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

August 16, 2012

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

SUBJECT: SOUTH TEXAS PROJECT, UNIT 2 – REQUEST FOR RELIEF RR-ENG-3-08 FROM AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE SECTION XI REQUIREMENTS FOR REPAIRS TO CLASS 3 VALVES IN ESSENTIAL COOLING WATER SYSTEM PIPING (TAC NO. ME8159)

Dear Mr. Richards:

By letter dated March 12, 2012 (Agencywide Document Access and Management System (ADAMS) Accession No. ML12079A034), as supplemented by letters dated April 10 and August 6, 2012 (ADAMS Accession Nos. ML12110A085 and ML12228A293, respectively), STP Nuclear Operating Company (the licensee) submitted for U.S. Nuclear Regulatory Commission (NRC) review and approval Relief Request RR-ENG-3-08 for South Texas Project (STP), Unit 2. The licensee requested relief from the requirements of IWA-5250 of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for repairs of flaws found in bodies of the two Class 3 cast aluminum-bronze valves in Essential Cooling Water system piping pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(ii).

Specifically, the licensee requested to defer code repairs of the flaws until after STP, Unit 2, was returned to service scheduled for mid-April 2012 at the time of the request. The relief is needed due to nonavailability of necessary parts until after the scheduled restart date.

On April 13, 2012, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff verbally authorized the use of RR-ENG-3-08 (ADAMS Accession No. ML121040361) up to the time the valve is repaired, but no later than October 28, 2012. The enclosed safety evaluation documents the technical basis of the verbal authorization.

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

If you have any questions, please contact Balwant K. Singal at 301-415-3016 or by e-mail at Balwant.Singal@nrc.gov.

Sincerely,

CF Lyon for

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

Docket No. 50-499

Enclosure:
As stated

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF NO. RR-ENG-3-08

ESSENTIAL COOLING WATER SYSTEM VALVE REPAIRS

RELIEF FROM AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESEEL CODE, SECTION XI REQUIREMENTS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 2

DOCKET NO. 50-499

1.0 INTRODUCTION

By letter dated March 12, 2012 (Agencywide Document Access and Management System (ADAMS) Accession No. ML12079A034), as supplemented by letters dated April 10 and August 6, 2012 (ADAMS Accession Nos. ML12110A085 and ML12228A293, respectively), STP Nuclear Operating Company (the licensee) submitted for U.S. Nuclear Regulatory Commission (NRC) review and approval Relief Request RR-ENG-3-08 for South Texas Project (STP), Unit 2. The licensee requested relief from the requirements of IWA-5250 of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for repairs of flaws found in bodies of the two Class 3 cast aluminum-bronze valves in Essential Cooling Water (ECW) system piping pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(ii).

Specifically, the licensee requested for defer code repairs of the flaws until after STP, Unit 2, was returned to service scheduled for mid-April 2012 at the time of the request. The relief is needed due to nonavailability of necessary parts until after the scheduled restart date.

On April 13, 2012, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff verbally authorized the use of RR-ENG-3-08 (ADAMS Accession No. ML121040361) up to the time the valve is repaired, but no later than October 28, 2012.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4) require that ASME Code Class 1, 2, and 3 components (including supports) will 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.

The regulations in 10 CFR 50.55a(a)(3), state, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be authorized by Director, Office of Nuclear Reactor Regulation, if the licensee demonstrates that: (i) the proposed alternative provides 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.

3.0 TECHNICAL EVALUATION

3.1 Component for Which Relief is Requested

The affected components are ASME Code Class 3, aluminum-bronze, ECW system blowdown isolation valves 2-EW-FV-6936 and 2-EW-FV-6937. These valves are designed to meet the requirements specified in ASME Code, Section III, 1974 Edition through 1975 Winter Addenda with Code Case 1761-1.

In its letter dated March 12, 2012, the licensee stated, in part, that

> The ECW System is designed to supply cooling water to various safety-related systems for normal plant operation, normal shutdown, and during and after postulated design-basis accidents. Valves 2-EW-FV-6936 and 2-EW-FV-6937 have a safety-related function to stop ECW blowdown on a safety injection signal.... The valves are normally closed and opened only when blowdown is needed.... The flawed components are located downstream of components cooled by the ECW system. Each ECW blowdown valve can be isolated by means of a normally open, manually operated isolation valve immediately upstream.

3.2 Applicable Code Edition and Addenda

ASME Code, Section XI, 2004 Edition.

3.3. Applicable Code Requirement (as stated by the licensee)

> ASME [Code,] Section XI, IWA-5250(a)(3) requires that the source of leakage be evaluated for repair or replacement in accordance with IWA-4000[1]. Relief from the requirements of IWA-5250(a)(3) is requested so that code repair of the through-wall flaw at these locations may be deferred until the parts required for replacing the valves can be obtained.

[1] In its letter dated March 12, 2012, the licensee stated that ASME Code Section XI, IWA-5250(a)(3) requires that the source of leakage be evaluated for repair or replacement in accordance with IWA-4000 or IWA-7000. Requirement IWA-7000 was imposed in error. By letter dated August 6, 2012, the licensee corrected the error and deleted reference to IWA-7000.

3.4 Reason for Request

In its letter dated March 12, 2012, the licensee stated, in part, that

> The flaws in valve 2-EW-FV-6936 were first found on July 28, 2011, during six-month visual examination of ECW large bore piping. Unit 2 was in Mode 1 at 100% power. Note that refueling outage 2RE15 began October 29, 2011. Through an oversight, the need for a relief request was not identified at the time of discovery, and repairs were not completed during 2RE15 prior to Unit 2 restart and breaker closure on November 22, 2011. Subsequent to restart of Unit 2, an unrelated and unplanned shutdown occurred on November 29, 2011. Unit 2 remains shutdown in Mode 5 with restart planned for mid-April 2012 [at the time of the relief request].
>
> The second set of dealloying flaws was found in valve 2-EW-FV-6936 on November 30, 2011, during a follow-up visual examination. Unit 2 was at Mode 5 at 0% power.
>
> A dealloying flaw was visually identified in valve 2-EW-FV-6937 on January 9, 2012, during the periodic examination of ECW large bore piping. Unit 2 was in Mode 5 at 0% power.

The licensee plans to repair these flaws by replacing the degraded valve bodies. The licensee has ordered the replacements parts and plans to complete the repairs by October 28, 2012.

In its letter dated March 12, 2012, the licensee stated, in part, that

> Time required to obtain the parts precludes resolution of the flaws prior to the planned Unit 2 startup date. Delaying unit restart until completion of repairs would constitute a hardship without a compensating increase in the level of quality and safety.

Therefore, relief is requested on the basis of hardship.

3.5 Proposed Alternative

In its letter dated March 12, 2012, the licensee stated, in part, that

> ASME Section XI, IWA-5250, requires that leakage be evaluated for corrective action and implies that any component with through-wall leakage must be repaired or replaced regardless of the leakage rate. The expectation of ASME [Code] Section XI is that through-wall leaks are repaired at the time of discovery. Replacement of the affected valves is to be deferred until parts are available. The work can be performed when the unit is in Mode 1. More frequent monitoring has been implemented to detect changes in the condition of the flaws and initiate compensatory actions as needed until a repair can be implemented.

The licensee's proposed alternative is based on demonstrating that the degraded flange will maintain its structural integrity until the valves are replaced. The licensee presented the results of flaw characterization, flooding and pump and cooling reservoir margin assessments, and flaw evaluations. In addition, the licensee proposed an augmented monitoring plan to ensure the structural integrity of the degraded valves.

3.6 Duration of the Proposed Alternative

The licensee plans to replace the valves by October 28, 2012, after the replacement parts are received.

3.7 NRC Staff Evaluation

The regulations in 10 CFR 50.55a(a)(2) state that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Code specified in 10 CFR 50.55a paragraphs (b) through (g). However, 10 CFR 50.55a(a)(3) permits licensees to apply alternatives to the requirements of the ASME Code, Section XI, when authorized by the NRC.

3.7.1 General

NRC Generic Letter (GL) 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," dated June 15, 1990 (ADAMS Accession No. ML031140590), and ASME Code Case N-513-3, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping, Section XI, Division 1," dated January 26, 2009, subject to conditions specified in NRC Regulatory Guide 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," October 2010 (ADAMS Accession No. ML101800536), provide guidance concerning temporary repairs. In its request for additional information (RAI) dated March 22, 2012 (ADAMS Accession No. ML120820096), the NRC staff requested the licensee to identify, for each of the requirements contained in both the code case and GL 90-05, whether the current relief request a) meets the requirement; b) the requirement is not applicable; or c) how the current relief request provides an acceptable level of safety with respect to each requirement.

By letter dated April 10, 2012, in response to the NRC staff's RAI, the licensee stated that both Code Case N-513-3 and Generic Letter 90-05 are either not applicable to valves or exclude valves from the evaluation. The NRC staff considers that the analysis methodology prescribed in Code Case N-513-3 and GL 90-05 could be applied, although these documents may not be directly applicable to valve bodies. In its RAI response, the licensee provided a listing of specific evaluation criteria from GL 90-05 and their relevance to the relief request. The following summarizes the results of the comparison provided by the licensee.

1. Flaw Detection During Plant Operation and Impracticality Determination. The licensee stated that the replacement of the valves can be performed on-line, but the replacement parts will not be available until July 2012. The licensee also stated that, based on its experience, valve replacement is preferred over weld repairs on aluminum-bronze valves.

2. Root Cause Determination and Flaw Characterization. The licensee determined the root cause of the degradation to be dealloying of aluminum-bronze. Ultrasonic testing (UT) or radiographic testing (RT) cannot determine the flaw geometry and a flaw can only be examined visually.

3. Flaw Evaluation. The licensee analyzed the flaws as through-wall defects even though no apparent crack was observed. There was also no apparent leakage from the degradation indicating dealloying was identified by residue buildup on the outside surface of the valve body. In addition, the multiple flaws were accounted for by assuming one large defect and assuming the pipe wall was 100 percent dealloyed. The licensee also stated that the scope of the NRC staff guidance provided by GL 90-05 excludes valves and the wall-thinning approach does not apply to the condition being evaluated.

4. Augmented Inspection. The licensee stated that this type of flaw cannot be assessed by either UT or RT. The licensee conducts VT-2 examinations of visible areas of ECW piping for evidence of dealloying at 6-month intervals. The licensee also stated that this frequency is sufficient for timely identification of flaws due to dealloying.

The NRC staff concludes that the licensee's proposed alternative provides an acceptable level of quality and safety, as discussed in this safety evaluation even though Code Case N-513-3 and GL 90-05 are not directly applicable to the dealloying of the subject valves.

3.7.2 Flaw Characterization

On July 28, 2011, during periodic examination of ECW large bore piping, the licensee found two indications of a through-wall flaw on the inlet flange of ECW valve 2-EW-FV-6936. In its letter dated March 12, 2012, the licensee stated, in part, that:

> Residue deposits indicating dealloying at the 10 o'clock (Figure 3) and 2 o'clock (Figure 4) positions (looking upstream from the valve inlet). Prior to cleaning, each indication was less than ½-inch in diameter. Cleaning reduced each indication to less than 1/8-inch in diameter. Subsequent inspection on November 30, 2011, found additional residue at the 2 o'clock position with little apparent change at the 10 o'clock position. Visual examination found no cracks, and there is no measurable leakage.
>
> The November 30, 2011, inspection identified an additional three dealloying indications on the same valve body. The indications are on the inlet flange of 2-EW-FV-6936 between the 11 o'clock and 1 o'clock positions (between the first two indications).

On January 9, 2012, the licensee found an additional dealloying indication on a similar ECW valve, 2-EW-FV-6937. The licensee found the indication at the 2 o'clock position and determined it was less than 1/8-inch in diameter surrounded by multiple pinpoints of secondary indication. The total area affected is about 1-1/2 inch.

In each case, the licensee found no cracks or no measurable leakage through visual examination. The affected valves are 4-inch, 150 pounds (lbs) air-operated valves with a minimum wall thickness in the affected area of 1/4-inch.

The licensee identified the root cause of the subject flaw as dealloying. The licensee stated that the root cause of dealloying is a combination of corrosion and stress. The dealloying process normally initiates from a crevice, such as the area behind a backing ring, a fabrication-induced flaw, or a casting flaw.

The NRC staff concludes that the licensee has appropriately characterized these flaws as dealloying. The staff believes that dealloying in this case is similar to dealloying seen in other susceptible aluminum-bronze components.

3.7.3 Flooding Assessment

The ECW system is a low-pressure system with normal operating pressure of approximately 50 pounds per square inch gauge (psig), and a design pressure of 120 psig. Normal system temperature is 47 to 100 degrees Fahrenheit (°F). Per STP's Technical Specification (TS) 3.7.4, "Essential Cooling Water System," at least three independent ECW loops are required to be operable in Modes 1, 2, 3, and 4. With only two ESW loops operable, at least three loops are required to be restored to operable status within seven days, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With two or more ECW loops inoperable, at least two loops are required to be restored to operable status within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The requirements of the Configuration Risk Management Program (CRMP) can be applied to the requirements of TS 3.7.4. The licensee indicated that through engineering review it has determined that that even though the affected valves are degraded by dealloying, they remain operable.

The affected ECW blowdown valves are located downstream of components cooled by the ECW. Each ECW valve can be isolated by means of a normally open, manually operated isolation valve immediately upstream. This isolation valve provides the ability to isolate the blowdown valve while maintenance is being performed during normal operation.

The licensee stated that flooding in the area due to the ECW system is enveloped by the worst-case flow from an opening in a local pipe due to a critical crack with an area equivalent to a rectangle of length one-half the pipe diameter and a width equal to one-half the pipe wall thickness. The licensee's analysis is presented in Appendix 9A of the STP Updated Final Safety Analysis Report, "Assessment of the Potential Effects of Through-wall Cracks in ECWS Piping." The licensee calculated a flow rate of 8 gallons per minute (gpm) through a postulated crack in the ECW pipe. However, the flood volume over a 7-day period is enveloped by the licensing basis flooding analysis with a maximum flood rate of approximately 5,750 gpm for 30 minutes.

The licensee further stated that the leakage from the ECW piping flows to the Mechanical Auxiliary Building (MAB) sumps. Sump level alarms are available to warn operators if leakage exceeds the sump capacity. In its RAI dated March 22, 2012, the NRC requested the licensee to discuss the flaw size and leak rate that would cause the alarm to activate. By letter dated

April 10, 2012, the licensee stated that when the sump level reaches 36 inches, the alarm activates. Flow through the valves is throttled to 300 gpm, and each of the ECW sump pumps has a capacity of 390 gpm. The licensee stated that even if a crack develops, loss of structural integrity of the pipe will not result in excessive flow to the sump. In addition, the licensee states that the ECW pumps and cooling reservoir have adequate design margin and make-up capability to account for postulated leakage. The NRC staff concludes that the licensee has performed appropriate flooding evaluation and that MAB sump has sufficient capacity to manage the leakage so that the flooding in the MAB will not be a safety concern.

3.7.4 Inspections

The licensee stated that it will perform monthly walkdown of the affected area to detect changes in size of the discolored area or leakage until the valves are replaced. The licensee stated that it will re-evaluate the structural integrity and monitoring frequency if significant changes in the condition of the dealloyed areas are found. The NRC staff requested the licensee to describe the criteria to be used to determine significant changes in the dealloying. In its letter dated April 10, 2012, the licensee defined significant changes as:

- Residue buildup over an area greater than twice the original area of residue buildup
- Residue buildup or leakage in a new area on the valve body
- A change from residue buildup to measureable liquid leakage
- Indication of crack development

The licensee stated that it will increase the inspection frequency to weekly if the flaw growth is projected to compromise the integrity of the valve prior to its replacement. In addition, the licensee stated that these valves are in a location that is subject to daily observation by plant personnel.

ASME Code Case N-513-3, paragraph 2(f) requires that for through-wall leaking flaws, leakage shall be observed by daily walkdowns to confirm the analysis conditions used in the evaluation remain valid. In an RAI dated March 22, 2012, the NRC staff requested the licensee to explain why monthly inspections are sufficient in this case. In its letter dated April 10, 2012, the licensee stated that current monitoring levels have shown that the dealloying process progresses very slowly. The licensee has observed these indications over a period of months and the progress of the degradation has been inconsequential or non-existent. Therefore, the licensee will detect any change in the degradation before the ECW train is challenged. The NRC staff concludes that the proposed monthly inspections are acceptable due to the slow progression of the dealloying process. In addition, daily observations by the plant personnel will also help in detecting changes and additional degradation before ECW system integrity is compromised.

The licensee has stated that examinations of all ECW train piping are performed every 6 months. No additional areas of dealloying have been found. If identified, dealloying flaws are addressed under the station corrective action program. In an RAI dated March 22, 2012, the NRC staff requested that the licensee explain the rationale for not implementing the increased inspection on the third ECW loop since degradation was found on the other two ECW loops. In its letter dated April 10, 2012, the licensee stated that the dealloying in the second valve was

found during a walk down at 6-month interval and that period is consistent with the historically slow progression of the dealloying process. The staff believes that the 6-month examination of the ECW train piping is sufficient in finding new degradation due to the slow progression of the dealloying process.

The licensee stated that the dealloying flaws can only be detected by VT-2 examinations after the degradation has reached the outside surface. In an RAI dated March 22, 2012, the NRC staff requested the licensee to explain how the inspection procedures account for the inside diameter (ID) degradation. In its letter dated April 10, 2012, the licensee stated that its flaw evaluation assumes that the affected pipe has experienced 100 percent dealloying. The licensee stated that inspection procedures cannot detect through-wall dealloying and this type of degradation can only be verified by destructive examination. The staff agrees that the only way to verify the through-wall extend of the dealloying is by performing destructive examination of the degraded pipe. The results of the staff's review of the flaw evaluation procedures assuming 100 percent dealloyed materials are described in Section 3.7.5 of this safety evaluation. The NRC staff concluded that the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the valves for the authorization period of this relief request.

3.7.5 Flaw Evaluation

The licensee calculated that the critical crack length for maintaining structural integrity of the valve body to be 3 inches. In an RAI dated March 22, 2012, the NRC staff requested that the license provide details of the analyses including the fracture mechanics procedures and the material properties assumed. In its letter dated April 10, 2012, the licensee stated that the fracture analyses considered both plastic collapse and fracture. The licensee described the methodology and analyses in provided APTECH Engineering Services, Inc. Report AES-C-1964-1, "Calculation of Critical Bending Stress for Dealloyed Aluminum-Bronze Castings in the ECW system," dated December 1, 1993. This report was submitted as an attachment to Relief Request RR-ENG-35, dated August 10, 2000 (ADAMS Accession No. ML003742174).

The licensee conducted two separate analyses. The first is a net section collapse analysis assuming an idealized through-wall crack. This through-wall crack represents the dealloyed area of the pipe wall. Using the tensile properties of the non-dealloyed material, the licensee calculated the critical bending stress as a function of crack length using Net-Section-Collapse equations from ASME Code Section XI Appendix H. The licensee then limits this behavior to a critical bending stress of 38 kilopounds per square inch (ksi), assuming the pipe wall is unflawed but 100 percent dealloyed. For this analysis, the licensee uses a dealloyed ultimate strength of 30 ksi. In an RAI dated March 22, 2012, the NRC staff requested the licensee to describe the basis of the 100 percent dealloyed tensile strength and for a discussion of how 100 percent dealloyed material can leak and still have residual strength. In its letter dated April 10, 2012, the licensee stated that the phenomena of dealloying has been discussed with the NRC staff in detail in analysis titled, "Failure Analysis and Structural Integrity Evaluation of Leaking Small Bore Aluminum-Bronze Cast Valve Bodies and Fittings in Essential Cooling Water System," submitted by Houston Lighting and Power Company in a letter dated November 1, 1988 (ADAMS Legacy Document No. 8811100009). The NRC staff reviewed this reference and determined that the report does not contain sufficient data to fully characterize the tensile strength of dealloyed material. In addition, the ASME Code, Section XI, net section collapse analysis requires the material flow stress be used in the analyses. The flow stress is typically

defined as the average of the yield and ultimate strength and is lower than the material ultimate strength. In the licensee's reference, no yield strength was reported due to the small size of the specimen tested. Therefore, the NRC staff concluded that the licensee's calculated bending stress limit of 38 ksi may be non-conservative when the ultimate strength is used in the analysis procedures.

In addition to the net section collapse analyses, the licensee also conducted linear elastic fracture analysis. Using accepted fracture relationships and a lower bound plane strain fracture toughness from crack-tip opening displacement experiments on non-dealloyed material, the licensee generated critical bending stress as a function of through-wall crack length. The licensee compared the fracture and net-section collapse behavior and demonstrated that the fracture behavior is limiting. Using plant-specific loads for pressure, dead weight, expansion, end loading, operating basis earthquake (OBE), safe shutdown earthquake (SSE), and postulated pipe break, the licensee stated that significant margin is available for both the fracture and collapse analyses. In an RAI dated March 22, 2012, the NRC staff requested that the license provide details on the margin calculations. In its letter dated April 10, 2012, the licensee stated that the critical (i.e., maximum allowable) bending stress, based on the limiting analysis is 18.19 ksi for a 3-inch-long defect, which compared to the applied bending stress of 1.7 ksi derived from deadweight and seismic loads represents significant margin.

Even though the NRC staff considers that the licensee's analysis methodology is acceptable, the staff does not conclude the appropriate material properties were used or fully characterized. Since the ID extent of the degradation is unknown, the staff concludes that a through-wall flaw in 100 percent dealloyed material would be the limiting case and should be assumed in the licensee's analysis to support its proposed alternative. This analysis would require the fracture toughness and flow stress of the dealloyed material. The staff expects that the maximum allowable bending stress from such an analysis would be less than that calculated in the limited case by the licensee and the margin between the allowable and applied bending stress would be less than the margin calculated by the licensee. Therefore, the staff does not agree with the applicability of the licensee's flaw evaluation, as it pertains to dealloyed material.

The NRC staff also does not conclude that appropriate material properties were used in a conservative and limiting fracture analyses performed in support of this relief request. Since the amount of degradation on the ID and through-wall is unknown, the staff concludes that more appropriate analysis would utilize the 100 percent dealloyed strength and toughness properties. Therefore, the staff does not agree with the applicability of the licensee's flaw evaluation, as it pertains to dealloyed material.

However, the NRC staff has determined that the slow progression of the dealloying process, coupled with the proposed augmented inspection program and the daily surveillance of the area containing the valves, results in reasonable assurance of a noticeable change to the visible degradation before instantaneous valve failure. In addition, since these blowdown valves are on the discharge side of the piping system and can be isolated by means of a manually operated isolation valve, a failure will not reduce flow or pressure to the safety-related components, damage surrounding components or cause flooding in excess of the sump capacity. Therefore, the NRC staff concludes that immediate replacement of the two subject valves in accordance with the ASME Code, Section XI, will not result in a significant increase in safety or quality when compared to the proposed alternative in the relief request.

3.7.6 Compliance with Regulations

As stated above, the licensee submitted the relief request based on 10 CFR 50.55a(a)(3)(ii). The licensee stated that the relief request is to allow deferral of the code repair of the flaws until after STP, Unit 2, is returned to service.

To qualify for 10 CFR 50.55a(a)(3)(ii), the licensee must determine if 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 stated that the code repair was not possible at the time of the request due to nonavailability of necessary parts. After identifying the dealloying in the fall of 2011, the licensee issued a purchase order to procure the replacement valves. These valves have a long lead time and will not be available before the planned restart of Unit 2 (scheduled for mid-April 2012 at the time of the request). The replacement parts are scheduled to be delivered by July 28, 2012. The licensee also looked for replacement valves of different materials, but was unsuccessful in finding a suitable for the application valve with a shorter lead time. Therefore, the NRC staff considers this to be a hardship and concludes that the licensee's proposal is consistent with the requirements of 10 CFR 50.55a(a)(3)(ii).

4.0 CONCLUSION

As set forth above, the NRC staff concludes that that the ASME Code requirements to repair the degraded valves before the end of the STP, Unit 2, unscheduled outage (scheduled for mid-April 2012 at the time of the request), results in a hardship due to the unavailability of replacement valves. Even though the staff does not agree with the applicability of the licensee's flaw evaluation with respect to dealloyed valves, the staff concludes that the proposed inspection schedule and procedures, coupled with the replacement date, provide reasonable assurance of noticeable change to the visible degradation before instantaneous valve failure, and that complying with the specified 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(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative for deferral of code repair/replacement of degraded ECW system blowdown isolation valves up to the time that the valves are replaced but no later than October 28, 2012.

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

Principal Contributor: David Rudland, NRR/DE/EPNB

Date: August 16, 2012

If you have any questions, please contact Balwant K. Singal at 301-415-3016 or by e-mail at Balwant.Singal@nrc.gov.

Sincerely,

/RA by FLyon for/

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

Docket No. 50-499

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