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October 28, 2013
NOC-AE-13003041
10 CFR 54
STI: 33764926

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
2013 Annual Update to the South Texas Project
License Renewal Application (TAC NOS. ME4936 and ME4937)

Reference: STPNOC Letter dated October 25, 2010, from G. T. Powell to NRC Document Control Desk, "License Renewal Application" (NOC-AE-10002607) (ML103010257)

By the referenced letter, STP Nuclear Operating Company (STPNOC) submitted an application to the Nuclear Regulatory Commission (NRC) for the renewal of Facility Operating Licenses NPF-76 and NPF-80, for South Texas Project (STP) Units 1 and 2, respectively. The application included the License Renewal Application (LRA), and the Applicant's Environmental Report – Operating License Renewal Stage. As required by 10 CFR 54.21(b), each year following submittal of the LRA, an amendment to the LRA must be submitted that identifies any change to the current licensing basis (CLB) that materially affects the contents of the LRA, including the Updated Final Safety Analysis Report (UFSAR) supplement.

This LRA update covers the period from September 1, 2012 through August 31, 2013. Enclosure 1 identifies STP LRA changes that are being made to: (1) reflect the CLB that materially affect the LRA; and (2) reflect completed enhancements and commitments. Changes to LRA pages described in Enclosure 1 are depicted as line-in/line-out pages provided in Enclosure 2.

License Renewal Application revised regulatory commitments are provided in Enclosure 2. There are no other regulatory commitments in this letter.

Should you have any questions regarding this letter, please contact Arden Aldridge, STP License Renewal Project Lead, at (361) 972-8243, or Ken Taplett, STP License Renewal Project regulatory point-of-contact, at (361) 972-8416.

A147
NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 28, 2013
Date

G. T. Powell

G. T. Powell
Site Vice President

KJT

Enclosures:
1. STPNOC License Renewal Application (LRA) Changes Reflected in 2013 Annual LRA Update
2. STP LRA Changes with Line-in/Line-out Annotations

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# Enclosure 1

# STPNOC License Renewal Application (LRA)

# Changes Reflected in

# 2013 Annual LRA Update

# STPNOC License Renewal Application (LRA) Changes Reflected in 2013 Annual LRA Update

| Following Changes Materially Affect the LRA | |
|---|---|
| **Reason for Change** | **Affected LRA Sections or Tables** |
| Revised Section 2.1.2.3.5, "Station Blackout" to reflect Change Notice 3013 to the Updated Final Safety Analysis Report (UFSAR) that deleted the section on "Station Blackout Coping Duration". (Note 1) | 2.1.2.3.5 |
| Revised Final Safety Analysis Report Supplement to reflect that the PWR Reactor Internals program is a new program that has now been implemented. | A1.35 |
| South Texas Project UFSAR Change Notice 3073 deleted a discussion in Appendix 9A of the UFSAR that relief requests for leaks in piping diameters of one inch or less are exempted per the ASME code. The aging management program was revised to delete any indication that components with indications of through-wall dealloying, associated with piping of one inch in diameter or less, would not be replaced by the end of the next refueling outage. | A1.37 |
| Clarify that procedures will be enhanced to perform a remote VT-1 of stud insert #30 "in Unit 2 only" concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area. | B2.1.3 |
| Revised program description to reflect that the PWR Reactor Internals program is a new program that has now been implemented. | B2.1.35 |
| South Texas Project UFSAR Change Notice 3073 deleted a discussion in Appendix 9A of the UFSAR that relief requests for leaks in piping diameters of one inch or less are exempted per the ASME code. Aging management program revised to delete any indication that components with indications of through-wall dealloying, associated with piping of one inch in diameter or less, would not be replaced by the end of the next refueling outage. | B2.1.37 |
| Updated implementation schedule for License Renewal Commitment Item #27 to implement the PWR Reactor Internals Program as described in MRP-227A to reflect that the commitment has been completed. | Table A4-1 |
| Revised Commitment Item #37 to take groundwater samples to reflect that the commitment has been completed. | Table A4-1 |
| Revised Commitment Item #42 to reflect that the remote VT-1 inspection of stud insert #30 is "in Unit 2 only". | Table A4-1 |
| Revised Commitment Item #43 to reflect that the commitment to remove seal cap enclosures for the valves described in Unit 1 has been completed. | Table A4-1 |

Note 1: Change notice 3013 to the Updated Final Safety Analysis Report deleted the section on "Station Blackout Coping Duration" from Section 8.3.4 "Station Blackout". STP established and the NRC acknowledged in a Safety Evaluation that the AAC power source meets the requirements of 10 CFR 50.63 and has sufficient capacity and capability to provide power to the SBO loads within 10 minutes of the onset of SBO. Therefore, the SBO Rule is satisfied for not requiring a coping analysis.

# Enclosure 2

# STP LRA Changes with Line-in/Line-out Annotations

### 2.1.2.3.5 Station Blackout

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for SBO (10 CFR 50.63).

STPNOC provided the NRC with its original response to the SBO Rule in correspondence dated April 17, 1989. The NRC issued its initial Safety Evaluation on July 17, 1991. On July 28, 1994, a STP self-assessment determined that the STP units were not fully meeting the requirements of the SBO Rule. On August 1, 1994 STPNOC prepared a justification for continued operation which implemented the 73 miles per hour hurricane shutdown criterion, reduced the coping duration from 8 hours to 4 hours, and eliminated crediting the auxiliary ESF transformers for SBO coping. On March 1, 1995 STP provided the NRC a revised position to the SBO Rule. The NRC issued a Safety Evaluation Report for the Revised Station Blackout Position on July 24, 1995.

UFSAR Section 8.3.4 discusses SBO coping duration, alternate AC power source, ~~condensate inventory for decay heat removal, class 1E battery capacity, compressed air requirements, effect of loss of ventilation, containment isolation, reactor coolant inventory~~ and quality assurance program requirements.

The STP SBO analysis was performed using the guidance provided in NUMARC 87-00, Rev. 0 and the coping time (the postulated maximum SBO duration) was determined to be four hours. The STP SBO position credits any one of the three standby diesel generators as the AAC source. Each standby diesel generator can energize an independent train of auxiliary feedwater, essential cooling water, component cooling water, steam generator power operated relief valves, high head safety injection, and EAB/Control Room HVAC. Each standby diesel generator meets or exceeds the NUMARC 87-00, Appendix B, criteria for capacity, capability and connectability.

STP's offsite power system is in accordance with GDC 17 and provides two separate paths of power from the transmission system to the ESF buses as shown on Figure 2.1-2. Recovery from an SBO focuses on restoration of an AC power source. This can be from the onsite diesel generators, or from an offsite source. STP has the following paths of offsite power.

- 345 kV switchyard to main and unit auxiliary transformers
- 345 kV switchyard to standby transformer 1
- 345 kV switchyard to standby transformer 2

The main and unit auxiliary transformers are connected to the switchyard through disconnect G019 (Unit 1), and G029 (Unit 2) which connects to the switchyard via switchyard circuit breakers Y510 and Y520 (Unit 1), and Y590 and Y600 (Unit 2). The unit auxiliary transformer, the iso-phase bus, the main transformer, the overhead transmission lines, the switchyard breakers and switchyard breaker control cables and connections are within the scope of license renewal.

Standby transformers 1 and 2 are connected to the 345 kV switchyard north (Unit 1) and south (Unit 2) bus via disconnect S014 (Unit 1), and S024 (Unit 2). The standby transformers, the overhead transmission lines, and the switchyard disconnects are within the scope of license renewal.

A position paper was created to summarize the results of a review of the SBO documentation for STP. The position paper identifies the SSCs credited with coping and recovering from a SBO. The SSCs identified in the SBO position paper were used in scoping evaluations to identify SSCs that demonstrate compliance with 10 CFR 50.63.

License renewal drawing LR-STP-ELEC-00000E0AAAA schematically shows the portions of the plant AC electrical distribution system, including the SBO recovery path, that are included within the scope of license renewal and is summarized in Figure 2.1-2, Station Blackout Recovery Path.

SSCs classified as satisfying criterion 10 CFR 54.4(a)(3) related to station blackout are identified as within the scope of license renewal.

## A1.35 PWR REACTOR INTERNALS

The PWR Reactor Internals program manages cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload for reactor vessel components that provide a core structural support intended function. The program implements the guidance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* (MRP-227-A) and EPRI 1016609, *Inspection Standard for PWR Internals* (MRP-228). The program manages aging consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

Program examination methods include visual examination (VT-3), enhanced visual examination (EVT-1), volumetric examination, and physical measurements. The program provides both examination acceptance criteria for conditions detected as a result of monitoring the primary components, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

The PWR Reactor Internals program is a new program and has been ~~will be~~ implemented ~~within 24 months after the issuance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* MRP-227-A~~.

## A1.37 Selective Leaching of Aluminum Bronze

The Selective Leaching of Aluminum Bronze program manages loss of material due to selective leaching of aluminum bronze (copper alloy with greater than eight percent aluminum) components exposed to raw water within the scope of license renewal. The Selective Leaching of Aluminum Bronze program is an existing program that is implemented by STP procedure. The procedure directs that every six months (not to exceed nine months), an inspection of all aluminum bronze components be completed. STP has buried piping with less than eight percent aluminum content, and that is not susceptible to dealloying. However, there are welds in which the filler metal is a copper alloy with greater than eight percent aluminum material. Therefore, the procedure directs that a yard walkdown be performed above the buried piping with aluminum bronze welds, from the intake structure to the unit and from the unit to the discharge structure to look for changes in ground conditions that would indicate leakage. If a leak from below-grade weld is discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive examination. Aluminum bronze (copper alloy with greater than 8 percent aluminum) components which are found to have indications of through-wall dealloying are evaluated, and scheduled for replacement by the corrective action program. Components with indications of through-wall dealloying~~, associated with piping greater than one inch in diameter,~~ will be replaced by the end of the next refueling outage.

Volumetric examinations of aluminum bronze material components that demonstrate external leakage will be performed where the configuration supports this type of examination to conclude with reasonable assurance that cracks are not approaching a critical size.

Destructive examination of each leaking component removed from service will be performed to determine the degree of dealloying until 10 percent of the susceptible components in the ECW system are examined. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Metallurgical testing of leaking aluminum bronze material components in the ECW system removed from service will be performed to update the structural integrity analyses, to confirm load carrying capacity and to determine the degree of dealloying by destructive examination. Metallurgical testing of the removed leaking component will be performed until at least three different size components of two samples each are tested, and at least nine total samples are tested. The metallurgical testing will include fracture toughness testing of test samples that include a crack in the dealloyed material where sufficient sample size supports bend testing. Additionally, the samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Beginning 10 years prior to the period of extended operation for each 10-year interval, periodic metallurgical testing of aluminum bronze material components will be performed to update the structural integrity analyses, confirm load carrying capacity, and determine degree of dealloying. For each 10-year interval beginning 10 years prior to the period of extended operation, 20 percent of leaking above ground components removed from service, but at least one, will be tested every five years. Tensile test samples from a removed component will be tested to include both leaking and non-leaking portions of the component. If at least two leaking components are not identified two years prior to the end of each 10-year testing interval, a risk-ranked approach based on those components most susceptible to degradation will be used to identify candidate components for removal and testing so at least two components are tested during the 10-year interval. The samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. The samples will be destructively examined to determine the degree of dealloying and the presence of cracks. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

An engineering evaluation will be performed at the end of each test to determine if the sample size requires adjustment based on the results of the tests.

The acceptance criterion for ultimate tensile strength value of aluminum bronze material is greater than or equal to 30 ksi. The acceptance criterion for the fracture toughness is greater than or equal to 65 ksi $in^{1/2}$ for aluminum bronze castings and at welded joints in the heat affected zones. The acceptance criterion for yield strength is equal to or greater than one-half of the ultimate strength. If a criterion is not met, the condition will be documented in the corrective action program to perform a structural integrity analysis to confirm that the load carrying capacity of the tested material remains adequate to support the intended function of the ECW system through the period of extended operation.

## B2.1.3Reactor Head Closure Studs

### Program Description

The Reactor Head Closure Studs program manages cracking and loss of material by conducting ASME Section XI inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings. The program includes periodic visual, surface, and volumetric examinations of reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings and performs visual inspections of the reactor vessel flange closure during primary system leakage tests. The STP program implements ASME Section XI code, Subsection IWB, 2004 Edition. Reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings are identified in ASME Section XI Tables IWB-2500-1 and are within the scope of license renewal. The program implements recommendations in NUREG-1339 and NRC Regulatory Guide 1.65, *Material and Inspection for Reactor Vessel Closure Studs,* to address reactor head stud bolting degradation except for yield strength of existing bolting materials. STP uses lubricants on reactor head closure stud threads after reactor head closure stud, nut, and washer cleaning and examinations are complete. The lubricants are compatible with the stud material and operating environment and do not include $MoS_2$ which is a potential contributor to stress corrosion cracking.

In conformance with 10 CFR 50.55a(g)(4)(ii), the STP ISI Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition of the Code specified twelve months before the start of the inspection interval. STP will use the ASME Code Edition consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

Potential cracking and loss of material conditions in reactor vessel flange stud hole threads, reactor head closure studs, nuts, washers, and bushings are detected through visual, surface, or volumetric examinations in accordance with ASME Section XI requirements in STP procedures every ten years. A remote VT-1 of stud insert #30 <u>(Unit 2 only)</u> is performed concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area. These inspections are conducted during refueling outages. Reactor vessel studs are removed from the reactor vessel flange each refueling outage. Studs, nuts, washers, and bushings are stored in protective racks after removal. Reactor vessel flange holes are plugged with water tight plugs during cavity flooding. These methods assure the holes, studs, nuts, washers, and bushings are protected from borated water during cavity flooding. Reactor vessel flange leakage is detected prior to reactor startup during reactor coolant system pressure testing each refueling outage. The STP program has proven to be effective in preventing and detecting potential aging effects of reactor vessel flange stud hole threads, closure studs, nuts, washers, and bushings.

### NUREG-1801 Consistency

The Reactor Head Closure Studs program is an existing program that is consistent, with exception to NUREG-1801, Section XI.M3, Reactor Head Closure Studs.

**Exceptions to NUREG-1801**

Program Elements Affected:

*Scope of Program (Element 1)*

Regulatory Guide 1.65 states that the ultimate tensile strength of stud bolting material should not exceed 170 ksi. One closure head insert has a tensile strength of 174.5 ksi. STP credits inservice inspections that are within the scope of this AMP, which are implemented in accordance with the STP Inservice Inspection Program, Examination Category B-G-1 requirements, as the basis for managing cracking in these components. This is in accordance with the "parameters monitored or inspected" and "detection of aging effects" program elements in NUREG 1801, Section XI.M3. In addition, the studs, nuts and washers are coated with a lubricant which is compatible with the stud materials, and the studs, nuts, and washers are protected from exposure to boric acid by removing them and plugging the reactor vessel flange holes during cavity flooding. Replacement reactor head closure bolting obtained in the future (not currently installed or on site as spare parts) will be fabricated from material with an actual measured yield strength less than 150 ksi.

*Corrective Actions (Element 7)*

NUREG-1801, Section XI.M3 specifies the use of Regulatory Guide 1.65 requirements for closure stud and nut material. STP uses SA-540, Grade B-24 (as modified by Code Case 1605) stud material. The use of this material has been found acceptable to the NRC for this application within the limitations discussed in Regulatory Guide 1.85, *Materials Code Case Acceptability.*

**Enhancements**

*Scope of Program (Element 1)*

Procedures will be enhanced to preclude the future use of replacement closure stud assemblies fabricated from material with an actual measured yield strength greater than or equal to 150 ksi. The use of currently installed components and any spare components currently on site is allowed.

Detection of Aging Effects (Element 4)

Procedures will be enhanced to perform a remote VT-1 of stud insert #30 <u>(Unit 2 only)</u> concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area.

**Operating Experience**

Review of plant-specific operating experience has not revealed any program adequacy issues with the Reactor Head Closure Studs program for reactor vessel closure studs, nuts, washers, bushings, and flange thread holes. No cases of cracking due to SCC or IGSCC have been identified with STP reactor vessel studs, nuts, washers, bushings, and flange stud holes.

Review of the Refueling Outage Inservice Inspection Summary Reports for Interval 2 indicates there were no repair/replacement items identified with reactor vessel closure studs, nuts, washers, bushings, or flange thread holes. None of the repair/replacement items indicate any implementation issues with the STP ASME Section XI Program for reactor closure studs, nuts, washers, bushings, or flange thread holes.

The ISI Program at STP is updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the ISI Program to reference more recent editions of ASME Section XI at the end of each inspection interval ensures the ISI Program reflects enhancements due to operating experience that have been incorporated into ASME Section XI.

**Conclusion**

The continued implementation of the Reactor Head Closure Studs program provides reasonable assurance that aging effects will be managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B2.1.35 PWR Reactor Internals

### Program Description

The PWR Reactor Internals program manages cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload for reactor vessel components that provide a core structural support intended function. The program implements the guidance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* (MRP-227-A) and EPRI 1016609, *Inspection Standard for PWR Internals* (MRP-228, Rev. 0). The program manages aging consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A, Westinghouse designated expansion components in Table 4-6 of MRP-227-A, and the Westinghouse designated existing components in Table 4-9 of MRP-227-A. Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

Program examination methods include visual examination (VT-3), enhanced visual examination (EVT-1), volumetric examination, and physical measurements. Bolting ultrasonic examination technical justifications in MRP-228 have demonstrated the indication detection capability to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting. For some components, the MRP-227-A methodology specifies a focused visual (VT-3) examination, similar to the current ASME Code, Section XI, Examination Category B-N-3 examinations, in order to determine the general mechanical and structural condition of the internals by (a) verifying parameters, such as clearances, settings, and physical displacements; and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. In some cases, VT-3 visual methods are used for the detection of surface cracking when the component material has been shown to be tolerant of easily detected large flaws. In some cases, where even more stringent examinations are required, enhanced visual (EVT-1) examinations or ultrasonic methods of volumetric inspection, are specified for certain selected components and locations.

The program provides both examination acceptance criteria for conditions detected as a result of monitoring the primary components, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

The PWR Vessel Internals program is a new program that has been~~will be~~ implemented ~~within 24 months after the issuance of MRP-227-A, PWR Internals Inspection and Evaluation Guideline~~. The program will include future industry operating experience, as it is incorporated into the future revisions of MRP-227-A, to provide reasonable assurance for long-term integrity of the reactor internals. The reactor vessel internals included in the scope of the PWR Reactor Internals program are identified in Element 1. The scope of the program does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

**Aging Management Program Elements**

The results of an evaluation of each element against the 10 elements described in Appendix A of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* are provided below.

*Scope of Program – Element 1*

The scope of the program applies the guidance in MRP-227-A which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of Westinghouse reactor vessel internals. The scope of the PWR Reactor Internals program includes components that provide a core structural support intended function and are managed by the Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A and applicable MRP-227-A methodology license renewal applicant action items. MRP-227-A Table 4-9 also identifies existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

The STP reactor vessel internals are divided into the following major component groups: the lower core support assembly (including the entire core barrel assembly, baffle-former assembly, neutron shield panel, core support plate, and energy absorber assembly), the upper core support (UCS) assembly (including the upper support plate, support column, control rod guide tube assembly, upper core plate, and protective skirt), the incore instrumentation support structures (including the instrumentation columns (exit thermocouples), upper/lower tie plates, and instrumentation columns (BMI)), and miscellaneous alignment/interface components (including internals hold-down spring, upper core plate guide pins, and radial support keys including clevis inserts).

The following reactor vessel internals are included in the scope of the PWR Reactor Internals program:

1. Control rod guide tube assembly and Bolting

- Guide plate (cards) [Primary component]

- Lower flange welds and adjacent base metal (Addressed in AMR by Component Type of "RVI Control Rod Guide Tube Assembly") [Primary component]

- Guide Tube Support Pins (Split Pins) (Addressed in AMR by Component Type of "RVI Control Rod Guide Tube Bolting") [Existing programs component]

2. Core barrel assembly

- Upper core barrel flange weld and adjacent base metal (Addressed in AMR by Component Types of "RVI Core Barrel Assembly") [Primary component]

- Core barrel assembly–former bolting [Expansion component]

- Core barrel flange (Addressed in AMR by Component Types of "RVI Core Barrel Assembly") [Expansion component and Existing programs component]

- Core barrel axial welds and adjacent base metal [Expansion component]

- Core barrel girth welds and adjacent base metal [Primary component]

- Core barrel outlet nozzle welds and adjacent base metal [Expansion component]

- Lower core barrel flange weld and adjacent base metal (Addressed in AMR by Component Types of "RVI Core Barrel Assembly") [Primary component]

3. Baffle–former assembly and bolting

- Baffle-edge bolting [Primary component]

- Baffle–former bolting [Primary component]

- Baffle-former assembly [Primary component]

4. Alignment and interfacing components

- Internals hold-down spring [Primary component]

- Radial support key clevis insert bolts [Existing programs component]

- Upper core plate guide pins [Existing programs component]

5. Instrumentation support structures

- Instrumentation columns – BMI [Expansion component]

6. Upper core support assembly

- Upper core support protective skirt [Existing programs component]

- Upper Core Plate [Expansion component]

7. Lower Core Support Structure

- Core Support Plate Forging [Expansion component]

The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

The STP reactor vessel internals configuration does not include the lower internals assembly (lower support column bodies and lower core plate) noted in MRP-227-A.

The PWR Reactor Internals program is consistent with the following MRP-227-A assumptions (determination of applicability) which are based on PWR representative internals configurations and operational histories.

(1) STP has operated for less than 30 years of operation with high leakage core loading patterns. Operation with high leakage core loading was followed by implementation of a low-leakage fuel management pattern for the remaining operating life.

(2) STP operates at fixed power levels and does not usually vary power based on calendar or load demand schedule.

(3) STP has not implemented any design changes beyond those identified in industry guidance or recommended by Westinghouse.

*Preventive Actions – Element 2*

The PWR Reactor Internals program does not prevent degradation due to aging effects, but provides measures for monitoring to detect the degradation prior to loss of intended function. Preventive measures to mitigate aging effects such as loss of material and cracking include monitoring and maintaining reactor coolant water chemistry consistent with the guidelines of EPRI TR 1014986, *PWR Primary Water Chemistry Guidelines*, Volume 1. The primary water chemistry program is described separately in the Water Chemistry program (B2.1.2).

*Parameters Monitored or Inspected – Element 3*

The PWR Reactor Internals program monitors the following aging effects by inspection in accordance with the guidance of MRP-227-A or ASME Section XI Category B-N-3:

(1). Cracking
Cracking is due to stress corrosion cracking (SCC), primary water stress corrosion

cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or fatigue /cyclical loading. Cracking is monitored with a visual inspection for evidence of surface breaking linear discontinuities or a volumetric examination. Surface examinations may also be used to supplement visual examinations for detection and sizing of surface-breaking discontinuities.

(2). Loss of Material
Loss of Material is due to wear. Loss of material is monitored with a visual inspection for gross or abnormal surface conditions.

(3). Loss of Fracture Toughness
Loss of fracture toughness is due to thermal aging or neutron irradiation embrittlement. The impact of loss of fracture toughness is indirectly monitored by using visual or volumetric examination techniques to monitor for cracking and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

(4). Dimensional Changes
Dimensional Changes are due to void swelling and irradiation growth, distortion or deflection. The program supplements visual inspection with physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

(5). Loss of Preload
Loss of preload is caused by thermal and irradiation-enhanced stress relaxation or creep. Loss of preload is monitored with a visual inspection for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections.

The PWR Reactor Internals program manages the aging effects noted above consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A. MRP-227-A also identifies Existing Program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3. See the component list in element 1 to identify Primary, Expansion, and Existing components.

*Detection of Aging Effects – Element 4*

The PWR Reactor Internals program detects aging effects through the implementation of the parameters monitored or inspected criteria and bases for Westinghouse designated Primary Components in Table 4-3 of MRP-227-A and for Westinghouse designated Expansion Components in Table 4-6 of MRP-227-A. The aging effects of a third set of MRP-227-A internals locations identified in Table 4-9 of MRP-227-A are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

One hundred percent of the accessible volume/area of each component will be examined for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g. bolting), the minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible).

If defects are discovered during the examination, STP enters the information into the STP corrective action program and evaluates whether the results of the examination ensure that the component (or set of components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP-17096-NP.

*Monitoring and Trending – Element 5*

The program provides both examination acceptance criteria (See Element 6) for conditions detected as a result of monitoring the primary components as described in Element 4, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria (See Element 6) are required to be dispositioned through the corrective action program (See Element 7), which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

*Acceptance Criteria – Element 6*

Examination acceptance for the Primary and Expansion component examinations are consistent with Section 5 of MRP-227-A. ASME Section XI section IWB-3500 acceptance criteria apply to Existing Programs components. The following examination acceptance criteria apply to the STP reactor vessel internals:

Visual examination (VT-3) and enhanced visual examination (EVT-1)

For existing program components, the ASME Code Section XI, Examination Category B-N-3 provides the following general relevant conditions for the visual (VT-3) examination of removable core support structures.

(1) Structural distortion or displacement of parts to the extent that component function may be impaired,

(2) Loose, missing, cracked, or fractured parts, bolting, or fasteners,

(3) Corrosion or erosion that reduces the nominal section thickness by more than 5 percent,

(4) Wear of mating surfaces that may lead to loss of function; and

(5) Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

In addition, for the visual examinations (VT-3) of Primary and Expansion components, the PWR Reactor Internals program is consistent with the more specific descriptions of relevant conditions provided in Table 5-3 of MRP-227-A. EVT-1 examinations are used for detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids. EVT- 1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. The relevant condition applied for EVT-1 examination is the same as found for cracking in ASME Section XI section 3500 which is crack-like surface breaking indications.

Volumetric examination

Individual bolts are accepted (pass/fail acceptance) based on the detection of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt, it is assumed to be non-functional and the indication is recorded. Bolted assemblies are evaluated for acceptance based on meeting a specified number and distribution of functional bolts. Acceptance criteria for volumetric examination of STP reactor internals bolting are consistent with Table 5-3 of MRP-227-A.

Physical Measurements

Continued functionality of the internals hold down spring is confirmed by direct physical measurement. The examination acceptance criterion for this measurement is consistent with Table 5-3 of MRP-227-A and requires that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.

*Corrective Actions – Element 7*

The following corrective actions are available for the disposition of detected conditions that exceed the examination acceptance criteria:

(1) Supplemental examinations to further characterize and potentially dispose of a detected condition consistent with Section 5.0 of MRP-227-A;

(2) Engineering evaluation that demonstrates the acceptability of a detected condition consistent with WCAP-17096-NP;

(3) Repair, in order to restore a component with a detected condition to acceptable status (ASME Section XI); or

(4) Replacement of a component with an unacceptable detected condition (ASME Section XI)

(5) Other alternative corrective action bases if previously approved or endorsed by the NRC.

Relevant indications failing to meet applicable acceptance criteria are repaired or replaced in accordance with plant procedures. Appropriate codes and standards are specified in both the "ASME Section XI Repair, Replacement, and Post-Maintenance Pressure Testing" procedure and in design drawings. Quality assurance requirements for repair and replacement activities are also included in the STP Operations Quality Assurance Plan.

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing corrective actions. The QA program includes elements of corrective action, confirmation process and administrative controls, and is applicable to the safety-related and non-safety related systems, structures, and components that are subject to aging management review.

*Confirmation Process – Element 8*

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing the confirmation process. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-safety related systems, structures and components that are subject to aging management review.

*Administrative Controls – Element 9*

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing administrative controls. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-safety related systems, structures and components that are subject to aging management review.

*Operating Experience — Element 10*

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. The experience reviewed includes NRC Information Notice 84-18, Stress Corrosion Cracking in PWR Systems and NRC Information Notice 98-11, Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants. Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube split pins has also been reported.

Several other items with existing or suspected material degradation concerns that have been identified for PWR components are wear in thimble tubes and potentially in control guide cards and observed cracking in some high-strength bolting and in control rod guide tube alignment (split) pins. The latter are conditions that have been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load.

Based on industry operating experience, STP replaced the Alloy-750 guide tube support pins (split pins) with strained hardened (cold worked) 316 stainless steel pins during

Refueling Outage 1RE12 (Spring 2005) for Unit 1 and Refueling Outage 2RE11 (Fall 2005) for Unit 2. The replacement was conducted to reduce the susceptibility for stress corrosion cracking in the split pins. There were no cracked Alloy X-750 pins discovered during the replacement process.

The ASME Code, Section XI, Examination Category B-N-3 examinations of core support structures conducted during Refueling Outage 1RE15 (Fall 2009) for Unit 1, and Refueling Outage 2RE14 (Spring 2010) for Unit 2, did not identify any conditions that required repair, replacement or evaluation.

The ISI Program portion of the PWR Reactor Internals program at STP is updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the ISI Program to reference more recent editions of ASME Section XI at the end of each inspection interval ensures the ISI Program reflects enhancements due to operating experience that have been incorporated into ASME Section XI.

With exception of the ASME Section ISI portions, the PWR Reactor Internals program will be a new program and has no direct programmatic history. A key element of the MRP-227-A program is the reporting of aging of reactor vessel components. STP, through its participation in PWR Owners Group and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or INPO, as appropriate.

As additional Industry and applicable plant-specific operating experience become available, the OE will be evaluated and appropriately incorporated into the program through the STP Corrective Action and Operating Experience Programs. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. This process will confirm the effectiveness of this new license renewal aging management program by incorporating applicable OE and performing self assessments of the program.

**Conclusion**

The implementation of the PWR Reactor Internals program provides reasonable assurance that aging effects will be adequately managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## B2.1.37 Selective Leaching of Aluminum Bronze

### Program Description

The Selective Leaching of Aluminum Bronze program manages loss of material due to selective leaching for aluminum bronze (copper alloy with greater than eight percent aluminum) components exposed to raw water within the scope of license renewal. This plant-specific program will use requirements of the Selective Leaching of Materials program (B2.1.17) specifically relating to aluminum bronze components. The selective leaching of aluminum bronze is applied in addition to the Open-Cycle Cooling Water program (B2.1.9).

The Selective Leaching of Aluminum Bronze program is an existing program that is implemented by plant procedure. This procedure directs that every six months (not to exceed nine months), an inspection of aluminum bronze (copper alloy with greater than eight percent aluminum) components be completed. STP has buried copper piping with less than eight percent aluminum content that is not susceptible to dealloying. However, there are welds in which the filler metal is copper alloy with greater than eight percent aluminum material. Therefore, the procedure directs that a yard walkdown be performed above the buried piping with aluminum bronze welds, from the intake structure to the unit and from the unit to the discharge structure to look for changes in ground conditions that indicate leakage. Aluminum bronze (copper alloy with greater than 8 percent aluminum) components which are found to have indications of through-wall dealloying are evaluated, and scheduled for replacement by the corrective action program. Components with indications of through-wall dealloying~~, greater than one inch,~~ will be replaced by the end of the next refueling outage. Periodic destructive examinations of aluminum bronze material components will be performed to update the structural integrity analyses, confirm load carrying capacity, and determine degree of dealloying.

### Aging Management Program Elements

The results of an evaluation of each element against the 10 elements described in Appendix A of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* are provided below.

*Scope of Program (Element 1)*

The Selective Leaching of Aluminum Bronze program manages loss of material due to selective leaching for aluminum bronze (copper alloy with greater than eight percent aluminum) pumps, piping welds and valve bodies exposed to raw water within the scope of license renewal. These aluminum bronze (copper alloy with greater than eight percent aluminum) components with raw water internal environments are susceptible to loss of material due to selective leaching (dealloying).

STP has analyzed the effects of dealloying and found that the degradation is slow so that rapid or catastrophic failure is not a consideration. A structural integrity analysis performed when dealloying was first identified confirmed that 100 percent dealloyed aluminum bronze material retains sufficient load carrying capacity. This structural integrity analysis determined that the leakage can be detected before the flaw reaches a limiting size that would affect the intended functions of the essential cooling water and essential cooling water screen wash system.

Volumetric examinations of aluminum bronze material components that demonstrate external leakage will be performed where the configuration supports this type of examination to conclude with reasonable assurance that cracks are not approaching a critical size.

Destructive examination of each leaking component removed from service will be performed to determine the degree of dealloying until 10 percent of the susceptible components in the ECW system are examined. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Metallurgical testing of leaking aluminum bronze material components in the ECW system removed from service will be performed to update the structural integrity analyses, to confirm load carrying capacity and to determine the degree of dealloying by destructive examination. Metallurgical testing of the removed leaking component will be performed until at least three different size components of two samples each are tested, and at least nine total samples are tested. The metallurgical testing will include fracture toughness testing of test samples that include a crack in the dealloyed material where sufficient sample size supports bend testing. Additionally, the samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. Degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

As part of the testing described above, six samples from three aluminum bronze components removed from service in 2012 will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. The aluminum bronze samples exposed to ECW system raw water environment will come from a pump shaft line casing pipe and from two small cast valve bodies. The pump shaft line casing pipe was removed from service in 2012 and the two small cast valve bodies will be removed from service in 2012. The components to be sampled have been exposed to ECW system raw water environment since the ECW system entered service. Priority will be given to selecting 100% dealloyed component samples. STP will complete this testing prior to the end of 2012.

Beginning 10 years prior to the period of extended operation for each 10-year interval, periodic metallurgical testing will be performed to confirm that the load carrying capacity of aged dealloyed aluminum bronze material in the ECW system remains adequate to support the intended function of the system during the period of extended operation. For each 10-year interval beginning 10 years prior to the period of extended operation, 20 percent of leaking above ground components removed from service, but at least one, will be tested every five years. Tensile test samples from a removed component will be tested to include both leaking and non-leaking portions of the component. If at least two leaking components are not identified two years prior to the end of each 10-year testing interval, a risk-ranked approach based on those components most susceptible to degradation will be used to identify candidate components for removal and testing so at least two components are tested during the 10-year interval. The component will be sectioned to size the inside surface flaws, if present, and/or mapping of the dealloyed surface areas for determining the degree of the dealloying. The samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

An engineering evaluation will be performed at the end of each test to determine if the sample size requires adjustment based on the results of the tests. The structural integrity analyses will be updated as required to validate adequate load carrying capacity.

Plant procedure directs that every six months (not to exceed nine months) an inspection of all susceptible aluminum bronze (copper alloy with greater than eight percent aluminum) above ground components be completed to identify any components that show evidence of dealloying. Aluminum bronze (copper alloy with greater than 8 percent aluminum) components which are found to have indications of through-wall dealloying are evaluated, and scheduled for replacement by the corrective action program. Components ~~greater than one inch~~ will be replaced by the end of the subsequent refueling outage.

STP has buried copper alloy piping with less than eight percent aluminum that is not susceptible to dealloying. However, there are welds in which the filler metal is copper alloy with greater than eight percent aluminum material. Therefore, the procedure directs that a yard walkdown be performed above the buried piping aluminum bronze welds, from the intake structure to the unit and from the unit to the discharge structure to look for changes in ground conditions that indicate leakage. If leaking below-grade welds are discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive examination.

*Preventive Actions (Element 2)*

The Selective Leaching of Aluminum Bronze program does not prevent degradation due to aging effects but provides for inspections to detect aging degradation prior to the loss of intended functions, replacement of degraded components, and testing to confirm load carrying capacity of aged dealloyed aluminum bronze material.

The Open-Cycle Cooling Water program (B2.1.9) uses an oxidizing biocide treatment (sodium hypochlorite and sodium bromide) to reduce the potential for microbiologically influenced corrosion.

*Parameters Monitored or Inspected (Element 3)*

The Selective Leaching of Aluminum Bronze program includes visual inspections every six months (not to exceed nine months) for dealloying in all susceptible aluminum bronze (copper alloy with greater than eight percent aluminum) components. During these inspections, if evidence of through-wall dealloying is discovered, the components are evaluated and scheduled for replacement by the corrective action program. Components~~, greater than one inch,~~ will be replaced by the end of the next refueling outage.

During the walkdown of the buried essential cooling water piping, the ground is observed for conditions that would indicate leakage due to selective leaching. Whenever aluminum bronze materials are exposed during inspection of the buried essential cooling water piping, the components are examined for indications of selective leaching. If leaking below-grade welds are discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive examination.

*Detection of Aging Effects (Element 4)*

The Selective Leaching of Aluminum Bronze program includes visual inspection of aluminum bronze (copper alloy with greater than eight percent aluminum) components to determine if selective leaching of these components is occurring. Every six months (not to exceed nine months), an inspection of susceptible above ground aluminum bronze (copper alloy with greater than eight percent aluminum) components is completed to identify any components that show evidence of dealloying. Every 6 months, walkdown is performed above the buried essential cooling water piping containing copper alloy welds with an Aluminum content greater than 8 percent. During the walkdown, the soil is observed to identify conditions that may be an indication of leakage due to selective leaching. Whenever aluminum bronze materials are exposed during inspection of the buried essential cooling water and ECW screen wash system piping, the components are examined for indications of selective leaching. If leaking below-grade welds are discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive examination.

Aluminum bronze (copper alloy with greater than 8 percent aluminum) components which are found to have indications of through-wall dealloying are evaluated, and scheduled for replacement by the corrective action program. Components~~, greater than one inch,~~ will be replaced by the end of the next refueling outage.

Volumetric examinations of aluminum bronze material components that demonstrate external leakage will be performed where the configuration supports this type of examination to conclude with reasonable assurance that cracks are not approaching a critical size.

Destructive examination of each leaking component removed from service will be performed to determine the degree of dealloying until 10 percent of the susceptible components in the ECW system are examined. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Metallurgical testing of leaking aluminum bronze material components in the ECW system removed from service will be performed to update the structural integrity analyses, to confirm load carrying capacity and to determine the degree of dealloying by destructive examination. Metallurgical testing of the removed leaking component will be performed until at least three different size components of two samples each are tested, and at least nine total samples are tested. The metallurgical testing will include fracture toughness testing of test samples that include a crack in the dealloyed material where sufficient sample size supports bend testing. Additionally, the samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. Degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

As part of the testing described above, six samples from three aluminum bronze components removed from service in 2012 will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. The aluminum bronze samples exposed to ECW system raw water environment will come from a pump shaft line casing pipe and from two small cast valve bodies. The pump shaft line casing pipe was removed from service in 2012 and the two small cast valve bodies will be removed from service in 2012. The sample components have been exposed to ECW system raw water environment since the ECW system entered service. Priority will be given to selecting 100% dealloyed component samples. STP will complete this testing prior to the end of 2012.

Beginning 10 years prior to the period of extended operation for each 10-year interval, periodic metallurgical testing will be performed to confirm that the load carrying capacity of aged dealloyed aluminum bronze material in the ECW system remains adequate to support the intended function of the system during the period of extended operation. For each 10 year interval beginning 10 years prior to the period of extended operation, 20 percent of leaking above ground components removed from service, but at least one, will be tested every five years. Tensile test samples from a removed component will be tested to include both leaking and non-leaking portions of the component. If at least two leaking components are not identified two years prior to the end of each 10-year testing interval, a risk-ranked approach based on those components most susceptible to degradation will be used to identify candidate components for removal and testing so at least two components are tested during the 10-year interval. The component will be sectioned to size the inside surface flaws, if present, and/or to map the dealloyed surface areas for determining the degree of the dealloying. The samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

An engineering evaluation will be performed at the end of each test to determine if the sample size requires adjustment based on the results of the tests. The structural integrity analyses will be updated as required to validate adequate load carrying capacity.

*Monitoring and Trending (Element 5)*

The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

The ultimate tensile strength results from the metallurgical aluminum bronze material testing will be monitored and trended. Trending provides monitoring of the degree of dealloying, the degree of cracking, and the ultimate tensile strength for aging aluminum bronze material through the period of extended operation. Upon completion of each test, the data trended will be evaluated against the acceptance criteria for ultimate tensile strength.

*Acceptance Criteria (Element 6)*

Dealloying of aluminum bronze components is a well known phenomenon at STP. A long term improvement plan was developed in May 1992. As a result of these analyses, aluminum bronze (copper alloys with greater than eight percent aluminum) components are visually inspected every six months (not to exceed nine months). Upon discovery of visual evidence of through-wall dealloying, components are evaluated, and scheduled for replacement by the corrective action program. Components~~, greater than one inch,~~ will be replaced by the end of the next refueling outage. Due to the slow nature of dealloying, this replacement interval provides reasonable assurance that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

The acceptance criterion for ultimate tensile strength value of aluminum bronze material is greater than or equal to 30 ksi. The acceptance criterion for the fracture toughness is greater than or equal to 65 ksi $in^{1/2}$ for aluminum bronze castings and at welded joints in the heat affected zones. The acceptance criterion for yield strength is equal to or greater than one-half of the ultimate strength. If a criterion is not met, the condition will be documented in the corrective action program to perform a structural integrity analysis to confirm that the load carrying capacity of the tested material remains adequate to support the intended function of the ECW System through the period of extended operation.

*Corrective Actions (Element 7)*

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing corrective actions. The QA program includes elements of corrective action, and is applicable to the safety-related and nonsafety-related systems, structures and components that are subject to aging management review.

*Confirmation Process (Element 8)*

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing confirmation processes and administrative controls. The QA program includes elements of corrective action, and is applicable to the safety-related and nonsafety-related systems, structures and components that are subject to aging management review.

*Administrative Controls* (Element 9)

See Element 8.

*Operating Experience (Element 10)*

A review of the STP plant-specific operating experience indicates that macrofouling, general corrosion, erosion-corrosion, and through-wall dealloying have been observed in aluminum bronze components. STP has analyzed the effects of the through-wall dealloying and found that the degradation is slow so that rapid or catastrophic failure is not a consideration. STP has determined that the leakage can be detected before the flaw reaches a limiting size that would affect the intended functions of the essential cooling water and essential cooling water screen wash system. A long range improvement plan and engineering evaluation were developed to deal with the dealloying of aluminum bronze components when dealloying has been identified. Based on these analyses, the approach has been to evaluate components, and schedule replacement by the corrective action program. Components with indications of through wall dealloying~~, associated with piping greater than one inch in diameter,~~ will be replaced by the end of the next refueling outage. A monitoring and inspection program provides confidence in the ability to detect the leakage.

**Enhancements**

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

*Scope of Program (Element 1)*

Procedures will be enhanced to:

Perform volumetric examinations of aluminum bronze material components that demonstrate external leakage where the configuration supports this type of examination to conclude with reasonable assurance that cracks are not approaching a critical size.

Perform destructive examination of each leaking component removed from service to determine the degree of dealloying until 10 percent of the susceptible components in the ECW system are examined. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Prior to the period of extended operation, metallurgical testing of leaking aluminum bronze material components in the ECW system removed from service will be performed to update the structural integrity analyses, to confirm load carrying capacity and to determine the degree of dealloying by destructive examination. Metallurgical testing of the removed leaking component will be performed until at least three different size components of two samples each are tested, and at least nine total samples are tested. The metallurgical testing will include fracture toughness testing of test samples that include a crack in the dealloyed material where sufficient sample size supports bend testing. Additionally, the samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

As part of the testing described above, test six samples from three aluminum bronze components removed from service in 2012 for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. The aluminum bronze test samples exposed to ECW system raw water environment are to come from a pump shaft line casing pipe and from two small cast valve bodies. The pump shaft line casing pipe was removed from service in 2012 and the two small cast valve bodies will be removed from service in 2012. Priority shall be given to selecting 100% dealloyed component samples.

Beginning 10 years prior to the period of extended operation for each 10-year interval, periodically test samples of above ground ECW system components removed from service for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. For each 10 year interval beginning 10 years prior to the period of extended operation, 20 percent of leaking components removed from service, but at least one, will be tested every five years. Tensile test samples from a removed component shall be tested to include both leaking and non-leaking portions of the component. If at least two leaking components are not identified two years prior to the end of each 10-year testing interval, a risk-ranked approach will be used based on those components most susceptible to degradation to identify candidate components for removal and testing so at least two components are tested during the 10-year interval. The component will be sectioned to size the inside surface flaws, if present, and/or mapping of the dealloyed surface areas for determining the degree of the dealloying. The samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Perform an engineering evaluation at the end of each test to determine if the sample size requires adjustment based on the results of the tests.

Perform a structural integrity analysis to confirm that the load carrying capacity of the tested material remains adequate to support the intended function of the ECW system through the period of extended operation.

*Parameters Monitored and Inspected (Element 3)*

Procedures will be enhanced to indicate that whenever aluminum bronze materials are exposed during inspection of the buried essential cooling water piping, the components are examined for indications of selective leaching. If leaking below-grade welds are discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive metallurgical examination.

*Detection of Aging Effects (Element 4)*

Procedures will be enhanced to:

Indicate that whenever aluminum bronze materials are exposed during inspection of the buried essential cooling water piping, the components are examined for indications of selective leaching. If leaking below-grade welds are discovered by surface water monitoring or during a buried ECW piping inspection, a section of each leaking weld will be removed for destructive examination.

Perform volumetric examinations of aluminum bronze material components that demonstrate external leakage where the configuration supports this type of examination to conclude with reasonable assurance that cracks are not approaching a critical size.

Perform destructive examination of each leaking component removed from service to determine the degree of dealloying until 10 percent of the susceptible components in the ECW system are examined. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Metallurgical testing of leaking aluminum bronze material components in the ECW system removed from service will be performed to update the structural integrity analyses, to confirm load carrying capacity and to determine the degree of dealloying by destructive examination. Metallurgical testing of the removed leaking component will be performed until at least three different size components of two samples each are tested, and at least nine total samples are tested. The metallurgical testing will include fracture toughness testing of test samples that include a crack in the dealloyed material where sufficient sample size supports bend testing. Additionally, the samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

As part of the testing described above, test six samples from three aluminum bronze components removed from service in 2012 for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. The aluminum bronze test samples exposed to ECW system raw water environment are to come from a pump shaft line casing pipe and from two small cast valve bodies. The pump shaft line casing pipe was removed from service in 2012 and the two small cast valve bodies will be removed from service in 2012. Priority shall be given to selecting 100% dealloyed component samples.

Beginning 10 years prior to the period of extended operation for each 10-year interval, periodically test samples of above ground ECW system components removed from service for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. For each 10 year interval beginning 10 years prior to the period of extended operation, 20 percent of leaking components removed from service, but at least one, will be tested every five years. Tensile test samples from a removed component shall be tested to include both leaking and non-leaking portions of the component. If at least two leaking components are not identified two years prior to the end of each 10-year testing interval, a risk-ranked approach will be used based on those components most susceptible to degradation to identify candidate components for removal and testing so at least two components are tested during the 10-year interval. The component will be sectioned to size the inside surface flaws, if present, and/or mapping of the dealloyed surface areas for determining the degree of the dealloying. The samples will be tested for chemical composition including aluminum content, mechanical properties (such as yield and ultimate tensile strengths) and microstructure. Ultimate tensile strength will be trended and compared to the acceptance criterion. The degree of dealloying and cracking will be trended by comparing examination results with previous examination results.

Perform an engineering evaluation at the end of each test to determine if the sample size requires adjustment based on the results of the tests.

Perform a structural integrity analysis to confirm that the load carrying capacity of the tested material remains adequate to support the intended function of the ECW system through the period of extended operation.

*Monitoring and Trending (Element 5)*

Procedures will be enhanced to:

Trend the degree of dealloying and cracking by comparing examination results with previous examination results.

Trend ultimate tensile strength results from the metallurgical aluminum bronze material testing.

Upon completion of each test, evaluate the data trended against the acceptance criteria for ultimate tensile strength.

*Acceptance Criteria (Element 6)*

Procedures will be enhanced to:

Specify the acceptance criterion for ultimate tensile strength value of aluminum bronze material is greater than or equal to 30 ksi.

Specify the acceptance criterion for fracture toughness is 65 ksi $in^{1/2}$ for aluminum bronze castings and at welded joints in the heat affected zones.

Specify the acceptance criterion for yield strength is equal to or greater than one-half of the ultimate strength.

Initiate a corrective action document when the acceptance the criterion is not met.

**Conclusion**

The continued implementation of the Selective Leaching of Aluminum Bronze program provides reasonable assurance that aging effects will be managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

## A4 LICENSE RENEWAL COMMITMENTS

Table A4-1 identifies proposed actions committed to by STPNOC for STP Units 1 and 2 in its License Renewal Application. These and other actions are proposed regulatory commitments. This list will be revised, as necessary, in subsequent amendments to reflect changes resulting from NRC questions and STPNOC responses. STPNOC will utilize the STP commitment tracking system to track regulatory commitments. The Condition Report (CR) number in the Implementation Schedule column of the table is for STPNOC tracking purposes and is not part of the amended LRA.

*Table A4-1 License Renewal Commitments*

| Item # | Commitment | LRA Section | Implementation Schedule |
|---|---|---|---|
| 27 | Implement the PWR Reactor Internals program as described in LRA Section B2.1.35. | B2.1.35 | ~~Within 24 months after the issuance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* MRP-227-A~~<br><br>Completed<br><br>CR 10-23602 |
| 37 | Groundwater samples will be taken at multiple locations around the site every three months for at least 24 consecutive months. The samples will analyze for pH, sulfates, and chlorides. This sampling plan will begin no later than September 2012. | B2.1.32 | ~~September 2012~~<br>Completed<br><br>CR 11-20856-1 |
| 42 | Enhance the Reactor Head Closure Studs program procedures to:<br>• perform a remote VT-1 of stud insert #30 (Unit 2 only) concurrent with the volumetric examination once every 10 years to verify no additional loss of bearing surface area. | B2.1.3 | Starting with the current (Third Interval) 10-year ASME Section XI |

| | | | |
|---|---|---|---|
| | | | inspection interval<br><br>CR 12-15170 |
| 43 | The seal cap enclosures from Unit 2 Safety Injection System Check Valve SI0010A and from Unit 1 and Unit 2 Chemical Volume Control System Check Valves CV0001, CV0002, CV0004, and CV0005 will be permanently removed. After removal of the seal cap enclosures, the component bolting will be replaced or inspected for intergranular stress corrosion cracking. | B2.1.7 | ~~2012 Refueling Outage (~~Unit 1 completed~~);~~<br><br>2013 Refueling Outage (Unit 2)<br><br>CR 12-21155 |