SECTION 4.0 TIME-LIMITED AGING ANALYSES

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4.0 TIME LIMITED AGING ANALYSES

4.1 INTRODUCTION

Under the 10 CFR 54 License Renewal Rule (the Rule), an analysis, calculation, or evaluation is a "Time-Limited Aging Analysis" (TLAA) only if it meets all six of the defining criteria per 10CFR54.3(a):

> *Time-limited aging analyses*, for the purposes of this part, are those licensee calculations and analyses that
> (1) Involve systems, structures, and components within the scope of license renewal;
> (2) Consider the effects of aging;
> (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
> (4) Were determined to be relevant by the licensee in making a safety determination;
> (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions...; and
> (6) Are contained or incorporated by reference in the CLB [current licensing basis].

4.1.1 Identification of TLAAs

The methods used to identify PBAPS plant-specific TLAAs are consistent with the NRC draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP) and with NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule."

The Rule requires a list of TLAAs, including license exemptions which are based on TLAAs, be included in a license application. There are no exemptions granted that are based upon time-limited aging analyses.

Exelon assembled a list of generic potential TLAAs from the SRP and from industry guidance, performed a search of PBAPS current licensing basis (CLB) and supporting documents to confirm the occurrence of plant-specific instances of TLAAs suggested by the NRC and industry guidance, and to identify additional potential plant-specific (or unit-specific) TLAAs. Exelon then screened the resulting list of potential TLAAs against the six 10 CFR 54.3 criteria, above.

The Rule requires that these TLAAs then be evaluated to demonstrate that

(i) The analyses remain valid for the period of extended operation;
(ii) The analyses have been projected to the end of the period of extended operation; or
(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

[10 CFR 54.21(c)]

Exelon dispositioned each TLAA by one of these three methods. This section provides the list of TLAAs, a summary of each evaluation, and a description of each disposition.

4.1.2 Summary of Results

This review identified six general categories of TLAAs applicable to PBAPS. They are described in Sections 4.2 through 4.7 of this Section, with their dispositions, and are listed in Table 4.1-1, List of Time-Limited Aging Analyses (TLAAs).

Table 4.1-1 List of Time-Limited Aging Analyses (TLAAs)

TLAA	Description	Disposition Category	Section
1.	Reactor Vessel Neutron Embrittlement		4.2
	10 CFR 50 Appendix G Reactor Vessel Rapid Failure Propagation and Brittle Fracture Considerations: Charpy Upper-Shelf Energy (USE) Reduction and RTNDT Increase, Reflood Reshock Analysis	Revision of the Analysis and Validation of the Analysis for the period of Extended Operation	4.2.1
	Reactor Vessel Thermal Limit Analyses: Operating Pressure-Temperature Limit (P-T Limit) Curves	Revision of the Analysis	4.2.2
	Reactor Vessel Circumferential Weld Examination Relief	Revision of the Analysis	4.2.3
	Reactor Vessel Axial Weld Failure Probability	Validation of the Analysis for the period of Extended Operation	4.2.4
2.	Metal Fatigue		4.3
	Reactor Vessel Fatigue	Management of the Aging Effect	4.3.1
	Reactor Vessel Internals Fatigue and Embrittlement		4.3.2
	Reactor Vessel Internals Fatigue Analyses	Validation of the Analysis for the period of Extended Operation and Management of the aging effect	4.3.2.1
	Reactor Vessel Internals Embrittlement Analyses	Validation of the Analysis for the period of Extended Operation	4.3.2.2
	Effect of Fatigue and Embrittlement on End-of-Life Reflood Thermal Shock Analysis	Validation of the analysis for the period of extended operation	4.3.2.3

Table 4.1-1 List of Time-Limited Aging Analyses (TLAAs) (Continued)

TLAA	Description	Disposition Category	Section
	Piping and Component Fatigue and Thermal Cycles		4.3.3
	Fatigue Analyses of Group I Primary System Piping	Management of the aging effect	4.3.3.1
	Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction in Group II and III Piping and Components	Validation of the analysis for the period of extended operation	4.3.3.2
	Design of the RHR System for a Finite Number of Cycles	Validation of the analysis for the period of extended operation	4.3.3.3
	Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)	Management of the aging effect	4.3.4
3	Environmental Qualification of Electrical Equipment	Management of the aging effect	4.4
4.	Loss of Prestress in Concrete Containment Tendons Not Applicable. (PBAPS containments do not have prestress tendons.)	Not Applicable	4.5

Table 4.1-1 List of Time-Limited Aging Analyses (TLAAs) (Continued)

TLAA	Description	Disposition Category	Section
5.	Containment Fatigue		4.6
	Fatigue Analyses of Containment Boundaries: New Loads Analysis of Tori, Torus Vents, and Torus Penetrations	Validation of the analysis for the period of extended operation and Management of the aging effect	4.6.1
	New Loads Fatigue Analysis of SRV Discharge Lines and External Torus-Attached Piping	Validation of the analysis for the period of extended operation	4.6.2
	Expansion Joint and Bellows Fatigue Analyses - Drywell to Torus Vent Bellows	Validation of the analysis for the period of extended operation	4.6.3
	Expansion Joint and Bellows Fatigue Analyses - Containment Penetration Bellows	Validation of the analysis for the period of extended operation	4.6.4
6.	Other Plant Specific TLAAs		4.7
	Reactor Vessel Corrosion Allowances	Validation of the analysis for the period of extended operation	4.7.1

Table 4.1-1 List of Time-Limited Aging Analyses (TLAAs) (Continued)

TLAA	Description	Disposition Category	Section
	Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"	Management of the aging effect	4.7.2
	Fracture Mechanics of ISI-Reportable Indications for Group I Piping: As-forged laminar tear in a Unit 3 Main Steam elbow near weld 1-B-3BC-LDO discovered during preservice UT	Validation of the analysis for the period of extended operation	4.7.3

4.1.3 Identification of Exemptions

The rule requires a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3.

A search of docketed correspondence, the operating licenses, and the Updated Final Safety Analyses Report (UFSAR) identified and listed all exemptions in effect. Each exemption in effect was then evaluated to determine if it involved a TLAA as defined in 10 CFR 54.3.

There are no exemptions granted that are based upon time-limited aging analyses.

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The PBAPS Unit 2 and 3 reactor vessels are described in UFSAR Chapter 4. Reactor vessel materials are subject to embrittlement, primarily due to exposure to neutron radiation. "Embrittlement" means the material will adsorb less energy during a crack or rupture, and therefore that a crack could more easily propagate under load.

In addition, adsorbed energy is temperature dependent. In most materials adsorbed energy increases with temperature up to a maximum (the "upper-shelf energy," USE). Neutron embrittlement decreases USE. Because fracture energy is low at low temperature, operating pressure-temperature limit curves (P-T curves) are included in Technical Specifications which dictate the limit to which the vessel can be pressurized at a given temperature. RT_{NDT}, nil-ductility transition reference temperature, is determined for vessel materials before irradiation and indicates temperatures above which impact tests will demonstrate an acceptable USE. Neutron embrittlement raises this transition temperature. This increase (• RT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile fashion. The P-T curves are determined by the RT_{NDT} and • RT_{NDT} calculations for the licensed operating period.

These limits and effects are calculated on the basis of lifetime neutron fluence, are part of the licensing basis, and support safety determinations. Their calculations are therefore TLAAs. The supporting calculation of vessel neutron fluence is similarly a TLAA. The increases in neutron fluence and RT_{NDT} (• RT_{NDT}) also affect the bases for relief from circumferential weld inspection and its associated supporting calculation of limiting axial weld conditional failure probability. Circumferential weld examination relief and axial weld failure probability are thereby also TLAAs.

4.2.1 10 CFR 50 Appendix G Reactor Vessel Rapid Failure Propagation and Brittle Fracture Considerations: Charpy Upper-Shelf Energy (USE) Reduction and RT_{NDT} Increase, Reflood Thermal Shock Analysis

Although the scope of reactor vessel embrittlement concerns includes the heads, nozzles, nozzle safe ends, and closure studs, the neutron fluence is highest in the beltline region of the vessel shell. Differences in vessel materials at different locations might somewhat offset or balance this effect, but for the PBAPS vessels the limiting USE occurs in the beltline region.

Fluence, USE, and RT_{NDT}

The tests done on vessel materials under the code of record supplied limited Charpy impact data. It was therefore not possible to develop original Charpy upper shelf energy values using the ASME III NB-2300, S'1972 (et seq.) methods invoked by 10 CFR 50 Appendix G. A 10 CFR 50 Appendix G equivalent margin analysis was therefore used to demonstrate compliance with the 40-year USE requirement, and to support the RT_{NDT} determinations.

The predicted rise in RT_{NDT} due to temperature-dependent precipitation and neutron fluence, and the consequent operating margins, were calculated for a 40 year design with 32 effective full-power years (EFPY) of operating life.

Reflood Thermal Shock Analysis

An end-of-life thermal shock analysis for a recirculation line break followed by ECCS injection was performed on a typical BWR-4 reactor vessel [Reference 4.19]. A subsequent analysis for BWR-6 vessels is equally applicable to BWR-4 vessels [Reference 4.20]. These analyses assume end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature and therefore, end-of-life neutron fluence. These analyses are therefore TLAAs.

Analysis

Fluence, USE, and RT_{NDT}

An acceptable minimum vessel end-of-life USE will be confirmed using the Boiling Water Reactor Vessel and Internals Program Report 74 methodology [BWRVIP-74, Ref. 16].

The vessel end-of-life RT_{NDT} will be recalculated for a 60-year licensed operating life (54 EFPY) under Code Case N640.

Exelon has not yet performed these calculations. Exelon will initiate these calculations after the GE fluence methodology has been approved by the NRC.

Reflood Thermal Shock Analysis

The critical location for the fracture mechanics analysis is at ¼ of vessel thickness from the inside. For this event, the peak stress intensity occurs approximately 300 seconds after LOCA. At 300 seconds, this analysis shows that the temperature of the vessel wall at 1.5 inches deep (which bounds the PBAPS case) is approximately 400 °F. Even after 60 years of irradiation, the vessel beltline material adjusted reference temperature will be low enough to ensure that at 400 °F, the material is in the Charpy upper shelf region. Therefore, the analysis remains bounding and valid for the license renewal term.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Revised end-of-life fluence, USE, and RT_{NDT} will be calculated for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). These calculations will be completed and acceptable values will be confirmed prior to the end of the initial operating license term for PBAPS.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The reflood thermal shock analysis was examined and remains valid and bounding for the period of extended operation.

4.2.2 Reactor Vessel Thermal Limit Analyses: Operating Pressure-Temperature Limit (P-T Limit) Curves

PBAPS Units 2 and 3 Technical Specifications, Section 3.4.9, contain P-T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and also limit the maximum rate of change of reactor coolant temperature. The criticality curve provides limits for both heatup and criticality, calculated for a 40 year design, 32 EFPY operating period. The P-T curves are not presently limited by beltline neutron fluence embrittlement.

Analysis

Exelon will perform these calculations for a 60 year, 54 EFPY operating period. Exelon will initiate these calculations after the GE fluence methodology has been approved by the NRC.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Revised vessel P-T limit curves will be calculated for the extended licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii). These calculations will be completed and acceptable values will be confirmed prior to the end of the initial operating license term for PBAPS.

4.2.3 Reactor Vessel Circumferential Weld Examination Relief

Relief from RPV circumferential weld examination requirements under Generic Letter 98-05 is based on probabilistic risk assessments which predict an acceptable probability of failure per reactor operating year, with vessel metallurgical conditions and indication sizes and frequencies assumed to be those expected at the end of a licensed operating period, nominally 32 EFPY for the original 40-year term.

The additional changes in metallurgy and additional growth of indications expected over an extended licensed operating period make any such relief a TLAA, if the relief is extended.

Exelon has been granted relief from the requirements for inspection of RPV circumferential welds for the remainder of the current 40-year licensed operating period [Reference 4.14 and 4.15]. The justification is consistent with BWRVIP-05 guidelines, showing metal chemistry limits and predicted effects at 32 EFPY within the limits assumed by the generic BWRVIP-05 analysis.

Analysis

Exelon will apply for an extension of this relief for the 60-year extended licensed operating period, with justification based on the guidelines of BWRVIP-74 when approved, or based on the guidelines of BWRVIP-05.

The procedures and training that will be used to limit the frequency of cold over-pressure events to the number specified in the relief request extension, during the license renewal term, are the same as those used in the current licensed operating period. Details of these procedural and training provisions are provided in the relief request for the current licensed operating period [Ref. 4.13].

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Exelon will apply for an extension of the existing relief from circumferential weld inspection for the 60-year extended licensed operating period. This extension will require revision of supporting analyses and evaluations for the 60-year extended licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii). Exelon will apply for this extension in time for it to be granted prior to the end of the initial operating license term for PBAPS.

4.2.4 Reactor Vessel Axial Weld Failure Probability

The Boiling Water Reactor Owner's Group Vessel and Internals Program recommendations for inspection of reactor vessel shell welds [BWRVIP 05, Ref. 4.15] contain generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year.

BWRVIP-05 showed that this axial weld failure rate of 5×10^{-6} per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds, as described in Section 4.2.3.

Analysis

Supporting calculations of expected 60-year neutron fluence, USE, and RT_{NDT} will be completed before the current 40-year operating license expires. Exelon has not yet performed these calculations. Exelon will initiate these calculations after the GE fluence methodology has been approved by the NRC. Exelon has reasonable assurance that the PBAPS vessel parameters will permit a demonstration that the expected failure probability of the limiting axial weld in each unit will still remain less than the 5×10^{-6} per reactor year.

The operator training and procedures that will be used to limit the frequency of cold over-pressure events during the license renewal term are the same as those currently used, as described in the "Analysis" of "Reactor Vessel Circumferential Weld Examination Relief," Subsection 4.2.3, above.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Supporting analyses will be completed to confirm that the assumed limit on the axial weld failure probability remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). Exelon will complete these analyses prior to the end of the initial operating license term for PBAPS.

4.3 METAL FATIGUE

4.3.1 Reactor Vessel Fatigue

Reactor Vessel Fatigue Analyses, RPV Nozzle Thermal Cycle Count, and Reactor Vessel Stud Fatigue Analyses

The PBAPS Unit 2 and Unit 3 reactor vessel fatigue analyses, which include the vessel shell, head, nozzles, nozzle safe ends, and closure studs, depend on cycle count assumptions that assume a 40-year operating period. Applicable analyses have been revised to incorporate licensing changes for power uprate and other operational changes. The analyses demonstrate that the 40-year cumulative usage factors (CUFs) for the critical components of the vessel are below the ASME Code Section III design value of 1.0, except for the closure studs which are included in a fatigue management program that provides for dispositioning if that program indicates the code design value will be exceeded. The current analyses of record are TLAAs.

Analysis

The existing program maintains a count of cumulative reactor pressure vessel thermal and pressure cycles to ensure that licensing and design basis assumptions are not exceeded. An improved program is being implemented which will use temperature, pressure, and flow data to calculate and record accumulated usage factors for critical RPV locations and subcomponents.

Existing reactor vessel fatigue analyses have been reviewed to establish a bounding set of RPV locations for inclusion in the fatigue management program. All locations with 40-year CUFs expected to exceed 0.4 are included. CUF equations will be updated as necessary to incorporate any analysis revisions, and plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored components.

The following information on the Core ΔP/SBLC Nozzle is included because of a commitment to BWRVIP-27 [Ref. 4.17]. Exelon is using the license renewal Appendix B of the BWRVIP-27 guidelines for inspection and flaw evaluation of the standby liquid control system and core plate • P lines and their common nozzles. This appendix commits each applicant who invokes it to list fatigue of this nozzle (core • P/SBLC nozzle) as a TLAA, and to describe the usage factor and aging management plan or other disposition.

The original PBAPS design analysis found that the stresses and the expected number of significant cycles in the core • P/SBLC nozzles were in accordance with Section III, Paragraph N-415.1 of the code of record, and were therefore less than those that required a fatigue analysis. Therefore, no CUF was calculated. Any CUF which might be calculated would be negligible. The fatigue management program will monitor other, higher-usage factor locations. Any potentially-significant increase in the CUF for these core • P/SBLC nozzles will be indicated by a significant increase, above predicted values, in the CUFs monitored in these other locations.

Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" is discussed in Section 4.7.2.

Disposition: Aging Management, 10CFR54.21(c)(1) (iii)

This TLAA will require management of the aging effects (10 CFR 54.21(c)(1)(iii)). Refer to Appendix B, Section B.4.2, Fatigue Management Activities. The required implementing actions will be completed prior to the end of the initial operating license term for PBAPS.

The fatigue management program will monitor at least the reactor vessel, reactor vessel internals, and piping components listed in Table 4.3.1-1 (in both units, except as indicated).

Table 4.3.1-1: Fatigue Monitoring Program Locations

Location
Reactor Pressure Vessel (RPV) Feedwater nozzles (Loops A and B)
RPV support skirt
RPV closure studs
RPV shroud support
RPV Core Spray nozzle safe end
RPV Recirculation inlet nozzle
RPV Recirculation outlet nozzle
RPV Refueling containment skirt
RPV Jet Pump shroud support
Residual Heat Removal (RHR) 24" return line (Loop A)
RHR 20" supply line (Loops A and B)
RHR Tee (Loop A)
RHR Tee (Loop B)
Feedwater piping (Node 754)
Main Steam piping (Node 606)
Torus Penetrations Unit 2
Torus Penetrations Unit 3
Torus Shell

4.3.2 Reactor Vessel Internals Fatigue and Embrittlement

This discussion of fatigue in the reactor vessel internals also includes evaluation of radiation embrittlement.

4.3.2.1 Reactor Vessel Internals Fatigue Analyses

Original Core Shroud, Shroud Support and Jet Pump Assembly Evaluation: Fatigue in these components is from two sources, system cycles and vibrations. The vibration effects were based on tests and analysis of standard vessel internals under forced recirculation flow. Vibration effects in the core shrouds, shroud supports, and jet pump assemblies were evaluated in a standard-plant analysis applicable to PBAPS. The design analysis was based on the cyclic stress criteria of ASME Section III.

The core shroud and jet pump assembly analysis was performed for a plant where the configuration (leg type shroud support) applies to PBAPS. Therefore, the calculated fatigue usage is expected to be a reasonable approximation for this plant. The limiting 40 year cumulative fatigue usage factor is 0.35.

Revised Core Shroud Support Analysis: Fatigue analyses of the core shroud supports were reevaluated for effects of increased recirculation pump starts with the loop outside thermal limits. The reevaluation used conservative estimates of the expected number of these starts in 40 years, and conservative methods of estimating the fatigue usage factor. The evaluation resulted in an estimated upper bound on the 40-year usage factor of 0.834.

Unit 3 Core Spray Pipe to Tee Box Weld Crack Repair Bracket: Repair brackets and attachment welds to the piping installed in PBAPS Unit 3 in 1985 were analyzed in accordance with the requirements of ASME Section III Article NB-3200.

Section III of the ASME code does not strictly apply to internals, which are not code pressure boundary items, and is used as a guide only. However, comparison to the code design CUF of 1.0 was used to support safety determinations.

Analysis

Core Shroud, and Jet Pump Assembly: Maximum CUF for the original analyses of vibration and fatigue in the core shroud and jet pump assembly is 0.35 for a 40-year life. Therefore, the predicted 60-year CUF would not exceed 0.525 (0.35CUFx1.5), which is less than the code design value of 1.0 CUF, and no aging management activity is required.

Core Shroud Supports: The usage factor at the critical core shroud support locations will be managed by the fatigue management program cycle counting and fatigue usage factor tracking program used by the fatigue management program described in Section B.4.2. The fatigue management program will ensure that the fatigue effects in the shroud supports will be adequately managed and will be maintained within the code design value for the period of extended operation. The critical shroud support locations monitored by the fatigue management program are 'RPV shroud support' and the 'RPV Jet Pump shroud support' in Table 4.3.1-1.

Unit 3 Core Spray Pipe to Tee Box Weld Crack Repair Bracket: The greatest contributor to the high usage factor in the Unit 3 Core Spray pipe to tee box weld crack repair bracket fatigue analysis is not dependent on the licensed operating period but is due to a single seismic event. The usage factor will not change significantly with an increase in the licensed operating period, and no aging management activity is required.

Disposition: Validation, 10 CFR 54.21(c)(1)(i); and Aging Management, 10 CFR 54.21(c)(1)(iii)

The core shroud, jet pump assembly, and Unit 3 Core Spray pipe to tee box weld crack repair bracket fatigue evaluations have been reviewed and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Fatigue in the critical locations of the Jet Pump shroud support and the RPV shroud support will be managed by the fatigue management program. (10 CFR 54.21(c)(1)(iii)). Refer to Section B.4.2, Fatigue Management Activities. The required implementing actions will be completed prior to the end of the initial operating license term for PBAPS.

4.3.2.2 Reactor Vessel Internals Embrittlement Analyses

These analyses affect the core shroud and top guides. The fluences expected for 40 years may cause shroud and top guide embrittlement or irradiation-assisted stress corrosion cracking (IASCC), and must therefore be addressed for the 60 year period. The expected 40-year fluence of the most irradiated point on the inner surface of the shroud is 2.7×10^{20} nvt (>1 Mev).

Analysis

BWRVIP-26 [Ref. 4.18] lists 5×10^{20} nvt as the threshold fluence beyond which the components will be significantly affected [Ref. 4.18 § 2.1.1].

Shroud: The expected 60-year fluence on the shroud,

$$2.7 \times 10^{20} \text{ nvt} \times 60/40 = 4.5 \times 10^{20} \text{ nvt},$$

is below the 5×10^{20} nvt damage threshold.

Top Guide: License Renewal Appendix C to BWRVIP-26 states that the generic fluence for 60 years on the top guide is 6×10^{21} nvt. Although this 60-year fluence will be above the 5×10^{20} nvt damage threshold, the tensile stresses in this component are very low. At these low stresses a fracture is not a concern, and embrittlement is, therefore, not a threat to the intended function. These critical locations in the top guide are exempt from inspection under the approved BWRVIP-26 and no aging management activity is required.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The reactor vessel internals embrittlement analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.2.3 Effect of Fatigue and Embrittlement on End-of-Life Reflood Thermal Shock Analysis

Radiation embrittlement and fatigue usage may affect the ability of certain internals, particularly the core shroud support plate, to withstand an end-of-life reflood thermal shock following a recirculation line break.

These thermal shock analyses assume end-of-life fatigue and embrittlement effects and are considered TLAAs.

Analysis

The effects of embrittlement and fatigue on the end-of-life reflood thermal shock analyses were evaluated. The thermal shock analyses were validated for the 60-year extended operating term. The effects of embrittlement are not significant at higher usage factor locations, and the effects of fatigue are not significant at locations where embrittlement is significant. The net effect in each analyzed location is acceptable. The thermal shock analyses are, therefore, acceptable for the extended operating period.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The effect of fatigue and embrittlement on end-of-life reflood thermal shock analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.3 Piping and Component Fatigue and Thermal Cycles

This section includes fatigue-related TLAAs arising within design analyses of:

- Group I, II, and III piping and components, except the reactor vessel and internals (Group I, II, and III are described in UFSAR App. A)
  - Group I piping with an ASME III Class 1 Fatigue Analysis
  - Group I, II and III piping to USAS-B31.1 (ANSI-B31.1)
  - Group II and III components to ASME III Class C (1965) - ASME VIII Division 2 Alternative Rules
- Residual Heat Removal (RHR) system operating cycles

RHR components are treated separately because specific analyses for the RHR system and its components were identified as TLAAs.

4.3.3.1 Fatigue Analyses of Group I Primary System Piping

All Group I piping was originally designed in accordance with USAS B31.1, 1967 Edition. The Recirculation and RHR piping in Group 1 was replaced under the GL 88-01 IGSCC Correction Program. The replacement piping was analyzed to ASME Section III Class 1 rules. ASME Section III requires an explicit fatigue analysis for Class 1 components. The ASME Code limits the design CUF to less than 1.0.

USAS B31.1 does not require an explicit fatigue analysis. Therefore, CUF values were not originally calculated for the remainder of the PBAPS Group I piping designed to this code. However, a simplified fatigue analysis was developed to estimate CUFs and to construct the algorithms used by the fatigue management program to estimate current CUFs from operating data.

Both the ASME Section III Class 1 fatigue analyses and the fatigue management program analyses of USAS B31.1 piping are TLAAs.

<u>Analysis</u>

Group I piping fatigue issues will be managed by the fatigue management program. Additional evaluations have been developed to establish a comprehensive and bounding set of Group I piping locations for inclusion in the fatigue management program. All locations with 40-year CUFs expected to exceed 0.4 will be included. The CUF modeling equations will be updated as necessary in the future to incorporate any analysis revisions, and all necessary plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored locations.

Disposition: Aging Management, 10 CFR 54.21(c)(1) (iii)

This TLAA will require management of the aging effects (10 CFR 54.21(c)(1)(iii)). Refer to Section B.4.2, Fatigue Management Activities. The required implementing actions will be completed prior to the end of the initial operating license term for PBAPS.

4.3.3.2 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction in Group II and III Piping and Components

The original piping codes used in the design of Group II and III piping required the use of a stress range reduction factor in the evaluation of calculated stresses due to thermal expansion. The reduction factors were based on the anticipated number of equivalent full-temperature cycles over the 40 year life of the plant. The assumed thermal cycle count is therefore a TLAA applicable to all piping systems designed to these codes.

Analysis

The number of equivalent full-temperature cycles involved in the extended operation of PBAPS is significantly less than the 7,000 cycles used as the threshold number for applying stress range reduction factors in the applicable code piping analyses. The 40 year life expected total number of cycles associated with these transients is 700. For conservatism, additional cycles were added for feedwater temperature changes due to load changes. Based on the accumulated cycles to date, an additional 1,000 cycles is appropriate to cover these types of feedwater transients. Therefore, the total number of cycles assumed for the 40-year plant life is conservatively less than 2,000. For the extended licensed operating period the total number of thermal cycles for piping analyses would, proportionately, be less than 3,000 during the entire plant life, which is still significantly less than the 7,000-cycle threshold. Therefore, the code stress range reduction factor remains at 1 and the stress range reduction factor used in all the original piping analyses will not be affected by extending the operating period to 60 years. The existing piping analyses within the scope of license renewal containing assumed thermal cycle counts are valid for the period of extended operation.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The existing piping analyses containing assumed thermal cycle counts have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.3.3 Design of the RHR System for a Finite Number of Cycles

The NSSS vendor specified the RHR System for a 40-year design life, and specified a finite number of cycles for each of its elevated-temperature operating modes. The specified number of cycles for each mode was based on BWR operating experience and engineering judgment at the time the plant was designed.

No description of these design operating cycles of the RHR systems was found in the PBAPS licensing basis documents. However, the design requirements of the RHR piping system are described in UFSAR Appendix A, Sections A.1 and A.9. Thermal cycles of the RHR piping and valves are, therefore, treated as TLAAs.

Analysis for RHR Valve and Piping Thermal Cycles

Group II RHR piping, and some Group I RHR piping, is designed to an ANSI B31.1 or USAS B31.1 allowable secondary stress range determined by a finite number of maximum-range thermal cycles, without a detailed fatigue analysis.

The ANSI B31.1 threshold number for applying a reduction factor for RHR valves is 7,000 equivalent full-temperature cycles. The total number of cycles assumed for the original 40-year plant life is, conservatively, less than 1,000. For the period of extended operation, the number of thermal cycles for piping analyses would be proportionately increased to 1,500, which is still significantly less than the 7,000 cycle threshold. The code stress range reduction factor therefore remains at 1.0 and is not affected by extending the operating period to 60 years.

Group I RHR piping inside the Drywell, which was analyzed to ASME Section III Class 1 rules, is discussed in Section 4.3.3.1.

Disposition for RHR Valve and Piping Thermal Cycles: Validation, 10 CFR 54.21(c)(1)(i)

The RHR valve and piping thermal cycle analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.3.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

ASME Section III uses stress versus design cycle curves (S-N curves) based on tests in air to determine fatigue usage factor. The environment of a stressed component can affect fatigue life. This concern was the subject of a series of NRC communications, including three Generic Safety Issues (GSIs). Of these GSI 190 [Ref. 4.3] identified NRC staff concerns about the potential effects of reactor water environments on component fatigue life during the period of extended operation. GSI-190 was closed in December 1999 [Ref. 4.4]. The NRC staff concluded that environmental effects have a negligible impact on core damage frequency and no generic regulatory action is required. However, as part of the closure of GSI-190, the staff also concluded that applicants for license renewal should address the effects of reactor coolant environment on component fatigue life as part of their aging management programs.

Analysis of Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping

Much of the conservatism in the design basis calculation process is due to design basis transient definitions. Exelon evaluated the effect of the reactor coolant environment on fatigue life, based on the EPRI fatigue study most applicable to PBAPS [Ref. 4.5]. The evaluation found that the effect of conservative design basis transient definitions, by themselves, provides a design margin, which encompasses any potential increase in fatigue usage factor due to reactor water environmental effects.

The evaluation found that a revised calculation would predict 60-year cumulative fatigue usage factors (CUFs), with environmental effects, that would be at least an order of magnitude less than the originally-calculated design basis 40-year CUFs, without environmental effects. At the sample location with the least difference between the revised and original calculation, the revised value would be only about eight per cent of the calculated design basis CUF. At all other sample locations the revised CUFs would be less than five per cent of the original CUFs. See Table 4.3.4-2.

Latest Environmental Fatigue Methodology

Exelon evaluated the effect of the reactor coolant environment on fatigue life based on EPRI Report TR-110356 [Ref. 4.5], the EPRI fatigue study most applicable to PBAPS. The NRC staff raised some issues [Ref. 6] with respect to the generic studies published by EPRI, including EPRI Report TR-110356. Those issues were considered in the assessment of the environmental fatigue at PBAPS.

Fatigue data have been generated by Argonne National Laboratory (ANL) in simulated reactor water environments since the EPRI generic studies. These data have resulted in improved environmental correction factor (F_{en}) correlations, which are documented in NUREG/CR-6583 for carbon and low-alloy steel [Ref. 4.7], and in NUREG/CR-5704 for stainless steel [Ref. 4.8]. These improved correlations were not available at the time all of the EPRI generic studies were performed. The NRC was concerned that the EPRI correlations do not reflect the Argonne data and are not as conservative as the NUREG correlations.

The improved NUREG/CR-6583 correlations for carbon and low-alloy steels do not differ substantially from those used in the EPRI generic studies. However, the change in strain threshold has a potentially significant effect, and that effect was therefore evaluated. The environmental effect on carbon steel was recalculated for one of the examples of EPRI TR-105759 [Ref. 4.9]. This is a carbon steel BWR feedwater piping location with a design basis 40-year CUF of 0.1409.

The EPRI TR-105759 method used an alternating *stress* threshold of 30 ksi, equivalent to a NUREG/CR-6583 alternating *strain* threshold of about 0.10%, to adjust the incremental fatigue usage. At this location this threshold included eight out of thirty-one load pairs, giving an additional (environmental) fatigue usage of 0.0477, for a 40-year adjusted total of 0.1886. The overall environmental multiplier (F_{en}) in this case was 1.34 (1.66 for the eight affected load pairs).

The NUREG/CR-6583 correlation would reduce the alternating strain threshold to 0.07%. This can be approximated in the EPRI TR-105769 methods by reducing the alternating stress threshold to 21 ksi. which would require an environmental adjustment for six additional load pairs, for a total design fatigue usage of 0.0803, before environmental effects, for the fourteen load pairs. Assuming that the F_{en} multiplier of 1.66 would continue to apply for the fourteen affected load pairs, the estimate for the adjusted fatigue usage factor would be 0.1409 - 0.0803 + 1.66 (0.0803) = 0.194. The overall correction factor F_{en} multiplier increases from 1.34 to only 1.38, or about 4 per cent.

Therefore, because the additional load pairs that would have to be included contribute relatively small increments to the total CUF, the change in the strain range threshold under the NUREG/CR-6583 correlation does not cause a significant increase in the CUF calculated by EPRI TR-105769 methods. Therefore, the results of the EPRI TR-105769 generic studies provide a reasonable estimate of potential environmental fatigue effects for carbon and low-alloy steel components, and remain valid without further modification.

For austenitic stainless steels, the differences between the NUREG/CR-5704 correlation and the EPRI generic studies are more significant. For the case of relatively low temperature (<200°C), a low (bounding) strain rate, and either high or low dissolved oxygen, the environmental correction factor is 2.55. For relatively high temperature (>200°C), low dissolved oxygen, and a low (bounding) strain rate, the environmental correction factor may be as high as 15.35, although there is a reduction above 250°C where the environmental factor decreases to about 3.20 at 340°C. These factors are higher than those obtained from the methods of the EPRI generic studies.

For most of the component locations evaluated in the EPRI generic studies, the Argonne data and NUREG/ CR-5704 correlation would still demonstrate that the 60-year CUF is less than 1.0, including reactor water environmental effects. Using a detailed, temperature-dependent F_{en} environmental correction factor approach has an advantage for these cases, since most of the penalizing thermal transients lie below the 200°C threshold. The calculated environmental shift is therefore relatively low provided separate multipliers are used for the portions of the transient which are above and below 200°C. However, for locations most sensitive to environmental effects the environmentally-adjusted CUF still increases over that calculated in the EPRI generic studies by a factor of about two. For BWR stainless steel materials a conservative, analysis correction factor of 2.0 was therefore applied to the EPRI generic study results to account for the more recent laboratory data and correlation. See <u>Table 4.3.4-2</u>.

<u>Application of the EPRI Generic Studies to PBAPS</u>

As mentioned earlier, the most applicable evaluation for PBAPS with respect to the EPRI generic studies is EPRI Report No. TR-110356. Those results are considered directly applicable to PBAPS. First, the results documented in that report apply to a BWR-4, which is identical to the PBAPS design. Therefore, the Class 1 systems associated with the plants are the same, which defines the characteristics of the thermal transients in these systems. These similarities are most clearly observed in the plant heat balance diagrams and thermal cycle definitions. In particular, the thermal cycle definitions for the RPV nozzles provide a good measure of the thermal characteristic similarities between plants, because they represent fluid variations based on the combinations of several plant systems prior to entering the RPV. The heat balance diagram and several key thermal cycle diagrams for the generic BWR-4 evaluated in EPRI Report No. TR-110356 were compared to similar diagrams for PBAPS. Comparison of these diagrams allows the following conclusions to be made:

- The feedwater inlet temperatures are within 2°F of each other (381.4°F for PBAPS vs. 383°F for generic BWR-4).
- The feedwater flow rates are within 5% of each other (14.247 Mlb/hr for PBAPS vs. 13.574 Mlb/hr for generic BWR-4). A similar situation exists for the steam and core flow rates.

- The recirculation inlet temperatures are within 2°F of each other (531.4°F for PBAPS vs. 529°F for generic BWR-4).
- The recirculation flow rates are the same for both plants (34.2 Mlb/hr for both plants).
- The dome pressures are within 3% of each other (1,050 psig for PBAPS vs. 1,020 psig for generic BWR-4).
- All like transients have the same profiles (i.e., they have the same "size and shape").

Further similarities between PBAPS and the generic BWR-4 evaluated in EPRI Report No. TR-110356 is demonstrated in Table 4.3.4-1, where the design basis transient types and quantities for both plants are compared.

As a result of the above comparisons, the design basis transient definitions associated with the plants are very similar, as expected for similar BWR-type plants. Therefore, it is reasonable to utilize the results and conclusions documented in EPRI Report No. TR-110356 for PBAPS, with some modification to incorporate the results of more recent laboratory testing (as described above).

Table 4.3.4-2 shows the CUF results from EPRI Report TR-110356, with modifications to account for the more recent data in NUREG/CR-6583 and NUREG/CR-5704, as described above. The original design basis CUF for each of the TR-110356 sample plant locations is also shown for comparison. Table 4.3.4-2 clearly demonstrates that the conservatism of design basis transient definitions encompasses all environmental effects. The marginal effect of the reactor coolant environment on CUF, projected to 60 years, is at least a factor of 12.9 below the original design basis CUF for all locations.

The BWR-4 evaluated in EPRI Report TR-110356 did not consider hydrogen water chemistry (HWC), as evidenced by the plots of dissolved oxygen in that report. Both units at PBAPS have implemented HWC. The maximum effect of the change in dissolved oxygen as a result of HWC implementation is adequately addressed by the conservative penalty factors described above.

Two materials issues may affect the application of the EPRI TR-110356 BWR-4 generic study to PBAPS. First, EPRI Report TR-110356 conservatively assumed the sulfur content, where applicable, was a maximum. Second, although the material types (i.e. stainless versus carbon or low-alloy steel) are similar between the two plants, differences were identified and were considered appropriately in all fatigue evaluations. Material types of most BWRs are very similar, as evidenced by the Table 4.3.4-3 comparison between PBAPS and the older vintage BWR-4 evaluated in NUREG/CR-6260 [Ref. 4.10]. Therefore, differences in materials do not have any effect on the application of the results of EPRI TR-110356 to PBAPS.

The six locations investigated in NUREG/CR-6260 for the older vintage BWR are listed in Table 4.3.4-3. The older vintage BWR results are applicable to PBAPS because the design transients and B31.1 piping design methodology are similar. The equivalent locations are included in the PBAPS fatigue management program, and the projected 60-year CUF values (using design basis transient severity to encompass environmental effects) are shown in Table 4.3.4-3. The CUF estimates for each location are based on plant operational data evaluated through November 12, 2000.

Table 4.3.4-3 shows that all BWR locations from NUREG/CR-6260 are included in the fatigue management program at PBAPS. The CUF values for all monitored locations, except for the Unit 3 RPV support skirt (CUF_{60} = 1.02), are projected to remain within the allowable value of 1.0 for the duration of the license renewal period. For the PBAPS Unit 3 RPV support skirt location, continued refinement of CUF estimates as additional operating experience is collected by the fatigue management program is expected to yield acceptable CUF results for the entire license renewal period.

The fatigue management program includes two locations that use stress-based fatigue (SBF) as the fatigue estimation basis, the RPV feedwater nozzles and the RPV support skirt. The CUF value for the RPV support skirt does not require any modification for potential environmentally assisted fatigue effects because this knuckle region, outside the RPV, is not exposed to reactor water. The CUF values for the RPV feedwater nozzle locations (one in each of two feedwater loops) are used for information only, because this component has an alternate aging management program as discussed in Section 4.7.2.

All of the remaining RPV, Class 1 piping, and torus locations included in the fatigue management program use a cycle-based fatigue (CBF) method. This method uses conservative design basis equations and actual plant transient counts to estimate the CUF. Equations were developed for each of the remaining limiting RPV locations and for limiting locations in each Class 1 piping system and the torus.

Table 4.3.4-2 and Table 4.3.4-3 show that the PBAPS fatigue management program conservatively estimates CUF based on design basis transient definitions, and that these estimates and the scope of the program are adequate to ensure that fatigue effects, including effects of the reactor coolant environment, will be kept below the design CUF limit of 1.0 throughout the license renewal period. The PBAPS fatigue management program also includes other locations beyond those evaluated in NUREG/CR-6260, and thereby provides more comprehensive CUF management. See Table 4.3.1-1 and Section B.4.2.

Disposition for Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping: Aging Management, 10 CFR 54.21(c)(1)(iii)

Exelon has implemented a fatigue management program. The fatigue management program tracks cumulative usage factor (CUF) for several bounding Class 1 pressure boundary and torus locations. The program is described in detail in Section 4.3 and Section B.4.2, "Fatigue Management Activities." The fatigue management program includes limiting locations in the reactor pressure vessel (RPV), Class 1 piping, and the torus. Exelon evaluated the current CUF based upon actual operating history for both PBAPS units to date to develop a baseline for this program. Exelon will continue to use the fatigue management program to monitor CUF for the Class 1 pressure boundary and torus locations throughout the license renewal period. See Table 4.3.1-1 for the list of monitored locations, and Table 4.3.4-3 and its notes, which describes how the NUREG/CR-6260 locations are monitored.

Much of the conservatism in the design basis calculation process is due to design basis transient definitions. As demonstrated in the "Analysis," above, the conservative design basis transient definitions by themselves bound any potential effects of the reactor water environment on fatigue usage factor. The fatigue monitoring program cycle-based fatigue (CBF) equations for the RPV, Class 1 piping, and the torus locations are conservatively based on such transient definitions. Therefore, the use of design basis transient severity in the fatigue monitoring program CUF equations is another conservatism that more than compensates for potential reactor water environmental effects.

The fatigue management program monitors at least those locations for which the 40-year CUF equaled or exceeded 0.4 in the original design basis analysis. This screening criterion includes sufficient margin, together with the conservative load case definitions, to account for potential reactor water environmental effects for the extended licensed operating period.

Table 4.3.4-1 Design Basis Plant Transient Comparison for the BWR-4 in EPRI Report No. TR-110356 vs. PBAPS

Transient	BWR-4 No. of Cycles	PBAPS Units 2 and 3 No. of Cycles
Boltup	123	66
Design Hydrostatic Test	130	130
Startup	117	161
Turbine Roll & Increase to Rated Power	not specified	161
Daily Reduction to 75% Power	10,000	10,000
Weekly Reduction to 50% Power	2,000	480
Rod Pattern Change (Rod Worth Test)	400	400
Loss of Feedwater Heaters, Turbine Trip with 100% Steam Bypass, Unit 1 = Turbine Trip at 25% Power	10	10
Loss of Feedwater Heaters, Partial Feedwater Heater Bypass	70	70
SCRAM, Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40	80
SCRAM, All Other	140	107
Rated Power Normal Operation	not specified	not specified
Reduction to 0% Power	111	161
Hot Standby	111	2,600
Shutdown/Vessel Flooding	111	161
Unbolt	123	66
Refueling	not specified	not specified
Pre-Operational Blowdown	10	not included
Loss of Feedwater Pumps, Isolation Valves Close	5	10
Reactor Over Pressure with Delayed SCRAM, Feedwater Stays On, Isolation Valves Stay Open	1	1
Single Relief or Safety Valve Blowdown	8	8
Automatic Blowdown	1	1
Improper Start of Cold Recirculation Loop	1	1
Sudden Start of Pump in Cold Recirculation Loop	1	1
Improper Startup with Recirculation Pumps Off & Drain Shut Off	1	not included
Pipe Rupture and Blowdown	1	1
Natural Circulation Startup	3	not included
Loss of AC Power, Natural Circulation Restart	5	not included
Feedwater Temperature Reduction	not included	2,000
Excessive Heatups (160°F/hr)	not included	10
Excessive Cooldowns (160°F/hr)	not included	10
Code Hydrostatic Test	0	3

Table 4.3.4-2 Revised Fatigue Usage Results for BWR (Including Environmental Effects)

Case No.	Location	Projected 60 Year Usage Factor from TR-110356 (with F_{en})	Correction Factor to Account for NUREG/ CR-6583 or NUREG/ CR-5704	Revised 60-Year Usage Factor (with F_{en})[(1)]	Design Basis 40-Year Fatigue Usage[(2)]	Margin[(3)]
1	1 = CRD Penetration	0.034	2.0	0.068	0.875	12.9
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
2	1 = CRD Penetration	0.013	2.0	0.026	0.875	33.7
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
3	1 = CRD Penetration	0.016	2.0	0.032	0.875	27.3
	2 = FW Loop A Safe End	0.009	2.0	0.018	0.471	26.2
	3 = FW Loop A Nozzle Forging	0.001	1.0	0.001	< 0.1	~100
	4 = FW Loop B Safe End	0.009	2.0	0.018	0.471	26.2
	5 = FW Loop B Nozzle Forging	0.001	1.0	0.001	< 0.1	~100

Notes:

1. The "Revised 60-Year Usage Factor" is equal to the "Projected 60-Year Usage Factor from TR-110356" multiplied by the "Correction Factor to Account for NUREG/CR-6583 or NUREG/CR-5704."

2. As documented in the governing design basis fatigue analysis report for the plant evaluated in TR-110356. These values do not include environmental effects.

3. The "Margin" is equal to the "Design Basis Fatigue Usage" divided by the "Revised 60-Year Usage Factor."

Table 4.3.4-3 Locations Evaluated in NUREG/CR-6260 for Older Vintage General Electric Plant (BWR-4) vs. PBAPS, and Projected 60-Year CUFs

NUREG/CR-6260 Location	NUREG/CR-6260 Material	Addressed by PBAPS Program?	PBAPS Material	Projected 60-Year CUF for PBAPS(6)
Reactor Vessel (Lower Head to Shell Transition)	Low Alloy Steel	YES (1)	Low Alloy Steel	U2 = 0.85 U3 = 1.02
Feedwater Nozzle (Bore)	Low Alloy Steel	YES (2)	Carbon Steel	U2 = 0.57 U3 = 0.59
Recirculation System (RHR Return Line Tee)	Stainless Steel	YES	Stainless Steel	U2 = 0.78 U3 = 0.54
Core Spray System (Nozzle)	Low Alloy Steel	YES (3)	---	---
Core Spray System (Safe End)	Stainless Steel	YES	Stainless Steel	U2 = 0.12 U3 = 0.13
Residual Heat Removal Line (Tapered Transition)	Stainless Steel	YES (4)	Stainless Steel	U2 = 0.63 U3 = 0.52
Feedwater Line (RCIC Tee)	Carbon Steel	YES (5)	Carbon Steel	U2 = 0.05 U3 = 0.04

Notes:

1. The support skirt is monitored by the PBAPS fatigue management program. For managing fatigue effects this is an acceptable substitute for the shell region evaluated in NUREG/CR-6260, since it is more limiting for PBAPS.

2. The feedwater nozzle safe end is monitored by the PBAPS fatigue management program. For managing fatigue effects this is an acceptable substitute for the bore region evaluated in NUREG/CR-6260, since it is more limiting for PBAPS.

3. The core spray nozzle safe end is monitored by the PBAPS fatigue management program. For managing fatigue effects this is an acceptable substitute for the nozzle region evaluated in NUREG/CR-6260, since it is more limiting for PBAPS.

4. The limiting locations in the Class 1 portions of the RHR line were selected for PBAPS.

5. The limiting location in the feedwater piping was selected for PBAPS.

6. Before environmental effect factors F_{en}, if applicable. At PBAPS the equivalent location for the first entry is on the vessel exterior, for which no F_{en} applies. See Note 1.

4.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The thermal, radiation, and cyclic aging analyses of plant electrical equipment within the scope of 10 CFR 50.49 have been identified as time-limited aging analyses for PBAPS.

4.4.1 Electrical Equipment Environmental Qualification Analyses

10 CFR 50.49 requires that an environmental qualification (EQ) program be established to demonstrate that certain electrical equipment located in "harsh" plant environments (i.e., those areas of the plant that could be subject to the harsh environment effects of a loss-of-coolant accident, high energy line break, or post loss-of-coolant accident radiation) are qualified to perform their safety function in those harsh environments after the effects of in-service aging. The PBAPS EQ program complies with all applicable regulations and manages equipment thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Environmentally qualified equipment must be refurbished, replaced or have its qualification extended prior to reaching the aging limits established in the aging evaluation.

Analysis

Aging evaluations for environmentally qualified equipment that specify the qualified life of at least 40 years are considered time-limited aging analyses (TLAAs) for license renewal. The following is a list of identified TLAAs for EQ of electrical equipment:

- GE Co. 4kV pump motors and associated cable
- EGS Grayboot connectors
- Raychem insulated splices for class 1E systems
- Bussman Co. and Gould Shawmut fuses and fuse holders
- EGS quick disconnect connectors
- Limitorque motor operated valve actuators
- Namco position switches
- ASCO solenoid valves, trip coils and pressure switches
- UCI splice tape

- Rosemount 1153 Series B transmitters
- GE Co. control station
- Agastat relays
- Static-O-Ring pressure switches
- Cutler Hammer motor control centers
- NDT International accoustical monitors
- Target Rock solenoid valves
- PYCO RTDs and thermocouples
- ITT Barton differential pressure switches
- Atkomatic solenoid valves
- Reliance fan motors and SGTS auxiliaries
- Brown Boveri load centers
- Valcor solenoid valves
- GE Co. radiation elements
- Pyle National plug connectors
- General Atomic radiation monitors
- GE electrical penetrations
- Buchanan terminal blocks
- GE terminal blocks
- Marathon terminal blocks
- Weidmueller terminal blocks
- Amp Inc. terminal lugs
- Scotch Insulating Tape
- GE SIS cable
- Brand Rex cable

- ITT Suprenant 600v control cable
- Okonite 600v power and control cable
- Rockbestos cable
- Foxboro pressure transmitters
- Patel conduit seals
- Jefferson coaxial cable
- Anaconda cable
- HPCI system equipment
- Masoneilan electropneumatic Transducer
- Westinghouse Y panels and associated transformers
- Barksdale pressure switches
- H_2 and O_2 analyzer
- Avco pilot solenoid valves
- Rosemount model no. 710-DU trip units
- Westinghouse manual transfer switch

Disposition: Aging Management, 10CFR54.21(c)(1)(iii).

The aging effects of the EQ equipment identified in this TLAA will be managed during the extended period of operation by the EQ program activities described in Section B.4.1.

4.4.2 GSI-168, Environmental Qualification Of Electrical Components

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI [Reference 4.11]. In the letter, the NRC states: "With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the Statements of Consideration is to provide a technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration also indicated that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time."

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for PBAPS. The PBAPS program (Section B.4.1) evaluates the qualified lifetime of equipment in the EQ program. The existing EQ program requires that equipment qualified for 40 years be reanalyzed prior to entering the period of extended operation. The EQ program requires inclusion of any changes managed by closure of GSI-168. Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a license renewal application at this time.

4.5 LOSS OF PRESTRESS IN CONCRETE CONTAINMENT TENDONS

PBAPS containments have no prestress tendons.

4.6 CONTAINMENT FATIGUE

Subsequent to the original design, elements of the PBAPS containment were re-analyzed in response to discoveries by General Electric and others of unevaluated loads due to design basis events and Safety Relief Valve (SRV) discharge. The load definitions include assumed pressure and temperature cycles resulting from SRV discharge and design basis loss of coolant accident (LOCA) events, and combinations thereof. This re-evaluation was in two parts: generic analyses applicable to each of the several classes of BWR containments, and Mark I Containment Program plant-unique analyses (PUA). The scope of the analyses included the tori (also referred to as "Pressure Suppression Chambers"), the drywell-to-torus vents, SRV discharge piping, other torus-attached piping and its penetrations, and the torus vent bellows.

In the absence of hydrodynamic loads, fatigue is not a concern in containment design except at penetrations or other stress concentration areas. Drywell shell plate was not evaluated for fatigue in the original design and the PUA did not re-evaluate the drywell, drywell penetrations, or process penetration bellows, all of which are attached to the drywell. The licensing and design basis documents do not reflect the existence of any fatigue analysis for the drywell or its penetrations. However, the drywell process bellows were originally specified for a finite number of operating cycles, and the design of these bellows is therefore a TLAA.

4.6.1 Fatigue Analyses of Containment Pressure Boundaries: Analyses of Tori, Torus Vents, and Torus Penetrations

The tori were originally evaluated for a maximum of 800 SRV events.

For the stress cycles associated with SRV and other dynamic events, the PUA calculated a maximum design life fatigue usage factor for the torus of 0.89 and "less than 1" for the drywell-to-torus vents. Current calculations show a usage factor of 0.662 for the most-limiting Unit 2 torus penetration and 0.992 for the most limiting Unit 3 penetration. All of these examples except the Unit 2 limiting penetration would exceed the code allowable CUF of 1.0 if multiplied by 1.5, the ratio of proposed licensed operating years to presently-licensed operating years.

Analysis

Validation: For most torus, vent, and torus penetration locations, the predicted 40-year CUF is less than 0.666, and could be validated for 60 years. However, a CUF of 0.666 provides no analytical or event margin. This validation will, therefore, be applied only to locations with a calculated 40-year CUF of 0.4 or less.

Aging Management: Locations whose 40-year CUF exceeds 0.4 are included in the fatigue management program. Since only the SRV load cases contribute to fatigue during normal operation, normal operation may continue so long as the contribution from SRV lifts has not exceeded 1.0 minus the contribution expected from the postulated worst-case LOCA event. As part of the fatigue management program, the analyses will be revised to show that this usage factor limit is not expected to be exceeded in the period of extended operation. This will be confirmed for the duration of the extended operating period by monitoring fatigue at the high-usage-factor locations in the containment tori, torus vents, and penetrations with the fatigue management program. The fatigue management program CUF modeling equations will be updated as necessary, and transient events will be tracked to ensure that the CUF remains less than 1.0. See Appendix B, Section B.4.2.

This program also permits reanalysis of high-usage-factor locations. Conservatism in original containment PUA may permit the calculated 40-year usage factor to be reduced below the value (0.4) for which fatigue monitoring would be required.

Disposition: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, (iii)

Most locations have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i), as discussed above.

However, portions of this TLAA will require management of the aging effects (10 CFR 54.21(c)(1)(iii)). Refer to Appendix B, Section B.4.2, Fatigue Management Activities.

4.6.2 Fatigue Analysis of SRV Discharge Lines and External Torus-Attached Piping

SRV discharge lines and external torus-attached piping were originally analyzed separately from the tori and torus vents. The analysis included the SRV lines, all piping and branch lines (including small-bore piping less than four inches nominal) attached to the tori, pipe supports, valves, flanges, equipment nozzles, and equipment anchors. The worst-case predicted fatigue usage factors are lower than those found in the tori and vents, and permit a simpler disposition.

PUA Addendum 1 states that 800 SRV actuations were used. PUA Addendum 1 calculated a maximum usage factor for SRV discharge lines and torus-attached piping of 0.202.

Analysis

The fatigue analysis for SRV discharge lines and external torus-attached piping is dispositioned by validation:

$$(U_{max, 40} = 0.202) \times 60/40 = (0.303 = U_{max, 60}) < 1.0,$$

where <1.0 is the design criterion. The maximum usage factor will not exceed the design criterion of 1.0 if multiplied by 1.5, the ratio of proposed licensed operating years (60) to presently-licensed operating years (40).

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The fatigue analysis for SRV discharge lines and external torus-attached piping has been evaluated and remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.6.3 Expansion Joint and Bellows Fatigue Analyses - Drywell to Torus Vent Bellows

The predicted fatigue usage factors for the drywell to torus vent bellows are lower than those calculated for the torus, and permit a simpler disposition. The PUA calculated a "negligible" fatigue usage factor for the drywell-to-torus vent bellows.

Analyses

The predicted fatigue usage factors for the drywell to torus vent bellows are "negligible," and, therefore, will remain negligible compared to the <1.0 design criterion even if increased by a factor of 1.5 for the extended licensed operating period.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The predicted fatigue usage factors for the drywell to torus vent bellows have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.6.4 Expansion Joint and Bellows Fatigue Analyses - Containment Process Penetration Bellows

At PBAPS, the only containment process piping expansion joints subject to significant thermal expansion and contraction are those between the drywell shell penetrations and process piping. These bellows are designed for a stated number of operating and thermal cycles. The design of containment boundary components for a stated number of cycles over the design life is a TLAA.

The PUA does not include any reanalysis of process penetration bellows.

Some PBAPS process expansion joints have been replaced with components designed to later code and specification requirements. Both the original and replacement components were designed for a number of equivalent full-temperature thermal cycles far in excess of their specifications.

Analysis

The bellows originally supplied were designed for “greater than 10,000” cycles; the replacement bellows were designed for "greater than 50,000” cycles. The bellows were designed to the requirements of ASME Code, Section III, that specified a minimum of 200 “startup-shutdown” cycles and a minimum of 1500 “normal operating” cycles. The as-supplied design cycles of the original and replacement bellows far exceed the requirements of the original specifications if extrapolated to 60 years, and similarly far exceed a very conservative estimate of the governing thermal cycles that might be expected to occur in the process piping in 60 years.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The expansion joint and bellows fatigue analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7 OTHER PLANT SPECIFIC TLAAS

4.7.1 Reactor Vessel Main Steam Nozzle Cladding Removal Corrosion Allowance

Original reactor vessel corrosion allowances were conservative values intended to encompass 40 years of operation but without reliance on a particular corrosion rate. However, a subsequent calculation to justify removal of the main steam nozzle cladding used a time-dependent corrosion rate for 40 years, and is thereby a TLAA.

Analysis

Corrosion data for unclad portions of the vessel interior were evaluated, and predict a loss of about 0.030 inches in 60 years. The main steam nozzle clad removal calculation was validated to confirm that the 1/16 inch (.065") corrosion allowance is conservative for 60 years of operation.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The reactor vessel main steam nozzle clad removal corrosion allowances have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.2 Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"

Generic Letter 81-11 and NUREG–0619 were initiated because cracking was observed in BWR feedwater nozzles. The causes included thermal sleeve bypass leakage and consequent rapid thermal cycling on inner surfaces, on-off low-flow feedwater control, and the clad-base metal interface.

The concern was extended to Control Rod Drive Hydraulic System returns as well. At PBAPS, these returns were capped to eliminate the thermal cycling concern at this nozzle.

Analysis

Modifications were installed to reduce or eliminate the causes and an inspection program was invoked to detect and manage any future cracking. Improved triple thermal sleeves with dual piston ring seals were installed, the clad was removed from the nozzle bores and blend radii, and the low-flow controllers were improved. The PBAPS units now use the improved BWR Owners Group (BWROG) inspection and management methods [Ref. 4.12] in lieu of NUREG-0619 methods. The BWROG methods depend on a fracture mechanics analysis and ultrasonic inspection (UT) from the vessel and nozzle exteriors.

The fracture mechanics analysis has cyclic duty inputs and calculates their effects (Criterion 2). The expected cycle accumulation rate over the current 40-year licensed operating period is used to predict crack growth. At PBAPS the analysis predicts that growth between the assumed initial flaw size and an upper analytic limit will require about 60 years. This predicted time for a flaw to reach an upper analytic limit is not used to justify a 60-year operating life. Instead, the BWROG method determines the inspection (or "reinspection") interval as a fraction of this flaw growth time. Therefore, although the fracture mechanics analysis is based on the calculated effects on the number of cycles expected in the current 40-year licensed operating period, it is not "time-limited ... by the current operating term," and thereby fails Criterion 3. It is not a TLAA.

However, GL 81-11 nozzle cracking effect must be acceptable for the period of extended operation if the fatigue design of the reactor vessel is to remain acceptable. Fatigue design of the reactor vessel clearly is a TLAA, and this effect must, therefore, be acceptable if the reactor vessel fatigue TLAA is to be acceptable.

The UFSAR description of this issue includes an evaluation of combined effects of long-term thermal cyclic fatigue (which affects the entire vessel and nozzle wall), and the rapid surface-effect thermal cycles of concern here (which affect only the inner surface of the nozzle of the vessel). This evaluation is a TLAA.

However, these fatigue effects are separable, and the revised safety determination which depends on the improved, NRC-approved BWROG inspection methods no longer depends on this combined fatigue evaluation.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(ii)

The basis for the safety determination documented in the UFSAR will be reexamined and the alternative dispositions consistent with NRC-approved BWROG improved inspection methods will be invoked. The revised methods include aging management activities necessary to manage this effect, which consist of inspections with acceptance criteria and prescribed actions if criteria are not met. These activities are included in the Reactor Pressure Vessel and Internals Inservice Inspection (ISI) Program described in Section B.2.7. In addition, periodic review of the fracture mechanics analysis, in conjunction with the fatigue management program, discussed in Section B.4.2, will ensure that the fracture mechanics evaluation remains bounding and applicable for its intended purpose.

This reexamination of the basis for the safety evaluation will be completed before the current licensed operating period expires.

4.7.3 Fracture Mechanics of ISI-Reportable Indications for Group I Piping: As-forged laminar tear in a Unit 3 Main Steam elbow near weld 1-B-3BC-LDO discovered during preservice UT

A preservice ultrasonic volumetric examination (UT) discovered an imbedded as-forged laminar tear in the Unit 3 main steam elbow material. The indication did not extend to the weld.

Although this piping was not in general subject to an ASME III Class 1 fatigue analysis, a stress and 40-year-life fatigue analysis was performed. The analysis found that the 40-year CUF will not exceed 1.0 (the code acceptance limit) and that primary, secondary, and primary plus secondary stresses are within code. The calculated 40-year CUF is 0.012, which at worst could rise to 0.036 if the lamination extended to the weld joint. This analysis is a TLAA.

Analysis

The fatigue analysis for the as-forged laminar tear in a Unit 3 Main Steam elbow near weld 1-B-3BC-LDO discovered during preservice UT is dispositioned by validation.

Even if the laminar tear were to extend to the weld, the CUF would still not exceed 0.054, which does not exceed the code design criterion (1.0). The condition is, therefore, acceptable for a 60-year operating life.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The fatigue analysis for the as-forged laminar tear in a Unit 3 main steam elbow near weld has been evaluated and remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.8 REFERENCES FOR SECTION 4

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4.4 Memorandum, Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, to William D. Travers, Executive Director for Operations, "Closeout of Generic Safety Issue 190, Fatigue Evaluation of Metal Components for 60 Year Plant Life", USNRC, December 26, 1999

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APPENDIX A UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT

APPENDIX A UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT

Introduction

The aging management activity descriptions presented in this appendix represent commitments for managing aging of the in-scope systems, structures and components during the period of extended operation. These descriptions will be incorporated in the PBAPS, Units 2 and 3 Updated Final Safety Analysis Report (UFSAR) following issuance of the renewed operating license.

As part of the license renewal effort, it must be demonstrated that the aging effects applicable for the components and structures within the scope of license renewal are adequately managed during the period of extended operation.

In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. In some cases, aging management reviews revealed that existing activities should be enhanced to adequately manage applicable aging effects. In a few cases, new activities were developed to provide added assurance that aging effects are adequately managed.

Summary descriptions of TLAAs will be incorporated in the PBAPS, Units 2 and 3 Updated Final Safety Analysis Report (UFSAR) following issuance of the renewed operating license.

Activities Credited for Managing Aging in the Renewal Term

PBAPS has numerous activities that detect and monitor aging effects. This supplement to the UFSAR only describes those activities which PBAPS is crediting for the purposes of complying with the license renewal rule.

Each aging management activity presented in this Appendix is characterized as one of the following:

- **Existing Activity:** A current activity that will continue to be implemented during the period of extended operation.
- **Enhanced Activity:** A current activity that will be modified during the extended period of operation. Enhancements will be implemented as discussed in this Appendix.

- **New Activity:** An activity that does not currently exist, which will manage aging during the extended period of operation. These activities will be implemented as shown in this Appendix.

- **Time Limited Aging Analyses Activity:** An activity that has been credited by a time-limited aging analysis described in Section 4.0

Time Limited Aging Analyses Summaries

Summary descriptions of time-limited aging analyses are provided.

A.1 EXISTING AGING MANAGEMENT ACTIVITIES

A.1.1 Flow Accelerated Corrosion Program

The PBAPS Flow Accelerated Corrosion (FAC) Program activities manage loss of material in pipes and fittings by monitoring the condition of piping susceptible to FAC induced wall thinning. The FAC Program provides for prediction of the amount of wall thinning in carbon steel pipes and fittings through analytical evaluations and periodic examinations of locations most susceptible to FAC induced loss of material. The program includes analyses to determine critical locations, baseline inspections to determine the extent of thinning at these locations, and follow-up inspections to confirm the predictions. The FAC Program provides reasonable assurance that loss of material of carbon steel pipe and fittings is detected and addressed prior to loss of intended function of the piping.

A.1.2 Reactor Coolant System Chemistry

PBAPS reactor coolant system (RCS) chemistry activities manage loss of material and cracking of components exposed to reactor coolant and steam through measures that monitor and control reactor coolant chemistry. These activities include monitoring and controlling of reactor coolant water chemistry to ensure that known detrimental contaminants are maintained within pre-established limits. Reactor coolant is monitored for indications of abnormal chemistry conditions. If such indications are found, then measurements of impurities are conducted to determine the cause, and actions are taken to address the abnormal chemistry condition. Whenever corrective actions are taken to address an abnormal chemistry condition, sampling is utilized to verify the effectiveness of these actions. The RCS chemistry activities provide reasonable assurance that intended functions of components exposed to reactor coolant and steam are not lost due to loss of material or cracking aging effects.

A.1.3 Closed Cooling Water Chemistry

The PBAPS closed cooling water (CCW) chemistry activities manage loss of material, cracking and reduction of heat transfer in components exposed to closed cooling water through measures that monitor and control cooling water chemistry. These activities include periodic monitoring and controlling of chemistry parameters and corrosion inhibitors. If parameter limits are exceeded, corrective actions are taken to restore parameters within the acceptable range. The CCW chemistry activities provide reasonable assurance that the intended functions of components in a CCW environment are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

A.1.4 Condensate Storage Tank Chemistry Activities

PBAPS condensate storage tank (CST) chemistry activities manage loss of material and cracking of components exposed to condensate storage tank water in the RCIC, HPCI, CRD, core spray and condensate storage systems. In addition, CST chemistry activities manage reduction in heat transfer in the HPCI gland seal condenser, and the RCIC and HPCI turbine lubricating oil coolers. CST water is monitored periodically to assure that purity is maintained within pre-established limits. If parameter limits are exceeded, corrective actions are taken to restore parameters within the acceptable range. CST chemistry activities provide reasonable assurance that intended functions of components exposed to CST water are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

A.1.5 Torus Water Chemistry Activities

PBAPS torus water chemistry activities manage loss of material and cracking of components exposed to torus grade water in the RHR, HPCI, RCIC, core spray and main steam systems. In addition, torus water chemistry activities manage cracking of stainless steel component supports submerged in torus grade water, and reduction of heat transfer in RHR heat exchangers. Torus grade water is monitored periodically to assure that purity is maintained within pre-established limits. Torus water chemistry activities provide reasonable assurance that intended functions of components exposed to torus grade water are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

A.1.6 Fuel Pool Chemistry Activities

PBAPS fuel pool chemistry activities manage loss of material for fuel pool gates, fuel storage racks, fuel pool liner, component supports, fuel preparation machines, refueling platform mast, and loss of material and cracking for fuel pool cooling and cleanup system components exposed to fuel pool water. Fuel pool water is monitored periodically to assure that purity is maintained within pre-established limits. Fuel pool chemistry activities provide reasonable assurance that intended functions of components contacted by fuel pool water are not lost due to loss of material or cracking aging effects.

A.1.7 High Pressure Service Water Radioactivity Monitoring Activities

PBAPS high pressure service water radioactivity monitoring activities manage loss of material and cracking in RHR heat exchangers through routine sampling and isotopic analysis of the HPSW system water contained within the RHR heat exchangers to confirm the absence of radioactive isotopes that do not occur naturally. High pressure service water radioactivity monitoring activities provide reasonable assurance that loss of material and cracking are detected and addressed prior to loss of intended function.

A.1.8 Inservice Inspection (ISI) Program

The inservice inspection (ISI) aging management program, as augmented to address the requirements of GL 88-01, consists of those portions of the PBAPS ISI program that are being utilized for managing aging in pressure retaining piping and components in the scope of license renewal. However, the reactor pressure vessel components and internals in the PBAPS ISI program are not included in the ISI aging management program. The PBAPS ISI program complies with the requirements of the 1989 edition of the ASME Section XI code and includes requirements for inspections of ASME Class 1, 2, and 3 pressure retaining components. In addition, it provides for condition monitoring of ASME Class 1,2 and 3 piping and equipment supports and integral support anchors. The ISI program provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.1.9 Primary Containment Inservice Inspection Program

The primary containment ISI program consists of inspections that manage loss of material in the primary containment for Class MC pressure-retaining components, their integral attachments, and Class MC component supports; and loss of sealing for the drywell internal moisture barrier at the juncture of the containment wall and the concrete floor. The program complies with subsection IWE of the ASME Section XI Code 1992 Edition including 1992 Addenda or alternatives, as approved by the NRC. The primary containment ISI program provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.1.10 Primary Containment Leakage Rate Testing Program

The primary containment leakage rate testing program is that portion of the PBAPS primary containment leakage rate testing program that is being credited for license renewal. The primary containment leakage rate testing program provides for aging management of pressure boundary degradation due to loss of material in a wetted gas environment in containment atmosphere control and dilution, RHR, and primary containment isolation systems penetrating primary containment. The primary containment leakage rate testing program also manages change in material properties and cracking of gaskets and O-rings for the primary containment pressure boundary access penetrations. The program complies with the requirements of 10CFR50 Appendix J, Option B. and provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.1.11 Inservice Testing (IST) Program

The Inservice Testing (IST) aging management program is that portion of the PBAPS IST program that is being credited for license renewal. The IST aging management program manages flow blockage of system components from the ECW pump through the ESW and ECW system piping to the ECT. In addition, the program manages reduction of heat transfer of the RHR heat exchangers through flow testing of the torus water path. IST program activities are conducted in accordance with the ASME O&M Code 1990 Edition. IST program activities provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.1.12 Reactor Materials Surveillance Program

The PBAPS Reactor Materials Surveillance (RMS) program manages loss of fracture toughness in the reactor pressure vessel beltline region consistent with the requirements of 10 CFR 50, Appendix H and ASTM E185. The RMS program provides for periodic withdrawal and testing of in-vessel capsules to monitor the effects of neutron embrittlement on the reactor vessel beltline materials. The results of this testing are used to determine plant operating limits. The RMS program contains sufficient dosimetry and materials to monitor irradiation embrittlement during the period of extended operation and provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.1.13 Standby Liquid Control System Surveillance Activities

The standby liquid control system (SBLC) surveillance activities are credited with managing the aging effects of loss of material and cracking in components of the

SBLC that are on the suction side of the SBLC pumps. The surveillance activities monitor the SLCS solution tank liquid level in accordance with a PBAPS procedure. The SBLC components covered by this surveillance include the solution tank, piping, and valves on the suction side of the SBLC pumps. The extent and frequency of this monitoring provides reasonable assurance that loss of material and cracking are detected and addressed prior to loss of intended function.

A.1.14 Crane Inspection Activities

PBAPS crane inspection activities manage loss of material for the structural members, rails, and rail anchorage for the circulating water pump structure gantry crane, and rails and monorails for the cranes and hoists located in a sheltered environment. These annual crane inspections provide reasonable assurance that loss of material is detected and addressed prior to loss of intended function.

A.1.15 Conowingo Hydroelectric Plant (Dam) Aging Management Program

The Conowingo Hydroelectric Plant dam is subject to the FERC 5-year inspection program. This program consists of a visual inspection by a qualified independent consultant approved by FERC, and is in compliance with Title 18 of the Code of Federal Regulations, Conservation of Power and Water Resources, Part 12 (Safety of Water Power Projects and Project Works), Subpart D (Inspection by Independent Consultant). The NRC has found that mandated FERC 5-year inspection programs are acceptable for aging management.

A.1.16 Maintenance Rule Structural Monitoring Program

The maintenance rule structural monitoring program is that portion of the PBAPS structural monitoring program that is being credited for license renewal. The maintenance rule structural monitoring program complies with 10CFR50.65 and utilizes visual inspections in managing aging effects for structures and components within the scope of license renewal that are not covered by other existing inspection programs. The structures and components include emergency cooling tower and reservoir reinforced concrete walls in contact with raw water, structural steel components outside primary containment exposed to the outdoor environment, emergency cooling water outdoor piping support anchors, and penetration seals and expansion joints. Maintenance rule structural monitoring program activities provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

A.2 ENHANCED AGING MANAGEMENT ACTIVITIES

A.2.1 Lubricating and Fuel Oil Quality Testing Activities

Lubricating and fuel oil quality testing activities manage loss of material, cracking and reduction of heat transfer in components that contain or are exposed to lubricating oil or fuel oil. Lubricating and fuel oil quality testing activities provide for sampling and testing of lubricating oil in components in emergency diesel generator (EDG), high pressure coolant injection (HPCI), high pressure service water (HPSW), core spray (CS), and reactor core isolation cooling (RCIC) systems. Lubricating and fuel oil quality testing activities also provide for sampling and testing of fuel oil in the EDG and diesel driven fire pump fuel oil systems. Lubricating and fuel oil quality testing activities include sampling and analysis of lubricating oil and fuel oil for detrimental contaminants. The diesel driven fire pump fuel oil sampling methods will be enhanced to improve water detection capabilities. Analyses of the diesel driven fire pump and EDG fuel oil samples will be enhanced to add testing for microbes in any water detected. The lubricating and fuel oil quality testing activities provide reasonable assurance that aging effects on system components will be detected and addressed prior to loss of intended function of the components. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.2 Boraflex Management Activities

The Boraflex management activities provide for aging management of the spent fuel rack neutron poison material. These activities involve monitoring the condition of Boraflex by routinely sampling fuel pool silica levels and periodically performing in-situ measurement of boron-10 areal density. Existing processes will be enhanced by including the requirement and frequency for in-situ measurement of boron-10 areal density in one or more PBAPS procedures. The Boraflex management activities monitor the condition of Boraflex to provide reasonable assurance that its degradation will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.3 Ventilation System Inspection and Testing Activities

PBAPS ventilation system inspection and testing activities manage aging of filter plenum access door seals and fan flex connections in the standby gas treatment system and the control room ventilation system. These activities also include inspections of fan flex connections for the standby gas treatment system, the control room ventilation system, the battery room and emergency switchgear ventilation system exhaust fans, and the ESW booster pump room ventilation supply fans. These activities will be enhanced by adding inspections of fan flex connections in the diesel generator building ventilation system, the pump structure ventilation system and the battery room and emergency switchgear ventilation system supply fans. PBAPS ventilation system inspection and testing activities provide reasonable assurance that change in material properties will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.4 Emergency Diesel Generator Inspection Activities

The emergency diesel generator (EDG) inspection activities provide for condition monitoring of EDG equipment within the scope of license renewal that are exposed to a gaseous, closed cooling water, lubricating oil or fuel oil environment. Loss of material in the starting air system air receivers is mitigated by daily removal of any accumulation of condensate. Loss of material and cracking in lubricating oil and fuel oil systems is mitigated by periodic inspections performed for underground storage tanks. Visual inspections for change in material properties of flexible hoses in the starting air system and the cooling water system are performed in accordance with a PBAPS procedure in connection with periodic EDG maintenance. This procedure will be enhanced to require inspections of the lubricating oil system and fuel oil system flexible hoses for a change in material properties. The aging management review also determined that the management of loss of material in the EDG exhaust silencer will be enhanced by periodic disassembly, cleaning, and inspection of an automatic drain trap to ensure its functionality in preventing condensation build up. The emergency diesel generator (EDG) inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function of the components. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.5 Outdoor, Buried and Submerged Component Inspection Activities

The outdoor, buried, and submerged component inspection activities provide for loss of material and cracking aging management of external surfaces of components subject to outdoor, buried, and raw water external environments. (Separately, the ISI program provides for monitoring of pressure boundary integrity for outdoor and buried components through pressure tests, flow tests, and inspections.) The submerged components include HPSW, ESW, ECW, and fire protection system pumps. HPSW and ESW system manual discharge pond isolation valves, condensate storage system piping and valves, and the external surfaces of the CSTs and the piping insulation jacketing at the CST are the components exposed to the outdoor environment. The buried components include HPSW, ESW, ECW, fire protection, and EDG fuel oil system piping, fire protection system fire main isolation valves, the EDG fuel oil storage tanks, the SGTS exhaust to the main stack, and the undersides of the CSTs which are in direct contact with compacted fill. The outdoor, buried, and submerged component inspection activities are implemented in accordance with PBAPS maintenance procedures and routine test procedures. The scope of components covered by these activities will be enhanced to include periodic visual inspection of the external surfaces of the CSTs, periodic visual inspection of the ECW pump casing and casing bolts, visual inspection of buried commodities whenever they are uncovered during excavation and enhanced inspections of the refueling water storage tank (as representative of the condition of the CST). These inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.6 Door Inspection Activities

The door inspection activities provide for managing the aging effects for hazard barrier doors that are exposed to the outdoor environment. The aging management review determined that the activities will be enhanced to include additional doors. In addition, the activities will be enhanced to include inspection for loss of material in hazard barrier doors in an outdoor environment. The door inspection activities also provide for managing the aging effects for gaskets associated with water-tight hazard barrier doors in both outdoor and sheltered environments. The inspection activities consist of condition monitoring of the gaskets associated with water-tight hazard barrier doors on a periodic basis. The hazard barrier doors inspection activities are condition monitoring activities that utilize inspections to provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.7 Reactor Pressure Vessel and Internals ISI Program

The BWR Vessels and Internals Project (BWRVIP) guidelines are implemented through the reactor pressure vessel and internals ISI program. The reactor pressure vessel and internals ISI program is that part of the PBAPS ISI program that provides for condition monitoring of the reactor vessel and internals using guidance provided by the BWRVIP and the BWR Owners Group alternate BWR feedwater nozzle inspection requirements. The PBAPS ISI program complies with requirements of an NRC approved Edition of the ASME Section XI Code, or approved alternative, and is implemented through a PBAPS specification. The PBAPS ISI program has been augmented to include various additional requirements, including those from the BWRVIP guidelines and the BWR Owners Group (BWROG) alternative to NUREG-0619 augmented inspection of feedwater nozzles for GL 81-11 thermal cycle cracking. The reactor pressure vessel and internals ISI program will be enhanced to assure that inspections are consistent with the relevant BWRVIP program criteria and NRC safety evaluation reports. The program utilizes early detection, evaluation and corrective actions that provide reasonable assurance that aging effects of reactor vessel components and internals will be detected and addressed prior to loss of intended function. Program enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.8 GL 89-13 Activities

The GL 89-13 activities provide for management of loss of material, cracking, flow blockage, and reduction of heat transfer aging effects in cooling water piping and components that are tested and inspected in accordance with the guidelines of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". The GL 89-13 activities include both condition monitoring and mitigating activities for managing aging effects in HPSW, ESW, and the ECW systems and in other systems' components using raw water as a cooling medium. System and component testing, visual inspections, UT, and biocide treatments are conducted to ensure that aging effects are managed such that system and component intended functions are maintained. Maintenance procedures will be enhanced to require inspection for specific signs of degradation, including corrosion, excessive wear, cracks and Asiatic clams. Also additional piping locations will be added to the UT inspection program. The GL 89-13 activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.9 Fire Protection Activities

The fire protection activities provide for inspections, monitoring, and performance testing of fire protection systems and components to detect aging effects prior to loss of intended function. Degradation of fire protection systems and components due to corrosion buildup, biofouling, and silting are detected by performance testing based on NFPA 24 standards. Periodic and maintenance inspections detect corrosion, fouling, and cracking in system components due to internal and external environment aging effects and detect aging effects in fire barriers. Monitoring of system pressure detects system leakage due to both internal and external aging effects. The scope of fire protection activities will be enhanced. The activities will require additional inspection requirements for deluge valves in the power block sprinkler systems, testing of sprinklers that have been in service for 50 years, inspection of diesel driven fire pump exhaust systems, inspection of diesel driven fire pump fuel oil system flexible hoses, inspection of fire doors for loss of material, and perform a one-time test of a cast iron fire protection component for loss of material due to selective leaching.

The fire protection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.10 HPCI and RCIC Turbine Inspection Activities

The HPCI and RCIC turbine inspection activities provide for aging management of the HPCI and RCIC turbine casings that are exposed to a wetted gas environment. The HPCI turbine inspection activities additionally provide for condition monitoring of components exposed to a lubricating oil environment. The inspection activities perform assessments of components for loss of material aging effects. The activities will be enhanced to inspect the HPCI lubricating oil system flexible hoses for a change in material properties. The HPCI and the RCIC turbine inspection activities are performed periodically in connection with turbine maintenance. The HPCI and RCIC turbine inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.11 Susquehanna Substation Wooden Pole Inspection Activity

The Susquehanna substation wooden pole inspection activity manages the aging effects of loss of material and change in material properties for a wooden takeoff pole at the Susquehanna substation. This pole provides the structural support for the conductors connecting the substation to the submarine cable that is used to transmit the alternate AC power for PBAPS from the Conowingo Hydroelectric Plant in compliance with the requirements of 10 CFR 50.63 for coping with station blackout. The inspection activity will be enhanced to ensure that it is performed every ten years in accordance with corporate specification. The wooden pole inspection activity provides reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.2.12 Heat Exchanger Inspection Activities

The heat exchanger inspection activities provide for periodic component visual inspections and cleaning of heat exchangers and coolers that are outside the scope of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". These activities manage aging effects of loss of material, cracking, and reduction of heat transfer effects for the HPCI gland seal condenser, the HPCI turbine lube oil cooler, and the RCIC turbine lube oil cooler.

These activities will be enhanced to require periodic inspection of the HPCI gland seal condenser tube side internals. These inspections provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function. Activity enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.3 NEW AGING MANAGEMENT ACTIVITIES

A.3.1 Torus Piping Inspection Activities

The PBAPS torus piping inspection activities will provide for identification of loss of material in carbon steel piping located at the water-gas interface in the torus compartment of the primary containment by monitoring the condition of a representative sample of the piping at a susceptible location. These activities will include a one-time inspection of the wall thickness of selected torus piping. The scope and frequency of subsequent examinations will be based on the results of the initial inspection sample. Torus piping inspection activities provide reasonable assurance that loss of material will be detected and addressed prior to loss of intended function. Torus piping inspection activities will be implemented prior to the end of the initial operating license term for PBAPS.

A.3.2 FSSD Cable Inspection Activity

PBAPS fire safe shutdown (FSSD) cable inspection activities will manage change in material properties of the PVC-insulated FSSD cables located in the drywell by monitoring the condition of a representative sample of the cables. FSSD cable inspection activities will identify anomalies in the PVC insulation surface that are precursor indications of loss of material properties of the PVC insulation. These activities provide reasonable assurance that loss of material properties of the PVC-insulated cables will be detected and addressed prior to loss of their intended function. The FSSD cable inspection activities will be implemented prior to the end of the initial operating license term for PBAPS.

A.4 TIME-LIMITED AGING ANALYSES ACTIVITIES

A.4.1 Environmental Qualification Activities

PBAPS environmental qualification (EQ) program ensures maintenance of qualified life for the electrical equipment important to safety within the scope of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." An aging limit (qualified life) is established for equipment within the scope of the EQ program and an appropriate action such as replacement or refurbishment is taken prior to or at the end of the equipment qualified life so that the aging limit is not exceeded. The PBAPS EQ program activities establish, demonstrate and document the level of qualification, qualified configuration, maintenance, surveillance and replacement requirements necessary to apply the qualification conclusions and the equipment qualified life.

A.4.2 Fatigue Management Activities

The fatigue management program counts fatigue stress cycles and tracks fatigue usage factors. The program will be enhanced to broaden its scope and update implementation methods, and will consist of analytical methods to determine stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors (CUFs). The program will calculate and track CUFs for bounding locations in the reactor pressure vessel (RPV), RPV internals, Group I piping, and containment torus. The fatigue management program enhancements will be implemented prior to the end of the initial operating license term for PBAPS.

A.5 TIME-LIMITED AGING ANALYSES SUMMARIES

In the descriptions of this section, Groups I, II, and III are the PBAPS pressure boundary safety groups described in UFSAR Appendix A, Section A.2

A.5.1 Reactor Vessel Neutron Embrittlement

The PBAPS Units 2 and 3 reactor vessels are described in UFSAR Sections 3.3 and 4.2. Reactor vessel materials are subject to embrittlement, primarily due to exposure to neutron radiation. Reactor vessel neutron embrittlement is a TLAA.

A.5.1.1 Reactor Vessel End-of-Life Fluence, USE, • RT_{NDT}, and P-T Limit Curves, Reflood Thermal Shock Analysis

The vessel end-of-life fluence, USE, • RT_{NDT}, and P-T limit curves will be recalculated for a 60-year licensed operating life. These calculations will be completed and acceptable values will be confirmed prior to the end of the initial operating license term for PBAPS.

The analyses of reactor vessel fluence, USE, • RT_{NDT}, and the operating limit P-T curves will be projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

The reflood thermal shock analysis was examined and remains valid and bounding for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.1.2 Reactor Vessel Circumferential Weld Examination Relief

Relief has been granted from the requirements for inspection of RPV circumferential welds for the remainder of the current 40-year licensed operating period. The justification for relief is consistent with Boiling Water Reactor Vessel and Internals Program BWRVIP-05 guidelines. Application for an extension of this relief for the 60-year period of extended operation will be submitted prior to the end of the initial operating license term.

The procedures and training that will be used to limit the frequency of cold over-pressure events to the number specified in the SER for the RPV circumferential weld relief request extension, during the license renewal term, are the same as those approved for use in the current period.

The analyses associated with reactor vessel circumferential weld examination relief will be projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.5.1.3 Reactor Vessel Axial Weld Failure Probability

BWRVIP-05 estimated the 40-year end-of-life failure probability of a limiting reactor vessel axial weld, showed that it was orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds, as noted above.

The re-evaluation of the axial weld failure probability for 60 years depends on vessel • RT_{NDT} calculations. Evaluation of plant-specific values will be completed prior to the end of the initial operating license term.

The analysis of reactor vessel limiting axial weld failure probability will be completed to confirm that the assumed limit on the axial weld failure probability remains valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21 (c)(1)(i).

A.5.2 Metal Fatigue

The thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs for PBAPS. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the PBAPS UFSAR.

A.5.2.1 Reactor Vessel Fatigue

Unit 2 and Unit 3 reactor vessel fatigue analyses depend on cycle count assumptions that assume a 40-year operating period. The effects of fatigue in the reactor vessel will be managed for the period of extended operation by the fatigue management program for cycle counting and fatigue usage factor tracking as described in Section A.4.2.

This aging management program will ensure that fatigue effects in vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.5.2.2 Reactor Vessel Internals Fatigue and Embrittlement

A.5.2.2.1 Reactor Vessel Internals Fatigue Analyses

Original Core Shroud, Shroud Support and Jet Pump Assembly Analysis: Fatigue in these components is from both system cycles and vibrations. Vibration effects in the core shroud, shroud supports, and jet pump assemblies were evaluated in a standard-plant analysis applicable to PBAPS. The design analysis was based on the cyclic stress criteria of ASME Section III.

Core shroud and jet pump assemblies: The evaluations were extended to 60 years with a usage factor less than the allowable of 1.0.

The fatigue analyses of the core shroud and jet pump assemblies have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Revised Core Shroud supports analysis: Fatigue analyses of the shroud supports were reevaluated for effects of increased recirculation pump starts with the loop outside thermal limits. The reevaluation used very conservative estimates of the expected number of these starts in 40 years, and conservative methods of estimating the fatigue usage factor. The evaluation therefore estimated an upper bound on the 40-year usage factor which is less than the code limit, but which cannot readily be validated for 60 years.

The usage factor at the critical shroud support locations will, therefore, be managed by the cycle counting and fatigue usage factor tracking used by the fatigue management program described in Section A.4.2, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). The fatigue management program will ensure that the fatigue effects in the shroud supports will be adequately managed for the period of extended operation.

Unit 3 Core Spray Pipe to Tee Box Weld Crack Repair Bracket: Repair brackets were installed in Unit 3 in 1985. The repair brackets and attachment welds to the piping were analyzed in accordance with the requirements of ASME Section III Article NB-3200.

The fatigue analysis of the Unit 3 Core Spray pipe to tee box weld crack repair bracket has been evaluated and remains valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.2.2.2 Reactor Vessel Internals Embrittlement Analyses

The reactor vessel internals embrittlement analyses affect the core shroud and top guides.

Shroud: An evaluation was performed and found that the expected 60-year fluence on the shroud is below the damage threshold.

Top Guide: An evaluation was performed and, although the expected 60-year fluence is above the damage threshold, critical locations in the top guide at PBAPS have low tensile stresses and thus are exempt from inspection under the BWRVIP-26 guidelines. That is, at these low stresses, a fracture is not a concern, and embrittlement is therefore not a threat to the intended function.

The existing analyses of the effects of embrittlement in core shrouds and top guides have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.2.2.3 Effect of Fatigue and Embrittlement on End-of-Life Reflood Thermal Shock Analysis

Radiation embrittlement and fatigue usage may affect the ability of certain internals, particularly the core shroud support plate, to withstand an end-of-life reflood thermal shock following a recirculation line break.

The existing analyses of the effects of fatigue and embrittlement on end-of-life reflood thermal shock analyses of reactor internals have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.2.3 Piping and Component Fatigue and Thermal Cycles

A.5.2.3.1 Fatigue Analyses of Group I Primary System Piping

All Group I piping was originally designed in accordance with USAS B31.1, 1967 Edition. The recirculation and RHR piping in Group 1 was replaced under the GL 88-01 IGSCC Correction Program. The replacement piping was analyzed to ASME Section III Class 1 rules. ASME Section III requires an explicit fatigue analysis for Class 1 components. USAS B.31.1 does not require an explicit fatigue analysis. However, a simplified fatigue analysis was developed to estimate the current usage factors from operating data for the fatigue management cycle counting and fatigue usage factor tracking program.

The effects of fatigue in Group I primary system piping will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as part of the Fatigue Management Activities, discussed in Section A.4.2. This aging management activity will ensure that fatigue effects in pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.5.2.3.2 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction in Group II and III Piping and Components

<u>Group II and III Piping</u>: Thermal cycle count is a consideration in all the codes associated with the design of Group II and III piping (e.g., USAS or ANSI B31.1). The applicable piping codes require the use of a stress range reduction factor in the evaluation of calculated stresses due to thermal expansion. The reduction factor is based on the anticipated number of equivalent full temperature cycles over the total number of years the plant is expected to be in operation.

The number of thermal cycles assumed for design of Group II and III piping has been evaluated and the existing stress range reduction factor remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.5.2.3.3 Design of the RHR System for a Finite Number of Cycles

Group I RHR piping inside the drywell that was analyzed to ASME III Class 1 rules is discussed in Section A.5.2.3.1.

Group II RHR piping and some Group I RHR piping is designed to a USAS or ANSI B31.1 allowable secondary stress range determined by a finite number of equivalent full-temperature thermal cycles, without a detailed fatigue analysis.

The ANSI B31.1 threshold number for applying a reduction factor for RHR valves is 7,000 equivalent full-temperature cycles. The total number of cycles assumed for the original 40-year plant life is, conservatively, less than 1,000. For the period of extended operation, the number of thermal cycles for piping analyses would be proportionately increased to 1,500, which is still significantly less than the 7,000 cycle threshold. The code stress range reduction factor therefore remains at 1.0 and is not affected by extending the operating period to 60 years

The RHR valve and piping thermal cycle analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.5.2.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Generic Safety Issue (GSI) 190 was identified by the NRC because of concerns about potential effects of reactor water environments on component fatigue life during the period of extended operation. The GSI was closed in December 1999 because the NRC concluded that environmental effects have a negligible effect on core damage frequency; however, license renewal applicants need to address the effects of the coolant environment on component fatigue life.

Exelon has reviewed the potential effects of reactor coolant on component fatigue life based on EPRI generic studies and on laboratory fatigue data in simulated reactor water environments that were generated by Argonne National Laboratory. This review concluded that environmental effects on fatigue life of components are encompassed by the conservatism of design basis transient definitions. However, to ensure that all CUFs remain below the code design limit throughout the extended operation period, Exelon will monitor selected locations in the fatigue management program in accordance with 10 CFR 54.21(c)(1)(iii).

A.5.3 Environmental Qualification Of Electrical Equipment

Electrical equipment included in the PBAPS Environmental Qualification Program which has a specified qualified life of at least 40 years involves time-limited aging analyses for license renewal. The aging effects of this equipment will be managed in the Environmental Qualification Program discussed in Section A.4.1, Environmental Qualification Activities, in accordance with the requirements of 10CFR54.21(c)(1)(iii).

A.5.4 Containment Fatigue

Subsequent to the original design, elements of the PBAPS containment were re-analyzed in response to discoveries of unevaluated loads due to design basis events and Safety Relief Valve (SRV) discharge. This re-evaluation was in two parts: Generic analyses applicable to each of the several classes of BWR containments, and plant-unique analyses (PUA). The scope of the analyses included the tori, the drywell-to-torus vents (torus vents), SRV discharge piping, other torus-attached piping and its penetrations, and the torus vent bellows.

A.5.4.1 Fatigue Analyses of Containment Pressure Boundaries: Analysis of Tori, Torus Vents, and Torus Penetrations

For low usage factor locations (40-year CUF < 0.4) the PBAPS new loads analyses of tori, torus vents, and torus penetrations have been evaluated and determined to remain valid for the extended period of operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

For higher usage factor locations in the analyses of tori, torus vents, and torus penetrations (40-year CUF ≥ 0.4) the effects of fatigue will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program described in Section A.4.2.

The Fatigue Management Activities will ensure that fatigue effects in vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.5.4.2 Fatigue Analysis of SRV Discharge Lines and External Torus-Attached Piping

SRV discharge lines and external torus-attached piping were analyzed separately from the tori and torus vents. The PBAPS analysis included the SRV lines, all piping and branch lines attached to the tori, pipe supports, valves, flanges, equipment nozzles, and equipment anchors.

The fatigue analyses of SRV discharge lines and external torus-attached piping have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.5.4.3 Expansion Joint and Bellows Fatigue Analyses - Drywell to Torus Vent Bellows

The predicted fatigue usage factors for the drywell torus vent bellows for the period of extended operation are negligible.

The PBAPS new loads fatigue analyses of the drywell-to-torus vent bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.5.4.4 Expansion Joint and Bellows Fatigue Analyses - Containment Process Penetration Bellows

The only containment process piping expansion joints subject to significant thermal expansion and contraction are those between the drywell shell penetrations and process piping. These are designed for a stated number of operating and thermal cycles.

The thermal cycle designs of PBAPS containment process penetration bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.5.5 Metal Corrosion Allowances

A.5.5.1 Reactor Vessel Corrosion Main Steam Nozzle Cladding Removal Allowance

The original vessel corrosion allowances were conservative values intended to encompass 40 years of operation but without reliance on a particular corrosion rate. The original allowances: therefore, do not depend on the 40 year design life. However, a subsequent calculation to justify removal of the main steam nozzle cladding used a time-dependent corrosion rate and is thereby a TLAA.

The reactor vessel corrosion allowances have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.6 Inservice Flaw Growth Analyses That Demonstrate Structural Integrity For 40 Years

Two flaw dispositions were identified that include TLAAs.

A.5.6.1 Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"

The PBAPS control rod drive hydraulic system return nozzles were capped. Therefore, cracking of the feedwater nozzles is the only aging effect that applies to PBAPS. Cracking of the feedwater nozzles was addressed by installing triple thermal sleeves with double-piston-ring seals, by removing the vessel clad from the nozzles, by installing improved low-flow feedwater controllers, and by adopting and maintaining an augmented inspection program to detect incipient problems currently using the NRC-approved BWR Owner's Group (BWROG) inspection and management methods.

The fracture mechanics evaluations which support the validity of the current examination methods are not TLAAs. However, the effects of the cracking phenomena must be managed to ensure the continued validity of the assumptions of fatigue analyses for the reactor vessel, which are TLAAs.

The aging effect is adequately addressed by the modifications already installed and by the inspection program already in place. No enhancements are required.

Any remaining or recurring effects of rapid-thermal-cycle damage at feedwater nozzle inner blend radii will be managed for the period of extended operation by the reactor vessel and internals inspection program described in Section A.2.7.

This aging management activity includes specific requirements for these nozzles and for this issue.

This program will ensure that any effects will be adequately detected, managed, and controlled, within the limits of the supporting fracture mechanics analyses, for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.5.6.2 Fracture Mechanics of As-forged Laminar Tear in a Unit 3 Main Steam Elbow

Preservice inspection discovered an as-forged laminar tear in a Unit 3 main steam elbow near weld 1-B-3BC-LDO. The original disposition included a fatigue analysis.

The fatigue analysis for the as-forged laminar tear has been evaluated and remains valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.5.7 References for Section A.5

(1) BWRVIP-05, EPRI Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." For the Boiling Water Reactor Owners Group (Proprietary), September 28, 1995, with supplementing letters of June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998.

(2) BWRVIP-26, EPRI Report TR-107285, "BWR Vessel and Internals Project: BWR Top Guide Inspection and Flaw Evaluation Guidelines", December 1996.

APPENDIX B AGING MANAGEMENT ACTIVITIES

APPENDIX B AGING MANAGEMENT ACTIVITIES

Introduction

The aging management activity descriptions are provided in this Appendix for each activity credited for managing aging effects based upon the aging management review results provided in Sections 3.1 through 3.6.

In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. In some cases, aging management reviews revealed that existing activities should be enhanced to adequately manage aging. In a few cases, new activities were developed to provide reasonable assurance that aging effects are adequately managed.

Each aging management activity presented in this Appendix is characterized as one of the following:

Existing Activity: A current activity that will continue to be implemented during the extended period of operation.

Enhanced Activity: A current activity that will be modified to manage aging during the extended period of operation.

New Activity: An activity that does not currently exist, which will manage aging during the extended period of operation.

Time Limited Aging Analyses Activity: An activity that has been credited by a time-limited aging analysis described in Section 4.0

B.1 Existing Aging Management Activities

B.1.1 Flow Accelerated Corrosion Program

ACTIVITY DESCRIPTION

The PBAPS flow accelerated corrosion (FAC) program activities predict, detect, and monitor wall thinning in piping and fittings due to flow accelerated corrosion. The FAC program is based on the EPRI guidelines in NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program." A PBAPS specification ensures that the FAC program will be implemented as required by NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning".

The FAC program provides for prediction of the amount of wall thinning in pipes and fittings through analytical evaluations and periodic examinations of locations most susceptible to FAC induced loss of material. Specifically, the program includes analyses to determine critical locations, baseline inspections to determine the extent of thinning at these locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

The susceptible piping systems are divided into two categories. Category 1 consists of piping systems, or portions of systems, that are susceptible to FAC and have a completed FAC Wear Rate Analysis in EPRI's CHECWORKS computer code. Category 2 consists of piping systems, or portions of systems, that are susceptible to FAC but do not have a completed FAC Wear Rate Analysis in the CHECWORKS computer code.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The FAC program provides for prediction, inspection, and monitoring of piping and fittings for a loss of material aging effect due to flow accelerated corrosion so that timely and appropriate action may be taken to minimize the probability of experiencing a FAC-induced consequential leak or rupture.

The FAC program elements are based on the recommendations identified in NSAC-202L-R2, which requires controls to assure the structural integrity of carbon steel lines containing high-energy fluids (two phase as well as single phase). The PBAPS FAC program manages loss of material in carbon steel piping and fittings. The PBAPS feedwater system is classified as Category 1.

The main steam system and the HPCI and RCIC steam line drains are classified as Category 2.

(2) Preventive Actions: The FAC program is a condition monitoring program that identifies loss of material aging effects prior to loss of intended function. No preventive or mitigative attributes are associated with the FAC program.

(3) Parameters Monitored/Inspected: The FAC program provides for inspection and monitoring of the wall thickness of piping and fittings susceptible to FAC-induced loss of material. Piping and fitting wall thickness reduction could challenge the maintenance of the pressure boundary intended function.

(4) Detection of Aging Effects: The FAC program provides for periodic ultrasonic inspections of components susceptible to FAC to validate analytical evaluations. The extent and schedule of inspections ensure that loss of material (wall thinning) of piping and fittings is detected prior to loss of intended function of the piping.

(5) Monitoring and Trending: The FAC program provides for analytical evaluations using parameters such as pipe material, geometry, hydrodynamic conditions, temperature, pH and oxygen content to predict wall thickness reduction due to FAC. Inspections of the piping verify the evaluations. The schedule of the next inspection is based on the remaining life determined after each inspection. If degradation is detected such that the wall thickness is less than the minimum predicted thickness, additional examinations are performed in similar and adjacent areas to bound the thinning. The FAC program provides reasonable assurance that structural integrity will be maintained between inspections.

(6) Acceptance Criteria: Inspection results are used to calculate the number of refueling or operating cycles remaining before the component reaches design code minimum allowable wall thickness. If calculations indicate that an area will reach design code minimum allowable thickness before the next scheduled outage the component is repaired, replaced or reevaluated.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Wall thinning problems in single-phase systems have occurred throughout the industry in feedwater and condensate systems, and in two-phase piping in extraction steam lines and moisture separator reheater and feedwater heater drains. The PBAPS HPCI and RCIC steam drain lines have experienced wall thinning due to FAC and this piping has been replaced. The FAC program was originally outlined in NUREG-1344 and implemented through GL 89-08. The FAC program has evolved through industry experience and is now described in NSAC-202L. Application of the FAC program has resulted in the replacement of piping identified as being subject to FAC before loss of material has challenged pressure boundary integrity. The FAC program has provided an effective means of ensuring the structural integrity of high-energy carbon steel systems. The NRC has audited industry programs based on the EPRI methodology at several plants and has determined that these activities can provide a good prediction of the onset of FAC so that timely corrective actions can be undertaken.

The PBAPS FAC program is updated to reflect the latest industry and plant experience. Modifications have been implemented at PBAPS due to discovery of pipe wall thinning or leakage. No failures, other than HPCI and RCIC steam drain lines, have occurred in any components at PBAPS within the license renewal boundary.

SUMMARY

The PBAPS FAC program activities predict, detect, and monitor wall thinning in piping due to flow accelerated corrosion. The FAC program is based on the EPRI guidelines in NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program." A PBAPS specification ensures that the FAC program is implemented as required by NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning".

Based on the use of industry guidelines, NRC requirements, and PBAPS operating experience, there is reasonable assurance that the PBAPS FAC program will continue to adequately manage the aging effects due to flow accelerated corrosion in carbon steel piping systems containing high energy fluids to maintain intended functions consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI NSAC-202L-R2, April, 1999, "Recommendations for an Effective Flow-Accelerated Corrosion Program"

(2) NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning"

(3) EPRI "CHECWORKS", "Computer Program, Database, and User's Manual"

(4) NUREG - 1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants"

B.1.2 Reactor Coolant System Chemistry

ACTIVITY DESCRIPTION

PBAPS reactor coolant system (RCS) chemistry activities consist of preventive measures that are used to manage loss of material and cracking in components exposed to reactor water and steam. RCS chemistry activities provide for monitoring and controlling of RCS water chemistry using PBAPS procedures and processes based on EPRI TR-103515, "BWR Water Chemistry Guidelines."

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: RCS chemistry activities manage loss of material and cracking in reactor, RPV Instrumentation, reactor recirculation, standby liquid control, feedwater, HPCI, RCIC, core spray, RHR, PCIS (RWCU), and main steam systems by monitoring and controlling detrimental contaminants.

(2) Preventive Actions: RCS chemistry activities include periodic monitoring and controlling of RCS water chemistry to ensure that known detrimental contaminants are maintained within pre-established limits, providing reasonable assurance that the aging effects of loss of material or cracking are managed.

(3) Parameters Monitored/Inspected: Conductivity of the reactor coolant is continuously monitored to provide indication of abnormal conditions and the presence of impurities. If conductivity becomes abnormal then measurements are conducted of impurities identified in EPRI TR-103515, such as chlorides and sulfates, to determine the cause.

(4) Detection of Aging Effects: RCS chemistry activities mitigate the onset and propagation of loss of material and cracking. No credit is taken for detection of aging effects.

(5) Monitoring and Trending: RCS water chemistry is monitored continuously to ensure purity is maintained within acceptable limits based on EPRI guidelines. Samples are taken and analyzed, and data are trended. The frequency of sampling is based on EPRI TR-103515 and varies with plant operating conditions. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling is utilized to verify the effectiveness of these actions.

(6) Acceptance Criteria: Maximum levels for various contaminants, including chlorides and sulfates, are maintained below system specific limits based on EPRI TR-103515.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: As chemistry control guidelines were evolving in the industry, PBAPS experience with reactor coolant system chemistry was similar to that of the industry. Cracking attributed to IGSCC was found in stainless steel recirculation and RHR system piping and loss of material was found in the HPCI and RCIC carbon steel steam line drains. Portions of the 304 stainless steel recirculation system, RWCU, and RHR piping were replaced with more IGSCC resistant, low carbon, 316 stainless steel. The HPCI and RCIC steam drain lines were also replaced.

The RCS water chemistry is maintained based on the recommendations of EPRI TR-103515 that have been developed based on industry experience. These recommendations have been shown to be effective and are adjusted as new information becomes available. Since the pipe replacement and improvements to chemistry activities, the overall effectiveness of RCS chemistry activities is supported by the excellent operating experience of reactor coolant and main steam systems at PBAPS.

SUMMARY

PBAPS reactor coolant system (RCS) chemistry activities are preventive aging management activities that assure potentially detrimental concentrations of impurities are not present in the reactor coolant. These measures manage loss of material and cracking in components exposed to reactor water and steam.

Based on the use of industry guidelines and industry and PBAPS operating experience, there is reasonable assurance that PBAPS RCS chemistry activities will continue to adequately manage the aging effects of loss of material and cracking associated with components exposed to the reactor coolant and steam environments so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI TR-103515, “BWR Water Chemistry Guidelines”, 2000 Revision

B.1.3 Closed Cooling Water Chemistry

ACTIVITY DESCRIPTION

PBAPS closed cooling water (CCW) chemistry activities consist of preventive measures that are used to manage loss of material, cracking, and reduction of heat transfer in components exposed to a closed cooling water environment. CCW chemistry activities provide for monitoring and controlling of CCW chemistry using PBAPS procedures and processes based on EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines."

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: CCW chemistry activities manage loss of material and cracking in systems and portions of systems within the emergency diesel generator and primary containment isolation systems subject to a closed cooling water environment by monitoring and controlling detrimental contaminants, and maintaining corrosion inhibitors to minimize corrosion. CCW chemistry activities also manage reduction of heat transfer in the EDG air coolant coolers and the EDG jacket coolant coolers.

(2) Preventive Actions: CCW chemistry activities include periodic monitoring and controlling of corrosion inhibitor concentrations within specified limits of EPRI TR-107396 to minimize corrosion and protect metal surfaces. Maintaining the system corrosion inhibitor concentration within the pre-established limits provides reasonable assurance that the aging effects of loss of material, cracking, and reduction of heat transfer are managed.

(3) Parameters Monitored/Inspected: The CCW chemistry monitoring and controlling activities minimize the aggressiveness of this environment. CCW chemistry is maintained per the recommendations of EPRI TR-107396. Nitrite, pH and methylbenzyl triazole (TTA) levels are monitored as chemistry control parameters. Chlorides, sulfates, nitrate, turbidity and metals are monitored on a regular basis as diagnostic parameters to provide indication of abnormal conditions. If parameter limits are exceeded, the chemistry control procedures require that corrective action be taken to restore parameters to within the acceptable range. Maintenance of corrosion inhibitor levels within EPRI TR-107396 guidelines mitigates loss of material, cracking, and reduction of heat transfer.

(4) Detection of Aging Effects: CCW chemistry activities mitigate the onset and propagation of loss of material, cracking, and reduction of heat transfer. No credit is taken for detection of aging effects.

(5) Monitoring and Trending: CCW chemistry is monitored to ensure corrosion inhibitors are being maintained within acceptable limits in accordance with EPRI guidelines. Samples are taken and analyzed, and data are trended. The frequency of sampling is based on EPRI TR-107396.

(6) Acceptance Criteria: Levels for concentration of nitrite and TTA are maintained within the limits specified in the EPRI TR-107396. Parameters maintained in CCW systems include pH (8.5 – 10.5), Nitrite (500 – 1100 ppm), and TTA (5 – 30 ppm).

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes.
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The industry operating experience demonstrates that the use of corrosion inhibitors in closed cooling water systems that are monitored and maintained by CCW chemistry activities is effective in mitigating loss of material, cracking, and reduction of heat transfer. No age related failures have occurred in the components within the scope of license renewal that are covered by PBAPS CCW chemistry activities.

SUMMARY

PBAPS CCW chemistry activities manage loss of material and cracking in components exposed to a closed cooling water environment. CCW chemistry activities provide for monitoring and controlling of CCW chemistry using PBAPS procedures and processes based on EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines".

Based on the use of industry guidelines and PBAPS operating experience, there is reasonable assurance that the PBAPS CCW chemistry activities will continue to adequately manage the aging effects of loss of material, cracking, and reduction of heat transfer in components exposed to a CCW environment so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines", October 1997

B.1.4 Condensate Storage Tank Chemistry Activities

ACTIVITY DESCRIPTION

Condensate storage tank (CST) chemistry activities consist of preventive measures that are used to manage aging in components of the RCIC, HPCI, CRD, core spray and condensate storage systems exposed to condensate storage tank water. CST chemistry activities provide for monitoring and controlling of CST water chemistry using PBAPS procedures and processes based on EPRI TR-103515, "BWR Water Chemistry Guidelines".

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: CST chemistry activities manage loss of material and cracking in the RCIC, HPCI, CRD, core spray and condensate storage system components exposed to condensate storage tank water. In addition, CST chemistry activities manage reduction of heat transfer in the HPCI gland seal condenser and the RCIC and HPCI turbine lubricating oil coolers. The aging effects are managed by monitoring and controlling detrimental contaminants in CST water.

(2) Preventive Actions: CST chemistry activities include periodic monitoring and controlling of CST water, to ensure known detrimental contaminants are maintained within pre-established limits, providing reasonable assurance that the aging effects of loss of material, cracking, and reduction of heat transfer are managed.

(3) Parameters Monitored/Inspected: Conductivity is maintained based on the EPRI guidance. Impurities identified in EPRI TR-103515, such as, chlorides and sulfates are monitored and controlled.

(4) Detection of Aging Effects: CST chemistry activities mitigate the onset and propagation of loss of material, cracking, and reduction of heat transfer aging effects. No credit is taken for detection of aging effects.

(5) Monitoring and Trending: CST water is monitored weekly to assure that purity is maintained within acceptable limits based on EPRI guidelines. Samples are taken and analyzed, and data are trended. The frequency of sampling is based on EPRI TR-103515.

(6) Acceptance Criteria: Maximum levels for various contaminants are maintained below system specific limits based on EPRI TR-103515. The acceptance criteria include the following parameter limits: conductivity $\leq$1 µS/cm, chlorides $\leq$10 ppb, and sulfates $\leq$10 ppb.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The overall effectiveness of CST chemistry activities is supported by the operating experience for systems, which are influenced by CST water chemistry. There has been no loss of component intended function in systems for which CST chemistry activities have been employed. CST water chemistry is maintained based on the recommendations of EPRI TR-103515. The EPRI recommendations have been developed based on industry experience, have been shown to be effective, and are adjusted as new information becomes available.

SUMMARY

CST chemistry activities use preventive measures to manage aging effects in components of the RCIC, HPCI, CRD, core spray and condensate storage systems exposed to condensate storage tank water. CST chemistry activities monitor and control CST water chemistry parameters using PBAPS procedures and processes based on EPRI TR-103515, "BWR Water Chemistry Guidelines".

Based on the use of industry guidelines and industry and PBAPS operating experience, there is reasonable assurance that the CST chemistry activities will continue to adequately manage the aging effects associated with systems and components exposed to condensate storage tank water so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI TR-103515, "BWR Water Chemistry Guidelines", 2000 Revision.

B.1.5 Torus Water Chemistry Activities

ACTIVITY DESCRIPTION

Torus water chemistry activities consist of preventive measures that are used to manage aging effects in components of the RHR, HPCI, RCIC, core spray and main steam systems exposed to torus grade water. Torus water chemistry activities provide for monitoring and controlling of torus water chemistry using PBAPS procedures and processes based on EPRI TR-103515, "BWR Water Chemistry Guidelines."

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: Torus water chemistry activities manage loss of material and cracking in RHR, HPCI, RCIC, core spray and main steam system components exposed to a torus grade water environment. In addition, torus water chemistry activities manage cracking of stainless steel component supports submerged in torus grade water, and reduction of heat transfer in RHR heat exchangers. The aging effects are managed by monitoring and controlling detrimental contaminants in torus grade water.

(2) Preventive Actions: Torus water chemistry activities include periodic monitoring and controlling of torus grade water chemistry to ensure that known detrimental contaminants are maintained within pre-established limits, providing reasonable assurance that the aging effects of loss of material, cracking, and reduction of heat transfer are managed.

(3) Parameters Monitored/Inspected: Conductivity is maintained based on the EPRI guidance. Impurities identified in EPRI TR-103515, such as chlorides and sulfates, are monitored and controlled.

(4) Detection of Aging Effects: Torus water chemistry activities mitigate the onset and propagation of loss of material, cracking, and reduction of heat transfer aging effects. No credit is taken for detection of aging effects.

(5) Monitoring and Trending: Torus grade water is monitored periodically to assure that purity is maintained within acceptable limits based on EPRI guidelines. Samples are taken and analyzed, and data are trended. The frequency of sampling is based on EPRI TR-103515.

(6) Acceptance Criteria: Maximum levels for various contaminants are maintained below system specific limits as specified in EPRI TR-103515. Parameters maintained include conductivity (< 5 µmho/cm), chloride (≤ 200 ppb), sulfate (≤ 200 ppb), total organic carbon (≤ 1000 ppb) and turbidity (2-25 ntu).

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Torus water chemistry is maintained per the recommendations of EPRI TR-103515 that have been developed based on industry experience and shown to be effective. These limits are adjusted as new information becomes available. Components containing a torus water environment within the scope of license renewal have not experienced any age related pressure boundary failure at PBAPS. There has been no age-related loss of function of a submerged stainless steel component support or RHR heat exchanger due to chemistry related degradation. Torus inspections conducted in 1997 revealed a decrease in the rate of corrosion of the torus structure in part due to improved torus water chemistry.

SUMMARY

Torus water chemistry activities consist of preventive measures that manage aging effects in components of the RHR, HPCI, RCIC, core spray and main steam systems exposed to torus grade water. Torus water chemistry activities monitor and control torus water chemistry using PBAPS procedures and processes based on EPRI TR-103515, "BWR Water Chemistry Guidelines".

Based on the use of industry guidelines and PBAPS operating experience, there is reasonable assurance that the torus water chemistry activities will continue to adequately manage the aging effects in components exposed to torus grade water so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI TR-103515, “BWR Water Chemistry Guidelines”, 2000 Revision

B.1.6 Fuel Pool Chemistry Activities

ACTIVITY DESCRIPTION

Fuel pool chemistry activities consist of preventive measures that are used to manage loss of material and cracking in structures and components exposed to fuel pool water. Fuel pool chemistry activities provide for monitoring and controlling of fuel pool water chemistry using PBAPS procedures based on EPRI TR-103515, "BWR Water Chemistry Guidelines".

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Activity:*** Fuel pool chemistry activities manage loss of material for fuel pool gates, fuel storage racks, fuel pool liner, component supports, fuel preparation machines, refueling platform mast, and loss of material and cracking for fuel pool cooling and cleanup system components by monitoring and controlling detrimental contaminants.

(2) ***Preventive Actions:*** Fuel pool chemistry activities include periodic monitoring and controlling of fuel pool water chemistry to ensure that contaminants are maintained within pre-established limits, providing reasonable assurance that the aging effects of loss of material and cracking are managed.

(3) ***Parameters Monitored/Inspected:*** Conductivity is maintained based on the EPRI guidance. Impurities identified in EPRI TR-103515, such as chlorides and sulfates, are monitored and controlled.

(4) ***Detection of Aging Effects:*** Fuel pool chemistry activities mitigate the onset and propagation of loss of material and cracking aging effects. No credit is taken for detection of aging.

(5) ***Monitoring and Trending:*** Fuel pool water is monitored weekly to ensure purity is maintained within acceptable limits based on EPRI guidelines. Samples are taken and analyzed, and data are trended. The frequency of sampling is based on EPRI TR-103515.

(6) ***Acceptance Criteria:*** Maximum levels for various contaminants are maintained below system specific limits based on EPRI TR-103515. The acceptance criteria include the following limits: conductivity $\leq$ 2 μmho/cm, chlorides $\leq$ 100 ppb, and sulfates $\leq$ 100 ppb.

(7) ***Corrective Actions:*** Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the

significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The overall effectiveness of fuel pool chemistry activities is supported by the operating experience for components and structures influenced by fuel pool water chemistry. There has been no loss of intended function for components or structures for which fuel pool chemistry activities have been employed. Fuel pool chemistry is maintained per the recommendations of EPRI TR-103515 that have been developed based on industry experience, shown to be effective and are adjusted as new information becomes available.

SUMMARY

Fuel pool chemistry activities use preventive measures to manage loss of material and cracking in structures and components exposed to fuel pool water. Fuel pool chemistry activities monitor and control fuel pool water chemistry using PBAPS procedures based on EPRI TR-103515, "BWR Water Chemistry Guidelines".

Based on the use of industry guidelines and industry and PBAPS operating experience, there is reasonable assurance that the fuel pool chemistry activities will continue to adequately manage the aging effects associated with structures and components exposed to fuel pool water so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI TR-103515, "BWR Water Chemistry Guidelines", 2000 Revision.

B.1.7 High Pressure Service Water Radioactivity Monitoring Activities

ACTIVITY DESCRIPTION

High pressure service water radioactivity monitoring activities consist of sampling and analysis of HPSW system water to confirm the absence of radioactive contaminants. These condition-monitoring activities manage loss of material and cracking in the RHR heat exchangers and are implemented through PBAPS procedures.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: PBAPS high pressure service water radioactivity monitoring activities provide for routine sampling and isotopic analysis of the HPSW system water contained within the RHR heat exchangers to confirm the absence of radioactive contaminants.

(2) Preventive Actions: High pressure service water radioactivity monitoring activities identify aging degradation of the RHR Heat Exchangers prior to loss of intended function. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: High pressure service water radioactivity monitoring activities monitor HPSW system water contained within the RHR heat exchangers for the presence of radioactive isotopes that do not occur naturally. Samples taken from selected system test points and the bottom head drains of the RHR heat exchangers are analyzed to confirm the functional integrity of the RHR heat exchanger pressure boundary internal components, including tubes.

(4) Detection of Aging Effects: High pressure service water radioactivity monitoring activities provide for routine sampling and analysis to detect loss of material and cracking in the RHR heat exchangers. The extent and schedule of the activities assures detection of component degradation prior to the loss of their intended functions.

(5) Monitoring and Trending: High pressure service water radioactivity monitoring activities provide for monitoring of aging degradation of the RHR heat exchangers. Weekly sampling and analysis provides for timely component degradation detection.

(6) Acceptance Criteria: The presence of identified power production isotopes is considered to be positive activity. The acceptance criterion requires that no positive activity be identified. Naturally occurring isotopes are not considered as contaminants or indications of positive activity.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Leakage and minor degradation have been detected in the RHR heat exchangers on the HPSW (raw water) side. The degradation has involved leakage of floating head gaskets, and degradation of internal baffle welds. Evaluations and appropriate corrective actions, including gasket modifications were implemented prior to loss of intended function.

<u>SUMMARY</u>

High pressure service water radioactivity monitoring activities consist of sampling and analysis of HPSW system water to confirm the absence of radioactive contaminants. These condition-monitoring activities manage loss of material and cracking in the RHR heat exchangers and are implemented through PBAPS procedures.

Based on PBAPS operating experience, there is reasonable assurance that high pressure service water radioactivity monitoring activities will continue to manage loss of material and cracking in the RHR heat exchangers so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

<u>REFERENCES</u>

None

B.1.8 Inservice Inspection (ISI) Program

ACTIVITY DESCRIPTION

The inservice inspection (ISI) program, as augmented to address the requirements of GL88-01, provides for condition monitoring of pressure retaining piping and components in the scope of license renewal except for the reactor pressure vessel components and internals. This activity is part of the PBAPS ISI program which complies with the requirements of 1989 Edition of the ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", and is implemented through a PBAPS specification. The PBAPS ISI program includes requirements for inspections of ASME Class 1, 2, and 3 pressure retaining components. In addition, it provides for condition monitoring of ASME Class 1,2 and 3 piping and equipment supports and integral support anchors in accordance with ASME Code Case N-491-1.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Activity:*** The ISI program provides for condition monitoring of:

- support members for ASME Class 2 and 3 piping and equipment submerged in raw water or torus grade water,
- ASME Class 3 portions of HPSW, ECW, and ESW systems, including the CS, HPCI, RCIC, and RHR pump room cooling coils, exposed to raw water,
- ECW system piping and equipment support members located in an outdoor environment,
- ASME Class 1 components in the main steam, reactor pressure vessel instrumentation, RCIC and HPCI systems subject to a steam environment,
- ASME Class 1 components in reactor recirculation, reactor pressure vessel instrumentation, SBLC, feedwater, RHR, RCIC, core spray, HPCI, and PCIS (reactor water cleanup) systems subject to a reactor grade water environment,
- SBLC components from the suction side of the SBLC pumps to the RPV injection isolation valve subject to a borated water environment,
- components in reactor head flange leakoff piping, scram discharge piping, and the steam supply and return portions of the RCIC and HPCI systems subject to a wetted gas environment, and
- reactor pressure vessel closure studs.

(2) ***Preventive Actions:*** The ISI program consists of condition monitoring activities that detect degradation of components before loss of intended function. No preventive or mitigating attributes are associated with these activities.

*(3) **Parameters Monitored/Inspected:*** The ISI program provides for the following condition monitoring in the following environments:

Raw water and torus grade water:

- loss of material monitoring for support members for ASME Class 2 and 3 piping and equipment submerged in raw water or torus water using VT-3 visual inspections for corrosion,
- loss of material and cracking monitoring for components in HPSW, ESW, and ECW systems including the CS, HPCI, RCIC and RHR pump room cooling coils, subject to raw water through flow testing or identification of leakage during pressure tests,

Outdoor:

- loss of material monitoring for ECW system piping and equipment support members located in an outdoor ambient environment using VT-3 visual inspections for corrosion,

Steam:

- loss of material and cracking monitoring for ASME Class 1 components in the main steam, reactor pressure vessel instrumentation, HPCI and RCIC systems through monitoring for leaks during pressure testing,
- loss of material monitoring for ASME Class 1 components in the main steam and HPCI systems by visually inspecting valves for corrosion and pressure retaining wall thickness reduction, when they are disassembled for maintenance,

Reactor grade water:

- loss of material and cracking monitoring for susceptible ASME Class 1 components in the reactor recirculation, reactor pressure vessel instrumentation, SBLC, feedwater, RHR, RCIC, core spray, HPCI and PCIS (reactor water cleanup) systems through monitoring for leaks during pressure testing,
- loss of material and cracking monitoring for susceptible ASME Class 1 components in the reactor recirculation, RHR, core spray, and PCIS (reactor water cleanup) systems by visually inspecting valves or reactor recirculation pump casings for evidence of these aging effects when they are disassembled for maintenance,
- loss of material monitoring for susceptible ASME Class 1 components in the feedwater, RCIC, and HPCI systems by visually inspecting valves for evidence of this aging effect when they are disassembled for maintenance,
- cracking monitoring for susceptible ASME Class 1 components in the reactor recirculation, RHR, core spray, and PCIS (reactor water cleanup) systems through surface and volumetric examinations of pressure retaining welds and their heat affected zones in piping,

- cracking monitoring for ASME Class 1 reactor pressure vessel closure studs through surface and volumetric examinations,

- loss of fracture toughness monitoring for susceptible ASME Class 1 components in the reactor recirculation and PCIS (reactor water cleanup) systems by visually inspecting reactor water cleanup system valves and reactor recirculation pump casings for evidence of this aging effect when they are disassembled for maintenance,

Borated water:
- loss of material and cracking monitoring for SBLC components from the suction side of the SBLC pumps to the RPV injection isolation valve by monitoring for visible leakage of susceptible component pressure boundaries during pressure testing,

Wetted gas:
- loss of material and cracking monitoring for susceptible components in reactor head flange leakoff piping, scram discharge piping, and the steam supply and return portions of the RCIC and HPCI systems through pressure testing.

(4) ***Detection of Aging Effects:*** The method, extent and schedule of the ISI program examinations provide reasonable assurance of detection of cracks, loss of material and loss of fracture toughness before loss of intended function.

(5) Monitoring and Trending: The ISI program provides for monitoring for the presence of aging degradation per the guidance provided in the ASME Section XI schedules or Code Case N-491-1. Documentation that facilitates comparison with previous and subsequent inspection results is maintained in accordance with ASME Section XI, IWA-6000.

(6) Acceptance Criteria: Relevant conditions detected during testing are evaluated in accordance with ASME Section XI Articles IWB-3000, IWC-3000 or IWD-3000, for classes 1, 2 and 3 respectively. Conditions detected in support members are evaluated in accordance with PBAPS implementing procedure acceptance criteria that is in agreement with Code Case N-491-1.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The ISI program complies with ASME Section XI including Code Case N-491-1. It is reviewed and approved every 10 years. ASME Section XI incorporates industry practice and experience that provides reasonable assurance that it provides early detection, evaluation and corrective actions.

In response to concerns with intergranular stress corrosion cracking (IGSCC), portions of the 304 stainless steel reactor recirculation, PCIS (reactor water cleanup) and RHR piping were replaced with more IGSCC resistant type 316 stainless steel. PBAPS has implemented extensive inspection programs through the ISI program to identify IGSCC.

SUMMARY

The inservice inspection (ISI) program, as augmented to address the requirements of GL 88-01, provides for condition monitoring of pressure retaining piping and components in the scope of license renewal except for those components covered by the reactor pressure vessel and internals ISI program.

This program is part of the PBAPS ISI program, which complies with the requirements of 1989 Edition of the ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", and is implemented through a PBAPS specification.

Based on the use of industry standards and PBAPS operating experience, there is reasonable assurance that the ISI program will adequately manage the identified aging effects for the piping and components so that intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Plant Components," American Society of Mechanical Engineers, New York, NY, 1989.
(2) GL 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Piping, January 25, 1988, including Supplement 1, February 4, 1992

B.1.9 Primary Containment Inservice Inspection Program

ACTIVITY DESCRIPTION

The PBAPS primary containment ISI program provides for inspections that manage loss of material in the primary containment for Class MC pressure-retaining components, their integral attachments, and Class MC component supports; and loss of sealing for the drywell internal moisture barrier at the juncture of the containment wall and the concrete floor. The program complies with subsection IWE of ASME Section XI, 1992 Edition including 1992 Addenda, in accordance with the provisions of 10 CFR 50.55a, and is implemented through a PBAPS specification. Class MC support inspection also meets the support examination criteria established by Code Case N-491-1.

EVALUATION AND TECHNICAL BASIS

***(1)* Scope of Activity:** The primary containment ISI program manages loss of material in pressure boundary components and supports of the drywell, pressure suppression chamber and vent system. The components monitored in the drywell are the shell, head, control rod drive removal hatch, equipment hatch, personnel airlock, access manhole, inspection ports and penetration sleeves. The components monitored in the pressure suppression chamber are the shell, ring girders, access hatches and penetrations. The components monitored in the vent system are the vent lines, vent header with downcomers, downcomer bracing and vent system supports. The primary containment ISI program also manages loss of sealing for the moisture barrier inside the drywell at the juncture of the containment wall and the concrete floor.

(2) Preventive Actions: The primary containment ISI program utilizes inspections for detection of conditions. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: The primary containment ISI program provides for visual examination of containment surfaces and Class MC component supports for evidence of loss of material that could affect structural integrity or leak tightness of the primary containment. The moisture barrier is examined for wear, damage, erosion, tears, cracks or other defects that could affect leak tightness.

(4) *Detection of Aging Effects:* The method, extent and schedule of the primary containment ISI program visual examinations provide reasonable assurance that evidence of loss of material or loss of sealing is detected prior to loss of intended function.

(5) Monitoring and Trending: The primary containment ISI program provides for periodic monitoring for the presence of aging degradation per the guidance provided in ASME Section XI.

(6) Acceptance Criteria: The acceptance criteria for the drywell, pressure suppression chamber, vent system, and drywell moisture barrier are in accordance with the requirements of ASME XI, Subsection IWE. MC component supports acceptance criteria is in accordance with Code Case N-491-1.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Indications of coating degradation and loss of material in certain wetted areas of the pressure suppression chamber structure were found at PBAPS in 1991. The interior surfaces were recoated and torus grade water chemistry was improved. Subsequent pressure suppression chamber inspections indicate that the rate of degradation has decreased significantly. No failure of containment components due to the loss of material or failure of the moisture barrier inside the drywell due to the loss of sealing has occurred at PBAPS. The development process for the ASME Code that forms the basis for the primary containment ISI program includes review and approval by industry experts thereby assuring that significant industry data has been considered.

<u>SUMMARY</u>

The PBAPS primary containment ISI program manages loss of material in the primary containment for Class MC pressure-retaining components, their integral attachments, and Class MC component supports; and loss of sealing for the drywell internal moisture barrier at the juncture of the containment wall and the concrete floor. The program complies with subsection IWE of ASME Section XI, 1992 Edition including 1992 Addenda, in accordance with the provisions of 10 CFR 50.55a, and is implemented through a PBAPS specification. Class MC support inspection also meets the support examination criteria established by

Code Case N-491-1.

Based on the application of industry standards and the PBAPS operating experience, there is reasonable assurance that the primary containment ISI program will continue to adequately manage the identified aging effects so that the primary containment intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Plant Components," American Society of Mechanical Engineers, New York, NY, 1992 Edition with the 1992 Addenda.

B.1.10 Primary Containment Leakage Rate Testing Program

ACTIVITY DESCRIPTION

The PBAPS Primary Containment Leakage Rate Testing Program complies with the requirements of 10CFR50 Appendix J, Option B. Containment leak rate tests are performed to assure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage rates specified in the PBAPS Technical Specifications. An integrated leak rate test (ILRT) is performed during a period of reactor shutdown at a frequency of at least once every ten years. Local leak rate tests (LLRT) are performed on isolation valves and containment pressure boundary access penetrations at frequencies that comply with the requirements of 10CFR50 Appendix J, Option B.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The primary containment leakage rate testing program is credited with managing the loss of material of pressure retaining boundaries of piping and components in a wetted gas environment for containment atmosphere control and dilution, RHR, and primary containment isolation systems. Two types of tests are implemented in the program. The ILRT is performed to measure the overall primary containment integrated leakage rate. LLRTs are performed to measure local leakage rates across each pressure containing or leakage limiting boundary for the primary containment isolation system containment penetrations. The method, extent and schedule of these tests will detect minor leakage prior to loss of intended function.

The primary containment leakage rate testing program also manages change in the material properties and cracking of gaskets and O-rings of the primary containment pressure boundary access penetration points including the drywell head, the equipment hatch, the airlock, control rod drive removal hatch, drywell head access hatch, stabilizer inspection ports and the two access hatches in the pressure suppression chamber.

(2) Preventive Actions: The primary containment leakage rate testing program does not prevent or mitigate degradation due to aging effects but provides measures for condition monitoring to detect the degradation prior to loss of intended function.

(3) Parameters Monitored/Inspected: The parameters monitored are leakage rates through penetrations, piping, valves, fittings, and other access openings.

The ILRT is a test of the pressure retaining capabilities of the containment as a whole. The LLRTs measure the pressure retaining integrity of individual containment penetrations and the local leak rate at access penetration points of containment pressure boundary. Gaskets and O-rings not meeting the allowable leakage rate are assumed to be degraded, and are visually examined, replaced and re-tested until the leakage rate is acceptable.

(4) Detection of Aging Effects: The primary containment leakage rate testing program detects containment pressure boundary piping and components loss of material by integrated containment and individual penetration pressure tests to verify the pressure retaining integrity of the containment.

The ILRT demonstrates the overall leak-tightness of the containment and systems within the containment boundaries. LLRTs demonstrate the leak-tightness of individual containment boundaries of the piping systems.

The program also detects local leaks and measures leakage across the leakage-limiting boundary of containment access penetrations whose design incorporated gaskets and O-rings. Leakage is an indication of change in material properties and cracking of the sealing materials.

The primary containment leakage rate testing program serves to detect aging degradation prior to loss of the pressure boundary function of selected portions of the primary containment.

(5) Monitoring and Trending: Since the primary containment leakage rate testing program must be repeated throughout the operating license period, the entire primary containment pressure boundary, including access penetrations whose design incorporated gaskets and O-rings, is being monitored and trended over time.

(6) Acceptance Criteria: The acceptance criteria are defined in the PBAPS Technical Specifications. These acceptance criteria meet the requirements in 10CFR50, Appendix J, Option B.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The primary containment leakage rate testing program activities at PBAPS have been effective in maintaining the pressure integrity of the containment boundaries, including identification of leakage within the containment atmosphere control and dilution, RHR, and primary containment isolation systems pressure boundaries. Degradation due to loss of material and failure of pressure boundary function has not occurred in any of the portions of these systems subject to a wetted gas environment.

The program found no age related pressure boundary integrity failures due to local leakage for gaskets and O-rings at penetration access points including the drywell head, the equipment hatch, the airlock, control rod drive removal hatch, drywell head access hatch, stabilizer inspection ports and the two access hatches in the pressure suppression chamber. Consequently, the program has been effective in preventing unacceptable leakage through the containment pressure boundary. PBAPS continues to demonstrate it's good operating history by electing to perform Option B of 10CFR50 Appendix J test requirements.

SUMMARY

The primary containment leakage rate testing program activities manage loss of material for pressure retaining boundaries of piping and components in a wetted gas environment for containment atmosphere control and dilution, RHR, and primary containment isolation systems. These activities also manage change in the material properties and cracking for gaskets and O-rings of the access penetration points for the primary containment pressure boundary. Based on compliance with PBAPS Technical Specification requirements and PBAPS operating experience, there is reasonable assurance that the primary containment leakage rate testing program activities will continue to adequately manage aging effects of loss of material, and change in materials and cracking of the identified primary containment components to preclude loss of intended function and maintain the current licensing basis during the period of extended operation.

REFERENCES

None

B.1.11 Inservice Testing (IST) Program

ACTIVITY DESCRIPTION

The inservice testing (IST) program that is being credited for license renewal is a portion of the PBAPS IST program.

The PBAPS IST program is implemented by a PBAPS specification and provides for inservice testing of Class 1, 2, and 3 pumps and valves in compliance with the ASME O&M Code, 1990 Edition, and 10 CFR 50.55a.

The IST program manages flow blockage in the ESW and ECW components exposed to raw water. In addition, the program manages reduction of heat transfer for the torus water path through the RHR heat exchangers.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The IST program manages flow blockage of system components from the ECW pump through the ESW and ECW system piping to the ECT. In addition, the program manages reduction of heat transfer of the RHR heat exchangers by performing periodic flow testing of the torus water path.

(2) Preventive Actions: The IST program consists of condition monitoring activities that detect flow restrictions prior to loss of intended function. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: The IST program detects flow blockage in ECW and ESW components by measuring ECW pump discharge flow and ESW booster pump discharge flow. The IST detects reduction of heat transfer of the RHR heat exchangers by measuring the flow output of the RHR pump through the associated heat exchanger.

(4) *Detection of Aging Effects:* IST program activities detect flow blockage and reduction of heat transfer aging effects in carbon steel and stainless steel components. Buildup of corrosion products and general silting and fouling contribute to flow blockage and reduction of heat transfer. The test methods, extent and schedule of the IST program activities provide for detection of flow blockage in the ESW and ECW components and reduction of heat transfer in the RHR heat exchangers prior to loss of intended function.

(5) Monitoring and Trending: The periodic testing schedule provides for detection of flow blockage and reduction of heat transfer aging effects. Corrective maintenance work orders are initiated for observations of low or inadequate flow. Deficiencies discovered during testing are monitored in accordance with ASME O&M Code requirements.

(6) Acceptance Criteria: Conditions detected during RHR flow testing are evaluated in accordance with the test procedure by verifying acceptable flow rates through the RHR heat exchangers. ECW system flow, from the ECW pump through the ESW booster pumps to the ECT, is evaluated in accordance with the test procedure by verifying acceptable flow rates at the test point near the ECT.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The IST program complies with ASME O&M Code. The IST program is reviewed and approved every 10 years. The ASME O&M Code incorporates industry practice and experience.

System modifications have been made to repair and replace piping and components due to leakage and degrading performance. In addition, the presence of corrosion, silting and clams has been discovered and evaluated through plant work order inspections. RHR heat exchanger leaks, degradation of baffle plate welds, and tube plugging events have been noted. Corrective actions were implemented prior to loss of function.

SUMMARY

The PBAPS IST program is implemented by a PBAPS specification and provides for inservice testing of Class 1, 2, and 3 pumps and valves in compliance with the ASME O&M Code, 1990 Edition, and 10 CFR 50.55a. The IST program manages the aging effects of flow blockage in the ESW and ECW components exposed to raw water and reduction of heat transfer for the torus water path through the RHR heat exchangers.

Based on the application of industry standards and the PBAPS operating experience, there is reasonable assurance that the IST program will continue to provide a method for early detection of flow blockage and reduction of heat transfer so that intended functions of the components will be maintained consistent with the current licensing basis through the period of extended operation

REFERENCES

(1) ASME Code for Operation and Maintenance of Nuclear Power Plants, American Society of Mechanical Engineers, New York, NY, 1990.

B.1.12 Reactor Materials Surveillance Program

ACTIVITY DESCRIPTION

PBAPS maintains a Reactor Materials Surveillance (RMS) program consistent with the requirements of 10 CFR 50, Appendix H and ASTM E185 to manage loss of fracture toughness in the RPV beltline material. This program contains sufficient dosimetry and materials to monitor irradiation embrittlement during the period of extended operation. This program will be incorporated into the industry Integrated Surveillance Program (ISP), as described in BWRVIP-78.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The RMS program includes periodic withdrawal and testing of in-vessel capsules to monitor the effects of neutron embrittlement on the reactor vessel beltline materials. The results of this testing are used to determine plant operating limits.

(2) Preventive Actions: The RMS program is a condition monitoring activity. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: Charpy V-Notch 30 ft.-lb. transition temperature, upper shelf energy, and neutron fluence are monitored, consistent with the requirements of ASTM E185.

(4) Detection of Aging Effects: The RMS Program monitors the effects of neutron embrittlement by evaluating the loss of fracture toughness.

(5) Monitoring and Trending: Monitoring is provided by the RMS Program, as well as industry experience contained in Regulatory Guide 1.99. The program provides for compilation of information concerning loss of fracture toughness of RPV components.

(6) Acceptance Criteria: The acceptance criteria are based upon the requirements of ASTM E185 and Regulatory Guide 1.99, Revision 2.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions, and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: PBAPS Units 2 and 3 have tested surveillance capsules containing plate and weld material, and the results are consistent with Regulatory Guide 1.99, Revision 2 predictions.

SUMMARY

The PBAPS reactor materials surveillance (RMS) program manages loss of fracture toughness in the RPV beltline material. The program is consistent with the requirements of 10 CFR 50, Appendix H and ASTM E185 and will be incorporated into the industry Integrated Surveillance Program (ISP), as described in BWRVIP-78.

Based on the application of industry standards and the PBAPS operating experience, there is reasonable assurance that the RMS program will continue to adequately manage fracture toughness in the RPV beltline material so that intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) EPRI Report TR-114228, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)", EPRI, Palo Alto, CA, December 1999

(2) Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", NRC, May 1988

B.1.13 Standby Liquid Control System Surveillance Activities

ACTIVITY DESCRIPTION

The standby liquid control (SBLC) system surveillance activities provide for managing the aging effects of loss of material and cracking in components of the SBLC system that are on the suction side of the SBLC pumps. The surveillance activities monitor the SBLC solution tank liquid level on a daily basis in accordance with a PBAPS procedure. The SBLC components covered by this surveillance include the solution tank, piping, and valves on the suction side of the SBLC pumps. The extent and frequency of this monitoring verifies the component intended functions.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The SBLC system surveillance activities manage the loss of material and cracking aging effects for the SBLC solution tank, piping, and valves, which are on the suction side of the SBLC pumps.

(2) Preventive Actions: The SBLC system surveillance activities provide a method to detect leakage due to loss of material or cracking prior to loss of intended function of the components. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The SBLC system surveillance activities verify the liquid level in the solution tank.

(4) Detection of Aging Effects: The daily surveillance of solution tank level will detect minor leakage in the SBLC components prior to excessive degradation from loss of material or cracking and prior to loss of intended component functions.

(5) Monitoring and Trending: The daily monitoring of solution tank liquid level is required by PBAPS Technical Specifications and provides for timely detection of loss of material or cracking in SBLC components that are on the suction side of the SBLC pumps.

(6) Acceptance Criteria: The SBLC system surveillance activities verify the required solution tank level is greater than 46% as specified in the PBAPS Technical Specifications to maintain intended system function.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: A review of both PBAPS and industry experience uncovered no age related loss of material or cracking in the SBLC components covered by this surveillance activity.

SUMMARY

The standby liquid control (SBLC) system surveillance activities manage loss of material and cracking for components of the SBLC system that are on the suction side of the SBLC pumps. PBAPS operating experience, and the use of daily surveillance of the SBLC solution tank liquid level to ensure it meets PBAPS Technical Specifications, provide reasonable assurance that the SBLC system surveillance activities will continue to adequately manage the effects of aging associated with components on the suction side of the SBLC pumps that are exposed to a borated water environment so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.1.14 Crane Inspection Activities

ACTIVITY DESCRIPTION

PBAPS crane inspection activities consist of inspections that are relied upon to manage loss of material for passive components of cranes and hoists. Crane inspection activities comply with the requirements of ASME B30.2, B30.11, B30.16 and B30.17 and are implemented through a PBAPS procedure.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: Crane inspection activities consist of inspections of the structural members, rails, and rail anchorage for the circulating water pump structure gantry crane located in an outdoor environment, and rails and monorails for the cranes and hoists located in a sheltered environment.

(2) Preventive Actions: Crane inspection activities include inspections to identify component aging effects prior to loss of intended function. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: Crane inspection activities verify structural integrity of crane and hoist elements required to maintain intended functions and comply with ASME B30.2, B30.11, B30.16 and B30.17. The activities include visual inspections for conditions such as corroded structural members, misalignment, flaking, sidewear of rails, loose tie down bolts, and excessive wear or deformation of monorail lower flange.

(4) Detection of Aging Effects: Crane inspection activities provide for inspections to identify deficiencies in components and degradation due to loss of material.

(5) Monitoring and Trending: Crane inspection activities monitor inspection results from previously identified findings and for newly developing conditions. The annual inspections provide for prediction of the onset of degradation and for timely implementation of corrective actions to prevent loss of intended function.

(6) Acceptance Criteria: Crane inspection activities provide for engineering evaluation of inspection results to assess the ability of the crane or hoist to perform its intended function. The acceptance criterion is no unacceptable visual indication of loss of material due to corrosion or wear.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: No incidents of failure of passive crane and hoist components due to aging have occurred at PBAPS. Loss of material in crane rails and monorails has been detected and managed by the crane inspection activities providing reasonable assurance that the intended functions of crane and hoist passive components will be maintained during the period of extended operation.

SUMMARY

PBAPS crane inspection activities manage loss of material for passive components of cranes and hoists. Crane inspection activities comply with the requirements of ASME B30.2, B30.11, B30.16 and B30.17 and are implemented through a PBAPS procedure.

Based on the application of industry standards, the extent and schedule of the inspections, and the PBAPS operating experience, there is reasonable assurance that the crane inspection activities will adequately manage the aging effects so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) ASME Safety Standard B30.2, "Overhead and Gantry Cranes."
(2) ASME Safety Standard B30.11, "Monorails and Underhung Cranes"
(3) ASME Safety Standard B30.16, "Overhead Hoists (Underhung)"
(4) ASME Safety Standard B30.17, "Overhead and Gantry Cranes (Top Running Bridge, Single Girder With Top or Running Trolley Hoist)"

B.1.15 Conowingo Hydroelectric Plant (Dam) Aging Management Program

ACTIVITY DESCRIPTION

The Conowingo Hydroelectric Plant dam is subject to the FERC 5-year inspection program. This program consists of a visual inspection by a qualified independent consultant approved by FERC, and is in compliance with Title 18 of the Code of Federal Regulations, Conservation of Power and Water Resources, Part 12 (Safety of Water Power Projects and Project Works), Subpart D (Inspection by Independent Consultant). The NRC has found that mandated FERC 5-year inspection programs are acceptable for aging management.

REFERENCES

(1) NRC Letter, May 5, 1999, Christopher I. Grimes (NRC) to Douglas J. Walters (NEI), License Renewal Issue No. 98-0100, "Crediting FERC-Required Inspection and Maintenance Programs for Dam Aging Management"

B.1.16 Maintenance Rule Structural Monitoring Program

ACTIVITY DESCRIPTION

The maintenance rule structural monitoring program is that portion of the PBAPS maintenance rule structural monitoring program that is being credited for license renewal.

The maintenance rule structural monitoring program provides for condition monitoring of structures and components within the scope of license renewal that are not covered by other existing inspection programs. The program complies with the 10CFR50.65 and is implemented through a PBAPS procedure.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The maintenance rule structural monitoring program provides for condition monitoring of:

- Emergency cooling tower and reservoir reinforced concrete walls in contact with raw water;
- Structural steel components outside primary containment exposed to the outdoor environment, including siding and exterior blowout panels;
- Emergency cooling water outdoor piping support anchors; and
- Penetration seals and expansion joint seals.

(2) Preventive Actions: The maintenance rule structural monitoring program is a condition monitoring program that utilizes inspections to identify aging effects prior to loss of intended function. No preventive or mitigating attributes are associated with this program.

(3) Parameters Monitored/Inspected: The maintenance rule structural monitoring program specifies visual inspection of:

- Emergency cooling tower and reservoir reinforced concrete walls in contact with raw water for evidence of a change in material properties due to leaching of calcium hydroxide;
- Structural steel components for loss of material;
- Emergency cooling water outdoor piping support anchors for corrosion; and
- Penetration seals and expansion joint seals for gaps, voids, tears and general degradation associated with cracking, delamination and separation, and change in material properties.

(4) Detection of Aging Effects: The method, extent and frequency of maintenance rule structural monitoring program inspections provide reasonable assurance of detection of change in material properties, loss of material, and cracking, delamination and separation aging effects prior to a loss of intended function.

(5) Monitoring and Trending: Structures and components are inspected at least once every four years, with provisions to perform trending and root cause analysis and increase the frequency of inspections in the event problems are identified.

(6) Acceptance Criteria: Maintenance rule structural monitoring program inspection results are documented and evaluated by qualified personnel. Evaluations are based on ensuring that the intended functions of the structure or component are maintained. The acceptance criteria are consistent with the recommended criteria in NUMARC 93-01, Revision 2.

- Acceptance criteria for the emergency cooling tower and reservoir walls are based on an evaluation of the walls' condition when compared to the condition from previous inspections in order to verify that no changes have occurred that may affect their ability to perform their intended functions.

- Acceptance criteria for structural steel are directed at finding corrosion that may affect its ability to perform its intended functions.

- Acceptance criteria for visual inspection of the emergency cooling water outdoor piping support anchors require that structures be free of corrosion that could lead to possible failure.

- Acceptance criteria for the inspections performed on penetration seals and expansion joint seals are provided on PBAPS drawings and in the inspection procedure for these seals. These documents are directed at finding any changes in the condition of these components that may affect their ability to perform their intended functions.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Aging effects have been detected and evaluated by the program before loss of intended function, providing reasonable assurance that the intended function of structures and components will be maintained during the period of extended plant operation. Specific PBAPS experiences include:

- Effective management of change in material properties due to contact of the emergency cooling tower and reservoir reinforced concrete walls with raw water by the detection and monitoring of leaching of calcium hydroxide. These walls have not experienced a loss of intended function.
- No loss of function of the emergency cooling water outdoor piping support anchors resulting from aging of the anchors. A review of industry experience shows that salt water corrosion and boric acid corrosion are the most common causes of loss of material for anchors. The anchors at PBAPS are not subjected to an environment containing either salt water or boric acid.
- Degraded conditions were found for some penetration and expansion joint seals. Most of the degradation was not attributed to aging effects. Corrective actions for all degraded conditions were taken prior to loss of any intended function.

SUMMARY

The maintenance rule structural monitoring program provides for condition monitoring of structures and components within the scope of license renewal that are not covered by other existing inspection programs. The program complies with the 10CFR50.65 and uses acceptance criteria that are consistent with the recommended criteria in NUMARC 93-01, Revision 2.

Based on the PBAPS operating experience and the use of acceptance criteria consistent with industry recommendations, there is reasonable assurance that the maintenance rule structural monitoring program will continue to adequately manage the identified aging effects on the structures and components so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) NUMARC 93-01, Revision 2 “Monitoring the Effect of Maintenance at Nuclear Power Plants”

B.2 ENHANCED AGING MANAGEMENT ACTIVITIES

B.2.1 Lubricating and Fuel Oil Quality Testing Activities

ACTIVITY DESCRIPTION

Lubricating and fuel oil quality testing activities manage loss of material, cracking, and reduction of heat transfer in components that contain or are exposed to lubricating oil or fuel oil. Lubricating and fuel oil quality testing activities are implemented through PBAPS procedures and include sampling and analysis of lubricating oil and fuel oil for detrimental contaminants. The presence of water or particulates may also be indicative of inleakage and corrosion product buildup. The aging management review determined that diesel driven fire pump fuel oil sampling methods will be enhanced to improve water detection capabilities. Analyses of the diesel driven fire pump and EDG fuel oil samples will be enhanced to add testing for microbes in any water detected.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: Lubricating and fuel oil quality testing activities provide for sampling and testing of lubricating oil in components in emergency diesel generator (EDG), high pressure coolant injection (HPCI), high pressure service water (HPSW), core spray (CS), and reactor core isolation cooling (RCIC) systems. Lubricating and fuel oil quality testing activities also provide for sampling and testing of fuel oil in the EDG and diesel driven fire pump fuel oil systems.

(2) Preventive Actions: The lubricating and fuel oil quality testing activities are aging management activities that are preventive in that reasonable assurance is provided that potentially detrimental concentrations of contaminants such as water and particulate are not present in the oil.

(3) Parameters Monitored/Inspected: Lubricating oil sample analyses are performed periodically in accordance with an approved PBAPS procedure. Samples are analyzed for attributes such as viscosity, moisture content, and pH.

Samples of new fuel oil deliveries are analyzed for water and sediment. Emergency diesel generator and diesel driven fire pump fuel oil storage tank samples are also periodically analyzed for the presence of water and particulate content of the fuel. Enhancements to the diesel driven fire pump fuel oil sampling techniques will be made to improve the methods for detection of water in the fuel. Sampling activities for water that may be detected in the EDG and diesel driven fire pump fuel oil systems will be enhanced to include an analysis for microbes.

(4) Detection of Aging Effects: Testing of lubricating oil for water and contaminants provides a means for detecting loss of material and cracking in the HPCI and RCIC, and EDG systems, and monitors for water inleakage in the HPCI and RCIC turbine lube oil coolers, HPSW and CS pump motor oil coolers, and the EDG lube oil cooler. Testing of fuel oil for the presence of corrosion particles or water provides a means for detecting loss of material for fuel oil storage tanks and underground fuel oil piping.

(5) Monitoring and Trending: Lubricating oil and fuel oil analyses are regularly scheduled and the results are evaluated to aid in the identification of potential adverse conditions.

(6) Acceptance Criteria: The lubricating and fuel oil quality testing activities are performed in accordance with approved PBAPS procedures which contain quantitative and qualitative acceptance criteria. Lubricating oil analysis acceptance criteria are based on deviations from the physical requirements identified in the oil type listing. The acceptability of lubricating oil test results is based upon comparison with new oil values, published data, or previous oil analysis results. Oil is acceptable if viscosity changes by no more than +15% to -10%, percent water is less than or equal to 0.10, and pH is within the required values for the type of oil being analyzed.

EDG fuel oil analysis acceptance criteria are contained in the PBAPS Technical Specifications and are based on the requirements of ASTM D2276-78 and ASTM D975-81. A fuel oil testing procedure based on ASTM D975-81 requires that new fuel oil contain no visible water or sediment. PBAPS Technical Specifications require periodic sampling of the EDG fuel oil for particulates and the presence of water. Tests for particulates use the methods specified in ASTM D2276-78 to provide assurance that the particulate limit of 10 mg/L is not exceeded. Plant procedures limit EDG fuel oil storage tank water accumulation to 100 ml/L for samples taken from the bottom of the tank, and EDG fuel oil day tank water accumulation to none present at the conclusion of the surveillance test. Diesel driven fire pump fuel oil analysis acceptance criteria are based on the requirements of ASTM D975-74, which requires that the fuel contain a maximum of 0.05%, by volume, of water and sediment. Fuel oil analysis for both the EDG and diesel driven fire pump fuel samples will be enhanced to analyze any water discovered in the storage or day tanks for the presence of microbes.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The overall effectiveness of the lubricating and fuel oil quality testing activities is supported by the operating experience PBAPS has had with lubricating oil and fuel oil systems. Minor contamination events such as sediment in the diesel driven fire pump fuel oil day tank (one event), water in the diesel driven fire pump fuel oil storage tank (two events), and water in the EDG fuel oil storage tanks (two events in 1988) have been detected and corrected in a timely manner. Since moving the diesel driven fire pump fuel oil storage tank indoors, there have been no incidents of water detected in the tank. There have been no age related component failures resulting in a loss of function of systems in lubricating oil or fuel oil environments.

SUMMARY

The lubricating and fuel oil quality testing activities are preventive aging management activities that assure potentially detrimental concentrations of water and particulate are not present in the oil. These activities also provide for detection of loss of material and cracking in certain components containing lubricating or fuel oil. Based on the use of industry standards and PBAPS operating experience there is reasonable assurance that the lubricating and fuel oil quality testing activities will continue to adequately manage the effects of aging associated with components exposed to lubricating oil and fuel oil environments so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) ASTM Standard D2276-1978, "Test Methods for Particulate Contaminant in Aviation Turbine Fuels"
(2) ASTM Standard D975-1974, "Specification for Diesel Fuel Oils"
(3) ASTM Standard D975-1981, "Specification for Diesel Fuel Oils"

B.2.2 Boraflex Management Activities

ACTIVITY DESCRIPTION

The Boraflex management activities provide for aging management of the spent fuel rack neutron poison material. These activities involve monitoring the condition of Boraflex by routinely sampling fuel pool silica levels and periodically performing in-situ measurement of boron-10 areal density. Existing processes will be enhanced by including the requirement and frequency for in-situ measurement of boron-10 areal density in PBAPS procedures.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activities: PBAPS Boraflex management activities manage the effects of spent fuel rack Boraflex material degradation to ensure that the intended function is maintained. These activities are based on EPRI guidelines and include routine monitoring and trending of silica in the spent fuel pool and periodically performing in-situ measurement of boron-10 areal density.

(2) Preventive Actions: The Boraflex management activities monitor the condition of Boraflex to ensure that its degradation is detected before a loss of intended function. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: Silica in fuel pool water is monitored for indication of leaching of boron from the matrix and degradation of the matrix itself. Measurement of the boron-10 areal density of in-service spent fuel storage rack panels is used to monitor neutron attenuation capability.

(4) Detection of Aging Effects: Boraflex degradation from change in material properties will result in increased levels of silica in fuel pool water or loss of boron-10 areal density. These parameters are monitored in accordance with EPRI guidelines at a frequency that assures identification of unacceptable aging effects before loss of intended function.

(5) Monitoring and Trending: Monitoring of change in material properties is accomplished through periodic measurement of boron-10 areal density of in-service spent fuel storage rack panels and sampling of silica levels in fuel pool water. This data is used to trend and predict performance of Boraflex.

(6) Acceptance Criteria: Analysis has shown that Boraflex will perform its intended function if degradation is maintained at less than a 10% uniform loss and at less than 10-cm. randomly distributed gaps. These parameter limits ensure that current licensing basis fuel pool reactivity limit ($k_{eff} \leq 0.95$ or 5% margin) is not exceeded. Spent fuel pool silica data is trended and compared in an industry-wide EPRI database. A sustained increasing trend in spent fuel pool silica concentration, inconsistent with previous seasonal/refueling changes, requires an engineering evaluation to determine the need for corrective action.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: NRC Information Notices IN 87-43, IN 93-70, and IN 95-38 address several cases of significant degradation of Boraflex in spent fuel pools. In response to these findings, the US NRC issued Generic Letter 96-04. The industry formed a Boraflex Working Group with EPRI and developed a strategy for tracking Boraflex performance in spent fuel racks, detecting the onset of material degradation, and mitigating its effects.

The Peach Bottom spent fuel racks and Boraflex have been in service since 1986. In-situ testing of representative Boraflex panels was conducted in 1996 for Unit 2 and 2001 for Unit 3. Test results identified Boraflex degradation; however, the degradations are less severe than experienced in the industry. Continued testing will identify unacceptable degradation prior to loss of intended function.

<u>SUMMARY</u>

The Boraflex management activities provide for aging management of the spent fuel rack neutron poison material. These activities involve monitoring the condition of Boraflex by routinely sampling fuel pool silica levels and periodically performing in-situ measurement of boron-10 areal density.

Based on the use of industry guidelines and PBAPS and industry operating experience, there is reasonable assurance that the Boraflex management activities will continue to adequately manage the effects of aging of spent fuel rack Boraflex so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks", June 26, 1996
(2) EPRI TR-103300, "Guidelines for Boraflex Use in Spent Fuel Storage Racks"
(3) EPRI TR-101986, "Boraflex Test Results and Evaluation"
(4) EPRI NP-6159, December 1998, "An Assessment of Boraflex Performance in Spent Nuclear Fuel Racks"
(5) NRC Information Notice (IN) 87-43, "Gaps in Neutron-Absorbing Material in High Density Spent Fuel Storage Racks", September 8, 1987.
(6) NRC Information Notice (IN) 93-70, "Degradation of Boraflex Neutron Absorber Coupons", September 10, 1993.
(7) NRC Information Notice (IN) 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks", September 8, 1995.

B.2.3 Ventilation System Inspection and Testing Activities

ACTIVITY DESCRIPTION

PBAPS ventilation system inspection and testing activities consist of inspections and tests that are relied upon to manage change in material properties in ventilation system components. The ventilation system inspection and testing activities are implemented through periodic surveillance tests and preventive maintenance work orders that provide for assurance of functionality of the ventilation systems by confirmation of integrity of selected components. The aging management review determined that scope of the components covered by these activities will be enhanced to provide added assurance of aging management.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: PBAPS ventilation system inspection and testing activities include surveillance tests that provide for inspection and leakage testing of the filter plenum access door seals in the standby gas treatment system and the control room ventilation system. These activities also include inspections of fan flex connections for the standby gas treatment system, the control room ventilation system, the battery room and emergency switchgear ventilation system exhaust fans, and the ESW booster pump room ventilation supply fans.

Ventilation system inspection and testing activities will be enhanced to include inspections of fan flex connections in the diesel generator building ventilation system, the pump structure ventilation system, and the battery room and emergency switchgear ventilation system supply fans.

(2) Preventive Actions: Ventilation system inspection and testing activities include the inspections and testing necessary to identify component aging degradation effects prior to loss of intended function. No preventive or mitigative attributes are associated with these activities.

(3) Parameters Monitored/Inspected: Ventilation system inspection and testing activities monitor and inspect for the presence of aging degradation by visual inspection and leakage testing. Pressure boundary integrity of fan flex connections and filter plenum access door seals is confirmed through inspections for evidence of changes in resilience, strength and elasticity. Testing of the filter plenum access door seals confirms their leak-tightness.

(4) Detection of Aging Effects: Ventilation system inspection and testing activities provide for periodic component inspections and leakage testing to detect change in material properties. The extent and schedule of the inspections and testing assures detection of component degradation prior to the loss of their intended functions.

(5) Monitoring and Trending: Ventilation system inspection and testing activities provide for monitoring and trending of aging degradation. Ventilation system components are periodically inspected which provides for timely component degradation detection. The inspection interval is dependent on the component and the system in which it resides. Components in the standby gas treatment system and the control room ventilation system are inspected and tested annually.

(6) Acceptance Criteria: Ventilation system inspection and testing activities acceptance criteria are defined in the specific inspection and testing procedures and confirm ventilation system operability by demonstrating that there is no significant pressure boundary leakage. Acceptance criterion for the filter plenum access door seals is lack of visual indication of smoke escaping through the seals during the smoke test.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: No physical degradation of metallic ventilation system components has been identified at PBAPS or by the industry in general. At PBAPS, the fan flex connection and filter plenum access door seal inspections have detected damaged components that were subsequently replaced in accordance with the inspection procedures. Torn and cracked fan flex connections for various ventilation fans have been detected during performance of inspection procedures. In these cases new flex connections were installed. In addition, access door seal leakage has been detected during performance of the seal leakage testing. New seals were installed as a result of the surveillance test process. In all cases the corrective actions, including component replacement, were taken prior to loss of intended function.

SUMMARY

The ventilation system inspection and testing activities assure that change in material properties is managed for fan flex connections and filter plenum access door seals. Based on the periodic inspection and testing and PBAPS operating experience, there is reasonable assurance that the ventilation system inspection and testing activities will continue to adequately manage the identified aging effects of the components so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.4 Emergency Diesel Generator Inspection Activities

ACTIVITY DESCRIPTION

The emergency diesel generator (EDG) inspection activities provide for condition monitoring of EDG equipment within the scope of licensing renewal that are exposed to a gaseous, closed cooling water, lubricating oil or fuel oil environment. Loss of material in the starting air system air receivers is mitigated by daily removal of any accumulation of condensate. Loss of material and cracking in lubricating oil and fuel oil systems is mitigated by periodic inspections. Visual inspections for change in material properties of flexible hoses in the starting air system and the cooling water system are performed in accordance with a PBAPS procedure in connection with periodic EDG maintenance. This procedure will be enhanced to require inspections of the lubricating oil and fuel oil system flexible hoses for a change in material properties. The aging management review also determined that the management of loss of material in the EDG exhaust silencer will be enhanced by periodic disassembly, cleaning, and inspection of an automatic drain trap to ensure its functionality in preventing condensation build up.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The EDG inspection activities manage the aging effects of loss of material, cracking, and change in material properties by:

- mitigating actions which ensure periodic removal of moisture from the starting air system air receivers.
- periodic inspections of the EDG lubricating oil and fuel oil systems for loss of material and cracking.
- periodic inspections of flexible hoses in the starting air and cooling water systems for a change in material properties.

The scope of the EDG inspection activities will be enhanced to:

- perform periodic inspections of EDG lubricating oil and fuel oil system flexible hoses for a change in material properties.
- periodically disassemble, clean, and inspect the EDG exhaust silencer drain trap to ensure its functionality to prevent condensation buildup and the resulting loss of material of the exhaust silencer.

(2) Preventive Actions: The EDG inspection activities provide mitigation methods to manage loss of material in the starting air system air receivers and the EDG exhaust silencer by ensuring periodic removal of moisture. The remaining EDG inspection activities provide inspection methods to identify aging effects, and as such, there are no preventive or mitigative attributes associated with them.

(3) Parameters Monitored/Inspected: The existing EDG inspection activities provide for:

- blowing down the EDG starting air system air receivers until no more moisture is present in the drain line.
- performance of visual inspections of the lubricating oil and fuel oil systems for loss of material and cracking, and performance of periodic UT inspections of the EDG fuel oil storage tanks for loss of material.
- performance of visual inspections of the starting air and cooling water system flexible hoses for a change in material properties.

EDG inspection activities will be enhanced to include:

- performance of visual inspections of the lubricating oil and fuel oil system flexible hoses for a change in material properties.
- periodic disassembly, cleaning, and inspection of the EDG exhaust silencer drain trap to ensure it is operating properly.

(4) Detection of Aging Effects: The starting air system air receiver inspections and the periodic exhaust silencer automatic drain trap preventive maintenance activities mitigate potential aging effects. Visual inspections of the EDG fuel oil day tanks and EDG lubricating and fuel oil system components, and visual and UT inspections of the EDG fuel oil storage tanks are performed to assess loss of material and cracking aging effects. Visual inspection of flexible hoses provides for detection of change in material properties by observation of swelling or cracking. PBAPS procedures for EDG maintenance contain requirements for visual examinations of starting air and cooling water system flexible hoses. This procedure will be enhanced to include inspections of lubricating and fuel oil system flexible hoses.

(5) Monitoring and Trending: Existing EDG inspection activities provide the following monitoring and trending activities:

- daily starting air system receiver inspections mitigate aging and require no monitoring or trending.
- EDG lubricating and fuel oil system examinations for loss of material and cracking range from every two years for engine mounted components to a 10-year inspection of the EDG fuel oil storage tank and day tank interiors.
- starting air and cooling water system flexible hose examinations for a change in material properties are conducted every two years.

Enhancements to EDG inspection activities will provide the following monitoring and trending activities:

- examinations of the EDG lubricating and fuel oil system flexible hoses for a change in material properties will be conducted every two years.

- the periodic preventive maintenance of the EDG exhaust silencer automatic drain trap will mitigate aging and requires no monitoring or trending.

(6) Acceptance Criteria: The EDG starting air system air receiver inspection contains the requirement to blow down the air receiver until there is no moisture in its drain line. Examinations for loss of material, visible cracking, and change in material properties aging effects are conducted in accordance with approved PBAPS procedures. Degraded components are repaired or replaced as required. The EDG exhaust silencer automatic drain trap preventive maintenance will ensure the trap is left in good working order.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The overall effectiveness of the EDG inspection activities is supported by the operating experience PBAPS has had with the starting air, engine exhaust, cooling water, lubricating oil, and fuel oil systems. Minor leakage events in the starting air, engine exhaust, cooling water, lubricating oil, and fuel oil systems have been detected and corrected in a timely manner. Due to numerous small leaks, portions of the EDG exhaust piping have been replaced. Water and sediment have been observed during the fuel oil storage tank inspections. During the 1995 - 1996 fuel oil storage tank inspections, the lowest tank shell UT reading was 0.375", which is equal to the original specified thickness for the shell. No age related failures have been observed in EDG system flexible hoses. There have been no starting air, engine exhaust, cooling water, lubricating or fuel oil system age related component failures resulting in a loss of function of the EDG.

<u>SUMMARY</u>

The emergency diesel generator (EDG) inspection activities provide for condition monitoring of EDG equipment within the scope of license renewal that are exposed to a gaseous, closed cooling water, lubricating or fuel oil environment.

Loss of material in the EDG starting air receivers is mitigated by periodic removal of any moisture present. Loss of material in the EDG exhaust silencer is mitigated by maintaining the exhaust silencer drain trap in good working order. Loss of material and cracking in the lubricating and fuel oil systems is detected by periodic inspections. A change in material properties in starting air, cooling water, lubricating and fuel oil system flexible hoses is detected by periodic inspections.

Based on PBAPS operating experience, there is reasonable assurance that the EDG inspection activities will continue to adequately manage the loss of material, cracking, and change in material properties aging effects in EDG equipment within the scope of license renewal so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.5 Outdoor, Buried, and Submerged Component Inspection Activities

ACTIVITY DESCRIPTION

The outdoor, buried, and submerged component inspection activities provide for management of loss of material and cracking of external surfaces of components subject to outdoor, buried, and raw water external environments. Separately, the ISI program provides for monitoring of pressure boundary integrity for outdoor and buried components through pressure tests, flow tests, and inspections. The outdoor, buried, and submerged component inspection activities are implemented in accordance with PBAPS maintenance procedures and routine test procedures that provide instructions for inspections. Component inspections include inspections of external surfaces for the presence of pitting, corrosion and other abnormalities. The visual inspections provide reasonable assurance that aging effects are being managed such that system and component intended functions are maintained. The aging management review determined that the scope of components covered by these activities will be enhanced to provide added assurance of aging management.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The outdoor, buried, and submerged component inspection activities provide for detection of degradation due to loss of material or cracking of external surfaces for outdoor, buried, and submerged components.

The submerged components include HPSW, ESW, ECW, and fire protection system pumps. Components exposed to the outdoor environment include HPSW and ESW system manual discharge pond isolation valves, condensate storage system piping and valves, the external surfaces of the CSTs, and the piping insulation jacketing at the CST. The buried components include HPSW, ESW, ECW, fire protection, and EDG fuel oil system piping; fire protection system fire main isolation valves; EDG fuel oil storage tanks; the SGTS exhaust to the main stack; and the underside of the CSTs.

The scope of these activities will be enhanced to include periodic visual inspection of the external surfaces of the CSTs, periodic visual inspection of the ECW pump casing and casing bolts, and visual inspection of buried commodities whenever they are uncovered during excavation. Inspection of the refueling water storage tank (RWST) will be performed as a representative inspection to determine the condition of the underside of the CSTs. The CSTs and RWST are of same material, construction and internal environment; thus the condition of the RWST is representative of the condition of the CSTs.

(2) Preventive Actions: The outdoor, buried, and submerged component inspection activities provide inspection methods to identify aging effects on external surfaces prior to loss of intended function. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The outdoor, buried, and submerged component inspection activities provide for inspection of external component surfaces of submerged pumps and outdoor valves for evidence of corrosion and cracking; inspection of buried commodities for the presence of coating degradation, if coated, or base metal corrosion and cracking, if uncoated; inspection of the external surfaces of the CSTs and inspection of outdoor condensate system piping insulation to verify that the jacketing is free of damage; volumetric inspection of the bottom of the RWST for corrosion as a representative inspection for the underside of the CST.

(4) Detection of Aging Effects: Outdoor, buried, and submerged components are visually inspected to identify loss of material and cracking aging effects. These inspections in combination with other condition monitoring methods in the PBAPS aging management activities provide for detection of aging effects prior to loss of intended function.

Outdoor valves are inspected during performance of component maintenance. These inspections provide for detection of external loss of material aging effects.

Outdoor insulation jacketing is periodically inspected as part of heat trace testing. The extent and schedule of the outdoor insulation inspections assure detection of loss of material before any jacketing leaks develop.

The excavating procedure will be enhanced to direct visual inspection of buried commodities whenever they are uncovered during excavation. The inspection of the external coating or the base metal, if uncoated, will detect any external degradation due to aging.

The above ground tank inspection procedure will be enhanced to include periodic visual inspection of the above ground external surfaces of the CSTs.

Inspections during component maintenance of submerged pumps and additional periodic inspections of the ECW pump will detect external casing degradation prior to loss of its pressure boundary function.

The inspection of the RWST will be enhanced to periodically perform volumetric inspection of the bottom of the RWST for loss of material as a representative inspection to determine the condition of the underside of the CSTs.

(5) Monitoring and Trending: Inspections of submerged pumps and outdoor valves are conducted as part of the maintenance process. In addition, the ECW pump will be periodically inspected as part of preventive maintenance. Buried commodities will be visually inspected whenever they are uncovered during excavation activities. The inspections of the RWST will be used to determine the condition of the underside of the CST. Degradation identified during the inspections is evaluated in accordance with procedure requirements.

Annual inspections of the outdoor piping insulation jacketing and the above ground exterior surfaces of the CSTs provide detection of corrosion degradation or damage to the jacketing or to the tanks.

(6) Acceptance Criteria: Identified loss of material or cracking will be evaluated to provide reasonable assurance that system and component functions are maintained. Indications of component degradation detected during the inspection processes will be evaluated by engineering and the appropriate corrective actions will be initiated. Degradation of the refueling water storage tank noted during its examination will result in the CSTs being evaluated for degradation.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions, and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Significant external surface degradation of outdoor, buried, or submerged components has not been identified to date at PBAPS except for the ECW pump. The performance lives of the HPSW, ESW and fire protection pumps are limited by wear of the pump internals. Inspections of the casings during maintenance have not detected significant corrosion degradation and the pumps are re-coated following re-assembly. The ECW pump is operated less frequently. Therefore, its performance life is dependent on external surface degradation. Enhanced periodic inspections of the pump casing and casing bolts will detect future pump casing corrosion degradation.

SUMMARY

The outdoor, buried, and submerged component inspection activities provide for determination of degradation due to loss of material or cracking of external surfaces for outdoor, buried, and submerged components. The outdoor, buried, and submerged component inspection activities are implemented in accordance with PBAPS maintenance procedures and routine test procedures that provide instructions for visual inspections. The scope of these activities will be enhanced to include periodic visual inspection of the external surfaces of the CSTs, periodic visual inspection of the ECW pump casing and casing bolts, visual inspection of buried commodities whenever they are uncovered during excavation, and enhanced inspections of the refueling water storage tank. The refueling water storage tank is not in the scope of license renewal. However, inspections of the refueling water tank as enhanced will be used for determining the condition of the underside of the CSTs.

Based on PBAPS operating experience and the enhanced inspections, there is reasonable assurance that the outdoor, buried, and submerged component inspection activities will manage loss of material and cracking for the identified external surfaces of components subject to outdoor, buried and raw water external environments so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.6 Door Inspection Activities

ACTIVITY DESCRIPTION

The door inspection activities provide for managing the aging effects for hazard barrier doors that are exposed to the outdoor environment. The aging management review determined that the activities will be enhanced to include additional doors. In addition, the activities will be enhanced to include inspection for loss of material in hazard barrier doors in an outdoor environment.

The door inspection activities provide for managing the aging effects for gaskets associated with water-tight hazard barrier doors in both outdoor and sheltered environments. The inspection activities consist of condition monitoring of the gaskets associated with water-tight hazard barrier doors on a periodic basis in accordance with PBAPS procedures.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The door inspection activities provide for inspections and evaluations of hazard barrier doors exposed to the outdoor environment and of gaskets for water-tight hazard barrier doors exposed to the outdoor and sheltered environments. The PBAPS procedures governing the inspections will be enhanced to identify additional doors and to include more inspection parameters linked to loss of material in hazard barrier doors in an outdoor environment.

(2) Preventive Actions: The hazard barrier doors inspection activities are condition monitoring activities that utilize inspections to identify aging effects prior to loss of intended function. There are no preventive or mitigating attributes associated with this activity.

(3) Parameters Monitored/Inspected: Hazard barrier doors exposed to the outdoor environment are and will be inspected for evidence of loss of material due to corrosion. Gaskets associated with water-tight hazard barrier doors in an outdoor environment are inspected to detect change in material properties. Gaskets for water-tight hazard barrier doors in a sheltered environment are inspected for evidence of change in material properties and cracking.

(4) Detection of Aging Effects: Inspections for loss of material of water-tight hazard barrier doors and inspections of associated gaskets for change in material properties, and cracking are performed and results are documented. Inspections for loss of material of other hazard barrier doors exposed to the outside environment will be performed and the results documented.

(5) Monitoring and Trending: The door inspection activities periodically monitor water-tight hazard barrier doors for loss of material due to corrosion and their gaskets for change in material properties and cracking. In addition, door inspection activities will periodically monitor other hazard barrier doors for loss of material due to corrosion.

(6) Acceptance Criteria: Acceptance criteria for hazard barrier doors and gaskets associated with water-tight hazard barrier doors are provided in PBAPS procedures. If an indication or evidence of a degraded condition is found, the information is forwarded to engineering for evaluation to determine if an unacceptable visual indication of loss of material, cracking or change in material properties exists.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions, and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: A review of the operating experience for hazard barrier doors and gaskets associated with water-tight hazard barrier doors found no degraded conditions due to loss of material, change in material properties, or cracking that resulted in loss of intended function.

Corrosion on hazard barrier doors was found in a few instances, mainly on those doors with one face exposed to an outdoor environment. This condition was typically due to drainage problems that allowed the water to run toward the door rather than away from it. Corrective actions were taken to eliminate the drainage problem and door degradation prior to loss of intended function.

There were a few instances of watertight door gasket replacements. The cause, in most cases, was manmade. Plant documentation cited a few instances of debris within the gasket folds preventing door closure. Debris was removed and gaskets inspected with no detrimental effects observed.

SUMMARY

The door inspection activities provide for managing the aging effects for hazard barrier doors that are exposed to the outdoor environment and for managing the aging effects for gaskets associated with water-tight hazard barrier doors in both outdoor and sheltered environments.

Based on the PBAPS operating experience there is reasonable assurance that the door inspection activities will continue to adequately manage the aging effects on hazard barrier doors in an outdoor environment and on gaskets associated with water-tight hazard barrier doors in outdoor and in sheltered environments so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.7 Reactor Pressure Vessel and Internals ISI Program

ACTIVITY DESCRIPTION

The BWR Vessels and Internals Project (BWRVIP) guidelines are implemented through the reactor pressure vessel and internals ISI program. The reactor pressure vessel and internals ISI program is that part of the PBAPS ISI program that provides for condition monitoring of the reactor vessel and internals using guidance provided by the BWRVIP and the BWR Owners Group alternate BWR feedwater nozzle inspection requirements.

The PBAPS ISI program complies with requirements of 1989 Edition of the ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", and is implemented through a PBAPS specification. The PBAPS ISI program has been augmented to include various additional requirements, including those from the BWRVIP guidelines and the BWR Owners Group (BWROG) alternative to NUREG-0619 augmented inspection of feedwater nozzles for GL 81-11 thermal cycle cracking.

The BWRVIP program is an industry developed effort based on over 20 years of service and inspection experience and is focused on detecting evidence of component degradation well in advance of significant degradation. The BWRVIP inspection and evaluation reports for reactor pressure vessel and internals components were submitted to the NRC for review and approval. These inspection and evaluations reports address both the current and license renewal periods.

The BWRVIP program was reviewed for its applicability to PBAPS design, construction, and operating experience. The review determined that reactor pressure vessel and internals components, including the materials of construction, are addressed by the BWRVIP inspection and evaluation reports. PBAPS operating parameters, including temperature, pressure, and water chemistry, are consistent with those used for the development of the inspection and evaluation reports. The reactor vessel and internals components that require aging management review are covered by the BWRVIP inspection and evaluation reports. The BWRVIP inspection and evaluation reports cover the design of PBAPS reactor pressure vessel and internals components. Therefore, it was concluded that the BWRVIP inspection and evaluation reports bound PBAPS design and operation.

The reactor pressure vessel and internals ISI program employs the BWRVIP program criteria documented in the final NRC safety evaluation reports except where specific exception has been identified to the NRC.

The aging management review determined that the reactor pressure vessel and internals ISI program will be enhanced to assure that the inspections are consistent with BWRVIP program criteria and the NRC safety evaluation reports.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The BWRVIP documents as implemented by the reactor pressure vessel and internals ISI program provide for examinations of reactor pressure vessel components and internals, for managing the aging effects of cracking and loss of material.

(2) Preventive Actions: The BWRVIP program and the reactor pressure vessel and internals ISI program consists of condition monitoring activities that utilize early detection, evaluation and corrective actions that address degradation of reactor pressure vessel components and internals before loss of intended function. No preventive or mitigating attributes are associated with these activities.

(3) Parameters Monitored/Inspected: The BWRVIP guidelines documents reviewed the function of each reactor pressure vessel and internals components. For those that could impact safety, the BWRVIP guidelines considered the mechanisms that might cause degradation of reactor pressure vessel and internals components and developed an inspection program that would enable degradation to be detected and evaluated before the components intended function is adversely affected. Details regarding inspection and evaluation are contained within the reactor pressure vessel and internals component-specific BWRVIP inspection and evaluation guidelines document. Additionally, the program provides for visual inspections of the top head for loss of material.

(4) Detection of Aging Effects: Reactor pressure vessel components and internals are inspected using ultrasonic, visual, and surface examinations as appropriate. The methods and the frequency of examination will be consistent with the applicable BWRVIP inspection and evaluation documents, and the BWROG "Alternate BWR Feedwater Nozzle Inspection Requirements", as incorporated in the ISI program specification.

(5) Monitoring and Trending: The reactor pressure vessel ISI program provides for monitoring for the presence of aging degradation per the guidance provided in the ASME Section XI schedules, the BWRVIP inspection and evaluation documents, and BWROG "Alternate BWR Feedwater Nozzle Inspection Requirements". The frequency of examination, as specified within these documents, varies for each component. The frequency is based on the component's design, flaw tolerance, susceptibility to degradation, and the method of examination used. Documentation that facilitates comparison with previous and subsequent inspection results is maintained.

(6) ***Acceptance Criteria:*** BWRVIP inspection and evaluation documents provide the basis for reactor vessel and internals inspection requirements, acceptance criteria, and corrective actions. Any degradation in reactor pressure vessel components is evaluated in accordance with Section XI required inspections. In addition, the BWROG "Alternate BWR Feedwater Nozzle Inspection Requirements" provide additional bases for acceptance criteria contained in the ISI program specification. BWRVIP inspection and evaluation documents applicable to PBAPS reactor pressure vessel and internals components are as follows:

Reactor Pressure Vessel And Internals BWRVIP Document Applicability

Reactor Pressure Vessel Components	**Reference**
Reactor pressure vessel components	BWRVIP-74
Vessel shells	BWRVIP-05
Shroud support attachments	BWRVIP-38
Nozzle safe ends	BWRVIP-75
Core support plate	BWRVIP-25
Core ΔP / SLC nozzle	BWRVIP-27
Core spray attachments	BWRVIP-48
Jet pump riser brace attachments	BWRVIP-48
Other attachments	BWRVIP-48
CRDH stub tubes	BWRVIP-47
ICM Housing penetrations	BWRVIP-47
Instrument penetrations	BWRVIP-49
Reactor Internals Components	
Shroud support	BWRVIP-38
Shroud	BWRVIP-76
Core support plate	BWRVIP-25
Core ΔP / SLC line	BWRVIP-27
Access hole covers	(Note 1)
Top guide	BWRVIP-26
Core spray lines	BWRVIP-18
Core spray spargers	BWRVIP-18
Jet pump assembly	BWRVIP-41
CRDH stub tubes	BWRVIP-47
CRDH guide tubes	BWRVIP-47
In-core housing guide tubes, LPRM & WRNMS dry tubes	BWRVIP-47
Note 1. GE SIL 462 for Unit 2 only.	

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

***(10) Operating Experience*:** A review of operating experience at PBAPS was conducted on degradations in water systems. The degradations mirrored that of the industry, in that most were attributed to cracking. The PBAPS reactor pressure vessel and internals ISI program provides for early detection, evaluation and corrective actions that are based on industry practice and experience, and are considered adequate to address degradation of reactor pressure vessel components and internals prior to loss of intended function.

SUMMARY

The reactor pressure vessel and internals ISI program activities manage cracking and loss of material for the reactor vessel and internals using guidance provided by the BWRVIP and the BWR Owners Group alternate BWR feedwater nozzle inspection requirements. These activities are implemented through the PBAPS ISI program specification. They utilize early detection, evaluation and corrective actions that address degradation of reactor pressure vessel components and internals.

Based on the use of industry guidelines and PBAPS operating experience, there is reasonable assurance that the PBAPS reactor pressure vessel and internals ISI program will continue to adequately manage the identified aging effects for the reactor vessel and internals to maintain the intended functions consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) GE NE-523-A71-0594, BWROG "Alternate BWR Feedwater Nozzle Inspection Requirements," Rev. 1, May 2000.

(2) NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, November 1980.

(3) NRC Generic Letter, GL 81-11, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG-0619)," U.S. Nuclear Regulatory Commission, 02/29/1981.

(4) ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Plant Components," American Society of Mechanical Engineers, New York, NY, 1989.

(5) "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," EPRI, Palo Alto, CA, September 1995 (EPRI Report TR-105697).

(6) "BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines (BWRVIP-27)," EPRI, Palo Alto, CA, April 1997, (EPRI Report TR-107286).

(7) "BWR Vessel and Internals Project, Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI, Palo Alto, CA, September 1997, (EPRI Report TR-108823).

(8) "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)," EPRI, Palo Alto, CA, December 1997, (EPRI Report TR-108727).

(9) "BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Evaluation Guidelines (BWRVIP-48)," EPRI, Palo Alto CA, February 1998, (EPRI Report TR-108724).

(10) "BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)," EPRI, Palo Alto, CA, March 1998 (EPRI Report TR-108695).

(12) "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Evaluation Guidelines (BWRVIP-74)," EPRI, Palo Alto, CA, September 1999, (EPRI Report TR-113596).

(13) "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," EPRI, San Jose, CA, October 1999, (EPRI Report TR-113932).

(14) "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report TR-106740, July 1996.

(15) "BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25)," EPRI Report TR-107284, December 1996.

(16) "BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26), " EPRI Report TR-107285, December 1996.

(17) "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41), " EPRI Report TR-108728, October 1997.

(18) "BWR Core Shrouds Inspection and Flaw Evaluation Guidelines (BWRVIP-76), " EPRI Report TR-114232, November 1999.

(19) General Electric Service Information Letter, SIL 462.

B.2.8 Generic Letter 89-13 Activities

ACTIVITY DESCRIPTION

The GL 89-13 activities provide for management of loss of material, cracking, flow blockage, and reduction of heat transfer aging effects in cooling water piping and components that are tested and inspected in accordance with the guidelines of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". The GL 89-13 activities include both condition monitoring and mitigating activities for managing aging effects in HPSW, ESW, and the ECW systems and in other systems' components using raw water as a cooling medium. System and component testing, visual inspections, UT, and biocide treatments are conducted to ensure that aging effects are managed such that system and component intended functions are maintained. The aging management review determined that several component maintenance procedures will be enhanced to require inspection for specific signs of degradation including corrosion, excessive wear, cracks and Asiatic clams. Also, additional piping locations will be added to the UT inspection program.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The GL 89-13 activities provide for managing loss of material, cracking, flow blockage, and reduction of heat transfer in components exposed to raw cooling water at PBAPS. These components include piping, piping specialties, pump casings, and valve bodies in HPSW, ESW, and ECW systems. RHR heat exchangers, HPSW and CS pump motor oil coolers, CS, HPCI, RCIC and RHR pump room cooling coils, and EDG jacket, air, and lube oil coolers are also included.

(2) Preventive Actions: The GL 89-13 activities provide for condition and performance monitoring of systems and components and utilize periodic biocide treatments and flushing of infrequently used systems to mitigate flow blockage aging effects due to biofouling.

(3) Parameters Monitored/Inspected: The GL 89-13 activities inspect and test to detect aging degradation in the HPSW, ESW, and ECW systems components and heat exchangers and coolers using raw water as the cooling medium. Procedures and work orders direct the inspection of components, including visual inspection for corrosion and cracking, and UT of system piping to detect wall thinning. System component performance testing for acceptable flowrates, pressures, and heat transfer and visual inspections for system and component fouling and silting are used to identify flow blockage and reduction of heat transfer aging effects.

Several component maintenance procedures will be enhanced to require inspection for specific signs of degradation including corrosion, excessive wear, cracks and Asiatic clams. Also, additional piping locations will be inspected for loss of material.

(4) Detection of Aging Effects: GL 89-13 activities provide for detection of aging effects prior to loss of intended functions. Loss of material and cracking is detected through component visual inspections. Loss of material in piping is detected through UT. System and component flow blockage and reduction of heat transfer are detected using a combination of system and component performance testing and component visual inspections during disassembly.

(5) Monitoring and Trending: System performance tests, piping UT, and periodic component visual inspections of pump motor oil cooling water loops and of pump process check valves provide for timely detection of loss of material, cracking and flow blockage. Other system components, which are primarily pumps and valves, are visually inspected for loss of material, cracking, and flow blockage during component maintenance. Heat exchanger performance and flow testing varies from once every six weeks to once every 48 months. Inspections and non-destructive testing of heat exchangers are used to determine the extent of biofouling, condition of surface coating, magnitude of localized pitting, and evidence of MIC.

(6) Acceptance Criteria: Engineering evaluations of identified aging degradation, including loss of material, cracking, and flow blockage aging effects, are used to confirm the component's ability to perform its intended function. Semi-annual biocide injections into the ESW and HPSW systems are performed per chemistry guidelines regarding concentration and treatment durations. Cooling water component visual inspections evaluate the presence of corrosion, pitting, erosion, MIC or other abnormalities. Heat exchanger testing measures the cooling flow rates and determines the heat removal rates and compares them with system requirements specified in plant procedures.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The PBAPS GL 89-13 activities implement the inspection and testing recommendations of NRC Generic Letter 89-13. Prior to issuance of GL 89-13, corrosion induced leakage and reduced system performance due to flow blockage had occurred. System modifications were required to replace and repair piping leaks and clean fouled heat exchangers, primarily in the ESW system. The ESW system operating configuration had relied on the use of normally closed process valves to various coolers, which would open to provide the required flow to meet plant operating conditions. These stagnant legs corroded and required replacement. Since the replacement, the system operational configuration was revised to maintain flow through the coolers to reduce corrosion and flow blockage. Also, GL 89-13 testing and inspections have been implemented.

GL 89-13 inspection activities have detected and evaluated the presence of corrosion, silting and clams. The system and component corrective actions were implemented prior to loss of system function. Existing GL 89-13 activities adequately manage the aging effects of loss of material, cracking, flow blockage, and reduction of heat transfer in components exposed to raw cooling water at PBAPS.

SUMMARY

The GL 89-13 activities manage loss of material, cracking, flow blockage, and reduction of heat transfer aging effects in cooling water piping and components that are tested and inspected in accordance with the guidelines of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". The GL 89-13 activities include both condition monitoring and mitigating activities for managing aging effects in HPSW, ESW, and the ECW systems and in other systems' components using raw water as a cooling medium. System and component testing, visual inspections, UT, and biocide treatments are conducted to ensure that aging effects are managed such that system and component intended functions are maintained.

Based on PBAPS operating experience, there is reasonable assurance that the GL 89-13 activities will adequately manage loss of material, cracking, flow blockage, and reduction of heat transfer aging effects in cooling water piping and components, that are tested and inspected in accordance with the guidelines of NRC Generic Letter 89-13, to maintain their intended functions, consistent with the current licensing basis, through the period of extended operation.

REFERENCES

(1) NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment
(2) NRC Generic Letter 89-13, Supplement 1, Service Water System Problems Affecting Safety-Related Equipment

B.2.9 Fire Protection Activities

ACTIVITY DESCRIPTION

The fire protection activities provide for inspections, monitoring, and performance testing of fire protection systems and components to detect aging effects prior to loss of intended function. Degradation of fire protection systems and components due to corrosion buildup, biofouling, and silting are detected by performance testing based on NFPA 24 standards. Periodic and maintenance inspections detect corrosion, fouling, and cracking in system components due to internal and external environment aging effects and detect aging effects in fire barriers. Monitoring of system pressure detects system leakage due to both internal and external aging effects. The aging management review determined that the scope of components covered by these activities will be enhanced to provide added assurance of aging management. In addition, a one-time test will be conducted to detect loss of material due to selective leaching.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: Existing fire protection activities provide for inspections, system monitoring, and/or system performance tests of:

- fire protection system pumps, piping, sprinklers, and valves,
- diesel driven fire pump fuel oil system pumps, valves, piping and tubing,
- buried fire main piping and valves,
- outdoor fire hydrants, hose connections and hose station block valves, and
- fire barrier penetrations seals, fire barrier doors, and fire wraps exposed to sheltered and outdoor environments.

The scope of fire protection activities will be enhanced to:

- require additional inspection requirements for deluge valves in the power block sprinkler systems,
- perform functional tests of sprinkler heads that have been in service for 50 years,
- inspect diesel driven fire pump exhaust systems,
- inspect diesel driven fire pump fuel oil system flexible hoses,
- inspect fire doors for loss of material and,
- perform a one-time test of a cast iron fire protection component.

(2) Preventive Actions: The fire protection activities provide system monitoring, performance testing, and inspections to identify aging effects prior to loss of intended function. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The existing fire protection activities provide for:

- visual inspections of the fire protection system piping, sprinklers, and valves to detect loss of material, cracking, and flow blockage,
- visual inspection of fire pumps for loss of material and flow blockage during corrective maintenance activities,
- visual inspections of the diesel driven fire pump fuel oil system pumps, valves, piping and tubing to detect loss of material and cracking,
- monitoring of fire protection system pressure to detect leakage of buried fire main piping and valves,
- flow tests to detect fire protection system blockage and component degradation in buried fire main piping and valves, outdoor fire hydrants, hose connections, and hose station block valves, and
- visual inspections of fire barrier penetrations seals, fire barrier doors, and fire wraps to detect changes in material properties, cracking, delamination, separation, and loss of material.

Fire protection activities will be enhanced to include:

- power block deluge valve visual inspection requirements to include examinations for loss of material, cracking. and flow blockage,
- functional testing for flow blockage of sprinkler heads that have been in service for 50 years,
- visual inspections to detect loss of material of the diesel driven fire pump exhaust system,
- visual inspections to detect a change in material properties of the diesel driven fire pump fuel oil system flexible hoses,
- visual inspections of fire doors for loss of material,
- testing of a cast iron fire protection component to detect loss of material due to selective leaching.

(4) Detection of Aging Effects: The existing fire protection activities provide for:

- periodic visual inspections of the fire protection system piping, sprinklers, and valves that will detect loss of material, cracking, and flow blockage prior to loss of intended function,
- visual inspection of fire pumps for loss of material and flow blockage during corrective maintenance activities,
- periodic visual inspections of the diesel driven fire pump fuel oil system pumps, valves, piping and tubing that will detect loss of material and cracking prior to loss of intended function,
- continuous monitoring of fire protection system pressure that will detect pressure boundary leakage of buried fire main piping and valves prior to loss of intended function,
- periodic flow tests that will detect fire protection system blockage and

component degradation in buried fire main piping and valves, outdoor fire hydrants, hose connections, and hose station block valves prior to loss of intended function and,
- periodic visual inspections of fire barriers that will detect loss of material in fire doors, and changes in material properties, cracking, delamination, separation and loss of material in fire barrier penetrations and fire wraps prior to loss of intended functions.

Fire protection activities will be enhanced to include:
- periodic visual inspection of power block deluge valves to detect loss of material, cracking and flow blockage prior to loss of intended function,
- functional testing of sprinkler heads that have been in service for 50 years to detect flow blockage,
- periodic visual inspections of the diesel driven fire pump exhaust system to detect loss of material prior to loss of intended function,
- visual inspections of the diesel driven fire pump fuel oil system flexible hoses to detect a change in material properties prior to loss of intended function, and
- added specificity for detection of loss of material in requirements for visual inspection of fire doors, and
- a one-time test of cast iron fire protection component to detect loss of material due to selective leaching.

(5) Monitoring and Trending: Existing fire protection activities provide for the following monitoring and trending activities:
- sprinkler systems are functionally tested for flow blockage on a periodic basis,
- fire main flow testing, and hydrant flushes and inspections, are performed on a periodic basis,
- the diesel driven fire pump fuel oil system is visually examined for loss of material and cracking on a periodic basis,
- fire main pressure is continuously monitored for leakage,
- specified sample quantities of fire barrier penetration seals are inspected every 24 months with the entire population being inspected every 16 years for change in material properties, cracking, delamination, and separation, and
- fire wraps on structural steel and on electrical raceways are periodically visually inspected for change in material properties and loss of material.

Enhancements to fire protection activities will provide for the following monitoring and trending activities:
- sprinkler system deluge control valves will be visually inspected for loss of material, cracking, and flow blockage following sprinkler system testing,
- a representative sample of sprinkler heads that have been in service for 50 years will be functionally tested for flow blockage and verification of proper

operation,
- the diesel driven fire pump exhaust system will be visually inspected for loss of material on a periodic basis,
- diesel driven fire pump fuel oil system flexible hoses will be visually examined for a change in material properties on a periodic basis,
- fire barrier doors will be visually inspected for loss of material on a periodic basis, and
- if the one-time test yields unfavorable results, the scope will be expanded to other components, based upon engineering evaluations.

Fire protection testing and inspections are performed in accordance with controlled PBAPS procedures. Any degradation identified during testing and component inspections is evaluated in accordance with procedural requirements. When applicable, trending of findings is performed to determine potential long term impact.

(6) Acceptance Criteria: Tests and inspections for flow blockage, loss of material, cracking, change in material properties, and cracking, delamination, and separation aging effects are conducted in accordance with approved PBAPS procedures. These procedures contain specific acceptance criteria to confirm the systems ability to maintain required system pressures and flow rates and specific acceptance criteria for components and fire barriers to confirm their functionality. The diesel driven fire pump engine manufacturer's representative is present during engine inspections and provides standards to ensure that inspections are properly performed and that the material condition of the exhaust and fuel oil system components is acceptable.

Acceptance criteria for fire barrier doors require that there be no visual indication of corrosion. Acceptance criteria for fire barrier penetrations seals and fire wraps require that they exhibit no change in material properties, cracking, delamination, separation and loss of material.

Acceptance criteria will be based upon component material specifications.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Buried cast iron components have typically demonstrated reliable performance in commercial and industrial applications for long operational periods. At PBAPS, repairs and replacements of several hydrants, fire pumps, and indoor piping have been required due to internal corrosion and wear. The presence of corrosion, silting, and clams have been noted during plant work order inspections. Modifications and work orders have repaired and replaced degraded fire barrier penetrations and fire barrier doors. Corrective actions were implemented prior to loss of system or barrier functions. The diesel driven fire pump fuel oil system has experienced minor leakage events that were detected and corrected in a timely manner. There have been no age related component failures resulting in a loss of function for the components covered by this aging management activity.

SUMMARY

The fire protection activities provide for inspections, monitoring, and performance testing of fire protection systems and components to detect aging effects prior to loss of intended function. Degradation of fire protection systems and components due to corrosion buildup, biofouling, and silting are detected by performance testing based on NFPA 24 standards. Periodic and maintenance inspections detect corrosion, fouling, and cracking in system components due to internal and external environment aging effects, and detect aging effects in fire barriers. Monitoring of system pressure detects system leakage due to both internal and external aging effects.

Based on industry and PBAPS experience, there is reasonable assurance that the fire protection activities as enhanced will adequately manage the internal and external environment aging effects on the fire protection system components and barriers so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) NFPA 24, “Standard for Outside Protection”, 1970

B.2.10 HPCI and RCIC Turbine Inspection Activities

ACTIVITY DESCRIPTION

The HPCI and RCIC turbine inspection activities provide for aging management of the HPCI and RCIC turbine casings that are exposed to a wetted gas environment. The HPCI turbine inspection activities additionally provide for condition monitoring of components exposed to a lubricating oil environment. The inspection activities perform assessments of components for loss of material aging effects. A PBAPS procedure will be enhanced to inspect the HPCI lubricating oil system flexible hoses for a change in material properties. The HPCI and the RCIC turbine inspection activities are performed periodically in connection with turbine maintenance in accordance with plant procedures.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The HPCI and RCIC turbine inspection activities focus on managing loss of material and change in material properties aging effects by the performance of periodic inspections of the turbine casings and HPCI lubricating oil system tank internals and flexible hoses.

(2) Preventive Actions: The HPCI and RCIC turbine inspection activities provide inspection methods to identify aging effects. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The HPCI and RCIC turbine inspection activities consist of visual inspections of the turbine casings and the HPCI lubricating oil tank internals for evidence of loss of material. These activities will also be enhanced to inspect the HPCI lubricating oil system flexible hoses for a change in material properties.

(4) Detection of Aging Effects: Visual inspections for evidence of loss of material are conducted in accordance with an existing PBAPS procedure. This procedure will be enhanced to perform a visual inspection of HPCI lubricating oil system flexible hoses for a change in material properties. Loss of material and change in material properties aging effects are identified and corrected prior to a loss of intended function.

(5) Monitoring and Trending: Visual examinations are conducted on a periodic basis. The examinations monitor the turbine casings, HPCI lubricating oil storage tank, and HPCI lubricating oil system flexible hoses for evidence of aging degradation.

(6) Acceptance Criteria: Examinations for pitting of turbine casings are conducted in accordance with approved PBAPS procedures. Engineering evaluations of identified turbine casing pitting are performed and appropriate corrective actions determined. Flexible hoses will be examined in accordance with approved PBAPS procedures and replaced when abnormal conditions are identified. The results of the examinations are documented.

HPCI lubricating oil tank internals are inspected for corrosion and scaling. Engineering evaluations of identified loss of material are performed and appropriate corrective actions determined.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: A review of the operating experience for PBAPS found that there have been no aging related turbine casing failures resulting in a loss of intended function of the HPCI or RCIC turbines. Minor HPCI lubricating oil system leakage events have been detected and corrected in a timely manner. There have been no HPCI lubricating oil age related component failures resulting in a loss of function.

SUMMARY

The HPCI and RCIC turbine inspection activities consist of visual inspections of the turbine casings and the HPCI lubricating oil tank internals for evidence of loss of material, and will be enhanced to inspect the HPCI lubricating oil system flexible hoses for a change in material properties. Based on PBAPS operating experience, there is reasonable assurance that the HPCI and RCIC turbine inspection activities will adequately manage the identified aging effects for the components so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.11 Susquehanna Substation Wooden Pole Inspection Activity

ACTIVITY DESCRIPTION

The Susquehanna Substation wooden pole inspection activity manages the aging effects of loss of material and change in material properties for a wooden pole at the Susquehanna Substation. This pole provides the structural support for the conductors tying the substation to the submarine cable which is used to transmit the alternate AC power for PBAPS from the Conowingo Hydroelectric Plant in compliance with the requirements of 10 CFR 50.63 for coping with station blackout. The existing process will be enhanced to ensure the inspection activity will be performed every ten years in accordance with a PBAPS procedure.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The Susquehanna Substation wooden pole inspection activity applies to the wooden pole adjacent to the Susquehanna Substation.

(2) Preventive Actions: The Susquehanna Substation wooden pole inspection activity is a condition monitoring activity. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The wooden pole is inspected for loss of material due to ant, insect, and moisture damage, and for change in material properties due to moisture damage.

(4) Detection of Aging Effects: Inspection on a ten-year interval by a qualified inspector will assure that aging effects are detected prior to loss of intended function.

(5) Monitoring and Trending: Condition monitoring for loss of material and change in material properties are provided in the corporate specification for inspection of wooden poles.

(6) Acceptance Criteria: The acceptance criteria for the inspection are provided in the corporate specification for inspection of wooden poles.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: Corporate experience shows that inspections of wooden poles once every ten years is adequate to detect aging degradation prior to loss of intended function.

SUMMARY

The Susquehanna Substation wooden pole inspection activity inspects the pole that provides the structural support for the conductors tying the substation to the submarine cable.

Based on corporate experience, there is reasonable assurance that the Susquehanna Substation wooden pole inspection activity will manage loss of material and change in material properties of the pole so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.2.12 Heat Exchanger Inspection Activities

ACTIVITY DESCRIPTION

The heat exchanger inspection activities provide for periodic component visual inspections and component cleaning of heat exchangers and coolers that are outside the scope of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". These activities include condition monitoring actions for managing loss of material, cracking, and reduction of heat transfer effects for heat exchangers and coolers in reactor grade water environment.

The aging management review has determined that the aging management of loss of material and cracking of the HPCI gland seal condenser will be enhanced by periodic inspection of the HPCI gland seal condenser tube side internals.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The heat exchanger inspection activities provide for aging management for the HPCI gland seal condenser, the HPCI turbine lube oil cooler, and the RCIC turbine lube oil cooler through the cleaning and inspection of the heat exchangers on the water side. The scope of the activities will be enhanced to include periodic inspection of the HPCI gland seal condenser tube side internals.

(2) Preventive Actions: The heat exchanger inspection activities detect loss of material, cracking, and reduction of heat transfer aging effects prior to loss of intended function. There are no preventive or mitigating attributes associated with these activities.

(3) Parameters Monitored/Inspected: The heat exchanger visual inspections are performed in accordance with PBAPS procedures to identify degradation associated with loss of material, cracking, and reduction of heat transfer aging effects.

(4) Detection of Aging Effects: Loss of material and cracking degradation are detected through component surface visual inspections of the HPCI and RCIC turbine lube oil coolers on the water side. The existing maintenance procedures for the HPCI gland seal condenser will be enhanced to include periodic inspection of the condenser tube side internals to provide assurance of aging management for loss of material and cracking of the HPCI Gland Seal Condenser.

During disassembly, visual inspection for fouling would identify conditions, which could result in the reduction of heat transfer.

(5) Monitoring and Trending: The periodic component visual inspections and cleaning are conducted as part of HPCI and RCIC turbine inspections, and provide for timely detection of loss of material, cracking, and reduction of heat transfer prior to loss of intended function.

(6) Acceptance Criteria: Engineering evaluations of identified aging degradation including loss of material, cracking, flow blockage, and loss of heat transfer aging effects are used to confirm the ability of the component to perform its intended functions.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions, and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The heat exchanger inspection activities implement inspection and cleaning of heat exchangers. The PBAPS operating experience review identified no losses of pressure boundary integrity or heat transfer capability for these components as a result of aging degradation.

<ins>SUMMARY</ins>

The heat exchanger inspection activities provide for periodic component visual inspections and component cleaning of heat exchangers and coolers that are outside the scope of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment". These activities include condition monitoring actions for managing loss of material, cracking, and reduction of heat transfer for heat exchangers and coolers in reactor grade water environment.

Based on PBAPS operating experience, there is reasonable assurance that the heat exchanger inspection activities will continue to manage loss of material, cracking, and reduction of heat transfer for heat exchangers and coolers in reactor grade water environment so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.3 NEW AGING MANAGEMENT ACTIVITIES

B.3.1 Torus Piping Inspection Activities

ACTIVITY DESCRIPTION

Torus piping inspection activities will consist of a one-time inspection of selected piping to verify the integrity of carbon steel piping located at the water-gas interface in the torus compartment of the primary containment. The aging management review determined that it would be prudent to conduct these activities to assure there is no unacceptable loss of material. The activities will be based upon the guidance provided in ASME Section V, 1989 Edition and will be implemented through a PBAPS procedure. The results of this inspection will bound the piping that runs above the water and is subjected to a humid but less corrosive environment of air/nitrogen.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The torus piping inspection activities will examine a susceptible location on a representative sample of carbon steel piping exposed to the water-gas interface environment in the torus to assure there is no unacceptable loss of material. The Torus Piping Inspection Activity will provide confirmation that the Main Steam SRV, HPCI Turbine, and RCIC Turbine piping discharging to the Torus is in acceptable condition. The results of this inspection will bound the Torus connected piping for the gas environments. The gas environment piping that runs above the waterline to the turbines, condensers, and SRVs is subjected to a humid but less corrosive environment of air/nitrogen.

(2) Preventive Actions: The torus piping inspection activities will be condition monitoring activities that identify loss of material aging effects. No preventive or mitigating attributes will be associated with the torus piping inspection activities.

(3) Parameters Monitored/Inspected: The torus piping inspection activities will provide for a one time inspection of wall thickness to assure there is no unacceptable loss of material that could potentially challenge the maintenance of the pressure boundary intended function of torus piping. In order to determine the condition of the carbon steel piping, located near the waterline, an ultrasonic (UT) test to measure the wall thickness at a sample location will be performed. The results of this inspection will bound the Torus connected piping for the gas environments.

(4) Detection of Aging Effects: The torus piping inspection activities will be undertaken to provide reasonable assurance that there is no loss of material at the water-gas interface that would result in loss of intended function of the carbon steel piping in the torus.

(5) Monitoring and Trending: Results of the torus piping inspection activities will be evaluated. The scope and frequency of subsequent examinations will be based on the results of the initial inspection sample.

(6) Acceptance Criteria: The torus piping inspection activities acceptance criteria will be used to ensure that there is no unacceptable loss of material of the carbon steel piping in the torus. Apparent unacceptable indications of corrosion will be evaluated by further engineering analysis and if warranted, additional inspections performed. The inspection acceptance criteria will provide assurance that the minimum wall thickness requirements for the torus piping continue to be met during the period of extended operation.

(7) Corrective Actions: Identified deviations will be evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: There has been no loss of intended function of PBAPS torus piping due to age related loss of material. The torus piping inspection activities are new, and therefore, there is no operating history associated with these activities. A review of industry experience shows no failures of torus piping at the water-gas interface.

SUMMARY

Torus piping inspection activities will consist of a one-time inspection of selected piping to verify the integrity of carbon steel piping located at the water-gas interface in the torus. The activities will be based upon the guidance provided in ASME Section V, 1989 Edition and will be implemented through a PBAPS procedure.

Based on PBAPS and industry experience and the use of industry guidance for conducting the inspection, there is reasonable assurance that the torus piping inspection activities will manage loss of material for carbon steel piping located at the water-gas interface in the torus so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) ASME Boiler and Pressure Vessel Code, 1989 Edition, Section V, "Nondestructive Examination", Subsection A, "Nondestructive Methods of Examination", Article 5, "Ultrasonic Examination Methods for Materials and Fabrication"

(2) ASME Boiler and Pressure Vessel Code, 1989 Edition, Section V, "Nondestructive Examination", Subsection B, "Documents Adopted by Section V", Article 23, "Ultrasonic Standards"

B.3.2 FSSD Cable Inspection Activity

ACTIVITY DESCRIPTION ACTIVITIES

FSSD cable inspection activities will consist of inspections of the PVC insulated fire safe shutdown (FSSD) cables located in the drywell. These cables are all MSRV discharge line thermocouple wires. This inspection will manage a change in material properties of the PVC insulation.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The FSSD cable inspection activities will include evaluation of PVC insulated fire safe shutdown (FSSD) cables that are within the scope of license renewal and are installed in the drywell.

(2) Preventive Actions: FSSD Cable inspection activities will be conducted for condition monitoring purposes. No preventive or mitigating attributes will be associated with FSSD cable inspection activities.

(3) Parameters Monitored/Inspected: The PVC insulation will be visually inspected for surface anomalies such as embrittlement, discoloration, or cracking.

(4) Detection of Aging Effects: FSSD cable inspection activities will identify anomalies in the PVC insulation surface that are precursor indications of a loss of material properties for PVC insulated cables.

(5) Monitoring and Trending: Sample size of the inspection will be identified in the inspection activity. The PVC insulated FSSD cables will be inspected once every 10 years.

(6) Acceptance Criteria: Acceptance will require that no unacceptable visual indications of insulation surface anomalies exist that would suggest that the insulation has degraded, as determined by engineering evaluation. An unacceptable indication will be defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

(7) Corrective Actions: Identified deviations will be evaluated within the PBAPS corrective action process, which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process: The PBAPS corrective action process includes:
- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: No age-related PVC insulated FSSD cable failures have occurred at PBAPS.

SUMMARY

FSSD cable inspection activities will consist of inspections of the PVC insulated fire safe shutdown (FSSD) cables located in the drywell.

Based on industry experience with cable aging and inspections, there is reasonable assurance that FSSD cable inspection activities will manage a change in material properties of the PVC insulation so that the intended functions are maintained consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

B.4 TIME-LIMITED AGING ANALYSES ACTIVITIES

B.4.1 Environmental Qualification Activities

ACTIVITY DESCRIPTION

The PBAPS environmental qualification (EQ) program maintains the qualified life of the electrical equipment important to safety within the scope of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." An aging limit (qualified life) is established for equipment within the scope of the PBAPS EQ program and an appropriate action such as replacement or refurbishment is taken prior to or at the end of the equipment qualified life so that the aging limit is not exceeded. Environmental qualification binders (EQBs) are maintained to demonstrate and document the qualified life of the equipment.

The PBAPS EQ program activities establish, demonstrate and document the level of qualification, qualified configuration, maintenance, surveillance and replacement requirements necessary to meet the requirements of 10 CFR 50.49.

The PBAPS EQ program includes maintenance of supporting documentation, such as input information, references, calculations, analyses, EQ related correspondence, qualification test reports and certifications.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The PBAPS EQ program includes certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49.

***(2) Preventive Actions**:* 10 CFR 50.49 does not require actions that prevent aging effects. The PBAPS EQ program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit), (b) refurbishment, replacement, or requalification of installed equipment prior to reaching these aging limits, and (c) where applicable, requiring specific installation, inspection, monitoring, or periodic maintenance actions to maintain equipment aging effects within the qualification.

(3) Parameters Monitored/Inspected: EQ component aging limits are not typically based on condition or performance monitoring. However, per Regulatory Guide 1.89 Rev. 1, such monitoring programs are an acceptable basis to modify aging limits. Monitoring or inspection of certain environmental, condition or equipment parameters may be used to ensure that the equipment is within its qualification or as a means to modify the qualification.

(4) Detection of Aging Effects: 10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring of aging effects may be used as a means to modify component aging limits.

(5) Monitoring and Trending: 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. EQ program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition or component parameters may be used to ensure that a component is within its qualification or as a means to modify the qualification.

(6) Acceptance Criteria: 10 CFR 50.49 acceptance criteria is that an in-service EQ component is maintained within its qualification including (a) its established aging limits and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the aging limits of each installed device. When monitoring is used to modify a component aging limit, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods.

(7 & 8) Corrective Actions & Confirmation Process: If an EQ component is found to be outside its qualification, corrective actions are implemented in accordance with the PBAPS corrective action process. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When emerging industry aging issues are identified that affect the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Confirmatory actions, as needed, are implemented as part of the PBAPS corrective action process.

(9) Administrative Controls: The PBAPS EQ program is subject to administrative controls, which require formal reviews and approvals. The PBAPS EQ program will continue to comply with 10 CFR 50.49 throughout the renewal period including development and maintenance of qualification documentation demonstrating a component will perform required functions during harsh accident conditions. The PBAPS EQ program documents identify the applicable environmental conditions for the component locations. The PBAPS EQ program qualification files are maintained in an auditable form for the duration of the installed life of the component. The PBAPS EQ program documentation is controlled under the quality assurance program.

(10) Operating Experience: The PBAPS EQ program includes consideration of operating experience to modify qualification bases and conclusions, including aging limits. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during accident conditions after experiencing the detrimental effects of in-service aging.

SUMMARY

Under 10 CFR 54.21c(1)(iii), EQ programs, which implement the requirements of 10 CFR 50.49 are viewed as aging management programs for license renewal. The PBAPS EQ program complies with all applicable regulations and manages component thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods, and provides reasonable assurance that the components within the scope of license renewal will maintain their intended function consistent with the current licensing basis for the period of extended operation.

REFERENCES

(1) Code of Federal Regulations, Title 10, Part 50, Section 49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants

(2) NRC Regulatory Guide 1.89, Rev. 1, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants, June 1984

B.4.2 Fatigue Management Activities

ACTIVITY DESCRIPTION

The fatigue management program counts fatigue stress cycles and tracks fatigue usage factors. The program will be enhanced to broaden its scope and update implementation methods, and will consist of analytical methods to determine stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors (CUFs). The program will calculate and track CUF for bounding locations in the reactor pressure vessel (RPV), RPV internals, Group I piping, and containment torus, in order to manage fatigue in these components and structures. The program will rely upon the FatiguePro cycle counting and fatigue usage factor tracking program. FatiguePro is an EPRI-licensed computerized data acquisition, recording and tracking program that is being customized for PBAPS use.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Activity: The fatigue management program will consist of automated cycle counting and fatigue CUF tracking activities that will monitor critical components of the RPV and Group I piping reactor coolant pressure boundary, reactor internals, and torus structure.

Bounding locations for monitoring these components will be determined by an evaluation of the ASME Section III fatigue analyses of the reactor vessels, analyses of the vessel internals, ASME Section III Class 1 piping fatigue analyses, and the Mark I Containment Plant Unique Analyses. Simplified fatigue analyses will be performed to establish usage factors for USAS B31.1 Group I piping.

(2) Preventive Actions: The fatigue management program will monitor component and structure conditions. It includes no preventive or mitigating activities.

(3) Parameters Monitored or Inspected: The fatigue management program will monitor plant transients that contribute to the fatigue usage for each of the monitored reactor coolant pressure boundary, reactor internals, and torus structure components included in the program scope.

(4) Detection of Aging Effects: The fatigue management program will continuously monitor plant operational events, will calculate usage factors for all monitored locations, and will compare the accumulated data to allowable values; and will thereby identify the need for any corrective actions.

This process and the associated trending will allow appropriate corrective or mitigating actions to be taken, thereby maintaining structural safety factors originally considered in the plant design basis, and thereby preventing loss of the intended function.

(5) Monitoring and Trending: The fatigue management program will monitor and trend fatigue CUF and will allow corrective measures to be implemented in time to ensure that structural margins required by codes used in the original plant design are maintained throughout the operating life of the plant. The recording and assessment frequency reasonably ensures that normal operating transients which might occur during operation will not compromise these limits. The activities also will include provisions to identify deviations from expected fatigue CUF so that appropriate corrective or mitigating actions can be taken.

(6) Acceptance Criteria: The acceptance criterion consists of maintaining the CUF below the appropriate design code limit. This acceptance criterion will ensure that original structural margins considered in the plant design are maintained throughout the period of extended operation, and will thereby prevent loss of the intended function.

(7) Corrective Actions: Identified deviations are evaluated within the PBAPS corrective action process which includes provisions for root cause determinations and corrective actions to prevent recurrence as dictated by the significance of the deviation.

(8) Confirmation Process**:** The PBAPS corrective action process includes:

- Reviews to assure that proposed actions are adequate;
- Tracking and reporting of open corrective actions; and
- For root cause determinations, reviews of corrective action effectiveness.

(9) Administrative Controls: All credited aging management activities are subject to administrative controls, which require formal reviews and approvals.

(10) Operating Experience: The fatigue monitoring program was developed by EPRI, for the industry, in response to NRC concerns that early-life operating cycles at some units had caused fatigue usage factors to accumulate faster than anticipated in the design analyses. This fatigue monitoring program was therefore designed to ensure that the code limit will not be exceeded in the remainder of the licensed life. The fatigue management program will include operating experience provisions to ensure that corrective actions and experience are incorporated into future corrections and improvements.

SUMMARY

The fatigue management activities will adequately manage fatigue of reactor pressure vessel (RPV), RPV internals, Group I piping, and containment torus components consistent with the current licensing basis for the period of extended operation.

REFERENCES

None

APPENDIX C COMMODITY GROUPS (Optional)

Appendix C - Not Used

APPENDIX D TECHNICAL SPECIFICATION CHANGES (Later, if required)