Safety Evaluation Report

with Open Items Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3

Docket Nos. 50-277 and 50-278

Exelon Generation Company, LLC (Exelon)

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September, 2002

ABSTRACT

This document is a safety evaluation report regarding the application to renew the operating licenses for Peach Bottom Atomic Power Station, Units 2 and 3. The application was filed by the Exelon Generation Company LLC, (Exelon) by letter dated July 2, 2002. The Office of Nuclear Reactor Regulation has reviewed the Peach Bottom Atomic Power Station, Units 2 and 3, license renewal application for compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” and prepared this report to document its findings.

In its submittal of July 2, 2002, the Exelon requested renewal of the Peach Bottom, Units 2 and 3, operating licenses (License Nos. DPR-44 and DPR-56, respectively), which were issued under Section 104b of the Atomic Energy Act of 1954, as amended, for a period of 20 years beyond the current license expiration dates of August 8, 2013, and July 2, 2014, respectively. The Peach Bottom Atomic Power Station is a two-unit nuclear power plant located in York County and Lancaster County in southeastern Pennsylvania. Each unit consists of a General Electric boiling-water reactor nuclear steam supply system designed to generate 3458 megawatts thermal or 1093 megawatts electric.

The NRC license renewal project manager for Peach Bottom, Units 2 and 3, is David Solorio. Mr. Solorio may be contacted by calling 301-415-1973 or by writing to the License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001.

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ACRONYMS

AAC	Alternate ac
AASHTO	American Association of State Highway and Transportation Official
ACI	American Concrete Institute
ACSR	AluminumConductor Steel Reinforced
ADS	automatic depressurization system
AMP	Aging Management Program
AMR	aging management review
ANL	Argonne National Laboratory
AO	abnormal occurrence
APCSB	Auxiliary and Power Conservation Systems Branch
ARI	alternate rod insertion
ART	anticipatory reactor trip
ASCO	American Switch Co.
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BESVS	Battery and Emergency Switchgear Ventilation Systems
BPT	Branch technical position
BWR	boiling water reactor
BWROG	boiling water reactor owners group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAC	containment atmosphere control (system)
CAD	containment atmospheric dilution (system)
CASS	cast austenitic stainless steel
CCW	closed cooling water
CDF	core damage frequency
CFR	Code of Federal Regulations
CLB	current licensing basis
CRD	control rod drive
CRDHS	control rod drive housing supports
CRL	component record list
CRVS	control room ventilation system
CST	condensate storage tank
CUF	cumulative usage factor
DBA	Design-basis accidents
DBD	design baseline document
DBE	design basis event
DGBVS	diesel generator building ventilation system
DRF	dose reduction factor
ECCS	emergency core cooling system
ECP	electrochemical potential
ECT	emergency cooling tower
ECW	emergency cooling water (system)
EDG	emergency diesel generator
EFPY	effective full-power years
EPDM	ethylene propylene dienyl monomer
EPRI	Electric Power Research Institute

EQ	environmental qualification
ESF	engineered safety feature
ESW	emergency service water (system)
FAC	flow-accelerated corrosion
FERC	Federal Energy Regulatory Commission
FMP	fatigue monitoring program
FPP	Fire Protection Program
FSAR	final safety analysis report
FSSD	fire safe shutdown
GDC	General Design Criteria
GL	generic letter
GSI	Generic Safety Issues
HEDL	Hanford Engineering and Development Laboratory
HELB	high-energy line break
HEPA	high-efficiency particulate air
HPCI	high-pressure coolant injection (system)
HPSW	high-pressure service water (system)
HVAC	heating, ventilation, and air conditioning
HWC	hydrogen water chemistry
HX	heat exchanges
I & C	instrumentation and controls
IASCC	irradiation assisted stress corrosion cracking -
ICEA	Insulated Cable Engineers Association
ICM	Instrument Control Monitor
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate test
IN	information notice
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
IPE	individual plant evaluation
IPEEE	individual plant examination of external events
ISI	inservice inspection
IST	inservice testing
LEFM	linear elastic fracture mechanics
LER	licensee event report
LLRT	local leak rate tests
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	loss of coolant accident
LPCI	low-pressure coolant injection (system)
LPRM	local power range monitor
LRA	license renewal application
LRC	Level Recorder Controller
LWR	light-water reactor
MCC	motor control center
MCRE	main control room envelope
MCRE	main control room envelope
MIC	microbiologically influenced corrosion
MOV	motor-operated valve

MR	maintenance rule
MSIV	main steam isolation valve
MSRV	main steam relief valve
NCR	nonconformance report
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufactures Associates
NFPA	National Fire Protection Association
NMCA	Noble Metals Chemical Addition
NPAR	nuclear plant aging research
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSR	non-safety related
NSSS	nuclear steam supply systems
NSW	normal service water
NUMARC	Nuclear Management and Resources Council
OE	operating experience
OFS	orificed fuel support
ORNC	Oak Ridge National Laboratory
P&ID	piping and instrumentation diagram
PBAPS	Peach Bottom Atomic Power Station
PCIS	primary containment isolation system
PECO	Philadelphia Electric Company
PLI	project level instruction
PM	preventive maintenance
P-T	pressure-temperature
PSVS	pump structure ventilation system
PUA	plant-unique analysis
PWR	pipe whip restraint
QAP	quality assurance procedure
RAI	request for additional information
RBM	rod block monitor
RCIC	reactor core isolation cooling (system)
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal (system)
RMS	radiation monitoring system
RPS	reactor protection system
RPV	reactor pressure vessel
RRS	reactor recirculation system
RTNDT	nil-ductility transition reference temperature
RVID	Reactor Vessel Integrity Database
RWM	rod worth minimizer
RWCU	reactor water cleanup
RWST	refueling water storage tank
SBLC	standby liquid control (system)
SBO	station blackout
SCC	stress corrosion cracking

SE	safety evaluation
SECY	Secretary of the Commission Office of the (NRC)
SER	safety evaluation report
SGIG	safety grade instrument gas (system)
SGTS	standby gas treatment system
SIL	Service Information Letter
SLC	standby liquid control
SOER	Significant Operating Experience Reports
SPOTMOS	suppression pool temperature monitoring system
SRM	source range monitor
SRP-LR	Standard Review Plan - license renewal
SRV	safety relief valve
SCs	structures and components
SSCs	systems, structures, and components
SV	safety valve
SSWP	Susquehanna Substation Wooden Pole
TID	total integrated dose
TLAAs	time-limited aging analyses
TTA	thenyltrifluoroacetone
UFSAR	updated final safety analysis report
UL	Underwriters Laboratories, Inc.
USAS	United States of America Standards
USE	upper-shelf energy
USI	unresolved safety issue
WRNM	wide range neutron monitor
XLPE	cross-linked polyethylene
XLPO	cross-linked plyolifin

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This document is a safety evaluation report (SER) on the application to renew the operating licenses for Peach Bottom Atomic Power Station, Units 2 and 3, filed by Exelon Generation Company, LLC, (Exelon) (hereafter referred to as Exelon or the applicant).

By letter dated July 2, 2001, Exelon submitted its application to the U.S. Nuclear Regulatory Commission (NRC) for renewal of the operating licenses for Peach Bottom Atomic Power Station, Units 2 and 3, for an additional 20 years. The NRC staff reviewed the Peach Bottom license renewal application (LRA) for compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings. The NRC's license renewal project manager for Peach Bottom Atomic Power Station, Units 2 and 3, is David Solorio. Mr. Solorio may be contacted by calling 301-415-1973 or by writing to the License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001.

In its application, Exelon requested renewal of the operating licenses issued under Section 104b of the Atomic Energy Act of 1954, as amended, for Peach Bottom Atomic Power Station, Units 2 and 3 (License Nos. DPR-44 and DPR-56, respectively) for a period of 20 years beyond the current license expiration dates of August 8, 2013 and July 2, 2014, respectively. The Peach Bottom Atomic Power Station is a two-unit boiling water reactor located in York County and Lancaster County in southeastern Pennsylvania. Each unit consists of a General Electric boiling-water reactor nuclear steam supply system designed to generate 3458 megawatts thermal or 1093 megawatts electric. Details concerning the plant and the site are found in the updated final safety analysis report (UFSAR) for each unit.

The license renewal process proceeds along two tracks: a technical review of safety issues and an environmental review. The requirements for these two reviews are stated in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review is based on Exelon's application for license renewal and on the applicant's answers to requests for additional information (RAIs) from the NRC staff. Exelon has also supplemented its answers to the RAIs in meetings and docketed correspondence. The public can review the LRA and all pertinent information and material, including the UFSARs, at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852-2738. In addition, the Peach Bottom Atomic Power Station, Units 2 and 3, LRA and significant information and material related to the license renewal review are available on the NRC's Website at www.nrc.gov. through the NRC's electronic reading room.

This SER summarizes the findings of the staff's safety review of the Peach Bottom Atomic Power Station, Units 2 and 3, and describes the technical details considered in evaluating the safety aspects of its proposed operation for an additional 20 years beyond the term of the current operating licenses. The staff reviewed the LRA in accordance with the NRC regulations and the guidance presented in the NRC "Standard Review Plan (SRP) for the Review of License Renewal Applications for Nuclear Power Plants," dated July 2001.

1.2 License Renewal Background

Pursuant to the Atomic Energy Act of 1954, as amended, and NRC regulations, operating licenses for commercial power reactors are issued for 40 years. These licenses can be renewed for up to 20 additional years. The original 40-year license term was selected on the basis of economic and antitrust considerations, not technical limitations. However, some individual plant and equipment designs may have been engineered on the basis of an expected 40-year service life.

In 1982, the NRC anticipated interest in license renewal and held a workshop on nuclear power plant aging. That led the NRC to establish a comprehensive program plan for nuclear plant aging research (NPAR). On the basis of the results of that research, a technical review group concluded that many aging phenomena are readily manageable and do not involve technical issues that would preclude extending the life of nuclear power plants.

In 1986, the NRC published a request for comment on a policy statement that would address major policy, technical, and procedural issues related to life extension for nuclear power plants.

In 1991, the NRC published the license renewal rule in 10 CFR Part 54. The NRC participated in an industry-sponsored demonstration program to apply the rule to pilot plants and develop experience to establish implementation guidance. To establish a scope of review for license renewal, the rule defined age-related degradation unique to license renewal. However, during the demonstration program, the NRC found that many aging mechanisms occur and are managed during the period of the initial license. In addition, the NRC found that the scope of the review did not allow sufficient credit for existing programs, particularly for the implementation of the Maintenance Rule, which also manages plant aging phenomena.

As a result, in 1995 the NRC amended the license renewal rule in 10 CFR Part 54. The amended rule established a regulatory process that is simpler, more stable, and more predictable than the previous license renewal rule. In particular, 10 CFR Part 54 was clarified to focus on managing the adverse effects of aging rather than on identifying all aging mechanisms. The rule changes were intended to ensure that important systems, structures, and components (SSCs) will continue to perform their intended function in the period of extended operation. In addition, the integrated plant assessment (IPA) process was clarified and simplified to be consistent with the revised focus on passive, long-lived structures and components (SCs).

In parallel with these efforts, the NRC pursued a separate rulemaking effort to amend 10 CFR Part 51 to focus the scope of the review of environmental impacts of license renewal, and fulfill, in part, the NRC's responsibilities under the National Environmental Policy Act of 1969 (NEPA).

1.2.1 Safety Review

License renewal requirements for power reactors are based on two key principles:

(1) The regulatory process is adequate to ensure that the licensing basis of all currently operating plants maintains an acceptable level of safety, with the possible exception is the detrimental effects of aging on the functionality of certain SSCs during the period of

extended operation, and a few other safety issues may arise only during the period of extended operation

(1) The plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term.

In implementing these two principles 10 CFR 54.4 defines the scope of license renewal as including those plant SSCs (a) that are safety-related, (b) whose failure could affect safety-related functions, (c) that are relied on to demonstrate compliance with the Commission's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout.

Pursuant to 10 CFR 54.21(a)(1), the applicant must review all SSCs that are within the scope of the rule to identify SCs that are subject to an aging management review (AMR). SCs that are subject to an AMR are those that perform an intended function without moving parts or without a change in configuration or properties and that are not subject to replacement based on a qualified life or specified time period. As required by 10 CFR 54.21(a), the applicant must demonstrate that the effects of aging will be managed in such a way that the intended function or functions of the SCs that are within the scope of license renewal will be maintained, consistent with the current licensing basis, for the period of extended operation.

Active equipment, however, is considered to be adequately monitored and maintained by existing programs. The detrimental effects of aging on active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance indicators, and maintenance. The surveillance and maintenance programs and activities for active equipment, as well as other aspects of maintaining the plant design and licensing basis, are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(b), within a year of submitting the LRA and at least 3 months before the scheduled completion of the NRC's review of the application each applicant is required to submit an amendment to the LRA that identifies any changes to the CLB for its facilities that materially affect the contents of the LRA, including the FSAR supplement.

Another requirement for license renewal is the identification and updating of time-limited aging analyses. During the design phase for a plant, certain assumptions are made about the initial operating term of the plant, and these assumptions are incorporated into design calculations for several of the plants SSCs. In accordance with 10 CFR 54.21(c)(1), these calculations must be shown to be valid for the period of extended operation or must be projected to the end of the period of extended operation, or the applicant must demonstrate that the effects of aging on these SSCs will be adequately managed for the period of extended operation. Pursuant to 10 CFR 54.21(c)(2), each applicant must provide a list of the exemptions granted pursuant to 10 CFR 50.12 and still in effect that are based on the TLAAs as defined in 10 CFR 54.3. Pursuant to CFR 54.21(c)(2), each applicant must also provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

Pursuant to 10 CFR 54.21(d), each application is required to include a supplement to the FSAR. This supplement must contain a summary description of the programs and activities for managing the effects of aging, and the evaluation of TLAAs for the period of extended operation.

In July 2001, the NRC issued Regulatory Guide 1.188, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating License”; NUREG-1800, “Standard Review Plan for the Review of License Renewal Application for Nuclear Power Plants” (SRP-LR); and NUREG-1801, “Generic Aging Lessons Learned (GALL) Report.” These documents describe methods acceptable to the NRC staff for implementing the license renewal rule, as well as techniques used by the NRC staff in evaluating applications for license renewals. The draft versions of these documents were issued for public comment on August 31, 2000 (64 FR 53047). The staff assessment of public comments was issued as NUREG-1739, “Analysis of Public Comments on the improved License Renewal Guidance Documents.” The regulatory guide endorsed an implementation guideline prepared by the Nuclear Energy Institute (NEI) as an acceptable method of implementing the license renewal rule. The NEI guideline is NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54–The License Renewal Rule," Revision 3 issued in April 2001. The staff used the RG1.188, along with the SRP, to review this application and to assess topical reports on license renewal issues as submitted by industry groups.

1.2.2 Environmental Review

In December 1996, the staff revised the environmental protection regulations in 10 CFR Part 51 to facilitate environmental reviews for license renewal. The staff prepared the "Generic Environmental Impact Statement [GEIS] for License Renewal of Nuclear Plants," NUREG-1437, in which it examined the possible environmental impacts associated with renewing operating licenses for nuclear power plants. For certain types of environmental impacts, the GEIS establishes generic findings applicable to all nuclear power plants. These generic findings are identified as Category 1 issues in 10 CFR Part 51, Subpart A, Appendix B. Pursuant to 10 CFR 51.53(c)(3)(i), an applicant for license renewal may incorporate these generic findings in its environmental report.

Analyses of the environmental impacts of license renewal that must be evaluated on a plant-specific basis are identified as Category 2 issues in 10 CFR Part 51, Subpart A, Appendix B. Such analyses must be included in an environmental report in accordance with 10 CFR 51.53(c)(3)(ii).

In accordance with NEPA and the requirements of 10 CFR Part 51, the NRC staff performed a plant-specific review of the environmental impacts of renewal of the Peach Bottom, Units 2 and 3, operating licenses, including a determination of whether there is new and significant information not considered in the GEIS. A public meeting was held on November 7, 2001, near the Peach Bottom site as part of the NRC's scoping process to identify environmental issues specific to the plant. The results of the environmental review and analysis process, and a preliminary recommendation on the license renewal action, were documented in the NRC's draft plant-specific Supplement 10 to the GEIS, issued June 2002.

On July 31, 2002 (during the 75-day public comment period for the draft supplement to the GEIS), another public meeting was held near the site. At this meeting, the staff described the environmental review process and answered questions from members of the public to assist them in formulating their comments on the review process and the draft supplement to the GEIS.

On the basis of (1) the analysis and findings in the "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," NUREG-1437; (2) the environmental report submitted by the applicant; (3) consultation with other Federal, State, and local agencies; (4) the staff's independent review; and (5) the staff's consideration of public comments received during the scoping period and comment period for the June 2002 draft, the staff made a preliminary recommendation in the draft Supplement 10 to NUREG-1437 that the Commission determine that the adverse environmental impacts are not so great that preserving the option of license renewal for energy planning would be unreasonable.

The final Supplement 10 to the GEIS is scheduled to be issued February 2003.

1.3 Summary of the Principal Review Matters

The requirements for renewing operating licenses for nuclear power plants are described in 10 CFR Part 54. The staff performed its technical review of the Peach Bottom Atomic Power Station, Units 2 and 3, license renewal application in accordance with Commission guidance and the requirements of 10 CFR Part 54. The standards for renewing a license are contained in 10 CFR 54.29.

In 10 CFR 54.19(a), the Commission requires a license renewal applicant to submit general information. Exelon submitted this general information in an enclosure to its July 2, 2001, application for renewed operating licenses for Peach Bottom Atomic Power Station, Units 2 and 3. The applicant supplemented this information in a letter dated August 23, 2001. The staff reviewed the enclosure and the supplemental information.

In 10 CFR 54.19(b), the Commission requires that LRAs include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The applicant stated the following in its renewal application regarding this issue:

> The current indemnity agreement for Peach Bottom Atomic Power Station, Units 2 and 3 states, in Article VII, that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the Attachment to the agreement, which is the last to expire. Item 3 of the Attachment to the indemnity agreement, as revised by Amendment No. 10, lists two license numbers, DRP-44 and DRP-56. Should the license numbers be changed upon issuance of the renewed licenses, Exelon requests that the conforming changes be made to Article VII and Item 3 of the Attachment, and to any other sections of the indemnity agreement as appropriate.

The staff will use the original license number for the renewed license. Therefore, there is no need to make conforming changes to the indemnity agreement, and the requirements of 10 CFR 54.19(b) have been met.

In 10 CFR 54.21, the Commission requires that each application for a renewed license for a nuclear facility must contain (a) an integrated plant assessment (IPA), (b) description of current licensing basis changes made during the NRC review of the application, (c) an evaluation of time-limited aging analyses (TLAAs), and (d) a final safety analysis report (FSAR) supplement.

On July 2, 2001, the applicant submitted the information required by 10 CFR 54.21(a) and (c) in the Enclosure of its LRA.

In 10 CFR 54.22, the Commission states requirements regarding technical specifications. The applicant did not request any changes to the plant technical specification in its LRA.

The staff evaluated the technical information required by 10 CFR 54.21 and 54.22 in accordance with the NRC's regulations and the guidance provided in the SRP. The staff's evaluation of this information is documented in Chapters 2, 3, and 4 of this SER.

The staff's evaluation of the environmental information required by 10 CFR 54.23 is documented in the draft plant-specific supplement to the GEIS (NUREG-1437, Supplement 10), which states the considerations related to renewing the licenses for Peach Bottom Atomic Power Station, Units 2 and 3.

1.3.1 Boiling Water Reactor Vessel Internals Project (BWRVIP) Topical Reports

In accordance with 10 CFR 54.17(e), Exelon also incorporated by reference several BWRVIP topical reports into the Peach Bottom LRA. The purpose of the topical reports is to generically demonstrate that the aging effects for reactor coolant system components are adequately managed for the period of extended operation under a renewed license. Exelon incorporated the following BWRVIP topical reports into its application:

- BWRVIP-05, "BWR RPV Shell Weld Inspection Recommendations," September 1995
- BWRVIP-18, "Core Spray Internals Inspection and Flaw Evaluation Guidelines," July 1996
- BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," October 1999
- BWRVIP-26, "Top Guide Inspection and Flaw Evaluation Guidelines," December 1996
- BWRVIP-27, "Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines," April 1997
- BWRVIP-38, "Shroud Support Inspection and Flaw Evaluation Guidelines," September 1997
- BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," October 1997
- BWRVIP-47, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines," December 1997
- BWRVIP-48, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines," March 1998

- BWRVIP-49, “Instrument Penetration Inspection and Flaw Evaluation Guidelines,” March 1998

- BWRVIP-74, “BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines,” September 1999

- BWRVIP-75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313),” October 1999

- BWRVIP-76, “BWR Core Shroud Inspection and Flaw Evaluation Guidelines,” December 1999

All the BWRVIP reports listed above have been approved by the staff with the exception of BWRVIP-76. The staff is presently reviewing the responses from the Owners Group, and is expected to issue a safety evaluation report by December 2002.

The applicant committed to follow the BWRVIP reports as approved by the staff. The staff finds this commitment to be acceptable for aging management of the systems and components addressed in the subject BWRVIP reports.

1.4 Summary of Open Items

As a result of its review of the license renewal application for the Peach Bottom Atomic Power Station Units 2 & 3, including the additional information submitted to the NRC through May 22, 2002, the staff identified the following issues that remained open at the time this report was prepared. A issue is open if Exelon has not presented a sufficient basis for its resolution. Each open item has been assigned a unique identifying number, which identifies the section in this report in which the open item is described. For example, open item 3.0-1 is discussed in Section 3.0 of this report.

Item	Description
2.3.2.7.2-1	The staff considers the applicant’s response is partially acceptable since prefilters, HEPA filters, and charcoal filters are governed by technical specification (TS) requirements or plant procedures which provide for their replacement in accordance with TS surveillance requirements or plant procedures. The staff does not agree that the demisters, fire spray nozzles, and heating coils should be excluded from AMR because of any one if these components should fail, the intended function of the filtration unit may not be accomplished. **This is Open Item 2.3.2.7.2-1.**
2.3.2.7.2-2	The applicant responded that the components identified above are part of the secondary containment as shown by the flag “SC” on drawing LR-M-391, sheets 1 and 2, Rev. A. As such, the valve bodies, ductwork, and tubing are shown in Table 2.3.2-8 in LRA Section 2.3.2.8. The staff considered the applicant’s response to the RAI acceptable since the components were subject to an AMR and were identified in Table 2.3.2-8 of the LRA. However, the applicant needs to indicate that valve bodies

include damper housings for the SGTS dampers, if any, in LRA Table 2.3.2-8. **This is part of Open Item 2.3.2.7.2-2.** The additional part of this item is discussed in Section 2.3.8.2 of this SER.

(From Section 2.3.2.8.2 of this SER) In a letter dated May 22, 2002, the applicant clarified that the components referred to by the staff as dampers in RAI 2.3.2.8-1 are actually air-operated valves. These valves are secondary containment isolation valves; their associated valve bodies are subject to an AMR and are listed in Table 2.3.2-8. Also, the applicant indicated that the test connections identified by the staff are considered to be in the ducting component group, which the applicant has included in the AMR results provided in LRA Table 2.3.2-8. The staff finds the applicant's RAI response to be acceptable, as it clarifies that the passive, long-lived components in question are subject to an AMR in accordance with 10 CFR 54.21(a)(1). However, the applicant needs to indicate that valve bodies include the damper housings for the secondary containment system dampers, if any (as shown in LRM-391), in LRA Table 2.3.2-8. **This is the other part of Open Item 2.3.2.7.2-2.**

2.3.3.8.2-1 In RAI 2.3.3.8-2, the staff stated that LRA Table 2.3.3-8 did not identify the components and their housings listed below, although these components, including their housings, support the intended function of the CRVS to comply with the requirements of the Appendix A to 10 CFR Part 50, GDC 19. These components are shown on license renewal drawing LR-M-384, sheet 1, as falling within the scope of license renewal but are not listed in Table 2.3.3-8 of the LRA. The staff requested that the applicant provide a justification for the exclusion of these components and their housings from an AMR. Housings and components excluded are:

- reheat coil 00E072, drawing LR-M-384, sheet 3, location H2
- thermowell for temperature transmitter TT00174, drawing LR-M-384, sheet 3, location H2
- louver, drawing LR-M-384, sheet 1, location D8
- preheat coil 00E068, sheet 1, at location D7
- HEPA filters OAF041, drawing LR-M-384, sheet 1, location G6, and OBF041at location F6
- HEPA filters OAF050, drawing LR-M-384, sheet 1, location G5 and OBF050 at location F5

The staff indicated that if the filter media for the components identified above were excluded on the basis that these media components are routinely replaced (consumables), the applicant should describe the plant-specific monitoring program and the specific performance standards and criteria for periodic replacement.

In a response to RAI 2.3.3.8-2, the applicant stated that heating coil enclosures (reheat and preheat coils) were inadvertently omitted from the

LRA tables. These components should be included in LRA Table 2.3.3-8 as having a pressure boundary function in a sheltered, ventilation atmosphere environment. The applicant further indicated that there is no thermowell for temperature transmitter TT00174. The temperature element is a capillary type and penetrates the ventilation duct through a bulkhead type fitting. The bulkhead fitting is considered as part of the ventilation ductwork hardware for license renewal. The louver shown on license renewal drawing LR-M-384, sheet 1, at location D8, is mounted in a wall opening at the ventilation intake and does not include any pressure boundary housing or enclosure. The applicant confirmed that heating coil enclosures are subject to an AMR and should be included in LRA Table 2.3.3-8. As stated above, the staff found the inclusion of the heating coil enclosures in Table 2.3.3-8 acceptable because they meet the requirements of 10 CFR 54.21(a)(1).

The filter media for the components identified above are short-lived and passive and are not subject to an AMR. Periodic testing and inspection programs include filter performance such that system intended functions are maintained. The filters are monitored during the annual filter train surveillance tests, including verification of acceptable maximum differential pressure. System filters are replaced as conditions warrant; therefore an AMR is not required. The staff considers the applicant's response to RAI 2.3.3.8-2 partially acceptable. However, the filter housings of the HEPA filters were excluded from the LRA Table 2.3.2-8 and the applicant failed to provide justification for this exclusion in its response. The applicant needs to include these housings in LRA Table 2.3.2-8 to indicate that they are subject to an AMR or justify their exclusion from an AMR. **This is Open Item 2.3.3.8.2-1.**

2.3.3.8.2-2 The staff considers the applicant's response to RAI 2.3.3.8-5 incomplete because the system's safety-related radiation, cooling, and toxic protection functions are required to meet Appendix A to 10 CFR Part 50, GDC 19. LRA Section 2.3.3.8 states that the control room air conditioning ventilation subsystem (of CRVS) provides ventilation for the control room during normal, abnormal, accident, and accident conditions. Also, the UFSAR subsection 10.13.4 states that the emergency cooling and ventilation system for the control room and other safety-related equipment rooms are installed in seismic Class I structure and are provided with 100% redundancy. Therefore, the applicant needs to include the CRVS subsystem components listed below within the scope of license renewal and subject to an AMR (in LRA Tables 2.3.3-8 and 3.3-8) in accordance with 10 CFR 54.4 and 10 CFR 54.21 (a)(1) or justify their exclusion:

<u>LRA Drawing LR-M-384, Sheet 2</u>

- Housings for supply fans (OAV028/OBVO28),
- Cooling coils (OAE069/OBE069)
- Ductwork and damper housings

LRA Drawing LR-M-384, Sheet 3

- Housings for two balance dampers at F7 and G7
- Housings for return air fans (OAV029/OBV020)
- Ductwork and damper housings

Additionally, if the fiter media and filter housings for the supply roll filter and bag filter (OOF038/OOF057, as shown in LRA Drawing LR-M-384, Sheet 2) were excluded on the basis that these media components are routinely replaced (i.e., they are consumables) the applicant should describe the plant specific monitoring program and the specific performance standards and critieria for periodic replacement. **This is Open Item 2.3.3.8.2-2.**

2.3.3.9.2-1

In RAI 2.3.3.9-1, the staff noted that LRA Table 2.3.3-9 does not list the heating coils and their housings 0AE073 and 0BE073 as being subject to an AMR, although these components are shown at locations F5 and C5 on license renewal drawing LR-M-399, sheet 1, as being within the scope of license renewal. The staff believes that these components provide a passive boundary function for the BESVS. Accordingly, the staff requested the applicant to provide its justification for the exclusion of the above components from Table 2.3.3-9 of the LRA. In a letter dated May 22, 2002, the applicant responded that the subject heating coils are steam heating coils that are installed inside the fan unit (0AV034, 0BV034) enclosure housing, and do not provide a passive boundary function for the BESVS. However, the fan enclosures (housings) are included in LRA Table 2.3.3-9.

The staff considers failure of a steam heating coil pressure boundary to cause steam leakage into the BESVS ventilation duct, thereby degrading HVAC unit performance. The staff believes that these heating coils do fall within the scope of license renewal and are subject to an AMR. **This is Open Item 2.3.3.9.2-1.**

2.3.3.18.2-1

In response to RAI 2.3.3.18-2, the applicant stated that the components identified by the staff are within the scope of license renewal and subject to an AMR. However, not all of the components are part of the cranes and hoists and thus not all are not listed in Table 2.3.3-18 of the LRA. Structural crane components such as bridge girders, trolley, trolley rails, crane rails, clips, and bolts are included in the component group listed in Table 2.3.3-18. Crane girders, columns, beams, base plates, and anchors are a part of the building structural steel and included in the structural steel component group listed in LRA Table 3.5-1, 3.5-2, 3.5-4, 3.5-5, 3.5-10, or 3.5-11. The applicant identified that the content of Table 2.3.3-18 is consistent with NUREG-1801, Section VII B, and the table on page VII B-3. The staff reviewed LRA Tables 3.5-1, 3.5-2, 3.5-4, 3.5-5, 3.5-10, and 3.5-11. In addition, the staff reviewed the Generic Aging Lessons Learned (GALL) Report, Section VII, Table VII B-3, to verify if the SSCs listed by the applicant in Table 2.3.3-18 as within scope are consistent with the GALL Report. On the basis of this review, the staff determined that the SSCs and their AMR results were included in the component groups in the tables identified by the applicant. However, the staff could not determine from the applicant's response how the SSCs in RAI 2.3.3.18-2 were captured within the scope of license renewal. The tables only provide the SSCs and the AMR results, but it is unclear to the staff how Section 2.3.3.18 uniquely identifies and lists these SSCs as being within the scope of license renewal. For example, the staff is unable to determine which of the component types listed in the structural steel component group in LRA Tables 2.4-1 and 3.5-1 captures the containment crane girder. **This is Open Item 2.3.3.18.2-1.**

2.3.3.19.2-1 In a letter dated May 21, 2002, the applicant responded to the RAIs. The applicant identified components of non-safety-related systems (listed above) which fall within the scope of license renewal and are subject to an AMR. However, the applicant's RAI response did not supply sufficient information to allow the staff to determine, with reasonable assurance, that all of the SSCs with the potential for non-safety to safety-related interactions had been identified and included within the scope of license renewal. The staff asked the applicant to do the following: (1) define the procedure and criteria used to determine the credibility of the spatial interactions of the hazard systems with equipment within the scope of license renewal. Identify the plant area where the potential interactions with safety-related equipment are postulated to occur; (2) explain how non-fluid-containing systems having potential spatial interaction with safety-related systems were evaluated; (3) define the criteria used to designate hazard systems; (4) describe the plant walkdown mentioned in the applicant's May 21, 2002, letter to the NRC and how the results were used to determined which non-safety-related systems, structures, and components were brought within scope; and (5) discuss the means by which information that formed the basis for the applicant's conclusions for including the non-safety-related systems within the scope will be documented, auditable, and retrievable, in accordance with 10 CFR 50.37. **This is Open Item 2.3.3.19.2-1.**

2.4.7.2-1 Based on the applicant's responce to the RAI, the staff reviewed the technical information in USFAR Section 9.2. The staff found that the USFAR Section 9.2.3, "Safety Design Basis," states that the liquid radwaste system prevents the inadvertent release of significant quantities of liquid radioactive material from the site boundary of the plant which could result in radiation exposures to the public in excess of the limits specified in 10 CFR Part 100. USFAR Section 9.2.9 states that leaks or spills from the liquid radwaste system are retained by secondary enclosures such as water-tight dikes and the water-tight dikes support the liquid radwaste system, by providing a barrier, in meeting its safety design of ensuring that a radioactive release to the public in excess of 10 CFR Part 100 limits is prevented. The applicant should include the water-tight dikes within the scope of license renewal and subject them to an AMR or justify their exclusion. **This is Open Item 2.4.7.2-1.**

3.0.3.6.2-1 In response to RAIs B.1.8-1 and B.1.8-2, the applicant stated that the ISI program is not credited with managing the aging effects of ASME Code class piping in several plant systems, including HPCI, core spray, PCIS, RCIC, and RHR. Instead, the applicant stated the aging was adequately managed by Reactor Coolant System Chemistry (B.1.2), Condensate Storage Tank Chemistry Activities (B.1.4), Closed Cooling Water Chemistry (B.1.3), or Torus Water Chemistry Activities (B.1.5), as applicable. These programs provides chemistry controls only and do not include provisions for any inspections to verify the effectiveness of the programs. Water chemistry programs are designed to mitigate aging effects and not designed to confirm that the aging effect has not occurred. Confirmation of the effectiveness of chemistry programs is needed because they may not be effective in managing aging effect particularly in low or stagnant flow areas and lead to unacceptable degradation. Therefore, it is the staff's position that the applicant should perform inspections, through either the ISI program or one-time inspections, which are credited for license renewal, to verify the effectiveness of the chemistry program credited for managing the effects of aging. **This is Open Item 3.0.3.6.2-1.**

3.0.3.11.2-1 As stated above under Scope of Program, in response to RAI 3.5-1 the applicant committed to manage loss of material, cracking, and change in material properties for all accessible concrete and masonry block structures. To be consistent with this commitment made in response to RAI 3.5-1, the applicant needs to clarify that the parameters inspected for the maintenance rule structural monitoring program will be revised to include inspection of the concrete components, which credit this program, for cracking, loss of material, and change in material properties. **This is part of Open Item 3.0.3.11.2-1.** The additional part of this open item related to the acceptance criteria is discussed below.

The above acceptance criteria are adequate to detect the aging of the component groups that originally credited this program; however, as a result of the applicant's response to RAI 3.5-1, several additional concrete components now credit the maintenance rule structural monitoring program. To be consistent with the commitment made in response to RAI 3.5-1, the applicant needs to add additional acceptance criteria for the concrete components which now credit this program. **This is the other part of Open Item 3.0.3.11.2-1.**

3.0.3.16.2-1 The staff noted that the applicant is committing to inspect the diesel-driven fire pump flexible hoses, but has not provided the kind of details regarding the inspection activities that were provided for the EDG fuel and system flexible hoses. Therefore, the staff cannot conclude that the inspection activities for the diesel-driven fire pump fuel and system flexible hoses provide adequate aging management. The applicant needs to provide information for the fire fuel and system pump flexible fuel and system hoses comparable to that provided for the EDG flexible hoses. **This is part of Open Item 3.0.3.16.2-1.** Additional parts of this open item are discussed below under detection of aging effects, monitoring and trending, and acceptance criteria.

The staff finds that the detection of aging effects as discussed above is acceptable with the exception of the visual inspection of the diesel-driven fire pump fuel oil system flexible hoses. **This is part of Open Item 3.0.3.16.2-1.**

The staff finds that the applicant's methodology will provide effective monitoring and trending of the aging effects and is therefore acceptable with the exception of the frequency of the visual inspections of the diesel-driven fire pump fuel oil system flexible hoses. **This is part of Open Item 3.0.3.16.2-1.**

The staff finds these criteria reasonable and acceptable because they will provide an effective means of detecting changes in material properties such that the effects of aging will be detected and evaluated before failure would occur with the exception of the acceptance criteria for the visual inspection of the diesel-driven fire pump fuel oil system flexible hoses. **This is part of Open Item 3.0.3.16.2-1.**

3.1.3.2.1-1 The application does not identify the aging effect of cracking due to stress corrosion cracking and cyclic loading for valve closure bolting in the reactor pressure vessel instrumentation system. Bolting that is heat treated to a high-hardness condition and exposed to a humid environment within containment could be susceptible to SCC. NUREG-1399, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," indicates that the bolting material with yield strength greater than 150 ksi is susceptible to SCC. For high-strength bolting, the effects of cyclic loading are generally seen in conjunction with SCC in causing crack initiation and growth. In RAI 3.1-1, the staff requested the applicant to take into account the above information and review industry and plant experience to assess whether these aging effects are applicable for valve closure bolting in the reactor pressure vessel instrumentation system. If such an aging effect is present, the applicant should submit an aging management program to manage cracking in valve closure bolting in the reactor pressure vessel instrumentation system. In response to RAI 3.1-1, the applicant provided the following justification for why cracking due to SCC is not considered an applicable aging effect for valve closure bolting in the reactor pressure vessel instrumentation system: PBAPS implemented changes as a result of NRC generic correspondence on bolt cracking. PBAPS has a materials control program in place, which requires an evaluation of all chemicals and consumables to minimize the potential for damage to plant equipment. These administrative controls prevent the introduction of lubricants or sealants that may damage closure bolting. PBAPS does not have a history of closure bolting cracking. The vast majority of bolting failures due to SCCs have occurred at PWRs. Boric acid environment is the primary contributor to these SCC failures. Since PBAPS is a BWR and does not have a boric acid environment, bolting does not experience conditions conducive to stress corrosion crack initiation and propagation. Therefore, cracking due to SCC is not considered an applicable aging effect for closure bolting. In evaluating the susceptibility of bolting material, the applicant did not address the effect of the humid environment within containment and the possibility of high yield strength (>150 ksi) for bolting material. **This is part of Open Item 3.1.3.2.1-1.** Additional parts of this open item are discussed below under the loss of material and loss of preload in Section 3.1.4.2.1.

The ISI program will not detect the loss of material on the inside of the carbon steel pipe; therefore is not adequate to assess the effectiveness of the RCS chemistry program to mitigate loss of material in carbon steel components. Therefore, the applicant needs to provide periodic inspections to confirm the effectiveness of the RCS chemistry program for carbon steel components. **This is part of Open Item 3.1.3.2.1-1.**

The AMR did not identify loss of material as an aging effect because several mitigative actions are in place to avoid direct contact between a continuous moisture source and the bolting. These actions include grease coating of bolting during installation, use of antisweat insulation for bolting where the operating temperature is below ambient, and timely repair of any system leakage. However, the applicant does not identify any activities to assess and maintain the effectiveness of grease coating and antisweat insulation. **This is part of Open Item 3.1.3.2.1-1.**

Loss of preload can be caused by factors other than degradation induced by human activities, such as vibration, cyclic loading, gasket creep, and stress relaxation. **This is part of Open Item 3.1.3.2.1-1.**

(From Section 3.1.4.2.1, Effects of Aging) The application does not identify the aging effect of cracking due to stress corrosion cracking and cyclic loading for closure bolting of the recirculation pumps and valves in the recirculation system. Bolting that is heat treated to a high-hardness condition and exposed to a humid environment within containment could be susceptible to SCC. NUREG-1399, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," indicates that the bolting material with yield strength greater than 150 ksi is susceptible to SCC. For high-strength bolting, the effects of cyclic loading are generally seen in conjunction with SCC in causing crack initiation and growth. This issue is discussed in greater detail in Section 3.1.3.2.1 of the SER. **This is part of Open Item 3.1.3.2.1-1**.

(From Section 3.1.4.2.1, Effects of Aging) The applicant does not identify loss of material due to corrosion as an aging effect for recirculation pump closure bolting and valve closure bolting in the reactor recirculation system. This issue is discussed in greater detail in Section 3.1.3.2.1 of the SER. **This is part of Open Item 3.1.3.2.1-1**.

(From Section 3.1.4.2.1, Effects of Aging) The applicant does not identify loss of material due to wear as an aging effect for recirculation pump closure bolting and valve closure bolting in the reactor recirculation system. In response to RAI 3.1-1, the applicant stated that wear is caused by vibration and prying loads, both of which are event-related mechanisms. Therefore, loss of material due to wear should be excluded from an aging management review. The staff disagrees because vibrations and prying loads that can occur during normal operation and maintenance activities can cause loss of material due to wear. **This is part of Open Item 3.1.3.2.1-1.**

(From Section 3.1.4.2.1, Effects of Aging) However, the staff does not consider the hydrostatic pressure tests adequate because it will not detect the loss of material on the inside of the carbon steel pipe, therefor it will not confirm the effectiveness of the RCS chemistry program to prevent loss of material in these components. **This is part of Open Item 3.1.3.2.1-1.**

(From Section 3.1.4.2.1, Effects of Aging) The applicant does not identify loss of preload as an aging effect for recirculation pump closure bolting and valve closure bolting in the reactor recirculation system. This issue is discussed in greater detail in Section 3.1.3.2.1 of this SER. **This is part of Open Item 3.1.3.2.1-1**.

3.6.1.2.1-1 The staff acknowledges that the EPR-insulated replacement cable is more resistant to water-treeing. However, the staff still does not accept the applicant's positions that moisture is not an aging effect requiring aging management for these cables. The staff believes that the discussion and conclusion of the paper, "Assessment of Field Aged 15kV and 35kV Ethylene Propylene Rubber Insulated Cables," do not support the applicant's position that moisture is not an aging effect requiring management at PBAPS. For example, the paper concludes that aging of the EPR-insulated cables can be characterized by an increase in moisture content, growth of water trees, drop in insulation elongation, increase in dissipation factor, and decrease in AC and impulse voltage breakdown strength. Further, the data for water trees, elongation, dissipation factor, and AC and impulse strength indicate that EPR insulated cable deterioration appears to result from moisture permeating the insulation of the cable. Therefore, the applicant has not provided a sufficient technical justification for not requiring an aging management program for inaccessible medium-voltage cables and has not proposed to prevent such cables from being exposed to significant moisture, such as inspecting for water collection in cable manholes and conduit and draining water, as needed. **This is part of Open Item 3.6.1.2.1-1.** The additional part of this open item is discussed in Section 3.6.3.2.1 of this SER.

(From Section 3.6.3.2.1 of this SER) However, as discussed in Section 3.6.1.2.1, the staff does not accept the applicant's position that moisture is not an aging effect requiring an aging management for these cables. The staff is concerned that the applicant has not provided a sufficient technical justification for not requiring an aging management program for buried cables, not specifically designed for a wet environment. **This is the other part of Open Item 3.6.1.2.1-1.**

3.6.1.2.2-1 The applicant should provide a technical justification for high range radiation monitor and neutron monitoring instrumentation cables to demonstrate that visual inspection will be effective in detecting damage before current leakage can affect instrument loop accuracy. **This is Open Item 3.6.1.2.2-1.**

4.5.2-1 The staff is concerned that multiple failures of top guide beams are possible when the threshold fluence for IASCC is exceeded. According to BWRVIP-26, multiple cracks have been observed in top guide beams at Oyster Creek. In addition, baffle-former bolts on PWRs that exceeded the threshold fluence have had multiple failures. In order to exclude the top guide beam from inspection when its fluence exceeds the threshold value, the applicant must demonstrate that failures of multiple beams (all beams that exceed the threshold fluence) will not impact the safe shutdown of the reactor during normal, upset, emergency, and faulted conditions. If this can not be demonstrated, the applicant should propose an aging management program (AMP) for these components which contain the elements in Branch Technical Position RLSB-1 of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," July 2001. **This is Open Item 4.5.2-1.**

1.5 Summary of Confirmatory Items

As a result of the staffs' review of Exelon's application for license renewal, including the additional information and clarifications submitted subsequently, the staff identified the confirmatory items listed below, as of the time this report was prepared. Confirmatory items are those for which Exelon has not yet provided adequate documentation. In addition, confirmatory items may include significant matters that need to be considered as possible license conditions or technical specification requirements, depending on the form of the resolution. Each confirmatory item has been assigned a unique identifying number, which identifies the section in this report in which the confirmatory item is described. For example confirmatory item 3.0-1 is discussed in Section 3.0 of this report.

Item	Description
3.0.3.3.2-1	In the May 14, 2002, response, the applicant also stated that the PBAPS closed cooling water chemistry activities are based on EPRI TR-107396. The EPRI guidelines define control parameters as those that assist with maintaining system chemistry control and define diagnostic parameters as those that assist with corrective actions if improvement in system control is required. As diagnostic parameters, the chlorides, fluorides, sulfates, nitrates, turbidity, and metals are trended. On August 6, 2002, via teleconference the staff requested additional information regarding the chloride and fluoride acceptance criteria. The applicant responded during the call that the acceptance criterion parameters for the chlorides and fluorides is < 10 ppm. The staff requests that the applicant confirm this information in writing. **This is Confirmatory Item 3.0.3.3.2-1**.
3.0.3.11.2-1	To be consistent with the commitment made in response to RAIs 3.5-1 and 3.5-2, the applicant needs to clarify that the scope of the maintenance rule structural monitoring program will be revised to include the above concrete and structural steel components, which now credit this program. These additional commitments will require changes to the UFSAR Supplement (Appendix A of the LRA) for the maintenance rule structural monitoring program to add the additional components to the list in the supplement. **This is Confirmatory Item 3.0.3.11.2-1.**
3.0.3.13.2-1	The staff requested that the applicant address the frequency of inspections of the ECW pump. During a teleconference on August 8, 2002, the applicant indicated that the ECW pumps are inspected every 10 years. **This is part of Confirmatory Item 3.0.3.13.2-1**. The additional part of this item is related to frequency of RWST inspections and is discussed below. 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. The staff requested that the applicant address the frequency of inspections of the RWSTs. During a teleconference on August 8, 2002, the applicant indicated that the RWSTS are inspected every 4 years. **This is the other part of Confirmatory Item 3.0.3.13.2-1**.
3.0.3.14.3-1	The summary description of the door inspection program provided in Section A.2.6 of Appendix A to the LRA does not reflect the additional commitment made by the applicant to include monitoring of hazard barrier doors in a sheltered environment for loss of material due to corrosion. **This is Confirmatory Item 3.0.3.14.3-1.**
3.0.3.17.2-1	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 applicant further stated that the scope of the activities would be enhanced to include periodic inspection of the HPCI gland seal condenser tube side internals. The staff requested that the applicant indicate what percentage of the subject heat exchangers are inspected. During a teleconference on August 6, 2002, the applicant indicated that all tubes of the HPCI gland seal condenser and the HPCI turbine lube oil cooler heat exchangers are visually inspected. **This is part of Confirmatory Item 3.0.3.17.2-1.** The additional part of the confirmatory item is discussed below under the acceptance criteria program element.

The staff requested clarification regarding inspection procedures used to determine acceptability of the heat exchanger tubes. During a teleconference on August 6, 2002, the applicant indicated that the subject heat exchangers are very small heat exchangers and that all tubes are fully disassembled throughly cleaned and visually inspected. In addition, the applicant sited various inspection procedures that are used. **This is the other part of Confirmatory Item 3.0.3.17.2-1**.

3.0.3.19.2-1 The applicant stated that the one-time piping inspection activities will be undertaken to provide reasonable assurance that there is no loss of material or cracking, as adequate for the system material and environment, that would result in loss of pressure boundary intended function of the piping. Qualified personnel following procedures consistent with the ASME Code will perform the nondestructive examinations. The staff requested the applicant to provide information regarding when this one-time inspection would occur. By teleconference call, on August 8, 2002, the applicant indicated that this one-time inspection will occur before the end of plant life, between the years 30 to 40. **This is a Confirmatory Item 3.0.3.19.2-1**.

3.0.3.20.3-1 The applicant describes the reactor materials surveillance program as an existing program in Section A1.12 of the LRA. The program uses periodic testing of metallurgical surveillance samples to monitor the loss of fracture toughness of the reactor pressure vessel beltline region materials consistent with the requirements of 10 CFR Part 50, Appendix H, and ASTM E185. The applicant does not include a summary of the BWR Integrated Surveillance Program, which it intends to use at Peach Bottom. In RAI 3.1-17, the staff requested the applicant to include information about the BWR Integrated Surveillance Program, which should include reference to BWRVIP reports. In response to this RAI, the applicant stated that Section A.1.12 description has been revised to include information about the BWR Integrated Surveillance Program, which is one alternative that may be used at PBAPS to comply with 10 CFR Part 50, Appendix H. **This is Confirmatory Item 3.0.3.20.3-1**.

3.0.4-1 With the applicant's commitment to include in the UFSAR Supplement a new section, Section A.1.17, that describes how the CAP would provide a description of how the attributes of corrective action, confirmation process, and administrative controls are met for the aging management programs the staff will be able to conclude that an adequate program summary has been provided in accordince with the requirements of 10 CFR 54.21(d). **This is Confirmatory Item 3.0.4-1**.

3.2.1.2.2-1 Program Scope: The applicant described the program scope of the HPCI and RCIC turbine inspection activities as focusing on managing loss of material and change in material properties by the performance of periodic inspections of the turbine casings and HPCI lubricating oil system tank internals and flexible hoses. In LRA Table 3.2-1(aging management results for RCIC system), the HPCI and RCIC turbine inspection activities AMP is listed as the aging management program for lubricating oil tanks with lubricating oil as the applicable environment. Wetted gas environment is also in the program scope of the AMP. Therefore, the staff requested the applicant to identify the reference to the AMP being applied to components in a wetted gas environment. By letter dated April 29, 2002, the applicant responded that LRA Table 3.2-1 identifies a number of carbon steel and stainless steel components in a wetted gas environment. For carbon steel components in a wetted gas environment, the applicable aging management activity is referenced in the table. The aging management review has determined that the stainless steel components in the HPCI system (LRA Table 3.2-1) that are exposed to an internal environment of wetted gas do not have any aging effects that require aging management. The applicant stated that therefore no aging management activity is identified for these components in Table 3.2-1. The staff found the scope of the program to be acceptable because the LRA and the additional information provided to the staff the have adequately addressed the components whose aging effects can be managed by the application of the HPCI and RCIC turbine inspection activities. The staff notes that during a conference call on August 21, 2002, the applicant stated the flexible hoses were stainless steel rather than an elastomer of neoprene and rubber. In a call and electronic mail on September 6, 2002, the applicant stated that the stainless steel flexible hose was a gland seal bleed-off line subjected to a wetted gas internal environment and a sheltered air external environment (see LRA Table 3.2-1, page 3-24 third row titled “Elastomer Flex Hoses”) and do not require aging management. Therefore, the flexible hoses would not be covered by this program. The staff finds this acceptable because the stainless steel hoses subject to a wetted gas and sheltered environment do not require aging management. The applicant needs to confirm this information in writing. **This is part of Confirmatory Item 3.2.1.2.2-1.** The additional parts of this confirmatory item are discussed below in the parameters monitored or inspected, detection of aging effects, monitoring and trending, and acceptance criteria program elements.

3.6.1.2.2-1 However, to be consistent with the commitment made in response to RAI 3.6-1, the applicant needs to provide a summary of description of the

B.3.3, “Non-EQ accessible cable aging management activity” in the UFSAR Supplement. **This is Confirmatory Item 3.6.1.2.2-1.**

3.6.2.2.2-1 The applicant proposed an aging management program, “Non-EQ Accessible Cable Aging Management Activity,” for connectors, splices, and terminal blocks in a letter dated April 29, 2002. This program applies to electrical connectors, splices, and terminal blocks within the scope of license renewal that are installed in adverse localized environments caused by heat or radiation in the presence of oxygen. The staff found that the submitted aging management activity is essentially a visual inspection that addresses age-related degradation of connections that can result from exposure to high values of heat or radiation. In addition, fuse holders/blocks are classified as specialized type of terminal block because of the similarity in design and construction. Terminal blocks are passive components subject to an AMR for license renewal and so are fuse holders. During a conference call on September 5, 2002, the applicant stated that it will include fuse holders in the scope of the proposed AMP, Non-EQ accessible Cable Aging Management Activity (B.3.3), and this AMP will manage the aging effects for fuse connectors, splices, and terminal blocks as well as fuse holders. **This is Confirmatory Item 3.6.2.2.2-1.**

4.1.2-1 In a separate licensing action, the applicant has submitted a license amendment for a power uprate to increase the maximum allowed operating power level. This power uprate is based on the increased accuracy of feedwater flow monitors. The higher power level may result in higher reactor coolant temperatures, increased reactor coolant flow, and/or increased neutron fluence. On July 23, 2002, the staff held a conference call with the applicant to ask if the the effects of the power uprate were considered during its evaluation of the TLAAs or that the analysis results are bounding for the higher power level. The applicant stated that the effects of the power uprate were considered. **This is Confirmatory Item 4.1.2-1.**

4.1.3-1 The applicant indicated that it did not expect the number of design transients assumed in these CUF calculations to be exceeded during the period of extended operation. Therefore, the Peach Bottom pipe break postulations remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1). The staff finds that the applicant’s response is acceptable because the existing calculations are bounding for the period of extended operation. The staff concludes that the applicant has adequately evaluated the TLAA related to pipe breaks as required by 10 CFR 54.21(c). The UFSAR update needs to include a summary of the activities for the evaluation of this TLAA. **This is Confirmatory Item 4.1.3-1.**

4.1.3-2 By letter dated February 6, 2002, the staff requested additional information, per RAI 3.3-3, as to why the crane load cycle limit was not included as an TLAA. The applicant responded in a letter dated May, 6, 2002, in which it stated that it will update the UFSAR Supplement to include load cycles for the reactor building overhead bridge cranes, turbine hall cranes, emergency diesel generator bridges, and circulating water pump structure gantry crane as a TLAA in Section 4.7.4 of the LRA. In the response, the applicant stated that the cranes are predominantly used to lift loads which are significantly lower than the crane's rated load capacity. For example, the reactor building cranes will undergo less than 5000 load cycles in 60 years based on the projected number of lifts during refueling outages, handling of spent fuel storage casks, and testing. The other cranes are expected to experience significantly fewer load cycles than the reactor building cranes. Thus, the number of lifts at or near their rated load is low compared to the design limit of 20,000 load cycles. The applicant stated that the load cycles for these cranes were evaluated for the period of extended operation and it was determined that the analyses associated with crane design, including the load cycle limit, remain valid for the period of extended operation and, therefore, meet the requirements of 10 CFR 54.21(c)(1)(i). The staff agrees with the applicant's conclusion that the cranes will continue to perform their intended function throughout the period of extended operation as required by 10 CFR 54.21 (c)(1) and finds the applicant's response acceptable. The update of the UFSAR Supplement is as required by 10 CFR 54.21(c)(1) is **Confirmatory Item 4.1.3-2.**

4.2.1.2-1 The UFSAR Supplement needs to include the additional information contained in the applicant's response to RAI 4.2-3 regarding the evaluation of this TLAA. **This is Confirmatory Item 4.2.1.2-1.**

4.2.3.2-1 The UFSAR Supplement needs to include the additional information contained in the applicant's response to RAI 4.2-6 regarding the evaluation of this TLAA. **This is Confirmatory Item 4.2.3.2-1.**

4.2.4.2-1 The UFSAR Supplement needs to include the additional information contained in the applicant's response to RAI 4.2-7 regarding the evaluation of this TLAA. **This is Confirmatory Item 4.2.4.2-1.**

4.3.2-1 With the applicant's commitment to include in the UFSAR Supplement a description of the corrective actions to address closure studs as provided above in the response to RAI 4.3-1; and perform plant specific calculations for the locations identified in NUREG/CR-6260 for an older vintage BWR plant considering applicable environmental factors provided in NUREG/CR-6583 and NUREG/CR-5704 as provided above in response to RAI RAI 4.3-6; the staff concludes that the UFSAR Supplement will include and appropriate summary description of the programs and activities to manage aging as required by 10 CFR 54.21(d). The applicant needs to provide the revised UFSAR Supplement that includes these commitments. **This is Confirmatory Item 4.3.2-1.**

1.6 Summary of Proposed License Conditions

As a result of the staffs' review of Exelon's application for license renewal, including the additional information and clarifications submitted subsequently, the staff identified 4 license conditions. The first license condition requires the applicant to include the UFSAR Supplement in the next UFSAR update required by 10 CFR 50.71 (e). The second license condition requires that, prior to operation in the renewal term, the applicant will notify the NRC of its decision to implement either the staff-approved reactor vessel integrated surveillance program, or plant-specific program, and provide the appropriate revision to the UFSAR Supplement summary descriptions of the vessel surveillance material testing program. The third license condition requires that the future inspection activities identified in the UFSAR Supplement be completed before the beginning of the extended period of operation. The fourth license condition requires the applicant to submit inspection details of an aging management program to manage the effects of aging on reactor vessel closure studs prior to implementation of such a program by license amendment request.