3/18/03

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T license. Therefore, there is no need
to i the requirements of 10 CFR
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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.

delete "draft" *

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 _P Inspection and Flaw Evaluation Guidelines," April 1997

component and is used to identify the safety-related components in the plant. The UFSAR includes information on the plant, presents the design bases and the limits on the plant's operation, presents the safety analyses of the SSCs and of the facility as a whole, and identifies the intended functions of structures. DBDs are comprehensive system-level documents that provide the design bases and include system functions, controlling parameters, and design features for various operating and accident conditions. In addition, DBDs discuss the regulatory requirements, commitments, codes and standards, and system configuration changes that are reflected in the design basis of the system. The evaluation against license renewal scoping criterion 54.4(a)(1) for mechanical and electrical systems is taken from the evaluation against the corresponding MR scoping criterion described in the LRA. The applicant then performed additional scoping activities to identify systems and structures within the scope of license renewal. For structure-level scoping, a comprehensive list of plant structures to be evaluated for license renewal scoping was produced from the MR bases documentation, the UFSAR and other plant design documentation. Seismic Class I structures were included within the scope of license renewal under scoping criterion 10 CFR 54.4(a)(1). Structural component listings were downloaded from the CRL and added to the license renewal database. Certain types of structural components and commodity items are not identified in the CRL (e.g., equipment pads and pedestals and equipment supports). Such components and commodity items were identified by review of design drawings and plant walkdowns and added to the license renewal database. Some structural components may also be listed as components of mechanical and electrical systems in the CRL.

The scoping results are documented, reviewed, and approved on a license renewal scoping form and entered in the license renewal database. The format of the scoping form is defined in Exhibit LR-C-14-3 of PBAPS procedure LR-C-14, "License Renewal Process." A scoping form is prepared for each system and structure and includes references to the applicable UFSAR sections, design drawings, and DBDs. The form also includes answers to several scoping questions related to system intended functions, applicable supporting systems, and whether any components were realigned into or out of the system (the system boundary realignment methodology is discussed in Section 2.1.2.1.4 of this report). The scoping form is generated as a report from the license renewal database into which the scoping data is entered during the review process. Boundary drawings for the various disciplines in the form of marked-up piping and instrumentation drawings (P&IDs), electrical single-line drawings, and site plan drawings were prepared to identify the major electrical systems and plant structures within the scope of license renewal. The documents are also reviewed and approved by both the license renewal team and PBAPS system managers. mechanical systems

2.1.2.1.2 Non-safety-related Systems, Structures, and Components

With respect to the non-safety-related criteria in 10 CFR 54.4(a)(2), the applicant stated, that a review of the UFSAR and other CLB documents has been performed to identify the non-safety-related and non-safety-related quality SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 54.4(a)(1)(i), (ii), or (iii). Component listings for non-safety-related systems were downloaded from the CRL and reviewed to check for any safety-related components. This review assured that safety-related components associated with system interfaces are captured regardless of which system they were assigned to in the CRL. Any safety-related components found in non-safety-related systems were included in the license renewal database. The specific functions of such components were determined by review against the plant CLB on a case-by-case basis to identify the appropriate system and system

safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations. ~~The answers to these scoping questions were transferred electronically from the MR scoping documentation to the license renewal database and then confirmed during the system scoping review.~~

Systems and structures that are in the scope of license renewal scoping criterion 10 CFR 54.4(a)(3) are identified by review of appropriate plant documentation. For 10 CFR 50.48 and 10 CFR 50.63, the review is documented in license renewal position papers. The reviewer uses the position papers and the CRL to answer the questions on the scoping and screening form. For 10 CFR 50.62, the required components are identified in the controlled CRL database. The equipment within the scope of 10 CFR 50.49 is identified by a controlled data field in the CRL and is addressed in LRA Section 4.4 under the time-limited aging analysis (TLAA) evaluations. For 10 CFR 50.61, no review is performed since it is not applicable to boiling water reactors.

2.1.2.1.4 System Boundary Realignment

A significant aspect of the licensee's scoping and screening methodology involved the use of system boundary realignment. Interfaces between systems were examined and realigned, as necessary, to ensure that interfacing components were associated with the appropriate system for license renewal. For example, a valve in an out-of-scope system that provides an isolation boundary interface with an in-scope system would be considered in the scope of license renewal. The valve is "realigned" to the in-scope system and the remainder of the out-of-scope system remains out-of-scope. Similar realignments are used to address out-of-scope systems that interface with the primary containment boundary. Electrical distribution systems interface with many systems, including many mechanical systems, and the interface point is often an electrical isolation device such as a fuse or circuit breaker. These electrical isolation devices are typically considered part of the mechanical system because their function is to provide electrical isolation of these systems. The applicant examined these interfaces to confirm interfacing components had been identified in the correct system for license renewal. For example, a fuse in an out-of-scope mechanical system that has an isolation boundary interface with an in-scope electrical system was considered in the scope of license renewal. The fuse was realigned to the in-scope electrical system, and the out-of-scope mechanical system remained out-of-scope.

In some cases, components were realigned to support specific intended functions. For example, at PBAPS the main steam isolation valves (MSIVs) are air-operated and require compressed gas to perform their intended function. These valves do not rely on the instrument air distribution system but instead utilize a dedicated instrument air accumulator. Accordingly, the MSIVs instrument air accumulators are required to support the intended function of the MSIVs. For purposes of system scoping, these instrument air accumulators were realigned from the instrument air system to the main steam system. System boundary realignment is described on page 2-5 of the LRA.

Non-Safety-related Systems	Safety-Related Systems With Components Realigned to Non-Safety- Related System
Primary Containment Leak Test System	Primary Containment Isolation System
Reactor Building Ventilation System	RHR System Core Spray System HPCI System RCIC System
Reactor Building Closed Cooling Water System	Primary Containment Isolation System
Reactor Water Cleanup System	Reactor Recirculation System Primary Containment Isolation System
Chilled Water System	Primary Containment Isolation System
Instrument Nitrogen System	Primary Containment Isolation System Main Steam System
Instrument Air System	Main Steam Safety-Grade Instrument Gas System Battery and Emergency Switchgear Ventilation System
Service Air System	Primary Containment Isolation System
Plant Equipment and Floor Drain System	Primary Containment Isolation System
Process Sampling System	Primary Containment Isolation System
Torus Water Cleanup System	Primary Containment Isolation System
Post-accident Sampling System	Primary Containment Isolation System
Traversing In Core Probe System	Primary Containment Isolation System

As a result of the applicant's system boundary realignment, the staff was unable to adequately review the implementation of the boundary realignment using the information presented in the Peach Bottom LRA. Therefore, the staff issued RAIs to the applicant on January 23 and March 12, 2002. The staff's RAI of January 23, 2002, asked the applicant to describe the realignment process and the rationale for its use. The staff's RAI of March 12, 2002, requested the applicant to provide (1) a brief description of each of these out-of-scope systems whose components were realigned to be in-scope, (2) a textual description of the types of components realigned, and (3) details regarding the intended function for each realigned component in the context of license renewal and how the realigned components met the criteria of 10 CFR 54.4(a)(1), (2), or (3). In addition, the RAI requested the applicant to provide a means to identify, in an unambiguous and traceable manner, the components realigned to systems within the scope of license renewal back to the out-of-scope systems. The applicant responded to this RAI by letter on May 22, 2002. The staff's RAI of January 23, 2002, questioned how the realignment was done and the March 12, 2002, RAI questioned the results of the realignment process as presented in the LRA in Sections 2.3 through 2.5. The applicant's response to the staff's RAI, dated February 28, 2002, described the following five cases for system boundary realignment:

grouping those SCs as a commodity. The staff's evaluation of the primary containment isolation system is provided in Section 2.3.2.3 of this document.

Case 4 involves the realignment of shared components of the instrument air and instrument nitrogen systems, which are non-safety-related, to (1) the safety grade instrument gas, (2) the backup instrument nitrogen to ADS, and (3) the battery and emergency switchgear ventilation system (BESVS). In the February 28, 2002, RAI response, the applicant stated that the plant design includes a safety grade backup source of compressed gas for the safety-related systems which share components with the above-mentioned non-safety-related systems. As previously stated, the staff's evaluation of the BESVS is in Section 2.3.3.9 of this document. Also, the staff's evaluations of other realignments involving the instrument air and nitrogen systems are in Section 2.3.3.12 (safety grade instrument gas), and 2.3.3.13 (backup instrument nitrogen to ADS), of this document.

Case 5 involves the realignment of piping and components of the reactor building ventilation system to the boundary of the RHR, core spray, high-pressure coolant injection, and RCIC systems. In the May 22, 2002, response to the staff's RAI 2.2-1.2, the applicant stated that the cooling intended function for all components cooled by the emergency service water (ESW) system is included under the ESW system intended function of component cooling. Further, the HPCI, RCIC, RHR, and core spray system room coolers are cooled by the ESW system. The applicant also stated that the ESW system performs the room cooling function by providing cooling water to the room coolers and therefore the function of room cooling is not included as an intended function of the HPCI, RCIC, RHR, and core spray systems.

Because the components responsible for cooling were realigned to the HPCI, RCIC, RHR, and core spray systems, the system intended function of room cooling is removed from the scope of license renewal. The system intended function of room cooling meets the scope of the Rule in §54.4(a)(2). However, realignment of SCs to extend the boundary of HPCI, RCIC, RHR, and core spray obscures the room cooling function since the supported systems rely on the room coolers to remain functional before and after a design basis event but do not include room cooling as a system level intended function. The staff's evaluations of the system boundary realignment of SCs are in Sections 2.3.2.5 (RHR), 2.3.2.1 (HPCI), 2.3.2.2 (core spray), and 2.3.2.4 (RCIC) of this document.

not true for HPCI + RCIC

Non-Safety-related Systems Affecting Safety-Related Systems

The staff evaluated the applicant's methodology for scoping SSCs meeting the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(2). The implementation of the methodology for the potential spatial interaction between non-safety and safety-related systems resulted in the expansion of systems boundaries for the following systems:

- reactor recirculation system
- reactor pressure vessel instrumentation system
- core spray system
- residual heat removal system
- fuel pool cooling and cleanup system
- control rod drive system
- radiation monitoring system
- emergency service water system

On the basis of the above review, ~~with the exception of Open~~ Item ~~2.3.2.7.2-2~~, the staff did not find any other omissions by the applicant of SSCs within the scope of license renewal.

2.3.2.8.3 Conclusions

On the basis of its review, with the exception of Open Item ~~2.3.2.7.2-2~~, the staff concludes there is reasonable assurance that the applicant has adequately identified the secondary containment SSCs that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.3.3 Auxiliary Systems

In Section 2.3.3, "Auxiliary Systems (AUX)," of the Peach Bottom Atomic Power Station, Units 2 & 3, License Renewal Application (the LRA), Exelon (the applicant) described the systems, structures and components (SSCs) of the AUX that are subject to aging management review (AMR) for license renewal.

2.3.3.1 Fuel Handling Systems

2.3.3.1.1 Summary of Technical Information in the Application

In Section 2.3.3.1, "Fuel Handling Systems," of the LRA, the applicant describes the structural components of the fuel handling systems that are within the scope of license renewal and subject to an AMR. Additional information concerning fuel handling systems is given in Sections 10.3 and 10.4 of the Peach Bottom UFSAR.

In Section 2.1 of the LRA, the applicant described its process for identifying the structural/civil components within the scope of license renewal and subject to an AMR. Based on its methodology, the applicant, in Table 2.2-1 identifies the fuel handling system components within the scope of license renewal and describes the results of its scoping methodology in Section 2.3.3.1 of the LRA.

As stated in Section 10.4.2, "Fuel Servicing Equipment," of the Peach Bottom UFSAR, the fuel preparation machines located in each fuel storage pool are used to remove and install channels to support inspection or servicing of fuel assemblies. The fuel preparation machines are also used for the placement of new fuel assemblies into the spent fuel pool. These machines are designed to be removed from the pool for servicing. In addition, Section 10.4.6, "Refueling Equipment," describes the use and purposes of the refueling platform. The refueling platform is used primarily as a means of transporting fuel assemblies back and forth between the reactor well and the storage pool. The platform travels on rails extending along each side of the reactor well and fuel pool. The platform supports the fuel grapple and the frame-mounted and monorail auxiliary hoists. Platform operations are controlled from either auxiliary hoist control pendants or refuel grapple controller consoles. Other cranes and hoists used during refueling operations, including the fuel channel handling hoists, the control rod drive (CRD) jib crane and the reactor building ~~cask hoist~~, are discussed in LRA Section 2.3.3.18, "Cranes and Hoists."

The applicant's scoping methodology captures fuel handling systems within the scope of license renewal that meet the intent of 10 CFR 54.4(a) because they perform the following "structure level" intended function:

Crane

and increasers are fittings and are part of the piping component group, and therefore are within the scope of license renewal and subject to an AMR. Based on the above clarification, the staff found the applicant's response to RAI 2.3.3.2-1 to be acceptable.

On drawing LR-M-363, sheets 1 and 2, in the fuel storage pool, there is an unidentified component indicated by a circle at location F4. The staff believes that this component may perform one or more intended functions, such as pressure boundary, which justify its inclusion within the scope of license renewal. However, this component is not identified on the legend (drawing LR-M-300). In RAI 2.3.3.2-2, the staff asked the applicant to identify this component and indicate where in the LRA it is included within the scope of license renewal and subject to an AMR. In a letter dated May 22, 2002, the applicant clarified that the "hole" on the drawing is not a component, but represents two siphon breaker holes to prevent siphoning of water. The staff considers the clarification provided in the applicant's response to RAI 2.3.3.2-2 to be acceptable.

In Table 2.3.3-2 of the LRA, a restricting orifice is listed as a component requiring an AMR. However, pressure boundary is the only intended function listed for this component. In RAI 2.3.3.2-3, the staff questioned whether flow restriction should also be listed as an intended function for this component. In a letter dated May 22, 2002, the applicant stated that the restricting orifice was installed in the RHR to fuel pool discharge line during plant construction to give a pressure drop large enough to prevent the upstream valves from vibrating open. However, the addition of RHR pump discharge control valves, after the original plant construction, provides sufficient flow control that the restricting orifice is no longer needed. Therefore, the restricting orifice is not required to provide the flow restriction (throttle) intended function. ~~The staff found the applicant's exclusion of this component from the scope of license renewal to be acceptable, as the component does not perform an intended function that meets~~ the criteria of 10 CFR 54.21(a)(1).

Not true. Component is included ... scope but flow restriction function is not.

On the basis of the above review, the staff did not find any omissions by the applicant of SSCs within the scope of license renewal.

2.3.3.2.3 Conclusions

On the basis of its review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the fuel pool cooling and cleanup system SSCs that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.3.3.3 Control Rod Drive System

2.3.3.3.1 Summary of Technical Information in the Application

As described in the LRA, the control rod drive (CRD) system is a reactivity control system that utilizes pressurized demineralized water to rapidly insert control rods in the core upon receipt of a scram signal. The system also provides control rod manipulation and positioning for power adjustments, and serves as a source of cooling water for the Graphitar seals of the CRD mechanisms.

The CRD system serves as a source of purge water for the reactor water cleanup pumps and reactor recirculation pump seals. The system also serves as a source of injection water to reactor vessel level instrumentation reference legs to mitigate the accumulation of gases.

In a letter dated May 22, 2002, the applicant responded that the radwaste exhaust vent and the ductwork leading to it are within the scope of license renewal and are subject to an AMR. These components (ductwork and exhaust hoods) are included in LRA Table 2.3.3-9. License renewal drawing LR-M-399, sheet 4, Rev. A, is in error, and will be revised to identify the exhaust vent and associated ductwork as in-scope. The staff considers the applicant's response to be acceptable.

As stated in applicant's response to RAI 2.2-1.1(b) (refer to SER Section 2.2.3), the instrument air system piping, tubing, and valve bodies that are required to support the safety-related pneumatic system pressure boundary were realigned from the instrument air system to the BESVS for license renewal. The normal source for compressed gas to the pneumatic controls is from the non-safety-related instrument air system. However, portions of the pneumatic controls in the BESVS are safety-related, as are the nitrogen bottles, which are the safety-related source for compressed gas to the pneumatic controls. The subject piping and tubing with associated valves is shown as cross-hatched (pneumatic piping and tubing symbol) and is highlighted as falling within the scope of license renewal on boundary drawings LR-M-399 sheets 1 and 4, Rev. A.

As discussed above, portions of the instrument air system were realigned to the BESVS. In a letter dated October 30, 2001, the staff identified certain components that were omitted from Tables 2.3.3-9 and the corresponding table in Section 3.3. In a November 16, 2001, response, applicant stated that when LRA Table 2.3.3-9 was prepared, the BESVS component groups in the gas environment AMR were inadvertently omitted. Additionally, the applicant stated that LRA Table 2.3.3-9 requires the addition of "dry gas" in the "Environment" column for both the "valve bodies" and "pipe" entries. The applicant further explained that the valve bodies are brass material, and the pipe is copper material. In its May 22, 2002, response to the staff's March 12, 2002, RAIs 2.2-1.1(a) and (b), the applicant clarified which systems or portions thereof were realigned, and revised LRA Table 3.3-9. The revision adds pipe to the component group of piping which performs the intended function of pressure boundary. The staff finds the addition of the components in the dry gas environment to be acceptable because they perform an intended function, as described in 10 CFR 54.21(a)(1), without moving parts or without a change in configuration or properties.

On the basis of the above review, ~~with the exception of Open Item 2.3.3.9.2-1~~, the staff did not find any other omissions by the applicant of SSCs within the scope of license renewal.

2.3.3.9.3 Conclusions

On the basis of its review, ~~with the exception of Open Item 2.3.3.9.2-1~~, the staff concludes there is reasonable assurance that the applicant has adequately identified the battery and emergency switchgear ventilation SSCs that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.3.3.10 Diesel Generator Building Ventilation System

2.3.3.10.1 Summary of Technical Information in the Application

In Section 2.3.3.10 of the LRA, the applicant identified the boundaries of the diesel generator building ventilation system (DGBVS) and the DGBVS components within the scope of license

main control room complex, radwaste building, and reactor auxiliary bay. The resistive coatings are within the scope of license renewal and subject to an AMR and, therefore, should be included in the scope of fire protection activities as described in LRA Appendix B.2.9.

Using the information provided in the LRA and the UFSAR, the staff sampled several cases of hazard barriers and elastomers to determine whether the applicant properly identified them as being subject to an AMR in Table 2.4-14 of the LRA. On the basis of the above review, the staff did not find any omissions by the applicant of SSCs within the scope of license renewal.

2.4.14.3 Conclusion

On the basis of its review, the staff concludes there is reasonable assurance that the applicant has adequately identified the hazard barrier and elastomer SSCs that are within the scope of license renewal and subject to an AMR in accordance with 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.4.15 Miscellaneous Steel

2.4.15.1 Summary of Technical Information in the Application

In Section 2.4.15 of the LRA, the applicant described the miscellaneous steel. The commodity group of miscellaneous steel includes platforms, grating, stairs, ladders, steel curbs, handrails, kick plates, instrument tubing trays, and manhole covers. These structural steel components are generally installed throughout Peach Bottom plant structures. Some structural steel components are exposed to the outdoor environment. These steel components are treated as commodities because of similarities in design, material, and/or environment.

In Section 2.1 of the LRA, the applicant described its process for identifying the structural/civil components falling within the scope of license renewal and subject to an AMR. In addition to the structures falling within the scope of license renewal presented in this section, the applicant evaluated several structural component groups such as miscellaneous steel, as commodities. Commodity groups were determined based upon similar design or similar materials and similar environments. For each of the structural commodities, the applicant provided the following information:

- a general description of the commodity
- list of the components or component groups that require an AMR, and the associated component intended functions and environments

On the basis of this methodology, the applicant identified, in Table 2.4-15, the structural components in the miscellaneous steel commodity group subject to an AMR. Table 2.4-15 of the LRA lists structural support, fluid containment, shelter, protection, and/or radiation shielding as the intended functions of the miscellaneous steel commodity group.

2.4.15.2 Staff Evaluation

The staff reviewed Section 2.4.15 of the LRA, the associated sections of the Peach Bottom UFSAR, relevant staff's SERs, and the IPE and IPEEE to determine whether there is reasonable assurance that the miscellaneous steel system components and supporting structures within the

Acceptance Criteria: BWRVIP I&E reports provide the basis for Peach Bottom reactor pressure and vessel internals inspection requirements, acceptance criteria, and proper corrective actions. The applicant has incorporated these applicable I&E reports into the Peach Bottom LRA by specific reference. BWRVIP I&E reports applicable to PBAPS RPV and vessel internals components are as follows:

Component	Reference	SER Date	Accession # for SER
Reactor pressure vessel components	BWRVIP-74	10/18/01	ML012920549
Vessel shells	BWRVIP-05	03/07/00	ML003690281
Shroud support and attachments	BWRVIP-38	03/01/01	ML010600211
Shroud	BWRVIP-76	the end of 2003	N/A
Nozzle safe ends and piping	BWRVIP-7~~5~~ 4	09/15/00	ML003751105
Core support plate	BWRVIP-25	12/07/00	ML003775989
Core ΔP/SLC line and nozzle	BWRVIP-27	12/20/99	ML993630179
Core spray, jet pump riser brace, and other attachments	BWRVIP-48	01/17/01	ML010180493
Core spray lines and spargers	BWRVIP-18	12/07/00	ML003775973
Top guide	BWRVIP-26	12/07/00	ML003776119
Jet pump assemblies	BWRVIP-41	05/01/01	ML011570560
CRDH stub tubes and guide tubes, ICM housing guide tubes and penetrations	BWRVIP-47	12/07/00	ML003775765
Instrument penetrations	BWRVIP-49	03/31/02	NUDOCS ? different from SER
Integrated Surveillance Program Plan	BWRVIP-78	02/01/02 (40 years)	ML020380691
Intergrated Surveillance Program: Implementation Plan	BWRVIP-86	02/01/02 (40 years)	ML020380691

The acceptance criteria for cracking in the feedwater nozzle are presented in the industry report GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements," May 2000. The staff finds that the acceptance criteria, as presented in the referenced BWRVIP reports and in GE-NE-523-A71-0594-A, Revision 1, are acceptable.

While the review of BWRVIP-76, which deals with cracking and inspections of the core shroud, has not been completed, PBAPS has indicated by letter dated May 6, 2002, that it will incorporate the NRC approved BWRVIP-76 programs into its aging management activities. The renewed license will be conditioned to require that, prior to operation in the renewal term, the applicant will notify the NRC of its decision to implement the staff approved BWR core shroud inspection and flaw evaluation guideline program or a plant-specific program, and provide adequate revisions to the UFSAR Supplement summary description of the program.

The staff has completed the review of the integrated surveillance (ISP) program that is documented in BWRVIP-78 and BWRVIP-86. However, this program is only applicable for 40 years. The staff expects to receive a revised integrated surveillance program for review that is applicable for 60 years, which will be based on the technical criteria in BWRVIP-78 and BWRVIP-86.

confirmation process, administrative controls, and operating experience. The applicant indicated that the corrective actions, confirmation process, and administrative controls are part of the site-controlled quality assurance program. The staff's evaluation of these three elements is provided separately in Section 3.0.4 of this SER. The remaining seven elements are evaluated below.

Program Scope: The components within the scope of the Fire Protection Activities program are the sprinklers and fire hydrant valves and hose rack valves of the fire protection system. These components include the 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 penetration 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.
- Perform a one-time test of a cast iron fire protection component.

The staff finds acceptable the scope of the components and systems within fire protection activities, including the enhancements.

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 actions associated with these activities, and the staff did not identify the need for any.

Parameters Monitored or Inspected: The existing fire protection activities provide for visual inspections and/or monitoring 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.
- Visual inspections of fire barrier penetration seals, fire barrier doors, and fire wraps to detect changes in material properties, cracking, delamination, separation, and loss of material.

exposed to reactor coolant water. The applicant will use the RCS chemistry program, ISI program, and FAC program to manage loss of material for carbon steel piping, piping specialties, and valve bodies. The applicant will use the RCS chemistry program to manage loss of material for stainless steel or low alloy steel piping (tubing) and valve bodies.

Cracking was identified for the stainless steel pipe, tubing, and valve bodies in a reactor coolant environment. Cracking of stainless steel materials may occur in reactor coolant environment, and therefore may be an applicable aging effect for the stainless steel surfaces exposed to reactor coolant. The applicant will use the RCS chemistry program to manage the ~~loss of material~~ cracking associated with stainless steel pipe, tubing, and valve bodies in a reactor coolant environment.

3.4.3.2.2 Aging Management Programs

The applicant stated that the RCS chemistry program, ISI program, and FAC program will be used to manage the loss of material associated with carbon steel or low alloy steel piping, piping specialties, and valve bodies. The RCS chemistry program will be used to manage the cracking ~~loss of material~~ associated with stainless steel pipe, tubing, and valve bodies in a reactor coolant environment.

A detailed description of each of the programs identified above is included in Appendix B to the LRA, along with a demonstration that the identified aging effects will be effectively managed for the period of extended operation. The staff's detailed review of the different aging management activities and their ability to adequately manage the applicable aging effects is provided in Sections 3.0.3.1, 3.0.3.2, and 3.0.3.6 of this SER. As a result of its review, the staff did not identify any concerns or omissions in the aging management activities used to manage the feedwater system.

3.4.3.3 Conclusions

The staff has reviewed the information in Section 3.4, "Aging Management of Steam and Power Conversion Systems," of the LRA. The staff considered both industry and plant-specific experience. On the basis of its review, the staff concludes that the applicant's identification of the aging effects associated with the feedwater system is consistent with published literature and industry experience. The staff further concludes that the applicant has adequate aging management programs to effectively manage the aging effects of the feedwater system and that there is reasonable assurance that the intended functions of the system will remain consistent with the CLB during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.5 Aging Management of Structures and Component Supports

3.5.1 Containment Structure

3.5.1.1 Technical Information in the Application

The aging management review results for the containment structure, which consists of the

not true for non-EQ cables

1.5 to obtain the 60-year value ~~and then adding the accident dose~~. All other cable insulation types were bounded by this analysis. No cables requiring aging management as a result of radiation effects were identified.

A review of cable insulation aging effects from temperature required a more detailed elimination process. Cable populations were grouped according to their common cable insulation material type and voltage application (power, control, or instrumentation). For each cable insulation material type, a 60-year limiting service temperature was established. This value was compared to the bounding cable service temperature to determine if it was below the 60-year limiting service temperature. Ohmic heating was considered for power cables and for control cables that are routed with power cables, where applicable to determine the bounding service temperature. A summary of each cable group review follows:

- Computer Cable Groups

  Computer cable groups are not in the scope of license renewal and were eliminated from the temperature review.

- Fibre Optic & Bare Ground Cable Groups

  Fibre optic cable insulation material is unaffected by thermal aging. Bare ground cables have no insulation and were determined not to be within the scope of license renewal.

- Instrumentation Cable Groups

  Instrumentation cable groups with cross-linked polyethylene (XLPE), polyethylene, cross-linked polyolefin (XLPO), hypalon, Teflon-based, and polypropylene insulation were determined to have 60-year limiting service temperature greater than the bounding ambient temperature of PBAPS. Two bounding ambient temperatures were determined: one bounding ambient temperature for containment and another bounding ambient temperature for all other plant areas.

- XLPE Power & Control Cable Groups

  XLPE insulated cable groups can operate continuously at their bounding service temperature for greater than 60-years. The 60-year limiting service temperature is greater than bounding ambient temperature and its associated ohmic heating temperature rise.

- EPR Power & Control Cable Groups

  EPR (ethylene polymer rubber) cable groups supplying loads not in the scope of license renewal were eliminated from review. The remaining EPR cable groups were determined to be routed in areas outside containment and have 60-year limiting service temperature greater than the bounding ambient temperature and its associated ohmic heating temperature rise.

- PE Power and Control Cable Groups

  The routing of PE (polyethylene) power and control cable groups was determined and local ambient temperature field measurements were conducted in bounding cases. The 60-year limiting service temperature for PE insulation groups was greater than the bounding ambient temperature and its associated ohmic heating temperature rise.

- PVC Cable Groups

  Poly-vinyl-chloride (PVC) cables groups and individual cables from the remaining PVC cable groups supplying loads not in the scope of license renewal were eliminated from review. The remaining PVC cables were reviewed to identify cables with 60-year limiting service temperatures greater than the bounding service temperature. Thirty cables relied upon for fire safe shutdown (FSSD) were determined to require aging management.

- Miscellaneous Cable Groups

  Miscellaneous cables groups not in the scope of license renewal loads were eliminated from review. Miscellaneous cable groups were also reviewed to eliminate cables with a 60-year limiting service temperature greater than the bounding ambient temperature. Individual cables within the remaining group were reviewed to identify cables within the scope of the environmental qualification aging management activity or cables supplying loads not within the scope of license renewal. None of the miscellaneous cables were identified as requiring management.

3.6.1.1.2 Aging Management Program

Table 3.6-1 of the LRA provides the aging management review results for cables. In this table, no aging management activity is identified except for PVC insulated fire safe shutdown cables. The applicant states that a cable replacement program was initiated in 1995 to replace "suspected" cables subject to the water-treeing. No cable failures have occurred at PBAPS since the cable replacement program was initiated. Therefore, moisture is not an aging effect requiring management at PBAPS. The applicant also states that the maximum operating doses of insulation material (1.5 times the existing radiation design value plus the accident dose) will not exceed the 60-year service limiting radiation dose. The maximum operating temperature of insulation material will also not exceed the maximum temperature for 60-year life. The applicant concludes that no aging management programs are required for cables due to heat or radiation.

The fire safe shutdown (FSSD) inspection activity is a new aging management program. The applicant reviewed the PVC cable groups and determined that 30 cables relied upon for fire safe shutdown require aging management. These cables have a 60-year service temperature greater than the bounding service temperature. These cables are located in the drywell and are all MSRV discharge line thermocouple wires. The inspection will manage change in material properties of the PVC insulation.

moisture simultaneously with significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific test of test performed will be determined prior to the initial test. Each test performed for a cable may be a different type of test. This activity will provide reasonable assurance that aging effects on the conductor insulation are detected and addressed such that the intended function of these cable will be maintained for the period of extended operation. This activity will be implemented prior to the end of the initial operating license term for PBAPS.

The staff reviewed proposed Section B.3.5 of the UFSAR Supplement (Appendix B of the LRA) and verified that the information provided in the UFSAR Supplement for the aging management of systems and components discussed above is equivalent to the information in NUREG-1800 and therefore provides an adequate summary of program activities as required by 10 CFR 54.21(d).

Conclusions

The staff concludes that the applicant has demonstrated that the aging effects associated with inaccessible medium-voltage cables not subject to 10 CFR 50.49 environmental qualification requirements will be adequately managed so there is reasonable assurance that the intended functions of the systems and components will be maintained consistent with the CLB during the period of extended operation as required by 10 CFR 54.2(a)(3). The staff also concludes that the UFSAR Supplement contains an adequate summary description of the program activities for managing the effects of aging for the systems and components discussed above as required by 10 CFR 54.21(d).

For accessible Non-EQ cables installed in adverse localized environments due to heat or radiation, in Section 2.5.1 of the LRA, the applicant states that the maximum operating doses of insulation material (1.5 times the existing radiation design value plus the accident dose) will not exceed the 60 year-service limiting radiation dose. The applicant also states that the maximum operating temperature of insulation material will not exceed the maximum temperature for 60-year life. Therefore, it concludes that no aging management is required for aging effects due heat or radiation. Additionally, on January 2, 2002, the applicant stated that a plant walk down was conducted outside containment (i.e., excluding the drywell and steam tunnel) to identify any adverse localized equipment environments. It was concluded that only the drywell PVC cables credited for fire safe shutdown required an aging management activity. The staff finds that this conclusion is not consistent with the aging management program and activities for electrical cables and connections exposed to adverse localized environments caused by heat or radiation, because conductor insulation material used in cables may degrade more rapidly than expected.

The radiation levels most equipment experience during normal service have little degrading effect on most materials. However, some localized areas may experience higher-than-expected radiation conditions. Areas prone to elevated radiation levels include areas near primary reactor coolant system piping or the reactor-pressure-vessel; areas near waste processing systems and equipment (e.g., gaseous waste system, reactor purification system, reactor water cleanup system, and spent fuel pool cooling and cleanup system); and areas subject to radiation streaming. The most common adverse localized equipment are those

The applicant discussed the piping and component fatigue analyses in Section 4.3.3 of the LRA. The applicant designates reactor coolant pressure boundary piping as Group I piping. The applicant indicated that all Group I piping was originally designed to United States of America Standards (USAS) B31.1, 1967. This code did not require an explicit fatigue analysis of piping components. The applicant indicated that the Group I recirculation piping and RHR piping were replaced because of IGSCC concerns and that the replaced piping was analyzed to ASME Section III Class 1 requirements, which include an explicit fatigue analysis. The applicant indicated that a simplified fatigue analysis was developed for the remainder of the Group I piping to estimate CUFs from the operating data. The applicant indicated that fatigue of the Group I piping will be managed by the FMP in accordance with 10 CFR 54.21(c)(1)(iii).

The applicant designates the remainder of the safety-related piping as Group II and III. This piping was designed to the requirements of USAS B31.1. USAS B31.1 requires a reduction in the allowable bending loads if the number of full range thermal bending cycles exceeds 7,000. The applicant's evaluation indicated that the expected number of thermal bending cycles will not exceed the 7,000 limit during the period of extended operation and that the analyses remain valid for the period of extended operation in accordance with 54.21(c)(1)(i).

The applicant discussed the evaluation of the effects of the reactor coolant environment on the fatigue life of components in Section 4.3.4 of the LRA. The applicant relied on industry generic studies to address this issue.

4.3.2 Staff Evaluation

The components of the RCS were designed to codes that contained explicit criteria for fatigue analysis. Consequently, the applicant identified fatigue analyses of these RCS components as TLAAs. The staff reviewed the applicant's evaluation of the identified RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The design criterion for ASME Class 1 components involves calculating the CUF. The fatigue damage in the component caused by each thermal or pressure transient depends on the magnitude of the stresses caused by the transient. The CUF sums the fatigue damage resulting from each transient. The design criterion is that the CUF not exceed 1.0. The applicant monitors limiting locations in the RPV, RVI, and RCS piping for fatigue usage through the FMP. The applicant relies on the FMP to monitor the CUF and manage fatigue in accordance with the provisions of 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the FMP is in provided below.

The applicant indicated that all component locations where the 40-year CUFs are expected to exceed 0.4 are included in the FMP. Section 4.3.1 of this SE lists the component locations monitored by the FMP. These locations have been identified in the reactor vessel, vessel internals, reactor coolant system piping, and torus. The applicant indicated that the existing FMP maintains a count of cumulative reactor pressure vessel thermal and pressure cycles to ensure that licensing and design basis assumptions are not exceeded. The applicant also indicated that 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. In RAI 4.2-2, the staff requested that the applicant describe how the monitored data will be used to calculate usage factors and to indicate how the fatigue usage will be estimated prior to implementation of the improved program.

4.3.2

Although the letter dated August 6, 1999, identified the staff's concerns regarding the EPRI procedure and its application to PWRs, the technical concerns regarding the application of the Argonne National Laboratory (ANL) statistical correlations and strain threshold values are also relevant to BWRs. In addition to the concerns referenced above, the staff identified additional concerns regarding the applicability of the EPRI BWR studies in its review of the Hatch LRA. EPRI topical report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant, and EPRI topical report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," used plant transient data from a newer vintage BWR-4 plant. The applicant indicated that these issues were considered in the assessment of metal fatigue at Peach Bottom.

The applicant discussed the impact of the environmental correction factors for carbon and low-alloy steels contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and the environmental correction factors for austenitic stainless steels contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels," on the results of the EPRI studies. The applicant indicated that the impact of the new carbon steel data was not significant. The applicant applied a correction factor of 2.0 to the EPRI generic study results to account for the new stainless steel data.

The applicant indicated that EPRI topical report TR-110356 contained studies that are directly applicable to Peach Bottom because they involved a BWR-4 that is identical to the Peach Bottom design. However, the only components evaluated in TR-110356 are the feedwater nozzle and the control rod drive penetration locations. The staff had previously expressed concerns regarding the applicability of the measured data contained in EPRI topical report TR-110356 to another facility in its review of the Hatch LRA.

The applicant provided the sixty-year CUFs projected for Peach Bottom Units 2 and 3 at the locations evaluated for an older vintage BWR in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," dated March 1995, in Table 4.3.4-3 of the LRA. The applicant indicated that these locations are monitored by the FMP, and that the environmental factors have been adequately accounted for by the conservatism in the design basis transient definitions. The applicant indicated that the vessel support skirt is monitored in lieu of the shell region identified in NUREG/CR-6260 because it is a more limiting fatigue location. The applicant also indicated that, since the location is on the vessel exterior, the environmental fatigue factors do not apply. The staff agrees with the applicant's statement.

In RAI 4.3-6, the staff requested that the applicant provide an assessment of the six locations identified in NUREG/CR-6260 considering the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 reports for Peach Bottom Units ~~1~~ 2 and ~~2~~ 3.

In its May 1, 2002, response, the applicant committed to perform plant-specific calculations for the locations identified in NUREG/CR-6260 for an older vintage BWR plant considering the applicable environmental factors provided in NUREG/CR-6583 and NUREG/CR-5704. The applicant committed to complete these calculations prior to the period of extended operation and take appropriate corrective actions if the resulting CUF values exceed 1.0. The staff finds the applicant's commitment to complete the plant-specific calculations described above prior to

Exelon reserves the right to modify this position in the future, based on the results of industry activities currently underway, as well as based on the results of any other methodology improvements that may be made associated with environmental fatigue. It is understood that any such modifications will be subject to NRC approval prior to implementation at PBAPS.

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In Attachment 3 to a letter from M. P. Gallagher to USNRC dated January 14, 2003, the applicant provided a revised Reactor Pressure Vessel and Internals ISI Program (B.2.7) which indicates Peach Bottom will perform augmented inspections for the top guide similar to the inspections of Control Rod Drive Housing (CRDH) guide tubes. The sample size and frequency for CRDH guide tubes is a 10% sample of the total population within 12 years; one half (5%) to be completed within six years. The method of examination is an enhanced visual examination (EVT-1). EVT-1 are utilized to examine for cracks. The program will be implemented prior to the end of the initial operating license term for Peach Bottom. The applicant also stated that it might modify the above agreed-upon inspection program should the BWRVIP-26, "BWR Vessels and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)," be revised in the future. This is acceptable to the staff because any modifications to the BWRVIP-26 program through the BWRVIP are reviewed and approved by the staff. Since the aging effect is IASCC, the staff requested the applicant to clarify whether the inspection sample would be in top guide locations that receive the greatest amounts of neutron fluence. In a letter from M. P. Gallagher to USNRC dated January 29, 2003, the applicant concluded that future locations for the top guide inspections will be in the center or close to the center of the core in the high fluence region. The conclusion is based on the applicant's experiences with prior CRDH inspections. Since the applicant has proposed an inspection program which will be able to detect IASCC in locations which receive high neutron fluence, the staff considers the program acceptable; therefore, Open Item 4.5.2-1 is closed.

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

Radiation embrittlement and fatigue usage may affect the ability of certain reactor vessel internals (RVI), particularly the core shroud support plate, to withstand an end-of-life reflood thermal shock following a recirculation line break. The applicant evaluated the effects of embrittlement and fatigue on the end-of-life reflood thermal shock analysis. 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. Based on the applicant's evaluation of the impact of fatigue and embrittlement on RVI components, the staff concludes that reflood thermal shock will not significantly affect the capability of RVI components to perform their intended functions during the 60-year extended operating term. The impact of reflood thermal shock on the reactor vessel is discussed in Section 4.2.1 of this SER.

4.5.3 Conclusions

The staff concludes that, ~~with the exception of Open Item 4.5.2-1,~~ 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). Because of the above open item the staff cannot conclude that the UFSAR Supplement provides an adequate description of the evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d). Pending resolution of the open item, the staff will determine if the UFSAR Supplement contains an appropriate summary description.

The effect of fatigue and embrittlement on end-of-life reflood thermal shock analysis have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The staff has also reviewed the UFSAR Supplement and the staff concludes the