March 13, 2002

APPLICANT: Exelon Generation Company, LLC (Exelon)

FACILITIES: Peach Bottom Atomic Power Station, Units 2 and 3

SUBJECT: TELECOMMUNICATION WITH EXELON GENERATING COMPANY TO DISCUSS INFORMATION IN SECTIONS 3.1 AND 4.1 OF THE PEACH BOTTOM LICENSE RENEWAL APPLICATION

On January 22, 2002, after the NRC staff reviewed information provided in Sections 3.1 and 4.1 of the license renewal application (LRA), a conference call was conducted between the staff and representatives of Exelon Generating Company to clarify information presented in the application pertaining to Sections 3.1and 4.1. The information discussed, the applicant's responses, and the follow-up actions are in Attachment 1.

A draft of this telephone conversation summary was provided to the applicant to allow them the opportunity to comment on the contents of its input prior to the summary being issued.

/***RA***/

Raj K. Anand, Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Attachments: As stated

cc w/attachments: See next page

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SUMMARY OF TELECOMMUNICATION WITH
EXELON GENERATING COMPANY
PEACH BOTTOM UNITS 2 AND 3

3.1 Reactor Coolant System

RAI 3.0-1

This is a global RAI applicable to all systems.

The application does not identify the aging effects of cracking due to stress corrosion cracking, cyclic loading, wear, loss of pre-load, and loss of material for closure bolting for valves and pumps in any system. The applicant should review industry and plant experience to assess whether these aging effects are applicable for closure bolting. If such aging effects are present, the applicant should submit an aging management program to manage these aging effects in the closure bolting.

Response to 3.0-1:

The applicant stated that NEI 95-10 Revision 3, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 the License Renewal Rule, which is endorsed by NRC REG Guide 1.188, does not consider bolting as a component. Based on this guideline, PBAPS LRA did not include it as a line item under component groups, although an AMR was performed for these piece parts. The environment that bolting would see would be external environments. External environments could be sheltered, outdoor, buried or submerged (raw water environment). These environments are described in PBAPS LRA section 3.0.

Loss of preload: Bolting pre-load is a design condition. Peach Bottom and industry operating experience has shown that proper closure bolting pre-load is effective in preventing mechanical joint leakage. A loss of pre-load would be detected by joint leakage before there is a catastrophic failure. Most loss of pre-load events are attributed to human error. According to the conclusion stated in the June 5, 1998, NRC letter from C.I.Grimes to D.Walters of NEI, in the subject matter of LR Issue No. 98-0013, Degradation Induced Human Activities , degradation events induced by human activities need not be considered as a separate aging effect and should be excluded from an aging management review.

Wear and cyclic loading: Both are caused by vibration and prying loads aging mechanisms. Both of these are event related degradation mechanisms, and based on the NRC letter indicated above in the loss of pre-load paragraph, need not be considered as a separate aging effect and should be excluded from an aging management review.

Cracking due to stress corrosion cracking: SCC occurs through the combined actions of stress (either applied or residual), a corrosive environment, and a susceptible material. All three conditions must be present simultaneously to produce SCC. SCC is characterized by the base metal not being attacked over most of its surface while fine cracks have propagated through the microstructure. The threshold values for stress and corrosion are difficult to determine. Major suspected causes of SCC in fasteners are the use of lubricants containing sulfur compounds and the use of sealants containing fluorides or halides. Bolting materials susceptible to SCC are stainless steel and high-strength low alloy steel.

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 cracking of closure bolting.

Loss of material: For the plant sheltered environment, the presence of a continuous moisture source will typically not be in direct contact with threaded fasteners. In addition, during plant operation the drywell is inerted with nitrogen which reduces the oxygen concentration to less than 4% to render the atmosphere non-flammable. Lack of oxygen in the drywell has the added benefit of minimizing the potential for corrosion degradation. In general, moisture on the external surfaces of threaded fasteners could be caused by high humidity and resulting condensation or by system leakage. Plant sheltered environmental conditions during normal operations vary with the humidity ranging from 10% to 90%. To guard against condensation, anti-sweat insulation was specified for all piping and components where the process operating temperature is between 30-60°F or is below ambient. During installation, closure bolting is coated with grease to aid in obtaining proper pre-load. System leakage, when present, is repaired in a timely manner as part of the plant inspections, testing, and corrective actions activities and is not considered to be a long-term moisture source. PBAPS does not have a history of closure bolting loss of material when the bolting is located in a sheltered environment. Since the relevant conditions that contribute to the onset of general corrosion are being controlled, general corrosion is not considered an aging mechanism for closure bolting located in the plant sheltered environment. The applicant further stated that closure bolting located in outdoor, buried, and submerged environments is unprotected and general corrosion, pitting and crevice corrosion are applicable loss of material aging mechanisms that cause loss of material aging effect requiring management. The Outdoor, Buried and Submerged Component Inspection Activities as described in Appendix B.2.5 manage these aging effects.

Discussion: The applicant's response is not sufficient because it lacked adequate details. Therefore, the staff is requesting for more detail information in the following RAI:

The application does not identify the aging effects of cracking due to stress corrosion cracking (SCC), cyclic loading, wear, loss of preload, and loss of material for closure bolting for valves and pumps in any 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. Vibration, cyclic loading, gasket creep and stress relaxation could cause loss of preload. Carbon steel bolting exposed to a humid environment within containment could be susceptible to loss of material.

The applicant should take into account the above information and review industry and plant experience to assess whether these aging effects are applicable for closure bolting. If such aging effects are applicable, the applicant should submit an aging management program to manage these aging effects in the closure bolting.

RAI 3.1.3.1-1

To determine whether the applicant has identified all applicable aging effects for components in the reactor pressure vessel (RPV) and internals, reactor pressure vessel instrument system, and reactor recirculation system, the applicant is requested to identify their process for reviewing industry experience and Peach Bottom experience related to aging effects in these components. Describe where the review of industry and Peach Bottom experience is documented so that it can be verified through future inspections and identify the BWRVIP documents that describe the industry experience for RPV, internals, reactor pressure vessel instrument systems, and reactor recirculation system components.

Response to 3.1.3.1-1

The applicant stated that PBAPS LRA Section 3.0 Aging Management Review Results, on page 3-8 under Aging Effects contains a summary of the types of documents reviewed in preparing the application. This paragraph is reproduced below:

The systematic assessment of aging effects was based on the collective experience of the nuclear power industry available in pertinent industry literature and specific PBAPS operating experience. Identification of those aging effects that require management incorporated information developed from available industry experience and PBAPS experience. The evaluation process included a review of pertinent industry operating experience as contained in NRC generic communications such as Information Notices, Generic Letters and Bulletins. In addition, PBAPS specific experience was reviewed including plant maintenance history, modifications, nonconformance reports, and Licensee Event Reports. GE service information letters, operating experience assessment reports, topical information from various industry working groups, and plant condition reports were also reviewed.

The methodologies employed to prepare Peach Bottom Aging Management Review Reports (AMRs) assured that aging effects noted in these communications were appropriately considered in the PBAPS LRA. A list of generic communications considered is included in each AMR.

For Reactor Pressure Vessel and RPV Internals, BWRVIP relied on extensive review of applicable industry operating experiences and examination results to develop appropriate inspection and evaluation guidelines. PBAPS LRA Appendix B.2.7 Reactor Pressure Vessel and Internals ISI Program in the Activity Description section states:

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

The applicant further stated the BWRVIP program was reviewed for its applicability to PBAPS design, construction, and operating experience Therefore, it was concluded that the BWRVIP inspection and evaluation reports bound PBAPS design and operation.

Appendix B.2.7 references the BWRVIP documents used for PBAPS LRA.

Discussion: Applicant's response is acceptable to staff. No further action is needed.

RAI 3.1.3.1-2

1) In Table 3.1-1 of the LRA, the applicant has identified cumulative fatigue damage as an aging effect for "other nozzles." According to Table 4.3.1-1 of the LRA, "other nozzles" appears to include RPV recirculation inlet and outlet nozzles. Verify that "other nozzles" includes the RPV recirculation inlet and outlet nozzles. Provide justification for not identifying cumulative fatigue damage as an aging effect for the remaining RPV nozzles (e.g., core spray nozzle).

2) In Table 3.1-1 of the LRA, the applicant does not identify cumulative fatigue damage as an aging effect for nozzle safe-ends. However, BWRVIP-74 states that fatigue usage factors for safe-ends follow the same pattern as nozzles. Table 4.3.1-1 of the LRA includes RPV core spray nozzle safe-end as a fatigue monitoring program location. Provide technical justification for not including cumulative fatigue damage of safe-ends as an aging effect in Table 3.1-1.

Response to 3.1.3.1-2:

a)The applicant stated that Table 4.3.1-1 is a listing of the fatigue monitoring program locations. This table does not list all RPV nozzles and safe ends for which a fatigue analysis is a TLAA and for which cumulative fatigue damage is an aging effect, but only those locations expected to be used by the fatigue monitoring program.

All locations with a design-basis predicted 40-year CUF of 0.4 or greater are included, plus the highest usage factor in an analysis segment if less than 0.4, plus locations which field or industry experience suggest, including some in B31.1 piping, plus ECCS locations important to postulated accident scenarios. Tracking the fatigue usage factor for these locations will ensure that fatigue effects at all other locations with lower predicted usage factors will remain within acceptable limits.

Both "other nozzles" and safe ends are in fact included in the evaluation of RPV fatigue, as described in LRA Section 4.3.1. Therefore "other nozzles" in Table 3.1-1 is both correct and inclusive.

b)Table 3.1-1 should have stated that safe ends are included in the evaluation of RPV fatigue, as indicated in the response to a), above.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-3

Table 3-1 of BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," identifies cumulative fatigue as an aging effect for vessel flanges and stabilizer brackets. But Table 3.1-1 of the LRA does not identify cumulative fatigue damage as an aging effect for these two components. Provide the technical basis for excluding cumulative fatigue damage as an aging effect for these two components.

Response to 3.1.3.1-3

The applicant stated that stabilizer brackets are not included in the list of 17 components in Table 3-1 of BWRVIP-74, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines . However, BWRVIP-74, section 3.2.2.6 identifies stabilizer brackets under vessel external attachments as having a potential for a significant fatigue usage. Our review of the original RPV vendor calculations indicates that the CUF for the Peach Bottom Units 2 & 3 stabilizer brackets was 0.17 for a 40-year life, which is significantly under 1.0 and therefore is not an issue for Peach Bottom license renewal. {Reference: GE e-mail dated 11/2/01, attaching cover page of Report # 10 Stress Analysis of the Internal Brackets and Brackets on the Shell, dated September, 1970 with the results page.}

The BWRVIP-74 table 3-1 does identify cumulative fatigue damage as an aging effect for the vessel flanges. However, BWRVIP-74, in section 3.2 on fatigue, under subsection 3.2.2, vessel flange is not identified as one of the locations of significance. Moreover in the fatigue mechanism discussion in section 3.2.1 of the BWRVIP-74, concluding paragraph states, The variation in calculated value for the vessel shell and the vessel flange strongly suggests that the assumptions used in these analyses vary widely. By using consistent and realistic assumptions, low cumulative usage factors will most likely result [23]. Reference 23 is the BWR RPV License Renewal Industry Report, Revision1, EPRI Report TR-103836, July 1994.

EPRI Report TR-103836 discusses fatigue in vessel flange in subsection 4.2.2.9. This subsection discusses thermal and mechanical fatigue cycling of the vessel flange, sampling of fatigue usage factors, and more detailed calculation results. The concluding paragraph states, The low fatigue usage factors, coupled with successful operating experience, leads to the conclusion that fatigue will not be a significant age related degradation mechanism for the vessel flange during the license renewal period. This conclusion applies to all vessel flange designs.

Therefore, Table 3.1-1 of the PBAPS LRA does not identify cumulative fatigue damage as an aging effect for vessel flange.

Discussion: Applicant’s response should include the calculated vessel flange CUF for the Peach Bottom. The staff will issue a RAI.

RAI 3.1.3.1-4

Void swelling is not identified as an aging effect for any component of the reactor pressure vessel and internals. The applicant is requested to supply the peak neutron fluence for the reactor internals at the end of the license renewal term. Using this neutron fluence as basis, provide data

that indicates void swelling is not an aging effect during the license renewal term. If it is an aging effect, identify the aging management program that will ensure the function of the internals is not degraded (result in cracking or change in critical dimensions) during the license renewal term.

Response:

The applicant stated that void swelling is not an aging effect. Rather, it is an aging mechanism, and the effects of concern would be swelling or cracking. EPRI TR-107521, Generic License Renewal Technical Issues Summary , EPRI, April 1998, addresses data gathered from Liquid-Metal-Cooled Fast Breeder Reactors (LMFBRS), and how it may possibly be related to a PWR component (baffle-former bolt) that is in almost direct contact with the fuel in a PWR. A BWR does not have components located in a similar location, and thus, can reasonably be expected to experience less fluence. Past studies of void swelling by ANL, ORNL, HEDL and GE have shown that the threshold fluence for void swelling is approximately 1 $X10^{22}$ n/cm^2-, which is well in excess of the fluences experienced by BWR components. Secondly, the EPRI report notes that field experience does not support void swelling being a significant issue. The lowest temperature for which this phenomenon is conjectured to occur is 300^0C (572^0F), which is higher than the internals that either Peach Bottom unit will experience. Further, the RPV and Internals ISI program that implements the NRC staff approved BWRVIP program for BWR internals addresses the key aspects of the internals components and provides inspection criteria where appropriate to manage aging. The BWRVIP Program that is implemented at Peach Bottom plant is adequate to address aging of the internals.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-5

Table 3.1-1 of the LRA indicates that the CASS components in jet pump assemblies and CASS fuel supports have no aging effects requiring management because the ferrite content is less than 20 vol.%. However, according to the criteria stated in the May 19, 2000, NRC letter from C. I. Grimes to D. Walters, if the molybdenum content of these components is not low (@0.5 wt.%) and the ferrite content is greater than 14 vol.%, these components are considered susceptible to thermal embrittlement. CASS components with niobium are also considered susceptible to thermal embrittlement.

For all CASS components that are susceptible to significant thermal embrittlement, the applicant may perform a flaw tolerance analysis. The flaw tolerance analysis should follow the methodology and criteria in Code Case N-481. Piping and reactor vessel internals that are potentially susceptible to thermal embrittlement and can not satisfy the flaw tolerance criteria must be inspected with a technique that is capable of detecting a quarter thickness crack with a 6-to-1 aspect ratio in the CASS component.

Describe which CASS components are susceptible to thermal embrittlement and will require a flaw tolerance analysis? Describe the proposed aging management program for components that are susceptible to thermal embrittlement and cannot demonstrate adequate flaw tolerance?

Response to 3.1.3.1-5:

The applicant stated that research on the jet pump assembly and orificed fuel support materials indicates that they were manufactured to the low moly ASTM SA 351, Grade CF-8. All of these

castings at Peach Bottom are statically cast, except the jet pump inlet-mixer adapter castings that are centrifugally cast. Calculated delta ferrite percentages (based on ASTM A800 and the certified material test reports) indicate that the maximum percentage of delta ferrite in any of the statically cast components is below 20%.

According to Table 2, CASS Thermal Aging Susceptibility Screening Criteria, stated in the May 19, 2000 NRC letter from C.I.Grimes to D.Walters, grade CF8, low Moly content and <20% delta ferrite material are not susceptible to thermal aging for statically or centrifugally cast components. Table 3.1-1 of the LRA reflects this result.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-6

The CASS components in the jet pump assemblies and CASS fuel supports are susceptible to neutron irradiation embrittlement if these components experience neutron fluence greater than 10^{17} n/cm2. The applicant is requested to supply data about whether neutron fluence experienced by the CASS components during the license renewal period will exceed 10^{17} n/cm2 fluence. If so, then the applicant should provide an aging management program to manage irradiation embrittlement in these CASS components or provide the basis for the conclusion that neutron irradiation embrittlement need not be managed.

Response to 3.1.3.1-6

The Plant Hatch License Renewal Safety Evaluation Report, October 2001, section 3.2.3 staff evaluation of effects of aging for reactor assembly system under neutron and thermal embrittlement acknowledges that irradiation embrittlement of CASS components becomes a concern only if cracks are present in the components. Our review of the industry experience and plant experience has not identified any cracking in these components. Further, the BWRVIP-41 report, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines , requires inspection of several jet pump assembly welds, which are more susceptible to cracking than the CASS components and will therefore serve as an indication of the potential need for more extensive inspections later in life.

In the case of the orificed fuel support (OFS), irradiation embrittlement may make the OFS more susceptible to cracking from impact loads, such as a dropped fuel bundle. Since this is event related, corrective action would include inspection for damage prior to resuming operation. As such, no aging management program is necessary to manage the effects of irradiation.

The BWRVIP guidelines are implemented at PBAPS through the Reactor Pressure Vessel and Internals ISI program, which is augmented to the PBAPS 10-year ISI program. The PBAPS LRA, Appendix B.2.7, RPV and Internals ISI Program credits BWRVIP-41 for inspection of jet pump assembly.

Based on the above, we believe the RPV and Internals ISI program adequately manages the aging effects of irradiation embrittlement.

Discussion: Applicant's response is acceptable to staff. However, the staff indicated that the applicant need to provide a basis for not requiring additional inspection for cracks in the OFS. The staff will issue a RAI.

RAI 3.1.3.1-7

The application identifies cracking as an aging effect for stainless steel components in the reactor pressure vessel instrumentation system exposed to reactor coolant environment. Identify whether the cracking results from stress corrosion or thermal fatigue, identify the butt weld locations within the system and the pipe size for all effected components. For components less than 4 inches in diameter identify whether the components are susceptible to stress corrosion cracking or thermal fatigue resulting from turbulent penetration or thermal stratification, and identify the inspection method for detecting the cracking.

Response to 3.1.3.1-7

The applicant stated that the RPV Instrumentation system is not prone to sudden changes in temperature which could cause high cycle fatigue and therefore is not susceptible to thermal fatigue resulting from turbulent penetration or thermal stratification.

Stress Corrosion Cracking: The RPV instrumentation system piping is 2 and less in diameter and does not have any butt weld connections. The majority of the piping in this system is 1 and less. The NRC staff concern has excluded socket welded pipe and fittings as indicated in the resolution of Open item 3.2.3.2.3-1 of plant Hatch SER. The Aging Management Activities identified for managing cracking due to SCC are Reactor Coolant System Chemistry (Appendix B.1.2) and ISI (Appendix B.1.8) as defined in the PBAPS LRA Table 3.1-3. The ISI program requires system hydro testing for this system in accordance with Section XI of the ASME Code. We believe these two programs are adequate in managing cracking due to SCC in 2 and less diameter reactor coolant pressure boundary piping.

Discussion: The applicant response is not sufficient because it lacked adequate details for piping greater than 2-inches and less than 4-inches. Therefore, the staff is requesting for more detail information in the following RAI:

The applicant identifies cracking as an aging effect for stainless steel components in the reactor coolant system exposed to reactor coolant environment. Identify whether the cracking results from stress corrosion or thermal fatigue, identify the butt weld locations within the system and the pipe size for all effected components. For components less than 4 inches in diameter, identify whether the components are susceptible to stress corrosion cracking or thermal fatigue resulting from turbulent penetration or thermal stratification, and identify the aging management program for detecting cracking.

RAI 3.1.3.1-8

The applicant has not identified cracking as an aging effect for carbon steel piping in the reactor pressure vessel instrumentation system. The applicant is requested to present an evaluation of industry experience and plant specific experience assessing whether cracking due to cyclic loading from vibration or thermal loads resulting from turbulent penetration or thermal stratification is an aging effect for the carbon steel piping in the reactor pressure vessel instrumentation system.

Response to RAI 3.1.3.1-8

The applicant stated that cyclic loading includes vibrational fatigue and high cycle thermal fatigue. Vibrational fatigue is the result of a design deficiency. Failures due to vibration are typically detected

early in component service life and actions to prevent recurrence are taken. Vibrational fatigue is not age related. Vibrational fatigue is not an applicable aging mechanism for extended operation.

The only carbon steel piping in the reactor pressure vessel instrumentation system is the wide range level instrument tap coming off of the two-inch carbon steel reactor head vent line. This one-inch line is carbon steel from the two-inch head vent line to a flange connection with the stainless steel instrument line. The subject one inch carbon steel pipe is a six inch long nipple, installed in a socket welded reducing bushing in a two inch socket welded tee in the reactor head vent line. The tee is located a short distance from the vessel head vent flange connection. Peach Bottom has not experienced cracking in this line. The review of industry experience has not identified an issue with cracking of this line.

This small section of carbon steel piping is normally exposed to a saturated steam environment. This is a static instrument line installed on the steam side of the level instrument condensing chamber, and is installed to allow any condensate to flow back to the reactor vessel. The condensate would be at the same saturated steam temperature. Therefore, this piping is not subject to any high cycle thermal fatigue.

Discussion: Applicant's response is acceptable to staff. No further action is needed.

RAI 3.1.3.1-9

The applicant has identified loss of material as an aging effect for stainless and carbon steel components in the reactor pressure vessel instrumentation system exposed to reactor coolant. Describe how these components that are susceptible to loss of materials will be inspected as part of the ISI program.

Response to RAI 3.1.3.1-9

Table 3.1-3 does not identify any carbon steel components in the reactor pressure vessel instrumentation system exposed to reactor coolant. The subject components exposed to reactor coolant are constructed of stainless steel. As indicated in Table 3.1-3, the RCS Chemistry (LRA Appendix B.1.2) and ISI Program (LRA Appendix B.1.8) aging management activities manage this aging effect. The ISI Program aging management activity includes periodic hydrostatic pressure tests that confirm the integrity of the reactor pressure vessel instrumentation system piping and components.

Discussion: The applicant's response is not sufficient because it lacked adequate details. Therefore, the staff is requesting for more detail information in the following RAI:

The applicant identifies loss of material as an aging effect for stainless and carbon steel components in the reactor pressure vessel instrumentation system. The applicant identifies (a) RCS Chemistry Program to mitigate this aging effect and (b) ISI Program, which includes periodic hydrostatic pressure tests, to confirm the integrity of these components. These pressure tests are not adequate to confirm the effectiveness of the RCS Chemistry Program to prevent loss of material. Please describe an aging management program to confirm the effectiveness of the RCS Chemistry Program, i.e., to confirm that the stainless steel and carbon steel components in the reactor pressure vessel instrumentation system are not susceptible to loss of material.

RAI 3.1.3.1-10

(a) Loss of material due to galvanic corrosion can occur when two dissimilar metals (i.e., carbon steel and stainless steel) are in contact in the presence of oxygenated water. The applicant is requested to identify whether the carbon steel piping of the reactor pressure vessel instrumentation system is connected to stainless steel components. If so, then does the aging effect of loss of material include damage due to galvanic corrosion? How will the ISI program presented in Section B.1.8 of the LRA manage this aging effect?

(b) The applicant is requested to identify whether the carbon steel piping of the reactor recirculation system is connected to stainless steel components. If so, then does the aging effect of loss of material include damage due to galvanic corrosion? How will the ISI program presented in Section B.1.8 of the LRA manage this aging effect?

Response to RAI 3.1.3.1-10(a)

The applicant stated that the steam side of the wide range level instrument tap comes off the reactor head vent line. This instrument line is carbon steel from the head vent line to a flange connection with a stainless steel instrument line. The aging effect of loss of material includes potential damage due to galvanic corrosion. As indicated in Table 3.1-3, the RCS Chemistry (LRA Appendix B.1.2) and ISI Program (LRA Appendix B.1.8) aging management activities manage this aging effect. The RCS Chemistry aging management activity monitors and controls conductivity, which acts to minimize the rate of galvanic corrosion. Industry and plant operating experience has determined that galvanic corrosion has not been a problem for boiling water reactors within the reactor coolant pressure boundary. The ISI Program aging management activity includes periodic hydrostatic pressure tests that confirm the integrity of the flanged connection.

Response to RAI 3.1.3.1-10(b)

The applicant stated that the only carbon steel piping and valves included in the Reactor Recirculation system are the piping and valves associated with the reactor vessel bottom head drain. The bottom head drain line is a 2-inch carbon steel line from the reactor bottom head to a connection with a 2-inch stainless line. The aging effect of loss of material includes potential damage due to galvanic corrosion. As indicated in Table 3.1-4, the RCS Chemistry (LRA Appendix B.1.2) and ISI Program (LRA Appendix B.1.8) aging management activities manage this aging effect. The RCS Chemistry aging management activity monitors and controls conductivity, which acts to minimize the rate of galvanic corrosion. The ISI Program aging management

activity includes periodic hydrostatic pressure tests that confirm the integrity of the piping connections.

Discussion: The applicant's response is not sufficient because it lacked adequate details. Therefore, the staff is requesting for more detail information in the following RAIs:

RAI 3.1.3.1-10(a)

Loss of material due to galvanic corrosion can occur when two dissimilar metals (i.e., carbon steel and stainless steel) are in contact in the presence of oxygenated water. The applicant is requested to identify whether the carbon steel piping of the reactor pressure vessel instrumentation system is connected to stainless steel components. If so, then does the aging effect of loss of material include damage due to galvanic corrosion? The applicant has identified the RCS Chemistry Program to mitigate this aging effect. Please describe an aging management program to confirm the effectiveness of the RCS Chemistry Program to prevent loss of material from galvanic corrosion. (See RAI 3.1.3.1-9)

RAI 3.1.3.1-10(b)

The applicant is requested to identify whether the carbon steel piping of the reactor recirculation recirculation system is connected to stainless steel components. If so, then does the aging effect of loss of material include galvanic corrosion? The applicant has identified the RCS Chemistry Program to mitigate this aging effect. Please describe an aging management program to confirm the effectiveness of the RCS Chemistry to prevent loss of material from galvanic corrosion.

RAI 3.1.3.1-11

The valve bodies, valve bonnets, and valve closure bolting in the reactor pressure vessel instrumentation system are subject to ASME Code fatigue analysis. But the applicant has not identified cumulative fatigue damage as an aging effect for these components. Provide the technical basis for not considering cumulative fatigue damage as an aging effect for these components.

Response to 3.1.3.1-11

The applicant stated that RPV Instrumentation system piping is designed to the requirements of ANSI B31.1. This code applies only to piping and does not require an explicit fatigue analyses. Therefore, CUF values were not calculated for this system piping. PBAPS LRA section 4.3.3 addresses piping and component fatigue and thermal cycles for piping designed to requirements of ANSI B31.1.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-12

According to NUREG-0313, Rev. 2, a CASS component is susceptible to stress corrosion cracking if the carbon content is greater than 0.035 wt% or ferrite content less than 7.5 vol.%. In a statically cast CASS component (i.e., pump casing), the ferrite distribution is not uniform and could be below 7.5 vol.% at some locations on the inside surface of the component. In addition, if the ferrite content of the weld metal used to perform repair at the inside surface of the pump casing is less than 7.5 vol.%, the pump casing is susceptible to stress corrosion cracking. The applicant is requested to present technical justification for not including cracking as an aging effect for the CASS pump casings in the reactor recirculation system.

Response to RAI 3.1.3.1-12

The applicant stated that cracking is considered an applicable aging effect for the pump casings in the Reactor Recirculation system. This aging effect was inadvertently excluded from LRA Table 3.1-4. In the first row of Table 3.1-4, the Component Group Casting and Forging should include both Pump Casings and Valve Bodies. The aging effect of cracking will be managed by the RCS Chemistry and ISI Program aging management activities.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-13

The applicant is requested to present an evaluation of the BWR industry-wide response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." The staff would specifically like to know whether the applicant, in response to the NRC Bulletin, identified any unisolable sections of piping connected to the Peach Bottom RCS that can be subjected to stresses from temperature stratification or temperature oscillations induced by leaking valves. The staff needs this information to assess the effectiveness of the ISI Program, presented in Section B.1.8 of the LRA, to manage cracking of the reactor coolant system components.

Response to RAI 3.1.3.1-13

The applicant stated that Exelon response to NRC Bulletin 88-08 was provided to the NRC by letter dated September 16, 1988. As indicated in the response, the design of the Peach Bottom station does not contain any unisolable sections of piping that are potentially subject to thermal cycling fatigue from cold water leaks into the RCS during normal operation. The response concludes that the Peach Bottom station does not contain any unisolable sections of RCS piping that can be subject to stresses of the type defined in the Bulletin.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.1-14

The applicant is requested to present the Peach Bottom experience related to loss of material in carbon steel and stainless steel components in the reactor recirculation system. Does industry experience indicate that the carbon steel piping and valve bodies in the reactor recirculation system experience loss of material due to flow-accelerated corrosion? If so, describe how loss of

material in the affected components would be managed by the Flow Accelerated Corrosion Program described in Section B.1.1 of the LRA.

Response to RAI 3.1.3.1-14

The applicant stated that the only carbon steel piping and valves included in the Reactor Recirculation system are the piping and valves associated with the reactor vessel bottom head drain. This is a two-inch carbon steel line that includes a vent connection (PBAPS Unit 3 only) that uses normally closed carbon steel valves. There is no industry or Peach Bottom experience with flow accelerated corrosion (FAC) in this line. The normal flow rate in this line is less than 6 feet per second, so based on EPRI guidance in NSAC-202L, FAC would not be expected to occur. The Peach Bottom LRA does not credit the FAC Program for aging management in the Reactor Recirculation System.

Discussion: Applicant's response is acceptable to staff. No further action is needed.

RAI 3.1.3.1-15

The applicant is requested to submit justification for not identifying loss of material due to wear as an aging effect for recirculation pump seal flange in the reactor recirculation system.

Response to RAI 3.1.3.1-15

The applicant stated that the recirculation pump seal flange is considered a subcomponent of the pump casing. Removal and installation of the recirculation pump seal cartridge is controlled by a maintenance procedure, which also includes pressure testing of the seal cartridge. The seal flange is a machined surface sealed with an O-Ring seal. The seal cartridge is installed and removed carefully to avoid damage to the pump shaft. The seal cartridge flange is inspected for wear during the seal rebuild, in accordance with the rebuild maintenance procedure. This is considered a routine maintenance practice, and not an aging management activity. The seal flange is not subject to significant wear in the sheltered environment or in the reactor coolant environment. Flange bolting is addressed by response to RAI 3.0-1.

Discussion: Applicant's response is acceptable to staff. No further action is needed.

RAI 3.1.3.1-16

The applicant has identified loss of material as an aging effect for stainless steel in the reactor recirculation system exposed to reactor coolant. Describe how these components that are susceptible to loss of material be inspected as part of the ISI program.

Response to RAI 3.1.3.1-16

The applicant stated that in Table 3.1-4, the RCS Chemistry (LRA Appendix B.1.2) and ISI Program (LRA Appendix B.1.8) aging management activities manage this aging effect. The ISI Program aging management activity includes periodic inspections and hydrostatic pressure tests that confirm the integrity of the reactor recirculation system piping and components.

Discussion: Applicant's response is acceptable to staff. No further action is needed.

RAI 3.1.3.1-17

Components of the reactor recirculation system, such as piping and recirculation pump subcomponents (casing, cover, seal flange and closure bolting, valve bodies, bonnets and closure bolting) are subject to cumulative fatigue damage due to plant heatup, cooldown, and other operational transients. Cumulative fatigue damage has not been identified as an aging effect for any of the component in the reactor recirculation system. Provide the technical basis for excluding cumulative fatigue damage as an aging effect for the reactor recirculation system components that are within the scope of license renewal.

Response to RAI 3.1.3.1-17

The applicant stated that cumulative fatigue damage has been addressed in TLAA Section 4.3.

Cumulative fatigue for Reactor recirculation piping designed to ASME Section III, class 1 requirements is addressed in the TLAA section 4.3.3.1. Reactor recirculation system piping designed to the requirements of ANSI B31.1 does not require an explicit fatigue analyses. Therefore, CUF values were not calculated for this system piping. PBAPS LRA section 4.3.3.2 addresses piping and component fatigue and thermal cycles for piping designed to requirements of ANSI B31.1

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.2-1

a) The applicant s reactor coolant system chemistry program is based on the guidance presented in EPRI TR-103515, "BWR Water Chemistry Guidelines, 2000 Revision." The reviewers note that the staff has not approved EPRI TR-103515, 2000 Revision, for generic use. The latest revision reviewed by the staff is the 1996 revision (Reference: September 18, 1998 letter from D.S. Hood, NRC to J.H. Mueller, Niagara Mohawk Power Corporation). Therefore, the applicant is requested to identify the changes in the water chemistry program that result from the use of the guidelines from the1996 Revision to the 2000 Revision of EPRI TR-103515.

b) To determine the effectiveness of the EPRI TR-103515 BWR water chemistry guidelines, identify components at Peach Bottom that have had stress corrosion cracking or loss of material since the EPRI TR-103515 water chemistry guidelines were instituted at Peach Bottom. Identify the changes in water chemistry that have been instituted to eliminate or mitigate cracking or loss of material in these components.

c) The reactor coolant system chemistry AMP, presented in Section B.1.2 of the LRA, continuously monitors coolant conductivity, and measures the impurities such as chlorides and sulfates only when the conductivity measurements indicate presence of abnormal conditions. Does EPRI TR 103515, 2000 Revision guidelines require that the sulfates and chlorides be measured daily and continuous monitoring of the desolved oxygen concentration in the reactor feedwater/condensate system and the control rod drive water? The applicant is also requested to

provide information about whether normal or hydrogen water chemistry is employed at Peach Bottom plants because it determines the parameters to be monitored. When hydrogen water chemistry (HWC) is in service, does EPRI TR-103515, 2000 Revision requires that electrochemical potential (ECP) be monitored?

Response to RAI 3.1.3.2-1

a) The applicant stated that the Peach Bottom Atomic Power Stations (PBAPS) reactor water chemistry control program is based on EPRI TR-103515, "BWR Water Chemistry Guidelines 2000 Revision. PBAPS believes that it is important to maintain the flexibility to modify its plant chemistry control program based on the collective industry operating experience of similar reactors. Therefore, over time, PBAPS expects to revise its plant chemistry control program to reflect changes in industry guidance presented in the EPRI BWR Water Chemistry Guidelines (TR-103515).

The 2000 revision of EPRI BWR Water Chemistry Guidelines differs from the 1996 revision in the following areas for Reactor Water Power operation:

1. In the 2000 revision to the EPRI BWR Water Chemistry Guidelines, chlorides and sulfates no longer need to be measured on a daily basis provided that reactor water conductivity is trended to ensure that the action level 1 limits are not exceeded. At PBAPS, chloride and sulfate are measured 3 times a week, provided that reactor water conductivity remains below an administrative limit, which was set to assure that chlorides and sulfates action level 1 limits are not exceeded. This provides adequate assurance that chloride and sulfate levels are controlled below action level 1 limits. If the reactor water conductivity exceeds its administrative limit, chloride and sulfate sampling frequency is increased based on the significance of the transient. In this case, sampling frequency is at least once per day.

2. In the 2000 revision to the EPRI BWR Water Chemistry Guidelines, plants with HWC or HWC with Noble Metals Chemical Addition (NMCA) no longer need to measure ECP on a continuous basis. Even in the 1996 version of the EPRI BWR Water Chemistry Guidelines, alternate methods (e.g., Main Steam Line Radiation) could be used for estimating ECP. PBAPS is a HWC with NMCA plant that uses ECP and alternate methods for estimating ECP. PBAPS is not committed to measure ECP on a continuous basis and would use alternative methods if ECP measurements were not available.

3. The 2000 revision to the EPRI BWR Water Chemistry Guidelines, allows Plants with HWC or HWC with NMCA to go to higher action level 2 and 3 levels for chloride and sulfate. Action level 2 was increased from >20 ppb to > 50 ppb and Action level 3 was increased from >100 ppb to > 200 ppb. This additional flexibility is allowed based on the increased protection of reactor coolant system and reactor assembly components provided by HWC or HWC with NMCA.

4. The 2000 revision to the EPRI BWR Water Chemistry Guidelines, also added Reactor Water Iron as a new diagnostic parameter to its Reactor Water Chemistry Guidelines. PBAPS has implemented this change.

b) As stated in LRA Appendix B1.2 attribute 10, As chemistry control guidelines were evolving in the industry, PBAPS experience with reactor coolant system chemistry was similar to that of the industry. Cracking attributed to IGSCC was found in stainless steel recirculation and RHR system

piping and loss of material was found in the HPCI and RCIC carbon steel steam line drains. Portions of the 304 stainless steel recirculation system, RWCU, and RHR piping were replaced with more IGSCC resistant, low carbon, 316 stainless steel. The HPCI and RCIC steam drain lines were also replaced.
The RCS water chemistry is maintained based on the recommendations of EPRI TR-103515 that have been developed based on industry experience. These recommendations have been shown to be effective and are adjusted as new information becomes available. Since the pipe replacement and improvements to chemistry activities, the overall effectiveness of RCS chemistry activities is supported by the excellent operating experience of reactor coolant and main steam systems at PBAPS. For example, no IGSCC cracking has been identified in the recirculation system piping since it was replaced in 1985 and 1988. PBAPS implemented the EPRI chemistry guidelines in 1986 and has continued to revise plant procedures as the guidelines are updated. PBAPS uses the BWRVIP program to monitor the condition of reactor vessel internals. An annual summary report is sent to the NRC from the BWRVIP with results of BWR plant inspections.

c) Chloride and sulfate measurement frequency changes are discussed in part a of this RAI response. The described analysis process will provide adequate assurance that chloride and sulfate levels are controlled below action level 1 limits.

PBAPS does have a continuous dissolved oxygen monitor on the condensate, feedwater and reactor water systems. Since under normal operations control rod drive water comes from the condensate system, an additional dissolved oxygen monitor is not provided on the control rod drive water system.

PBAPS is a HWC plant with NMCA applied. As described in part a of this RAI response, the 1996 revision of the EPRI BWR Water Chemistry Guidelines does not require that ECP be measured on a continuous basis, alternate methods (e.g., Main Steam Line Radiation) could be used for estimating ECP. PBAPS is not committed to measure ECP on a continuous basis and would use alternative methods if ECP measurements were not available.

Discussion: The applicant has provided acceptable response to the staff's RAI except that they have not indicated when EPRI TR 103515, 2000 Revision requires continuous monitoring of the developed oxygen concentration in the feedwater/condensate system and the control rod drive water. The staff will issue RAI that addresses this concern.

RAI 3.1.3.2-2

In Section B.2.7, "Reactor Pressure Vessel and Internals ISI Program, of the LRA, the applicant stated that the vessel internals requiring aging management within the scope of license renewal are shroud, shroud supports, access hole covers, core support plate, core q P/SLC line, top guide, core spray piping and spargers, control rod guide tubes, jet pump assemblies, CRDH guide tubes, in-core housing guide tubes, and dry tubes. The applicant has not submitted information about any repair to core shroud or other internals, but NUREG-1544, "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components," published in 1994, refers to the PECO Energy Company's submittal of the Peach Bottom core shroud repair designs to NRC for review. The applicant is requested to provide information about whether the Peach Bottom core shrouds and other internals have been repaired, and if so then whether the repair hardware for those components is within the scope of the reactor pressure vessel and internals ISI program.

Response to RAI 3.1.3.2-2 - The applicant stated that NUREG-1544 provided the shroud repair modification for NRC review. But this repair was not implemented at either Peach Bottom units.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.3.2-3

To evaluate whether the reactor materials surveillance program presented in Section B.1.12 of the LRA provides sufficient data for monitoring the extent of neutron irradiation embrittlement during the license renewal period, the staff requests that the applicant determine whether the existing Peach Bottom reactor surveillance program or the integrated surveillance program would be revised to satisfy the following attributes:

> Capsules shall be removed periodically to determine the rate of embrittlement and at least one capsule with a neutron fluence not less than once or greater than twice the peak beltline neutron fluence must be removed before the expiration of the license renewal period.
>
> Capsules shall contain material to monitor the impact of irradiation on the limiting beltline materials and must contain dosimetry to monitor neutron fluence.
>
> If capsules are not being removed from Peach Bottom during the license renewal period, the applicant shall supply operating restrictions (i.e., inlet temperature, neutron spectrum and flux) to ensure that the RPV is operating within the environment of the surveillance capsules, and must supply ex-vessel dosimetry for monitoring neutron fluence.

The applicant has indicated in Section B.1.12 of the LRA that it plans to implement the provisions of the Integrated Surveillance Program (ISP) as described in BWRVIP-78. The staff requests that the applicant provide the schedule for implementing the ISP at Peach Bottom. The staff also request that the applicant indicate how the proposed ISP would satisfy the ISP criteria in Appendix H, 10 CFR Part 50 and the attributes discussed above.

Response to RAI 3.1.3.2-3

The applicant stated that the BWRVIP has developed an ISP and submitted it to NRC for review and approval. The ISP is documented in BWRVIP-78, BWR Vessels and Internals Project: BWR Integrated Surveillance Program Plan, issued December 1999, and it's companion document, BWRVIP-86, BWR Vessels and Internals Project: BWR Integrated Surveillance Program Implementation Plan. One of the provisions of the ISP is for surveillance capsule material withdrawal and testing during the license renewal period. As noted in section 2.1 of BWRVIP-78, the ISP complies with the provisions of 10CFR50 Appendix H. The ISP currently provides for 13 capsules to be available for testing during the renewal period for the BWR fleet.

Exelon is aware of the provisions of Appendix H, and understands that the RPV must be operated within parametric limits that assure vessel integrity with regard to embrittlement and fracture toughness. However, there is not yet a demonstrated need to provide operating restrictions. Should the ISP be approved by the NRC, PBAPS will be bounded by the 13 representative capsules that are available for testing during the renewal period for the BWR fleet.

Exelon plans to implement the provisions of the ISP currently described in BWRVIP-78 and BWRVIP-86. Should the ISP not be approved by the NRC, or it should be modified such that Peach Bottom is not covered by the ISP, then Exelon will develop a RPV material surveillance program for the period of extended operation. This plant-specific program, if needed, will include the following actions:

Capsules will be removed periodically to determine the rate of embrittlement and at least one capsule with a neutron fluence not less than once or greater than twice the peak beltline neutron fluence will be removed before the expiration of the license renewal period. Capsules will contain material to monitor the impact of irradiation on the limiting beltline materials and must contain dosimetry to monitor neutron fluence.

If capsules are not being removed from Peach Bottom during the license renewal period, the applicant will supply operating restrictions (i.e., inlet temperature, neutron spectrum and flux) to ensure that the RPV is operating within the environment of the surveillance capsules, and must supply ex-vessel dosimetry for monitoring neutron fluence.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.4-1 - UFSAR Update

The reviewers found that the applicant presents adequate general information of the reactor coolant system chemistry program in Section A.1.2 of the LRA but does not identify the supporting documents (e.g., EPRI water chemistry guidelines). The applicant is requested to include in the UFSAR update the supporting documents by reference in Section A.1.2. The revision of the water chemistry guidelines need not be included in the UFSAR update.

Response to RAI 3.1.4-1

The applicant stated that Section A.1.2 is revised to read as follows:

A.1.2 Reactor Coolant System Chemistry

PBAPS reactor coolant system (RCS) chemistry activities manage loss of material and cracking of components exposed to reactor coolant and steam through measures based on EPRI TR-103515, BWR Water Chemistry Guidelines, that monitor and control reactor coolant chemistry. These activities include monitoring and controlling of reactor coolant water chemistry to ensure that known detrimental contaminants are maintained within pre-established limits. Reactor coolant is monitored for indications of abnormal chemistry conditions. If such indications are found, then measurements of impurities are conducted to determine the cause, and actions are taken to address the abnormal chemistry condition. Whenever corrective actions are taken to address an abnormal chemistry condition, sampling is utilized to verify the effectiveness of these actions. The RCS chemistry activities provide reasonable assurance that intended functions of components exposed to reactor coolant and steam are not lost due to loss of material or cracking aging effects.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

RAI 3.1.4-2 - UFSAR Update

The applicant describes the Reactor Materials Surveillance Program as an existing program in Section A.1.12 of the LRA and identifies the supporting documents (10 CFR 50, Appendix H and ASTM E18) by reference. The applicant is requested to include information about the BWR Integrated Surveillance Program, which it intends to use at Peach Bottom.

Response to RAI 3.1.4.2 The applicant stated that see Appendix A response.

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.

4.1 Identification of TLAAs

RAI 4.1-1

Table 4.1-1 of the LRA identifies flaw growth analysis as a TLAA for feedwater nozzle and control rod drive return line nozzle. The table does not identify the flaw growth analyses for other reactor coolant pressure boundary components as TLAAs. Flaws in Class 1 components that exceed the size of allowable flaws defined in IWB-3500 of the ASME Code need not be repaired if they are analytically evaluated to the criteria in IWB-3600 of the ASME Code. The analytic evaluation requires the applicant to project the amount of flaw growth due to fatigue and stress corrosion cracking mechanisms, or both, where applicable, during a specified evaluation period. The applicant is requested to identify all Class 1 components that have flaws exceeding the allowable flaw limits defined in IWB-3500 and that have been analytically evaluated to IWB-3600 of the ASME Code, and to submit the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 for the period of extended operation.

Response to 4.1-1

The applicant stated that as part of the effort to identify all potential TLAAs Exelon reviewed all preservice and inservice inspection summary reports. Exelon reviewed all dispositions, which might have included an IWB-3600 evaluation.

The only other flaw evaluated with time-dependent methods similar to IWB-3600 for the licensed operating period is a laminar indication in a Unit 3 Main Steam elbow. See Section 4.7.3 of the License Renewal Application, which describes the condition, the original fatigue calculation, and the basis for its validation for the extended licensed operating period.

No other flaws evaluated with time-dependent methods similar to IWB-3600 extended to the end of the current licensed operating period, and therefore no other flaw evaluations met Criterion 3, "Does the analysis involve time-limited assumptions defined by the current operating term, for example, 40 years?"

Discussion: Applicant's response is acceptable to staff. The staff will issue a RAI.