4 TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

4.1.1 Introduction

The applicant describes its identification of time-limited aging analyses (TLAAs) in Section 4.1.1, "Identification of Time-Limited Aging Analyses," of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has identified the TLAAs as required by 10 CFR 54.21(c) and described them in its UFSAR Supplement as required by 10 CFR 54.21(d).

In Section 4.1 of the application, the applicant described the requirements for the technical information to be reported in the application regarding time-limited aging analyses (TLAAs), as stated in 10 CFR 54.21(c). These include a list of TLAAs, as defined in 10 CFR 54.3, "Definitions," and a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 that are based on TLAAs. The applicant also described the criteria used to identify TLAAs at Peach Bottom, Units 2 and 3. These criteria are the same as the six criteria stated in 10 CFR 54.3 for identifying TLAAs.

The identified TLAAs were evaluated and the results are described in Sections 4.1 through 4.7 of this SER. As required by 10 CFR 54.21(c), the applicant has provided a list of TLAAs in Table 4.1-1 of the LRA. The applicant also stated that no plant-specific exemptions based on TLAAs have been granted at Peach Bottom.

4.1.2 Summary of Technical Information in the Application

The applicant evaluates calculations for Peach Bottom against the six criteria specified in 10 CFR 54.3 to identify the TLAAs. The applicant identifies the following TLAAs:

- Reactor vessel neutron embrittlement

  - 10 CFR Part 50 Appendix G reactor vessel rapid failure propagation and brittle fracture considerations: Charpy upper shelf energy (USE) reduction and RT_{NDT} increase, reflood thermal shock analysis
  - Reactor vessel thermal limit analysis: operating pressure-temperature limit (P-T limit) curves
  - Reactor vessel circumferential weld examination relief
  - Reactor vessel axial weld failure probability

- Metal fatigue

  - Reactor vessel fatigue
  - Reactor vessel internals fatigue and embrittlement
  - Reactor vessel internals fatigue analyses
  - Reactor vessel internals embrittlement analyses
  - Effect of fatigue and embrittlement on end-of-life reflood thermal shock analysis

- Piping and component fatigue and thermal cycles
- Fatigue analyses of Group I primary system piping
- Assumed thermal cycle count for allowable secondary stress range reduction in Group II and III piping and components
- Design of the RHR system for a finite number of cycles
- Effects of reactor coolant environment on fatigue life of components and piping (Generic Safety Issue 190)

• Environmental qualification of electrical equipment
• Loss of prestress in concrete containment tendons not applicable
• Containment fatigue

  - Fatigue analyses of containment boundaries: new loads analysis of torus, torus vents, and torus penetrations
  - New loads fatigue analysis of SRV discharge lines and external torus-attached piping
  - Expansion joint and bellows fatigue analyses (drywell-to-torus-vent bellows)
  - Expansion joint and bellows fatigue analyses (containment penetration bellows)

• Other plant-specific TLAAs

  - Reactor vessel corrosion allowances
  - Generic Letter 81-11 crack growth analysis to demonstrate conformance to the intent of NUREG-0619
  - Fracture mechanics of ISI-reportable indications for Group I piping: as-forged laminar tear in a Unit 3 main steam elbow

Pursuant to 10 CFR 50.21(c)(2), the applicant stated that no exemptions granted under 10 CFR 50.12 on the basis of a TLAA were identified. The applicant states that a technical alternative (as defined in 10 CFR 50.55a(a)(3)(i)) to requirements to inspect circumferential welds on the reactor pressure vessel has been approved by NRC. This TLAA is discussed in Section 4.2.3 of this SER.

In a separate licensing action, the applicant has submitted a license amendment for a power uprate to increase the maximum allowed operating power level. This power uprate is based on the increased accuracy of feedwater flow monitors. The higher power level may result in higher reactor coolant temperatures, increased reactor coolant flow, and/or increased neutron fluence. On July 23, 2002, the staff held a conference call with the applicant to ask if the the effects of the power uprate were considered during its evaluation of the TLAAs or that the analysis results are bounding for the higher power level. The applicant stated that the effects of the power uprate were considered. In response to Confirmatory Item 4.1.2-1, by letters dated November 26 and December 19, 2002, the applicant indicated that as part of the power uprate, a separate RPV fracture toughness evaluation was performed. The evaluation confirmed that the combined effects of license renewal and power uprate on fluence, adjusted reference temperature, and upper shelf energy at the end of the license renewal period are bounded by the values provided in the license renewal application Furthermore, no additional aging effects that require management are applicable due to the small increase in steam flow resulting from the power uprate. The applicant has adequately addressed the effects of the power uprate and license renewal by confirming the results of the power uprate and license extension are bounded by the results identified in the license renewal application.

4.1.3 Staff Evaluation

TLAAs are defined in 10 CFR 54.3 as analyses that meet the following six criteria:

- involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a)
- consider the effects of aging
- involve time-limited assumptions defined by the current operating term (for example, 40 years)
- were determined to be relevant by the applicant in making a safety determination
- involve conclusions or present the basis for conclusions related to the capability of the system, structure, or component to perform its intended functions, as delineated in 10 CFR 54.4(b)
- are contained or incorporated by reference in the CLB

In addition, to the TLAAs listed in Section 4.2 through 4.7 of the LRA, the staff identified three other potential TLAAs. The evaluation of these potential TLAAs is provided below.

<u>Flaw Growth Analyses</u>

Feedwater and Control Rod Drive Nozzles

Table 4.1-1 of the LRA identifies flaw growth analysis as a TLAA for feedwater nozzles and control rod drive return line nozzles. The table, however, 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 or stress corrosion cracking mechanisms, or both where applicable, during a specified evaluation period. In RAI 4.1-1, the staff requested the applicant 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 submit the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 for the period of extended operation. In response, the applicant stated that Exelon reviewed all preservice and inservice inspection summary reports as part of the effort to identify all potential TLAAs. 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 (discussed in Section 4.7.3 of the LRA). This section describes the condition, the original fatigue calculation, and the basis for validating the calculation 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. Since no other flaw evaluations met TLAA criteria, the staff find the applicant's response that such flaw evaluations were not TLAAs acceptable.

Pipe Break Locations

The applicant did not identify postulated pipe breaks locations based on the cumulative usage factor (CUF) as a TLAA for Peach Bottom. Although the applicant identified the fatigue usage factor calculation as a TLAA, the applicant did not identify the pipe break criteria as a TLAA. The usage factor calculation used to identify postulated pipe break locations meets the definition of a TLAA as specified in 10 CFR 54.3. In a teleconference on May 6, 2002, the staff requested the applicant to provide a description of the TLAA performed to address the pipe break criteria for Peach Bottom. In addition, the staff requested the applicant to identify any postulated pipe breaks locations based on CUF and describe the TLAA performed for these locations.

The applicant's June 10, 2002, response indicated that pipe breaks had been postulated at Peach Bottom locations where the CUF exceeds 0.1. The applicant also indicated that it did not expect the number of design transients assumed in these CUF calculations to be exceeded in 60 years of plant operation. Therefore, the CUF calculations which form the basis for the Peach Bottom pipe break postulations remain valid for the period of extended operation.

The Peach Bottom Unit 2 recirculation system piping was replaced in 1985-86 and the Unit 3 piping in 1988-89. The replacement was designed to ASME Section III Class 1 requirements. Peach Bottom UFSAR Appendix A.10.3.3 states that for the recirculation system piping, breaks have been assumed to occur at intermediate locations where the cumulative usage factor (CUF) exceeds 0.1. This piping was reanalyzed in 2001 to consider extended operation and no new breaks were identified. The analysis for extended operation used a piping life of 47 years for Unit 2 and 44 years for Unit 3, not 60 years, because the original piping has been replaced. The same screening criterion, 0.1 CUF, was used in all of the analyses. In addition, as identified in LRA Table 4.3.1-1, the reactor pressure vessel recirculation inlet and outlet nozzles and the residual heat removal system tee connections to the recirculation pipe are also included as monitoring locations in LRA Appendix B.4.2, "Fatigue Management Activities."

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

The applicant's November 26, 2202, response to the open and confirmatory items referenced the CUF criteria in UFSAR Section A.10.3.3 used for postulating pipe breaks in the recirculation piping pipe breaks. The applicant also indicated that the reactor pressure vessel recirculation inlet and outlet nozzles and the RHR tee connections to the recirculation line are included in fatigue management program discussed in Section A.4.2 of the UFSAR Supplement. The staff finds that the applicant's UFSAR update contains an appropriate summary description of the activities to evaluate TLAAs related to fatigue as required by 10 CFR 54.21(d).

Crane Load Cycle Limit

In Section 4.1 of the LRA, the applicant did not identify a crane load cycle limit as a TLAA for the cranes within the scope of license renewal. Normally, based on the design code of the crane, a load cycle limit is specified at rated capacity over the crane's projected life. Therefore, it is generally necessary to perform a TLAA relating to crane load cycles estimated to occur up to the end of the extended period of operation.

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

On November 26, 2002, the applicant provided the UFSAR Supplement. In Section A.5.7 of the UFSAR Supplement, the applicant provided a summary description of its evaluation of this TLAA for the period of extended operation. The description contains the basis for determining that the analyses associated with crane design, including the load cycle limit, remain valid for the period of extended operation and therefore, meet the requirements of 10 CFR 54.21(c)(1)(i). On the basis of its review of the information provided in Section A.5.7 of the UFSAR Supplement, the staff concludes that the applicant has provided adequate summary description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d) and therefore, the confirmatory Item 4.1.3-2 is closed.

4.1.4 Conclusions

The staff has reviewed the information provided in Section 4.1 of the Peach Bottom LRA. The NRC staff concludes that the applicant has adequately identified the TLAAs as required by 10 CFR 54.21(c), and that no 10 CFR 50.12 exemptions have been granted on the basis of the TLAA as defined in 10 CFR 54.3. The staff also concludes that the applicant has adequately evaluated the TLAAs related to pipe breaks and the crane load cycle limit as required by 10 CFR 54.21 (c).

4.2 Reactor Vessel Neutron Embrittlement

4.2.1 10 CFR Part 50 Appendix G Reactor Vessel Rapid Failure Propagation and Brittle Fracture Considerations: Charpy Upper Shelf Energy (USE) Reduction and RT_{NDT} Increase, Reflood thermal shock analysis

4.2.1.1 Summary of Technical Information in the Application

The applicant described its evaluation of this TLAA in LRA Section 4.2, "Reactor Vessel Neutron Embrittlement."

Neutron Irradiation Embrittlement

Neutron irradiation causes a decrease in the Charpy upper shelf energy (USE) and an increase in the adjusted reference temperature (ART) of the reactor pressure vessel (RPV) beltline materials. The ART impacts the plant's pressure-temperature (P-T) limit and RPV integrity evaluations. BWRVIP-74 report contains integrity evaluations of the BWR RPV circumferentially oriented welds and the BWR RPV axially oriented welds. Therefore, in order to demonstrate that neutron embrittlement does not significantly impact BWR RPV integrity during the license renewal term, the applicant must determine the end-of-life fluence and the end-of-life RT_{NDT}, determine the validity of the reflood thermal shock analysis, and evaluate the impact of neutron irradiation on the Charpy USE reduction, P-T limits, RPV circumferential welds, and RPV axial welds.

Neutron Fluence and RT_{NDT}

The application does not contain the calculations for determining the end-of-life fluence and end-of-life RT_{NDT}. The application indicates that the applicant will initiate the calculations for end-of-life fluence using the GE fluence methodology after the NRC approves it. Then the applicant will recalculate the vessel end-of-life RT_{NDT} for a 60-year licensed operating life (54 EFPYs) according to Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves [ASME Code] Section XI, Division 1."

Reflood Thermal Shock Analysis

The applicant has reviewed the reflood thermal shock analysis for Peach Bottom. For the reflood thermal shock event, the peak stress intensity at 1/4 of vessel thickness from inside occurs at about 300 seconds after the LOCA. At 300 seconds, the analysis shows that the temperature of the vessel wall at a depth of 38.1 mm (1.5 inches) is approximately 204 °C (400 °F). The applicant expects that the vessel beltline material ART, even after 60 years of irradiation, will be low enough to ensure that the material is in the Charpy upper shelf region at 204 °C. Therefore, the analysis will be bounding and valid for the license renewal term.

Charpy Upper Shelf Energy (USE)

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to demonstrate that BWR RPVs could meet margins of safety against fracture equivalent to those required by Appendix G of the ASME

Code Section XI for Charpy USE values less than 68 J (50 ft-lb). General Electric (GE) performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, “BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines,” BWRVIP-74, September 1999. This updated analysis incorporates the effects of irradiation for 54 effective full-power years (EFPYs), which corresponds to 60 years of operation at 90% power. The updated analysis determined that the generic materials considered would maintain the margins for USE required by 10 CFR Part 50 Appendix G. The application indicates that the applicant plans to review the generic analyses with respect to their applicability for the Peach Bottom license renewal term. This review will determine whether the generic analyses are applicable and whether the critical materials would retain sufficient USE to satisfy 10 CFR Part 50 Appendix G requirements for 54 EFPYs. The applicant plans to complete this review and confirm the acceptable value for USE before the end of the initial operating license term for Peach Bottom.

4.2.1.2 Staff Evaluation

<u>Neutron Irradiation Embrittlement</u>

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to ensure adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. For the RPV, this appendix requires an evaluation of the Charpy USE and an evaluation of the ART to determine pressure-temperature limits for the RPV. Neutron irradiation causes a decrease in the Charpy USE and an increase in the ART of the RPV beltline materials. The staff’s evaluation of the impact of irradiation on the reflood thermal shock analysis and Charpy USE is discussed in this section. The staff's evaluation of the impact of irradiation on pressure-temperature limit, RPV circumferential weld, and RPV axial weld integrity analyses is discussed in SER Sections 4.2.2.2, 4.2.3.2, and 4.2.4.2, respectively. Since each of these evaluations depends on the neutron fluence received by the RPV, neutron fluence is also discussed in these sections.

<u>Neutron Fluence and RT_{NDT}</u>

The RT_{NDT}, reflood thermal shock analysis, Charpy USE, P-T limit, circumferential weld, and axial weld integrity evaluations are all dependent upon the neutron fluence. The applicant states that it will initiate the calculations for end-of-life fluence for a 60-year licensed operating period (54 EFPYs) using the GE fluence calculation methodology (NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation") after the NRC approves it.

In order to determine whether neutron irradiation embrittlement will satisfy the time-limited aging analysis criterion in 10 CFR Part 54.21(c)(1), the staff issued RAI 4.2-1 requesting the applicant to determine the adjusted reference temperature (ART) and the Charpy upper shelf energy (USE) at the end of the license renewal period (60 years of operation). These analyses require that the applicant determine the peak neutron fluence at the end of the license renewal period. Therefore, in RAI 4.2-1, the staff also requested the applicant to calculate the peak neutron fluence at the clad-steel interface and the 1/4 thickness (1/4T) location in the reactor vessels at the end of the license renewal period using a methodology approved by the staff and adhering to the guidance in Regulatory Guide (RG) 1.190,"Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

In response to RAI 4.2-1, the applicant submitted the following estimates of neutron fluence and adjusted reference temperature for Peach Bottom Units 2 and 3. The applicant response for estimates of upper shelf energy is presented later in this section under the heading Charpy upper shelf energy (USE).

Neutron fluence: For Units 2 and 3, the 54 EFPYs RPV peak fluence predictions are 2.2 x 10^{18} n/cm^2 at the inner vessel wall and 1.6 x 10^{18} n/cm^2 at 1/4T location. The neutron fluence calculation was performed using the methodology of NEDC-32983P, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,” which was approved by the NRC in a letter dated September 14, 2001, from S.A. Richards (NRC) to J.F. Klapproth (GE). Since the neutron fluence evaluation was performed in accordance with a methodology that was approved by the staff, the results are acceptable and may be utilized for the evaluations discussed in SER Sections 4.2.2.2, 4.2.3.2, and 4.2.4.2.

The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (delta RT_{NDT}), and a margin (M) term. The delta RT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation methods. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

The 54 EFPYs ART for the limiting beltline material for Unit 2 (Shell # 2 Heat C2873-1) at 1/4T is 70 °F. The 54 EFPYs ART for the limiting material for Unit 3 (Shell # 2, Heat C2773-2) at 1/4T is 97 °F. These values for ARTs were confirmed by the staff using the neutron fluence value of 1.6E18 n/cm^2, the initial RT_{NDT} values, and the Cu and Ni contents for the limiting beltline materials from the Peach Bottom Updated Final Safety Analysis Report, Volume 1. The Cu and Ni contents for the limiting beltline material are 0.12 and 0.57 wt%, respectively, for Unit 2, and 0.15 and 0.49 wt%, respectively, for Unit 3. The initial RT_{NDT} for the limiting beltline material is -6 °F for Unit 2 and 10 °F for Unit 3. A margin value of 34 °F was used for confirming the ARTs. The staff finds the ART consistent with RG 1.99, Revision 2, and acceptable.

Reflood Thermal Shock Analysis

The applicant has reviewed the reflood thermal shock analysis for Peach Bottom. For the reflood thermal shock event, the peak stress intensity at 1/4 of vessel thickness from inside occurs about 300 seconds after the LOCA. At 300 seconds, the analysis shows that the temperature of the vessel wall at a depth of 38.1mm (1.5 inches) is approximately 204 °C (400 °F). The applicant states that the reflood thermal shock analysis for 40-years of operation (32 EFPYs) will be bounding and valid for the license renewal term because the vessel beltline material ART, even after 60 years of irradiation, is expected to be low enough to ensure that the material is in the Charpy upper shelf region at 204 °C. In RAI 4.2-2, the staff requested the applicant to present the technical basis for expecting the vessel beltline material ART after 60 years of irradiation to be low enough so that the material is in the Charpy upper shelf region at 204 °C. In response, the applicant referred to its response to RAI 4.2-1, which indicated that the ART for the limiting plate material for Peach Bottom Unit 2 is 70 °F and for Unit 3 is 97 °F, which is well below the 204 °C (400 °F) 1/4T temperature predicted for the thermal shock event at the time of peak stress intensity. The reflood thermal shock analysis is, therefore, bounding and valid for the license renewal term.

Charpy Upper Shelf Energy (USE)

Section IV.A.1a of Appendix G to 10 CFR Part 50 requires, in part, that the RPV beltline materials have Charpy USE in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted a topical report entitled “10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels,” to demonstrate that BWR RPVs could meet margins of safety against fracture equivalent to those required by Appendix G of the ASME Code Section XI for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of ASME Code Section XI, in accordance with Appendix G of 10 CFR Part 50. In this report, the BWROG derived through statistical analysis the unirradiated USE values for materials that originally did not have documented unirradiated Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) USE values in accordance with RG 1.99, Rev. 2. According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

General Electric (GE) performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, “BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines,” BWRVIP-74, September 1999. The staff review and approval of EPRI TR-113596 is documented in a letter from C. I. Grimes to C. Terry dated October 18, 2001. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Revision 2. Using this methodology and a correction factor of 65% for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPYs for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical position MTEB 5-2. Using the RG methodology, the lowest irradiated Charpy USE at 54 EFPY for BWR non-Linde 80 submerged arc welds is projected to be 43 ft-lb. EPRI TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 beltline plates and BWR non-Linde 80 submerged arc welds are 23.5% and 39%, respectively. Since this is a generic analysis, the staff issued RAI 4.2-3 requesting the applicant to submit plant-specific information to demonstrate that the beltline materials of the Peach Bottom Units 2 and 3 RPVs meet the criteria in the report at the end of the license renewal period. The applicant was specifically requested to submit the information specified in Tables B-4 and B-5 of EPRI TR-113596. In response to RAI 4.2-3, the applicant stated that the predicted percent decrease of the beltline material USE values at 1/4T and 54 EFPYs was estimated using BWRVIP-74 and RG 1.99, Revision 2. The equivalent margin analysis was performed using information presented in Tables B-4 and B-5 of EPRI TR-113596. RG 1.99, Revision 2, predicted percent decrease in USE for the limiting beltline plate material at the end of the license renewal period is 14% for Unit 2 and 16% for Unit 3; both predicted values of USE are less than the generic value of 23.5% reported in EPRI TR-113596. Similarly, the RG 1.99, Revision 2, predicted percent decrease in USE for limiting weld material (non-Linde 80 weld material at both units) at the end of license renewal period is 21% for both Unit 2 and Unit 3, which is less than the generic value of 39% reported in EPRI TR-113596. The predicted values for the decrease in USE for limiting beltline weld and plate materials for Units 2 and 3 were confirmed by the staff using the 54 EFPYs neutron fluence values at 1/4T provided by the applicant and the values of the Cu contents for the limiting materials from the Peach Bottom Updated Final Safety Analysis Report, Volume 1. The 54 EFPYs neutron fluence at 1/4T for the limiting beltline plate and weld materials of both units is 1.6E18 n/cm^2. The Cu contents for the limiting beltline materials are 0.182 wt% for weld and 0.13 wt% for plate for Unit 2, and 0.182 wt% for weld and 0.15 wt% for plate for Unit 3. The staff finds the applicant response acceptable because the percent decrease in USE for plant-specific limiting plate and weld materials at Units 2 and 3 is bounded by the corresponding generic results obtained by the equivalent margin analysis presented in EPRI TR-113596 as mentioned above. Therefore, the Charpy USE values at 54 EFPYs for the limiting plate and weld materials at Units 2 and 3 are greater than the minimum allowable value of 35 ft-lb, which demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of Section XI of the ASME Code, in accordance with Appendix G of 10 CFR Part 50, throughout the license renewal period. The UFSAR Supplement needs to include the additional information contained in the applicant’s response to RAI 4.2-3 regarding the evaluation of this TLAA. In a letter dated November 26, 2002, responding to this Confirmatory Item, the applicant provided a revision to Section A.5.1.1 of the UFSAR Supplement, which describes the USE analyses performed by the applicant, and adequately addresses the issue.

4.2.1.3 Conclusions

The staff has reviewed the information in LRA Section 4.2.1, “10 CFR 50 Appendix G Reactor Vessel Rapid Failure Propagation and Brittle Fracture Considerations: Charpy Upper Shelf Energy (USE) Reduction and RT_{NDT} Increase, Reflood Thermal Shock Analysis.” On the basis of this review, the staff concludes that the applicant has adequately evaluated the TLAA related to 10 CFR Part 50 Appendix G reactor vessel rapid failure propagation and brittle fracture considerations (Charpy upper shelf energy (USE) reduction, RT_{NDT} increase, and reflood thermal shock analysis), as required by 10 CFR 54.21(c)(1)(i). The staff has also reviewed the UFSAR Supplement and the staff concludes that, the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

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

4.2.2.1 Summary of Technical Information in the Application

Peach Bottom Technical Specification 3.4.9 presents P-T limit curves for heatup and cooldown, and also limit the maximum rate of change of reactor coolant temperature. At Peach Bottom, the criticality curve presents limits for both heatup and criticality are calculated for a 40-year design (32 EFPY). The application indicates that the applicant will determine the P-T limits for 60 years (54 EFPY), in accordance with 10 CFR 54.21(c)(1)(ii), after the GE fluence methodology has been approved by the NRC.

4.2.2.2 Staff Evaluation

The P-T limit curves are based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, "Reactor Vessel Structural Integrity, " Revision 1; GL 92-01, Revision 1, Supplement 1; RG 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock." GL 88-11 advised applicants that the staff would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper shelf energy resulting from neutron radiation. GL 92-01, Revision 1, requested that applicants submit their RPV data for their plants to the staff for review. GL 92-01, Revision 1, Supplement 1, requested that applicants submit and assess data from other applicants that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by the methodology of Appendix G Section XI of the ASME Code.

SRP Section 5.3.2 presents an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 the thickness (1/4T) of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating cooldown and heatup P-T limit curves are the 1/4T and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The ASME Code Appendix G methodology requires that applicants determine the ART at the end of the operating period.

The applicant plans to calculate vessel P-T limit curves for 60 years (54 EFPYs) after the NRC has approved GE fluence calculation methodology. As discussed in Section 4.2.1.2 of the SE, the staff has approved the GE fluence calculation methodology that is documented in topical report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation." This topical report was approved by the NRC in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GE). In RAI 4.2-5, the staff requested the applicant to submit P-T limit curves for a 60-year (54 EFPYs) design for Peach Bottom using the GE methodology. In response, the applicant stated that the vessel P-T limit curves for 54 EFPYs have been completed. The plant technical specifications will be modified to incorporate these P-T limit curves when the current curves reach their operational limits. The curves will be submitted to the NRC as a license amendment prior to the end of the initial operating license term for Peach Bottom. The staff finds the applicant's response acceptable because the change in P-T curves will be implemented by the license amendment process.

4.2.2.3 Conclusions

The staff has reviewed the information in LRA Section 4.2.2, "Reactor Vessel Thermal Limit Analyses: Operating Pressure-Temperature Limit (P-T Limit) Curves." On the basis of this review, the staff concludes that the applicant has adequately evaluated the reactor vessel operating pressure-temperature limit curves TLAA, as required by 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.2.3 Reactor Vessel Circumferential Weld Examination Relief

4.2.3.1 Summary of Technical Information in the Application

Sections 4.2.3 and A.5.1.2 of the LRA discuss inspection of the Peach Bottom RPV circumferential welds. These sections of the LRA indicate that Peach Bottom will use an approved technical alternative in lieu of ultrasonic testing of RPV circumferential shell welds. The BWRVIP presented the technical bases in EPRI TR-113596 for supporting the elimination of RPV circumferential welds from the inservice inspection programs for BWRs. These technical bases are approved for the current license term and are applicable to Peach Bottom.

Appendix E of the NRC's safety evaluation report (SER), "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report " USNRC, July 28, 1998, documents an evaluation of the impact of license renewal from 32 to 64 EFPYs on the conditional probability of vessel failure. The SER reports that the frequency of cold overpressurization events results in a total vessel failure probability of approximately 5×10^{-7}. The SER conservatively evaluates an operating period of 10 EFPYs greater than what is realistically expected for a 20-year license renewal term, i.e., 48 to 54 EFPYs. Therefore, this analysis supplies a basis for BWRVIP-05 to be approved as a technical alternative from the current inservice inspection requirements of ASME Section XI for volumetric examination of the circumferential welds as they may apply in the license renewal period.

In LRA Section 4.2.3, "Reactor Vessel Circumferential Weld Examination Relief," the applicant states that the procedures and training used to limit the frequency of cold overpressure events to the specified number in the current licensed operating period will also be used during the license renewal term. The applicant will apply for an extension of the subject relief for the 60-year extended licensed operating period prior to the end of the initial operating license term for Peach Bottom.

4.2.3.2 Staff Evaluation

Sections 4.2.3 and A.5.1.2 of the LRA discuss inspection of the Peach Bottom RPV circumferential welds. These sections of the LRA indicate that Peach Bottom will use an approved technical alternative in lieu of ultrasonic testing of RPV circumferential shell welds.

The technical alternative is discussed in the staff's final SER of the BWRVIP-05 report, which is enclosed in a July 28, 1998 letter to Carl Terry, BWRVIP Chairman. In this letter, the staff concludes that since the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and the core damage frequency (CDF) of any BWR plant, since that continued inspection would result in a negligible decrease in an already acceptably low value, elimination of the ISI for RPV circumferential welds is justified. The staff's letter indicated that BWR applicants may request relief from inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) the applicants have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the frequency specified in the report. The letter indicated that the requirements for inspection of circumferential RPV welds during an additional 20-year license renewal period would be reassessed, on a plant specific basis, as part of any BWR LRA.

Section A.4.5 of report BWRVIP 74 indicates that the staff's SER conservatively evaluated the BWR RPVs to 64 effective full power years (EFPYs), which is 10 EFPYs greater than what is realistically expected for the end of the license renewal period. Since this was a generic analysis, the staff issued RAI 4.2-6 requesting the applicant to submit plant-specific information to demonstrate that the Peach Bottom beltline materials meet the criteria specified in the report. To demonstrate that the vessel has not become embrittled beyond the basis for the technical alternative, the applicant must supply (1) a comparison of the neutron fluence, initial RT_{NDT}, chemistry factor, amounts of copper and nickel, delta RT_{NDT} and mean RT_{NDT} of the limiting circumferential weld at the end of the renewal period to the 64 EFPYs reference case in Appendix E of the staff's SER, and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting circumferential weld and the reference case. Should the applicant request relief from augmented ISI requirements for volumetric examination of circumferential RPV welds during the period of extended operation, the applicant is requested to demonstrate that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) the applicant has implemented operator training and established procedures that limit the frequency of cold overpressure events to the frequency specified in the report. In response to the RAI, the applicant compared the limiting circumferential weld properties for Peach Bottom Units 2 and 3 to the information in Table 2.6-4 and Table 2.6-5 of the staff SER on BWRVIP-05 dated July 28, 1998.

The NRC staff used the mean RT_{NDT} value for materials to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPYs in the staff SER dated July 28, 1998. The mean RT_{NDT} value is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}) and the mean value of the adjustment in reference temperature caused by irradiation (delta RT_{NDT}); it does not include a margin (M). The neutron fluence used in this evaluation was the neutron fluence clad-weld (inner) interface. The mean RT_{NDT} for Peach Bottom Units 2 and 3 is determined to provide a comparison with the values documented in the staff SER. The 54 EFPYs mean RT_{NDT} values thus determined are12 °F and 17 °F for Units 2 and 3, respectively. The staff confirmed these values of mean RT_{NDT} using the data for 54 EFPYs neutron fluence at the clad-weld interface provided by the applicant and the data for Ni and Cu contents in the girth welds from the Peach Bottom Updated Final Safety Analysis Report, Volume 1. For Unit 2, the 54 EFPYs fluence is 1.8E18 n/cm^2, and Cu and Ni contents are 0.056 and 0.96 wt%, respectively. For Unit 3, the 54 EFPYs fluence is 1.4E18 n/cm^2, and Cu and Ni contents are 0.102 and 0.942 wt%. These 54 EFPYs values mean that RT_{NDT} values for Units 2 and 3 are bounded by the 64 EFPYs mean RT_{NDT} value of 70.6 °F used by NRC for determining the conditional failure probability of a circumferential girth weld. The 64 EFPYs mean RT_{NDT} value from the staff SER dated July 28, 1998, is for a Chicago Bridge and Iron (CB&I) weld because CB&I welded the girth welds in the Peach Bottom vessels. Since the Peach Bottom 54 EFPYs value is less than the 64 EFPYs value from the staff SER dated July 28, 1998, the staff concludes that the Peach Bottom RPV conditional failure probability is bounded by the NRC analysis.

The procedures and training used to limit cold overpressure events will be the same those approved by the NRC when Peach Bottom requested to use the BWRVIP-05 technical alternative for the current term (letter from James Hutton of PECO Nuclear to NRC dated February 7, 2000). The staff find the applicant's response to RAI 4.2-6 acceptable because the 54 EFPYs mean RT_{NDT} value for the circumferential weld is bounded by the NRC analysis in the staff SER dated July 28, 1998, and Peach Bottom will be using procedures and training to limit cold overpressure events during the period of extended operation. The UFSAR Supplement needs to include the additional information contained in the applicant’s response to RAI 4.2-6 regarding the evaluation of this TLAA. In a letter dated November 26, 2002, responding to this Confirmatory Item, the applicant provided a revision to Section A.5.1.1.3 of the UFSAR Supplement, which describes the analysis of the circumferential welds and adequately addresses this issue.

4.2.3.3 Conclusions

The staff has reviewed the information in LRA Section 4.2.3, “Reactor Vessel Circumferential Weld Examination Relief.” On the basis of this review, the staff concludes that the applicant has adequately evaluated the reactor vessel circumferential weld examination relief TLAA, as required by 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes that, the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.2.4 Reactor Vessel Axial Weld Failure Probability

4.2.4.1 Summary of Technical Information in the Application

The staff’s SER, enclosed in a letter dated March 7, 2000, to Carl Terry, BWRVIP Chairman, discusses the failure frequency for RPV axial welds and the BWRVIP analysis of the RPV failure frequency for axial welds. The SER indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5 x 10^{-6} per reactor year, given the assumptions on flaw density, distribution, and location described in this SER. Since the BWRVIP analysis was generic, the applicant plans to perform plant-specific analyses to confirm that the axial weld failure probability for the Peach Bottom RPVs remains below 5 x 10^{-6} per reactor year during the period of extended operation, in accordance with 10 CFR Part 54.21(c)(1)(i). The application indicates that the applicant plans to complete these analyses prior to the end of the initial operating license term for Peach Bottom.

4.2.4.2 Staff Evaluation

In its July 28, 1998, letter to Carl Terry, BWRVIP Chairman, the staff identified a concern about the failure frequency of axially oriented welds in BWR RPVs. In response to this concern, the BWRVIP supplied evaluations of axial weld failure frequency in letters dated December 15, 1998, and November 12, 1999. The staff's SER on these analyses is enclosed in a March 7, 2000 letter to Carl Terry. The SER indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5×10^{-6} per reactor year, given the assumptions on flaw density, distribution, and location described in this SER. Since the results apply only for the initial 40-year license period of BWR plants, applicants for license renewal must submit plant-specific information applicable to 60 years of operation.

The BWRVIP identified the Clinton and Pilgrim reactor vessels as the reactor vessels with the highest mean RT_{NDT} in the BWR fleet. The staff confirmed this conclusion in the SER enclosed in the March 7, 2000, letter by comparing the information in the BWRVIP analysis and the information in the Reactor Vessel Integrity Database (RVID) for all BWR RPV axial welds. The results of the staff calculations are presented in Table 1. The staff calculations used the basic input information for Pilgrim, with three different assumptions for the initial RT_{NDT}. The calculations of the actual Pilgrim condition used the docketed initial RT_{NDT} of -44 °C (-48 °F) and a mean RT_{NDT} of 20 °C (68 °F). A second calculation, listed as "Mod 1" in Table 1, uses an initial RT_{NDT} of -18 °C (0 °F) and a mean RT_{NDT} of 47 °C (116 °F) consistent with the BWRVIP calculations. A third calculation, with an initial RT_{NDT} of -19 °C (-2 °F) and a mean RT_{NDT} of 46 °C (114 °F), was chosen to identify the mean value of RT_{NDT} required to provide a result which closely matches the RPV failure frequency of 5×10^{-6} per reactor-year.

Table 1: Comparison of Results from Staff and BWRVIP

Plant	Initial RT_{NDT} (°F)*	Mean RT_{NDT} (°F)	Vessel Failure Freq.	
			Staff	BWRVIP
Clinton	-30	91	2.73E-6	1.52E-6
Pilgrim	-48	68	2.24E-7	-------
Mod 1 **	0	116	5.51E-6	1.55E-6
Mod 2 ***	-2	114	5.02E-6	-------

* °C = 0.56 x (°F – 32)
** A variant of Pilgrim input data, with initial RT_{NDT} = 0 °F.
*** A variant of Pilgrim input data, with initial RT_{NDT} = -2 °F.

Since the BWRVIP analysis was generic, the staff issued RAI 4.2-7 requesting the applicant to submit plant-specific information to demonstrate that the Peach Bottom beltline materials meet the criteria specified in the report. To demonstrate that the vessel has not become embrittled beyond the basis for the staff and BWRVIP analyses, the applicant was requested to submit (1) a comparison of the neutron fluence, initial RT_{NDT}, chemistry factor, amounts of copper and nickel, delta RT_{NDT}, and mean RT_{NDT} of the limiting axial weld at the end of the renewal period to the reference cases in the BWRVIP and staff analyses; and (2) an estimate of the conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting axial welds and the reference case. If this comparison does not indicate that the RPV failure frequency for axial welds is less than 5×10^{-6} per reactor year, the applicant must submit a probabilistic analysis to determine the RPV failure frequency for axial welds.

The applicant presented plant-specific information in response to RAI 4.2-7 to demonstrate that Peach Bottom beltline materials meet the criteria specified in this SER. The SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet, and vessel failure probability calculations determined for Clinton should bound those for the BWR fleet. The NRC used mean RT_{NDT} for the comparison. The mean RT_{NDT} values in the staff's SER were determined using the neutron fluence at the clad/weld (inner) interface, and did not include a margin term. The 54 EFPYs mean RT_{NDT} values for axial welds at clad-weld interface in both Peach Bottom Units 2 and 3 are the same and equal to 11 °F. The staff confirmed this value by using the 54 EFPYs neutron fluence data (2.2E18 n/cm^2) provided by the applicant and the data for Cu and Ni contents (0.182 and 0.181 wt%, respectively) in the axial welds from the Peach Bottom Updated Final Safety Analysis Report, Volume 1; these data are the same for the limiting beltline region axial welds for Units 2 and 3. A comparison of the mean RT_{NDT} value (91 °F) for the Clinton axial weld given in Table 1 with the Peach Bottom value (11 °F) shows that the NRC analysis of Clinton axial welds bounds the Peach Bottom axial welds. Since the Peach Bottom 54 EFPYs value is less than the Clinton value, the staff concludes that Peach Bottom is bounded by the NRC analysis that is enclosed in the March 7, 2000, letter to Carl Terry, and the staff finds the applicant's response acceptable. The UFSAR Supplement needs to include the additional information contained in the applicant's response to RAI 4.2-7 regarding the evaluation of this TLAA. In a letter dated November 26, 2002, responding to this Confirmatory Item, the applicant provided a revision to Section A.5.1.1.4 of the UFSAR Supplement, which describes the analysis of the axial welds and adequately addresses this issue.

4.2.4.3 Conclusions

The staff has reviewed the information in LRA Section 4.2, "Reactor Vessel Neutron Embrittlement." On the basis of this review, the staff concludes that the applicant has adequately evaluated the reactor vessel neutron embrittlement TLAA, as required by 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes that, the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity as a result of metal fatigue, which initiates and propagates cracks in the material.

The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for piping and components and, consequently, fatigue is part of the current licensing basis (CLB) for Peach Bottom. The applicant identified fatigue analyses as TLAAs for piping and components. The staff reviewed Section 4.3 of the LRA, which discusses fatigue of piping and components, to determine whether the applicant has adequately evaluated the TLAAs as required by 10 CFR 54.21(c).

4.3.1 Summary of Technical Information in the Application

The applicant discussed the fatigue analyses of the Peach Bottom Unit 2 and 3 reactor pressure vessel (RPV) components in Section 4.3.1 of the LRA. The applicant indicated that the analyses have been revised to incorporate changes for power uprate and other operational changes. The applicant's revised analyses indicated that the vessel closure studs may exceed the ASME Code fatigue cumulative usage factor (CUF) limit during the current term of operation and, therefore, included the closure studs in its fatigue management program (FMP). The applicant further indicated that all RPV locations with calculated CUFs that exceed 0.4 are included in the FMP. The FMP monitors plant transients that contribute to the fatigue usage for the following components:

- RPV feedwater nozzles (Loops A and B)
- RPV support skirt
- RPV closure studs
- RPV shroud support
- RPV core spray nozzle safe end
- RPV recirculation inlet nozzle
- RPV recirculation outlet nozzle
- RPV refueling containment skirt
- RPV jet pump shroud support
- residual heat removal (RHR) return line (Loop A)
- RHR supply line (Loops A and B)
- RHR tee (Loops A and B)
- feedwater piping
- main steam piping
- torus penetrations
- torus shell

The applicant discussed the fatigue analyses of the reactor vessel internals (RVI) in Section 4.3.2.1 of the LRA. The applicant indicates that the core shroud, shroud support, and jet pump assembly evaluation were based on a standard plant design and that the core shroud supports were reevaluated to account for the effects of increased recirculation pump starts with the loop outside the thermal limits.

The applicant discussed the RVI embrittlement analysis in Section 4.3.2.2 of the LRA. The applicant's evaluation indicated that the effect of fatigue and embrittlement on end-of-life reflood thermal shock remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

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.

The applicant's May, 1, 2002, response indicated that the FatiguePro monitoring system will be implemented to monitor selected component locations. FatiguePro uses measured temperature, pressure, and flow data to either monitor the number of cycles of design basis transients or to directly compute the stress history to determine the actual fatigue usage for each transient. The applicant indicated that most component locations will be monitored by an automated cycle counting module that will count each licensing basis transient experienced by the plant based on input from monitored plant instruments. The applicant will incorporate the cycle counts obtained since initial plant startup for these component locations. Monitoring of the RPV feedwater nozzles and the RPV support skirt will include a fatigue usage computation based on temperature, pressure, and flow data obtained from monitored plant instruments. The applicant will estimate that the prior fatigue usage for the feedwater nozzles and the RPV support skirt assuming a linear accumulation of fatigue based on the design fatigue values. The applicant indicates that the future monitoring will be used to demonstrate the conservatism of the assumption of a linear accumulation of fatigue based on the design values. The staff considers the applicant's improved program an acceptable method to monitor fatigue of the critical components.

The applicant indicated that the closure studs are projected to have a CUF > 1.0 during the current period of operation and that the studs are included in the FMP. In RAI 4.3-1, the staff requested the applicant to provide additional discussion regarding the projected CUF for the closure studs.

The applicant's May 1, 2002, response indicated the fatigue evaluation of the reactor vessel closure studs is based on very conservative analysis techniques. The fatigue usage of the closure studs is being monitored by the FMP. The applicant indicated that corrective action will be initiated prior to reaching a CUF of 1.0 and that corrective actions would include one or more of the following options:

- refinement of the fatigue analysis to lower the CUF to below 1.0
- Repair/replacement of the studs
- manage the effects of fatigue by an inspection program

The applicant committed to provide the NRC with the inspection details of the aging management program for staff review and approval prior to implementation if the last option is selected. An aging management program under this option would be a departure from the design basis CUF evaluation described in the UFSAR Supplement, and therefore, would require a license amendment pursuant to 10 CFR 50.59. In view of the above, the staff finds the applicant's proposed corrective actions an acceptable approach to manage fatigue of the closure studs. However, in accordance with 10 CFR 54.21(d), this information needs to be added to the UFSAR Supplement, and was the subject of Confirmatory Item 4.3.2-1 discussed below.

The applicant indicated that a fatigue evaluation of the core shroud and jet pump assembly was performed for a plant where the configuration applies to Peach Bottom. The applicant further indicated that the fatigue analyses were reevaluated for the effects of increased pump starts with the loop outside thermal limits. The applicant indicated that fatigue of the critical locations of the jet pump shroud support and RPV shroud support would be managed by the FMP. In RAIs 4.3-3 and 4.3-4, the staff requested that the applicant provide further clarification regarding the revised analysis considering an increase in recirculation pump starts and its impact on the fatigue usage of the core shroud and jet pump assembly.

The applicant's May 1, 2002, response indicated that although the shroud support is not an ASME component, it was included in the original ASME Code Section III design basis evaluation for the reactor pressure vessel. The applicant further indicated that the core shroud and jet pumps are not ASME components and do not have design basis fatigue evaluations. The applicant indicated the discussion in the LRA regarding the core shroud and jet pump assembly refers to a location on the core shroud support structure where the jet pump adapter is attached.

The applicant's May 1, 2002, response also described the reevaluation of the core shroud support structure. The Peach Bottom technical specifications require that the temperature difference between an idle recirculation loop and the vessel coolant be 50 °F or less prior to pump restart. Since Peach Bottom experienced recirculation pump starts outside the technical specification limit, a reevaluation was triggered. The applicant accounted for the fatigue associated with these events by using the results from the design basis sudden pump start event. The design basis sudden pump start is a more severe thermal transient than the events that have occurred at Peach Bottom. The calculated fatigue usage from the design basis event is multiplied by the ratio of the temperature difference from the actual pump start to the temperature from the design basis event to obtain the fatigue usage for each pump start event at Peach Bottom. The applicant provided the results from a sample calculation to demonstrate the conservatism of the procedure. On the basis of the results of the applicant's sample calculation, the staff finds the applicant's evaluation provides an acceptable method to estimate the fatigue usage resulting from the recirculation pump start events experienced at Peach Bottom.

The applicant's FMP tracks transients and cycles of RCS components that have explicit design basis transient cycles to ensure that these components stay within their design basis. Generic Safety Issue (GSI) 166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of these components. Although GSI-166 was resolved for the current 40-year design life of operating plants, the staff initiated GSI-190 to address license renewal. The resolution of GSI-166 for the 40-year design life relied, in part, on conservatism in the existing CLB analyses. This conservatism included the number and magnitude of the cyclic loads postulated in the initial component design. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address license renewal. The NRC closed GSI-190 in December, 1999, concluding:

> The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to mange the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe breaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in

10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant indicated that there is sufficient conservatism in the fatigue analyses of components at Peach Bottom to account for the effects of the environment on the design fatigue curves. The applicant relied on the results of generic industry studies to support this argument. The staff has previously commented on these generic industry studies.

By letter dated February 9, 1998, the Electric Power Research Institute (EPRI) submitted two technical reports dealing with the fatigue issue. EPRI topical reports TR-107515, “Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant,” and TR-105759, “An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Evaluations” were part of an industry attempt to resolve GSI-190. As recommended in SECY 95-245, the EPRI analyzed components with high usage factors, using environmental fatigue data. The staff has open technical concerns regarding the EPRI reports. The staff’s technical concerns were transmitted to the Nuclear Energy Institute (NEI) by letter dated November 2, 1998, and NEI responded to the staff’s concerns in a letter dated April 8, 1999. The staff submitted its assessment of the response in a letter to NEI, dated August 6, 1999. As indicated in the staff’s letter, the NEI response did not resolve all of the staff’s technical concerns regarding the EPRI reports.

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 2 and 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 the period of extended operation acceptable. However, in accordance with 10 CFR 54.21(d), this information needs to be added to the UFSAR Supplement.

The applicant indicated that Group II and III piping systems were designed to the requirements of USAS B31.1. The applicant performed an evaluation of the number of cycles expected for the period of extended operation. The applicant’s evaluation indicated that the number of cycles is expected to be substantially less than the 7,000 cycle limit during the period of extended operation. Therefore, the existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant indicated that the NSSS vendor specified a finite number of cycles for each of the elevated-temperature operating modes of the RHR system. The applicant also indicated that it found no description of these design operating cycles in the Peach Bottom licensing basis documents. The applicant indicated that the Group 1 RHR piping inside the drywell was analyzed to the ASME Section III Class 1 requirements. The applicant further indicated that an evaluation of the remaining Group I and Group II piping indicated that the number of thermal cycles would be substantially less the 7,000 cycle limit applicable to piping designed to USAS B31.1. In RAI 4.3-5, the staff requested the applicant to provide further clarification regarding the NSSS vendor specification.

In its May 1, 2002, response, the applicant indicated that the vendor specification contained a description of certain thermal cycles for the original system design. The applicant found no licensing basis requirements (other than design code cycle limits) like those contained in the USAS B31.1 piping design code. The applicant also stated that design to the vendor-specified cycles is not a TLAA, except as it may be included within the design code requirements. The applicant reviewed the design specifications and design codes for components such as pumps and heat exchangers to determine whether they incorporated thermal cycle design considerations. The applicant indicated that no such requirements were identified. As a consequence, the applicant concluded that the only consideration for thermal cyclic loading that needed to be considered was the USAS B31.1 cycle limit. The staff considers the applicant's clarification of this issue satisfactory.

The applicant's UFSAR Supplement for metal fatigue is provided in Section A.4 of the LRA. The applicant describes the FMP in Section A.4.2 and its assessment of metal fatigue for the reactor vessel, reactor vessel internals and piping and components in Section A.5.2. As discussed previously, the applicant indicated that corrective actions to address the fatigue of the reactor vessel closure studs would be initiated prior to the period of extended operation. With the applicant's commitment to include in the UFSAR Supplement a description of the corrective actions to address closure studs as provided above in the response to RAI 4.3-1; and perform plant specific calculations for the locations identified in NUREG/CR-6260 for an older vintage BWR plant considering applicable environmental factors provided in NUREG/CR-6583 and NUREG/CR-5704 as provided above in response to RAI RAI 4.3-6; the staff concludes that the UFSAR Supplement will include an appropriate summary description of the programs and activities to manage aging as required by 10 CFR 54.21(d). This was identified as Confirmatory Item 4.3.2-1 in the draft safety evaluation.

By letter dated November 26, 2202, responding to this Confirmatory Item, the applicant provided a revision to the UFSAR Supplement. The revised UFSAR supplement contains a description of the applicant's proposed corrective actions to address fatigue of the reactor vessel closure studs and the applicant's commitment to evaluate the impact of the reactor water environment on the fatigue life of the components identified in NUREG/CR-6260 for an older vintage BWR. On the basis of the applicant's revised UFSAR supplement, Confirmatory Item 4.3.2-1 is closed.

Fatigue Monitoring Program

Summary of Technical Information in the Application

In Appendix B.4.2 of the LRA, the applicant describes an existing aging management program, the FMP, that is designed to track cyclic and transient occurrences to ensure that reactor coolant pressure boundary components remain within ASME Code Section III fatigue limits. The applicant indicates the FMP will be enhanced to broaden its scope and update its implementation methods. The applicant further indicates that the program will use a computerized data acquisition, recording and tracking system.

Staff Evaluation

The staff's evaluation of the FMP focused on how the program manages fatigue through effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls and operating experience.

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site controlled corrective actions program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to aging management review. The staff evaluation of the applicant's corrective actions program is provided separately in Section 3.0.4 of this SER. The corrective actions program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining 7 elements are discussed below.

Program Scope: The scope of the program includes the reactor pressure vessel (RPV), reactor vessel internals (RVI), Group I piping reactor coolant pressure boundary and the torus structure. The staff considers the scope of the FMP, which includes components, including components of the reactor coolant pressure boundary, with fatigue analyses, to be acceptable.

Preventive and Mitigative Actions: The applicant referred to the cycle counting procedure as the preventative action for this program. The staff did not identify a need for any additional preventive or mitigative actions.

Parameters Inspected or Monitored: The applicant monitors the transients that contribute to the fatigue usage of the components discussed in Section 4.3 of the SE. The staff finds that monitoring these selected high fatigue usage locations provides an acceptable method to monitor the fatigue usage due to design transients for the RPV, RVI, Group 1 reactor coolant pressure boundary piping, and torus structure.

Detection of Aging Effects: The program continuously monitors operational transients and updates the fatigue analyses of the monitored components . This provides assurance that the fatigue analyses of record remain valid during the period of extended operation. The staff finds this monitoring acceptable.

Monitoring and Trending: As stated previously, the program continuously monitors the operational transients that contribute to the fatigue usage of the monitored components to assure that the fatigue analyses of record remain valid during the period of extended operation. The staff finds that the applicant’s continuous monitoring is sufficient to allow for timely corrective actions and is, therefore, acceptable.

Acceptance Criteria: The acceptance criteria consists of maintaining the fatigue usage below the code limit. By meeting these limits, the applicant provides assurance that the monitored components remain within their design limits. Therefore, the staff considers this criteria acceptable.

Operating Experience: The applicant’s program was developed in response to concerns that early-life operating cycles at some units caused fatigue usage to accumulate faster than anticipated in the design analysis. The applicant has selected a sample of critical locations to monitor the fatigue usage accumulation. The staff finds that the applicant has adequately considered operating experience in selecting the locations to be monitored.

The staff reviewed Section A.4.2 of the UFSAR Supplement (Appendix A of the LRA) to verify that the information provided in the UFSAR Supplement for the aging management associated with the FMP is equivalent to the information in NUREG-1800. The staff concludes that the UFSAR Supplement provides an adequate summary of program activities as required by 10 CFR 54.21(d).

Conclusions

The applicant references the FMP in its discussion of the fatigue TLAAs as a program to assure that design fatigue limits are not exceeded during the period of extended operation. The staff considers the applicant's program, which monitors the number of plant transients that were assumed in the fatigue design, an acceptable method to manage the fatigue usage of the RCS components within the scope of the program. Therefore, the staff concludes that the FMP will adequately manage thermal fatigue of RCS components for the period of extended operation as required by 10 CFR 54.21(a)(3). The staff also concludes that the UFSAR Supplement contains an adequate summary description of the program activities associated with the FMP for managing the effects of aging as required by 10 CFR 54.21(d).

4.3.3 Conclusions

The staff has reviewed the information in Section 4.3 of the LRA regarding the fatigue analysis of the reactor vessel, reactor vessel internals and piping at Peach Bottom. The applicant's evaluation of Group II and III piping indicates that the analyses will remain valid for the period of extended operation. The applicant monitors the fatigue usage of critical reactor vessel, reactor vessels internals and Group I piping components using its FMP. The staff concludes that the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.4 Environmental Qualification

The 10 CFR 50.49 environmental qualification (EQ) program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ components includes all long-lived passive and active electrical and instrumentation and control (I&C) components and commodities that are located in a harsh environment and are important to safety, including safety-related and Q list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The staff has reviewed LRA Section 4.4, "Environmental Qualification of Electrical Equipmne," LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1) for evaluating the EQ TLAA. Paragraph (1) of 10 CFR 54.21(c) requires that a list of EQ TLAA must be provided. The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) analyses have been projected to the end of the period of extended operation, or (iii) the effect of aging on the intended functions will be adequately managed for the period of extended operation. The staff also reviewed LRA Section 4.4.2, "GSI-168, 'Environmental Qualification of Low Voltage Instrumentation and Controls (I&C) Cables."

On the basis of this review, the staff requested additional information in a letter to the applicant dated October 26, 2001. The applicant responded to this request for additional information (RAI) in a letter to the staff dated January 2, 2002.

4.4.1 Electrical Equipment Environmental Qualification Analyses

4.4.1.1 Summary of Technical Information in the Application

The Peach Bottom EQ program complies with all applicable regulations and manages equipment thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Environmetally qualified equipment must be refurbished, replaced, or have its qualification extended prior to reaching the aging limits established in the aging evaluation. Aging evaluations for environmental qualified equipment that specify a qualified life of at least 40 years are considered TLLAs for license renewal. The following is a list of TLAAs for EQ of electrical equipment.

- GE Co. 4kV pump motors and associated cable
- EGS Grayboot connectors
- Raychem insulated splices for class 1E systems
- Bussman Co. and Gould Shawmut fuses and fuse holders
- EGS quick disconnect connectors
- Limitorque motor-operated valve actuators
- Namco position switches
- ASCO solenoid valves, trip coils, and pressure switches
- UCI splice tape
- Rosemount 1153 Series B transmitters
- GE Co. control station
- Agastat relays
- static O-ring pressure switches
- Cutler Hammer motor control centers
- NDT International accoustical monitors
- Target Rock solenoid valves
- PYCO Resistance Temperature Detectors (RTDs) and thermocouples
- ITT Barton differential pressure switches
- Atkomatic solenoid valves
- Reliance fan motors and SGTS auxiliaries
- Brown Boveri load centers
- Valcor solenoid valves
- GE Co. radiation elements
- Pyle National plug connectors
- General Atomic radiation monitors
- GE electrical penetrations
- Buchanan terminal blocks
- GE terminal blocks
- Marathon terminal blocks
- Weidmueller terminal blocks
- Amp Inc. terminal lugs
- Scotch insulating tape
- GE SIS cable
- Brand Rex cable
- ITT Suprenant 600V control cable
- Okonite 600V power and control cable
- Rockbestos cable

- Foxboro pressure transmitters
- Patel conduit seals
- Jefferson coaxial cable
- Anaconda cable
- HPCI system equipment
- Masoneilan electropneumatic transducer
- Westinghouse Y panels and associated transformers
- Barksdale pressure switches
- H_2 and O_2 analyzer
- Avco pilot solenoid valves
- Rosemout model no. 710-DU trip units
- Westinghouse manual transfer switch

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

4.4.1.2 Staff Evaluation

The staff reviewed Section 4.4.1 of the Peach Bottom LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). In addition, the staff met with the applicant to obtain clarifications and reviewed the applicant's response to the staff's request for additional information.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(iii)

For the list of electrical equipment identified in Section 4.4.1 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the aging effects of the EQ equipment identified in this TLAA will be managed during the extended period of operation by the EQ program activities described in Section B.4.1 of the LRA.

The staff reviewed the EQ program to determine whether it will assure that the electrical and I&C components covered under this program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation. The staff's evaluation of the component qualification focused on how the program manages the aging effect through effective incorporation of the following 10 elements: program scope, preventive action, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

Program Scope: The Peach Bottom EQ program includes certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49. The staff considers the scope of the program acceptable.

Preventive Actions: 10 CFR 50.49 does not require actions that prevent aging effects. The Peach Bottom EQ program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit), (b) refurbishment , replacement, or requalification of installed equipment prior to reaching these aging limits, and (c) where applicable, requiring specific installation, inspection, monitoring, or periodic maintenance actions to maintain equipment aging effects within the qualification. The staff considers these are acceptable because 10 CFR 50.49 does not require actions that prevent aging effects.

Parameter Monitored or Inspected: EQ component aging limits are not typically based on condition or performance monitoring. However, per RG 1.89 Rev. 1, such monitoring program are an acceptable basis to modify aging limits. Monitoring or inspection of certain environmental, condition or equipment monitoring may be used to ensure that the equipment is within its qualification or as a means to modify qualification. The staff considers this monitoring appropriate because the program objective is to ensure the qualified life of devices established is not exceeded.

Detection of Aging Effects: 10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring of aging effects may be used as a means to modify component aging limits. The staff considers the applicant's program to use monitor of aging effects as a means to modify component aging limits acceptable.

Monitoring and Trending: 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. EQ program actions that could be viewed as monitoring include monitoring how long qualified component have been installed. Monitoring or inspection of certain environmental, condition or component parameters may be used to ensure that a component is within its qualification or a means to modify the qualification. The staff considers this is acceptable since 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging.

Acceptance Criteria: 10 CFR 50.49 acceptance criteria is that an in-service EQ component is maintained within its qualification including (a) its established aging limits and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the aging limits of each installed device. When monitoring is used to modify a component aging limit, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods. The staff considers this is acceptable since it is consistent with 10 CFR 50.49 requirements of refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device.

Corrective Actions, Confirmation Process, and Administrative Controls: If an EQ component is found to be outside its qualification, corrective actions are implemented in accordance with the PBAPS corrective action process. When unexpected adverse conditions are identified during operational or maintenance activities that effect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When emerging industry aging issues are identified that affect the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Confirmatory actions, as needed, are implemented as part of the PBAPS corrective actions. The PBAPS EQ program is subject to administrative controls, which require formal reviews and approvals. The PBAPS EQ program will continue to comply with 10 CFR 50.49 throughout the renewal period including development and maintenance of qualification documentation demonstrating a component will perform required functions during harsh accident conditions. The PBAPS EQ program documents identify the applicable environmental conditions for the component locations. The PBAPS EQ program qualification files are maintained in an auditable form for the duration of the installed life of the component. The PBAPS EQ program documentation is controlled under the quality assurance program. The staff considers this acceptable because corrective actions, confirmation process, and administrative controls are implemented in accordance with the requirement of 10 CFR 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, that will insure adequacy of corrective actions, confirmation process, and administrative controls.

Operating Experience: The Peach Bottom EQ program includes consideration of operating experience to modify qualification bases and conclusions. Including aging limits. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during accident conditions after experiencing the detrimental effects of in-service aging. The staff finds that the applicant has adequately addressed operating experience.

The results of the environmental qualification of electrical equipment in Section 4.4. indicate that the aging effects of the EQ of electrical equipment identified in the TLAA will be managed during the extended period of operation under 10 CFR 54.21(c)(1)(iii). However, no information is provided in the submittal on the attribute of a reanalysis of an aging evaluation to extend the qualification life of electrical equipment identified in the TLAA. The important attributes of a reanalysis are the analytical methods, the data collection and reduction methods, the underlying assumptions, the acceptance criteria, and corrective actions. The staff requested the applicant to provide information on the important attributes of reanalysis of an aging evaluation of electrical equipment identified in the TLAA to extend the qualification under 10 CFR 50.49(e).

The applicant responded, in the letter dated January 2, 2002, that the reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the Peach Bottom EQ program. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to Peach Bottom quality assurance program requirements, which requires the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Methods

The Peach Bottom EQ program analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years).

The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods

Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis per the Peach Bottom EQ Program. Temperature data used in an aging evaluation is to be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperature used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperature for an evaluation. Any changes to material activation energy values as part of a reanalysis are to be justified on a plant-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cycling aging.

Underlying Assumptions

The Peach Bottom EQ Program EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modification and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions

Under Peach Bottom EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification can not be extended by reanalysis, the component is be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

The staff finds that the above response acceptable because it now addresses the reanalysis attribute.

4.4.1.3 Conclusions

The staff has reviewed the information in LRA Section 4.4.1 “Electrical Equipment Environmental Qualification Analyse” for the Peach Bottom Units 2 and 3 and concluded that the applicant has submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1) and that the applicant has adequately evaluated the time-limited aging analyses for EQ of electrical equipment consistent with 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes the applicant has provided an adequate description of its evaluation of this TLAA and the associated program for effectivley managing aging for the period of extended operation as required by 10 CFR 54.21(d).

4.4.2 GSI-168, Environmental Qualification of Low Voltage Instrumentation and Control (I&C) Cables

4.4.2.1 Summary of Technical Information in the Application

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

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

4.4.2.2 Evaluation

GSI-168, “Environmental Qualification of Low Voltage Instrumentation and Control (I&C) Cables,” was developed to address environmental qualification of electrical equipment. The staff guidance to the industry (letter dated June 2, 1998 from NRC (Grimes) to NEI (Walters) states:

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues, specifically at this time; and

- An acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

For the purpose of license renewal, as discussed in the statements of consideration (SOC) (60 FR22484, May 8, 1995), there are three options for addressing issues associated with a GSI:

- If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution in the LRA.

- An applicant can submit a technical rationale that demonstrate the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.

- An applicant can develop a plant-specific aging management program that incorporates the resolution of the aging issue.

For addressing issues associated with GSI-168, the applicant continues to manage the effects of aging in accordance with the CLB and considers the evaluation of the EQ TLAA to be technical rationale that demonstrate that the CLB will be maintained during the period of extended operation. The staff finds that the applicant has addressed the issues associated with GSI-168.

4.4.2.3 Conclusions

The staff concludes that the applicant has adequately addressed the issues associated with GSI-168. The applicant will continue to manage the effects of aging in accordance with the CLB and considers the evaluation of the EQ TLAA to be the technical rationale that demonstrates that the CLB will be maintained during the period of extended operation in accordance with 10 CFR 54.21(c)(1). The staff has also reviewed the UFSAR Supplement and the staff concludes the applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.5 Reactor Vessel Internals Fatigue and Embrittlement

4.5.1 Summary of Technical Information in the Application

Core Shroud and Top Guide

BWRVIP-26 [Ref.: EPRI topical report TR-107285, “BWR Vessel and Internals Project: BWR Top Guide Inspection and Flaw Evaluation Guidelines,” December 1996] lists 5×10^{20} n/cm^2 as the threshold fluence beyond which the components will be significantly affected. The expected 60-year fluence on the shroud, 2.7×10^{20} n/cm^2 x 60/40 = 4.5×10^{20} n/cm^2, is below the 5×10^{20} n/cm^2 damage threshold. License Renewal Appendix C to BWRVIP-26 states that the generic fluence for 60 years on the top guide is 6×10^{20} n/cm^2. The application indicates that although this 60-year fluence will be above the 5×10^{20} n/cm^2 damage threshold, the tensile stresses in this component are very low. At these low stresses fracture is not a concern, and embrittlement is, therefore, not a threat to the intended function. These critical locations in the top guide are exempt from inspection under the approved BWRVIP-26 and no aging management activity is required.

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

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

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

4.5.2 Staff Evaluation

Core Shroud and Top Guide

The BWRVIP inspection program for the core shroud and top guide is discussed in topical report EPRI TR-107285, "BWR Vessel and Internals Project, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)," December 1996. This report was approved by the staff in a letter from C.I. Grimes (NRC) to C. Terry (BWRVIP) dated December 7, 2000. In its safety evaluation of this report, the staff concluded that due to susceptibility to irradiation-assisted stress corrosion cracking (IASCC), applicants referencing the BWRVIP-26 report for license renewal should identify and evaluate the projected accumulated neutron fluence as a potential TLAA issue.

BWRVIP-26 lists 5×10^{20} n/cm^2 as the threshold fluence beyond which components will be susceptible to IASCC. Since the expected 60-year fluence on the shroud, is below the 5×10^{20} n/cm^2 damage threshold, the core shroud should not be susceptible to IASCC.

The staff in a telephone call on June 17, 2002, with the applicant discussed the impact of neutron radiation on the integrity of top guide components. BWRVIP-26 states that the generic fluence on the top guide for 60 years is 6×10^{20} n/cm^2, which exceeds the 5×10^{20} n/cm^2 damage threshold. The applicant stated that the location on the top guide that will see this high fluence is the grid beam. This is location 1, as identified in BWRVIP-26, Table 3-2, "Matrix of Inspection Options." In its evaluation of the top guide assembly, including the grid beam, General Electric (GE) assumed a lower allowable stress value, acknowledging the high fluence value at this location. The conclusion of this analysis, and the fact that a single failure at this location has no safety consequence, was that no inspection was considered necessary.

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

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 applicant has provided an adequate description of its evaluation of this TLAA for the period of extended operation as required by 10 CFR 54.21(d).

4.6 Containment Fatigue

The applicant stated that, subsequent to the original design, elements of Peach Bottom containments were reanalyzed for fatigue due to unevaluated pressure and temperature cycles discovered by GE and others, resulting from design basis events, including loss of coolant accidents, safety relief valve discharge, and combinations of loads resulting from these conditions. The re-evaluation consisted of (1) generic analyses applicable to each of several classes of BWR containments and (2) plant-unique analyses (PUA) from the Mark 1 Containment Program. The scope of these analyses included the tori, the drywell-to-torus vents, SRV discharge piping, other torus-attached piping and its penetrations, and the torus vent bellows.

Since there are no hydrodynamic loads acting on the containment, fatigue is not considered in containment design except at penetrations or other stress concentration areas. The drywell shell plate was not evaluated for fatigue in the original design; the PUA also did not reevaluate the drywell, the drywell penetrations, or the process piping penetration bellows which are attached to the piping. No fatigue analyses were identified in the licensing and design basis documents for Peach Bottom for these components. However, the drywell process bellows were originally specified for a finite number of operating cycles, and the design of these bellows is therefore identified as a TLAA.

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

4.6.1.1 Summary of Technical Information in the Application

The applicant stated that the tori were originally evaluated for a maximum of 800 SRV events. For the stress cycles associated with SRV and other dynamic events, the PUA calculated maximum design life CUFs in excess of 0.666 for locations on the torus and drywell-to-torus vents. The CUFs for these locations will therefore exceed the ASME Section III Code allowable of 1.0 for the period of extended operation. For most torus, vent, and torus penetration locations the predicted CUF is less than 0.666. However, this CUF value does not provide analytical or event margin. The applicant has therefore chosen a calculated CUF of 0.4 or less as the validation limit for 60 years of operation. Locations whose 40-year CUF exceeds 0.4 will be included in the Fatigue Management Program (FMP), described in Section B.4.2 of the Application.

The FMP counts fatigue stress cycles, tracks fatigue usage factors, and calculates CUFs from modeling equations. For the torus, vent, and torus penetration the CUF model is made up of contributions resulting from normal operation and design basis worst case LOCA cyclic transients. The applicant stated that during normal operation, only SRV load cases contribute to fatigue. As part of the FMP, the fatigue analyses will be revised to show that the SRV contribution will not exceed the Code CUF limit during the period of extended operation. This will be confirmed for the duration of the extended operating period by monitoring fatigue at the high-usage-factor locations in the tori, torus vents and penetrations with the FMP, and tracking the CUFs at these locations using the CUF modeling equations, based on the monitored plant transients. These equations will be updated as necessary, and transient events will be tracked to ensure that the CUF due to normal operating transients will remain less than 1.0. The FMP also permits fatigue reanalysis of the high-usage-factor locations. Conservatism in the original containment PUA may permit the reduction of the total calculated CUFs below the limiting value of 0.4, for which fatigue monitoring would be required. Most locations have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). Those that do not remain valid will require management of the aging effects, in accordance with 10 CFR 54.21(c)(1)(iii).

4.6.1.2 Staff Evaluation

The applicant has performed fatigue analyses of the tori, torus vents and torus penetrations that include new Peach Bottom loads. A limit of CUF =0.4 for 40 years as an acceptance criterion was selected to determine if the analyses will remain valid for the period of extended operation. Those locations with CUF<0.4 will remain valid, pursuant to 10 CFR 54.21(c)(1)(i). For those locations that exceed the threshold, the effects of fatigue will be managed during the period of extended operation by the FMP cycle counting and fatigue CUF tracking program, pursuant to 10 CFR 54.21(c)(1)(iii).

4.6.1.3 Conclusions

Pursuant to 10 CFR 54.21(c), the staff finds the proposed acceptance limit CUF of 0.4 acceptable. The staff also finds the use of the FMP, to ensure that fatigue effects will be adequately managed and will be maintained within Code design limits for the period of extended operation, reasonable and acceptable. The applicant has also provided an adequate summary of the information related to the fatigue analysis of the tori, torus vents and penetrations in Section A.5.4.1 of the UFSAR Supplement as required by 10 CFR 54.21(d).

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

4.6.2.1 Summary of Technical Information in the Application

The SRV discharge lines and external torus-attached piping were analyzed separately from the tori and the torus vents. The analysis included the SRV lines and all piping and branch lines, including small-bore piping attached to the tori, pipe supports, valves, flanges, equipment nozzles and equipment anchors. The applicant stated that the highest fatigue CUF, calculated in the PUA on the basis of 800 SRV actuations was 0.202. The applicant concludes that the fatigue analyses of this piping will remain valid for the period of extended operation.

4.6.2.2 Staff Evaluation

The applicant has described a conservative approach to determining the fatigue evaluation of the SRV discharge lines and external torus-attached piping. The staff finds this approach reasonable and acceptable.

4.6.2.3 Conclusions

Pursuant to 10 CFR 54.21(c)(1)(i), the staff finds that the applicant’s evaluation of the fatigue analyses of the SRV discharge lines and external torus-attached piping demonstrate that these TLAAs will remain valid for the period of extended operation. The applicant has also provided an adequate summary of the information related to the fatigue analysis of the SRV discharge lines and external torus-attached piping in Section A.5.4.2 of the UFSAR Supplement as required by 10 CFR 54.21(d).

4.6.3 Expansion Joints and Bellows Fatigue Analyses: Drywell-to-Torus Vent Bellows

4.6.3.1 Summary of Technical Information in the Application

The applicant has stated that the PUA-calculated fatigue usage factors for the drywell to torus vent bellows are negligible.

4.6.3.2 Staff Evaluation

The staff considers the results of the PUA for these components reasonable and acceptable.

4.6.3.3 Conclusions

Pursuant to 10 CFR 54.21(c)(1)(i), the staff finds that the applicant's evaluation of the fatigue analysis of the drywell-to-torus vent bellows demonstrates that these TLAAs will remain valid for the period of extended operation. The applicant has also provided an adequate summary of the information related to the fatigue analysis of the drywell-to-torus vent bellows in Section A.5.4.3 of the UFSAR Supplement as required by 10 CFR 54.21(d).

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

Expansion Joint and Bellows Fatigue Analyses: Containment Process Penetration Bellows has been identified as a TLAA for the purposes of license renewal. The staff reviewed LRA Section 4.6.4 to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c).

4.6.4.1 Summary of Technical Information in the Application

The applicant stated that at Peach Bottom, the only containment process piping expansion joints and bellows subjected to significant thermal expansion and contraction cycling are those between the drywell shell penetrations and process piping. The design of containment boundary components for a stated number of cycles over the design life constitutes a TLAA, in accordance with 10 CFR 54.3. Some process expansion joints have been replaced with components designed to later code and specification requirements. These bellows were designed to the requirements of ASME Code Section III and specified a minimum of 200 "startup-and-shutdown" cycles and a minimum of 1,500 "normal operating" cycles. Both the original and replaced components were designed for a number of equivalent full-temperature thermal cycles in excess of their specifications. The bellows were initially designed and supplied for operation in excess of 10,000 operating and thermal cycles. The replacement bellows were designed for operation in excess of 50,000 cycles. The PUA did not include any reanalysis of the expansion joints.

4.6.4.2 Staff Evaluation

Based on the applicant's description, the design cycles of the original and replacement bellows exceed the requirements of the original specifications and the estimate of the thermal cycles that might be expected to occur during the period of extended operation. The fatigue analyses of the penetrations therefore demonstrate ample margin for continuing operation during the period of extended operation.

4.6.4.3 Conclusions

Pursuant to 10 CFR 54.21(c)(1)(i), the staff finds that the applicant's evaluation of the fatigue analysis of the expansion joint and bellows demonstrates that these TLAAs will remain valid for the period of extended operation. The applicant has also provided an adequate summary of the information related to the fatigue analysis of the containment process penetration bellows in Section A.5.4.4 of the UFSAR Supplement as required by 10 CFR 54.21(d).

4.7 Other Plant-Specific TLAAs

4.7.1 Reactor Vessel Main Steam Nozzle Cladding Removal Corrosion Allowance

4.7.1.1 Summary of Technical Information in the Application

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

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

4.7.1.2 Staff Evaluation

In response to RAI 4.7-1, the applicant identified the basis for the corrosion rate and the sources for the data. Based on the average of the available data, corrosion rates were determined for high- and low-temperature operating conditions. Assuming 54 years at high temperature and 6 years at low temperature (90% availability for 60 years of operation), and doubling the average corrosion rate, the amount of corrosion for 60 years of operation was estimated to be 0.030 inch. The analysis is acceptable to the staff because the analysis used the average of all available data and conservatively doubled the average corrosion rate to estimate the amount of corrosion for 60 years of operation. Based on the applicant's conservative analysis of the predicted loss of material resulting from corrosion during 60 years of operation, the staff concludes that the corrosion allowance identified when the clad was removed from the main steam nozzles is valid for 60 years of operation.

4.7.1.3 Conclusions

The reactor vessel main steam nozzle clad removal corrosion allowances have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The applicant has also provided an adequate summary of the information related to the above analysis in Section A.5.5.1 of the UFSAR Supplement as required by 10 CFR 54.21(d).

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

4.7.2.1 Summary of Technical Information in the Application

The applicant describes its evaluation of the feedwater nozzle and control rod drive return line nozzle cracking TLAA in LRA Section 4.7.2, "Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*," and in Section A.5.6, "Inservice Flaw Growth Analyses that Demonstrate Structural Integrity for 40 Years," of Appendix A, "Updated Final Safety Analysis Report (UFSAR) Supplement," of the LRA. The applicant proposes to manage crack growth associated with the TLAA by an NRC-approved BWR Owners Group (BWROG) inspection program.

By late 1970s, inservice inspections (ISIs) discovered cracking on the inside surface of feedwater and control rod drive return line (CRDRL) nozzles at several BWR plants in the United States. The cracking was attributed to thermal cycling due to turbulent mixing of relatively cooler CRDRL water and leaking feedwater with hot downcomer flow. The CRDRL nozzles have been capped at Peach Bottom Units 2 and 3 to eliminate cracking due to thermal cycling.

The applicant has taken the following three actions as recommended by NUREG-0619 and Generic Letter 81-11 to reduce or eliminate the causes of cracking of feedwater nozzles: (a) installation of improved triple thermal sleeves with dual piston ring seals, (b) removal of cladding from the nozzle bore and blend radii, and (c) improvement of the low-flow controller. The applicant now uses the NRC-approved improved BWROG inspection and management methods in lieu of NUREG-0619 methods. The BWROG methods depend on a fracture mechanics analysis and ultrasonic inspection from the vessel and nozzle exterior. The fracture mechanics analysis is used to determine the inspection interval. This analysis is not a TLAA because it does not involve time-limited assumptions defined by the current operating term.

The nozzle crack growth, however, must be acceptable for the period of extended operation to ensure the continued validity of the assumptions of fatigue analyses for the reactor pressure vessel, which are TLAAs.

The feedwater nozzle is subject to the combined effect of long-term, low-cycle thermal fatigue due to heatup, cooldown, and other operational transients (which affects the entire vessel, including the nozzle wall) and high-cycle thermal fatigue due to leaking feedwater (which only affects inner surface of the feedwater nozzle). The UFSAR description of this issue includes an evaluation of this combined effect, which is a TLAA. However, these two fatigue effects are separable. Table 3.1-1 of the LRA includes both cumulative fatigue damage and cracking as aging effects due to fatigue for BWR feedwater nozzle. The applicant proposes the use of NRC-approved BWROG inspection methods, which no longer depend on this combined fatigue evaluation, to manage cracking due to rapid thermal cycling, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

4.7.2.2 Staff Evaluation

The relatively cooler water leaking past the loosely fitted thermal sleeves installed inside the feedwater nozzles has caused cracking of these nozzles in a large number of BWR plants in the United States during 1970s. The cracks were discovered on the inside surface of the nozzles at the blend radius and bore. The leaking water (also called bypass leakage) turbulently mixed with hot downcomer flow in the annulus between the nozzle and thermal sleeve and put high-cycle fatigue loads on the nozzle inside wall. The cracks initiated by the high-cycle fatigue are arrested at a shallow depth (~6 mm) because the thermal stresses induced by the high-cycle fatigue have steep gradients and shallow depth. These cracks are further propagated by low-cycle fatigue due to plant heatup, cooldown, and feedwater on-off transients. These transients produce large, throughwall, stress cycles on the nozzle wall and in time could drive the cracks to significant depth. Such cracking has been discovered in the feedwater nozzles at Peach Bottom Units 2 and 3.

Similarly, the relatively cooler water passing through the CRDRL nozzle turbulently mixes with hot downcomer flow and causes cracking on the inside surface of the nozzle and also on the wall of the reactor pressure vessel beneath the nozzle. Such cracking has been discovered at the CRDRL nozzles at Peach Bottom Units 2 and 3. The applicant reports that these nozzles were capped after the cracks were repaired and are no longer susceptible to damage due to rapid thermal cycles. Therefore, the staff concludes that cracking of the CRDRL nozzles no longer requires aging management for license renewal at Peach Bottom Units 2 and 3.

NUREG-0619 recommended that the licensees take the following six actions to reduce the potential for initiation and growth of cracks in the inner nozzle areas: (1) remove the cladding from the inner radii; (2) replace loose-fitting or interference-fitting sparger thermal sleeves; (3) evaluate the acceptability of the flow controller; (4) modify operating procedures to reduce thermal fluctuations; (5) reroute reactor water cleanup system (RWCU) discharge to both feedwater loops; and (6) conform to the inspection interval specified in Table 2 of NUREG-0619. In 1981, the NRC staff issued Generic Letter 81-11 to amend the recommendations in NUREG-0619, thereby allowing plant-specific fracture mechanics analysis in lieu of hardware modifications.

The first three of the NUREG-0619 recommendations have been implemented at Peach Bottom Units 2 and 3: cladding has been removed from the nozzle bores and blend radii, improved triple thermal sleeves with dual piston ring seals have been installed, and the low-flow controllers have been improved. The implementation of these recommendations has been effective in preventing cracking of the feedwater nozzle. An industry report, GE-NE-523-A71-0594-A, Revision 1, “Alternate BWR Feedwater Nozzle Inspection Requirements,” May 2000, states that no new cracking has been identified in the BWR feedwater nozzles since 1984.

The feedwater nozzle is susceptible to the combined effect of low-cycle thermal and mechanical fatigue due to heatup, cooldown, and feedwater on-off transients and high-cycle thermal fatigue due to bypass leakage. The evaluation of this combined effect is a TLAA. The applicant, however, states that these two fatigue effects are separable and proposes two different aging management programs to manage them. The aging effect of low-cycle fatigue is cumulative fatigue damage, whereas the aging effects of high-cycle thermal fatigue is cracking. Several of the NUREG-0619 recommendations implemented at Peach Bottom Units 2 and 3 have reduced the potential for racks due to rapid thermal cycling damage. Consequently, the susceptibility to crack initiation at the feedwater nozzle blend radius and bore has also been reduced. This reduced susceptibility to cracking is supported by the significant field experience with the successful prevention of cracks in feedwater nozzles since implementation of the NUREG-0619 recommendations, as mentioned earlier. So the remaining aging effect of high-cycle fatigue is the growth of an existing crack that was initiated earlier by rapid thermal cycling caused by bypass leakage. Therefore, the staff conclude that the separation of two fatigue effects, cumulative fatigue damage and crack growth, is justified.

NUREG-0619 identified the inservice inspection requirements based on the state-of-the-art in the late 1970s. The required inservice inspection included both ultrasonic testing (UT) of the entire nozzle and dye-penetrant testing (PT) of various portions of blend radius and bore. Since the issuance of NUREG-0619, significant advances have been made in UT inspection technology, and significant field experience has been gained on the successful prevention of cracks in feedwater nozzles. As a result of these improvements, BWROG proposed that UT inspections replace the PT inspections specified in NUREG-0619, and that UT inspection intervals be based on sparger-sleeve configurations and specific UT inspection methods as described in the report GE-NE-523-A71-0594-A, Revision 1. This report specifies UT of specific regions of the nozzle inner blend radius and bore. The nozzle inner blend radius region is more limiting from a fracture mechanics point of view than the bore region. The UT examination techniques and personnel qualifications are in accordance with the guidelines of GE-NE-523-A71-0594-A, Revision 1. The examination techniques include manual, automatic and phased-array UT methodologies. In a letter from SA. Richards to W. Glenn Warren, dated March 10, 2000, “Final Safety Evaluation of BWR Owners Group Alternative BWR Feedwater Nozzle Inspections,” the NRC staff accepted the proposed BWROG inspection methods and fracture mechanics analysis. These NRC-approved BWROG inspection methods and inspection intervals are currently being used at Peach Bottom. The applicant proposes to continue the use of these inspection methods during the extended period of operation.

The BWROG inspection methods require fracture mechanics analysis to estimate the time required for an assumed crack (an initial crack depth of ~6 mm [0.5 inch]) to reach the generic allowable value (1 inch) or to reach an allowable value based on plant-specific analysis. Plant-specific analysis must follow the recommendations of Section 5.6 of the report GE-NE-523-A71-0594-A, Revision 1. The BWROG method determines the inspection interval as a fraction of the time taken for this crack growth. The magnitude of the fraction and therefore the size of the inspection interval depend on the thermal sleeve-sparger design configuration, the UT inspection technique employed, and the specific region of the nozzle inspected. The maximum allowable inspection interval for the nozzle inner blend radius is 10 years. This fracture mechanics analysis is not a TLAA because it is used to determine the inspection interval and not to determine whether the crack growth at the end of the current 40-year licensed operating period is acceptable, and so does not involve time-limited assumptions for the current operating term. The GE generic fracture mechanics evaluation show that there is significant margin available to the allowable depth of 1 inch. The report recommends that the fatigue crack growth curves from Section XI of the ASME Code be utilized in the fracture mechanics analysis. To predict crack growth, Peach Bottom performed the fracture mechanics analysis of feedwater nozzle subjected to thermal cycles expected during the extended period of operation. Analysis at Peach Bottom predicts that growth from the assumed initial flaw size to the allowable value will take about 60 years.

The NRC-approved BWROG inspection methods, along with acceptance criteria and corrective actions are included in the aging management program presented in LRA Section B.2.7, “RPV and Internals ISI Program.” The evaluation of this program is presented in Section 3.0.3.9 of this

SER. In addition to these inspections, the applicant proposes to do a periodic review of the fracture mechanics analysis, in conjunction with the fatigue management program presented in Section B.4.2 of the LRA, to ensure that the fracture mechanics evaluation remains bounding and applicable for its intended purpose. The staff finds the applicant's commitments acceptable.

4.7.2.3 Conclusions

The staff has reviewed the information presented in LRA Section 4.7.2, "Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*." On the basis of this review, the staff concludes that the applicant has adequately evaluated this TLAA, as required by 10 CFR 54.21(c)(1). Specifically, the staff concludes that the RPV and Internals ISI program will ensure that any cracking in the feedwater nozzle will be adequately detected and managed, within the limits of the supporting fracture mechanics analyses, for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). The applicant has also provided an adequate summary of the information related to the above analysis in Section A.5.6.1 of the UFSAR Supplement as required by 10 CFR 54.21(d).

4.7.3 Fracture Mechanics of ISI-Reportable Indications for Group 1 Piping: As-forged Laminar Tear in a Unit 3 Main Steam Elbow Near Weld 1-B-3BC-LDO Discovered During Preservice UT

4.7.3.1 Summary of Technical Information in the Application

The applicant reported that a preservice UT volumetric examination discovered an imbedded as-forged laminar tear in the Unit 3 main steam elbow material. The UT indication did not extend to the weld.

To determine the effect of the flaw on the life of the steam line, the applicant performed an ASME Section III Class 1 fatigue analysis of the main steam elbow with the flaw, considering 40 years of operation. The analysis determined that the primary, secondary, and primary plus secondary stresses are within the Code allowable limits, and calculated a 40-year cumulative usage factor (CUF) of 0.012. The applicant stated that if the laminar tear extended to the weld joint, the CUF would rise to 0.036, and would not exceed to 0.054 for the period of extended operation. These values are below the Code design limit of 1.0.

4.7.3.2 Staff Evaluation

Ordinarily, fatigue analyses of steam lines in accordance with ASME Section III Class 1 are not required, since these are not Class 1 components. However, for the elbow with flaws, the applicant chose to perform an ASME Section III Class 1 fatigue analysis and demonstrate that the calculated CUF is below the Code design limit of 1.0 for 40-year operation and also for the period of extended operation. A CUF of 1.0 is considered the approximate threshold at which a fatigue crack may initiate and propagate. The staff's interpretation is that the applicant's intent was to consider the discovered flaw as a local discontinuity in the elbow geometry. The effect of the flaw is accounted for by the introduction of a fatigue strength reduction factor, or an equivalently stress concentration factor, as specified in the ASME Section III Subsection NB design rules. By reporting that the CUF is considerably below the design limit of 1.0, the staff concludes that the applicant has provided reasonable assurance that the flaw will not propagate during operation during the 40-year life of the plant and the period of extended operation.

4.7.3.3 Conclusions

Pursuant to 10 CFR 54.21(c)(1)(i), the staff finds that the applicant's evaluation of the effect of a laminar tear discovered during a preservice ultrasonic examination on the structural integrity of the steam line elbow by an ASME Section III Class 1 fatigue analyses is acceptable, and that the applicant has demonstrated that this TLAA will remain valid for the period of extended operation. The applicant has also provided an adequate summary of the information related to the fatigue evaluation of a laminar tear discovered during a preservice inspection in a steam line elbow in Section A.5.6.2 of the UFSAR Supplement as required by 10 CFR 54.21(d).