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10CFR54

May 1, 2002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Peach Bottom Atomic Power Station, Units 2 and 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Response to Request for Additional Information Related to Time-Limited Aging Analyses

Reference: Letter from R. K. Anand (USNRC) to M. P. Gallagher (Exelon), dated February 7, 2002

Dear Sir/Madam:

Exelon Generation Company, LLC (Exelon) hereby submits the enclosed responses to the request for additional information transmitted in the reference letter. For your convenience, attachment 1 restates the questions from the reference letter and provides our responses.

If you have any questions or require additional information, please do not hesitate to call.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on 5/2/02

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosures: Attachment 1

cc: H. J. Miller, Administrator, Region I, USNRC
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS

A001

ATTACHMENT 1

Exelon Generation Company, LLC (Exelon)
License Renewal Application (LRA)
Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3

Request for Additional Information

4.0 Time-Limited Aging Analyses

4.1 Identification of TLAAs

RAI 4.1-1

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

Response:

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

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

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

4.2 Reactor Vessel Neutron Embrittlement

RAI 4.2-1

The applicant describes its evaluation of reactor vessel neutron embrittlement time-limiting analyses in Section 4.2 of the LRA. The evaluation shows that 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 EFPY) using the GE fluence methodology after the NRC approves it.

In order to determine whether neutron irradiation embrittlement will satisfy the time-limited aging analyses criteria in 10 CFR Part 54.21(c)(1) the applicant must 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, the applicant is requested to calculate the peak neutron fluence at the clad-steel interface and the 1/4 thickness location in the reactor vessels at the end of the license renewal period using a methodology approved by the staff and adheres to the guidance in Regulatory Guide (RG) 1.190,"Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Note: The staff approved a neutron fluence calculation methodology submitted by GE Nuclear Energy (NEDC-32983P,"General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation") in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GENE).

Response:

Neutron fluence: For Unit 2, the 54 EFPY RPV peak fluence prediction is 2.2E18 n/cm^2 at the inner vessel wall. The 54 EFPY fluence prediction is 1.6E18 n/cm^2 at ¼ T. For Unit 3, the 54 EFPY RPV peak fluence prediction is 2.2 E18 n/cm^2 at the inner vessel wall. The 54 EFPY fluence prediction is 1.6 E18 n/cm^2 at ¼ T. 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).

ART: The adjusted reference temperature (ART = Initial RT_{NDT} + ΔRT_{NDT} + margin) was predicted for each beltline material, based on the methods of Regulatory Guide 1.99, Rev. 2. The ART for the limiting beltline material for Unit 2 (Shell # 2 Heat C2873-1) at 54 EFPY is 70°F. For Unit 3, the limiting ART at 54 EFPY is 97°F (Shell # 2, Heat C2773-2).

USE: See response to RAI 4.2-3.

RAI 4.2-2

The applicant has reviewed the reflood thermal shock analysis for Peach Bottom in Section 4.2.1 of the LRA. For the reflood thermal shock event, the peak stress intensity at 1/4 of vessel thickness from inside occurs about 300 seconds after LOCA. At 300 seconds, the analysis shows that the temperature of the vessel wall at 38.1-mm (1.5-inch) depth location is approximately 204° C (400° F). The applicant states that the reflood thermal shock analysis for 40-years of operation (32 EFPY) 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. The applicant is requested to present 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.

Response:

As indicated in our response to RAI 4.2-1, the ART for the limiting plate material for PBAPS Unit 2 is 70°F and for Unit 3 is 97°F, which is well below the 204°C (400°F) ¼ t 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. The Equivalent Margin Analysis requirements are also met. See response to RAI 4.2-3.

RAI 4.2-3

EPRI TR-113596, "BWRVIP BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999, performs a generic analysis and determines that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds are 23.5 percent and 39 percent, respectively. Since this is a generic analysis, the applicant is requested to submit plant-specific information to demonstrate that the beltline materials of Peach Bottom Units 2 and 3 RPVs meet the criteria specified in the BWRVIP-74 report at the end of the license renewal period. The applicant is requested to submit the information specified in Tables B-4 and B-5 of EPRI TR-113596.

Response:

Predicted % decrease of the beltline material USE values at 54 EFPY was performed using the BWRVIP-74 and Regulatory Guide 1.99, Rev. 2. The Equivalent Margin Analysis demonstrates that the BWRVIP-74 requirements are met. The Equivalent Margin Analysis was performed using information provided in Tables B-4 and B-5 of EPRI TR-113596 as shown below.

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 2

Beltline Limiting Plate Material (shell # 2 Heat C2873-1)

Surveillance Plate USE:

% Cu = 0.10 (Reference 1)

Capsule Fluence = 1.8E17 n/cm^2 (Reference 1)

Measured % Decrease = See note 1 below

R.G.1.99 Predicted % Decrease = 8% (R.G.1.99, Rev. 2)

Limiting Beltline Plate USE:

% Cu = 0.12

54 EFPY ¼ T Fluence = 1.6E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 14% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

14% < 23.5%, so vessel plates are bounded by equivalent margin analysis

Note:

1. Per Reference 1, Table 5-5, the Upper Shelf Energy (USE) values for unirradiated and irradiated plate materials are as follows:
Unirradiated Plate (ft-lbs, Longitudinal/Transverse): 126/82
Irradiated Plate (ft-lb, Longitudinal/Transverse): 135/88

The above data shows the irradiated USE is higher than the unirradiated USE.

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 2

Beltline Limiting Girth Weld Material
(Weld BC, Heat S-3986 Lot 3876, Linde Flux 124)

<u>Surveillance Weld USE:</u>

% Cu = N/A

Capsule Fluence = N/A

Measured % Decrease = N/A

R.G.1.99 Predicted % Decrease = N/A

<u>Limiting Beltline Girth Weld USE:</u>

% Cu = 0.06

54 EFPY ¼ T Fluence = 1.3E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 13% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

13% < 39%, so vessel girth welds are bounded by equivalent margin analysis

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 2

Beltline Limiting Electroslag Weld Material (welds C1, C2, and C3)

<u>Surveillance Weld USE:</u>

% Cu = 0.10 (Reference 1)

Capsule Fluence = 1.8E17 n/cm^2 (Reference 1)

Measured % Decrease = see note 1 below

R.G.1.99 Predicted % Decrease = 10% (R.G. 1.99, Rev. 2)

<u>Limiting Beltline Electroslag Weld USE:</u>

% Cu = 0.182

54 EFPY ¼ T Fluence = 1.6E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 21% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

21% < 39%, so vessel electroslag welds are bounded by equivalent margin analysis

Note:

1. Per Reference 1, Table 5-5, the Upper Shelf Energy (USE) values for unirradiated and irradiated plate materials are as follows:
   Unirradiated Plate (ft-lbs): 110
   Irradiated Plate (ft-lb): 113

   The above data shows the irradiated USE is higher than the unirradiated USE.

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 3

Beltline Limiting Plate Material (shell #2, Heat C2773-2)

Surveillance Plate USE:

% Cu = 0.13 (Reference 2)

Capsule Fluence = 1.6E17 n/cm^2 (Reference 2)

Measured % Decrease = 10% (Reference 2)

R.G.1.99 Predicted % Decrease = 10% (R.G. 1.99, Rev.2)

Limiting Beltline Plate USE:

% Cu = 0.15

54 EFPY ¼ T Fluence = 1.6E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 16% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

16% < 23.5%, so vessel plates are bounded by equivalent margin analysis

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 3

Beltline Limiting Girth Weld Material
(Weld EF, Heat IP4217 Lot 3929, Linde Flux 124)

Surveillance Weld USE:

% Cu = N/A

Capsule Fluence = N/A

Measured % Decrease = N/A

R.G.1.99 Predicted % Decrease = N/A

Limiting Beltline Girth Weld USE:

% Cu = 0.102

54 EFPY ¼ T Fluence = 1.0E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 14% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

14% < 39%, so vessel girth welds are bounded by equivalent margin analysis

Equivalent Margin Analysis Plant Applicability Verification Form
For Peach Bottom Atomic Power Station Unit 3

Beltline Limiting Electroslag Weld Material (welds E1, E2, and E3)

Surveillance Weld USE:

% Cu = 0.11 (Reference 2)

Capsule Fluence = 1.6E17 n/cm^2 (Reference 2)

Measured % Decrease = 9% (Reference 2)

R.G.1.99 Predicted % Decrease = 10%

Limiting Beltline Electroslag Weld USE:

% Cu = 0.182

54 EFPY ¼ T Fluence = 1.6E18 n/cm^2

R.G. 1.99 Predicted % Decrease = 21% (R.G. 1.99, Rev. 2, Fig.2)

Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)

21% < 39%, so vessel electroslag welds are bounded by equivalent margin analysis

Reference:

1. B.J. Branlund, "Peach Bottom Atomic Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," GE-NE, San Jose, CA, December 1991, (SASR 88-24, Revision 1)
2. T.A. Caine, "Peach Bottom Atomic Power Station, Unit 3 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," GE-NE, San Jose, CA, July 1995, (SASR 90-50, Revision 1)

RAI 4.2-4

In Section 4.2.1 of the LRA, the applicant states that it will recalculate the vessel end-of-life RT_{NDT} (54 EFPY operating period) according to Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves." But this code case is not about calculating the end-of-life RT_{NDT}. It is about the use of the reference fracture toughness curve K_{Ic}, as found in Appendix A of ASME Section XI, in lieu of K_{Ia}, as given by Fig. G-2210-1 in Appendix G for the development of P-T limit curves. The applicant is requested to resolve this inconsistency about the use of Code Case N-640.

Response:

The clarification requested by the reviewer is correct. The citation of Code Case N-640 properly belongs to Section 4.2.2. The second paragraph of Analysis sub-section in Section 4.2.1 should read

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

Accordingly, the Disposition sub-section of Section 4.2.2 should read,

> <u>Disposition: Revision, 10 CFR 54.21(c)(1)(ii)</u>

Revised vessel P-T limit curves will be calculated for the extended licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii). The calculations may use the alternative fracture toughness curve K_{Ic} of ASME Section XI Appendix A in lieu of K_{Ia} from Appendix G Figure G-2210-1, as permitted by Code Case N-640. These calculations will be completed and acceptable values will be confirmed prior to the end of the initial operating license term for PBAPS.

RAI 4.2-5

In Section 4.2.2 of the LRA, the applicant states that it will calculate vessel P-T limit curves for a 60 years, 54 EFPY operating period, after the NRC has approved GE fluence methodology. The applicant is requested to submit P-T limit curves for a 60-year design (54 EFPY operating period) for Peach Bottom using the neutron fluence calculation methodology discussed in RAI 4.2-1.

Response:

The vessel P-T limit curves for 54 EFPY have been completed and will be submitted to the NRC as a license amendment prior to the end of the initial operating license term for PBAPS.

RAI 4.2-6

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 BWR Vessel and Internals Project BWRVIP-05 Report, which is contained in a July 28, 1998 letter to Carl Terry, BWRVIP Chairman. This indicates BWR applicants may request relief from inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating: (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount 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 license renewal application.

Section A.4.5 of Report BWRVIP-74 indicates that the staff's SER conservatively evaluated BWR RPVs to 64 effective full power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. It also discusses the impact of radiation embrittlement on circumferential RPV welds. Since this was a generic analysis, the applicant must submit plant-specific information to demonstrate that the Peach Bottom beltline materials meet the criteria specified in the report. To demonstrate that each of the Peach Bottom Unit 2 and 3 vessels has not been embrittled beyond the basis for the technical alternative, the applicant is requested to 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 EFPY 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) they have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the report.

Response:

The following provides the limiting circumferential weld properties for PBAPS Units 2 and 3 as compared to the information in Table 2.6-4 and Table 2.6-5 from the staff SER dated July 28, 1998 on BWRVIP-05.

The NRC staff used materials and fluence data in Tables 2.6-4 and 2.6-5 to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY. The Mean RT_{NDT} used by the NRC has been compared to the PBAPS values for 54 EFPY.

A review of NRC NUREG 1803 - Safety Evaluation Report related to the License Renewal of Edwin I. Hatch Nuclear Plant Units 1 and 2, page 4-26, indicates that 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 PBAPS Units 2 and 3 values shown in the Table provide values to match the staff SER. The Units 2 and 3 values at 54 EFPY are bounded by the 64 EFPY Mean RT_{NDT}. Although a conditional failure probability has not been calculated, the fact that the PBAPS 54 EFPY value is less than the 64 EFPY value the staff used leads to the conclusion that the PBAPS RPV conditional failure probability is bounded by the NRC analysis.

The procedures and training used to limit cold over-pressure events will be the same as that approved by the NRC when PBAPS requested the BWRVIP-05 technical alternative be used for current term (Letter from James Hutton of PECO Nuclear to USNRC dated February 7, 2000.) There is nothing unique about the renewal term in this regard.

Circumferential Weld

Group	CB&I 32 EFPY (Note 1)	CB&I 64 EFPY (Note 1)	PB Unit 2 54 EFPY	PB Unit 3 54 EFPY
Cu%	0.10	0.10	0.06	0.102
Ni%	0.99	0.99	0.97	0.942
CF	134.9	134.9	82	137
Fluence at clad/weld interface (10^{19}n/cm^2)	0.51	1.02	0.18	0.14
ΔRT_{NDT} w/o margin (^{0}F)	109.5	135.6	44	67
$RT_{NDT(U)}$ (^{0}F)	-65	-65	-32	-50
Mean RT_{NDT} (^{0}F)	44.5	70.6	12	17
P(F/E) NRC	2E-7	1.76E-5	---	---
P(F/E) BWRVIP	1E-6	---	---	---

Note 1: CB&I welded the girth welds in the PBAPS RPVs.

RAI 4.2-7

Sections 4.2.4 and A.5.1.3 of the LRA discuss inspection of the Peach Bottom RPV axial welds. These sections of the LRA state that Peach Bottom will perform plant-specific analyses following the generic analyses presented in BWRVIP-05 report. These analyses support a conclusion of an NRC SER, enclosed in the March 7, 2000 letter to Carl Terry, BWRVIP Chairman, 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 the 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.

Since the BWRVIP analysis was generic, the applicant is requested 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 been embrittled beyond the basis for the staff and BWRVIP analyses, the applicant is 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 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.

Response:

In the following table, the limiting axial weld properties for PBAPS Units 2 and 3 are compared to the information in Table 2.6-4 and Table 2.6-5 from the staff SER on BWRVIP-05. The NRC issued a supplement to the final safety evaluation of BWRVIP-05 report on March 7, 2000. The supplement required the limiting axial weld to be compared with data in Table 3 of that document. For PBAPS, the comparison was made to the Clinton plant information. The supplemental 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. A review of NRC NUREG-1803, "Safety Evaluation Report Related to the License Renewal of Edwin I. Hatch Nuclear Plant Units 1 and 2," page 4-26, indicates that the mean RT_{NDT} values in the staff's SER was determined using the neutron fluence at the clad/weld (inner) interface, and did not include a margin term. The PBAPS Units 2 and 3 values shown in the Table provide values to match the staff SER. A comparison of the Mean RT_{NDT} values from the NRC report with the PBAPS data shows that the NRC analysis bounds the PBAPS welds. Although a conditional failure probability has not been calculated, the fact that the PBAPS 54 EFPY value is less than the Clinton value the staff used leads to the conclusion that PBAPS plant is bounded by the NRC analysis.

Axial Weld

Group	B&W 32 EFPY (Note 1)	B&W 64 EFPY (Note 1)	Clinton	PBAPS Unit 2 54 EFPY	PBAPS Unit 3 54 EFPY
Cu%	0.25	0.25		0.182	0.182
Ni%	0.35	0.35		0.181	0.181
CF	142.5	142.5		94.5	94.5
Fluence at clad/weld interface (10^{19}n/cm^2)	0.125	0.25		0.22	0.22

Group	B&W 32 EFPY (Note 1)	B&W 64 EFPY (Note 1)	Clinton	PBAPS Unit 2 54 EFPY	PBAPS Unit 3 54 EFPY
ΔRT_{NDT} w/o margin (^0F)	66	88.9		56	56
$RT_{NDT(U)}$ (^0F)	10	10	-30	-45	-45
Mean RT_{NDT} (^0F)	76	98.9	91	11	11
P(F/E) NRC	5.98E-2	1.87E-1	2.73E-6	---	---
P(F/E) BWRVIP	8.12E-3	---	1.52E-6	---	---

Note 1: B&W completed the axial welds on PBAPS RPVs.

4.3 Metal Fatigue

RAI 4.3-1

Section 4.3.1 of the LRA indicates that the reactor vessel closure studs are projected to have a CUF>1.0 during the current period of operation. The LRA further indicates that the studs are included in the fatigue management program (FMP). Provide the reason the projected CUF for the closure studs is expected to exceed 1.0 during the current operating period. Discuss the potential corrective actions that will be implemented prior to the period of extended operation.

Response:

The fatigue evaluation for the reactor vessel closure studs is based on very conservative analysis techniques that, in turn, lead to a somewhat artificially inflated CUF. In addition, various scaling factor approaches have been applied over time to conservatively incorporate effects of modified plant operations (i.e., power uprate), and increased cycle counts have been incorporated into the evaluation to account for actual event accumulation. Despite these conservatisms present in the CUF estimate for the studs, and in view of the fact that the studs are replaceable components, Exelon has chosen to continue using the existing evaluation for the immediate future, while commensurately considering corrective actions consistent with ASME Code, Section XI, Nonmandatory Appendix L. However, Exelon recognizes that to-date, the NRC has not endorsed the currently existing Appendix L approach. The primary NRC concerns with Appendix L include crack aspect ratio and acceptable fatigue crack growth rates (including environmental effects).

The approach to be used for the fatigue management program will include one or more of the following options:

1. Refinement of the fatigue analysis to lower the CUF to below 1.0, or
2. Repair/replacement of the studs, or

3. Manage the effects of fatigue by an inspection program (e.g., periodic non-destructive examination of the studs at certain inspection intervals).

The reactor vessel closure studs are monitored in the improved fatigue-monitoring program. As soon as the CUF value approaches 1.0, the above corrective action will be triggered.

Should Exelon select Option 3 (i.e., inspection) to manage fatigue, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to implementation.

RAI 4.3-2

Section 4.3.1 of the LRA indicates that an improved program is being developed which will use temperature, pressure, and flow data to calculate and record accumulated usage factors for critical RPV locations and subcomponents. Describe how the monitored data will be used to calculate the usage factors for the monitored components. Indicate how the fatigue usage of the monitored components is estimated for the time prior to implementation of the improved program.

Response:

As discussed in Section 4.3.1 of the PBAPS LRA, Exelon is implementing the FatiguePro fatigue monitoring system for tracking cycles and CUF in critical plant component locations. FatiguePro monitors CUF for the selected locations in one of two ways:

1. *Stress-Based Fatigue Monitoring*: Stress-based fatigue (SBF) monitoring consists of computing a "real time" stress history for a given component from actual temperature, pressure, and flow histories via a finite element based Green's Function approach. CUF is then computed from the computed stress history using appropriate cycle counting techniques, and appropriate ASME Code, Section III fatigue analysis methodology. SBF monitoring is intended to duplicate the methodology used in the governing ASME Code, Section III stress report for the component in question, but uses actual transient severity in place of design basis transient severity.

2. *Cycle-Based Fatigue Monitoring*: Cycle-based fatigue (CBF) monitoring consists of a two-step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles:

   (a) Automated Cycle Counting: Categorization and counting of plant transients is accomplished by the FatiguePro automated cycle counting (ACC) module. The ACC module counts each transient that is defined in the plant licensing basis based on the mechanistic process or sequence of events experienced by the plant (as determined from monitored plant instruments). This approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by FatiguePro is captured for each monitored component, for ready comparison to design basis transient severity. Transients defined in the PBAPS Updated Final Safety Analysis Report are identified and implemented in the ACC module. Any additional system-specific transients that are experienced by the Group I piping systems, which contribute significantly to the calculated CUF, are also monitored.

(b) CUF Computation: CUF computation calculates fatigue directly from counted transients and parameters, as determined by the ACC module, for the monitored components. CUF is computed via a design-basis fatigue calculation where the fatigue table from the governing stress report is used as a basis, but actual numbers of cycles are substituted for assumed design basis numbers of cycles. The CUF calculations are conservative in that design basis transient severity is assumed.

Limiting components throughout the Group I pressure boundary were selected for monitoring that bound or represent all other components. The components identified in NUREG/CR-6260 for the older vintage BWR plant are also encompassed by the locations selected for monitoring. Inclusion of Group I piping systems into the fatigue management program provides a complete structural assessment of the Group I pressure boundary. The monitored locations and the fatigue computation method employed are summarized in Table 1.

For the time period prior to FatiguePro implementation, fatigue usage was estimated in one of two ways. For the SBF components, the initial CUF was determined based on a linear projection of the design basis CUF, including the increased cycle counts resulting from tracking actual plant operation. For example, if the design 40-year CUF for an SBF component is 0.70, and the improved program was implemented after 20 years of plant operation, the initial CUF was estimated to be (20/40) * 0.70 = 0.35. Continued CUF monitoring into the future will be used to demonstrate the conservatism of this estimate (i.e., show that the rate of actual CUF accumulation is less than the rate of design basis fatigue accumulation). For the CBF components, the initial CUF estimate was determined based on the cycle counts to-date since initial plant startup, and the design basis fatigue calculation methodology described above. These initial CUF estimates therefore considered all cycles experienced by PBAPS to-date, and assumed design basis severity for each event.

Table 1
Monitored Components and Method of CUF Calculation

Location	Fatigue Estimation Basis
RPV feedwater nozzles (Loops A and B)	SBF
RPV support skirt	SBF
RPV closure studs	CBF
RPV shroud support	CBF
RPV core spray nozzle safe end	CBF
RPV recirculation inlet nozzle	CBF
RPV recirculation outlet nozzle	CBF
RPV refueling containment skirt	CBF
RPV jet pump shroud support	CBF
Residual heat removal (RHR) 24" return line (Loop A)	CBF
RHR 20" supply line (Loops A and B)	CBF
Recirc. whip restraints (Unit 2 Loop A)-not included in LRA Table	CBF

Core spray piping – not included in LRA Table *(Reviewed, bounded by RPV Core Spray nozzle, not monitored)*	CBF
Feedwater piping	CBF
Main steam piping	CBF
RHR Tee (Loop A)	CBF
RHR Tee (Loop B)	CBF
Feedwater piping (Node 754)	CBF
Main steam piping (Node 606)	CBF
Torus penetrations Unit 2	CBF
Torus penetrations Unit 3	CBF
Torus shell	CBF

RAI 4.3-3

Section 4.3.2.1 of the LRA indicates that fatigue analyses of the core shroud supports were reevaluated for effects of increased recirculation pump starts with the loop outside thermal limits. Describe the reevaluations that were performed considering an increase in recirculation pump starts. Indicate the reason that the reevaluations were necessary.

Response:

At issue are thermal events associated with the PBAPS Technical Specification requirement that limits the temperature difference (ΔT) between an idle recirculation loop and the vessel coolant to be within 50°F of each other prior to pump start. Specifically of concern is the ΔT following initiation of the first of two idle recirculation pumps. Since PBAPS has experienced events of this type in the past, the plant Technical Specifications requirements triggered the evaluations in question.

The design basis Sudden Start event provides conservative and convenient criteria for accounting for the actual events experienced at the PBAPS units. This design basis event is much more severe than the events actually experienced at PBAPS, as the actual ΔTs were significantly lower than the ΔT evaluated for the design event. Since the design basis event consists of a very conservative step change in temperature, the fatigue contribution for this event is driven almost exclusively by the ΔT during the event. Therefore, the ΔT of the actual events were compared to the ΔT of the design basis events to establish partial cycle counts for these events. This method leads to revised CUF estimates that are linear with respect to ΔT, which is conservative compared to more traditional CUF estimates where stress is linearly related to ΔT. The conservatism of this method is demonstrated in Figure 1. Since the Sudden Start event is a primary contributor to the shroud support CUF value, this component was selected for evaluation of these events. Other affected RPV and piping locations were also evaluated, but were less limiting than the shroud support from a CUF perspective. Note that although the shroud support is not an ASME Code pressure boundary component, it was considered in this evaluation since it was included as a part of the original ASME Code, Section III design basis evaluation for the reactor pressure vessel.

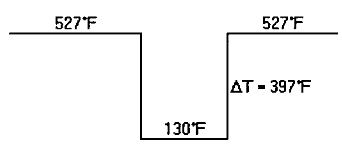
The fatigue analyses that were performed for the core shroud support essentially consisted of a cycle counting evaluation, using partial cycles, to ascertain the acceptability of the events experienced at PBAPS. The design basis Sudden Start event has an analyzed limit of 40 cycles, and the past accumulation of events was determined to be 7.8 events for the limiting PBAPS unit, which is acceptable.

The event in question, the shroud support location, and all past occurrences are all included as a part of Exelon's improved fatigue monitoring program implemented at PBAPS, so this issue will continue to be monitored throughout the period of extended operation.

Figure 1

Sample Calculation Demonstrating Conservative Nature of Partial Cycle Methodology

Design basis event definition:



Assume that 6 actual events are experienced, with ΔTs of 397°F, 350°F, 300°F, 200°F, 100°F, and 50°F. The following calculations are performed:

Method #1: Linear CUF Method (same as used by Exelon)

Low alloy steel material	S_m =	26.7	ksi	E_{curve} =	3.00E+07	psi
	m =	2.0		$E_{analysis}$ =	3.00E+07	psi
	n =	0.2				

S_{max} (ksi)	S_{min} (ksi)	S_{range} (ksi)	K_e	S_{alt} (ksi)	$N_{allowable}$	Actual ΔT (°F)	Ratio	n	ΔU
14.176	-14.176	28.352	1.000	14.176	400,000	397	1.000	1.000	2.500E-06
14.176	-14.176	28.352	1.000	14.176	400,000	350	0.882	0.882	2.204E-06
14.176	-14.176	28.352	1.000	14.176	400,000	300	0.756	0.756	1.889E-06
14.176	-14.176	28.352	1.000	14.176	400,000	200	0.504	0.504	1.259E-06
14.176	-14.176	28.352	1.000	14.176	400,000	100	0.252	0.252	6.297E-07
14.176	-14.176	28.352	1.000	14.176	400,000	50	0.126	0.126	3.149E-07
							Totals =	3.519	8.797E-06

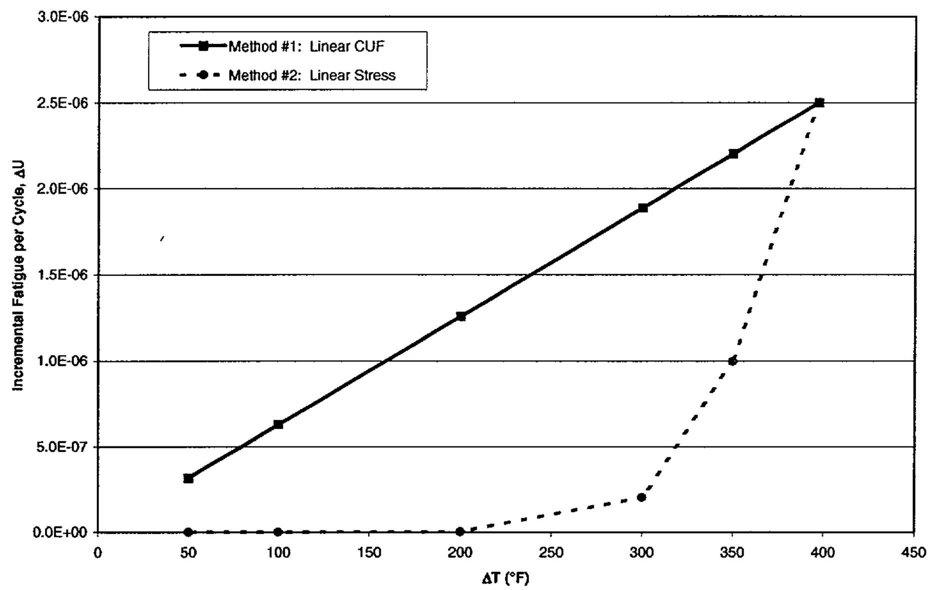
Therefore, 6 actual (partial) cycles were treated as 3.519 full (design basis) cycles for a CUF of 8.8×10^{-6}.

Method #2: Linear Stress Method (traditional approach)

Low alloy steel material	S_m =	26.7	ksi	E_{curve} =	3.00E+07	psi
	m =	2.0		$E_{analysis}$ =	3.00E+07	psi
	n =	0.2				

S_{max} (ksi)	S_{min} (ksi)	S_{range} (ksi)	Actual ΔT (°F)	Ratio	S_{range} (ksi)	K_e	S_{alt} (ksi)	$N_{allowable}$	n	ΔU
14.176	-14.176	28.352	397	1.000	28.352	1.000	14.176	400,000	1.000	2.500E-06
14.176	-14.176	28.352	350	0.882	24.995	1.000	12.498	1,001,900	1.000	9.981E-07
14.176	-14.176	28.352	300	0.756	21.425	1.000	10.712	4,947,800	1.000	2.021E-07
14.176	-14.176	28.352	200	0.504	14.283	1.000	7.142	634,878,300	1.000	1.575E-09
14.176	-14.176	28.352	100	0.252	7.142	1.000	3.571	>1e6	1.000	0.000E+00
14.176	-14.176	28.352	50	0.126	3.571	1.000	1.785	>1e6	1.000	0.000E+00
								Totals =	6.000	3.702E-06

Thus, the same 6 actual (partial) cycles caused a CUF of 3.70×10^{-6}.



RAI 4.3-4

Section 4.3.2.1 of the LRA indicates that the limiting fatigue usage for the core shroud and jet pump assembly is based on the evaluation of a plant with a configuration similar to PBAPS. As discussed in RAI 4.3-3, the PBAPS core shroud supports were reevaluated for the effects of increased recirculation pump starts with the loop outside thermal limits. Indicate whether the increase in recirculation pumps starts has any impact on the fatigue usage of the core shroud and jet pump assembly.

Response:

As indicated in the response to RAI 4.3-3 above, the shroud support is not an ASME Code pressure boundary component, but it was considered in the evaluation of recirculation pump start events because it was included as a part of the original ASME Code, Section III design basis evaluation for the reactor pressure vessel. The core shroud and jet pumps are not ASME Code pressure boundary components, and therefore, do not have design basis fatigue evaluations. Aging management of both of these components, which address both IGSCC and fatigue concerns, is addressed by the Reactor Pressure Vessel and Internals ISI Program as discussed in Appendix B.2.7 of the PBAPS LRA.

The information included in Section 4.3.2.1 of the PBAPS LRA associated with the core shroud and jet pump assembly refers to a location documented in the PBAPS UFSAR that is identified as the "Jet Pump Shroud Support." This identification refers to a location on the shroud support structure where the jet pump adapter is attached. The CUF value for this location is based on generic BWR evaluation performed by General Electric associated with jet pump design, and the associated impact on the shroud support structure. That analysis was included as a part of the evaluation described in the response to RAI 4.3-3 above, and the location is included in Exelon's improved fatigue monitoring program implemented at PBAPS. Therefore, this issue will continue to be monitored throughout the period of extended operation.

RAI 4.3-5

Section 4.3.3.3 of the LRA indicates that the NSSS vendor specified the RHR system for a finite number of cycles for each of its elevated-temperature operating modes. The LRA also indicates that no description of these design operating cycles was found in the PBAPS licensing basis documents. According to the LRA, Group 1 RHR piping inside the drywell was analyzed to the ASME Section III Class 1 rules. The LRA further indicates that an evaluation of the remaining Group I and Group II piping projected that the number of thermal cycles would be substantially less the 7,000 cycle limit contained in USAS B31.1. Provide further clarification regarding the details of the NSSS vendor specification. Describe the basis for assuming the 7,000 cycle limit contained in USAS B31.1 satisfies the vendor specification.

Response:

Clarification of the NSSS Vendor Specification

Piping of the entire RHR system (including some valves) was originally designed to USAS B31.1 rules. However, Group 1 RHR Shutdown Cooling portions of the system, inside containment, were replaced with the Recirculation piping to mitigate IGSCC concerns. This replacement piping was analyzed under ASME III Class 1 rules, and the original B31.1 design no longer applies. LRA Section 4.3.3.1 addresses the RHR piping with a class I analysis.

For piping designed to USAS B31.1, the Code assumes no more than 7,000 equivalent full-range thermal cycles as the limit beyond which a stress range reduction factor must be applied. The statement that "No description of these [original vendor] design operating cycles was found in the PBAPS licensing basis documents" means that although there is a vendor specification description of certain thermal cycles for the original system design, there is no licensing basis which requires any thermal cycle design analysis, other than (1) the B31.1 thermal cycle limit, or (2) those thermal cycle considerations which might be required by codes and standards for components, and invoked by reference to those codes and standards. Design to the vendor-specified cycles is therefore not a TLAA, except as it may be included within code design requirements.

The specifications and design codes for components (pumps, heat exchangers, any valve standards other than B31.1) were examined to determine if any code basis might have existed which would have incorporated a thermal cycle design analysis or assumption into the licensing basis. The result of that investigation was negative. All RHR components, other than B31.1 piping and valves, were specified either for temperature ranges which did not require thermal cycle analysis or assumptions, or to codes whose date or addendum did not provide for thermal cycle analysis or assumptions.

Exelon therefore concluded that no design analysis for thermal cycles had been applied to any of the non-Class 1-analysis portions of the RHR system, other than the stress range reduction factor required under USAS B31.1 rules for piping.

<u>Basis for assuming the 7,000 cycle limit contained in USAS B31.1 satisfies the vendor specification</u>:

The NSSS vendor's original specification included 30 cycles of normal operating suppression pool cooling, and one cycle of end-of-life, post-accident suppression pool cooling with containment spray operation. In addition, normal operating shutdown cooling (which, from a separate source, would be 120 cycles) was also considered. The severity of these specified RHR cycles is no worse than the maximum-range thermal cycle. This specification therefore amounted to no more than 151 equivalent full-range thermal cycles under USAS B31.1 rules. As stated above, (1) there is no licensing basis for analysis of the vendor-specified cycles beyond the code rules, and therefore any such analysis is not a TLAA, and (2) a review of specific equipment specifications and codes discovered no such design, nor any other cyclic design bases, other than USAS B31.1. The disposition of this TLAA therefore did not specifically address the cycles specified by the NSSS vendor.

However, the disposition did address the B31.1 system design, and even if design to the vendor-specified cycles were a TLAA, 151 cycles is a small fraction of the number of cycles for which the system was designed under B31.1 rules.

The evaluation of USAS B31.1 piping systems found that 700 equivalent full-range thermal cycles would be expected in a 40-year lifetime based on the expected Recirculation System cycles, certainly no more than 1000 (neglecting feedwater transients, which do not affect RHR - see LRA Section 4.3.3.2). This is the basis for the following statement in the LRA validation:

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

RAI 4.3-6

Section 4.3.4 of the LRA contains a discussion of Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components For 60-year Plant Life." GSI-190 addresses the effect of the reactor water environment on the fatigue life of metal components. The discussion in Section 4.3.4 indicates that EPRI license renewal fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff does not agree with the contention that the EPRI fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. The staff identified several technical concerns regarding the EPRI studies. The staff technical concerns are contained in an August 6, 1999, letter to NEI. Although these concerns involved 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 has additional concerns regarding the applicability of the EPRI BWR studies to PBAPS. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant and EPRI 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 applicability of the EPRI fatigue studies to PBAPS has not been demonstrated. Provide the following additional information regarding resolution of the environmental fatigue issue:

a. Indicate whether the staff comments provided in the staff 's August 6, 1999, letter to NEI, which are applicable to PBAPS, have been considered in the assessment of the environmental fatigue issue at PBAPS. Discuss how the applicable staff comments were considered in the evaluation of environmental fatigue.

b. Discuss the applicability of the component fatigue assessments in the EPRI Reports TR-107943 and TR-110356 to components in PBAPS. The discussion should include a comparison of design transients, operating cycles and fabrication details for each component. In addressing fabrication details, compare pipe diameters and thicknesses at PBAPS with the components evaluated in the EPRI reports. This comparison should also include a comparison of the fabrication details at the tee connections. Also include a comparison of the hydrogen water chemistry used at PBAPS with the hydrogen water chemistry considered in the EPRI reports.

c. The staff assessed the impact of reactor water environment on fatigue life at high fatigue usage locations and presented the results in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," March 1995. Formulas currently acceptable to the staff for calculating the environmental correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and those for austenitic stainless steels are contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels." Provide an assessment of the 6 locations identified in NUREG/CR-6260 for an older vintage BWR-4 considering the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 reports for PBAPS.

Response:

Section 4.3.4 of the LRA describes Exelon's evaluation of the impact of the reactor water environment on the fatigue life of the components identified in NUREG/CR-6260. In particular, that evaluation relied on several industry background studies that have been performed by EPRI and NEI to address EAF effects in reactor coolant system components. In particular, one study for a newer vintage BWR-4 plant was used to provide an assessment of environmental effects for PBAPS on the locations identified in NUREG/CR-6260 for the older vintage BWR plant using a design basis transient severity approach.

Because of the more recent issues raised by the NRC staff relative to the use of the EPRI/GE F_{en} methodology (Reference EPRI Report No. TR-105759) in various industry applications (as highlighted in the NRC staff 's August 6, 1999 letter to NEI), as well as additional laboratory fatigue data in simulated LWR environments that have been generated by Argonne National Laboratory (ANL) for carbon, low-alloy, and stainless steels (as published in NUREG/CR-6583 and NUREG/CR-5704), Exelon has decided to perform plant-specific calculations for PBAPS for the locations identified in NUREG/CR-6260 for the older vintage BWR plant. For each of these locations, detailed environmental fatigue calculations will be performed using the appropriate F_{en} relationships from NUREG/CR-6583 (for carbon/low alloy steels) and NUREG/CR-6704 (for stainless steels), as appropriate for the material for each location. The detailed calculations will include calculation of an appropriate F_{en} factor for each individual load pair in the governing fatigue calculation so that an overall multiplier on CUF for environmental effects can be determined for each location. These calculations will be performed prior to entry into the period of extended operation, and appropriate corrective action will be taken if the resulting CUF values exceed 1.0.

When completed, the plant-specific calculations for PBAPS are expected to validate the conclusions identified with respect to the EAF effects documented in the LRA. This modified approach is expected to adequately address the concerns identified in Items (a), (b), and (c) of this RAI.

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

Per the NRC/Exelon telecommunication discussion on January 3, 2002 the following additional clarification is provided. The first sentence on page 4-26 of the LRA section 4.3.4 is revised to state: "For relatively high temperature (>200°C), low dissolved oxygen, and a low (bounding) strain rate, the environmental shift correction factor may be as high as 15.35." The last part of this sentence is deleted.

RAI 4.3-7

Table 4.3.4-3 of the LRA provides projected 60-year fatigue usage factors for selected PBAPS components. Confirm that the usage factors reported for the feedwater line (RCIC Tee) are correct.

Response:

Table 4.3.4-3 of the PBAPS LRA documents a projected CUF of 0.049 for the RCIC tee for the limiting PBAPS unit. This value is correct, based on cycles experienced to-date. The value was determined as discussed below.

The governing design basis for the PBAPS Class 1 piping is ANSI B31.1. As a result, design basis CUF calculations do not exist for most of the Class 1 piping. (Note that some piping, like the recirculation piping and portions of the reactor water cleanup and residual heat removal piping, have been replaced at PBAPS and have a governing Class 1 fatigue analysis.) However, as a part of implementation of the improved fatigue monitoring program at PBAPS, CUF evaluation was performed for the limiting locations of the major Class 1 piping systems. This was done in order to establish a CUF basis for the Class 1 piping to include in the improved fatigue monitoring program. The feedwater piping was one of the major systems included in this evaluation.

The limiting 40-year CUF value for the feedwater piping was determined to be 0.069 at the weld between a 12" 90° short radius elbow and a 24"x24"x12" tee. This location bounded all other locations in the feedwater line, including the RCIC tee. As a result, it was considered to also represent the RCIC tee location identified in NUREG-6260. As discussed in the response to RAI 4.3-2 above, this location is included in the improved fatigue monitoring program as a CBF component (i.e., "Feedwater Piping" location shown in Table 1). Therefore, if PBAPS experiences all events assumed in the design basis in the same quantities assumed in the design basis, the fatigue monitoring program will reproduce the "design" CUF value of 0.069 for this location.

As discussed in the response to RAI 4.3-2 above for the CBF components, the initial CUF estimate at the time the fatigue monitoring program was implemented, was determined based on the cycle counts to-date since initial plant startup, and the design basis fatigue calculation methodology. This initial CUF estimate (covering approximately 27 years of plant operation for the limiting PBAPS unit) was determined to be 0.022 based on all cycles experienced to-date, and assuming design basis severity for each event. The linearly extrapolated CUF value for 60 years is therefore 0.022 (60/27) = 0.049.

Since this location is included in Exelon's improved fatigue monitoring program implemented at PBAPS, this issue will continue to be monitored throughout the period of extended operation. This monitoring will include revised CUF projections as events are experienced.

4.7.1 Reactor Vessel Main Steam Nozzle Cladding Removal Corrosion Allowance

RAI 4.7.1-1

The applicant should provide the basis for concluding that there will be 0.030 inches of corrosion over the 60 years of operation. Was it based on actual corrosion data? How was the data collected? Was the data specific to Peach Bottom?

Response:

The corrosion data was calculated as follows using the references identified below. The data is not specific to Peach Bottom but generic to BWRs.

Based on averaging the available data, the following average corrosion rates were obtained:

High temperature- 43.7 $mg/dm^2 \bullet mo$
Low temperature- 24.1 $mg/dm^2 \bullet mo$

Note: the low temperature rate is based on exposure to BWR conditions, which reduces the corrosion rate significantly.

Assuming 54 years at high temperature and 6 years at low temperature (90% availability), and doubling the average corrosion rate to determine the allowance, the following result was obtained:

2000 mg/dm^2 = 0.001 inch (from NEDO-13424)

Corrosion allowance = 43.7*2*648 months + 24.1*2*72 months
= 56635 + 3470
= 60105 mg/dm^2
= 0.030 inches

Therefore, the corrosion allowance of 1/16 inch is acceptable for 60 years.

References:

1. E.G. Brush, "Corrosion and Corrosion Product Release of Low Alloy Steels Under Simulated BWR Power Plant Operating Conditions", NEDO-13424, July 1975.

2. Vreeland, D.C, Gaul, G.G., and Pearl, W.L., "Corrosion of Carbon and Other Steels in Out-of-Pile Boiling-Water-Reactor Environment," Corrosion, Volume 18, October 1962, pp. 368-377.

3. Pearl, W.L., and Wozadlo, G.P., "Corrosion of Carbon Steel in Simulated Boiling Water and Superheat Reactor Environments," Corrosion, Volume 21, August 1965, pp. 260-267.

4. NEDE-13160 (Chemical Engineering Quarterly Report, 4th quarter, 1970), January 1971.