NEXTera ENERGY
DUANE ARNOLD

March 9, 2010

NG-10-0091
10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Responses to Requests for Additional Information Regarding the Duane Arnold Energy Center License Renewal Application - Batch 4

References:
1. Letter, Richard L. Anderson (FPL Energy Duane Arnold, LLC) to Document Control Desk (USNRC), "Duane Arnold Energy Center Application for Renewed Operating License (TSCR-109)," dated September 30, 2008, NG-08-0713 (ML082980623)
2. Letter, Richard L. Anderson (FPL Energy Duane Arnold, LLC) to Document Control Desk (USNRC), "License Renewal Application, Supplement 1: Changes Resulting from Issues Raised in the Review Status of the License Renewal Application for the Duane Arnold Energy Center," dated January 23, 2009, NG-09-0059 (ML090280418)
3. Letter, Brian K. Harris (USNRC) to Christopher Costanzo (NextEra Energy Duane Arnold, LLC), "Request for Additional Information for the Review of the Duane Arnold Energy Center License Renewal Application - Batch 4 (TAC No. MD9769)," dated February 22, 2010 (ML100471037)

By Reference 1, FPL Energy Duane Arnold, LLC submitted an application for a renewed Operating License (LRA) for the Duane Arnold Energy Center (DAEC). Reference 2 provided Supplement 1 to the application. By Reference 3 the U.S. Nuclear Regulatory Commission Staff (Staff) requested additional information for the review of the LRA. Enclosure 1 contains the DAEC responses to the NRC Requests for Additional Information.

The need for certain LRA clarifications or changes has also been identified through other NextEra Energy activities and interactions with the Staff. These clarifications and changes are provided in Enclosure 2.

This letter revises one commitment, withdraws one commitment and adds new license renewal commitment number 50, as discussed in Enclosure 2. Enclosure 3 provides a revised LRA Appendix A, Section 18.4, Table A-1, Duane Arnold License Renewal

A001

NextEra Energy Duane Arnold, LLC, 3277 DAEC Road, Palo, IA 52324

NRR

Commitments, updated to reflect the license renewal commitment changes made in DAEC correspondence to date.

If you have any questions or require additional information, please contact Mr. Kenneth Putnam at (319) 851-7238.

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

Executed on March 9, 2010.

Christopher R Costanzo

Christopher R. Costanzo
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Enclosures:
1. DAEC Responses to NRC RAIs
2. Other LRA Changes and Clarifications
3. License Renewal Commitment List Updated to Reflect DAEC Correspondence to date.

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Senior Resident Inspector, DAEC, USNRC
License Renewal Project Manager, USNRC
M. Rasmusson (State of Iowa)

RAI 4.3.3-3

Background

License renewal application (LRA) Section 4.3.3, "Fatigue Of Class 1, 2 And 3 Piping And Components" and the applicant's response to request for additional information (RAI) 4.3.3-2, by letter dated October 13, 2009, states that Class 1 piping with B31.7 methodology is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii). Furthermore, the applicant stated that as part of the evaluation, 60 year cumulative usage factors (CUFs) were evaluated based on the numbers of cycles provided in LRA Table 4.3-1.

Issue

In order for the Class 1 piping with B31.7 methodology to be appropriately dispositioned in accordance with 10 CFR 54.21(c)(1)(ii), the limits for the number of cycles that are being tracked by the Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program and the surveillance test program (STP) must be the 60 year projected cycles and not the 40 year design cycles. If the limits for the number of cycles being tracked are not the same as those used in the evaluation to disposition in accordance with 10 CFR 54.21(c)(1)(ii), than the evaluation is not valid for the period of extended operation.

Request

1. Please confirm that the limits for the number of cycles that will be tracked by the Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program and the STP are the same number of cycles that were used in the evaluation to disposition the Class 1 piping with B31.7 methodology in accordance with 10 CFR 54.21(c)(1)(ii) and that these cycles will be incorporated into the updated final safety analysis report (UFSAR) update.

2. If the limits that will be tracked by the Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program and the STP are not the same cycles that were used in the evaluation to disposition the Class 1 piping with B31.7 methodology in accordance with 10 CFR 54.21(c)(1)(ii), please justify the validity of dispositioning the Class 1 piping with B31.7 methodology in accordance with 10 CFR 54.21(c)(1)(ii) and clarify the action that is taken to ensure the CUF is below 1.0 for the Class 1 piping with B31.7 methodology during the PEO.

DAEC Response to RAI 4.3.3-3

Part 1

The 60 year design cycle values given in LRA Table 4.3-1 are the numbers that were used in the fatigue evaluations discussed in LRA Section 4.3.3. Prior to the period of extended operation (PEO), the surveillance test procedure (STP) which implements the Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program will be updated to reflect the 60 year design cycle values provided in LRA Table 4.3-1. The Updated Final Safety Analysis Report (UFSAR) will also be revised to reflect these 60 year design cycle limits. Therefore, the limits for the numbers of design cycles tracked by the STP will be the same values that were used in the evaluations to disposition the Class 1 piping with B31.7 methodology (as discussed in LRA Section 4.3.3).

Part 2

Not Applicable

RAI 4.3.4-3

Background

In response to RAI 4.3.4-2 (part 4, 6 and 7) by letter dated October 13, 2009, a F_{en} factor of 1.49 was used for the Alloy 600 and SB166 components. Also in its response the applicant stated a methodology from 1995/1996 was used for calculating the F_{en} factor of 1.49. In 2007, the staff issued Regulatory Guide 1.207, entitled "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," based on NUREG/CR-6909, which incorporates more recent fatigue data for determining the F_{en} factor for Nickel Alloys.

Issue

Although Regulatory Guide 1.207 specifically applies to new reactors, the basis methodology for F_{en} described in NUREG/CR-6909 is considered by the staff to represent the most up-to-date method for determining the F_{en} factor for Nickel Alloys for license renewal considerations.

Request

1. Please clarify if the value of 1.49 for the F_{en} factor is the bounding/conservative value for the Alloy 600 and SB166 components as compared to the values calculated from NUREG/CR-6909 for Nickel Alloys. If not, please justify the use of the a F_{en} factor of 1.49 for the Alloy 600 and SB166 components.

2. Describe the planned actions to update the CUF calculations with F_{en} factor for Alloy 600 and SB166 components consistent with the methodology in NUREG/CR-6909 or other acceptable methods.

3. Describe how the assumed 72.425% overall hydrogen water chemistry (HWC) availability will be accounted for when managing these components, since this overall HWC availability has an effect on the CUF values and may fluctuate based on actual plant operations during the PEO.

DAEC Response to RAI 4.3.4-3

Part 1

The value of 1.49 for F_{en} for Alloy 600/SB166 components is less than the value that would be calculated from NUREG/CR-6909 for nickel alloys. A review indicates that using NUREG-6909 methodology would result in a value of F_{en} of 3.31. This calculation was based on a temperature of 546 °F, and maximum values for ε' and O' from NUREG/CR-6909. The following relationship was used to account for overall hydrogen water chemistry (HWC) availability:

Overall $F_{en} = 0.72425*F_{en\ HWC} + (1-0.72425)*F_{en\ NWC}$

Overall $F_{en} = 0.72425*(3.77) + (1-0.72425)*(2.11) = 3.31$.

As discussed in letter NG-09-0764 dated October 13, 2009, the 1.49 value for F_{en} of nickel alloys is determined based on the Alloy 600 methodology documented in "Status of Fatigue Issues at Argonne National Laboratory," presented at EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, O. Chopra, Snowbird, UT, August 22-23, 1996. The Staff found this method acceptable in the Three Mile Island License Renewal Safety Evaluation Report (June 2009), stating:

> The maximum F_{en} value of 1.49 for Nickel Alloy 600 can be found in the paper entitled "Status of Fatigue Issues at Argonne National Laboratory," presented by Omesh K. Chopra at the EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, August 1996. The staff notes that the EPRI technical information (presented by Omesh K. Chopra) quoted herein for Alloy 600 F_{en} calculations can be found in NUREG/CR-6335, titled "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," authored by J. Keisler, O. K. Chopra, and W. J. Shack, dated August 1995.

In addition, NUREG-1800, Rev 1, Section 4.3.3.2 Generic Safety Issue, states that formulas for calculating the environmental life correction factors are those contained in NUREG/CR–6583 for carbon and low-alloy steels, and in NUREG/CR-5704 for austenitic stainless steels, or an approved technical equivalent. NUREG/CR-6909 has not been approved for existing reactors. Regulatory Guide 1.207 provides that NUREG/CR-6909 only applies to new plants, stating that "Because of significant conservatism in quantifying other plant-related variables such as cyclic behavior, including stress and loading rates involved in cumulative fatigue [life] calculations, the design of the current fleet of reactors is satisfactory."

In addition, even if a value of 3.31 is used to calculate the 60 year environmental CUFs for the DAEC Alloy 600/SB166 locations, the CUF values remain below 1.0 and are, therefore, acceptable.

Part 2

DAEC has no current plans to update the CUF calculations with an F_{en} factor for Alloy 600 and SB166 components consistent with the methodology in NUREG/CR-6909.

Part 3

The Metal Fatigue of Reactor Coolant Pressure Boundary Program will ensure that HWC availability is monitored to ensure that the assumptions in the environmental fatigue evaluations remain valid. As a part of the existing License Renewal Commitment 35, the Reactor Vessel Transient Design Cycles surveillance test procedure (which implements the Program) will be revised to refer to the DAEC Chemistry Strategic Plan.

RAI 4.6-1

Background

"Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR) (NUREG 1800), Section 4.6.1.1 states that containment liner plates, metal containment, penetration sleeves may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on time-limited aging analysis (TLAAs).

Issue

LRA Section 4.6, "Fatigue of Primary Containment, Piping, and Components," states that the number of safety relief value (SRV) lifts throughout the Duane Arnold Energy Center's (DAEC's) operating history have not been consistently tracked. To address LRA requirements, the historical number of SRV lifts was needed; documentation was therefore researched to determine the number of the SRV lifts from 1974 until 2007. Using this information and projecting the results for 60 years provided a projected number of 334 single SRV lifts for 60 years and a projected number of 42 multiple lifts for 60 years.

Request

Provide the actual number of SRV lifts from 1974 to 2007. In addition, explain how the SRV lifts will be monitored during PEO since the number of SRV lifts were not tracked until 2007. The staff needs the above information and analyses that were performed in 2007 to confirm that the an evaluation of TLAA will remain valid for the period of extended operation as required by 10 CFR 54.21(c).(1)(ii).

DAEC Response to RAI 4.6-1

As stated in LRA Section 4.6, documentation was researched to determine the number of SRV lifts which had occurred from 1974 until 2007. This documentation included Licensee Event Reports, Monthly Operating Reports and Annual Reports of Safety and

Relief Valve Failures and Challenges. For conservatism, an additional 10% was added to the number of SRV lifts determined by this review.

The review determined that, from 1974 until the beginning of 2007, the numbers of SRV lifts which have occurred from normal operating conditions (including the 10% factor for conservatism) are:

- 219 single SRV lifts, and
- 33 multiple SRV lifts.

In 2007, for containment transient monitoring purposes, the "Reactor Vessel Transient Design Cycles" surveillance test procedure (STP) was revised to establish new requirements for monitoring SRV lifts and to incorporate the above totals of SRV lifts accumulated to date. The procedure requirements include recording the number of SRV lifts occurring since the last performance of the STP, computing the new totals of SRV lifts accumulated to date, and comparing the new totals with the assumptions in the Plant Unique Analysis Report (PUAR). This STP had already been in place to monitor the numbers of accumulated transient cycles for the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The STP is performed, at a minimum, once each fuel cycle. Including the requirement to monitor SRV lifts in this STP ensures that the number of lifts will be tracked and recorded throughout the PEO.

RAI 4.6-2

Background

SRP-LR (NUREG 1800), Section 4.6.1.1 states that the containment liner plates (including welded joints), penetration sleeves, etc., may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on TLAAs.

Issue

LRA Section 4.6.1 states that fatigue usage factors for the suppression chamber (torus) have been projected through the PEO. These fatigue usage factors have been determined based on an assumed number of thermal cycles for 60 years of operation. The LRA does not list the actual number of thermal cycles the suppression chamber (torus) has experienced until now.

Request

Explain how the numbers of thermal cycles are monitored for suppression chamber and actual count of thermal cycles the suppression chamber have experienced. The staff needs the above information that the an evaluation of the TLAA will remain valid for the PEO as required by 10 CFR 54.21(c).(1)(ii).

DAEC Response to RAI 4.6-2

The fatigue analysis of the suppression chamber is based on a defined number of single and multiple SRV lifts from normal operating conditions (NOC), and one small-break or intermediate-break accident. The CUF contribution from SRV lifts and the CUF contribution from the accident are summed to derive the maximum CUF for the suppression chamber. The maximum CUF value, as reported in LRA Section 4.6.1, is 0.467. This value is the sum of the usage factors from SRV lifts (740 single and 60 multiple from NOC) and a small-break accident (more limiting for fatigue than the intermediate-break accident). The only contributing events that need to be explicitly tracked are the SRV lifts.

As discussed in the response to RAI 4.6-1 above, SRV lifts are recorded and tracked via the "Reactor Vessel Transient Design Cycles" surveillance test procedure (STP). The numbers of SRV lifts accumulated as of the beginning of 2007 are 219 single SRV lifts and 33 multiple lifts (including the 10 % factor for conservatism). Projecting the number of SRV lifts for 60 years, using the relationship provided in the DAEC Response to RAI 4.3.1-2 in NG-09-0764 dated October 13, 2009, results in a projected number of 334 single SRV lifts for 60 years, and a projected number of 42 multiple lifts for 60 years. The number of SRV events assumed for the suppression pool fatigue analysis is 740 single lifts and 60 multiple lifts.

Thus the relevant suppression chamber cycles are monitored via the "Reactor Vessel Transient Design Cycles" STP.

RAI 4.6-3

Background

SRP-LR (NUREG 1800), Section 4.6.1.1 states that containment penetration bellows may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on TLAAs.

Issue

LRA Section 4.6.2 states that for vent lines bellows, thermal load is the largest contributor to displacements. The plant unique analysis report specifies 150 thermal load cycles. However, the LRA does not list the actual number of thermal cycles the bellows have experienced until now.

Request

Explain how the numbers of thermal cycles are monitored for containment bellows and provide the actual count of thermal cycles the bellows have experienced. The staff needs the above information and to confirm that the an evaluation of the fatigue of the bellows provided in the LRA will remain valid for the PEO as required by 10 CFR 54.21(c).(1)(ii).

DAEC Response to RAI 4.6-3

The most significant cyclic loading on the containment bellows is associated with accident loads that are not affected by license renewal. There is no significant non-accident source of normal cyclic thermal loads on the containment vessel and bellows beyond those that would be generally associated with startup/shutdown of the reactor and those associated with SRV lifts. Assuring that the numbers of startups/shutdowns and SRV lifts are within design assumptions will assure that the bellows remain within fatigue limits.

Reactor startup/shutdown cycles are recorded and tracked via the "Reactor Vessel Transient Design Cycles" surveillance test procedure (STP), which monitors the numbers of accumulated transient cycles for the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The number of cycles accumulated after approximately 30 years of operation and the number projected for 60 years are provided in LRA Table 4.3-1. In addition, as discussed in the response to RAI 4.6-1 above, SRV lifts are recorded and tracked via the same procedure. The number of SRV lifts accumulated at the beginning of 2007, and the number projected for 60 years are discussed in the response to RAI 4.6-2 above. Thus the cyclic loading of the containment bellows is adequately monitored via the "Reactor Vessel Transient Design Cycles" STP.

RAI 4.6-4

Background

SRP-LR (NUREG 1800), Section 4.6.1.1 states that the containment liner plates (including welded joints), penetration sleeves, etc., may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on TLAAs.

Issue

LRA Section 4.6.3 states that for suppression chamber external piping and penetrations have been projected through the period of extended operation. However, LRA does not list the actual number of thermal cycles that the suppression chamber external piping and penetrations have experienced until now. In addition, the LRA states that 334 single SRV lifts are projected for 60 years, and 42 multiple lifts are projected for 60 years of operation.

Request

Explain how the numbers of thermal cycles are monitored for suppression chamber external piping and penetrations, and provide the actual count of thermal cycles the suppression chamber external piping and penetrations have experienced. In addition, please provide the actual number of SRV lifts between 1974 through 2007 that were used to project the SRV lifts through the PEO. The staff needs the above information and analyses that were performed in 2007 to confirm that an evaluation of TLAA will remain valid for the PEO as required by 10 CFR 54.21(c).(1)(ii).

DAEC Response to RAI 4.6-4

The most significant cyclic loading on the piping penetrating containment is associated with accident loads that are not affected by license renewal. There is no significant non-accident source of normal cyclic thermal loads on the containment vessel and penetrating piping beyond those that would be generally associated with startup/shutdown of the reactor and those associated with SRV lifts. Assuring that the numbers of startups/shutdowns and SRV lifts are within design assumptions will assure that the piping and penetrations remain within fatigue limits.

Reactor startup/shutdown cycles are recorded and tracked via the "Reactor Vessel Transient Design Cycles" surveillance test procedure (STP), which monitors the numbers of accumulated transient cycles for the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The number of cycles accumulated after approximately 30 years of operation and the number projected for 60 years are provided in LRA Table 4.3-1. In addition, as discussed in the response to RAI 4.6-1 above, SRV lifts are recorded and tracked via the same procedure. The number of SRV lifts accumulated at the beginning of 2007, and the number projected for 60 years are discussed in the response to RAI 4.6-2 above. Thus the cyclic loading of the piping penetrating containment is adequately monitored via the "Reactor Vessel Transient Design Cycles" STP.

RAI 4.6-5

Background

SRP-LR (NUREG 1800), Section 4.6.1.1 states that metal containments may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on TLAAs.

Issue

LRA Section 4.6.4 states that the stress report includes a determination that the containment vessel is exempt from fatigue analysis. This determination is based on an assumed number of load fluctuations for 40 years of operation. After increasing this number to account for the additional cycles of a 60 year life, the fatigue analysis exemption remains valid.

Request

Explain why an assumed number of load fluctuations are used instead of the actual number experienced by the containment vessel. In addition, provide the basis that has been used in the stress report to determine that containment vessel is exempt for fatigue analysis. The staff needs the above information to confirm that the an evaluation of TLAA will remain valid for the PEO as required by 10 CFR 54.21(c).(1)(ii).

DAEC Response to RAI 4.6-5

The term "assumed" used in LRA Section 4.6.4 applied to the number of cycles assumed in the original exemption from fatigue evaluation performed during design, prior to the start of plant operation. The actual number of load fluctuations (startup cycles) at DAEC was used to determine the projected value for 60 years of operation. The relationship used to project the transient counts forward to 60 years is provided in the DAEC response to RAI 4.3.1-2 in letter NG-09-0764 dated October 13, 2009. This projection was based on trending the cycles experienced from the 1998 through 2005 time period of plant operation. The number of cycles determined by this projection is used to demonstrate that the containment vessel remains exempt from fatigue analysis for 60 years of operation.

As stated in LRA Section 4.6.4, the DAEC containment stress report includes an evaluation performed in accordance with Section III of the ASME Code, paragraph N-415.1, to determine whether a fatigue analysis is required for the containment vessel. The evaluation determined that the containment satisfies conditions of ASME Code Section III Paragraph N-415.1 for all specified conditions (for 40 years) and therefore, an analysis for cyclic operation is not required. This evaluation was reviewed to ensure that the containment vessel remains exempt from a fatigue analysis (for 60 years of operation).

General information regarding fatigue exemptions in accordance with ASME Section III Paragraph N-415.1 is provided in the DAEC Response to RAI 4.3.1-1 in letter NG-09-0764 dated October 13, 2009. As discussed in that response, if certain requirements outlined in Paragraph N-415.1 are met, a fatigue analysis is not required.

The paragraphs below show how Paragraph N-415.1 was used to demonstrate that the containment does not require analysis for cyclic operation for 40 years or 60 years of operation. As discussed in LRA Section 4.6.4, 100 cycles were assumed for the 40 year evaluation and 212 cycles are assumed for the 60 year evaluation. The range of pressure fluctuation used is -2 psi to +2 psi. The maximum temperature fluctuation used is 50 °F.

<u>Pressure Cycle</u>
For a value of S_a equal to $3S_m$, the corresponding permissible number of cycles is 3500. Since 3500 cycles > 100 cycles, the requirement of N-415.1(a) is met for 40 years of operation. Since 3500 cycles > 212 cycles, the requirement of N-415.1(a) is met for 60 years of operation.

<u>Pressure Fluctuation</u>
The 40 year evaluation demonstrated that this requirement is met by determining that a significant pressure fluctuation corresponds to 14.08 psi, which is greater than 4 psi (the range from -2 psi to +2 psi). Increasing the number of cycles does not affect this determination; therefore the requirement of N-415.1(b) is met for 60 years.

Temperature Variation between Adjacent Points

The 40 year evaluation addressed this requirement as "none specified;" therefore analysis for cyclic operation is not required by N-415.1(c). This remains unchanged for 60 years.

ΔT during Normal Operation

The 40 year evaluation addressed this requirement as "none specified;" therefore analysis for cyclic operation is not required by N-415.1(d). This remains unchanged for 60 years.

Components fabricated from Different Materials

The 40 year evaluation addressed this requirement by determining the significant temperature fluctuation to be 86.8 °F and showing it to be greater than the maximum temperature difference of 50°F. Increasing the number of cycles does not affect this determination; therefore the requirement of N-415.1(e) is met for 60 years.

Cyclic Mechanical Loads

The 40 year evaluation determined a value of S_a for 100 cycles of about 200,000 psi. This value is greater than the allowable used in the containment vessel design (= 3Sm = 52,500 psi). Therefore the requirement of N-415.1(f) is met for 40 years. Substituting 212 cycles (for 60 years), a value for S_a of about 150,000 psi is obtained. Since this value is greater than 52,500 psi, the requirement of N-415.1(f) is met for 60 years.

Therefore, the containment satisfies ASME Code Section III Paragraph N-415.1 for all specified conditions, and an analysis for cyclic operation is not required for 60 years of operation.

RAI 4.6-6

Background

SRP-LR (NUREG 1800), Section 4.6.1.1 states that penetration sleeves may be designed and/or analyzed in accordance with ASME Code requirements. The ASME Code contains explicit metal fatigue or cyclic considerations based on TLAAs.

Issue

LRA Section 4.6.5 states that the verification of adequacy of the fluid heads penetrations was determined by comparing allowable stresses (based on assumed number of cycles) and maximum stress intensities.

Request

Explain why an assumed numbers of cycles instead of actual number of cycles tracked since the start of plant operation were used to project the results for 60 years of operation. The staff needs the above information to confirm that the an evaluation of TLAA will remain valid for the PEO as required by 10 CFR 54.21(c)(1)(ii).

DAEC Response to RAI 4.6-6

An assumed number of cycles is used since Duane Arnold does not count the cycles on the flued heads individually. The assumed numbers of cycles on the flued heads are based on the numbers of reactor vessel transient cycles assumed to occur.

The original 40 year analyses demonstrated acceptability of the flued heads by comparing the maximum stress intensity and allowable stress for various load combinations and conditions. These analyses used an assumed number of normal and upset cycles that the flued heads could experience during 40 years. These assumed numbers of cycles are based on the numbers of reactor vessel design transient cycles assumed to occur, added to a number of cycles assumed to result from earthquakes. The number of earthquake cycles assumed in the analysis is larger than the number of cycles from any other event by at least a factor of 2. The numbers of cycles from all events were added to derive the total number of cycles used in the analysis.

The 60 year analyses in LRA Section 4.6.5 multiplied the number of 40 year assumed cycles by 1.5 to extrapolate the number to 60 years. This assumes that the number of cycles accumulated per year, including cycles contributed by earthquakes, is linear over the life of the plant. This is conservative since the number of earthquake cycles to be accounted for in the design should not increase due to license renewal.

Another conservatism in this assessment is that the total number of reactor vessel design cycles in 60 years, as reported in LRA Table 4.3-1, is actually smaller than the 40 year total, and the cycles experienced by the flued heads are based on the number of vessel cycles. Therefore, increasing the 40 year number of normal and upset design cycles for the flued heads by multiplying the total by 1.5 to account for 60 years is conservative.

Based on the 60 year analyses, as long as the number of reactor vessel transient cycles remains at or below those listed in LRA Table 4.3-1, the cycles on the flued heads will remain within acceptance criteria. The reactor vessel transient cycles are tracked by the Reactor Vessel Transient Design Cycles surveillance test procedure.

RAI B.3.15-X

<u>Background</u>

The “preventive actions” program element of the GALL Report AMP XI.M21 recommends maintaining system corrosion inhibitor concentrations within the specified limits of EPRI TR-107396 to minimize corrosion and SCC. The “acceptance criteria” program element recommends that corrosion inhibitor concentrations are maintained within the limits specified in the EPRI Closed Cooling Water Chemistry Guideline, which in Table 5-1, indicates that azoles are a monitored parameter unless it can be documented that there are no copper alloys in the system. The LRA indicates in Section 3.3.1.22 that the reactor building closed cooling water system contains copper; however, the basis documents do not appear to indicate that azoles are used in the reactor building closed cooling water system. RAI B.3.15-3 requested additional information describing why azoles are not used and monitored in the reactor building closed cooling water system as suggested in the EPRI Closed Cooling Water Chemistry Guidelines. The response to this RAI stated, "The Reactor Building Closed Cooling Water (RBCCW) system soluble copper historically has been less than 100 ppb. The industry best practices recommended range is less than 200 ppb. DAEC determined that the use of azoles in the RBCCW would be unlikely to provide a measurable reduction in the system soluble copper levels."

Issue

The staff notes that for a filtered sample (as described in the testing procedure provided) that the solubility of copper at pH values typical of closed systems is 100 ppb or less. The test results indicating 100 ppb dissolved copper do not, therefore, indicate that the corrosion rate of copper is low as all corroded copper in excess of approximately 100 ppb would precipitate and would not be measured in the test.

Request

Please provide additional information indicating that the corrosion rate of copper is sufficiently low that copper corrosion inhibitors are not required.

DAEC Response to RAI B.3.15-X

Copper is analyzed by an Inductively Coupled Plasma spectrophotometer (ICP). The ICP analyzes all copper in the sample, including dissolved and suspended particulate copper. Therefore, the DAEC copper analysis results represent total copper in the samples.

The total copper concentrations measured in the RBCCW system typically range between 35-140 ppb and have been stable in this range for several years. DAEC does not add Tolytriazole (TTA) or other types of azoles to this system. Historically, the copper in the RBCCW system has been stable and less than the maximum permissible value specified in the GE Water and Process Technologies Best Practices (< 200 ppb total copper concentration). For comparison, it is noted that total copper concentrations measured in another system using this analysis method have been observed as high as 1800 ppb even with the addition of azoles.

Based on this experience, DAEC has determined that the use of azoles in the RBCCW system is unnecessary.

Follow-up RAI B.3.37-1

Background

In letter NG-09-0764, dated October 13, 2009, the applicant responded to RAI B.3.37-1 related to the inspection interval of the Structures Monitoring Program. The applicant explained that based on operating experience, the five and ten year +/- one year inspection interval would be adequate to detect degradation.

Issue

GALL Report AMP XI.S6 states that inspection schedule for structures to be commensurate with industry codes, standards and guidelines. AMP XI.S6 further states that ACI 349.3R and ANSI/ASCE 11-90 provide an acceptable basis for addressing detection of aging effects. The staff recognizes that ACI 349-3R recommends a 10 year interval for structures subjected to a controlled interior environment. However, industry operating experience indicates that nuclear power plant structures under controlled interior environments are subjected to high radiation, high temperature, and water leakages.

Request

Provide a list of in-scope structures that will be inspected on a 10 year interval, along with their ambient environment and operating experience. This information is necessary for the staff to complete its review of AMP B.3.37 and to verify that that the DAEC inspection interval for structures is commensurate with industry operating experience, and codes and standards.

DAEC Response to Follow-up RAI B.3.37-1

The Maintenance Rule Structures Monitoring Program categorizes the environments for structures subject to the program as follows:

Harsh -- an area routinely subjected to outside ambient conditions, high moisture or humidity, very high ambient temperatures or frequent large cycling of temperatures (including freezing/thawing), frequent exposure to caustic materials, or extremely high radiation levels.

Mild -- an area which is not harsh.

The normal temperature in the Control Building is 75° F but varies somewhat from room to room. The normal temperature in the Pumphouse and Intake Structure is in the range of 70° to 85° F and varies somewhat with equipment operation and season. These buildings have environments classified as mild as noted in the table below. The normal temperature in the Reactor Building is in the range of 75° F to 90° F but varies somewhat depending on equipment operation and season; the Reactor Building contains two harsh areas as noted in the table. The normal temperature in the Turbine Building is in the range of 80° F to 100° F but varies somewhat depending on the room, equipment operating status and season; the Turbine Building contains one harsh area as noted in the table.

The table below shows the inspection frequency for the ACI-349-3R structure categories, and the DAEC license renewal in-scope structures and environments that correspond to each category.

Structure category per (ACI-349-3R)	Frequency of visual inspection	Structures in-scope of License Renewal	Environment
a) Below-grade structures	10 years (each ISI interval) (if exposed for any reason)	Reactor Building, Control Building, Pump house, Intake Structures, Turbine Building, Off-Gas vent stack	Mild
b) Structures exposed to natural environment (direct/indirect)	5 years (two per ISI interval)	Reactor Building Roof and Exterior, Control Building Roof and Exterior, Pump House Roof and Exterior, Intake Structures Roof and Exterior, Turbine Building Roof and Exterior, Miscellaneous Yard Structures, Off-Gas vent stack	Harsh
c) Structures inside primary containment	5 years (two per ISI interval)	Primary Containment	Harsh
d) Continuous fluid-exposed structures	5 years (two per ISI interval)	Pump House Pits and Intake Structure Pits	Harsh
e) Structures retaining fluid and pressure	5 years (two per ISI interval)	None	Harsh
f) Controlled interior environment	10 years (each ISI interval)	Reactor Building, Control Building, Pump house, Intake Structure, Turbine Building, Radwaste Building, Off-Gas Retention Building, Low Level Radwaste Storage Facility, Railroad Airlock, Machine shop	Mild
g) DAEC isolated locations with harsh environments**	5 years (two per ISI interval)	Reactor Building Main Plant Intake Coil Room and Steam Tunnel. Turbine Building Heater Bay.	Harsh

**These are specific areas within the listed DAEC structures which have harsh environments that warrant a 5 year inspection frequency. The other areas in these structures have mild environments that may be inspected on a 10 year frequency.

RAI B.3.40-1

Background

In the DAEC LRA Section B.3.40.7, "Parameters Monitored or Inspected," and Section B.3.40.8, "Detection of Aging Effects," the applicant stated that it will perform a volumetric examination on a minimum of 10 percent of the ASME Code Class 1 small bore butt welds in each inspection interval during the PEO.

Issue

The applicant's did not provide any technical basis regarding its sampling criteria.

Request

Provide a justification for the sampling criteria and explain why selecting ten percent each inspection interval is adequate.

DAEC Response to RAI B.3.40-1

The ten percent sampling criterion for the ASME Category B-F and B-J Class 1 piping welds is derived from the NRC-approved Risk-Informed Inservice Inspection methodology of EPRI TR-112657, Revised Risk-Informed Inservice Evaluation Procedure, 1999. The NRC originally approved this methodology in a letter dated October 28, 1999. The use of the EPRI TR-112657 methodology was approved by the NRC for the DAEC Risk-Informed Inservice Inspection (RI-ISI) Program during the third inspection interval in a letter dated January 17, 2003, and for the fourth interval in a letter dated January 31, 2007.

ASME Section XI, Subsection IWB, Table IWB-2500-1, specifies that Examination Categories B-F (Class 1 nozzle to safe end dissimilar metal welds) and B-J (Class 1 circumferential piping welds) require surface examination and not volumetric examination in piping less than NPS 4". The EPRI TR-112657 methodology applies the volumetric examination requirement to butt welds in piping between NPS 4" and NPS 2" instead of surface examination. The number of elements to be examined in Category B-J welds by this RI-ISI methodology is consistent with ASME Code Case N-560, and the selection of elements to be examined is consistent with ASME Code Case N-578 which emphasizes the selection of higher risk locations. Section 3.6.4.1 of EPRI TR-112657 states, "The number of elements to be volumetrically examined as part of the RI-ISI program is defined in ASME Code Case N-560 as 10 percent of the piping weld population." As discussed in letter NG-02-0259 dated March 29, 2002, DAEC has applied this ten percent criterion to the total number of Class 1 Category B-J welds, including both socket welds and non-socket welds. DAEC has also applied this ten percent sampling criterion to the population of Class 1 small-bore butt welds as stated in LRA Section B.3.40.1 (submitted in letter NG-09-0764 dated October 13, 2009).

In the October 1999 Safety Evaluation (SE) for the approval of TR-112657, in Section 3.3.1, Selection of Examination Locations, the NRC Staff discusses the ASME Code

Case N-560 guideline that the number of elements to be selected for examination by the application of this methodology should be 10 percent of the total weld population. The Staff found the sampling criteria acceptable, and stated in the last paragraph of SER Section 3.3.1, "The staff finds that this degradation review process, augmented by the selection of higher risk locations, is a systematic and reasonable method for considering engineering and risk insights in establishing a program to assess service-induced degradation due to variable, localized corrosion."

The DAEC RI-ISI Program was developed in accordance with the guidance of EPRI TR-112657. In a letter dated March 29, 2002, DAEC submitted the proposed RI-ISI Program for NRC review. The submittal contained an explanation of how 10 % of the population of Class 1 welds would be inspected to meet the EPRI TR-112657 guidelines. In a letter dated January 17, 2003, the NRC approved the use of the RI-ISI Program at DAEC. The approval letter stated that the results of the NRC staff's review indicate that the proposed RI-ISI program is an acceptable alternative to the ISI requirements of the ASME Code, Section XI, and therefore, the request for relief is authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the alternative provides an acceptable level of quality and safety. Subsequently, in a letter dated January 31, 2007, the NRC approved the continuation of the RI-ISI Program at DAEC for the fourth 10 year inspection interval.

It is recognized that the relief request for implementing an RI-ISI Program is not currently approved for the period of extended operation. 10 CFR 50.55a requires that inservice inspection of Class 1, 2, and 3 pressure retaining components, their integral attachments, and supports be conducted in accordance with the latest edition of ASME Section XI approved by the NRC twelve months prior to the start of a ten year interval. DAEC will start a new inspection interval at the start of the period of extended operation. For the fifth inspection interval (first during the period of extended operation), it is DAEC's current expectation that NRC approval will be requested to continue RI-ISI as an alternative inspection program that provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a (a)(3)(i). As discussed in LRA Section B.3.40.1 (submitted in letter NG-09-0764 dated October 13, 2009), DAEC will continue to perform volumetric examination of a minimum of ten percent of small bore Class 1 butt welds during the period of extended operation, whether or not the NRC approves continuation of the RI-ISI program.

In summary, the ten percent sampling criterion for ASME Category B-F and B-J piping welds is derived from the NRC-approved Risk-Informed Inservice Inspection methodology of EPRI TR-112657. This methodology has been approved by the NRC for use at DAEC. The continued use of the10% criterion for inspection of ASME Code Class 1 small-bore butt welds provides reasonable assurance that small-bore piping will continue to meet its current licensing basis requirements during the period of extended operation.

RAI B.3.40-2

Background

In the DAEC LRA Section B.3.40.7, "Parameters Monitored or Inspected," and Section B.3.40.8, "Detection of Aging Effects," the applicant stated that it will perform a VT-2 visual inspection during system leakage tests on its socket welds each refueling outage per the requirements of IWB-2500-1, examination Category B-P.

Issue

GALL XI.M.35 recommends volumetric examinations. The applicant's program deviates from GALL recommendations. The staff believes VT-2, or any form of a surface examination, is only leakage detection. It will not detect any degradation until a component leaks and loses its intended function. Therefore, by the time an indication is detected by VT-2, the subject component has already failed.

Request

Explain how visual inspection is sufficient in aging management of Class 1 socket welds. Address volumetric examination of socket welds. The staff has discussed with the applicant in length regarding this disagreement and will make it an open item until this issue is adequately addressed.

DAEC Response to RAI B.3.40-2

Introduction

Aging of Class 1 small-bore piping socket welds at DAEC is managed by multiple programs. These include the Water Chemistry Program; the ASME Section XI Inservice Inspection, Subsection IWB, IWC and IWD Program; and the ASME Code Class 1 Small-bore Piping Inspection Program. Together, these programs provide preventive measures to minimize the potential for cracking, and inspection and monitoring activities to assure that the condition of the welds remains acceptable in service.

Over the history of the nuclear industry, some Class 1 socket welds have been identified as degraded. The Licensee Event Report (LER) review discussed below indicates that nearly all, if not all, of this weld degradation has been discovered before gross failure occurred, during code-required surface examinations or as minor leakage during visual inspection or inservice monitoring. Leakage in a Class 1 socket weld would indicate there may be a reduction in margin in the leaking component, but the leakage would typically not have a significant impact on the safety functions of the Reactor Coolant System (RCS) or the small-bore line itself. The DAEC Technical Specifications address the potential for this type of leakage and define the responses required if leakage limits are exceeded. Therefore, the potential that a flaw could develop and cause leakage or gross failure in a Class 1 small-bore socket weld has relatively low safety significance.

The following discussion explains why the code-required surface examinations, coupled with visual inspection to detect the onset of leakage and identify the need for prompt repair, have provided and will continue to provide reasonable assurance that Class 1 small-bore socket welds are acceptable to support RCS integrity in a manner that prevents an accident and protects the health and safety of the public.

<u>Aging Effects/Mechanisms of Interest in Class 1 Small-bore Socket Welds</u>

Class 1 small-bore piping comprises part of the reactor coolant system pressure boundary, and therefore, is exposed to an internal environment of reactor coolant. For the purpose of this response, the primary aging effect of interest in Class 1 small-bore piping socket welds is cracking. As discussed in LRA Section B.3.39, cracking due to SCC is managed for the entire Reactor Coolant System by the Water Chemistry Program. This program minimizes the potential for SCC in Class 1 systems.

Fatigue cracking can also occur due to cyclic loading. A piping configuration that renders a socket weld vulnerable to damage from cyclic loading can be considered to have a design or fabrication deficiency, since the design of the component and its supports should fully account for expected cyclic loading. An example of such a design deficiency is where a pipe segment is configured such that its natural frequency is a harmonic of an excitation frequency from the associated operating equipment (e.g., pump pulse rate, rotational speed). The resulting sympathetic vibration in the pipe can amplify pipe oscillation and, ultimately, cause fatigue cracking. An example of a fabrication deficiency is where a pipe is inserted fully into the fitting socket without the required expansion gap. When such a joint is heated, there is insufficient room for expansion that can result in very high stresses in the weld and eventual weld cracking due to fatigue. As discussed below, and in EPRI-TR-1015138, "Nondestructive Evaluation, License Renewal - Small bore Piping Evaluation Process," December 2007, operating experience (OE) indicates that most failures of Class 1 socket welds have resulted from vibration fatigue.

Vibration fatigue is not an aging effect requiring management in the context of license renewal, and it is not addressed by the ASME Code Class 1 Small-bore Piping Inspection Program. Most vibration-induced failures are expected to occur early in plant life. While it is acknowledged that failures due to vibration may occur later in plant life due to infrequent or changing plant operations, procedure changes, operating events, etc., these failures are not age-related. The NRC has documented this position previously, including in NUREG 1931, "Safety Evaluation Report Related to the License Renewal of Susquehanna Steam Electric Station, Units 1 and 2," November 2009. Section 3.0.3.1.18, on page 3-70, of the SER states:

> The staff notes that in a letter dated September 30, 2008, the applicant revised LRA Section B.2.31 by deleting the discussions related to small bore piping failures attributed to vibrational (high-cycle) fatigue. The applicant made this change because the Small Bore Class 1 Piping Inspection Program is credited with managing age-related cracking due to stress corrosion or thermal and mechanical loading, but not with managing cracking due to high-cycle, vibrational fatigue, which is a short-term failure mechanism, not a long term aging mechanism.

The staff finds this LRA change acceptable because it deletes from the LRA the discussion of a short-term failure mechanism that is not managed by the Small Bore Class 1 Inspection Program, and because the Small Bore Class 1 Inspection Program, including all revisions to the LRA, is consistent with the corresponding AMP as described in the GALL Report.

Description and Safety Significance of DAEC Small-bore Socket Weld Population

The total population of Class 1 small-bore socket welds installed at DAEC is 118 welds. The DAEC Risk-Informed Inservice Inspection Program, using the NRC-approved methodology of EPRI-TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," 1999, has evaluated the safety significance of these Class 1 socket welds. The evaluation methodology of EPRI TR-112657 considers the failure potential due to applicable degradation mechanisms and the consequences of a failure as it assigns each pipe segment to an overall risk category. Potential degradation mechanisms considered in this evaluation include (as applicable) thermal fatigue, corrosion cracking, PWSCC, IGSCC, MIC, erosion-cavitation and flow-accelerated corrosion. A final overall risk category of high, medium, or low is assigned to each Class 1 pipe segment.

The resulting risk categories of the Class 1 small-bore socket welds installed at DAEC are:

- 0 Class 1 socket welds assigned to risk category "high"
- 11 Class 1 socket welds assigned to risk category "medium"
- 107 Class 1 socket welds assigned to risk category "low"

This clearly indicates that the potential for failures in the Class 1 small-bore socket weld population at DAEC does not contribute significant risk to plant nuclear safety.

Operating Experience Related to Class 1 Small-bore Socket Welds

As discussed in LRA Section B.3.40.14 (letter NG-09-0764 dated October 13, 2009), the OE review for the ASME Class 1 Small-bore Piping Inspection Program identified three relevant plant-specific events involving small-bore piping at DAEC. Only one event involved Class 1 piping.

The 1989 event on a Class 1 Recirculation Pump Suction Drain Line involved a pin hole leak discovered in a 2" socket weld at an elbow. The metallurgical evaluation of this defect indicated that the apparent cause was fatigue cracking due to vibration induced by the pump. Improper fit up at construction may have contributed to the failure. Since the replacement of this drain line in 1989, no further cracking of Class 1 small-bore piping has been observed at DAEC.

The other two events identified in the LRA program description involved non-Class 1 CRD withdrawal and insertion lines. The event in 1988 resulted from TGSCC initiated from the outside of the piping due to the lines being contaminated with chlorides leached from insulation in the vicinity. The event in 1990 involved cracking in fillet welds attaching the CRD lines to the drywell wall; the evaluation indicated that the cracking characteristics suggested that the cracking initiated and propagated by high cycle fatigue. Neither of the CRD withdrawal and insertion line events involved socket welds.

The LER review initially identified 139 LERs for defects in Class 1 and non-Class 1 small-bore piping socket welds in the commercial nuclear industry. This number considers all the events identified by the NRC staff and shared with the nuclear industry (ML093380259), and additional events identified by the industry and shared with the NRC (ML093380258), in January 2010.[1] These events were evaluated using best-available data with the following result:

Event Category	Total Removed from Further Consideration	Total Remaining to be Considered
Initial population of potential small-bore socket weld failure events identified by LER review	-	139
Less events in non-Class 1 piping	97	42
Less events found not to involve socket welds	2	40
Less events identified to result from construction defects	15[2]	25
Less events where cracks were due to fatigue	21[3]	4
Less events due to transgranular stress corrosion cracking from the outside due to chlorides leaching from the insulation	2	2
Events which had no determination of cause	2	0
Final population of known Class 1 socket weld failures due to SCC in a reactor coolant environment[4]	-	0

[2]Construction defects were identified as pipe bottomed in fitting socket, weld porosity or lack of fusion.

[3]Of the 21 fatigue-related events, 13 were specifically identified as due to cyclic loading/vibration. Five of the remaining events identified weld defects as contributors to fatigue cracking; two attributed cracking to excessive vibration of a reactor coolant pump; and one states the cause as high cycle fatigue with no further explanation.

[4]One leak through a weld defect was reported to result from IGSCC. The report stated that high stress existed due to poor fit-up and poor weld quality from original construction, and a weld repair had been made that also increased stress and caused sensitization of the material. This event was considered a fabrication defect and was not counted as a Class 1 socket weld failure due to SCC.

[1] Of 141 rows in the consolidated NRC/industry Licensee Event Report list, characterized elsewhere as 141 events, one was the title line and one was a duplicate entry. This results in 139 events.

The DAEC conclusions from this LER review are as follows:

- Most of the small-bore piping weld defects identified in the LER review occurred in lines that were not Class 1 and, therefore, would not be in the scope of the ASME Code Class 1 Small-bore Piping Inspection Program. Since non-Class 1 lines may not be subject to the same water chemistry controls as Class 1 lines, and may not be subject to the same design requirements or fabrication quality controls, non-Class 1 leakage or failure events should not automatically be viewed as being representative of Class 1 small-bore socket welds.

- The industry LER review identified a total of 40 Class 1 socket welds in commercial U.S. nuclear plants that have experienced defects. This review identified no gross failures of Class 1 small-bore socket welds.

- The industry LER review identified only one gross failure of a socket weld joint where the pipe separated from the fitting. This was in a non-Class 1 line in a condensate system. The other socket weld defects reported in the LERs were identified through the code-required surface examination or through visual detection of inservice leakage.

- The predominant causes for Class 1 small-bore socket weld defects that were identified in the LER review were indicated to be construction defects, fatigue, or an external stressor; none were concluded as being the result of aging by a mechanism such as SCC originating in a reactor coolant environment.

The fact that Class 1 small-bore socket welds have occasionally experienced leaks does not, in itself, suggest that the license renewal process would require additional aging management activities beyond those already in place under 10 CFR 50. As discussed in NUREG-1800, Section A.1.1, "The license renewal process is not intended to demonstrate absolute assurance that structures and components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions are maintained consistent with the CLB during the period of extended operation. "

The conclusion drawn from this OE review is that neither the industry LERs nor the DAEC-specific OE indicates that failure of Class 1 small-bore socket welds is a safety-significant issue related to aging. The review showed that there have been a relatively small number of Class 1 socket weld leaks in the history of the nuclear industry, but the predominant causes of the leaks were not related to aging. Only one gross failure of a small-bore socket weld was identified, and that failure was in a non-Class 1 line that would not be subject to the ASME Code Class 1 Small-bore Piping Inspection Program.

Both the DAEC-specific experience and the industry experience with Class 1 small-bore socket welds support the position that the very small potential for a small-bore socket weld failure does not represent a significant safety issue for DAEC; and the surface examinations and visual inspections currently being performed have been sufficient to identify and correct weld degradation prior to gross failure. The OE confirms that the existing activities for managing Class 1 small-bore socket welds provide reasonable assurance that the socket welds will continue to support the intended functions of the systems in which they are installed during the period of extended operation.

<u>DAEC LRA Describes Aging Management of Class 1 Small-bore Socket Welds</u>

As discussed in LRA Table 3.1-1, Item 3.1.1-48 (as modified in letter NG-09-0764 dated October 13, 2009), cracking in stainless steel Class 1 piping, fittings, and branch connections <NPS 4" exposed to reactor coolant are managed by the Water Chemistry Program; the ASME Section XI Inservice Inspection, Subsection IWB, IWC and IWD Program; and the ASME Code Class 1 Small-bore Piping Inspection Program.

The Water Chemistry Program monitors and controls reactor water chemistry parameters to minimize the potential for cracking and loss of material due to corrosion. This is a preventive program for RCS components, including small-bore piping.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program, implements the requirements of 10 CFR 50.55a and the NRC-approved Risk-Informed Inservice Inspection Program for Class 1 small-bore socket welds by performing surface examinations of small-bore piping component welds and visual inspections of all Class 1 components, including socket welds, during the RCS pressure test performed each refuel outage. Any weld defects identified during the surface or visual examinations would be repaired prior to reactor startup from that outage. It should be noted that this program also performs volumetric examinations of butt welds in Class 1 small-bore piping. Since the Class 1 small-bore piping butt welds are of similar materials and are exposed to a similar environment as the Class 1 small-bore socket welds, the condition of the butt welds determined by the volumetric examinations would be considered to provide meaningful information on the internal condition of the socket welds. The DAEC examinations of small-bore butt welds using volumetric techniques have found no indication of significant degradation.

As discussed in LRA Section B.3.40.1 (added by letter NG-09-0764 dated October 13, 2009), the ASME Code Class 1 Small-bore Piping Inspection Program will supplement the requirements of the ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program to require additional volumetric examinations of Class 1 small-bore socket welds <NPS 4" if a qualified technique is developed. The UT examinations of small-bore butt welds and the surface and visual examinations of small-bore socket welds, performed under this program, are the same examinations that are required using NRC-approved Risk-Informed Inservice Inspection methodology in the ASME Section XI Inservice Inspection, Subsection IWB, IWC and IWD Program.

In addition to these programs, the condition of Class 1 small-bore socket welds (and other reactor coolant system components) is monitored through Reactor Coolant System leak rate monitoring. This monitoring is performed in accordance with DAEC Technical Specifications 3.4.4, RCS Operational Leakage, and 3.4.5, RCS Leakage Detection instrumentation. This monitoring would provide an early indicator during plant operation if a leak were to develop in the Reactor Coolant System inside containment, including in a Class 1 small-bore socket weld.

<u>Reliable Volumetric Examination Method is Unavailable for Small-bore Socket Welds</u>

At present, there is no consistently reliable or qualified method available to detect and characterize potential flaws in small-bore socket welds. Research sponsored by the industry has been underway for some time, but, to date, has been unable to qualify NDE methodology that is capable of consistently identifying and characterizing flaws in small-bore piping welds. NUREG-1801 XI.M35, One-Time inspection of ASME Code Class 1 Small Bore Piping, indicates the intent that volumetric inspection techniques with demonstrated capability and a proven industry record be used to detect cracking in small-bore piping weld and base material. It also acknowledges, however, that the specific technique to be applied to Class 1 small-bore piping needs to be qualified before the examination.

As discussed in EPRI TR-1015155, "Nondestructive Evaluation: Volumetric Examination of Small bore Piping Welds - Phase 1," November 2007, volumetric examinations of socket welds in small-bore piping are limited by the geometry of the joint configuration and weld, and the small sizes of the lines in question. Attempts have been made to detect cracking in small-bore socket welds using both radiography and UT, but the results have been mixed.

In short, DAEC is unaware of any qualified examination technique, or a non-qualified technique that provides reliable results, to detect and characterize flaws in small-bore socket welds. The unqualified UT techniques that have been attempted at a small number of plants do not provide assurance that all defects will be identified, and have the potential to result in unnecessary maintenance when conservative responses need to be taken for UT indications that can not be interpreted.

<u>Comparison of DAEC ASME Code Class 1 Small-bore Piping Inspection Program with NUREG-1801 XI.M35</u>

As discussed in LRA Section B.3.40 (added by letter NG-09-0764 dated October 13, 2009), the DAEC ASME Code Class 1 Small-bore Piping Inspection Program is a plant-specific program. As such, consistency with NUREG-1801 is not required, nor has an assertion of consistency been made. However, it is instructive to compare the DAEC plant-specific program with the Monitoring and Trending element of the NUREG-1801 XI.M35 description of an ASME Code Class 1 Small-bore Piping Inspection Program.

NUREG-1801 XI.M35, in the element Monitoring and Trending, indicates that a one-time volumetric inspection is an acceptable method for confirming that cracking of ASME Code Class 1 small-bore piping, as a result of stress corrosion or thermal and mechanical loading, is not occurring in plants that have not experienced cracking due to these aging effects. Since DAEC has OE that showed some degradation in a small number of small-bore lines, DAEC has provided a plant-specific program that provides for periodic volumetric inspections in lieu of a one-time inspection. This XI.M35 element goes on to define its expectations for a plant-specific program that is created because of the one-time inspection results by stating:

> However, evaluation of the inspection results may indicate the need for additional examinations, i.e., a plant-specific AMP, consistent with ASME Section XI, Subsection IWB. This inspection should be performed at a sufficient number of

locations to assure an adequate sample. This number, or sample size, will be based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 smallbore piping locations.

The DAEC plant-specific program is consistent with ASME Section XI, Subsection IWB. As discussed above, the program has considered susceptibility and inspectability as well as other factors in determining the sample size for the small-bore weld inspections to be performed under the program. Small-bore socket welds are not considered to be inspectable by any means other than surface or visual examination since no qualified volumetric examination technique is available. Accordingly, all Class 1 small-bore piping (butt and socket welded) is subject to code-required surface and visual examinations, and the Class 1 small-bore piping weld sample that is subject to volumetric examination includes only butt welds. Therefore, the DAEC ASME Code Class 1 Small-bore Piping Inspection Program is consistent with the Monitoring and Trending element of NUREG-1801 XI.M35.

<u>Conclusion</u>

In view of the discussions above, the very small potential for complete failure of a Class 1 small-bore socket weld is not considered to have a significant impact on nuclear safety at DAEC. The Water Chemistry Program will continue to provide a preventive measure by maintaining an environment that minimizes the potential for corrosion or cracking in the RCS. The condition of small-bore butt welds that are volumetrically examined under the ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program will continue to serve as an indicator to confirm that significant SCC is not occurring in RCS small-bore piping, including socket welds. Even if degradation of Class 1 small-bore socket welds were to develop, the detection of surface flaws and through-wall leakage through existing code-required surface and visual examinations, respectively, provide an early warning to take corrective action, which minimizes the potential for complete failure of socket welded joints. These activities in themselves provide reasonable assurance that Class 1 small-bore socket welds will perform such that the intended functions are maintained consistent with the CLB during the period of extended operation.

Industry research continues into the development of NDE techniques that can reliably detect and characterize flaws in socket welds, and NEI and the nuclear industry are continuing to discuss this subject with the NRC. In the interim, until a qualified NDE technique is available for examination of Class 1 small-bore socket welds, DAEC will continue to comply with the ASME Code and the RI-ISI examination requirements for small-bore piping as defined in the ASME Code Class 1 Small-bore Piping Inspection Program. In addition, DAEC will monitor the industry research into small-bore piping examination methods and the interactions on the subject with the NRC. As discussed in LRA Section B.3.40.1, if a reliable volumetric examination technique is qualified for use on small-bore socket welds in the future, DAEC will perform volumetric examination of a sample of Class 1 small-bore socket welds each inspection interval in conjunction with the ASME Section XI Inservice inspection, Subsections IWB, IWC and IWD Program.

Response to RAI B.3.10-7 Follow-up Question Regarding CRD Safe End

During a conference call on February 2, 2010, it was noted that letter NG-10-0009, dated January 14, 2010, updated the response to RAI 3.10-7 to change the CRD Safe End line item in LRA Table 3.1.2-1 to list the BWR Control Rod Drive Return Line Nozzle Program as the aging management program for cracking, but the line item did not also list the Water Chemistry Program. DAEC acknowledged that the Water Chemistry Program was also applicable. Accordingly, this line item is further revised as follows:

In LRA Table 3.1.2-1, Summary of Aging Management Review Results - Nuclear Boiler, on page 3.1-55, the line item for Safe End - Control Rod Drive with an Aging Effect Requiring Management of Cracking is revised to appear as follows:

Component Type	Intended Function	Material	Environment	Aging Effects Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Line Item	Table 3.x-1 Item	Notes
Safe end - control rod drive	Pressure boundary	Stainless steel	Reactor Coolant (internal)	Cracking	BWR Control Rod Drive Return Line Nozzle Program	IV.A1-1 (R-68)	3.1.1-41	E
					Water Chemistry Program	IV.A1-1 (R-68)	3.1.1-41	A

RAI 3.1.2.2.7-1 Follow-up Question Regarding Main Steam Line Flow Restrictors

In an email dated January 15, 2010, the staff raised the following question about the response to RAI 3.1.2.2.7-1 in letter NG-09-0825 dated December 14, 2009:

> RAI 3.1.2.2.7-1 requested DAEC explain why SCC of CASS orifices would be managed with the Water Chemistry and OTI Programs and not the BWR Stress Corrosion Cracking and Water Chemistry Programs. The response to the RAI was to remove consideration of SCC for these CASS components because they are not subject to pressure stresses. However, weld residual stresses could cause SCC; therefore this aging effect needs to be managed. DAEC's response to RAI 3.4.2.4-1 was to eliminate loss of fracture toughness as an aging effect for CASS flow orifices because the material is centrifugally cast CF8 material with ferrite content less than 20%. However, this determination should made through a CASS Program and not in the LRA. It is noted that DAEC does not have an XI.M12 CASS Program.

DAEC Response to RAI 3.1.2.2.7-1 Follow-up Question

In LRA Table 3.4.2-4, DAEC provided the aging management review results for the main steam line flow restrictors. These devices are cast of SA-451 austenitic stainless steel and then inserted inside larger forged carbon steel piping and welded into place. The flow restrictors do not form the pressure boundary of the main steam piping. For managing the aging effect of cracking, the Water Chemistry and One-Time Inspection Programs were assigned. These line items for the flow restrictors were aligned with Table 3.1-1 item 3.1.1-41 for stainless steel in reactor coolant, which specifies use of the BWR Stress Corrosion Cracking and Water Chemistry Programs.

In RAI 3.1.2.2.7-1, the NRC questioned the assignment of the Water Chemistry and One-Time Inspection Programs instead of the programs cited in item 3.1.1-41, and requested justification why cracking was considered unlikely or occurring very slowly. DAEC responded to this RAI in letter NG-09-0825 on December 14, 2009. Upon further review, DAEC hereby withdraws the response provided for RAI 3.1.2.2.7-1 in DAEC Letter NG-09-0825, and provides the following information.

Management of Cracking in Flow Restrictors

As discussed above, the main steam line flow restrictors do not form the pressure boundary of the main steam piping. These devices are cast of SA-451 austenitic stainless steel and then inserted inside larger forged carbon steel piping and welded into place. DAEC has concluded that the GALL Volume 2 and Table 3.X-1 line items that were originally cited in the flow restrictor line items in LRA Table 3.4.2-4 were not appropriate for the configuration of flow restrictors welded inside forged carbon steel piping. While GALL Volume 2 does not contain aging management evaluations that align well with this configuration, the DAEC selections of Volume 2 Item IV.C1-9 and Table 3.1.1 item 3.1.1-41, which are applicable to a reactor coolant pressure boundary, introduced confusion. It has been concluded that the main steam system flow restrictors should originally have been aligned with appropriate items from GALL Chapter VIII instead of Chapter IV. In addition, DAEC acknowledges the staff's concern about residual stresses that could be present from welding the flow elements in place.

Accordingly, the following LRA changes are made to better represent the main steam line flow restrictor configuration at DAEC:

In LRA Table 3.1-1, Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 Reactor Coolant System, line item 3.1.1-57 on page 3.1-22, the Discussion entry is revised in its entirety to read, "Not applicable to the Reactor Coolant System Section."

In LRA section 3.4.1.4, Main Steam Isolation and Automatic Depressurization System, on page 3.4-5, under Aging Effects Requiring Management, the bullet "Loss of fracture toughness" is deleted.

In LRA Table 3.4.2-4, Summary of Aging Management Review Results - Main Steam Isolation and Automatic Depressurization System, on page 3.4-51, the line item for "Flow element Class 1" with Aging Effect Requiring Management of "Loss of fracture toughness" is deleted. In the Flow element Class 1 line item with Aging Effect Requiring Management of "Loss of material", the Notes entry is changed from A to C.

In LRA Table 3.4.2-4, Summary of Aging Management Review Results Main Steam Isolation and Automatic Depressurization System, on page 3.4-51, the first two line items for Flow element Class 1, with the aging effect of cracking, are replaced with the following single line item:

Component Type	Intended Function	Material	Environment	Aging Effects Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Line Item	Table 3.x-1 Item	Notes
Flow element Class 1	Throttle	Cast Austenitic Stainless Steel	Steam (External)	Cracking	Water Chemistry Program One-Time Inspection Program	VIII.B2-1 (SP-45)	3.4.1-13	A

While it is acknowledged that NUREG-1801 Volume 2 Line Item VIII.B2-1 lists stainless steel as the material and does not explicitly list CASS, the behavior of stainless steel in the steam environment is considered to appropriately represent the behavior of CASS in the steam environment of this configuration, and the Water Chemistry and One-Time Inspection Programs are the appropriate programs to manage this configuration.

<u>Fracture Toughness Issues</u>

In the follow-up question, the staff suggests that an XI.M12 CASS Program had to be established to make a determination whether particular CASS material was susceptible to a loss of fracture toughness. The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program, GALL XI.M12 states, "The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F)." Since the main steam system flow restrictors are neither primary pressure boundary nor reactor vessel internal components, nor are they constructed of SA-351 material, GALL XI.M12 is not applicable. The technical conclusion discussed in this response does not depend on the existence of an administrative program comparable to GALL XI.M12.

Clarification of LRA Section B.3.19.7 Regarding Electrical Connections Program

The Staff questioned whether the "scope of program" and "parameters monitored or inspected" elements of LRA Section B.3.19, Electrical Connections Program, were consistent in their treatment of high voltage switchyard connections. To resolve the question, the following LRA change is made.

In LRA section B.3.19, Electrical Connections Program, Subsection B.3.19.7, Parameters Monitored or Inspected, on page B-40, The sentence, "The following factors shall be considered for sampling: voltage level (medium and low voltage) ...." is revised to read as follows:

> The following factors shall be considered for sampling: voltage level (high, medium and low voltage), circuit loading (high load), and location (high temperature, high humidity, vibration, etc.).

Commitment for Core Rim Plate Hold-down Bolt Inspections

In response to discussions with the Staff regarding inspection methodology for core plate hold down bolts, License Renewal Commitment 47 (added in response to RAI 4.2.7-1 letter NG-09-0663 dated October 23, 2009) is withdrawn, and Commitment 37 is revised to read as follows:

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
37.	Reactor Internals	DAEC will ensure that aging of core plate hold down bolts is appropriately addressed by completing one of the following actions: • Install core plate wedges to eliminate the function of core plate hold down bolts. • Perform analysis of the core plate rim hold down bolts that demonstrates adequacy to perform their intended function including loss of pre-load in the period of extended operation including the effects of projected neutron fluence. Inspection of core plate hold down bolts will be performed in accordance with BWRVIP-25, or a deviation disposition will be developed/submitted in accordance with BWRVIP-94.	18.1.14 18.3.1.7	Prior to entering the period of extended operation

Commitment for Suppression Pool Recoating

In various telephone conference calls, DAEC has discussed its current plans to recoat interior surfaces of the suppression pool during a future refuel outage as determined to be necessary by the ASME Section XI Inservice Inspection, Subsection IWE Program. To follow up those discussions, the following new commitment is provided:

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
50	ASME Section XI, Inservice Inspection, Subsection IWE Program	Perform recoating of suppression pool interior surfaces below the water line.	18.1.4	Complete recoating prior to startup from the first refuel outage during the period of extended operation

Response to RAI BF76-3 Follow-up Question Regarding Brass Components

In an email dated February 17, 2010, the Staff requested clarification of the DAEC response to RAI BF76-3 (letter NG-10-0043 dated February 2, 2010) to address an additional line item in LRA Table 3.2.2-4 on page 3.2-54.

The component type in question, "Valve, damper," with material "Brass" and environment "Lube oil (internal)" shown on page 3.2-54, includes two brass valves containing <15% zinc. This line item currently cites Note 232. This line item should have cited Note 225 which states, "Crevice and pitting corrosion are not applicable aging mechanisms for copper alloy components with less than 15% zinc and aluminum bronze components with less than 8% aluminum in fuel oil and lube oil environments at Duane Arnold."

Accordingly, in LRA Table 3.2.2-4, Summary of aging Management Review Results, Reactor Core Isolation Cooling System, on page 3.2-54, for the line item "Valve, damper, of material Brass in a Lube Oil environment (internal), the Notes entries are changed from "232, I" to "225, I".

Response to RAI BF76-5 Follow-up Question Regarding PVC Components

In an email dated February 17, 2010, the Staff requested clarification of the potential radiation dose accumulation in components of the Reactor Building and Radwaste Building Sampling System.

The PVC/plastic components in question (pipe, pipe fittings, hoses, tubes and rupture disk) are in the North Open Area of elevation 786 in the Reactor Building. The normal 60 year dose for these components is 5.3E2 rads, well below 10E6 rads.

Clarification of Aging Management Review Results for Aluminum Spent Fuel Racks

In an email dated February 17, 2010, the Staff requested clarification of the aging management review results for the aluminum spent fuel racks in treated water.

A review of the Aging Management Reviews performed for aluminum components in the spent fuel pool identified that the PaR Spent Fuel Storage Racks had been identified as having aluminum exposed to treated water. This component was evaluated as having no aging effects requiring management and was not listed in the LRA. A review of GALL Volume 2 Section VII.A4-5, however, revealed that, while the PaR spent Fuel Storage Racks are not considered piping, they do contain material in environments that are comparable to this Gall line item. It is concluded that these aluminum components should have been listed in LRA Table 3.5.2-10 as having aging effects requiring management for loss of material due to pitting and crevice corrosion.

Accordingly, LRA changes are made to incorporate the aging management review results for the aluminum PaR Spent Fuel Storage Racks, as follows:

In LRA Section 3.3.2.2.10.2 on page 3.3-41, the third paragraph which reads, "The auxiliary systems at DAEC have no aluminum components exposed to a treated water environment" is replaced in its entirety with the following:

> At Duane Arnold, the aluminum PaR spent fuel pool storage racks exposed to treated water are managed for loss of material due to crevice and pitting corrosion by the Water Chemistry Program. The effectiveness of the Water chemistry Program will be confirmed by the One-Time Inspection Program through an inspection of a representative sample of components crediting this program, including susceptible locations such as areas of stagnant flow.

In LRA Table 3.5.2-10, Summary of Aging Management Review Results - Supports, a new line item is added to appear as follows:

Component Type	Intended Function	Material	Environment	Aging Effects Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Line Item	Table 3.x-1 Item	Notes
PaR spent fuel racks	Structure	Aluminum	Treated water (external)	Loss of material	Water Chemistry Program One-Time Inspection Program	VII.A4-5 (AP-38)	3.3.1-24	C

The Boral component that is contained within the PaR Spent Fuel Racks has been addressed separately in the LRA. Aging management of Boral has been discussed in the response to RAI 3.3.2.2.6 previously submitted in letter NG-09-0765 dated October 23, 2009.

RAI B.3.18-1 Follow-up Question Regarding Electrical Cables and Connections Used in Instrumentation Circuits Program

In emails dated January 5, 2010, the NRC transmitted a follow-up question related to the DAEC response to RAI B.3.18-1. The question was modified in an email dated March 1, 2010. The follow-up question and response are provided below.

<u>Background</u>

GALL AMP XI.E2, under Scope of Program, states that this program applies to electrical cable and connections used in circuits with sensitive, high voltage, low-level signals such as radiation monitoring and nuclear instrumentation that are subject to an aging management review. In the applicant's basis document LRAP-E002, under Scope of Program, its states that the cables in scope are in the nuclear instrumentation system and there are no radiation monitoring system cables in the scope of this aging management program. The radiation monitoring system cables are in the scope of license renewal because it performs an intended function. These cables are used in sensitive, high voltage, low level signal circuits. Exposure of these electrical cables to adverse localized environments caused by heat, radiation, or moisture can result in reduced insulation resistance (IR). Reduced IR can cause an increase in leakage current between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, high voltage, low-level signals such as high-range radiation monitoring system cables.

In response to the staff's concern, in a letter dated October 13, 2009, the applicant stated that instrumentation cables for radiation monitoring systems are not in the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program because either they are included in the 10CFR50.49 Environmental Qualification Program, or they are not located in adverse localized environments. The applicant also stated that Duane Arnold defines adverse localized environments for instrumentation cable as areas with radiation dose greater than 3×10^7 rads and/or temperature greater than 60°C (140°F). The radiation monitoring system instrumentation cables that are not included in the Environmental Qualification Program are in areas where the maximum design dose is 5.3×10^4 rads and maximum design temperature is 40°C (104°F). The radiation monitoring system instrumentation cables are designed and qualified for this environment.

Issue

The staff questioned the applicant's response. The high-range radiation monitors monitor radiation levels of specified liquid systems, gaseous system, and general areas throughout the plant; assists in controlling the release of radioactive materials, and provides personnel safety by warning of abnormal radiation levels. These monitors are typically installed in an adverse localized environment due to high heat and radiation. GALL AMP XI.E2, under Scope of Program, states that this program applies to electrical cable and connections used in circuits with sensitive, high voltage, low-level signals such as high-range radiation monitoring. The scope of the applicant AMP B.3.18 is not consistent with those in GALL XI.E2.

Question

Provide additional technical justification of why radiation monitoring cables are not required to be in scope of AMP B.3.18.

DAEC Response to RAI B.3.18-1 Follow-up Question

The issue is not a question of consistency with GALL XI.E2 but a question concerning the scoping and screening process. The radiation monitors that the Staff is questioning are not within the scope of license renewal and, therefore, are not within the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program. Duane Arnold does not consider the Area Radiation Monitor System, Environmental

Radiation Monitor System, Liquid Process Radiation Monitor System, Low Level Radwaste Processing and Storage Facility Kaman Radiation Monitor System, Stack Gas Kaman Radiation Monitor System, Offgas Radiation Monitor System, Reactor Building Kaman Radiation Monitor System, and Turbine Building Kaman Radiation Monitor System to meet the scoping criteria listed in 10 CFR 54.4(a)(1) – (3).

- These systems are not relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to maintain the integrity of the reactor coolant pressure boundary; aid in the capability to shut down the reactor and maintain it in a safe shutdown condition; or aid in the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10CFR50 § 50.34(a)(1), § 50.67(b)(2), or § 100.11, as applicable.

- No electrical failure in these systems could prevent satisfactory accomplishment of safety related functions.

- None of these systems are relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout.

Table 1 below lists the functions for these systems as described in LRA Table 2.2-2 as revised by NextEra Energy Letter NG-09-0823 dated December 2, 2009.

Table 1

System	Function
Area Radiation Monitor System	Notifies personnel of airborne radiation hazard.
Environmental Radiation Monitor System	Provides integrated measurements of direct radiation exposure at the boundary of the unrestricted area to confirm that the operation of the plant is in accordance with the requirements of 10 CFR 20.
Liquid Process Radiation Monitor System	Provides a clear indication whenever the radioactivity level in the stream reaches or exceeds a pre-established limit above the normal radiation level.
Low Level Radwaste Processing and Storage Facility Kaman Radiation Monitor System	Provides a clear indication whenever abnormal amounts of radioactivity exist in the LLRWSF facility and prompts operator actions.
Stack Gas Kaman Radiation Monitor System	Provides a clear indication whenever limits on the release of radioactive material to the environs are reached or exceeded.
Offgas Radiation Monitor System	Provides an alarm to operations personnel when the radioactivity level of the air ejector exceeds preset limits.
Reactor Building Kaman Radiation Monitor System	Provides a clear indication whenever abnormal amounts of radioactivity exist in the reactor building main exhaust stacks and prompts operator actions.
Turbine Building Kaman Radiation Monitor System	Provides a clear indication whenever abnormal amounts of radioactivity exist in the turbine building ventilation roof vents and prompts operator actions.

Radiation monitoring systems originally in the scope of license renewal are the Drywell Radiation Monitor, Main Steam Supply Line Radiation Monitor, Control Building Ventilation Radiation Monitor and the Reactor Building Ventilation Radiation Monitor.

- The Drywell Radiation Monitors and associated cables are in the Environmental Qualification (EQ) Program. This includes the entire length of cables from the detector to the Control Room panel. Therefore, these cables are not in the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program.

- The safety function of the Main Steam Supply Line Radiation Monitor was deleted per NRC-approved License Amendment 261 (ML063100647) and the system has been removed from the scope of license renewal. Duane Arnold UFSAR Section 11.5.1 contains details on why the safety function was removed. Therefore, these cables are not in the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program.

- The Control Building Ventilation Radiation Monitor, the Reactor Building Ventilation Radiation Monitor and associated cables are in mild environments (temperatures between 68° and 104°, 60 year normal total dose ≤ 5.3E4 Rads and between 20% and 90% humidity). These cables are not exposed (over any part of the length) to adverse localized environments (heat, radiation or moisture). Also these cables were installed new in 2005 when the analog system was replaced with a digital system. Therefore, these cables are not in the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program.

When Duane Arnold defined the scope of the Electrical Cables and Connections Used in Instrumentation Circuits Program, the engineers obtained a list of the coaxial and triaxial cables from the controlled Cable and Raceway Database. This database lists all cables that are in raceways (conduits and cable trays). Cables in systems that are not in the scope of license renewal were identified as not having a license renewal function. Cables in the EQ program were identified as having a license renewal function but aging management was covered by the EQ program. The cables that were left were included in the Electrical Cables and Connections Used in Instrumentation Circuits Program. Therefore, the only cables in the program are for the neutron monitoring systems (Intermediate Power Range Monitor and the Power Range Monitor systems).

Draft RAI B.3.25-X Regarding Diesel Driven Air Start Compressor Fuel Oil Tank

In an email dated February 23, 2010, the NRC transmitted draft RAI B.3.25-X. The draft RAI and response are provided below.

Background

The "detection of aging effects" program element of GALL AMP XI.M30 recommends that internal surfaces of tanks that are drained for cleaning are visually inspected to detect potential degradation, and an ultrasonic thickness measurement of the tank bottom surface be conducted to ensure that significant degradation is not occurring in locations where contaminants may accumulate, such as a tank bottom. However, during its audit, the staff found that the applicant's Fuel Oil Chemistry Program states that there are no equipment specific procedures required to validate the quality of the fuel oil in the diesel driven air start air compressor fuel oil tanks. In addition, it was also stated that these tanks are not subjected to periodic cleaning and visual or UT inspection, because the tanks are small, have high fuel turnover, and general inspections indicate no degradation, and as such this is not considered an exception to the GALL. By letter dated September 14, 2009, the staff issued RAI B.3.25-3 requesting that the applicant provide further justification for not performing any preventive/mitigative activities and interior visual or one-time UT examinations to confirm that degradation has not occurred in the diesel driven air start air compressor fuel oil tanks.

In its response dated October 13, 2009, the applicant modified Enhancement 3 as follows:

"Enhance the Program to assure that the frequencies for the periodic draining, cleaning and visual inspection of the diesel fuel oil day tanks, diesel fire pump day tank and diesel driven air start air compressor fuel oil tanks are on a schedule of every ten years."

Additionally, the applicant revised Commitment 16 to include:
"Enhance procedures to perform periodic (10 year) draining, cleaning and visual inspection of the diesel fuel oil day tanks, diesel fire pump day tank, and diesel driven air start air compressor fuel oil tanks."

Issue

In the LRA, the applicant did not provide a sufficient justification to not conduct UT thickness measurements on the diesel driven air start air compressor fuel oil tanks. Despite the tanks being small and having high fuel turnover, sediments and water can collect at the bottom of the tank and lead to corrosion. Additionally, the applicant stated that a general inspection indicated no degradation; however no thickness data was provided to demonstrate that corrosion that could challenge the component's function is not occurring.

Request

Provide sufficient justification for not performing an UT examination, or revise the enhancement so that the activities are consistent with the recommendations from GALL AMP XI.M30 for the diesel driven air start air compressor fuel oil tanks.

DAEC Response to Draft RAI B.3.25-X

The diesel driven air start air compressor fuel oil tanks are 3-3/4 gallon tanks made of thin gage sheet metal folded and welded, and mounted on the skid of the compressor. The tanks (1T477 and 1T478) are painted on the outside and the entire tank can be visually inspected from the outside as well as from the inside. A UT is not meaningful on these tanks due to the thickness of the sheet metal. Therefore, an ultrasonic inspection will not be performed. The diesel driven air start air compressor fuel oil tanks are in scope for (a)(2) leakage boundary only. A 10 year visual inspection of the outside coatings (paint) and accessible inside surfaces of these fuel oil tanks is adequate for detecting any aging effects. Accordingly, the LRA description of the Fuel Oil Chemistry Program is being revised to incorporate an additional exception and state that a 10 year visual inspection will be made to the diesel driven air start air compressor fuel oil tank coatings in lieu of an ultrasonic thickness measurement. The LRA changes are as follows:

LRA Section B.3.25.2, NUREG-1801 Consistency, on page B-52, is revised to read as follows:

> The program is consistent with five of the elements of NUREG XI.M30. Exceptions are taken to "Scope of Program," "Preventive Actions," "Parameters Monitored/Inspected," "Detection of Aging effects," and "Acceptance Criteria." These exceptions are listed below.

In LRA Section B.3.25.3, Exceptions to NUREG-1801, on page B-52, a new bullet is added to read as follows:

- Detection of Aging Effects

  The DAEC diesel driven air start air compressors fuel oil tanks (1T477 and 1T478) are constructed of thin gage sheet metal. As a result, an ultrasonic thickness measurement of the tank bottom surface is not meaningful. Visual examination of the tank exterior coatings and accessible inside surfaces will ensure aging effects are adequately managed.

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
1.	Buried Piping and Tanks Inspection Program	Implement Buried Piping and Tank Program [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.7	Prior to the period of extended operation
2.	BWR Vessel Internals Program	Perform an EVT-1 inspection of 5% of the top guide locations	18.1.14	Within six years of entering the period of extended operation
3.	BWR Vessel Internals Program	Perform an EVT-1 inspection of an additional 5% of the top guide locations	18.1.14	Within 12 years of entering the period of extended operation
4.	Electrical Cables and Connections Program	Implement an Electrical Cables and Connections Program and complete the first inspection prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.17	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
5.	Electrical Cables and Connections Used in Instrumentation Circuits Program	Implement an Electrical Cables and Connections Used in Instrumentation Circuits Program and complete the first inspection prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.18	Prior to the period of extended operation
6.	Electrical Connections Program	Implement an Electrical Connections Program and complete the one time inspection prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.19	Prior to the period of extended operation
7.	Electrical Penetration Assemblies Program	Implement an Electrical Penetration Assemblies Program. [Revised in DAEC letter NG-09-0764 in response to new Program Commitments RAI]	18.1.20	Prior to the period of extended operation
8.	External Surfaces Monitoring Program	Revise the inspection program to address inspector qualifications, types of components, degradation mechanisms, aging effects, acceptance criteria, inspection frequency, and periodic reviews to determine program effectiveness. The program will also specifically address inaccessible areas and include inspections of opportunity for possible corrosion under insulation. [Revised in DAEC letter NG-09-0764 in response to RAI B.3.21-2]	18.1.21	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
9.	Fire Protection Program	The DAEC Fire Barrier Penetration Seal Inspection surveillance procedure will be enhanced to include criteria for visual inspections of fire barrier wall, ceiling and floors to examine for any sign of degradation such as cracking, spalling and loss of material caused by freeze-thaw, chemical attack and reaction with aggregates by fire protection qualified inspectors. [Revised in DAEC letter NG-09-0764 in response to RAI B.3.22-1]	18.1.22	Prior to the period of extended operation
10.	Fire Protection Program	Enhance procedures to inspect the entire diesel driven fire pump fuel supply line for age related degradation.	18.1.22	Prior to the period of extended operation
11.	Fire Water System Program	Implement maintenance activities to perform volumetric examinations for pipe wall thinning of fire protection piping periodically during the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.23	Prior to the period of extended operation
12.	Fire Water System Program	Enhance procedures to include NFPA 25 criteria for sprinklers regarding replacing or testing	18.1.23	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
13.	Fire Water System Program	Enhance procedures to perform visual inspection of fire hydrants annually	18.1.23	Prior to the period of extended operation
14.	Fuel Oil Chemistry Program	Revise the program to require particulate testing of fuel oil samples from the diesel fire pump day tank	18.1.25	Prior to the period of extended operation
15.	Fuel Oil Chemistry Program	Enhance procedures to require sampling and testing of new fuel oil delivered to the diesel fire pump day tank; and to require that purchase orders and sampling procedures for diesel fuel delivered to and stored in the diesel fire pump day tank prohibit the delivery and use of biodiesel fuel. [Revised in letter NG-09-0764 in response to RAI B.3.25-1]	18.1.25	Prior to the period of extended operation
16.	Fuel Oil Chemistry Program	Enhance procedures to perform periodic (10 year) draining, cleaning and visual inspection of the diesel fuel oil day tanks, diesel fire pump day tank, and diesel driven air start air compressor fuel oil tanks. [Revised in letter NG-09-0764 in response to RAI B.3.25-4]	18.1.25	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
17.	Fuel Oil Chemistry Program	Implement procedures to require bottom thickness testing of the Standby Diesel Generator Day Tanks and the Diesel Fire Pump Day Tank. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.25	Prior to the period of extended operation
18.	Fuse Holders Program	Implement a Fuse Holders Program and complete the first test prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to RAI B.3.26-1 and New Program Commitments RAI]	18.1.26	Prior to the period of extended operation
19.	Inaccessible Medium Voltage Cable Program	Implement an Inaccessible Medium Voltage Cable Program and complete the first inspection or test prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.27	Prior to the period of extended operation
20.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program	Implement an Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.28	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
21.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program	Enhance procedures to monitor for corrosion and wear of the supporting steel and rails	18.1.29	Prior to the period of extended operation
22.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program	Enhance procedures to record usage of the reactor building and turbine building cranes	18.1.29	Prior to the period of extended operation
23.	Lubricating Oil Analysis Program	Enhance procedures to include diesel fire pump	18.1.30	Prior to the period of extended operation
24.	Metal Enclosed Bus Program	Implement a Metal Enclosed Bus Program and complete the first inspection prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.31	Prior to the extended operation
25.	One-Time Inspection Program	Implement a One-Time Inspection Program and complete the one-time inspections prior to the period of extended operation. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.32	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
26.	Reactor Vessel Surveillance Program	Implement a procedure to evaluate the BWRVIP ISP data as it becomes available. [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.35	Prior to the period of extended operation
27.	Reactor Vessel Surveillance Program BWRVIP-74-A BWR PRV Inspection and Flaw Evaluation Guidelines for License Renewal	Revise the Reactor Vessel Surveillance Program to implement the recommendations of BWRVIP-116 BWR Vessel and Internals Project Integrated Surveillance Program Implementation for License Renewal.	18.1.35	Prior to the period of extended operation
28.	Reactor Vessel Surveillance Program	Implement BWRVIP-116 with the conditions documented in Sections 3 and 4 of the NRC Staff's SE dated March 1, 2006 for BWRVIP-116	18.1.35	Prior to the period of extended operation
29.	Selective Leaching of Materials Program	Implement and complete a program to include one-time visual inspection and hardness measurement of selected components susceptible to selective leaching [Revised in DAEC letter NG-09-0764 in response to New Program Commitments RAI]	18.1.36	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
30.	Structures Monitoring Program	Enhance procedures to include structures and structural components not currently in Maintenance Rule Program	18.1.37	Prior to the period of extended operation
31.	Structures Monitoring Program	Enhance procedures to include periodic sampling of groundwater for pH, chloride and sulfate concentration on a 5 year periodicity. [Revised in letter NG-10-0043 in response to Follow-up RAI B.3.37-2]	18.1.37	Prior to the period of extended operation
32.	Structures Monitoring Program	Enhance procedures to include a elastomer inspection to prevent leakage through containment penetration.	18.1.37	Prior to the period of extended operation
33.	Structures Monitoring Program	Enhance procedures to include a requirement to contact the proper personnel to allow opportunistic inspection of the buried concrete foundation.	18.1.37	Prior to the period of extended operation
34.	Structures Monitoring Program	Enhance procedures to include opportunistic inspections of the buried concrete foundation on a 10 year periodicity.	18.1.37	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
35.	Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program	Enhance procedures to incorporate the requirements of NUREG/CR-6260 locations into the implementing procedures.	18.2.2	Prior to the period of extended operation
36.	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Implement a Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. [Revised in DAEC letter NG-09-0764 in response to New Program Commitment RAI]	18.1.38	Prior to the period of extended operation
37.	Reactor Internals	DAEC will ensure that aging of core plate hold down bolts is appropriately addressed by completing one of the following actions: • Install core plate wedges to eliminate the function of core plate hold down bolts. • Perform analysis of the core plate rim hold down bolts that demonstrates adequacy to perform their intended function including loss of pre-load in the period of extended operation including the effects of projected neutron fluence. Inspection of core plate hold down bolts will be performed in accordance with BWRVIP-25, or a deviation disposition will be developed/submitted in accordance with BWRVIP-94. [Revised in DAEC letter NG-10-0091]	18.1.14 18.3.1.7	Prior to entering the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
38.	Reactor Vessel Circumferential Weld TLAA	Submit a relief request to address the frequency requirements of the inservice inspection of the RPV circumferential welds. (BWRVIP-05).	18.3.1.4	Prior to the period of extended operation
39.	Quality Assurance Program (Corrective Action, Confirmation Process, Administrative Controls)	Expand the scope of its 10 CFR Part 50, Appendix B Quality Assurance program to include non-safety-related structures and components subject to an AMR for license renewal.	UFSAR 17.1.2	Prior to the period of extended operation
40.	Operating Experience	Perform an operating experience review of extended power uprate and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation.		Prior to the period of extended operation
41.	Bolting Integrity Program	Revise the implementing procedures for the ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program; ASME Section XI Inservice Inspection, Subsection IWF Program; External Surfaces Monitoring Program; Structural Monitoring Program; and Buried Piping and Tanks Program such that they specifically address the inspection of fasteners (bolting, washers, nuts, etc.) for signs of leakage, corrosion/loss of material, cracking, and loss of preload/loss of prestress, as applicable. [Added in letter NG-09-0764 in response to RAI B.3.6-02]	18.1.6	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
42.	BWR Penetrations Program	The implementing document for the BWR Penetrations Program will be revised to specify that guidance in BWRVIP-14, -59 and -60 will be used, as appropriate, depending on material, in the evaluation of crack growth in stainless steel, nickel alloys and low-alloy steels, respectively, when flaws are identified and evaluation required. [Added in letter NG-09-0764 in response to RAI B.3.10-5] [Revised in letter NG-10-0009]	18.1.10	Prior to the period of extended operation
43.	Fire Protection Program	The DAEC Fire Barrier Penetration Seal Inspection surveillance procedure will be enhanced to ensure a approximately 10% of each type of penetration seal is included in the 35 percent selection of fire penetration seals that are visually inspected at an 18 month interval. [Added in letter NG-09-0764 in response to RAI B.3.22-1]	18.1.22	Prior to the period of extended operation
44.	Fire Protection Program	The DAEC Surveillance Procedure for the CO2 Cardox System Operability Annual Test will be enhanced to include a step to perform an inspection for corrosion and mechanical damage to system components. [Added in letter NG-09-0764 in response to RAI B.3.22-1]	18.1.22	Prior to the period of extended operation
45.	ASME Class 1 Small-bore Piping Inspection Program	Implement an ASME Code Class 1 Small-bore Piping Inspection Program. [Added in letter NG-09-0764 in response to RAI B.3.3-2]	18.1.40	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
46.	BWR Vessel Internals Program	The BWR Vessel Internals Program will incorporate the crack growth rate evaluations specified in the BWRVIP-100-A report. Plant-specific inspection intervals will be developed for DAEC core shroud welds that are exposed to a neutron fluence value equal to or greater than 1 X 10^{21} n/cm^2 (E > 1 MeV), as needed. [Added in letter NG-09-0663 in response to RAI B.3.14-5]	18.1.14	Prior to the period of extended operation
47.		Not Used [Withdrawn in letter NG-10-0091]		
48.	Boral Surveillance Program	Implement a Boral Surveillance Program and complete the first in-situ neutron attenuation test of the PaR spent fuel racks. [Added in letter NG-10-0009]	18.1.41	Prior to the period of extended operation
49.	Fire Protection Program	Enhance procedures to inspect the 1 hour fire rated gypsum board wall that separates the control room computer room area from the front panel area for aging due to cracking. [Added in letter NG-10-0043]	18.1.22	Prior to the period of extended operation

TABLE A-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS[1]

Item No.	System, Component or Program	Commitment[2]	Section	Schedule
50.	ASME Section XI, Inservice Inspection, Subsection IWE Program	Perform recoating of suppression pool interior surfaces below the water line. [Added in letter NG-10-0091]	18.1.4	Complete recoating prior to startup from the first refuel outage during the period of extended operation

[1]Table is updated to reflect DAEC correspondence through 3/9/2010.
[2]In the table, the term "implement" means that the program is described in an approved procedure or other approved formal document; the test, inspection or monitoring procedure has been developed and approved; and the first test, inspection or monitoring activity has been scheduled.