NEXTera ENERGY
DUANE ARNOLD

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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

Response to Request for Additional Information Regarding Time-Limited Aging Analyses and Aging Management Programs of the Duane Arnold Energy Center License Renewal Application

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 (Florida Power & Light Company), "Request for Additional Information for the Review of the Duane Arnold Energy Center License Renewal Application – Time-Limited Aging Analysis Section 4.2, Aging Management Programs B3.12, B3.13, B3.14 and B3.35 (TAC No. MD9769)," dated September 24, 2009 (ML092580547)

By Reference 1, FPL Energy Duane Arnold, LLC submitted an application for a renewed Operating License (LRA) for the Duane Arnold Energy Center. Reference 2 provided Supplement 1 to the application. By Reference 3 the U.S. Nuclear Regulatory Commission (NRC) Staff requested additional information regarding Time-limited Aging Analyses (TLAA) and Aging Management Programs of the LRA.

Enclosure 1 to this letter contains the NextEra Energy Duane Arnold, LLC, (f/k/a FPL Energy Duane Arnold, LLC) responses to the Staff's request for additional information.

Enclosure 2 contains a non-proprietary version of an analysis requested in RAIs 4.2.4-1 and 4.2.5-2.

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In a telephone conference call on October 2, 2009, the NRC requested clarification about the aging management review results for stainless steel bolts in the High Pressure Coolant Injection (HPCI) System. NextEra Energy concluded that a line item for stainless steel bolting should be added to LRA Table 3.2.2-2. The LRA change is provided in Enclosure 3.

This letter contains two new license renewal commitments and one change to an existing license renewal commitment as indicated in the RAI responses. Enclosure 4 provides a revised LRA Appendix A, Section 18.4, Table A-1, Duane Arnold License Renewal 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 October 23, 2009.

Christopher R Costanzo

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

Enclosures:
1. DAEC Response to NRC Requests for Additional Information Regarding Time-Limited Aging Analyses and Aging Management Programs
2. Structural Integrity Associates Calculation Package DAEC-20Q-338 (Redacted, Non-proprietary version)
3. Change to LRA Table 3.2.2-2 Regarding Stainless Steel Bolting
4. Updated LRA Section 18.4, Table A-1, Duane Arnold License Renewal Commitments

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

RAI 4.2.3-1

The current RV pressure-temperature (P-T) limits are valid through 32 EFPY. The applicant does not address as to how it will manage any change in P-T limits as a result of a change in neutron fluence values during the extended period of operation. Therefore, the staff requests that the applicant state the following items in Section 4.2.3 of the (LRA):

(1) Changes in the P-T limits during the extended period of operation will be managed by using approved fluence calculations which address the unit's operating conditions (i.e., thermal power level, core design, etc.) in conjunction with surveillance capsule results.

(2) Any change in P-T limits will be implemented by the license amendment process (i.e., modifications of technical specifications) and will meet the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.60 and 10 CFR Part 50, Appendix G.

DAEC Response to RAI 4.2.3-1

In LRA Section 4.2.3, the first paragraph on the top of page 4.2-20, "Pressure Temperature limits ... higher neutron exposure." is revised in its entirety to read as follows:

> Pressure Temperature Limits for the Reactor Coolant System are currently specified in Technical Specification 3.4.9. Prior to exceeding 32 effective full power years (EFPY), DAEC will incorporate appropriate changes to reflect the higher neutron exposure. Changes in the P-T limits during the period of extended operation will be managed by using approved fluence calculations which address the unit's operating conditions (i.e., thermal power level, core design, etc.) in conjunction with surveillance capsule results. Changes to the P-T limits will be accomplished in accordance with the license amendment process and will meet the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G.

RAI 4.2.4-1

The applicant identified one circumferential weld, DE (VCB-A2), in the beltline region. Probability of failure (PoF) results were calculated for 60 years (54 EFPY) for the RV beltline circumferential weld, including the consideration of the low temperature overpressurization (LTOP) occurrence probability of 1×10^{-3} per year. The PoF for the beltline circumferential weld due to an LTOP event was calculated to be 1.5×10^{-9} per reactor-year at 54 EFPY and 2.5×10^{-11} per year for 90% inspection. Please provide the calculation/analyses used to derive the 54 EFPY PoF values for the beltline circumferential weld. Include the limiting circumferential weld surface fluence value, and the corresponding calculated mean nil-ductility reference temperature (RT_{NDT}).

DAEC Response to RAI 4.2.4-1

A redacted, non-proprietary, version of Structural Integrity Associates Calculation Package DAEC-20Q-338, RPV Circumferential and Axial Weld Examination Relief, is provided in Enclosure 2. The complete analysis is proprietary, since it is based on the methodology presented in the proprietary EPRI document, BWRVIP-05, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations. The material that has been redacted is based on the methodology of BWRVIP-05. The NRC has previously reviewed and accepted the methodology of BWRVIP-05 in a Safety Evaluation (SE), "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," July 28, 1998; and in a SE supplement, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," March 7, 2000 (ML003690281).

The requested information regarding the calculated mean RT_{NDT} has been extracted from the analysis and provided below.

From Table 4 of DAEC-20Q-338:
The maximum 54 EFPY surface fluence at the clad/metal interface for the circumferential weld DE is 5.85×10^{18} n/cm^2. The maximum 54 EFPY surface fluence at the clad/metal interface for the beltline axial welds E1 and E2 is 6.29×10^{18} n/cm^2.

Mean (irradiated) Reference Temperature $RT_{NDT} = RT_{NDT(u)} + \Delta RT_{NDT}$ (°F) where:
$RT_{NDT(u)}$ = Initial (unirradiated) Reference Temperature (°F)
ΔRT_{NDT} = shift in RT_{NDT} due to irradiation without margin (°F)

For the limiting circumferential weld fluence of 5.85×10^{18} n/cm^2 at 54 EFPY, the corresponding calculated RT_{NDT} is obtained using information from Table 4 of DAEC-20Q-338.

$RT_{NDT(u)}$ = -50°F
ΔRT_{NDT} = 31.8°F

Therefore, for the circumferential weld, the mean RT_{NDT} is -50 + 31.8 = -18.2°F.

For the limiting axial weld fluence of 6.29×10^{18} n/cm^2 at 54 EFPY, the corresponding calculated RT_{NDT} is obtained using information from Table 4 of DAEC-20Q-338.

$RT_{NDT(u)}$ = -50°F
ΔRT_{NDT} = 32.6°F for limiting Heat Number 432Z0471

Therefore, for the axial weld, the mean RT_{NDT} is -50 + 32.6 = -17.4°F for limiting Heat Number 432Z0471.

RAI 4.2.4-2

The applicant stated that it will utilize the same procedures and training used to limit RV cold over-pressure events as those approved by the NRC when the DAEC requested approval of the BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," technical alternative for the term of the current operating license. Include specific reference to the stated NRC approved request.

DAEC Response to RAI 4.2.4-2

The referenced request was submitted to the NRC by letter NG-04-0103, (Subject: Request to Implement NDE-R047 Addressing Boiling Water Reactor Shell Weld Inspection Recommendations of the Boiling Water Reactor Vessel And Internals Project (BWRVIP) Report BWRVIP-05), dated February 12, 2004 (ML040550520). The letter requested NRC approval of proposed Relief Request NDE-R047, which requested permanent relief for the remaining term of the DAEC operating license from certain requirements regarding volumetric examination of reactor pressure vessel (RPV) shell circumferential welds. By letter dated January 6, 2005, (Subject: Duane Arnold Energy Center - re: Alternatives for Examination of Reactor Pressure Vessel Circumferential Shell Welds Relief Request NDE-R047, TAC NO. MC2181)(ML043270051), the Staff provided its conclusion that the proposed alternative will provide an acceptable level of quality and safety, and authorized the proposed alternative for the remaining term of the DAEC operating license.

RAI 4.2.5-1

In the BWRVIP-05 safety evaluation (SE), the staff concluded that the applicant need not examine the RPV circumferential shell welds, if the corresponding volumetric examinations of the RV axial shell welds do not reveal the presence of an age-related degradation mechanism. Confirm whether previous volumetric examinations of the RV axial shell welds have shown any indication of cracking or other age-related degradation mechanisms in the welds.

DAEC Response to RAI 4.2.5-1

Volumetric examinations of the RPV axial shell welds have identified no recordable indications of cracking or age-related degradation mechanisms in the welds.

RAI 4.2.5-2

PoF results were calculated for 60 years (54 EFPY) for the RV beltline axial welds, including the consideration of the LTOP occurrence probability of $1\text{x}10^{-3}$ per year. The PoF for the limiting beltline axial weld due to an LTOP event was calculated to be 2.24

$x10^{-7}$ at 54 EFPY and $3.74x10^{-9}$ per year for 90% inspection. Please provide the calculation/analyses used to derive the 54 EFPY PoF values for the limiting beltline axial weld. Include the limiting axial weld surface fluence value, and the corresponding calculated mean RT_{NDT}.

DAEC Response to RAI 4.2.5-2

The requested information related to axial welds is provided in the response to RAI 4.2.4-1 above.

RAI 4.2.6-1

According to the applicant the reflood thermal shock analysis that is applicable to Duane Arnold Energy Center (DAEC) reactor vessel (RV) indicates that the peak stress intensity factor for DAEC RV is bounded by the fracture toughness value of the vessel material under the reflood thermal shock conditions. The staff requests that the applicant provide information regarding the following items:

(1) The methodology that was used to obtain the peak stress intensity factor of the RV material under the reflood thermal shock conditions,

(2) and effect of maximum vessel irradiation (E > 1 MeV) at 54 effective full power years (EFPY) at the mid-core inside RV location on the applicant's analysis.

DAEC Response to RAI 4.2.6-1

A BWR-6 vessel analysis was reevaluated to demonstrate acceptability of the DAEC vessel with respect to this TLAA. The BWR-6 analysis is, "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident" (Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper Gi/5, S. Ranganath). Since the DAEC is a BWR-4 with a smaller vessel diameter and thickness, the analysis was re-evaluated to determine acceptability for the DAEC. The peak stress intensity factor was determined from Figure 5 in the BWR-6 analysis. That analysis assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The critical location for the fracture mechanics analysis is at 1/4T. From Figure 5 of the BWR-6 analysis, the peak stress intensity factor, K, at 1/4T has a value of approximately 100 ksi$\sqrt{\text{inch}}$. Further discussion is provided below.

Since the 1979 analysis considers pressure stresses to be negligible once thermal stresses are fully developed, the DAEC reevaluation only evaluated the impact on thermal stress. Thermal stresses were computed in two axisymmetric cylindrical models: one model for the BWR-6 vessel dimensions from the 1979 evaluation, and the second model for the DAEC RPV. By evaluating the thermal transient shown in the 1979 analysis for both models, the applicability of the 1979 analysis results can be evaluated accordingly for application to the DAEC RPV.

The results indicated that the peak thermal hoop stress for DAEC is smaller than the peak thermal hoop stress for the BWR-6 model (12.69 ksi vs. 15.33 ksi). This is as expected, since the wall thickness is smaller and thermal stress tends to be lower for thinner walled components, and the relatively large radius of the RPV tends to have a negligible impact on thermal stress. Therefore, the thermal stresses from the 1979 fracture mechanics results are conservative for application to DAEC.

The results also indicate that the 1/4T temperature at the time of maximum stress for DAEC is lower than the 1/4T temperature for the BWR-6 model (399°F vs. 403°F). This is as expected, since the wall thickness is smaller for DAEC, so the transient cooldown effects are greater. Therefore, a revised 1/4T temperature of 399°F for DAEC should be used in place of the 1979 temperature results. From Figure 3 of the BWR-6 analysis, the temperature of the vessel wall at 1.5" deep (1/4T on the 6" thick BWR-6 vessel) is approximately 400°F at 300 seconds into the thermal shock event, which is in excellent agreement with the DAEC re-evaluation.

The BWR-6 analysis assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The critical location for the fracture mechanics analysis is at 1/4T. From Figure 5 of the BWR-6 report, the peak stress intensity factor, K, at 1/4T has a value of approximately 100 ksi$\sqrt{\text{inch}}$. The acceptability of this K on a plant-specific basis for DAEC was determined by considering a revised allowable fracture toughness applicable to the DAEC vessel for 54 EFPY. ART values were determined for the DAEC vessel for 54 EFPY of operation, computed in accordance with NRC RG 1.99 Revision 2 and using updated fluence values. The allowable material fracture toughness resides on the upper shelf of 200 ksi$\sqrt{\text{inch}}$ for a (T-RT_{NDT}) value of (399-158.1) = 240.9°F.

RAI 4.2.6-2

The DAEC stainless steel core shroud is subject to radiation embrittlement resulting in loss of structural integrity due to a low pressure coolant injection thermal shock transient. Therefore, the staff requests that the applicant include a reflood thermal shock analysis as a part of a time-limited aging analysis (TLAA) for the core shroud at DAEC.

DAEC Response to RAI 4.2.6-2

As defined in 10 CFR 54.3, to be considered a Time-Limited Aging Analysis (TLAA), the analysis must be contained in the current licensing basis (CLB). A review of the DAEC CLB did not identify any references or other indication that a reflood thermal shock analysis for the core shroud was ever performed for the DAEC. It would be expected that a reflood thermal analysis has not been required, since the DAEC core shroud has not experienced cracking and a tie rod repair has not been installed. (Reflood thermal shock would be expected to have an impact on RPV core shroud tie rod repairs and their ability to maintain tension during low pressure coolant injection events; therefore, such repairs would have necessitated a reflood analysis.)

Analysis of reflood thermal shock for the core shroud is not part of the DAEC CLB and, therefore, is not a TLAA for the DAEC.

RAI 4.2.7-1

The applicant stated that the loss of preload over time in core plate hold-down bolts due to stress relaxation is considered as a TLAA. Therefore, the staff requests that the applicant provide a TLAA analysis (if available) for the core plate hold down bolts for staff review and approval. If this analysis is not currently available, the applicant shall make a commitment to submit this analysis 2 years prior to entering the period of extended operation. The staff expects that this analysis shall use projected neutron fluence values to the end of the extended period of operation.

Since core plate wedges are not installed at DAEC, consistent with the inspection guidance specified in item 10 of Table 3-2 of the BWRVIP-25 report, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," the applicant shall continue enhanced visual inspection (EVT-1) of the core plate hold-down bolts. Therefore, the staff requests that the applicant confirm that it will continue performing EVT-1 inspections of the core plate hold-down bolts and use ultrasonic testing (UT) from a location above the core plate when such a UT technique is developed by the industry.

DAEC Response to RAI 4.2.7-1

A TLAA analysis for the core plate hold-down bolts that uses projected DAEC neutron fluence values for 54 EFPY is not currently available. Accordingly, in LRA Appendix A, Section 18.4, Table A-1 Duane Arnold License Renewal Commitments, a new license renewal commitment 47 is added to read as follows:

Item No.	System, Component or Program	Commitment	Section	Schedule
47.	Reactor Internals	DAEC will submit an analysis for loss of preload in core plate hold-down bolts due to stress relaxation to the NRC for Staff review. The analysis will use projected neutron fluence values for the end of the period of extended operation.	18.3.1.7	Two years prior to entering the period of extended operation

The BWR Vessel Internals Program (BWRVIP) continues to research effective methods to inspect the core plate rim hold-down bolts. The industry has found that EVT-1 inspections can be extremely difficult and are of limited value in inspecting the hold-down bolts. Therefore, DAEC has committed to performing VT-3 inspections of the hold-down bolts. Commitment 37 in LRA Appendix A, Section 18.4, Table A-1 Duane Arnold License Renewal Commitments, states:

> Inspect a sample of the rim hold-down bolts by VT-3 until an expanded technical basis for not inspecting is approved by the NRC.

RAI 4.2.7-2

Table IV.B1-17 of the Generic Aging Lessons Learned (GALL-NUREG-1801) requires that 5% of the top guide locations that are exposed to a neutron fluence exceeding the irradiation-assisted stress corrosion cracking (IASCC) threshold limit of 5×10^{20} (E > 1 MeV) prior to the period of extended operation, be inspected using an EVT-1 technique within 6 years after the period of extended operation. An additional 5% of the top guide locations with an exposure to a neutron fluence value greater than IASCC threshold limit shall be inspected within twelve years after entering into the period of extended operation. The staff requests that the applicant include a statement in Section 4.2.7 that refers to commitments 2 and 3.

DAEC Response to RAI 4.2.7-2

LRA Section 4.2.7 states that the effects of aging will be managed by the Water Chemistry Program and the BWR Vessel Internals Program. LRA Section B.3.14, BWR Vessel Internals Program, Subsection B.3.14.1, reiterates the referenced statements from NUREG -1801, XI.M9, by stating:

> Additionally, for top guides with neutron fluence exceeding the IASCC threshold (5E20, E>1MEV) prior to the period of extended operation, inspect five percent (5%) of the top guide locations using enhanced visual inspection technique, EVT-1 within six years after entering the period of extended operation. An additional 5% of the top guide locations will be inspected within twelve years after entering the period of extended operation.

LRA Section B.3.14.4 explicitly identifies these actions as required enhancements to the BWR Vessel Internals Program. These actions are, in turn, captured in Appendix A as Commitments 2 and 3.

To provide the requested cross-reference in LRA Section 4, the following LRA change is made:

In LRA Section 4.2.7, Reactor Internals, on page 4.2-23 under the Disposition summary for Irradiation Assisted Stress Corrosion Cracking, the following statements are added:

LRA Appendix A contains the following commitments regarding top guide inspections performed by the BWR Vessel Internals Program.

- Commitment 2: Perform an EVT-1 inspection of 5% of the top guide locations within six years of entering the period of extended operation.
- Commitment 3: Perform an EVT-1 inspection of an additional 5% of the top guide locations within 12 years of entering the period of extended operation.

RAI 4.2.7-3

The applicant stated that core shroud circumferential welds H3, H4, H5 and vertical welds V3 though V8 have exceeded the IASCC threshold neutron fluence value. Hence, the staff requests that the applicant confirm that it will implement the requirements specified in footnote 4 of Tables 2-1 and C-9 of the BWRVIP-76, "BWR Vessel Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," report for these core shroud welds.

DAEC Response to RAI 4.2.7-3

The DAEC will implement the requirements specified in footnote 4 of Tables 2-1 and C-9 of BWRVIP-76, "BWR Vessel Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines," for core shroud welds H3, H4, H5, and V3 through V8.

RAI B.3.12-1

Consistent with the requirement specified in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," DAEC included a portion of small bore piping (less than 4") in its American Society of Mechanical Engineers (ASME) Code, Section XI Inservice Inspection (ISI) program. In this context, the staff requests that the applicant provide the following information:

(1) Previous plant experience regarding the aging degradation of small bore piping welds,

(2) type of prior inspections that were performed thus far on the small bore piping welds,

(3) and inspection results followed by any corrective actions that were taken so far to prevent recurrence of any aging degradation in small bore piping welds.

DAEC Response to RAI B.3.12-1

In response to RAI B.3.3.2 in letter NG-09-0764 dated October 13, 2009, NextEra Energy incorporated a new ASME Code Class 1 Small-bore Piping Inspection Program into the LRA. The requested information can be found in that RAI response.

RAI B.3.12-2

DAEC stated that it identified intergranular stress corrosion cracking (IGSCC) in three recirculation riser nozzle-to-safe end welds (RRB-F002, RRD-F002 and RRF-002) and weld overlays were incorporated on these welds as a mitigation technique to prevent further aging degradation. The staff requests that the applicant provide the following information with respect to these welds:

(1) Type of the weld materials that were originally used to fabricate these welds,

(2) mitigation technique (e.g., stress improvement) that was implemented, if any, on these welds prior to the identification of IGSCC,

(3) confirm whether these cracks were through wall prior to the application of the weld overlays,

(4) and future inspection plans for these welds.

DAEC Response to RAI B.3.12-2

During refueling outage (RFO) 16 in 1999, inspections identified flaw indications on three recirculation riser nozzle-to-safe-end welds (RRB-F002, RRD-F002 and RRF-F002). Weld overlays were completed on the B and D riser F002 welds. On the RRF-F002 weld, the flaw was evaluated under the ASME Code and determined to be acceptable to leave as-is. In 2007, during RFO 20, ultrasonic examinations identified indications in welds RRF-F002 and RRC-F002; weld overlays were subsequently completed on both welds. This resulted in a total of four weld overlaid recirculation riser nozzle-to-safe-end welds: RRB-F002, RRC-F002 and RRD-F002 and RRF-F002.

<u>Part (1)</u>
The welds are composed of Alloy 82/182.

<u>Part (2)</u>
No mitigation techniques were implemented on these welds prior to the identification of IGSCC.

<u>Part (3)</u>
The cracks were not through-wall prior to the application of the weld overlays.

<u>Part (4)</u>
The weld overlays on RRB-F002 and RRD-F002, which were applied during RFO 16 (1999), were most recently examined during RFO 21 (2009). Examination results were acceptable. These two overlays are not scheduled for examination again during the current license period.

The weld overlays on RRC-F002 and RRF-F002 which were applied during RFO 20 (2007) are scheduled for examination during RFO 22 (2010).

RAI B3.13-1

The applicant stated that the frequency and the method of inspection specified in BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines," will be implemented for the attachment welds. If furnace-sensitized stainless steel components exist in these systems at DAEC, please provide details on how the aging management program is implemented, i.e. any additional augmented inspection program for the furnace-sensitized stainless steel materials. These requirements apply to, but are not limited to, jet pump raiser brace attachments, core spray piping bracket attachments, steam dryer support and hold down brackets, feedwater spargers, guide rods, and surveillance sample holders.

DAEC Response to RAI B3.13-1

No furnace-sensitized stainless steel materials have been identified in the components within the scope of the DAEC BWR Vessel ID Attachment Weld Program.

RAI B.3.14-1

The applicant should confirm whether it is incorporating hydrogen water chemistry (HWC) and/or noble metal chemical addition (NMCA) in its water chemistry program. The applicant should provide the following information with respect to the implementation of HWC and/or NMCA in its water chemistry program.

> Confirm the method of controlling HWC and/or NMCA in the RV. The applicant should explain how this implementation has affected the plant chemical parameters. Provide details on the methods for determining the effectiveness of HWC and/or NMCA by using the following parameters:
>
> (1) Electrochemical potential (ECP)
> (2) Feedwater hydrogen flow
> (3) Main steam line oxygen content
> (4) Hydrogen/oxygen molar ratio

If ECP is measured to verify the effectiveness of HWC and/or NMCA, provide information regarding the locations at which the ECP measurements are taken.

DAEC Response to RAI B.3.14-1

DAEC has incorporated hydrogen water chemistry (HWC) and Noble Metal Chemical Addition (NMCA) in its Water Chemistry Program in accordance with BWRVIP water chemistry guidelines. The method of controlling HWC and NMCA is defined in plant chemistry procedures.

Implementation of HWC and NMCA affects plant parameters in the following ways:

- Dissolved oxygen values in reactor recirculation system and reactor vessel changed from 200 ppb to ≤ 0.2 ppb.

- Feedwater dissolved oxygen values changed from 20 ppb to 40 ppb. The preferred range is 30-200 ppb in accordance with EPRI BWR Water Chemistry Guidelines.

- Reactor recirculation and RWCU inlet conductivity and pH values remained essentially constant with increasing feedwater hydrogen concentration (0.08 μS/cm and neutral pH, respectively). The overall effect is a reduction in reactor water conductivity. Also, note that implementation of zinc injection per the GEZIP program increased reactor conductivity from 0.055 μS/cm to the current value.

- Main Steam line oxygen content decreases from about 18 ppm to about 12.5 ppm when HWC is in-service.

- Chromates are significantly reduced to non-detectable levels with HWC in the reactor water.

Implementation of HWC and NMCA has been very effective in protection of the reactor recirculation piping and vessel internals, as illustrated by the following:

- Electrochemical Potential (ECP) data collection ensures reactor water ECP is <-230 mV she. A typical value for ECP is -470 mV she.

- Feedwater hydrogen flow is monitored to ensure approximately 6 scfm total is injected into the suction of the feedwater pumps (3 scfm each loop) at 100% power.

- Main Steam Line oxygen content is reduced. DAEC Data shows 12.5 ppm oxygen at 6 scfm and 18.0 ppm at 0 scfm.

- Hydrogen/oxygen molar ratio is maintained in accordance with the plant chemistry BWRVIP program procedure. Molar ratio can be determined by normally monitored feedwater H2 flow rate and reactor water dissolved oxygen or from the molar ratio model. If used, modeled molar ratio does not need to be run daily, but the analysis should bound the operating conditions for the current cycle or operational condition. Normally a 4:1 hydrogen/oxygen molar ratio is maintained at DAEC.

ECP is measured to verify the effectiveness of HWC and NMCA. The location of the ECP sensors is at the Crack Arrest Verification Sample Station ECP vessel with sample flow from the "B" loop reactor recirculation riser via the reactor recirculation sample valves.

RAI B.3.14-2

BWRVIP-18-A, "BWR Core Spray Inspection and Flaw Guidelines."

Some welds that are not accessible in core spray systems require inspections per Table 3-5 of the BWRVIP-18-A report. In this context, the staff requests that the applicant provide the following information:

(1) Type of prior inspections that were performed thus far on these welds,

(2) and inspection results followed by any corrective actions that were taken so far to prevent recurrence of any aging degradation of these welds.

DAEC Response to RAI B.3.14-2

Part (1)
BWRVIP-18-A, Table 3-5 includes inspection requirements for weld P1 (if accessible). Inspection is required every outage if inspection is by VT, and every other outage if by UT. BWRVIP-18-A Section 3.2.4 provides additional discussion regarding the P1 weld, and states that if P1 is completely inaccessible, a qualitative evaluation of P1 shall be performed if cracking is found in similar welds.

Weld P1 (Thermal Sleeve Connection) is inaccessible at DAEC; therefore, the inspection and corrective action requirements for P1 are based on the results of inspections of similar accessible welds.

The similar accessible welds for the DAEC P1 weld are P2, P3, P5, P6, P7, P8a, and P8b. Baseline examinations for these accessible welds were completed in refueling outage (RFO) 14 (1996) by EVT-1; no recordable indications were identified. EVT-1 has been performed every RFO since then with no recordable indications being identified in any of the similar accessible welds.

Part (2)
No cracking has been identified in similar accessible welds; therefore, no corrective actions have been required for the P1 weld. If cracking is detected, the flaw evaluation procedure of BWRVIP-18 Rev. 1 will be used.

RAI B.3.14-3

BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines."

(1) Identify the type of core plugs, if any, that are currently present in the core plate assembly (i.e., spring loaded plug or fillet all around plug) at DAEC,

(2) and identify the type of inspections that were performed thus far on these plugs, the results of the inspections and corrective actions taken to prevent recurrence of any aging degradation.

DAEC Response to RAI B.3.14-3

Part (1)
The installed type of core plug at DAEC is a spring-loaded plug type.

Part (2)
Inspections are not performed on the installed core plate plugs. The plugs are instead replaced on a frequency recommended by GE. The core plate plugs were last replaced in 1995. The preventive maintenance item for replacement on a 14 EFPY frequency is currently scheduled for 2010. GE is performing an evaluation to extend core plate plug service life from 14 EFPY to 18 EFPY. Replacement of the core plate plugs is expected to occur in 2012 or 2014 depending on the results of GE's evaluation.

RAI B.3.14-4

To ensure that the aging degradation in the following reactor vessel internals (RVI) components is adequately monitored, the staff requests that the applicant identify which of the following RVI components exhibited cracking, the corrective actions taken and any additional augmented inspections implemented as part of corrective actions:

(A) core spray system;
(B) core plate;
(C) steam dryer;
(D) top guide;
(E) control rod drive guide tube, stub tube, in-core housing and dry tube;
(F) and RV penetrations

DAEC Response to RAI B.3.14-4

Of the reactor vessel internals components listed above, indications of cracking have been identified only in the steam dryer and the dry tubes, as discussed below.

Indications indicative of cracking have been identified in the steam dryer. They have been evaluated and accepted for continued service on a cycle-by-cycle basis. The last inspection of the steam dryer was performed during the spring, 2009, refueling outage. The results of the 2009 inspection, performed per BWRVIP-139, showed that previously-identified indications had not changed. The 2009 examination also identified one new crack-like indication around the entire perimeter of the Tie Bar 4 to baffle plate weld. The indication appears to be due to fatigue. An evaluation was performed which determined that it is acceptable to operate with the cracked Tie Bar 4 to baffle plate weld for the next 24-month cycle, but recommended that Tie Bar 4 be repaired during the next refueling outage.

Indications were identified in 1988 in two dry tubes, resulting in replacement of the dry tubes with a new design. Visual examinations of the accessible portions of accessible dry tubes have been performed in 1998, 1999, 2007 and 2009. No indications of cracking have been found since the dry tube replacements.

RAI B.3.14-5

BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines."

(A) With regard to previous inspections of the core shroud components, the staff requests that the applicant provide the following information:

  (1) identify which core shroud welds showed indications during previous inspections,

  (2) and discuss tie rod repairs, if any, that were performed at DAEC.

(B) Reduction in ductility and fracture toughness can occur in stainless steel RVI components when they are exposed to high energy neutrons (E > 1 MeV). In August 2006, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) issued the staff-approved BWRVIP-100-A report, "Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," which discusses fracture toughness results for the irradiated stainless steel materials. For stainless steel materials with exposure to a neutron fluence value equal to or greater than 1 X 10^{21} n/cm^2 (E > 1 MeV), the BWRVIP-100-A report identified a lower fracture toughness value than that reported in Appendix C of the BWRVIP-76 report. During the license renewal period, core shroud welds and base materials may be exposed to neutron fluence values 1 X 10^{21} n/cm^2 (E > 1 MeV) or greater. Since the inspection frequency in the

BWRVIP-76 report is based on fracture toughness values which are less conservative than the BWRVIP-100-A report, the staff requests that the applicant make a commitment that it will incorporate the crack growth rate evaluations specified in the BWRVIP-100-A report and develop generic inspection intervals for 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).

(C) The applicant shall make the following statement in aging management program (AMP) B.3.14, "When a tie rod repair is considered as a repair option, the implications of the Hatch, Unit 1 tie rod repair cracking and the revised inspection guidelines, if any, developed by the BWRVIP will be used."

(D) The staff's guidance in Table IV.B1 of the GALL Report lists two potentially applicable aging effects for BWR core shrouds and core shroud repair assembly components that are made from either stainless steel (including cast austenitic stainless steel) or nickel alloys: (1) loss of material due to pitting and crevice corrosion (refer to GALL Aging Management Review [AMR] IV.B1-15), and (2) cumulative fatigue damage (refer to GALL AMR Item IV.B1-14). The BWRVIP-76 report does not address these aging effects, hence the staff requests that the applicant revise the LRA to include the aforementioned aging effects and perform necessary TLAA, if any, for the core shroud components.

DAEC Response to RAI B.3.14-5

Part (A)

(1) As reported to NRC in letter NG-95-1302 (Duane Arnold Energy Center ... Response to Generic Letter 94-03 Item 3, Inspection Results of Core Shroud Inspections), dated April 21, 1995, the DAEC core shroud inspection by ultrasonic examination was completed on March 29, 1995. Core shroud circumferential welds H-1 through H-7 were examined with no indications of intergranular stress corrosion cracking detected.

As reported to NRC in letter NG-01-0975 (Duane Arnold Energy Center ... Inservice Inspection Report), dated August 15, 2001 (ML012390015), core shroud circumferential welds H-1 through H-7 were again ultrasonically examined during refueling outage 17. No reportable indications were detected.

(2) Core shroud tie rod repairs have not been performed at the DAEC.

Part (B)

As discussed in BWRVIP-100-A, Section 4.1, Conclusions, generic inspection intervals can not be established for welds exposed to very high fluences, and the inspection intervals need to be evaluated on a case specific basis for fluence levels greater than approximately 1E21 n/cm^2. Therefore, the following commitment is made:

In LRA Appendix A, Section 18.4, Table A-1 Duane Arnold License Renewal Commitments, a new license renewal commitment 46 is added to read as follows:

Item No.	System, Component or Program	Commitment	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.	18.1.14	Prior to the period of extended operation

Part (C)

In LRA Section B.3.14, BWR Vessel Internals Program, Subsection B.3.14.1, Program Description, on pages B-32 and B-33, the following statement is added at the end of the existing text:

> If a tie rod repair is used in the future, the Hatch Unit 1 tie rod repair cracking operating experience will be considered, and revised inspection guidelines, if any have been developed by the BWRVIP, will be used.

Part (D)

As stated in LRA Section B.3.14, the BWR Vessel Internals Program incorporates the guidelines of the appropriate BWRVIP documents and ASME Section XI. NUREG-1801, XI.M9 BWR Vessel Internals, states that vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2. The ASME Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. This inspection also specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters, such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

Therefore, while BWRVIP-76 may not explicitly mention a loss of material due specifically to pitting and crevice corrosion, the inspections performed for ASME Section XI, Subsection IWB, examination category B-N-2 (which are included in the DAEC BWR Vessel Internals Program) implicitly address this aging effect/mechanism.

The DAEC CLB does not contain a fatigue evaluation for the core shroud; therefore, fatigue is not a TLAA for the DAEC core shroud.

RAI B.3.14-6

BWRVIP-41, "BWR Vessel and Internals Project, Jet Pump Assembly, Inspection and Flaw Evaluation."

(A) The staff requests that the applicant provide information regarding DAEC's plant-specific experience related to the cracking of the jet pump components, including any corrective actions or subsequent re-inspections if cracking has been observed.

(B) According to Table 3.3-1 of the BWRVIP-41 report, jet pump thermal sleeve welds have limited access for visual inspections and, therefore, the staff requests that the applicant provide the following information with respect to these welds. The staff determined that the applicant's response to the following items is essential in order to assess the applicant's capability in managing aging degradation in these welds:

(1) previous inspections and the results of the inspections that were performed on these welds,

(2) and if no inspections have been performed so far, the applicant's future plans to inspect these welds. A technical explanation is required if the applicant proposes to not inspect these welds during the license renewal period.

DAEC Response to RAI B.3.14-6

<u>Part (A)</u>

In the early 1990s, DAEC experienced cracking in jet pump restrainer set screw tack welds. In 1990, during refueling outage (RF0) 10, cracks were identified in five (out of a total of 64) jet pump restrainer set screw tack welds. For jet pumps #2, #4 and #10, one tack weld was cracked, and for jet pump #5, two tack welds were cracked. An evaluation by General Electric suggested that the likely cause was high cycle fatigue due to vibration. Continued operation for one cycle was determined to be acceptable provided only one tack weld on any adjustment screw was cracked, as was the case. Inspections during RFO 11 (in 1992) found two additional tack welds cracked, one for jet pump #9 and another one for jet pump #10. (No more than one tack weld on any adjustment screw was cracked.) During RFO 12 (in 1993), one additional set screw tack weld was found cracked (for jet pump #3). During RFO 12, a planned set screw mitigation process was performed on all set screws which applied two additional "mitigation" tack welds to each of the jet pump adjustment screws. No further cracking has been identified since the mitigation welds were applied.

Part (B)

The jet pump nozzle thermal sleeve welds (TS-1 and TS-2) are identified in BWRVIP-41 as inaccessible. There is currently no inspection technique developed for examining these welds. Therefore, welds TS-1 and TS-2 have not been previously visually inspected.

BWRVIP-41, Section 2.3.3.5 (Failure Consequences) recommends a plant specific review to determine if the failure of TS-1 or TS-2 would result in the Thermal Sleeve disengaging from the nozzle before the riser contacts the shroud. A review of construction drawings showed that the dimension from the riser pipe to the shroud is 5.495". The TS-2 weld is 20.06" in the nozzle bore, while TS-1 weld is 23.44" in the nozzle bore. This indicates that the riser pipe would contact the shroud prior to the thermal sleeve disengaging from the nozzle.

DAEC will continue to monitor the progress of the BWRVIP Committee in the development of an examination technique. It is important to note that this particular region is protected by Hydrogen Water Chemistry and, therefore, considered to be protected against intergranular stress corrosion cracking.

An additional weld (TS-1A) has been identified on the thermal sleeve connecting to the Jet Pump riser pipe inside the reactor vessel. This amounts to a total of eight welds (one for each Jet Pump Riser). Weld TS-1A is approximately one inch from the thermal sleeve to riser elbow weld. Three of the eight TS-1A welds were examined during the 2009 RFO; welds JP-01-02 TS1A, JP-03-04 TS1A, and JP-13-14 TS1A were examined by EVT-1. Coverage was limited; no reportable indications were identified.

RAI B.3.14-7

Access Hole Covers

The AMP does not address augmented inspection of access hole covers which is required in accordance with the requirements of GALL AMR Item IV-B.1.1-d.

DAEC Response to RAI B.3.14-7

As stated in LRA Table 3.1-1, Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 Reactor Coolant System, Item 3.1.1-49, on page 3.1-19, cracking in the core plate access hole covers is managed by the ASME XI Inservice Inspection, Subsections IWB, IWC & IWD Program, and the Water Chemistry Program. Due to the presence of a crevice in the access hole covers, augmented UT examinations will be done in accordance with BWRVIP-180.

RAI B.3.14-8

Reactor Vessel Flange Leak Detection Line

Crack initiation and growth due to thermal and mechanical loading or stress corrosion cracking (SCC) could occur in the BWR reactor vessel flange leak detection line. In accordance with the requirements of Table IV item A1.1.5 in the GALL report, a plant-specific AMP is to be evaluated for this component. In Appendix C of the LRA, the applicant stated that the AMP for this item will include implementation of BWR water chemistry and one time inspection. The staff determined that the one-time inspection is not adequate to identify cracking due to SCC in a timely manner. Hence, the staff requests that the applicant provide justification for using a one-time inspection as a part of the AMP for managing this aging effect in the reactor vessel flange leak detection line.

DAEC Response to RAI B.3.14-8

At DAEC, the Reactor Vessel Flange Leak Detection Line is constructed of ASTM A-106 Gr. B carbon steel material, not stainless steel or nickel alloy as addressed in the GALL item. Therefore, since the line is made of carbon steel, crack initiation and growth due to thermal and mechanical loading or SCC is not an applicable aging effect. The statement in Appendix C is intended to summarize the programs credited for management of the applicable aging mechanism (loss of material) for the leak-off line.

RAI B.3.35-1

The staff requests that the applicant include the following statements in the Updated Final Safety Analysis Report (UFSAR) and commitment table of the LRA:

(1) The applicant will obtain NRC approval of any change in the withdrawal schedules of the RV surveillance capsules,

(2) and if a standby capsule is removed from the RV without the intent to test it, the capsule will be stored in manner which maintains it in a condition which would permit its future use, including during the period of extended operation if necessary.

DAEC Response to RAI B.3.35-1

Commitment 28 in LRA Appendix A, Table A-1 Duane Arnold License Renewal Commitments, is revised as follows:

Item No.	System, Component or Program	Commitment	Section	Schedule
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, including the following: • NRC approval will be obtained for any change in the withdrawal schedules of the DAEC Reactor Vessel surveillance capsules. • If a standby capsule is removed from the DAEC Reactor Vessel without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary.	18.1.35	Prior to the period of extended operation

Enclosure 2 to NG-09-0663
Structural Integrity Associates Calculation Package DAEC-20Q-338
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