February 22, 2010

Mr. Christopher Costanzo
Vice President, Nuclear Plant Support
NextEra Energy Duane Arnold, LLC
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DUANE ARNOLD ENERGY CENTER LICENSE RENEWAL APPLICATION – BATCH 4 (TAC NO. MD9769)

Dear Mr. Costanzo:

By letter dated September 30, 2008, as supplemented by letter dated January 23, 2009, FPL Energy Duane Arnold, LLC, submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54) for renewal of Operating License No. DPR-49 for the Duane Arnold Energy Center. The staff of the U.S. Nuclear Regulatory Commission (the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its audit review from August 10, 2009 through August 14, 2009, the staff has identified areas where additional information is needed to complete the review. The staff's requests for additional information are included in the enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Ken Putnam, of your staff, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-2277 or by e-mail at Brian.Harris2@nrc.gov.

Sincerely,

/RA/

Brian K. Harris, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure:
As stated

cc w/encl: See next page

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DATE:	02/22/10	02/ 19 /10	02/22/10	02/22/10

OFFICIAL AGENCY RECORD

Letter to Christopher Costanzo from Brian K. Harris dated February 22, 2010

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DUANE ARNOLD ENERGY CENTER LICENSE RENEWAL APPLICATION – BATCH 4 (TAC NO. MD9769)

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DUANE ARNOLD ENERGY CENTER
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION

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 B.31.7 methodology is dispositioned in accrordance 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 B.31.7 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), then the evaluation is not valid for the period of extended operation (PEO).

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 B.31.7 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 B.31.7 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 B.31.7 methodology during the PEO.

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 fatgiue 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 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.

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).

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).

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).

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).

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).

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).

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.

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.

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.

RAI B.3.15-X

Background

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.