September 24, 2009

Mr. Christopher Costanzo
Vice President, Nuclear Plant Support
Florida Power & Light Company
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 – TIME-LIMITED AGING ANALYSIS SECTION 4.2, AGING MANAGEMENT PROGRAMS B.3.12, B.3.13, B.3.14 AND B.3.35 (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 (NRC or 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 review, 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 Mr. 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:	09/22/09	09/22/09	09/22/09	09/24/09

OFFICIAL AGENCY RECORD

Letter to Christopher Costanzo from Brian K. Harris dated September 24, 2009

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Duane Arnold Energy Center

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REQUEST FOR ADDITIONAL INFORMATION
RELATED TO TIME LIMITED AGING ANALYSES SECTION 4.2 AND AGING MANAGEMENT PROGRAMS B.3.12, B.3.13, B.3.14, AND B.3.35
LICENSE RENEWAL APPLICATION FOR DUANE ARNOLD ENERGY CENTER
DOCKET NO. 50-331

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.

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.

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.

ENCLOSURE

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 $1x10^{-3}$ per year. The PoF for the beltline circumferential weld due to an LTOP event was calculated to be $1.5x10^{-9}$ per reactor-year at 54 EFPY and $2.5x10^{-11}$ 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}).

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.

RAI 4.2.5-1

In the BWRVIP-05 safety evaluation (SE), the staff concluded that the applicant need not examine the reactor pressure vessel (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.

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 $1x10^{-3}$ 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}.

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 two 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.

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 six 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.

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.

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.

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.

RAI B.3.13-1

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.

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.

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.

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.

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

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 Boling 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 1×10^{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 1×10^{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 1×10^{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.

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.

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.

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.

RAI B.3.35-1

The staff requests that the applicant include the following statements in the Updated Final Safety Analysis Report 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.