NEXTera ENERGY
DUANE ARNOLD

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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 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 (NextEra Energy Duane Arnold, LLC), "Request for Additional Information for the Review of the Duane Arnold Energy Center License Renewal Application - Batch 2 (TAC No. MD9769)," dated October 16, 2009 (ML092870537)

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 for the review of the LRA.

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

This letter contains no new commitments or changes to existing commitments.

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

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

Executed on November 16, 2009.

Christopher R. Costanzo

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

Enclosure: DAEC Responses to NRC Requests for Additional Information

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
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RAI 4.3.1-1.1

Background

License renewal application (LRA) Section 4.3.1 states that in 1998 Duane Arnold Energy Center (DAEC) performed reassessment of DAEC Reactor Pressure Vessel (RPV) to remove excess conservatism from the existing fatigue calculations for all RPV components, and to incorporate transient cycles projected to occur at 40 years based on actual plant operation as of that time.

Issue

With the original design fatigue analysis and the 1998 reassessment together, DAEC has two sets of fatigue analyses results for the RPV components. However, it is unclear to the staff, which of the two analyses the "Cumulative Usage Factor (CUF) 40 year" column of LRA Table 4.3-2 represents since no CUF results are shown in the Updated Final Safety Analysis Report.

Request

Clarify which of the analyses that the "CUF 40 year" column of LRA Table 4.3-2 represents the original design analysis, or the 1998 reassessment. If it represents the 1998 reassessment then why do 60-year projected CUF values for some locations such as Main Closure and Control Rod Drive-Heat Rejection System Nozzle still show additional significant reductions when the excessive conservatism (including reduction in transient cycles based on projections) have already been removed in the reassessment analysis? If it represents the original design analysis, then what was the true purpose of the reassessment and where were the results used?

DAEC Response to RAI 4.3.1-1.1

The calculation that determined the values of 60 year CUFs used the following process. For each location, previous CUF calculations were identified, and the previous fatigue calculation from the governing stress report was reproduced. A revised calculation was then prepared to provide the fatigue usage for 60 years of operation using projected cycle counts for 60 years (as shown in LRA Table 4.3-1) and relevant Extended Power Uprate (EPU) effects. Because the process involved reproducing the 40-year CUF, there are locations where the value reported in LRA Table 4.3-2 may not match exactly the CUF reported in the governing stress report. For example, Table 4.3-2 lists the 40-year CUF for Shroud Support Pt 42 as 0.3197, whereas the 1998 reassessment (governing stress report for that location) lists the value as 0.320. The differences are insignificant, and the process ensured that the current basis was appropriately used in light of the fatigue evaluation issues identified in LRA Section 4.3.1 and discussed further in the response to RAI B.4.2-2 in letter NG-09-0764 dated October 13, 2009.

The end result of this process was that the "CUF 40 year" column of LRA Table 4.3-2 presents the values from the governing stress report, with minor round-off differences or corrections as discussed below. The governing stress report was the 1998 reassessment except as discussed below.

Main Closure Studs

The main closure region had been reevaluated to support reduced-pass stud tensioning used in the 2007 refueling outage. Therefore, that value is presented in LRA Table 4.3-2 as the 40-year CUF. As shown in Table 4.3-2, the 40-year CUF for the Main Closure Studs is 0.9191, and the 60-year CUF is 0.5842. As discussed in LRA Section 4.3.1, the decrease in the value of CUF for 60 years (as compared to the CUF for 40 years) is primarily because the 60 year evaluation assumes less than half as many bolt-up/unbolt cycles over the plant life. This results in a much smaller contribution from that transient to the overall usage factor for the component.

Shroud Support

As discussed in the DAEC response to RAI 4.3.1-1 in letter NG-09-0764 dated October 13, 2009, the 1998 reassessment had removed some, but not all, of the excess conservatism from the original analysis for Shroud Support Pt 21. As shown in the response to RAI 4.3.1-1, the 40-year CUF value for Pt 21 is 0.97 from the original Chicago Bridge & Iron (CB&I) report, and 0.773 from the 1998 reassessment. In addition, the evaluation performed in support of EPU refined the CUF calculation further. The EPU evaluation resulted in a 40-year CUF for Pt 21 of 0.1991. This is the value provided for the Shroud Support in LRA Table 4.3-2. Further details of the EPU evaluation of CUF for the Shroud Support were provided to the NRC in letter NG-01-0463 dated April 16, 2001.

The CUF values for Shroud Support Points 19, 42, and 44 were not evaluated for EPU. The values included in LRA Table 4.3-2 were obtained from the 1998 reassessment.

Feedwater Nozzle

The values for 40-year CUFs for the Feedwater Nozzle were obtained from the 1998 reassessment, with a correction. During the evaluation of CUFs for 60 years, an error was identified in the 1998 reassessment of the 40-year CUFs for the Feedwater Nozzle. The number of cycles used for the load pair "all other cycles" was 12,488 cycles, rather than the correct value of 12,588 cycles. This error was corrected in the 40-year CUF values reported in LRA Table 4.3-2. Note that the difference in CUFs due to this error in the 1998 reassessment is very small; for Thermal Sleeve Pt. 7 (the location with the largest difference in CUF due to the error) the value of CUF in the 1998 reassessment is 0.639, while the corrected CUF presented in LRA Table 4.3-2 is 0.6519.

CRD Penetration

The value of the 40-year CUF provided in LRA Table 4.3-2 for the RPV Wall Contour Grinding was obtained from the original analysis. As stated in the response to RAI 4.3.1-1 in NG-09-0764 dated October 13, 2009, the 1998 reassessment did not evaluate this location.

CRD-HSR (Control Rod Drive Hydraulic System Return) Nozzle Safe End

The value of the 40-year CUF provided in LRA Table 4.3-2 for the Control Rod Drive Hydraulic System Return Safe End was obtained from the 1998 reassessment. Since the RAI specifically refers to this nozzle as showing a significant reduction in CUF from 40 to 60 years, further discussion is warranted. The 1998 reassessment assumed that all of the cycles resulted in the maximum calculated stress range. The 60 year calculation "separated" the scram cycles (with a lower alternating stress (S_{alt})) from "all other transients" (with a higher S_{alt}) in the CUF calculation, instead of applying the maximum stress range to all transients.

Recirc Inlet Nozzle Safe End

The value of the 40-year CUF provided in LRA Table 4.3-2 for the Recirc Inlet Nozzle Safe End was obtained from the Replacement Safe End Stress Report. The 1998 reassessment did not consider the Replacement Safe End Stress Report, and, therefore, did not provide a valid CUF for the Recirc Inlet Nozzle Safe End.

Recirc Outlet Nozzle

During a review of stress evaluations, it was noted that piping reaction loads obtained in some of the Class 1 piping analyses appeared to be higher than the design basis reaction loads for certain RPV nozzles, including the Recirc Outlet Nozzle, as evaluated in the original RPV Stress Report. Therefore, further evaluation was performed. The fatigue evaluation performed for EPU had reduced the conservatism in the Recirc Outlet Nozzle analysis, resulting in a CUF of 0.411. Information regarding this reduction in analysis conservatism was provided to the NRC in letter NG-01-0463 dated April 16, 2001. The evaluation performed in response to the question about the reaction loads in the Recirc Outlet Nozzle took the CUF from the EPU evaluation into account and determined a revised CUF of 0.4084. This is the value reported in LRA Table 4.3-2.

RAI B.4.2-5

Background

LRA Section B.4.2.5, on operating experience, states that in 2007, a nuclear oversight evaluator found that procedural direction did not exist to record cumulative time spent in a hot-standby condition. The applicant states that the issue was addressed in accordance with the corrective action program.

Issue

The onsite basis document, LRAP-XM01, Rev. 1, states that after investigation it was concluded that hour count capturing was removed from the procedure. However, there was no discussion as to what extent would the lacking of hour and minute portion of the time records impair the accuracy of the monitored transient data.

Request

1. Provide the operating period, in terms of years and months, during which the issue existed.
2. Provide justification that the monitored transients are valid during this period.

DAEC Response to RAI B.4.2-5

Part 1

Procedural direction for counting transient cycles experienced by the reactor vessel is contained in a Surveillance Test Procedure (STP). This procedure is performed every refueling outage to count the transients experienced by the reactor vessel during the prior operating cycle, and the cumulative totals are incorporated into the procedure. A nuclear oversight assessor noticed that Revision 5 of the STP (effective date 9/28/05) contained a requirement for recording the hours spent in a hot standby/shutdown condition, and that Revision 6 (effective date1/25/07) did not contain that requirement. The issue was entered into the Corrective Action Program and a review was performed. The review determined that the hours in hot standby/shutdown should be recorded, and the requirement was put back into the STP in Revision 9 (effective date 6/5/07). As a result, the requirement was not specified in the procedure from January 25, 2007 to June 5, 2007, or approximately five months.

Part 2

Since Revision 5 of the STP contained the cumulative hours recorded through the 2005 refueling outage (RFO 19), and the requirement was put back into the procedure after the spring 2007 refueling outage (RFO 20), the 2007 performance of the STP did not record the hours in hot standby/shutdown for the period between RFO 19 and RFO 20. To correct this situation, transient events that occurred from RFO 19 through RFO 20 were reviewed to ensure that, if additional hours in hot standby/shutdown had occurred, they would be added to the cumulative total. The review concluded that there were no events resulting in additional hours in hot standby during Cycle 20, so the previous cumulative hour totals from the 2005 performance of the STP were added back into the STP in Revision 9. The issue was resolved with no loss of data regarding the hours in hot standby.

RAI B.4.2-6

Background

LRA Section B.4.2.5, on operating experience, states that in June 2000 and November 2006, the DAEC reactor bottom head and drain line pipe experienced rapid temperature drops related to reactor scrams.

Issue

The LRA states that DAEC took corrective actions to evaluate if the current design calculations require specific thermal cycle counting for attached piping, or do the calculations ensure that vessel cycle counting will adequately track attached piping cycles. However, the LRA does not indicate that fatigue analysis has been performed for the November 2006, event.

Request

1. Clarify whether a follow-up fatigue analysis was performed for the November 2006, incident and whether additional cycles were added to an appropriate representative transient to reflect both incidents (2000 and 2006). Identify the name of the representing transient.
2. Summarize the status of the DAEC plan of tracking thermal cycles for the attached piping, as stated in the corrective actions.

DAEC Response to RAI B.4.2.-6

Part 1
For both the 2000 and 2006 events, heatup/cooldown rates greater than 100 °F/hr were determined to have occurred in the bottom head drain piping, but not in the vessel bottom head or bottom head drain nozzle. This would be expected due to the differences in thickness between the piping and the vessel/nozzle. Since the vessel did not experience excessive heatup/cooldown rates, follow-up fatigue analyses for the vessel were not needed, nor were the events required to be counted against more severe transients. The transient in 2000 was counted as a "Scram to Hot Standby and Return to Power." The transient in 2006 was counted as a "Shutdown," since Cold Shutdown was entered.

Part 2
The corrective action evaluation concluded that a thermal transient monitoring and evaluation process is not needed for this piping. This was based on a review of the Class 1, ANSI B31.7, piping fatigue analysis that shows that the piping has been qualified for a much more severe transient (more than a 4000 °F/hr cooldown rate) for at least 10 cycles along with 28 other loading conditions and transients whose cumulative fatigue at the limiting location is less than one quarter of the allowable value (1.0).

This review also concluded that the specified design thermal transients for the bottom head piping closely match the reactor vessel bottom head design thermal cycles. In fact, the design cooldown rate for the vessel bottom head is very similar to the drain piping for the loss of recirculation flow after a scram, which is postulated to occur 10 times during plant life. Based on the similarities and conservatisms in the specified design cycles for both the vessel and piping for this type of transient, separate monitoring and evaluation of drain piping transient cycles are not warranted.

RAI 4.2.8-1

Background

Generic Aging Lessons Learned Table IV A1, Item IV.A1-4, paragraph (c) of the aging management program (AMP) column requires that the upper shelf energy (USE) or equivalent margin analysis (EMA) be performed as part of time-limited aging analysis for ferritic materials that are exposed to a neutron fluence value greater than 1×10^{17} n/cm^2 (E > 1 MeV).

Issue

Section 4.2.2, Table 4.2.2-1 in LRA does not include EMA for the N-2 and N-16 nozzle materials and the associated welds.

Request

The staff requests that the applicant submit the following information with respect to the applicant's EMA evaluation of these nozzles and the associated welds.

1. Initial (un-irradiated) USE values (if available) for the N-2 and N-16 nozzle materials and the associated welds. If the initial USE values are available, calculate the percent USE reduction for nozzle and the welds per the requirements specified Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

2. If the initial USE values for N-2 and N-16 nozzle materials and the associated welds are not available, an EMA per the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-74-A, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," is required.

DAEC Response to RAI 4.2.8-1

Initial USE values for the N-2 and N-16 nozzle materials and the associated welds are not available. However, copper (Cu) content for the nozzle materials and associated welds are available from Certified Material Test Reports (CMTRs). In addition, fluence was calculated for 54 effective full power years (EFPY) for these materials as a part of the recent fluence evaluation performed for the period of extended operation, as discussed in LRA Section 4.2.1. Therefore, these values can be used to perform EMA assessments for the nozzle materials and associated welds. A summary of the Cu content and fluence for these materials is shown in Table 1 below. As noted in Table 1, the EMA for the N-2 and N-16 nozzle weld materials is bounded by the EMA contained in the DAEC LRA for Lower Intermediate Weld E1, E2 (Heat Number 432Z0471) in LRA Table 4.2.1-9. The EMA assessment for the limiting N-2 and N-16 nozzle forging material is shown in Table 2.

From the results provided in Tables 1 and 2, it is concluded that the EMA evaluations for the N-2 and N-16 nozzle materials and the associated welds demonstrate acceptable results for 54 EFPY.

Table 1 EMA Input Values for N-2 and N-16 Nozzle Materials

Material	Copper (Cu) Content (wt. %)	1/4T Fluence at 54 EFPY (n/cm^2)
N-2 Nozzle Forging	0.180 (1)	5.22×10^{17} (3)
N-2 Nozzle Weld (4)	0.03 (2)	5.22×10^{17} (3)
N-16 Nozzle Forging (5)	0.180 (1)	2.21×10^{18} (3)
N-16 Nozzle Weld (4)	0.03 (2)	2.21×10^{18} (3)

Notes:
1. Cu value from Table 4.2.2-1 of the DAEC LRA.
2. Maximum Cu value from material-specific CMTRs for all weld materials used in nozzle weld.
3. Fluence value from Table 4.2.2-1 of the DAEC LRA. The value is the bounding value for both the nozzle forging and nozzle weld materials.
4. The EMA for Lower Intermediate Weld E1, E2 (Heat Number 432Z0471) in Table 4.2.1-9 of the DAEC LRA is bounding for these welds with a Cu content of 0.03 and a 1/4T 54 EFPY fluence of 4.81×10^{18} n/cm^2.
5. Limiting of the N-2 and N-16 nozzles; EMA shown in Table 2.

Table 2 EMA for the N-2 and N-16 Nozzle Forging Materials

BWR 3/6 Plate
(N-2 and N-16 Nozzles Forgings, Heat Nos. Q2Q6VW and Q2Q5VW)

Limiting (N-16) Nozzle Forging USE			
% Cu	=	0.180	
54 EFPY Peak ID Fluence	=	2.89×10^{18} n/cm^2	
54 EFPY 1/4t Fluence	=	2.21×10^{18} n/cm^2	
R.G. 1.99 Predicted % Decrease	=	18.9	
Adjusted % Decrease	=	NA	R.G. 1.99, position 2.2
18.9% ≤ 23.5%, so nozzles are bounded by Equivalent Margin Analysis (EMA)			

RAI 4.2.8-2

Background

In Section 4.2.5 of the LRA, DAEC stated that N-2 nozzle and N-16 nozzle have high adjusted reference temperature (ART) values.

Issue

The projected ART values of the nozzle materials were reported in LRA Table 4.2.3-1 however, the nozzle weld materials' ART values are not listed in this table.

Request

The staff requests that the applicant submit the following information which will be used to evaluate whether the weld materials used in Nozzles N-2 and N-16 would have adequate toughness until the end of the extended period of operation:

- Weld metal chemistry
- American Welding Society weld electrode/filler wire classification and the heat/lot number
- Initial (un-irradiated) reference nil ductility transition temperature (RT_{NDT}) of the weld materials
- 1/4T neutron fluence value of the nozzle welds at 54 effective full-power years
- ART values for the nozzle weld materials.

DAEC Response to RAI 4.2.8-2

The weld metal chemistry, AWS weld electrode/filler wire classification, heat/lot number, initial (un-irradiated) RT_{NDT}, 1/4t neutron fluence values at 54 EFPY, and ART values for the nozzle weld materials are provided in Table 3.

Table 3 ART Evaluation for N-2 and N-16 Nozzle Welds

Description	Wire Type	Heat No.	Lot No.	Initial RT_{NDT} (°F)	Chemistry (Maximum Values) Cu (wt %)	Chemistry (Maximum Values) Ni (wt %)	Chemistry Factor (°F)	Adjustments for 1/4 t: ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_i (°F)	ART_{NDT} (°F)
N-2/N-16 Nozzle Welds (bounding)	E 8018-G	412Z051, 08R4818, 659T568, 661Y494, 661Y439	K910A27A, K904A27A, H721A27A, F927A27A, E916A27A	-50	0.03	1.00	41.00	24.34	12.17	0.00	-1.3

Fluence Data

Location	Wall Thickness (in) Full	Wall Thickness (in) 1/4t	Fluence at ID (n/cm²)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm²)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Nozzle N-16 (bounding)	4.469	1.117	2.89E+18	0.765	2.21E+18	0.5936

Notes:
1. Wire type, Heat No., Lot No. and chemistry from material-specific CMTRs for all weld materials used in the N-2 and N-16 nozzle welds.
2. Initial RT_{NDT} estimated based on GE Document F1006A006, Revision 1, "Methods for Establishing Initial Reference Temperatures (RT_{NDT}) for Vessel Steels for Certain Plants," as the higher of the NDT or Transverse CVN 50 ft-lb - 60°F. In this case all Charpy V-notch data from weld material CMTRs exceeded 50 ft-lbs at a test temperature of 10 °F, so the Initial RT_{NDT} was estimated as 10 - 60 = -50 °F.
3. Chemistry Factor, ΔRT_{NDT}, Margin Terms, and ART determined in accordance with NRC Regulatory Guide 1.99 Rev. 2.
4. Fluence Data obtained from Table 4.2.2-1 of the DAEC LRA.