January 15, 2009

Mr. Ronnie L. Gardner
AREVA NP Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: THIRD REQUEST FOR ADDITIONAL INFORMATION REGARDING ANP-10285P, "FUEL ASSEMBLY MECHANICAL DESIGN TOPICAL REPORT" (TAC NO. MD7040)

Dear Mr. Gardner:

By letter dated October 2, 2007, which can be accessed through the U.S. Nuclear Regulatory Commission's (NRC) Agencywide Documents Access and Management System (ADAMS) Accession No. ML072840180, AREVA NP submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) ANP-10285P, "Fuel Assembly Mechanical Design," ADAMS ML072840180. The first set of request for additional information (RAI) was issued by the NRC on April 29, 2008 (ML081080360), and the AREVA NP responses were received on May 29, 2008 (ML081560318) and June 13, 2008 (ML081560318). The second set of request for additional information (RAI) was issued by the NRC on June 24, 2008 (ML081640135), and the AREVA NP responses were received on July 24, 2008 (ML082100438).

The NRC staff's review has determined that some areas of this report require additional information in order to complete the review. The specific information requested is contained in the enclosure to this letter. The draft RAI was discussed with your staff in meetings on October 16, 2008, and November 4, 2008. AREVA NP has agreed to provide a response by March 31, 2009.

If you have any questions regarding this matter, I may be reached at 301-415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr., Project Manager
EPR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-020

Enclosure: Request for Additional Information

cc: DC AREVA – EPR Mailing List

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3rd REQUEST FOR ADDITIONAL INFORMATION (RAI)

ANP-10285P, "U.S. EPR FUEL ASSEMBLY MECHANICAL

DESIGN TOPICAL REPORT"

DOCKET NO. 52-020

RAI-31. The following questions are related to a concern with the use of the RODEX2-2A code for SBLOCA analyses, particularly at burnup levels greater than 10 GWd/MTU. Because the code does not have a burnup dependent fuel thermal conductivity model, the code may under-predict fuel stored energy and fuel temperatures, and over-predict fuel thermal conductivity. The fuel temperature data used to verify this code above 10 GWd/MTU was very limited and based on measurements from only three rods that are not prototypical of current PWR fuel designs, including EPR. Also, the NRC confirmatory calculations with the FRAPCON code have been completed as per the input and output as calculated by S-RELAP5 provided in RAI-29 for SBLOCA. The confirmatory calculations demonstrated that the fuel centerline temperature is over 350°F higher than that calculated with the RODEX2-2A code at a peak LHGR of 13.4 kW/ft near the end of cycle 1.

a. What is the difference between the RODEX2 code approved by NRC 25 years ago and the RODEX2-2A code being used for SBLOCA as discussed in response to RAI-29?

b. An example calculation is needed on the impacts of higher fuel stored energy, for example 350°F higher fuel centerline temperature, and lower fuel thermal conductivity due to degradation with burnup at moderate to high burnup levels on SBLOCA peak cladding temperatures.

c. Because this may be impacted by a burnup dependence on fuel thermal conductivity, what is the process used to select the burnup level at which the SBLOCA analyses are performed?

d. Is the RODEX2-2A code or its models used for determining fuel temperatures for other analyses besides SBLOCA?

RAI-32. The original approval for RODEX3 was limited to applications below a burnup of 10 GWd/MTU, however the application provided for LBLOCA is above this burnup level. The concern with the use of the RODEX3A code is that the code does not have a burnup dependent fuel thermal conductivity model, therefore it may under-predict fuel stored energy and over-predict fuel thermal conductivity for LBLOCA analyses, particularly at burnup levels greater than 10 GWd/MTU.

a. What is the difference between the RODEX3 code approved by NRC 13 years ago in ANF-90-145P Volumes 1 and 2 and the RODEX3A code being used for initializing LBLOCA as discussed in RAI-29? Provide the reference for the NRC approval of RODEX3A.

b. The correction for RODEX3A biased prediction of fuel centerline temperatures as stated in Section 4.3.3.2.1 of EMF-2103 is negative, or decreasing the predicted

fuel temperatures, as a function of burnup. This appears to be incorrect, because this should be a positive addition to fuel centerline temperature. If this interpretation is correct, the positive addition in EMF-2103 to fuel centerline temperature appears to be much smaller than the percent per 10 GWd/MTU under-prediction in fuel centerline temperature recommended in the 1995 technical evaluation report (TER) of RODEX3. In addition, the degradation in fuel centerline temperature near peak LOCA conditions based on Halden data and several other recent fuel performance codes including the Halden, FRAPCON-3.3, and a recently approved AREVA fuel performance code RODEX4, results in approximately a 3.4 percent increase in fuel centerline temperature per 10 GWd/MTU starting from zero burnup levels at constant power and gap conductance. This degradation is also significantly greater than the correction provided in EMF-2013.

c. No S-RELAP5 predictions of initial conditions for RLBLOCA were found in the response to RAI-29 as requested, such as predicted centerline and volume average fuel temperatures, gap conductance, and rod pressures at the LHGR limits. Provide the initial centerline and volume average fuel temperature calculated with the S-RELAP5 code for the hot rod at the hot axial node along with rod pressure and gap conductance at the limiting burnup level, such that a comparison can be made to those calculated with the NRC code. This will help address the concern in question 31.b above on whether the RODEX3A code under-predicts fuel temperature.

d. Because it will be impacted by a burnup dependence on fuel thermal conductivity, what is the process used to select the burnup level at which the LBLOCA analyses are performed? Why is there a difference in burnup level at which RLBLOCA and SBLOCA are evaluated at 21 GWd/MTU and 32 GWd/MTU rod average, respectively? The power decrease with burnup due to depletion should be the same for the same core, and power should be the main driver for limiting conditions for both events.

e. Provide the peak LHGR limit versus fuel burnup for the EPR core based on the depletion calculation.

f. Is the RODEX3A code or its models used for determining fuel temperatures for other analyses besides LBLOCA?

RAI-33. The proposed bounding history methodology discussed in response to RAI-21 appears to be reasonably conservative, as long as the same identical core loading or less aggressive loadings at moderate to high burnup are used for all future core reloads. For example, the bounding history will change with different fuel management patterns and these future patterns may have a higher bounding history. The current methodology requires that if the planned power history, based on calculations of expected fuel operation, exceeds the bounding history, then the planned history will be used for the rod pressure analysis. Of concern is whether this approach of using planned power histories will become the norm, rather than a bounding history for the rod pressure analysis, because planned histories do not account for calculational uncertainties and possible uncertainties due to differences between planned and actual operating histories. For example, the bounding history will become less conservative if future cores and reloads have multiple rods that

exceed this bounding history, or if plant power upgrades are introduced. In order to address this concern, discuss when a new bounding power history would be developed based on new depletion calculations for a given fuel management. This applies when more than one rod from a fuel batch or core exceeds the previously assumed bounding history or when the level of conservatism, between bounding and calculated, is reduced by a given amount.

RAI-34. The original RAI-21 requested the applicant to "identify those transients that are bounding for the Condition 1 and 2 events in terms of fission gas release and rod pressure." The limiting transients were not identified as requested. Identification of the limiting Condition 2 transient is of particular interest to determine if the assumed time at power for this event is appropriate, and to evaluate whether other Condition 2 events may be more bounding in terms of fission gas release.

RAI-35. Provide the best estimate, without fabrication or code and model uncertainties, of rod pressure prediction using the bounding power history and transients provided in RAI-21. The fuel average temperature provided in Figure 5-6 of the topical report appears to be the average temperature, averaged radially and axially over the rod. Also, provide the peak axial centerline temperature and peak axial node fuel average temperature, radially averaged as a function of rod average burnup, for this calculation.

RAI-36. The original RAI-23 requested comparison of various M5 models to M5 data collected since the approval of M5. However, no COPERNIC M5 creep model comparisons to diameter change, or cladding strain, data were provided at low to moderate burnups where cladding creepdown is active. Also, cladding oxidation data were provided, but the data was not compared to the COPERNIC corrosion model used to predict M5 cladding corrosion. Provide the model comparisons to M5 cladding diameter change data at low to moderate burnups, and M5 corrosion data up to the burnups requested. Also, explain how best estimate predictions of cladding corrosion are determined based on the COPERNIC corrosion model predictions for M5.

RAI-37. Is the UTL 95/95 curve in Figure 5-2 of the TR for water channel closure used in determining a penalty for DNBR as a function of burnup? If not, explain and justify why this UTL curve is not applied for determining DNBR for a given burnup.

RAI-38. The response to RAI-25 indicates that no assembly growth data are currently available that are directly applicable to the EPR design. This is of particular concern for two reasons, the first being that there is currently a small margin to the gap between top nozzle-to-core plate, which becomes closed at end-of life, resulting in fuel assembly bowing based on the original assembly growth model. The second concern is that past experience has demonstrated that AREVA has underestimated assembly growth with M5 guide tubes. Therefore, it is possible that an NRC limitation or condition will be established on axial assembly growth until further data are provided to NRC, or design changes are implemented to substantiate that gap closure is highly unlikely at the burnup level requested.

RAI-39. Table RAI 25-1 shows that the bottom of the fuel rods are not seated as in other AREVA fuel designs. Discuss the impact of the fuel rod shifting down due to fuel assembly handling, resulting in closure of the bottom gap and the downward shift of the fuel column on fuel performance. Relating to RAI-27, did the flow tests in

HERMES-P, the PETER Flow Loop, and the EOL Autoclave tests for flow induced vibration (FIV) and fretting wear account for the fact that the bottom of the fuel rods are not seated for the EPR design, thus considering no support at the bottom of the fuel rods?

RAI-40. The strain data provided in Table RAI 28-1 is not applicable to the one percent elastic plus uniform plastic strain limit in the SRP, or the EPR design limit that is meant to prevent brittle failure in fuel cladding. The strain data provided in RAI-28 is based on total elongation data that includes total plastic strain, and not uniform plastic strain as required by the SRP limit. Cladding failure due to brittle fracture has been observed in Garde et al. (1996) and Hermann et al. (2007) with measured total plastic elongations between 1 to 3.6 percent from seven separate specimens, including three axial tensile and four burst test specimens. These specimens failed in a brittle manner with failure below the yield strength, at 60 to 85 percent of yield strength. Therefore, the plastic strain should be near zero, suggesting that maintaining total plastic elongation above one percent will not prevent brittle behavior. The measured uniform plastic strains from the tests in Garde et al. (1996) and Hermann et al. (2007) were less than 0.5 percent. Provide irradiated M5 strain data in terms of either elastic plus uniform plastic strain, or uniform plastic strain, confirming that the EPR one percent elastic plus uniform plastic limit is met at the maximum fluence and hydrogen level expected for the EPR fuel design.

<u>Referenced papers:</u>

A. M. Garde et al., "Effects of Hydride Precipitate Localization and Neutron Fluence on the Ductility of Irradiated Zircaloy-4," *Zirconium in the Nuclear Industry: 11th International Symposium, ASTM STP 1295*, p. 407, American Society for Testing and Materials, Garmisch-Partenkirchen, Germany (1996).

A. Hermann, S. K. Yagnik, and D. Gavillet, Effect of Local Hydride Accumulations on Zircaloy Cladding Mechanical Properties, 15th International Symposium on Zirconium in the Nuclear Industry, American Society for Testing Materials, Sun River, Oregon (2007).

RAI-41. Some of the design input conditions between the COPERNIC and RODEX2-2A codes for SBLOCA analyses, and the RODEX3A code for RLBLOCA analyses are not the same for the EPR fuel design analyses. The design input conditions in question include rod pressure, cladding strain, and fuel melt analyses. For example, the input initial rod pressure for COPERNIC and RODEX2-2A was 304.7 psi while for the RODEX-3A analysis it was 290 psi. The rod densification for COPERNIC was 1.5 percent TD while for RODEX2-2A and RODEX3A was 2.0 percent TD. Also, the enrichment for RODEX2-2A was only 3 percent while for COPERNIC and RODEX3A it was 4.8 and 4.95 percent, respectively. Explain why these differences exist.

RAI-42. Due to the extremely small margin for the shipping loads for bottom and top nozzles, further details of these calculations are necessary. Provide the details, including any assumed conservatisms, of both the analyses, including mesh sizes and structure of the mesh assumed if FEA was used for these calculations.

RAI-43. The strain limit on the guide tube end plug appears to be related to the unirradiated condition that is applicable to reactor startup of a fresh fuel assembly, but the strain

limit is significantly less following irradiation. What is the plastic strain experienced during subsequent startups of this assembly after being irradiated?

RAI-44. PNNL has performed a cladding strain analysis with the FRAPCON-3.3 code using the input provided for the COPERNIC strain analysis. The results of the FRAPCON-3 analysis has demonstrated that at a burnup of 28.5 GWd/MTU a 3.3 kW/ft lower cladding strain limit, and at a burnup of 53.5 GWd/MTU a 1.9 kW/ft lower limit for the UO_2 EPR fuel rods than calculated by COPERNIC in Table RAI 22-4. For the UO_2-Gd_2O_3 EPR fuel rods the LHGR limit calculated by FRAPCON-3 was approximately 2 kW/ft lower at 32.7 GWd/MTU and only 0.2 kW/ft lower at 52.9 GWd/MTU than calculated by COPERNIC in Table RAI 22-6.

a. Provide the values of the LHGR limits used for AOOs for the EPR design above burnups of 26 GWd/MTU for the UO_2 and UO_2-Gd_2O_3 EPR fuel rods, respectively, if they are different from the COPERNIC LHGR values in Tables RAI 22-4 and 22-6. Do these values include the uncertainty in COPERNIC predictions of cladding strain based on the strain predictions?

b. If these values do not include uncertainties in COPERNIC strain predictions, justify the lack of including predictive uncertainties in COPERNIC based on the general under-predictions demonstrated in Figure 7-44 of BAW-10231P of the TRANSRAMP IV ramp data.

RAI-45. The RODEX3A model has been used to determine fuel center line temperature, gap conductance and fuel thermal conductivity in support of EPR LBLOCA. The staff is concerned that the RODEX3A code and the procedures described on Page 4-93 of EMF-2103 revision 0 may lead to underestimating the fuel centerline temperature. Provide appropriate responses to the following questions.

a. Is the fuel temperature database for the correlation used by AREVA for RODEX3A applicable to EPR fuel and most current fuel designs? If yes, provide an explanation of the database's applicability to the U.S. EPR design.

b. Is the expanded RODEX4 database for fuel temperatures more applicable to EPR fuel design? If so, has it been used to verify the RODEX3A results? Provide comparisons of these data in terms of the ratio of predicted-minus-measured divided by measured temperature and also the predicted-minus-measured temperature.

c. The RODEX4 code treats the fuel conductivity in different fuel regions, for example the rim and bulk regions, while the RODEX3A code does not. Discuss the impact of this difference in integral thermal conductivity between the two codes on initial stored energy.

EMF-2103 revision 0, "Realistic Large break LOCA for Pressurized Water Reactors." August 2001.