AREVA

July 24, 2009
NRC:09:077

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response to U.S. EPR Design Certification Application RAI No. 231

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Ronda Pederson (AREVA NP Inc.), "U.S. EPR Certification Application RAI No. 231," June 25, 2009.

In Reference 1, the NRC provided a request for additional information (RAI) regarding the U.S. EPR design certification application. Technically correct and accurate responses to 12 of the 14 questions are enclosed with this letter.

The following table indicates the respective page(s) in the enclosure that contain AREVA NP's response to each of the subject questions:

Question #	Start Page	End Page
RAI 231 – 04.03-19	2	2
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The schedule for technically correct and complete responses to the remaining RAI No. 231 questions is provided below:

Question #	Response Date
RAI 231 – 04.04-54	August 7, 2009
RAI 231 – 04.04-55	August 7, 2009

AREVA NP considers some of the material contained in the enclosure to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the enclosure to this letter are provided.

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NRO

If you have any questions related to this submittal, please contact me by telephone at 434-832-2369 or by e-mail at sandra.sloan@areva.com.

Sincerely,

Sandra M. Sloan, Manager
New Plants Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: G. Tesfaye
Docket No. 52-020

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Ronda M. Pederson. I am Licensing Manager, U.S. EPR Design Certification, Regulatory Affairs for New Plants for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the enclosure to NRC:09:077, "*Response to U.S. EPR Design Certification Application RAI No. 231*", and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 24th

day of July 2009.

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2011

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2011

Response to

Request for Additional Information No. 231

6/25/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 04.03 - Nuclear Design
SRP Section: 04.04 - Thermal and Hydraulic Design
Application Section: FSAR Ch. 4

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

Question 04.03-19:

In reference to RAI 134, Question 04.03-10: Summarize the current fuel manufacturing practices that reduce in-reactor densification of US EPR fuel material.

Response to Question 04.03-19:

The main difference between current fuel manufacturing practices and those used when in-core densification was a concern is that pellets currently have a much higher initial density, in the range of 95 to 96 percent theoretical density (TD) versus the 92.5 percent TD that was typical of former pellets. Pellets can be manufactured at a higher density because of an increased understanding and control of the ceramic microstructure. This includes grain size/structure and pore size distribution by the use of additives and pore formers, as well as the ability to press pellets at higher pressures and sinter pellets at higher temperatures for a longer period of time.

A review of the various densification models used within the industry in the late 1970s is presented in NUREG-0085, "The Analysis of Fuel Densification," published in 1976. This document also established the technical basis for the resinter test. The document contains post-irradiation examination data from EPRI and other sources within the industry that support the use of the resinter test as a means of assessing in-pile densification behavior. The resinter test data from manufacturing provides confidence that modern manufacturing processes eliminate the potential for large in-reactor densification.

Because the pellets have been sintered for longer durations at a higher temperature, the pellets typically exhibit much lower in-reactor densification. This conclusion is supported by the pellet densification measured in the resinter tests performed on the pellets per NRC Regulatory Guide 1.126, which is designed to be representative of maximum in-reactor densification. In other words, the densification behavior that had been previously seen in the reactor for lower density fuel is now taking place during the sintering operation of the manufacturing process.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.03-20:

In reference to RAI 134, Question 04.03-10: Summarize the gamma scan fuel column gap measurements taken at the McGuire station, including identification of the maximum measured pellet stack gap and explanation of whether the measurements represent the burnup at maximum expected net densification.

Response to Question 04.03-20:

Three fuel rods from a 17x17 Mark-BW fuel assembly irradiated at the McGuire station to an exposure of approximately 42 MWd/kgU were examined with gamma scans to identify pellet stack gaps. Two of the three rods exhibited no gaps, and one rod had one pellet gap measuring 0.1 inch. The fuel column lengths were 145.3 in, 144.6 in, and 145.4 in (corrected for the gap), relative to the initial stack length of 144 in. The differences in irradiated stack lengths were not indicative of differential fuel rod growth, but the fact that the fuel columns are greater than as-built nominal indicate that they have surpassed the regime of maximum net densification and have entered into the regime of net swelling. This is expected since models such as those used in COPERNIC predict the maximum net densification to have occurred at very low burnup, less than 5000 MWd/mtU. However, the size of the gap produced even by the upper limit of densification would be very small and quickly overcome by the effects of swelling.

Figure 04.03-20-1 from the NRC-approved topical report for COPERNIC (BAW-10231PA), shows the model for pellet densification and swelling.

Figure 04.03-20-1—[ ] Densification Model

[ ]

Another important consideration when evaluating hot cell examinations of pellet stack gaps is whether any gaps seen in fuel rods measured at room temperature are indicative of gaps in the reactor. Because of thermal expansion of the fuel stack at operating temperatures, which is about 0.4 in (or four times the maximum gap seen in the McGuire examination), it is reasonable to expect that gaps may open up in the fuel column at some point between in-reactor operation and hot-cell examination, particularly at higher burnup where significant swelling has occurred and the plenum spring has experienced significant relaxation. However, the 0.1 inch maximum gap seen in the post-irradiation study may be considered conservatively bounding of the types of gaps that could potentially manifest themselves under operation at higher burnup. Furthermore, the 0.1 inch gap is within allowable tolerances, as analyses have shown that 0.1 inch pellet stack gaps are acceptable from a neutronic and fuel performance perspective.

Moreover, the 1985 EPRI Report (NP-3966-CCM) "Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods" reached the following conclusion:

"...modern, prepressurized fuel-rods loaded with nondensifying UO2 fuel pellets, are considered to be resistant to interpellet-gap formation and clad collapse. Therefore, the penalty for densification caused augmented power peaking should be removed from the analysis and licensing requirements of any reactor loaded exclusively with fuel-rods of such modern design"

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.03-22:

In reference to RAI 134, Question 04.03-14: Explain how the AMS flux mapping process performs a comparison of measured versus calculated power distributions.

Response to Question 04.03-22:

The aeroball measurement system (AMS) flux mapping process uses measured and calculated power reaction rate data in the three-dimensional inferred power distribution map process. The AMS flux mapping process is described in more detail in ANP-10287P (Reference 5 of U.S. EPR FSAR Tier 2, Section 4.3.5).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-48:

In reference to RAI 134, Question 04.04-24: The Question Response states that because the BWU-N correlation is used for the bottom-most spacer grid, the bundle length is not as important. Explain this statement, considering the BWU-N correlation is applicable to the HMP spacer design, and that the fuel assembly upper-most spacer is a HMP design spacer.

Response to Question 04.04-48:

As described in U.S. EPR FSAR, Tier 2, Section 4.4.4.1.1 the BWU-N correlation is applied downstream of the high mechanical performance (HMP) structural grids. The top-most HMP structural grid, however, is positioned above the top of the heated length, where the departure from nucleate boiling ratio is not calculated. Therefore, while it is accurate to state that the BWU-N correlation is applicable to both HMP structural grids, in practice it only needs to be applied to the bottom-most HMP structural grid. U.S. EPR FSAR, Tier 2, Section 4.4.4.1.1 will be reworded to improve clarity.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 4.4.4.1.1 will be revised as described in the response and indicated on the enclosed markup.

Question 04.04-49:

In reference to RAI 134, Question 04.04-26: Identify the design basis accident conditions evaluated to ensure that the top and bottom nozzles maintain engagement.

Response to Question 04.04-49:

The assembly liftoff design criterion is described in Reference 1, Section 5.1.9.

Section 5.1.9 of Reference 1 states, "The U.S. EPR fuel holddown springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating AOOs, except for the pump overspeed transient. The fuel assembly shall not compress the holddown spring to solid height for AOOs. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all AOOs and design basis accidents..."

The limiting anticipated operational occurrence (AOO) is described as the 120 percent over-speed transient. This AOO is not considered credible for the U.S. EPR but is a bounding flow condition of all other AOOs. It was therefore selected as the reference condition to demonstrate compliance with criteria. The limiting Design Basis Accident for fuel assembly liftoff is the vertical Safe Shutdown Earthquake condition. Reference 1 states that these conditions have been met for the U.S. EPR fuel assembly design.

References for Question 04.04-49:

1. ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report," October 2007.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-50:

In reference to RAI 134, Question 04.04-26: Explain how the statistical holddown methodology addresses the condition of relaxed holddown spring to demonstrate that holddown capability is not exceeded for the worst case loads during normal operation.

Response to Question 04.04-50:

To address the condition of relaxed hold-down springs, the statistical hold-down methodology allows for the percent relaxation to be input as a nominal value and a single-sided tolerance. The fast neutron fluence is calculated above the fuel stack to the average spring height. The hold-down spring relaxation, as a function of fluence, is determined from experimental results for irradiated spring material Inconel 718. The percent relaxation nominal value and tolerance depend on the statepoint and the fuel design being analyzed. The percent spring relaxation is calculated for each statepoint under consideration, typically at the beginning of life, the end of each cycle, and the end of life. According to Reference 1, the statistical hold-down methodology is used to arrive at the relaxed spring deflection and the worst case hold-down spring loads during normal operation. The results indicate that the hold-down capability will not be exceeded during normal operation.

References for Question 04.04-50:

1. BAW-10243PA, "Statistical Fuel Assembly Hold Down Methodology," September 2005

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-51:

In reference to RAI 134, Question 04.04-27: Identify the fuel manufacturing tolerances that are explicitly treated in the COPERNIC analysis of the US EPR fuel.

Response to Question 04.04-51:

Similar information was provided in the response to RAI-21 (Reference 1) for ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Topical Report". Reference 1 contains more details regarding the model and the methodologies used for the COPERNIC (Reference 2) analysis of the US EPR fuel rod performance.

The calculation of the bounding fuel rod internal gas pressure explicitly treats the manufacturing tolerances as required by the NRC approved COPERNIC fuel performance methodology. Other analyses employ nominal fuel rod characteristics along with appropriate uncertainties.

The allowances that are included for computer code modeling uncertainty and manufacturing tolerances with respect to the internal gas pressure and fission gas release are:

The approved NRC methodology (Reference 2) using the COPERNIC performance code does not require consideration of manufacturing tolerances for the predictions of other fuel rod performance limits and parameters. However, appropriate calculational uncertainties are applied for the other analyses.

References for Question 04.04-51:

1. ANP-10285Q4P, "Response to Third Request for Additional Information – ANP-10285P 'U.S. EPR Fuel Assembly Mechanical Design Topical Report' "

2. BAW-10231PA Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-52:

In reference to RAI 134, Question 04.04-27: Provide the calculated effect of each of the fuel manufacturing tolerances on the key fuel rod performance parameters, e.g., internal rod pressure, fuel pellet temperature, gap conductance, and cladding stress level.

Response to Question 04.04-52:

Quantitative assessment for rod internal gas pressure

The calculation of the bounding rod internal gas pressure uses the manufacturing tolerances specified in the Response to Question 04.04-51. Those parameters affect the fission gas release, the free gas volume, and the rod internal gas pressure. This is consistent with the NRC approved topical report for COPERNIC (Reference 1). The output from the COPERNIC calculation for the bounding rod internal gas pressure evaluations for the limiting fuel rod is provided in Table 04.04-52-1. This calculation is for the same UO_2 fuel rod in the 18-month equilibrium cycle shown in Figure 5-7 of the ANP-10285P Topical Report. The magnitudes of the parameters throughout the [ ] are presented in Figure 04.04-52-1 and Figure 04.04-52-2. The pressure impact of the parameters (including manufacturing tolerances) used in the bounding pressure calculation at ~62,000 MWd/tU for each parameter are provided in Table 04.04-52-1.

Qualitative assessment of other parameters

The approved NRC methodology (Reference 1) using the COPERNIC performance code does not require consideration of manufacturing tolerances for the predictions of other fuel rod performance limits and parameters. However, appropriate calculational uncertainties are applied for the other analyses. The impact of manufacturing tolerances on thermo-mechanical parameters such as gap conductance, fuel temperature, and clad strain is more difficult to quantify. Most of the computer code correlations and mechanistic behaviors have responses that are not singularly linear due to the interactions of temperature and burnup in such models as swelling and creep. Dimensional tolerances have little effect on the centerline and average fuel temperatures since the temperature rise is proportional to the linear heat rate and effective thermal conductivity of the fuel. The rise will follow the normalized radius. The fuel thermal conductivity is not affected by the rod dimensions, only the temperature and the burnup as a function of pellet radius.

Reference for Question 04.04-52:

1. BAW-10231PA Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004

Table 04.04-52-1—Bounding Pressure Penalties due to Code Uncertainties and Manufacturing Tolerances for 18 Month Cycle UO_2 Fuel Rod

Figure 04.04-52-1—Sample Fuel Rod Internal Pressures with and without Uncertainty Penalties

Figure 04.04-52-2—Sample Fuel Rod Internal Gas Pressure Penalties Due to Manufacturing and Modeling Uncertainties and Tolerances

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-53:

In reference to RAI 134, Question 04.04-27: Describe how pellet chipping is treated.

Response to Question 04.04-53:

The NRC approved COPERNIC code (Reference 1) does not explicitly account for pellet chipping in the fuel performance analyses. That is, no credit is taken for the increased free volume for rod internal pressure licensing analyses, and COPERNIC does not have mechanical models which account for the effects of chipping in the transient cladding strain licensing analyses. Chipping is expected to have the greatest impact on cladding strain if a pellet chip is wedged between pellet and cladding or a cladding surface is unsupported due to a missing pellet surface. Best practice improvements to manufacturing practices in recent years have seen order of magnitude improvements in the elimination of large missing pellet surfaces in fuel rods, and since this time no fuel failures have been attributed to missing pellet surface. Finite element calculations have determined that the reduction in the cladding failure threshold due to a missing pellet surface below AREVA NP's manufacturing specification requirement is bounded by the inherent margin of the operating guidelines based upon AREVA NP's ramp test database.

Reference for Question 04.04-53:

1. BAW-10231PA Revision 1, "COPERNIC Fuel Rod Design Computer Code," January 2004

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-54:

In reference to RAI 134, Question 04.04-29: The Question Response states that while fuel rod surface crud slightly reduces cladding surface heat transfer and increases cladding surface temperature, the probability of a steam film between the rod surface and coolant does not increase. Explain why an increase in cladding surface temperature does not increase the probability of the formation of a vapor film.

Response to Question 04.04-54:

A response to this question will be provided by August 7, 2009.

Question 04.04-55:

In reference to RAI 134, Question 04.04-29: Describe how the effects of crud are accounted for in the DNB correlations.

Response to Question 04.04-55:

A response to this question will be provided by August 7, 2009.

Question 04.04-56:

Summarize the applicability of the Juliette test loop to the US EPR, including physical configuration, scale, coolant mass flow, coolant system temperature and pressure, and core representation. This question is a follow-up item from the reactor systems audit held in Lynchburg on April 21 to April 24, 2009.

Response to Question 04.04-56:

The JULIETTE test mockup in the LeCreusot facility represents the following (at a scale of five to one):

- The four cold leg pipes:
  - Bends in cold legs preceded by a straight section long enough to obtain centered, uniform rotation upstream of each bend.
  - Removable twisted tapes used to simulate flow rotation induced by the reactor coolant pumps.
  - The reactor pressure vessel inlet nozzles.
- The full height of the downcomer annulus and all included structures (e.g. outlet nozzles and radial support keys).
- The RPV bottom head.
- The flow distribution device with all connecting parts.
- The core support plate with increased head loss coefficients that account for the effect of the core on flow distribution through the bottom core support structure. These loss coefficients represent the following losses:
  - The lower core plate.
  - The bottom nozzle.
  - The first support grid.
  - Part of the fuel rod bundle.
  - The first mixing grid.

Geometrical data for the JULIETTE test mockup are as follows:

The following approximate ranges of test conditions are covered:

The JULIETTE test mockup is representative of the U.S. EPR reactor vessel internals at a scale of five to one and includes all features that are considered to have a significant impact on flows, including the downstream effects of the core. The testing was conducted over a range of Reynolds numbers, by varying flow rates, and it was demonstrated that the Reynolds number has a negligible effect on the inlet flow distribution. Therefore, the inlet flow distributions from the JULIETTE testing are considered applicable for the U.S. EPR reactor.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-57:

Provide a comparison of the core inlet flow distributions between the US EPR and previous design 14-foot cores that had experienced fuel assembly. This question is a follow-up item from the reactor systems audit held in Lynchburg on April 21 to April 24, 2009.

Response to Question 04.04-57:

A comparison of the core inlet flow distributions for 14-foot cores is presented in Figure 04.04-57-1. The core inlet flow distributions for the central region of the core (less than approximately [ ]) are determined as the [ ]. As seen in Figure 04.04-57-1, the U.S. EPR core inlet profile has a maximum peak to average flow of approximately [ ], whereas the N3 and N4 plants have a maximum of peak to average flow of [ ], respectively.

Figure 04.04-57 – 1: Core Inlet Flow Distributions for N3, N4, and U.S. EPR

[ ]

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 04.04-58:

Assess the effects of the FDD and US EPR fuel assembly design improvements relative to flow-induced fuel assembly bow. This question is a follow-up item from the reactor systems audit held in Lynchburg on April 21 to April 24, 2009.

Response to Question 04.04-58:

The current operating experience (OE) of French 14-foot cores led to the identification of limited instances of grid to rod fretting (GTRF) and fuel assembly distortion. AREVA NP has extensive experience with the French 14-foot cores, both N3 and N4 variants (N3 plants are at 1300 MWe with 193 fuel assemblies; N4 plants are at about 1400 MWe with 205 fuel assemblies).

AREVA NP has been successful in improving performance of 14-foot fuel through fuel design changes. AREVA NP has corrected both the limited events of GTRF fuel rod failures and incomplete rod insertion that have historically affected some 14-foot cores. Additionally, the underlying mechanisms of the 14-foot core performance issues have been under review for some time. The GTRF issue is related to the highly peaked inlet flow conditions specific to the current 14-foot 4-loop cores.

The 14-foot 4-loop cores are known to have higher core flow rates and higher crossflow rates compared to their 12-foot core counterparts. The core hydraulic conditions play an important role in the historically experienced GTRF fuel rod failures at some 14-foot plants due to a large centrally peaked flow distribution which redistributes rapidly above the lower fuel assembly nozzles. This high crossflow region created elevated vibrations on the lower fuel rod region of the AFA3G fuel. A design change was made to add a second lower-end grid "twin grid" which stabilized the lower fuel rod region and eliminated the GTRF failures. The flow distribution differences between the French N3, N4, and the U.S. EPR plants are best illustrated in the Response to RAI 04.04-57. Though the N3 and N4 plants are similar, the N3 plants have experienced the majority of GTRF fuel rod failures at the intermediate ring of fuel assemblies where radial velocities are the highest due to this centrally peaked inlet flow distribution.

Core hydraulic conditions may also play an important role in the fuel assembly bow performance through the collective fuel assembly bow patterns. Such patterns can be observed at all pressurized water reactor plants. The reference to fuel assembly bow patterns illustrates that the entire core of fuel assemblies can bow in a manner that is not random but is affected by environmental flow conditions. The influence of core hydraulics on fuel assembly bow can not be specifically quantified at this time and, although the 14-foot fuel assembly performance has largely been addressed with fuel design changes, a flow distributor device (FDD) was implemented in the U.S. EPR design to reduce the hydraulic influence of flow distribution.

The FDD for the U.S. EPR was designed such that the U.S. EPR reactor inlet flow distribution would be similar to or better than that of the French N3 and N4 plants. The objective is to remain within the current operating experience of the French 14-foot reactor fleet. The FDD structure is designed to promote redistribution of the lower plenum flow field before approaching the lower core support plate. The FDD reduces the centrally peak flow from [ ] in the N3 plant to [ ] in the U.S. EPR. The FDD was tested and verified as discussed in the response to Question 04.04-56. The FDD represents a defense-in-depth approach by addressing a contributing cause of GTRF through plant design improvements. With respect to

fuel assembly bow, the benefit of the FDD can not be quantified. However, any improvement in balancing the inlet flow distribution will improve the optimization of axial loads (holdown and hydraulic) and reduce the potential for adverse hydraulic loading conditions.

The U.S. EPR fuel assembly design incorporates those specific design features that have been implemented and demonstrated to improve fuel performance at the currently operating French 14-foot plants. These proven mitigation features represent the most robust configuration of fuel components available. The notable features are MONOBLOC guide tube, lower high mechanical performance (HMP) (inconel) end grid, high thermal performance (HTP) intermediate grids, and FUELGUARD lower debris filter. The "twin grid" design used in the French plants is not used, and it is not necessary, because of the use of the inconel HMP lower end grid. The HMP is a highly robust and proven design. AREVA NP has experience in the U.S. with addressing high crossflow environments at B&W and CE plants. B&W plants have high cross flows at the loss of coolant accident holes and slots. CE plants experience high cross flows at the core inlet and outlet on the periphery of the core. The historical GTRF failures at the B&W and CE plants are being successfully managed by the use of HTP grids and the inconel HMP lower end grids. Full cores of the Mark B-HTP fuel assembly design are in place at B&W plants with great success and CE HTP fuel continues to demonstrate excellent fretting resistance when inconel lower HMP end grids are incorporated.

The assessment of the FDD and U.S. EPR fuel assembly design improvements relative to flow-induced fuel assembly bow has been provided. AREVA NP has incorporated lessons learned from the extensive experience with the French 14-foot reactors into both the U.S. EPR plant design (FDD) as well as incorporated proven fuel design features on the U.S. EPR fuel assembly design (MONOBLOC guide tube, HMP end grid, HTP spacer, and FUELGUARD lower debris filter) providing a defense-in-depth strategy.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

U.S. EPR Final Safety Analysis Report Markups

4.4.3.5 Load-Following Characteristics

Load follow using control rods and boron dilutions or additions are described in Section 4.3.2.4.15.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

A summary of the thermal and hydraulic characteristics of the reactor is presented in Table 4.4-1.

4.4.4 Evaluation

4.4.4.1 Critical Heat Flux

4.4.4.1.1 CHF Correlations

04.04-48

Two CHF correlations are applied to the high thermal performance (HTP) fuel assemblies for the U.S. EPR. The ACH-2 correlation (Reference 7) is applied downstream of HTP mixing grids and the BWU-N correlation (Reference 8) is applied downstream of the high mechanical performance (HMP) structural grids. The top-most HMP structural grid resides outside of the heated length of the fuel where no DNBR calculations are performed. Therefore, the BWU-N correlation is only applied downstream of the bottom-most HMP structural grid.

The ranges and limitations of the ACH-2 CHF correlation and the BWU-N CHF correlation are presented in Reference 7 and Reference 8, respectively.

4.4.4.1.2 Definition of DNBR

The DNBR for both a typical and thimble cell is defined as:

$$DNBR = \frac{q''_{CHF}}{q''_{local}}$$

Where:

$$q''_{CHF} = \frac{q''_{correlation}}{F}$$

q''_{CHF} =predicted critical heat flux

$q''_{correlation}$=critical heat flux from CHF correlation (ACH-2 or BWU-N)

F=nonuniform axial heat flux factor

q''_{local}=actual heat flux