Response to

Request for Additional Information No. 195 (2259), Revision 0

03/24/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 04.02 - Fuel System Design
Application Section: 4.2

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

Question 04.02-3:

FSAR Section 4.2.1.2.2 states that fuel densification and fission product swelling are addressed in AREVA NP topical report BAW-10231P-A, "COPERNIC Fuel Design Computer Code," January 2004. The topical report provides a description of the fuel densification and swelling model along with COPERNIC analysis results and comparison to measured data. Fuel densification (with solid swelling) results are presented for the Integrated Dry Route (IDR), Ammonium Di-Uranate (ADU) wet, and Ammonium Uranyl Carbonate (AUC) wet conversion processes.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, analysis results specifically applicable to the U.S. EPR fuel are required.

Provide evaluation results showing the fuel densification (with solid swelling) as a function of fuel rod burnup for the U.S. EPR fuel, including the appropriate initial pellet density (% TD) and uranium material conversion process. Assess the effect of fuel pellet Gd_2O_3 content up to the 8 weight percent design limit.

Response to Question 04.02-3:

The COPERNIC (Reference 1) analyses for AREVA NP's U.S. fuel, including the U.S. EPR fuel, use the IDR/ADU densification model. AUC fuel is not used commercially in the U.S.

The initial nominal U.S. EPR pellet density is 96% of theoretical. Figure 04.02-03-1 shows the progression of local fuel pellet average density versus burnup for a representative pellet near the axial center of the rod (elevation 2231 mm from the bottom of the fuel stack). Also depicted is the local linear heat generation rate. The data are derived from the rod internal pressure analysis of the highest burnup rod from the 18-month equilibrium cycle (Figures 5-5 – 5-7 of Reference 2).

As described in Section 6.2.4.2 of the COPERNIC topical report (Reference 1), **[**

] This is justified based upon experimental observations and literature.

Figure 04.02-03-1—UO_2 Pellet Average Density and Local Linear Heat Rate

FSAR Impact:

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

References:

1. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

2. ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report,” October 2007.

Question 04.02-4:

Section 4.2.1.3.2 of the FSAR states that the possibility of fuel rod cladding creep collapse is precluded by high theoretical density as-fabricated pellets which exhibit very low densification during operation and by manufacturing process inspections to prevent fuel column gaps.

ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Report" addresses fuel rod cladding collapse, referencing an NRC-approved methodology (BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5TM) in PWR Reactor Fuel") and stating that the creep rate for the M5TM cladding is approximately 50% slower than for Zircaloy-4 and that the fuel rod creep collapse lifetime is greater than the fuel rod design burnup limit of 62 GWD/MTU. NRC staff's February 4, 2000 safety evaluation of BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5TM) in PWR Reactor Fuel," however, discusses a disparity between the earlier Zircaloy-4 creep model (with M5TM adjustment factor) and a new M5TM creep model.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, additional information is required.

Provide the following information relative to the above discussion on fuel rod cladding creep:

a. Clarify which creep model is applied to the U.S. EPR fuel, and verify model applicability to the fuel rod burnup range from 0 to 62 GWD/MTU.

b. Confirm, clarify, or revise the statement given in ANP-10285P, "U.S. EPR Fuel Assembly Mechanical Design Report" that the M5™ creep rate is 50% slower than for Zircaloy-4.

c. Confirm the conclusion that the U.S. PER fuel rod creep collapse lifetime is greater than the fuel rod design burnup limit of 62 GWD/MTU.

Response to Question 04.02-4:

a. The creep model used for the creep collapse calculation is the Zircaloy-4 creep model multiplied by a factor of 0.9 for conservatism. This model applies to the analysis of the creep collapse lifetime up to the burnup limit of 62 GWd/mtU.

b. Appendix I of Reference 1 states that the M5® creep rate is 67% slower than the Zircaloy-4 creep rate, rather than the 50% value reported in Section 3.7 of Reference 1. Section 5.2.2 of ANP-10285P (Reference 2) will be revised to state that the creep rate of M5® is approximately 67% of Zircaloy-4, and that a multiplier of 0.67 or higher (for conservatism) can be applied to the CROV input for the M5® analysis.

   In Section 4.2.3.1.11 of the U.S. EPR FSAR, Tier 2, it is also incorrectly stated that the M5® creep rate is 50% slower than the Zircaloy-4 creep rate and that a multiplier of 0.5 or higher can be applied in the CROV input for the M5® analysis. This statement will be corrected in the U.S. EPR FSAR, Tier 2, Section 4.2.3.1.11.

c. The creep collapse lifetimes for power histories that are bounding for 18 and 24 month cycles are greater than 62 GWd/mtU.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 4.2.3.1.11 will be revised as described in the response and indicated on the enclosed markup.

References:

1. BAW-10227PA Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003.

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

Question 04.02-5:

In order for the staff to complete its evaluation of the U.S. EPR fuel assembly structural design relative to the requirements of GDC 10, additional information regarding nonoperational loading of the U.S. EPR fuel assembly is required.

Identify the nonoperational loads, and provide the basis for the axial and lateral loads that are given in FSAR Section 4.2.1.5.1.

Response to Question 04.02-5:

The non-operational loads given in U.S. EPR FSAR, Tier 2, Section 4.2.1.5.1 are shipping and handling load limits established to prevent fuel assembly damage. Fuel assemblies are monitored during shipping, handling, and storage to verify that these limits are not exceeded. Design analysis and testing verifies the ability of fuel assemblies to withstand loads up to these limits without permanent set or yielding of any assembly component. The 4g axial load is based on limiting the axial motion of the fuel column within the fuel rod assembly, which is constrained by the plenum spring, and limiting the fuel rod slip through the spacer. Similarly, the 6g lateral load limit is based on limiting the deflection of the spacer springs.

FSAR Impact:

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

Question 04.02-6:

In order for the staff to complete its evaluation of the U.S. EPR fuel assembly structural design relative to the requirements of GDC 27 as it relates to the capability of the reactivity control systems, additional information is required.

Justify the use of maximum shear stress theory to evaluate the U.S. EPR fuel assembly guide tube design and provide the evaluation results for the U.S. EPR fuel assembly guide tube.

Response to Question 04.02-6:

Reference 1, Section 5.1.1 states: “Stress intensities for U.S. EPR fuel assembly components shall be less than the stress limits based on ASME Code, Section III criteria.”

The use of stress intensity (SI), also referred to as the Maximum Shear Stress Theory, or Tresca Stress, is specified by the ASME Code, Section III, Subsection NG.

In addition to safety margins based on SI, additional safety margins against guide tube buckling ensure component stability. Specifically, the guide tubes are shown not to buckle and remain elastic, thereby ensuring control rods can be inserted, and meeting the design requirements of general design criteria (GDC) 27.

Reference 1, Section 5.1.1.2 addresses the guide tubes and provides margins for the evaluated results for anticipated operational occurrences. Reference 1, Table 5-17 provides the safety margins for the guide tubes for faulted conditions.

FSAR Impact:

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

Reference:

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

Question 04.02-7:

Section 4.2.1.5.5 of the FSAR states that the design bases for fuel assembly holddown springs is to maintain fuel assembly contact with the lower core plate, except during a reactor coolant pump overspeed transient. This does not meet the design acceptance criterion of SRP Section II.1.A.vii. The staff notes, however, that its safety evaluation of BAW-10239P-A, Revision 0, “Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report” found the pump overspeed transient exception acceptable for the advanced Mark-BW design.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR control rod (RCCA) design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to insertability of control rods.

Justify applicability of the above-described exception to the SRP Section II.1.A.vii acceptance criterion for the U.S. EPR fuel design with the bottom FUELGUARD™ nozzle. Specifically, assess the effect of the FUELGUARD™ nozzle on assembly liftoff.

Response to Question 04.02-7:

The Standard Review Plan (SRP) 4.2 Section II.1.A.vii states: “These worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs).” The U.S. EPR fuel assembly complies with this criterion for all normal operating conditions.

SRP 4.2 Section II.1.A.vii also states: “an evaluation of worst-case hydraulic loads should be performed for normal operation, AOOs, and accidents.” It has been determined that a 120% pump over-speed transient bounds all anticipated operational occurrences (AOOs) for hydraulic loads, and that the fuel assembly will not be damaged during this condition. SRP 4.2 Section II.1.A.vii does not state that the fuel assembly should remain in contact with the lower core support plate during AOOs.

The fuel assembly mechanical compatibility evaluation verifies that the bottom nozzle remains engaged with the lower core pins during AOOs and accidents. This continued engagement demonstrates that the control rods will be able to be inserted, and thus demonstrates compliance with general design criteria (GDC) 27.

Both the evaluation of liftoff margin for normal operations and the evaluation of hydraulic loads induced by AOOs consider the pressure drop characteristics of the FUELGUARD™ bottom nozzle. Pressure drop coefficients are determined by flow testing.

FSAR Impact:

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

Question 04.02-8:

Sections 4.2.1.7 and 4.2.4.6 describe the post-irradiation surveillance program for the U.S. EPR fuel and the RCCAs.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 and relative to the guidelines of SRP Section II Acceptance Criteria 4.C. as they relate to post-irradiation surveillance of the U.S. EPR fuel, additional information is required.

What actions are taken when: (1) plant instrumentation detects a fuel rod failure, and (2) where post-irradiation fuel examinations indicate unusual fuel system behavior. Include a description of the actions that may be taken, the disposition of failed fuel, and the root cause analyses that may be performed.

Response to Question 04.02-8:

The actions taken in response to indications of fuel rod failures or unusual fuel system behavior are specific to each situation. The actions are taken in conjunction with the plant owner. A description of the actions that may be taken is provided below.

If reactor coolant activities indicate a fuel rod leak has occurred, the indications are documented and tracked by the utility throughout the remainder of the cycle. An Operating Experience (OE) report is prepared by the utility and issued to the industry. Typically, follow-up investigations are planned based on the estimated rod burnup of the leaker(s). If a failure is detected, AREVA NP enters the failure event into it's corrective action program (WebCAP) and begins planning with the utility to remove the failed fuel rod(s) during the next refueling outage. The plant owner and AREVA NP decide whether or not to examine the failed fuel rod to determine the failure mechanism, either during the outage or shortly thereafter.

If post-irradiation examinations indicate unusual fuel system behavior, AREVA NP typically responds in three phases. Each situation is unique and the specific actions are tailored to the circumstances and taken in conjunction with the plant owner.

1. Document the behavior

   Document the unusual fuel system behavior using AREVA NP's WebCAP reporting system. Because the judgment of “unusual” behavior is subjective and is often based on experience, some discussions and investigations may take place prior to documentation.

2. Investigate the behavior

   The investigation may include an apparent cause or root cause analysis. The investigation scope includes some or all of the following elements: detailed examination of post-irradiation inspection evidence, a review of the manufacturing records, examination of archive materials from the batch, out-of-pile testing of the design, review of design documentation and evaluations, or other actions considered necessary.

3. Take compensatory and corrective actions as necessary.

   Since the response phases may overlap, corrective actions may be implemented as the investigation identifies actionable information.

The above descriptions are intended to be representative of the actions that may be taken in response to indications of fuel failures or unusual fuel system behavior. Each situation is unique and the specific actions are tailored to the situation.

FSAR Impact:

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

Question 04.02-9:

FSAR Table 4.2-1 provides a summary of the U.S. EPR fuel assembly design and lists the fissile enrichment to be less than or equal to 4.95 weight percent U^{235}. Section 3.2 of ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report” states that pellet enrichments used for the U.S. EPR may be as high as 5 weight percent U^{235}.

FSAR Figure 4.2-2 shows the U.S. EPR fuel rod assembly with a support tube to end of fuel rod dimension of 5.315 inches. Figure 3-3 of ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report” shows the same parameter dimension to be 2.048 inches.

Resolve the noted discrepancies and correct the documents as appropriate.

Response to Question 04.02-9:

U.S. EPR FSAR, Tier 2, Table 4.2-1 states that the weight percent of fissile enrichment is “less than or equal to 4.95 weight percent U^{235}.” This is the maximum enrichment AREVA NP uses for all currently fabricated fuel (under current manufacturing tolerances) to maintain the licensed limit of 5.00 weight percent U^{235}. This value of 4.95 weight percent U^{235} is consistent with the value shown in Table 3-6 of ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report.” The value of 5.00 weight percent U^{235} stated in Section 3.2 of ANP-10285P defines the actual license limit for fissile enrichment. The lower value may be changed if the fabrication tolerance is changed, but the sum of the maximum enrichment plus tolerance must be less than or equal to the 5.00 weight percent U^{235} license limit. The value for the maximum fissile enrichment will be changed in the U.S. EPR FSAR, Tier 2, Table 4.2-1 to be less than or equal to 5.00 w/o U-235.

The dimension shown in U.S. EPR FSAR, Tier 2, Figure 4.2-2 of 5.315 inches from the bottom of the fuel column to the bottom of the fuel rod is correct. The value shown in ANP-10285P, Figure 3-3 of 2.048 inches is incorrect, and should be 5.315 inches. (Figure 3-3 will be corrected when the approved version of the topical report is issued.) This is consistent with the support tube length **[ ]** and end plug length **[ ]** shown in ANP-10285P, Tables 3-11 and 3-12, respectively.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 4.2-1 will be revised as described in the response and indicated on the enclosed markup.

Question 04.02-10:

Section 4.2.2.9 describes a control rod end plug flex joint, designed to accommodate misalignment of the RCCA control rods with the fuel assembly guide tubes.

In order for the staff to complete its evaluation of the U.S. EPR control rod (RCCA) design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to insertability of control rods, additional information is required.

Provide the following additional information on the control rod flex joint:

a. Description and drawings, explaining the flex joint connection to the RCCA spider, the flex joint material, and how it functions to accommodate misalignment of the RCCA control rod with the fuel assembly guide tubes.

b. Discuss the operational experience with the control rod flex joint design feature, including any failures of the flex joint.

Response to Question 04.02-10:

The upper end plug (UEP), is machined with a reduced diameter shank (0.180" ± 0.002") and lower shoulder (0.260" ± 0.001") to provide lateral clearance with the interior diameter of the spider finger (see Figure 04.02-10-1). The clearance allows for elastic deflection of each individual UEP for any misaligned control rod, fuel assembly, upper internals, or fuel handling equipment. The UEP threads into the cavity of the spider finger, with a nominal diametral gap of 0.021" between the UEP lower shoulder and spider finger inner diameter. This clearance and the reduced shank diameter allows for elastic lateral deflection of the UEP, while maintaining stresses within design limits.

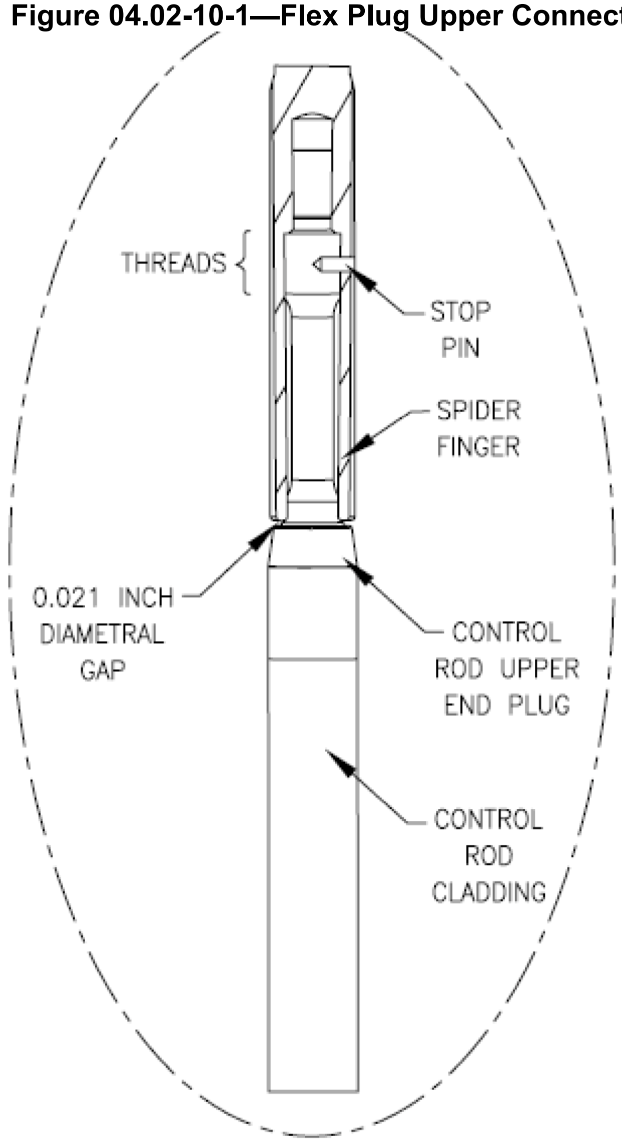
The UEP is threaded into the spider finger to a specified torque. The threaded connection (finger and UEP) is then drilled (normal to the thread axis) to accommodate a 0.094" diameter 308L stainless steel lock pin. The exposed end of the pin is tack welded to the finger.

The threaded and pinned UEP connection design has been used in the U.S. at thirteen plants with no reports of any degradation or failures. The rod cluster control assemblies supplied in the U.S. have used a stainless steel casting spider in lieu of the brazed spider design of the U.S. EPR. However, the method of pinning the UEP-to-spider attachment is similar for both spider designs.

FSAR Impact:

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

Figure 04.02-10-1—Flex Plug Upper Connection



Question 04.02-11:

FSAR Table 4.2-4 and Figure 4.2-12 provide Rod Control Cluster Assembly (RCCA) design data. The dimensions given for the hub diameter and the total height differ slightly between the table and the figure.

Correct the discrepancy and confirm that all the data provided in Table 4.2-4 and Figure 4.2-12 are correct.

Response to Question 04.02-11:

The data in U.S. EPR FSAR, Tier 2, Table 4.2-4 and Figure 4.2-12 were reviewed and the data in the table are correct. The dimension for the hub diameter in the figure is incorrect, and the dimension for the total height is inconsistent with the table due to a rounding error; the figure will be corrected

FSAR Impact:

U.S. EPR FSAR, Tier 2, Figure 4.2-12 will be revised as described in the response and indicated on the enclosed markup.

Question 04.02-12:

Section 4.2.3.1.4 states that crevice corrosion is not a likely corrosion mechanism for zirconium alloy material and that in general, zirconium alloys are very resistant to crevice corrosion.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, the following information is requested.

Justify the statements concerning M5™ fuel rod cladding resistance to crevice corrosion and provide the following information:

a. A description of post irradiation examination results of applicable PWR fuel relative to crevice corrosion.
b. An explanation of applicability of the fuel examination results to the M5™ fuel rod cladding used in the U.S. EPR fuel system design.

Response to Question 04.02-12:

Crevice corrosion is not a likely corrosion mechanism in zirconium alloy nuclear fuel assembly components; however, it is an important design and performance consideration. Accelerated localized corrosion can occur in crevices and other shielded areas on metal fuel assembly surfaces exposed to the primary coolant. This form of corrosion is usually associated with stagnant solution caused by holes, joints, crevices under bolt heads, etc.

In AREVA NP fuel assemblies, the areas of primary concern for crevice corrosion of zirconium alloy components include the spacer grid/fuel rod contact interface, the spacer grid strap intersections, and the area beneath any tenacious crud deposits. Preventing crevice corrosion involves design and surveillance considerations.

Design

The most effective design element for preventing or mitigating any type of corrosion is the choice of alloys. AREVA NP developed zirconium alloy M5® specifically for its excellent corrosion kinetics. Alloy M5® is a binary zirconium–niobium alloy that exhibits excellent resistance to uniform and localized corrosion, as has been demonstrated and documented in many poolside post-irradiation examinations (PIE) and detailed hot cell examinations. AREVA NP has provided the extensive uniform corrosion database for M5® and Zircaloy-4 fuel rod cladding to the NRC at the annual fuel performance meetings. Crevice corrosion has not been specifically addressed, but the only other form of corrosion that has been observed on alloy M5®—the localized galvanic corrosion that occurs on the cladding near the Inconel 718 upper and lower end grids—has been addressed. Visual, eddy-current, and metallographic measurements of this phenomenon reveal that the oxide thickness at these locations is only slightly greater than the uniform oxide thickness away from the end grids, and is less than the Zircaloy-4 uniform oxide thickness.

Another design element important for preventing crevice corrosion is control of manufacturing processes that could produce laps, holes, seams, etc. which could provide sites conducive to crevice corrosion or other localized corrosion mechanisms (pitting and delayed hydrogen cracking). Fuel rod cladding tube defects that may be produced by pilgering, polishing, handling, etc. are detected by eddy-current and ultrasonic inspection of every manufactured tube. Similarly, careful inspection of alloy M5™ sheet used in spacer grids detects any defects produced in rolling, handling and final surface preparation so they can be rejected before the grid straps are stamped

Surveillance

AREVA NP closely monitors fuel assembly performance by performing routine PIE examinations. The PIE inspections specifically designed to monitor the corrosion performance of alloy M5® components are the cladding, guide tube and spacer grid oxide thickness measurements, as well as detailed visual examinations of the entire fuel assembly. The PIE examinations of irradiated AREVA NP fuel included visual inspection for evidence of crevice and other types of localized corrosion. New designs and emerging performance issues are subject to frequent detailed destructive examinations performed in a hot cell. A typical fuel performance hot cell examination includes oxide thickness measurements by metallography and full rod visual and eddy-current examinations. In addition, detailed microscopy visual examinations may be performed (optical, scanning electron microscopy (SEM), and scanning transmission electron microscopy (STEM)).

In addition to the inspections performed on design features such as component surfaces, grid strap intersections, and grid stop/cladding interfaces, detailed examinations of visible anomalies (such as tenacious crud and unidentified stains) are also performed No crevice corrosion has been observed.

Conclusion

The only forms of corrosion observed on alloy M5® have been the very low uniform oxidation of fuel rod cladding, guide tubes, and spacer grid straps (a factor of five lower than for optimized, low-tin Zircaloy-4) and the galvanic type of localized corrosion near Inconel top and bottom spacer grids. Crevice corrosion has never been observed.

Fuel rod cladding, guide tubes, and spacer grids used in the U.S. EPR fuel assembly design will be produced using the identical M5® alloy used in all AREVA NP fuel assemblies in the U.S. and Europe, and the components will be fabricated using the same manufacturing techniques, procedures, and inspections. Thus, the fuel performance corrosion database for existing alloy M5® fuel assemblies is directly applicable to the U.S. EPR fuel system design.

This combination of low intrinsic corrosion rates for the M5® alloy, tight control of the fuel rod manufacturing process, regular PIE/surveillance, and lack of any observations of crevice corrosion to date, supports the conclusion that crevice corrosion will not occur in the U.S. EPR.

FSAR Impact:

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

Question 04.02-13:

Section 4.2.3.2.4 states that irradiation stability of the U.S. EPR fuel pellet is confirmed by performing analyses using the COPERNIC code.

In order for the staff to complete its evaluation of the performance of the U.S. EPR fuel against the requirements of GDC 10 as it relates to SAFDLs for normal operation and during AOOs, additional information is required.

Provide a description and the results of the COPERNIC analyses that were performed to confirm the irradiation stability of the U.S. EPR fuel pellet.

Response to Question 04.02-13:

Section C.I.4.2.3, item 2 (d) of Regulatory Guide 1.206 requires that the FSAR design evaluation include an assessment of the irradiation stability of the fuel, including fission product swelling and fission gas release. The irradiation stability of the U.S. EPR fuel pellets is confirmed by the NRC-approved thermal-mechanical models used in the COPERNIC (Reference 1) analyses of the U.S. EPR fuel rods, which include the effects of fission product swelling and fission gas release.

The relevant COPERNIC analyses performed for the U.S. EPR fuel include cladding strain analysis, fuel rod internal pressure analysis and fuel centerline melt analysis. The results of these analyses are presented in Reference 2, Sections 5.1.2, 5.1.8, and 5.2.4.

FSAR Impact:

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

References:

1. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

2. ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Topical Report”, October 2007

Question 04.02-14:

Section 4.2.3.6.1 states that calculations of RCCA rod internal gas pressure and cladding stresses have been performed and that for all the cases analyzed the cladding stresses remain within acceptable limits. Similarly, the neutron source rod cladding stresses are stated to remain below the stress limits.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR fuel control rod (RCCA) design relative to the requirements of GDC 27 as it relates to the capability of the reactivity control systems. The performance of the neutron source tubes must be evaluated relative to their potential effect on surrounding fuel rods relative to GDC 10.

Explain the source of control rod pressurization during normal operation, AOOs, and accident (LOCA) conditions.

What are the stress level acceptance criteria for conditions of normal operation, AOOs, and accident (LOCA) conditions, and how do the calculated internal control rod pressures and stress levels compare to the criteria?

Provide the analysis results for the calculated internal rod pressures and stresses with comparison to the acceptance criteria.

Response to Question 04.02-14:

As stated in U.S. EPR FSAR, Tier 2, Section 4.2.3.6.1, the rod cluster control assembly (RCCA) does not generate internal gases during operation. The control rod has an initial internal helium pressure of **[ ]** psig. During operation, the internal pressure is a function of helium temperature and changes to the internal void volume due to differential thermal expansion or contraction of the absorber and the cladding. The limiting criteria for internal control rod pressure is that the system pressure of 2250 psia is not exceeded during normal operations and anticipated operational occurrences (AOOs). The analysis shows that the maximum internal pressure under these conditions is **[ ]** psia.

Control rod cladding stresses, including consideration of differential pressure conditions, were determined for normal operations, AOOs, and postulated accidents. The bounding case considered a maximum reactor system design pressure of **[ ]** psia and a minimum control rod internal rod pressure of **[ ]** psia, with no credit taken for increased internal pressure at hot conditions, which maximizes the differential pressure across the cladding wall. A summary of the stress limits, stress analysis results and margins for normal operations and AOOs are summarized in Table 04.02-14-1 below.

Table 04.02-14-1—Summary of RCCA Cladding Stress Analysis

[ ]

Accident conditions were also evaluated to determine the control rod internal pressure. Maximum cladding temperatures were determined for different postulated accidents. The small break LOCA produced the maximum cladding temperature, **[ ]**. The resulting internal rod pressure was **[ ]** psia. The limiting criteria applied to control rod cladding under accident conditions is that the cladding does not rupture. The resulting maximum cladding stress was **[ ]** psi, which is significantly lower than the burst strength of 316L stainless steel of approximately 45,000 psi at the maximum cladding temperature.

FSAR Impact:

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

Question 04.02-15:

Section 4.2.4.1.4 states that more than 4,600 ion-nitrated HARMONI™ RCCAs have been delivered to operating reactors, including 572 RCCAs to U.S. PWRs since 1995.

Additional information is required in order for the staff to complete its evaluation of the U.S. EPR control rod design relative to the requirements of GDC 27 and the guidelines of SRP Section II.1.A.viii as they relate to the capability of the reactivity control systems.

Provide additional information on the design and performance experience of the HARMONI™ RCCAs relative to the U.S. EPR RCCA designs, including:

a. Identification of reactor facility, number of RCCAs delivered, and year of installation.

b. Any difference in the mechanical design, configuration, or materials of the delivered RCCAs relative to the U.S. EPR RCCAs.

c. Identification of any failures or degradations of the RCCAs.

Response to Question 04.02-15:

HARMONI™ rod cluster control assembly (RCCA) deliveries are listed in Tables 04.02-15-1 and 04.02-15-2 below:

Table 04.02-15-1—U.S. HARMONI™ RCCA Deliveries

Plant	Number of RCCAs Delivered	Year of Delivery to Site
Reactor 1	53	1996
Reactor 2	53	1995
Reactor 3	55	1995
Reactor 4	54	1997
Reactor 5	54	1997
Reactor 6 (15x15)	45	2000
Reactor 7 (15x15)	45	2000
Reactor 8	53	2001
Reactor 9	53	2001
Reactor 10	53	2002
Reactor 11	54	2002
Reactor 12	31	2008
Reactor 13	52	2008
Total	**655**	

Table 04.02-15-2: HARMONI™ RCCA delivered outside the U.S.

Country	Number of RCCAs Delivered
France	3131
Belgium	334
South Africa	154
China	332
Sweden	150
Spain	152
UK	43
South Korea	156

The design of the HARMONI™ RCCAs delivered to U.S. plants is similar to those for the U.S. EPR. The differences (e.g., RCCA length differences and the available clearance between the upper core plate and the fuel assembly, which impacts the spider hub and spring lengths) primarily arise from the different fuel assembly) configurations. In addition, the original 1988 HARMONI™ RCCAs used ion-nitrided 304L stainless steel, but In 1993, the cladding material was changed to 316L stainless steel. The most recent change has been a lengthened lower end plug to improve wear performance.

Notable similarities between the U.S. EPR RCCA and the HARMONI™ RCCAs previously delivered include:

1. Common among all HARMONI™ RCCAs, including those for the U.S. EPR and those previously delivered in the U.S. and in Europe, is ion-nitride hardening with a nominal depth of **[ ]** on the control rod outer surface. The hardened outer surface extends from a point just below the upper end plug weld down to, and including, the stainless steel lower end plug.
2. All materials used in the U.S. EPR HARMONI™ RCCAs and those delivered previously in U.S. plants are essentially the same, and include:
    a. 316L (or 304L for early designs) colddrawn stainless steel tubing for the cladding, with an ion-nitride outer surface.
    b. 308L upper and lower end plugs.
    c. Nickel alloy 718 spider reaction spring, colddrawn and age hardened.
    d. 304L stainless steel spring retainer bolt.
    e. 17-4PH stainless steel (UNS S17400) spring retainer cup.

f. 80% Ag - 15% In - 5% Cd (AIC) absorber (hybrid designs included B_4C absorber and thick-walled cladding).

g. Stainless steel spider (either 304L stainless steel or CF3, see discussion below).

Notable differences between the U.S. EPR RCCA and other HARMONI™ RCCAs previously delivered to U.S. sites include:

1. The U.S. EPR uses longer control rods to accommodate the nominal 14-foot U.S. EPR core, whereas previously delivered RCCA control rods in the U.S. are designed for 12-foot cores.
2. The U.S. EPR uses an all-AIC absorber material, whereas the Reactors 1, 2 and 3 (see table 04.02-15-1) all received hybrid control rods, which used a combination of AIC and B_4C pellets for neutron absorption. All other delivered RCCAs in the U.S. used an all-AIC absorber.
3. Due to the use of B_4C with the Reactors 1, 2, and 3 control components, the hybrid RCCA cladding wall thickness is 0.0385", compared to 0.0185" for the thin-wall all-AIC designs (including the U.S. EPR RCCA).
4. The U.S. EPR uses a full-length annular absorber, whereas all previously delivered RCCAs in the U.S. used solid (round bar) AIC material.
5. Spider configuration:
   - Previous U.S. deliveries used a CF3M stainless steel casting (integral hub, vanes, and fingers).
   - Outside the U.S. the spider assemblies used separate 304L wrought stainless steel components (hub, vanes, and fingers) that were brazed into an integral spider.
   - The U.S. EPR currently uses the brazed spider, however it may transition to the casting depending on the manufacturing methods chosen. Both are acceptable and have essentially equivalent functionality, corrosion resistance, and mechanical strength.
6. The lengths of the hub, spring, and retainer bolt are different depending on the specific reactor configuration (based on the clearance between upper core support plate and fuel assembly).
7. Most of the delivered RCCAs in the U.S. are comprised of 24 control rods that interface with a 17x17 fuel pin lattice. However, the HARMONI™ RCCAs for Reactors 6 and 7 use only 20 control rods, as they interface with a 15x15 fuel pin lattice.

Results of Inspections Performed To Date

U.S. inspections of HARMONI™ RCCAs have been performed at Reactor 2 (November2003), Reactor 4 (October 2005), and Reactor 5 (March, 2008). All three inspections showed some external surface wear of the cladding and lower end plug. No rodlet separations have occurred, nor has any cladding cracking or other indications of RCCA degradation been reported.

At Reactor 2 (thick-clad hybrid), 53 RCCAs were inspected, 27 of which showed wear depths **[ ]** or deeper at an axial location corresponding to guide card 2, where the cladding is un-coated. The deepest indication was **[ ]**.

In Reactor 2 RCCAs, 20 RCCAs had indications of wear **[ ]** or deeper in the ion-nitrided areas, usually at the solid end plug. The deepest wear measurement was **[ ]**, located at the solid end plug. Within the ion-nitrided cladding region, the deepest wear measurement was **[ ]** near the tip. At a guide card locations, the deepest wear scar was **[ ]** measured at guide card location 4.

At Reactor 4 (thin-clad, all-AIC), 53 RCCAs were inspected, and 23 showed wear depth greater than **[ ]**, with a maximum depth of **[ ]** located 0.8 inches from the tip (i.e., at the end plug weld). Most of the wear depth indications were at the lower tip region, although there were wear indications along the coated cladding length, with a maximum wear depth of **[ ]** at a location 79.0 inches from the tip. The Reactor 4 RCCA rods were supplied with short end-plugs, with a tip-to-weld centerline distance of 0.875" nominal. Current HARMONI™ RCCA rod lower end plugs, including the U.S. EPR RCCA, are longer, with a tip-to-weld centerline distance ranging from 1.663" to 2.291" (nominal). This provides added protection for the cladding and weld from guide tube wear.

At Reactor 5 (thin-clad, all-AIC), 53 RCCAs were inspected, and 22 showed wear greater than **[ ]** deep, the deepest being **[ ]** at the tip (solid lower end cap) of the rodlet. Along the cladding length, the deepest wear mark was **[ ]**, approximately 7.9" from the lower rodlet tip.

European inspection results showed the Harmoni™ RCCAs have performed well. Cladding cracking due to irradiation swelling of solid silver-indium-cadmium absorber has occurred, but this is a known phenomenon and is factored into the design life and inspection schedules of the components. There have also been three isolated cases of rodlets separating from the spiders due to missing stop pins.

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

corrosion is significantly lower than the established limit. Because crude formation occurs in part from cladding oxidation, the M5^{TM} cladding will have less crud buildup, even with other crud buildup factors remaining the same. Therefore, material wastage from mass transfer is greatly reduced.

4.2.3.1.8 Rod Bowing Attributable to Thermal, Irradiation, and Creep Dimensional Changes

Rod bowing is addressed in Section 4.2.3.5.6.

4.2.3.1.9 Consequences of Power-Coolant Mismatch

The consequences of power-coolant mismatch are addressed in Section 4.4.

4.2.3.1.10 Irradiation Stability of the Cladding

Considerable operating experience using M5^{TM} cladding has proven its irradiation stability. The effects of irradiation on the mechanical integrity of the cladding has been accounted for using the approved COPERNIC model (Reference 6) for performing the mechanical and thermal analyses, and the effects are shown to be acceptable for the currently approved burnup limit of 62 GWD/MTU established in the Extended Burnup Topical Report Reference 5.

4.2.3.1.11 Creep Collapse and Creepdown

04.02-4

The computer code CROV, a creep ovalization analysis program developed and certified for AREVA NP fuel rods, is used to evaluate the resistance of the U.S. EPR fuel rod cladding to creep collapse. Use of the CROV code in performing the creep collapse analysis for M5^{TM} cladding has previously been approved in the M5^{TM} Topical Report Reference 2. Inputs to the analysis include differential pressure, temperature gradients, and fast flux. The enveloping power histories from the COPERNIC thermal-hydraulic analysis (Reference 6) are used to initialize the creep collapse code. As discussed in Reference 2, the creep rate of M5TM is approximately ~~50~~67 percent slower than Zircaloy-4; therefore, a multiplier of ~~0.5~~67 or higher (for conservatism) is applied to the CROV input for the M5^{TM} analysis.

The following conservatisms were used in determining creep collapse over the life of the fuel rod:

- Minimum fuel rod pre-pressure.
- No fission gas release.
- A worst-case or enveloping power history.
- Worst-case cladding dimensions.

Table 4.2-1—U.S. EPR Fuel Assembly Design Summary

Parameter	Value or Description
Fuel Assembly Parameters	
Number of fuel rods per assembly	265
Number of guide tubes per assembly	24
Number of intermediate grids per assembly	8
Number of end grids per assembly	2
Fuel assembly envelope	8.426 in
Fuel rod pitch	0.496 in
Fuel Rod Parameters	
Cladding material	M5
Cladding outside diameter	0.3740 in
Cladding inside diameter	0.3291 in
Fuel column length	165.354 in
Overall fuel rod length	179.134 in
UO_2 Pellet Parameters	
Outside diameter	04.02-9 0.3225 in
Length	0.531 in
Fissile enrichment	≤ 5.00~~4.95~~ wt% U-235
UO_2-Gadolinia Pellet Parameters	
Outside diameter	0.3225 in
Length	0.531 in

Figure 4.2-12—Rod Cluster Control Assembly