CONTENTS

LIST OF FIGURES

No figures were included in the chapter.

LIST OF TABLES

No tables were included in the chapter.

4 REACTOR

4.0 Reactor

4.1 Summary Description

This section of the U.S. EPR Final Safety Analysis Report (FSAR) presents an overall description of the reactor core including the principal elements of the design, incore and excore instrumentation system, and information on the computer codes used in various aspects of the design of the core and fuel rods. The U.S. EPR is a light water cooled and moderated pressurized water reactor (PWR). It has a rated thermal power of 4,590 megawatts (MW) and operates at a pressure of 15.51 mega pascals (MPa) (2,250 pounds per square inch absolute (psia)). This reactor design is called an evolutionary design, as it is similar in many ways to the pressurized water reactors (PWRs) currently in operation in the United States (U.S.) and elsewhere in the world.

The U.S. EPR reactor core is made up of 241 fuel assemblies with each assembly containing 265 fuel rods and 24 guide tube locations arranged in a 17x17 array. Depending on the location of the fuel assembly in the core, the guide tubes in each fuel assembly may be used for rod control cluster assemblies (RCCA), also commonly referred to as control rods or incore nuclear instrumentation. Each of the 265 fuel rods in a fuel assembly contains uranium dioxide (UO_2) or uranium dioxide and gadolinium oxide (Gd_2O_3). Gadolinia is a neutron absorber and is used to establish a favorable radial power distribution. The concentration of gadolinia may be as high as 8 weight (wt) percent with up to 28 of the 265 fuel rods per fuel assembly containing gadolinia. The maximum enrichment of the uranium in the core is 5 wt percent. The enrichment varies depending on the position of the fuel assembly in the core. Typically, the rods with lower enrichment are placed on the periphery of the reactor core for reasons of neutron economy. The fuel rods are made of seamless zirconium tubing with zirconium end plugs welded at each end. The fuel rods are pressurized with helium during fabrication.

There are two methods of reactivity control for the U.S. EPR, rod control cluster assemblies and dissolved boron in the reactor coolant system (RCS). Eighty-nine RCCAs each containing 24 stainless steel neutron absorber (control) rods are used to control reactivity. For each of the 89 fuel assemblies that contain RCCAs, there is no incore instrumentation, since each of the 24 guide tubes is occupied by control rods. The control rod absorber material in the stainless steel tubes, which are the individual control rods, is 80 wt percent silver, 15 wt percent indium, and 5 wt percent cadmium. The RCCAs are controlled by the control rod drive system (CRDS). The boron concentration is varied by the chemical and volume control system (CVCS).

The reactor power level during operation is proportional to the difference in the hot and cold leg temperatures in each of the four primary coolant loops. The primary loop heat balance, as determined from loop temperature difference, is combined with fast response excore neutron detectors (i.e., located outside the core) to detect slow and fast changes in reactor power.

The incore instrumentation system is an Aeroball Measurement System (AMS). The AMS circulates steel balls containing vanadium into 40 radial core locations, each with 36 axial segments. The incore residence time of the steel balls is controlled. After the balls exit the core, the activation is measured. This provides a means to infer an axial and radial core neutron flux map, since the activation is proportional to the local reactor power.

4.2 Fuel System Design

4.2.1 Introduction

The design and safety objectives of the fuel system are to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

4.2.2 Summary of Application

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review. The fuel system design will not be included in the design aspects that are certified.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in FSAR Tier 2, Section 4.2, "Fuel System Design," summarized here in part, as follows:

The U.S. EPR fuel system consists of a reactor core containing 241 fuel assemblies and 89 control rods, or rod cluster control assemblies. Each fuel assembly is made up of 265 fuel rods and 24 guide tubes arranged in a 17x17 array. The assembly is held together by a bottom and top nozzle and guide tubes, with 10 spacer grids. Each fuel rod consists of a seamless M5^{TM} zirconium alloy tube containing enriched UO_2 ceramic material fuel pellets. Some of the fuel pellets also contain gadolinia (Gd_2O_3) as a burnable absorber for reactivity and power distribution control. The fuel rods are pressurized with helium during fabrication and leak-tested.

Each rod cluster control assembly is made up of 24 individual control rods fastened to a central hub spider assembly. The individual control rods consist of an absorber rod, composed of an 80 percent silver (Ag), 15 percent indium (In), and 5 percent cadmium (Cd) alloy, sealed within a stainless steel tube.

In addition to the fuel and control rods, fixed incore neutron detectors are positioned at six axial locations inside a guide tube of 12 different fuel assemblies throughout the core. The 152 fuel assemblies whose guide tube locations are unused by control rods contain stationary control component assemblies (SCCAs) positioned to restrict coolant bypass flow and to guide incore instrumentation into the fuel assembly. Some of the stationary control component assemblies contain neutron source material utilized for startup of the initial operating cycles.

Throughout FSAR Tier 2, Section 4.2, the applicant refers to licensing Topical Report ANP-10285P, Revision 0, "U.S. EPR Fuel Assembly Mechanical Design Topical Report," AREVA NP, Inc. (hereafter AREVA or the applicant), October 2007, which is currently under review by the staff. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.** The open item will be considered resolved when the final safety evaluation report (FSER) on Topical Report ANP-10285P is issued.

The content of FSAR Tier 2 sections follows the guidance contained in Regulatory Guide (RG) 1.206, "Combined License Applications (COLAs) for Nuclear Power Plants," June 2007, as summarized in the paragraphs below.

FSAR Tier 2, Section 4.2.1, “Design Bases,” describes the fuel system design bases relative to the fuel rod cladding, fuel material, spacer grids, assembly structural design, and the rod cluster control assemblies. The application addresses the mechanical properties, stress-strain limits, vibration and fatigue, and chemical properties of the fuel rod M5™ cladding material. The fuel rod cladding is described and analyzed in BAW-10227P-A, “Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel Framatome Cogema Fuels,” June 2003, and is justified for use in ANP-10263P-A, Revision 1, “Codes and Methods Applicability Report for the U.S. EPR,” AREVA NP, Inc., November 2007. The reasons why these topical reports apply to the U.S. EPR submittal are discussed below.

The thermal-physical properties of the fuel material and the effects of fuel densification are addressed in ANP-10285P, which is currently under review by the staff. The computer code COPERNIC, as described in BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” FRAMATOME ANP, January 2004, is used to perform the thermal-mechanical analyses to simulate the behavior of the fuel rod during irradiation and verify that the U.S. EPR fuel rod design meets design and safety criteria. The applicability of the COPERNIC code to the U.S. EPR is justified in ANP-10263P-A, “Codes and Methods Applicability Report for the U.S. EPR,” AREVA NP, Inc., November 2007. FSAR Tier 2, Section 4.2.1 provides the design bases for nonoperational loading, normal operational conditions, anticipated operational occurrences (AOOs), and postulated accident conditions.

FSAR Tier 2, Section 4.2.2, “Description and Design Drawings,” provides a description of the fuel assemblies, rod cluster control assemblies, and stationary control component assemblies. The fuel assembly is designed to achieve a maximum fuel rod average burnup of 62 GWD/MTU. Each fuel assembly may be loaded into any one of the 241 core locations, with proper orientation of the assembly fixed via a mating pin with the refueling machine grapple. The applicant describes the two types of spacer grids used for the U.S. EPR fuel assembly: The high mechanical performance (HMP) grid used at the upper and lower positions; and the high thermal performance (HTP) grid used at the eight intermediate positions.

The top and bottom nozzles are also described in FSAR Tier 2, Section 4.2.2. The bottom FUELGUARD™ nozzle includes a debris-resistant feature comprised of curved blades for improved debris filtering.

The applicant also describes the design of the stationary control component assembly that is utilized in nonrodded fuel assemblies in FSAR Tier 2, Section 4.2.2.10, “Stationary Control Component Assemblies Description.” The purposes of the stationary control assemblies are to restrict coolant flow bypass and guide the incore instrumentation probes into the core. Three types of stationary control assembly designs are described, applicable to various instrumented or non-instrumented fuel assembly configurations.

FSAR Tier 2, Section 4.2.3, “Design Evaluation,” provides the applicant’s design evaluation of the U.S. EPR fuel system. The applicant’s evaluation addresses fuel rod cladding stress, fretting, fatigue, and chemical reaction; fuel pellet purity and dimensional stability; and fuel assembly mechanical component performance, as described further in Section 4.2.4.3 of this report.

The fuel burnup experience and fuel rod failure history are addressed in licensing Topical Report ANP-10285P, “U.S. EPR Fuel Assembly Mechanical Design Report,” which is currently under review by the staff.

Structural performance of the rod cluster control assembly and stationary control component assembly is also evaluated in FSAR Tier 2, Section 4.2.3.6, “Reactivity Control, Neutron Source, and Thimble Plug Assemblies.”

FSAR Tier 2, Section 4.2.4, “Testing and Inspection Plan,” describes the various tests that have been performed or are planned to be performed, including in-reactor operating experience, prototype testing, component testing, new fuel testing, and post-irradiation surveillance.

ITAAC: There are no inspections, tests, analyses, and acceptance criteria (ITAAC) items for this area of review.

Technical Specifications: The Technical Specifications (TS) associated with FSAR Tier 2, Section 4.2 are given in FSAR Tier 2, Chapter 16, “Technical Specifications,”; FSAR Tier 2, Section 4.2.1, “Fuel Assemblies”; and FSAR Tier 2, Section 4.2.2, “Control Rod Assemblies.” The Related Core Operating Limits Report (COLR) is addressed in FSAR Tier 2, Section 16.5.6.3.

4.2.3 Regulatory Basis

The relevant requirements of Nuclear Regulatory Commission (NRC) regulations for this area of review, and the associated acceptance criteria, are given in Section 4.2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” (hereafter referred to as NUREG-0800 or the SRP) and are summarized below with the exception of some regulatory provisions that do not directly apply to this review. Review interfaces with other SRP sections can be found in Section 4.2 of NUREG-0800 and are summarized below. These regulations include:

1. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-water Nuclear Power Reactors,” and 10 CFR 52.47, “Contents of applications; technical information,” as they relate to the cooling performance analysis of the emergency core cooling system (ECCS) using an acceptable evaluation model and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

2. General Design Criterion (GDC) 10, “Reactor Design,” as it relates to assuring that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs.

3. GDC 27, “Combined Reactivity Control Systems Capability,” as it relates to control rod insertability under postulated accident conditions.

4. GDC 35, “Emergency Core Cooling,” as it relates to the reactor fuel system being designed such that the performance of the Emergency Core Cooling System will not be compromised following a postulated accident.

Acceptance criteria adequate to meet the above requirements include:

1. In order to meet the requirements of 10 CFR 50.46, as it relates to ECCS performance evaluation, the ECCS evaluation model should include a calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding.

2. In order to meet the requirements of GDC 10, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. SRP Acceptance Criteria Section 1.A, "Fuel System Damage," provides a complete list of the damage criteria.

3. In order to meet the requirements of GDC 35 as it relates to core coolability for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. SRP Acceptance Criteria Section 1.C, "Fuel Coolability," provides a complete list of the criteria.

4. In addition to the criteria applicable to the regulatory requirements described above, SRP 4.2 refers to the following RGs for criteria pertaining to analyses and results presented in the FSAR:

    - RG 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," March 1978, for acceptability of fuel densification models

    - RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, for a description of models, correlations, data, and methods for ECCS and loss-of-coolant accident (LOCA) evaluations

The staff's review of ECCS performance against the requirements of 10 CFR 50.46 and 10 CFR 52.47 is provided in U.S. EPR Safety Evaluation Report for FSAR Tier 2, Sections 6.3, "Emergency Core Cooling System," 15.0.3, "Radiological Consequences of Design Basis Accidents," and 15.6.5, "Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary."

4.2.4 Technical Evaluation

The staff has reviewed the U.S. EPR FSAR Tier 2, Section 4.2, "Fuel System Design," including applicable TS, and the initial test program to determine the acceptability of the design. Because of the reliance of FSAR Tier 2, Section 4.2 on the evaluations contained in ANP-10285P, the staff also reviewed the associated portions of ANP-10285P for applicability to the FSAR content. The staff's evaluation of FSAR Tier 2, Section 4.2 information was performed against the requirements of 10 CFR Part 50, GDC 10, GDC 27, and 10 CFR 50.46 and 52.47. There are no FSAR Tier 1 items for the fuel system design, since the applicant is not seeking certification for this area. .

The TS identified in Section 4.2.2 above specify the reactor core as being comprised of 241 fuel assemblies and 89 control rods. FSAR Tier 2, Chapter 16 TS 4.2.1 states that the fuel assemblies shall consist of a matrix of fuel rods made of zirconium-based alloy containing uranium dioxide fuel material. FSAR Tier 2, Chapter 16 TS 4.2.2 states that the control rod absorber material shall be silver-indium-cadmium, as approved by the NRC.

The staff determined that the TS conform to the descriptions and requirements provided in FSAR Tier 2, Section 4.2 and, therefore, concludes that they are sufficient for their intended purpose. A detailed discussion of the TS is contained in Chapter 16 of this report.

The acceptance criteria used in this evaluation are contained in SRP Section 4.2, "Fuel System Design," as summarized above in Section 4.2.3. Completeness of the content of FSAR Tier 2 section was reviewed against SRP Section 4.2 and the applicable portions of RG 1.206, and the results are summarized in the following sections of this report.

In addition to the material provided in FSAR Tier 2 section, the review was extended to the applicant's referenced licensing topical reports as necessary to evaluate the design.

FSAR Tier 2, Section 4.2.2 above provides an overview description of the U.S. EPR fuel assembly.

4.2.4.1 *Design Bases*

FSAR Tier 2, Section 4.2.1, "Design Bases," describes the bases for the fuel system design which are intended to meet the requirements of GDC 10, GDC 27, and GDC 35, and 10 CFR 52.47 and 50.46, and conform to the guidance in SRP Section 4.2. The individual components of the fuel assembly are addressed in the paragraphs below. This section of this report evaluates whether the bases and criteria are consistent with requirements and guidance for the fuel system design. The analyses that demonstrate these bases and criteria are addressed in Section 4.2.4.3 of this report.

Fuel Rod Cladding

The fuel rod cladding utilized for the U.S. EPR fuel assembly is the M5^{TM} material documented and approved for use in PWR fuel in BAW-10227P-A. M5^{TM} is a zirconium alloy with nominally one percent niobium added for improved material performance.

The material properties of the M5^{TM} fuel rod cladding are described in BAW-10227P-A, which states that the M5^{TM} alloy has a low cross section for thermal neutrons like Zircaloy but shows lower levels of corrosion, hydrogen pickup, and growth during incore operation. The design criteria for fuel rod cladding stress are based on unirradiated cladding properties for normal operation, AOOs, and accidents including those for combined seismic-LOCA loads, as substantiated in BAW-10227P-A. The use of unirratiated M5^{TM} properties to establish stress limits are conservative because the properties of yield and ultimate tensile stresses increase more than a factor of 3 quickly with irradiation. In addition, the M5^{TM} cladding stress criteria conform to the guidance in SRP 4.2 Section II.1.A.i and II.1.B.viii, which is based on Section III of the Boiler and Pressure Vessel (B&PV) Code of the American Society of Mechanical Engineers (ASME) that states: "Stress limits that are obtained by methods similar to those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) are acceptable. Other proposed limits must be justified." The staff has examined the stress limits for M5^{TM} cladding and finds them to be acceptable based on the preceding discussion.

The cladding strain criterion of ≤ one percent elastic plus plastic uniform hoop strain is contained in BAW-10227P-A. This strain criterion conforms to the acceptance criteria contained in SRP 4.2, Section II.1.A.i and II.1.A.vi. Therefore, the staff finds this cladding strain criterion to be acceptable.

The U.S. EPR fatigue criterion as described in FSAR Tier 2, Section 4.2.1.1.3, "Vibration and Fatigue of Cladding," states that the cumulative fatigue usage factor should not exceed 1.0. This conforms to regulatory guidance contained in SRP 4.2, Sections II.1.A.ii. Therefore, the staff finds this cladding fatigue criterion to be acceptable.

The design bases for vibration-induced fretting wear in FSAR Tier 2, Section 4.2.1.1.3 states that fretting wear is limited to prevent fuel rod failures based on fretting wear tests. There are two specific criteria to prevent fretting wear: (1) Limit cross flows; and (2) provide sufficient rod support with the grid spacer springs to limit vibration as specified in Section 5.1.4 of ANP-10285P. The ability of these measures to limit fretting wear has been demonstrated by fretting wear tests of the specific grid design. The acceptance criteria contained in SRP 4.2 Section II.1.A iii simply states that fretting wear should be limited and accounted for in analyses. Since the applicant proposed effective measures to limiting fretting wear and accounted for it in analyses, the staff finds the design bases for fretting wear to be acceptable.

The applicant evaluated fuel rod cladding oxidation and hydriding as potential fuel failure mechanisms in BAW-10186P-A, Revision 2, "Extended Burnup Evaluation," Framatome Cogema Fuels, January 2004. Further, the applicant established a corrosion and crud (combined) thickness design maximum of 100 microns (BAW-10227P-A) and a maximum hydrogen concentration in ANP-10285P applicable to the M5^{TM} cladding. As stated in the staff's safety evaluation of BAW-10186P-A, the M5^{TM} cladding has acceptable performance with respect to corrosion and crud buildup up to a rod average burnup of 62 giga-watt days per metric ton uranium (GWD/MTU). In addition, the data provided in the review of ANP-10285P suggests that both corrosion and hydrogen remain significantly below the corrosion and hydrogen limits proposed. However, the staff's review of ANP-10285P has identified an issue related to the applicant's proposed hydrogen limit for M5^{TM} cladding since the data provided are insufficient to justify the proposed limit. The fuel rod design bases pertaining to oxidation and crud buildup as described in the FSAR (combined thickness limited to 100 microns) conform to the criteria (SAFDLs) given in SRP Section II.1.A.iv and are acceptable. The applicant's proposed hydrogen limit for M5^{TM} cladding will be addressed in the staff's Safety Evaluation Report (SER) on Topical Report ANP-10285P**.** The design bases of the fuel rod cladding for oxidation and crud buildup conform to the requirements of GDC 10 and the applicable SRP Section 4.2 acceptance criteria and, therefore, are acceptable. The design bases for hydrogen levels will be addressed in the review of ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

Fuel Pellet Material

The design bases for fuel pellet thermal-physical properties are also addressed in this section of the FSAR, including the effects of fuel densification, pellet swelling, and control of contaminants during pellet manufacture. The NRC-approved methodology used to perform the fuel performance analyses using the COPERNIC fuel performance code for UO_2 fuel applications, including acceptance criteria for fuel performance, is documented in BAW-10231P-A, and its application to the U.S. EPR was approved by the staff in its safety evaluation of ANP-10263P-A.

In reviewing BAW-10231P-A, the staff determined that the COPERNIC code determines fuel densification in accordance with RG 1.126, and the staff's safety evaluation of BAW-10231P-A concludes that the fuel densification and swelling models in COPERNIC are acceptable. BAW-10231P-A provides a comparison of predicted UO_2 fuel pellet densification versus measured data for powders produced from a dry and two wet conversion processes. From this information it was not clear which powder conversion process and densification model would be used for the U.S. EPR. In Question 04.02-3 of RAI No 195, the staff requested that the powder conversion process applied for U.S. EPR fuel be defined along with the results of COPERNIC predicted pellet densification and swelling as a function of burnup. In an April 22, 2009, response to RAI 195, Question 04.02-3, the applicant provided the powder conversion process along with the calculated fuel pellet density versus burnup for U.S. EPR fuel. In the aforementioned response, the initial UO_2 pellet density is shown to correspond approximately to a 96 percent theoretical density (TD). The results provided in the applicant's response to the RAI are comparable to the results provided in BAW-10231P-A. Consequently, the staff considers RAI 195, Question 04.02-3 resolved. RG 1.126 recommends the use of methodologies to model fuel densification depending on the initial condition of the fuel and other conservative assumptions such as to minimize heat transfer, which will result in higher densification. The staff finds that densification effects included for the EPR fuel are consistent with the guidance in SRP 4.2 Sections II.1.B.ii, II.3.C.i, II.3.C.ii and II.3.C.iv and RG 1.126. Therefore, the staff finds that the EPR inclusion of densification effects acceptable.

Fuel Rod Performance

Fuel rod performance evaluations are performed by AREVA NP utilizing COPERNIC methodology (BAW-10231P-A,) and its application to the U.S. EPR was approved by the staff in its safety evaluation of ANP-10263P-A.

The criterion for creep collapse is that creep collapse be precluded during the in-reactor life of the fuel rod as per the guidance in SRP 4.2 Section II.1.B ii. The FSAR does not explicitly include a criterion for creep collapse but a criterion is provided in ANP-10285P, Section 5.2.2, for the U.S. EPR that states, "The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected incore life." However, FSAR Tier 2, Section 4.2.1.3.2, "Models Predictions," states that the possibility of fuel rod cladding creep collapse is precluded by high TD as-fabricated pellets, which exhibit very low densification during operation, and by manufacturing process inspections to prevent fuel column gaps. Therefore, the criterion of no creep collapse provided in ANP-10285P conforms to the guidance in SRP Section 4.2 Part II.1.B ii. The U.S. EPR creep collapse criterion is reviewed in the Safety Evaluation Report on ANP-10285P, which is ongoing**. RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

ANP-10285P includes a fuel pellet design criterion that for a 95 percent probability at a 95 percent confidence level, fuel pellet centerline melting shall not occur for normal operation and AOOs. This fuel melt criterion conforms to the acceptance criteria contained in SRP 4.2 Section II.1.B iv and is, therefore, acceptable.

The criteria for fuel rod internal pressure are defined in BAW-10227P, and in BAW-10183P-A, "Fuel Rod Gas Pressure Criterion," B&W Fuel Company, July 1995, such that the pellet-to-cladding gap will not increase during steady-state operation, and extensive departure from nucleate boiling (DNB) propagation will not occur. BAW-10227P-A and BAW-10183P-A are approved for application to the U.S. EPR per NRC safety evaluation of ANP-10263P-A. The EPR fuel design is similar to previously approved AREVA fuel designs with UO_2 fuel and M5^{TM} cladding such that the criteria for rod pressure in these topical reports remain applicable to EPR. In addition, ANP-10285P states that the design basis is that the fuel system will not be damaged due to excessive internal pressure. These criteria are evaluated for the U.S. EPR fuel assembly design in the staff's Safety Evaluation of ANP-10285P.

Spacer Grids

In the U.S. EPR design, the top and bottom end spacer grids are HMP design, and the eight intermediate spacer grids are HTP design. The spacer grids are designed to maintain the fuel rods in a coolable geometry and permit control rod insertion during design basis events. ANP-10285P states that no crushing deformation will occur during normal operation and operating basis earthquake conditions based on crush load tests with load limits taken as the 95/95 one sided confidence of the mean elastic limit for grids at beginning of life (BOL) conditions correcting for operating temperature. These design bases meet the requirements of GDC 35 and GDC 27, respectively and conform to SRP Section II.1.C.v and Appendix A.III.1. The Appendix A design criteria for spacer grid strength states, therefore, that average values are appropriate, and the allowable crushing load P(crit) should be the 95-percent confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. The staff has reviewed the information contained in the FSAR and has determined that additional information is needed for closure of this issue. **RAI 318, Question 04.02-16, which is associated with the above request, is being tracked as an open item.**.

Fuel Assembly Structural Design

FSAR Tier 2, Section 4.2.1.5.1, “Nonoperational Loading,” states that the nonoperational loading, with dimensional stability, is 4 times gravity (g) axial and 6 g lateral. However, the bases for these loads were not presented by the applicant. The staff requested that the bases for these loads be provided. In an April 22, 2009, response to RAI 195, Question 04.02-5, the applicant stated that non-operational axial and lateral load limits are established to prevent fuel damage during shipping and handling. The applicant verified the ability of fuel assemblies to withstand loads up to these limits without plastic deformation or yielding of any assembly component with design analysis and testing. The 4 g 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 to prevent permanent yielding. The design allowable stress intensities of austenitic steels under normal operating conditions and AOOs are based on ASME Code, Section III, “Rules for Construction of Nuclear Facility Components,” and are given as: The lower of one-third the minimum tensile strength or two-thirds the minimum yield strength at room temperature; and one-third the tensile strength or 90 percent of yield strength at operating temperature. Stress limits for the austenitic steel components are defined in terms of the design allowable stress intensity. These stress limits conform to SRP Section 4.2, Section II.1.A.ii and are, therefore, acceptable. Consequently, the staff considers RAI 195, Question 04.02-5 resolved. The staff examined the U.S. EPR fuel assembly structural component stress limits and determined that they are consistent with SRP Section 4.2, Part II.1.A.ii. The staff concludes that these stress limits meet the ASME Code, Section III and are, therefore, acceptable.

The fuel assembly structural design criterion provided in FSAR Tier 2, Section 4.2.1.5.3, “Postulated Accident Conditions,” for components other than fuel rods or spacer grids states that worst-case loads for a combined seismic-LOCA event are based on the methods described in SRP Section 4.2, Appendix A. The FSAR further states that the loads must not result in deflections or deformation that can interfere with reactor shutdown or cooling of the fuel rods based on GDC 27, GDC 35, and 10 CFR 50.34. The staff has reviewed the information contained in the FSAR and has determined that additional information is needed for closure of this issue. **RAI 318, Question 04.02-16, which is associated with the above request, is being tracked as an open item.**

The U.S. EPR fuel assembly design bases also include allowances for axial growth between the top nozzle and fuel rod shoulder, and between the fuel assembly and core internals to maintain a positive clearance to prevent fuel rod and assembly bowing due to hard contact between these components. This conforms to the design bases acceptance criteria of SRP Section 4.2, Section II.1.A.v pertaining to fuel assembly dimensional growth to prevent hard contact and assembly bowing and allow control rod insertion as required by GDC 27. Therefore, the staff concludes that this design basis is acceptable.

The design basis contained in FSAR Tier 2, Section 4.2.3.5.5, "Assembly Liftoff," prohibits the holddown springs from being fully compressed during any AOO and ensures that the top and bottom nozzles maintain engagement with the reactor internals during AOOs and postulated accidents. In addition, holddown springs on the top nozzle are designed to ensure that contact is maintained between the fuel assembly and lower core plate during normal operation and AOOs, with the exception of the worst-case reactor coolant pump overspeed transient. This conforms to SRP Section 4.2 Part II.1.A.vii, which indicates that the worst case loads for normal operation should not exceed holddown capability, resulting in no assembly liftoff of the fuel assembly. SRP Section 4.2 Part II.1.A.vi also states, "because unseating a fuel bundle may challenge control rod/blade insertion, an evaluation of worst-case hydraulic loads should be performed for normal operation, AOOs, and accidents." Therefore, these design bases conform to SRP Section 4.2, Part II.1.A.vii and are acceptable to the staff, with the exception of an reactor coolant pump overspeed event. The exception of assembly liftoff for the reactor coolant pump over-speed event is addressed in Section 4.4 of this report.

The fuel assembly design basis does not permit buckling of the guide tubes, nor does it permit control rod drop times to become degraded beyond the criteria provided in ANP-10285P. Accordingly, these design bases conform to the requirement of GDC 27 pertaining to control rod insertion ability. Further, the fuel assembly design bases provide for safe configuration following design basis seismic events including operating basis earthquake (OBE), safe-shutdown earthquake (SSE) and LOCA, in compliance with the design requirements of GDC 27 and GDC 35 by maintaining core reactivity control and coolability. The adequacy of the design to meet these criteria is discussed below in Section 4.2.4.3 of this report.

Vertical overlap of fuel assembly spacer grids is also ensured by design, thus providing for proper design contact interface between adjacent assemblies.

The design bases for fuel rod bow is that it should be limited to prevent fuel rod failures per SRP Section 4.2 Part II.1.A.v. Therefore, there are no specific limits for fuel rod bow. Rather, the effects of rod bowing are included in those analyses that are impacted by bowing (such as departure from nucleate boiling) per SRP Section 4.2 Part II.3.C.ii, to prevent fuel rod failures. FSAR Tier 2, Section 4.2.3.5.6, "Fuel Rod Bow," discusses the limit for rod bow that is provided in BAW-10186P-A Revision 2 and applied to all the applicant's fuel designs to prevent fuel damage and failure. The staff concludes that this design bases conforms to the guidance contained in SRP Section 4.2 Part II.1.A.v and is acceptable to the staff.

The guide tube and spacer grid corrosion and hydride limits are the same as for the M5^{TM} fuel rod cladding that is discussed above. The corrosion performance of the M5^{TM} material used for the guide tubes is addressed in BAW-10227P-A. The corrosion and hydride limits of ANP-10285P are under staff review. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

The U.S. EPR fuel assembly structural design bases described in the FSAR are acceptable to the staff with the exception of the open items identified above, based on conformance to the guidance contained in SRP Section 4.2. Accordingly, the staff finds that the proposed acceptance criteria for the fuel structure, with the exception of the open items above, complies with GDC 27, GDC 35, and 10 CFR 50.34.

Rod Cluster Control and Stationary Control Assemblies

The design bases for the RCCAs include functional requirements on reactivity control, drop times and insertability, mechanical strength, and chemical compatibility are included in FSAR Tier 2, Section 4.3, FSAR Tier 2, Section 4.2.1.6, "Rod Cluster Control and Neutron Source Assemblies," and ANP-10285P. The design considers operating conditions, such as system pressure, temperature effects, loads, irradiation effects, and materials compatibility. The staff has determined that the design bases account for all the items identified in SRP Section 4.2 Part II.1.A viii, and finds that the RCCA design bases are in compliance with the requirements of GDC 27.

The design bases for the stationary control component assemblies (SCCAs) include mechanical strength, chemical compatibility, and resistance to radiation degradation and are similar to those given for the RCCA in FSAR Tier 2, Sections 4.2.1.6 and 4.2.2.10, "Stationary Control Component Assemblies Description." These design bases are the same as those given in SRP Section 4.2, Part II.1.A.viii. Therefore, staff finds the design bases for the SCCAs acceptable.

Surveillance Programs

FSAR Tier 2, Section 4.2.1.7, "Surveillance Programs," provides an overview description of the U.S. EPR fuel assembly and control rod surveillance program. The fuel assembly surveillance program includes overall examination for mechanical damage plus measurements of oxide thickness; fuel rod diameter, length, and bow; fuel shoulder gap; and overall assembly growth.

The RCCA surveillance program includes material wear examinations and verification of control rod tube integrity.

SRP Section 4.2, Parts I.4, II.4 and IV.2 recommend that an online system fuel monitoring system be addressed. FSAR Tier 2, Section 4.2.4.5, "Online Fuel System Monitoring," simply stated that this was addressed in FSAR Tier 2, Section 9.3.2. The staff requested that information be provided on actions taken when plant monitoring detects a fuel failure. In an April 22, 2009, response to RAI 195, Question 04.02-8, the applicant provided a summary of the actions that may be taken upon detection of a failed fuel assembly during operation including documentation of the applicant's corrective program with investigations and compensatory actions taken as appropriate, depending on the specific nature of the situation. The staff has determined that the monitoring and action plan conform to the guidance contained in SRP Section 4.2, Parts I.4, II.4 and IV.2. Therefore, the staff considers RAI 195, Question 04.02-8 resolved. The staff concludes that the surveillance program conforms to the guidance in SRP Section 4.2, Parts I.4, II.4 and IV.2 and is acceptable.

4.2.4.2 *Design Description*

A detailed description of the fuel assembly design is provided in ANP-10285P. The staff's review of the fuel assembly design description is contained in the staff's Safety Evaluation of

ANP-10285P. Only the FSAR Tier 2, Section 4.2 information that differs from the content of the topical report is addressed in the staff's review presented in the following sections of this report.

Fuel Assembly

The staff has compared the fuel assembly design description provided in FSAR Tier 2, Section 4.2.2 against the information contained in AREVA NP Topical Report ANP-10285P. The staff identified two data discrepancies, the first included fissile enrichment differences in FSAR Tier 2, Table 4.2-1, and Section 3.2 of ANP-10285P and, the second involved differences in the dimension from the bottom of the fuel column to the bottom of the fuel rod in FSAR Tier 2, Figure 4.2-2 and Figure 3-3 of ANP-10285P. The staff issued a question to the applicant requesting an explanation of these discrepancies. In an April 22, 2009, response to RAI 195, Question 04.02-9, these discrepancies were corrected by the applicant. The staff finds this response acceptable.

The information provided in the FSAR Tier 2, Section 4.2, both directly and by reference to ANP-10285P, provides an adequate description of the fuel assembly and its components and, therefore, meets the acceptance criteria for design description of SRP Section 4.2 Part II.2. This design description in the FSAR is acceptable to the staff for the reasons stated above. The adequacy of the fuel assembly design, as described, to perform its intended safety functions is described below in Section 4.2.4.3 of this report.

Rod Cluster Control Assemblies

FSAR Tier 2, Section 4.2.2.9, "Rod Cluster Control Assemblies Description," provides a description of the U.S. EPR control rods, referred to as rod cluster control assemblies. The U.S. EPR utilizes a total of 89 RCCAs, with each RCCA comprised of 24 individual control rods fastened to a central hub spider assembly. The individual control rods consist of a silver-indium-cadmium (80 percent Ag + 15 percent In + 5 percent Cd) absorber rod sealed within a stainless steel tube. A pre-loaded spring is positioned inside the central hub to absorb the kinetic energy of a reactor scram insertion of the RCCA. The application also describes a control rod upper end plug flex joint designed to accommodate misalignment of the RCCA with the fuel assembly guide tubes. However, a description and operational experience of this flex joint was not provided in this section. The staff requested additional design and operational information on the end plug flex joint. In an April 22, 2009, response to RAI 195, Question 04.02-10, the applicant noted that the end plug flex joint has been implemented in 13 U.S. plants without degradation or failures. The applicant provided the flex joint design information and further noted that there is one difference between the previous U.S. spider design and that used in the U.S. EPR spider design: The former design was cast stainless steel and the latter is a brazed spider. This response was reviewed by the staff and the staff determined that the design description was adequate in that it provided sufficient information for the staff to perform its own confirmatory analysis. Therefore, the staff considers RAI 195, Question 04.02-10 resolved.

The applicant provides information in FSAR Tier 2, Table 4.2-4 and FSAR Tier 2, Figure 4.2-12 describing the design of the RCCAs. The staff noted two data discrepancies and requested that the applicant correct the identified discrepancies. In an April 22, 2009, response to RAI 195, Question 04.02-11, the applicant noted that the dimensions in FSAR Tier 2, Table 4.2-4 were correct, those in FSAR Tier 2, Figure 4.2-12 were incorrect and that the latter would be corrected. The staff reviewed U.S. EPR FSAR Tier 2, Revision 1, Section 4.2, dated May 29, 2009, and the changes were not included in the submittal. **Inclusion of these changes to FSAR Tier 2, Figure 4.2-12 is being tracked as Confirmatory Item 04.02-11.**

The staff has reviewed the information provided in FSAR Tier 2, Table 4.2-3, "RCCA, Source Assembly, and TPA Component Materials," FSAR Tier 2, Table 4.2-4, "RCCA Characteristics and Design Parameters," and FSAR Tier 2, Figures 4.2-12, "Rod Cluster Control Assembly," FSAR Tier 2, Table 4.2-13, "RCCA Control Rod," and FSAR Tier 2, Table 4.2-14, "RCCA Spider," and determined that the U.S. EPR RCCA design description is acceptable.. The adequacy of the RCCA design, as described, to perform its intended safety functions is described below in Section 4.2.4.3 of this report.

Stationary Control Component Assemblies

FSAR Tier 2, Section 4.2.2.10, "Stationary Control Component Assemblies Description," describes the SCCAs. The SCCAs are utilized in the 152 fuel assemblies that do not interface with RCCAs.

There are three types of SCCAs in the U.S. EPR fuel system design, as follows: (1) Thimble plug assemblies, used to restrict flow in noninstrumented guide tubes and provide a guide ring for instrumented guide tubes; (2) primary source assemblies, which are similar in design to the thimble plug assemblies, except one of the rods contains a Cf-252 neutron source for reactor startup during the initial cycle and possibly the second cycle of operation; and (3) secondary source assemblies for which all 24 rods contain cold-pressed antimony-beryllium (Sb-Be) pellets for incore activation that are used as a startup source for subsequent cycles of operation.

Both the primary source assemblies and the secondary source assemblies are eventually replaced with thimble plug assemblies once a neutron source is no longer needed for reactor startup. The SCCAs are similar in geometric configuration as the RCCAs, as described in FSAR Tier 2, Tables 4.2-3, 4.2-4, and 4.2-6, "SCCA Characteristics and Design Parameters." The staff concludes that the FSAR provides an adequate description of the SCCA components, and conforms to the guidance contained in SRP Section 4.2 Part II.2, and is therefore acceptable. The adequacy of the SCCA design, as described, to perform its intended safety functions is described below in Section 4.2.4.3 of this report.

4.2.4.3 *Design Evaluation*

FSAR Tier 2, Section 4.2.3, "Design Evaluation," provides an evaluation of the U.S. EPR fuel assembly against the requirements and acceptance criteria described in Section 4.2.3 of this report. The design evaluation addresses the fuel rod cladding, the fuel pellet, overall fuel rod performance, the spacer grids, and the fuel assembly. The performance of the RCCAs and SCCAs is also addressed. Where applicable, reference is made to the staff's Safety Evaluation of ANP-10285P.

Fuel Rod Cladding

Flow-induced fuel rod vibration can cause a bending stress and fretting wear on the fuel rod at a spacer grid contact point. The applicant addresses fuel rod vibration and its associated fretting and cladding stresses in FSAR Tier 2, Sections 4.2.3.1.1, "Vibration Analysis," FSAR Tier 2, Section 4.2.3.1.2, "Fuel Rod Internal and External Pressure and Cladding Stresses," and FSAR Tier 2, Section 4.2.3.5.7, "Fuel Rod Fretting," and in Topical Report ANP-10285P, Section 5.1.4.

The potential for chemical reaction of the fuel rod cladding is addressed in the FSAR through reference to BAW-10227P-A. In the staff's safety evaluation included in BAW-10227P-A, the staff concluded that the overall corrosion performance of the M5™ cladding is better than the Zircaloy-4 material. Therefore, M5™ cladding corrosion performance is acceptable up to 62 GWD/MTU rod average burnup. BAW-10227P-A has been approved by the staff for application to the U.S. EPR per ANP-10263P-A. The FSAR states that the predicted maximum oxide thickness is much less than the 100 micron design basis limit as described in Section 4.2.4.1 above. Fuel rod cladding corrosion and hydrogen absorption are addressed in FSAR Tier 2, Section 4.2.3.1.3, "Potential for Chemical Reaction," and in Topical Report ANP-10285P. In its review of ANP-10285P, the staff requested hydrogen data to demonstrate acceptable performance in terms of hydrogen pickup of the cladding. Hydrogen pickup data were provided that demonstrated very low hydrogen pickup up to 62 GWD/MTU rod average burnup. Therefore, the staff concludes that the hydrogen pickup levels are acceptable. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P, which is currently under review. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

There are no specific design criteria for crevice corrosion in fuel rod cladding in SRP Section 4.2. However, FSAR Tier 2, Section 4.2.3.1.4, "Fretting and Crevice Corrosion," addresses this failure mechanism by stating that crevice corrosion is not a likely corrosion mechanism for zirconium alloy material and that tight reactor water chemistry controls will limit the amount of detrimental dissolved oxygen and chlorine in the coolant. The staff requested that post irradiation examination results be provided to demonstrate whether M5™ fuel rod cladding use in the U.S. EPR will not undergo crevice corrosion. In an April 22, 2009, response to RAI 195, Question 04.02-12, the applicant provided information that demonstrated that crevice corrosion will likely not be limiting in the U.S. EPR fuel based on the low M5™ material uniform corrosion rates and the fact that no crevice corrosion has been observed during post irradiation examinations of M5™ cladding. The staff finds the response to be acceptable because crevice corrosion has not been shown to be a problem for M5™ in-reactor performance for current operating plants that are similar to the U.S. EPR. Therefore, the staff considers RAI 195, Question 04.02-12 resolved.

Stress corrosion cracking of the M5™ fuel rod material is addressed in the applicant's licensing Topical Report BAW-10227P-A. In a Safety Evaluation Report on BAW-10227P-A, the staff concluded that test data provided by the applicant demonstrate that the M5™ cladding is less susceptible to stress corrosion cracking than the applicant's Zircaloy-4 cladding. BAW-10227P-A has been approved by the staff for application to the U.S. EPR per ANP-10263P-A. Accordingly, the staff finds that M5™ cladding is acceptable in relation to stress corrosion cracking. Stress corrosion cracking in fuel rods is often referred to as pellet-cladding-interaction (PCI) such that cracking is chemically assisted by corrosive elements such as Iodine at the crack tip. SRP Section 4.2 Part II.1.B.vi, states that PCI is generally prevented by the cladding strain and fuel melting criteria discussed below. The staff's evaluation of both cladding strain and fuel melting will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

FSAR Tier 2, Section 4.2.3.1.6 states that fuel rod cladding fatigue usage was evaluated utilizing the methods described in the ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," for stresses, and BAW-10227P-A. Fuel rod cladding fatigue associated with normal operations, upset, and test transients is evaluated in Topical Report ANP-10285P, Section 5.1.3. In FSAR Tier 2, Section 4.2.3.1.6, the applicant reports that its analyses show that the cumulative fatigue usage factor is well within the allowable design limit. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

Fuel rod cladding collapse is addressed in licensing Topical Report ANP-10285P, Section 5.2.2. ANP-10285P also addresses fuel rod cladding collapse, and references an NRC-approved methodology in BAW-10227P-A. ANP-10285P states that the creep rate for the M5™ cladding is approximately 50 percent 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. The staff requested that the applicant clarify the creep model applied for this analysis and verify that the creep collapse lifetime is greater than the fuel rod design burnup limit. In an April 22, 2009, response to RAI No 195, Question 04.02-4, the applicant stated that the creep rate of M5™ is approximately 67 percent of Zircaloy-4 and that a factor of 0.9 will be conservatively applied to the Zircaloy-4 model in calculating M5™ material creep for creep collapse analyses. The staff has determined this approach to be conservative and acceptable. Therefore, the staff finds that the design criteria are met up to the 62 GWd/MTU burnup limit. **Inclusion of these changes to FSAR Tier 2, Section 4.2.3.1.11 is being tracked as Confirmatory Item 04.02-4.** In FSAR, Tier 2 Section 4.2, the applicant stated that this is conservative in relation to the creep collapse analysis (i.e., collapse is calculated at shorter times) with the higher creep rate. The staff is reviewing the information provided by the applicant in FSAR Tier 2, Section 4.2. The topical report states that the fuel rod cladding creep collapse lifetime is greater than the maximum design burnup of 62 GWD/MTU. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P**. RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

Fuel rod cladding strain is evaluated in licensing Topical Report ANP-10285P, Section 5.1.2. The FSAR and the topical report state that the transient linear heat generation rates (LHGRs) that induce a one percent cladding strain are much greater than the fuel rod is expected to experience. The COPERNIC code is used to determine the linear heat generation rate limits for cladding strain to determine that the one percent cladding strain limit is met for AOO events. In its review of the fuel rod design contained in ANP-10285P, the staff identified an issue related to the prediction of cladding strain and, therefore, the applicant may underpredict cladding strain and overpredict the LHGR limits for the U.S. EPR fuel design. This issue of fuel rod cladding strain will be evaluated in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

Fuel Pellet

FSAR Tier 2, Section 4.2.3.1.4 states that irradiation stability of the U.S. EPR fuel pellet is confirmed by analyses using the COPERNIC code but no examples were provided demonstrating the acceptable stability of the pellet. The staff requested a description and results of the COPERNIC analyses where irradiation stability is important. In an April 22, 2009, response to RAI 195, Question 04.02-13, the applicant stated that the stability of the fuel pellet during its irradiation lifetime was evaluated with the COPERNIC code (BAW-10231P-A) as provided in ANP-10285P. The response further stated that the effects of fuel densification and fission product swelling are included in the COPERNIC cladding strain, fuel rod internal pressure, and fuel centerline melt analyses. The inclusion of fuel stability in these analyses meets the criteria in SRP 4.2 Sections II.1.B.ii, II.3.C.i, II.3.C.ii and II.3.C.iv and RG 1.126 that calls for fuel densification to be accounted for in cladding strain, cladding collapse, fuel stored energy, fuel rod internal pressure, and fuel centerline melt analyses. The staff's evaluation of cladding strain will be provided in the Safety Evaluation Report on ANP-10285P**. RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

Overheating of the fuel pellet is addressed in licensing Topical Report ANP-10285P, Section 5.2.4. FSAR Tier 2, Section 4.2.3.2.3 states that the fuel pellet will not melt because the linear heat generation rate does not exceed the limit established in the centerline melt analysis. The applicant used the COPERNIC code (BAW-10231P-A) to verify that this design criterion is met by determining the linear heat rate limits for fuel melting for AOO events. In addition, FRAPCON 3.3 (NUREG/CR-6534 Vol. 4, "FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties") audit calculations have been performed by the staff that verify these LHGR limits are conservative for AOO events. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Fuel Rod Performance

Fuel rod performance calculations are performed with the COPERNIC code, BAW-10231P-A. The use of COPERNIC has been approved by the staff for application to the U.S. EPR for the reasons documented in ANP-10263P-A. The phenomenological models provided in the COPERNIC code, as identified in FSAR Tier 2, Section 4.2.3.3.1, "Fuel Rod Performance Predictions," conform to the guidelines of SRP Section II.3.C.i. The fuel design and operational parameter ranges specified in BAW-10231P-A have been verified by the staff as applicable to the U.S. EPR fuel by comparison of current operating PWRs and U.S. EPR operational parameters. The fuel rod performance analyses are described in Section 5.4.4 of ANP-10285P. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

The FSAR addresses the potential for fuel rod cladding rupture due to a rapid temperature transient on waterlogged fuel pellets. The applicant states that such fuel rod failure is of very low probability, based primarily on consideration of the nominal 96 percent theoretical pellet material density. The staff concurs with the applicant's assessment and its conclusion that the impact of waterlogged fuel in the U.S. EPR will be no more adverse than it is for current operating plants because the UO_2 fuel and its operating parameters for the U.S. EPR and current plants are similar.

Spacer Grids Evaluation

The methodology used to evaluate spacer grid loads is described in BAW-10133P-A-01, and the maximum load are described in ANP-10285P. The spacer grids are evaluated by the applicant for consequential damage due to external forces in FSAR Tier 2, Section 4.2.3.4, "Spacer Grids Evaluation," and licensing Topical Report ANP-10285P, Section 5.3.4. The FSAR states that the combined seismic plus LOCA load results in no grid deformation, and because the grids remain elastic, core coolable geometry is maintained. Spacer grid slip forces are also addressed in FSAR Tier 2, Section 4.2.3.4.2, "Spring Loads for Grids," and licensing Topical Report ANP-10285P, Section 5.1.1.5. The data reported in topical reports are used as input to the fuel assembly analysis models. The staff's evaluation of spacer grid loads will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Fuel Assembly Design Evaluation

FSAR Tier 2, Section 4.2.3.5, "Fuel Assembly Design Evaluation," identifies the design bases and the analytical methodologies applicable to the fuel assembly design evaluation. The identified design bases and analytical methodologies are referenced as applicable to the U.S. EPR fuel system design in either ANP-10263P-A or ANP-10285P. The applicability of BAW-10133P-A; ANF-89-060P-A, Supplement 1, "Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer," Siemens Power Corporation, February 1991; BAW-10172P-A, Rev 0, "Mark-BW Mechanical Design Report," Babcock & Wilcox, December 1989; and BAW-10239P-A, Revision 0, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," Framatome ANP, Inc., October 2004, to the U.S. EPR will be evaluated in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

FSAR Tier 2, Section 4.2.3.5 states that the design bases and design limits for the U.S. EPR fuel assembly are essentially the same as those for previously licensed designs. The FSAR refers to ANP-10285P for the detailed fuel assembly design evaluation, stating that fuel assembly evaluations show that the stresses are lower than the material allowable stresses for both normal operation and faulted conditions. ANP-10285P Section 5.1.1 Topical Report provides the applicable fuel assembly design evaluation as described in FSAR Tier 2, Section 4.2.3.5. In its review of ANP-10285P, the staff noted that the calculated stresses for the loads induced during shipping on the bottom and top nozzles show an extremely low margin to the stress limits. Actual load tests were performed on the bottom nozzles that demonstrated a significant margin to the point of plastic deformation. However, a finite element analysis (FEA) was presented that suggested a small margin for the top nozzle in the absence of testing. In its review of ANP-10285P, the staff requested additional details in its review of ANP-10285P from the applicant on the FEA used to determine that the stresses calculated for the top nozzle meets the stress limits and was performed in a satisfactory manner. This issue will be evaluated in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

FSAR Tier 2, Section 4.2.3.5.2, "Analysis of Combined Shock and Seismic Loading," provides a description of the analysis of combined seismic and LOCA loading and notes that the maximum loads meet the design load limits and remain in the elastic region. The analyses in this section address both horizontal (LOCA and seismic) and vertical (LOCA) loadings, concluding that the U.S. EPR fuel assembly meets the design criteria and remains in the elastic regime for both the safe-shutdown earthquake and SSE plus LOCA events.

ANP-10285P, Section 5.3.4 also provides the calculated loads and the 95/95 lower load limit for spacer grids based on crush and impact load tests of spacer grids for SSE plus LOCA events. Important inputs for these analyses, which are design dependent and specific to the U.S. EPR design, are the assembly stiffness and damping constants used in this analysis. The actual stiffness and damping constants and how they were determined are not presented in either FSAR Tier 2, Section 4.2 or ANP-10285P such that the staff could not evaluate whether they accurately represent assembly response to SSE plus LOCA events. Accordingly, the staff requested stiffness and damping constants used in the SSE analyses and a step-by-step summary of how they were determined from the shaker, stiffness, drop lateral pluck and impact tests. **RAI 318, Question 04.02-16, which is associated with the above request, is being tracked as an open item.**

The FSAR Tier 2, Section 4.2.1.1.2, “Stress-Strain Limits of Cladding,” states that maximum shear theory is used to evaluate the stresses in M5™ guide tubes. The staff requested that the use of maximum shear stress theory to evaluate the guide tubes be justified. In an April 22, 2009, response to RAI 195, Question 04.02-6, the applicant stated that 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. The SRP 4.2 Section II.1.A.i states that “stress limits that are obtained by methods similar to those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) (Reference 8) are acceptable.” Therefore, the use of maximum shear stress theory is consistent with SRP Section 4.2. In addition the applicant’s response noted that the guide tubes are shown not to buckle and remain elastic during normal operation and anticipated operational occurrences. The calculated guide tube stress intensities under faulted conditions are shown to be within design limits, as presented in Table 5-17 of ANP-10285P. The staff reviewed the applicant’s responses and finds the provided information acceptable and, therefore, considers RAI 195, Question 04.02-6 resolved.

FSAR Tier 2, Section 4.2.3.5.4, “Axial Growth,” addresses fuel assembly axial growth, including fuel rod-to-top nozzle, top nozzle-to-core internals, and total fuel assembly and fuel rod growth. The results of the applicant’s evaluation are provided by reference to ANP-10285P. ANP-10285P Section 5.1.7 provides the fuel assembly axial growth evaluation as described in the FSAR. The U.S. EPR fuel assembly growth is governed by M5™ guide tube growth. Recent data from fuel assemblies with M5™ guide tubes irradiated in Three Mile Island (TMI) Unit 1 and Catawba Unit 1 have suggested that the methods used to predict M5™ guide tube growth may significantly underpredict M5™ guide tube growth. If adequate clearances are not maintained between the fuel assembly and core internals, this can result in guide tube bowing or buckling that may impede control rod insertion. This issue of the methods for predicting M5™ guide tube growth for use in evaluating assembly to U.S. EPR core internal clearances is addressed in the staff’s Safety Evaluation of ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

FSAR Tier 2, Section 4.2.3.5.5, “Assembly Liftoff,” provides an evaluation of fuel assembly holddown. The FSAR states that the fuel assembly will maintain in contact with the lower core support plate during normal operation and AOOs, except for the bounding condition of a reactor coolant pump overspeed transient during which minimal liftoff will occur. In addition, the applicant states that assembly liftoff does not fully compress the holddown spring for any AOO and the top and bottom nozzles remained engaged with the reactor internals during all AOOs and postulated accidents. The staff was concerned with assembly liftoff for the pump overspeed event that could possibly impact control rod insertion in relation to GDC 27. Therefore, the staff requested that the impact of the U.S. EPR assembly including the FUELGUARD™ nozzle on liftoff and its impact on control rod insertion be evaluated. In an April 22, 2009, response to RAI 195, Question 04.02-7, the applicant stated that because the bottom FUELGUARD™ nozzle remains engaged with the lower core plate pins during transient and accident conditions including the coolant pump overspeed event, the guide tubes will remain aligned with the control rods such that control rod insertion is assured. RAI 195, Question

04.02-7 also requested information regarding assembly liftoff. The applicant's response indicated that this analysis is presented in FSAR Tier 2, Section 4.4. The staff concurs and the associated review of assembly liftoff is addressed in Section 4.4 of this report.

Fuel rod bow is addressed in FSAR Tier 2, Section 4.2.3.5.6, “Fuel Rod Bow,” and in ANP-10285P Section 5.1.6. The staff has identified that little rod bow data exists for fuel rods with M5 cladding and no data exists for designs similar to those for the EPR (14 foot rods and the same spacer grids). The staff’s evaluation will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Reactivity Control, Neutron Source, and Thimble Plug Assemblies

FSAR Tier 2, Section 4.2.3.6.1, “Internal Pressure and Cladding Stresses,” describes the analysis of the internal gas pressure and cladding material stresses of the RCCA and neutron source rods.

The FSAR states that because the Ag-In-Cd absorber material does not generate gases during irradiation, there is no source of gas generation that could pressurize the RCCA rods. As a result, the cladding stresses calculated by the applicant for normal operation, AOOs, and postulated accidents are within acceptable limits. Similarly, the FSAR states that cladding stresses for the neutron source rods remain below the stress limits. However, FSAR Tier 2, Section 4.2.3.6.1 does not provide details about the rod pressures and analysis stresses or how the latter compare to the stress limits. The staff requested that these stress levels and their comparison to stress limits be provided. In an April 22, 2009, response to RAI 195, Question 04.02-14, the applicant provided calculated RCCA cladding stress results that demonstrate adequate safety margins for normal operation, AOOs, and postulated accidents. The staff reviewed the information provided and finds the response satisfactory because stress limits were met, and concludes that there is reasonable assurance that the integrity of the RCCA cladding will be maintained during design basis operating conditions. Therefore, the staff considers RAI 195, Question 04.02-14 resolved.

FSAR Tier 2, Section 4.2.4.1.4 notes that more than 4,600 ion-nitrated HARMONI™ RCCAs have been delivered to operating reactors, including 572 of these in U.S. PWRs (similar to those proposed for the U.S. EPR). However, the performance of these RCCAs was not discussed. The staff requested that the reactor facility, number of RCCAs delivered and year of installation be provided along with a description of any differences between these past RCCAs and those proposed for the U.S. EPR. A description of any failures or degradations was also requested. The staff has reviewed the design comparisons and the operational and inspection results for the HARMONI™ RCCA components provided by the applicant in an April 22, 2009, response to RAI 195, Question 04.02-15. Since the provided information demonstrates no operational problems for RCCAs similar to those in the U.S. EPR design in current operating PWRs, the staff determined there is reasonable assurance that the U.S. EPR control rods will perform as designed and meet the requirements of GDC 27 relating to control rod insertability. Therefore, the staff considers RAI 195, Question 04.02-15 resolved.

4.2.4.4 *Testing and Inspection Plan*

FSAR Tier 2, Section 4.2.4, "Testing and Inspection Plan," describes the operating experience, prototype testing, and component testing of the U.S. EPR fuel system. In addition, the new fuel testing and inspection plans are discussed, as is the post-irradiation surveillance program.

Where applicable, reference is made to the staff's evaluation of the U.S. EPR fuel system testing and inspection plan, as will be contained in the Safety Evaluation Report on ANP-10285P.

Operating Experience

FSAR Tier 2, Section 4.2.4.1, "Operating Experience," describes the overall operating experience with fuel incorporating elements of the U.S. EPR fuel assembly design, including construction techniques, components, and material. The operating experience of the HTP spacer grid design, the M5^{TM} material, and the 14 foot assembly length is addressed in ANP-10285P Sections 4.1, 4.2, 4.3, and 4.4. The staff's evaluation will be provided in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Fuel Assembly Prototype Testing

FSAR Tier 2, Section 4.2.4.2, "Fuel Assembly Prototype Testing," provides a summary of the testing that was performed on a full-size prototype fuel assembly, including static tension and compression tests, lateral stiffness tests, shaker tests, lateral pluck tests, and vertical drop tests. In addition, fuel assembly hydraulic flow testing and control rod drive mechanism (CRDM) and RCCA testing were performed by the applicant. ANP-10285P Section 3.8, provides the details of these tests. The staff's evaluation is provided in the staff's Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Fuel Assembly Component Testing

FSAR Tier 2, Section 4.2.4.3, "Fuel Assembly Component Testing," provides a summary of the testing of individual components of the fuel assembly that was performed, as described in this section of the FSAR. The spacer grids, upper guide tube connection, bottom nozzle, and fuel assembly holddown springs were tested, as detailed in ANP-10285P Sections 3.9 and 5.1. The staff's evaluation is provided in the staff's Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

Testing and Inspection of New Fuel

FSAR Tier 2, Section 4.2.4.4, "Testing and Inspection of New Fuel," discusses the applicability of the applicant's Quality Assurance program to the manufacture and inspection of the U.S. EPR fuel assemblies, including quality control inspection, auditing, supplier specifications, design verification, and end product nondestructive examination. The staff determined the general provisions of the new fuel testing and inspection program to be acceptable because it conforms to SRP Section 4.2, Part II.4.A. The staff's review of the applicant's quality assurance program is documented in Chapter 17, "Quality Assurance," of this report. The staff is evaluating the need for additional information to include testing and inspection of new fuel. The staff is considering the need for additional information in connection with its review of Topical Report ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.**

On-Line Fuel System Monitoring

The U.S. EPR includes a nuclear sampling system designed to take liquid and gaseous samples from the primary coolant system for analysis. This system is described in FSAR Tier 2, Section 9.3.2, "Process Sampling Systems," and is addressed in Section 9.3.2 of this report.

Post-Irradiation Surveillance

FSAR Tier 2, Section 4.2.4.6, "Post-Irradiation Surveillance," describes the applicant's plans for post-irradiation examination (PIE) of its fuel and RCCA products, including onsite poolside inspections and measurements and laboratory hot cell examinations. The staff recommends such examinations and inspections as a means for improving operational fuel reliability.

4.2.5 Combined License Information Items

There are no combined license (COL) information items from FSAR Tier 2, Table 1.8-2 that affect this section.

4.2.6 Conclusions

The staff has reviewed the U.S. EPR fuel system design as submitted by the applicant in the U.S. EPR design certification FSAR Tier 2, Section 4.2. The staff's review has addressed the design bases; design description; design performance; and testing, inspection, and surveillance plans for the fuel and nonfuel components that comprise the U.S. EPR fuel system.

For the reasons set forth above, and with the exception of the open items identified above, the staff concludes that the U.S. EPR fuel system has been designed so that it will not be damaged as a result of normal operation and anticipated operational occurrences, and that fuel damage that may occur during postulated accidents will not be severe enough to prevent control rod insertion or impede core cooling. Further, with the exception of the open items identified above, the applicant has demonstrated compliance with 10 CFR Part 50 Appendix A, GDC 10, GDC 27, and GDC 35, (per Regulatory Basis Section 4.2.3 above).as follows:

- GDC 10: The reactor fuel system is designed with appropriate margin such that specified acceptable fuel design limits are not exceeded during conditions of normal plant operation and anticipated operational occurrences.
- GDC 27: The RCCA control rod mechanical design satisfies the stress limit design criteria, and control rod insertion will not be impeded.
- GDC 35: The reactor fuel system is designed such that the performance of the Emergency Core Cooling System will not be compromised following a postulated accident.

In addition, for the same reasons, the staff finds the fuel design will not compromise ECCS performance with respect to coolability and that 10 CFR 50.46(b)(4), "Coolable Geometry," is met in this regard.

Therefore, the regulatory requirements of 10 CFR 50 Appendix A, GDC 10, GDC 27, GDC 35, 10 CFR 50.46, and 10 CFR 52.47 are met in relation to fuel mechanical design.

The staff's conclusions are based on the following:

- The applicant has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analysis results. The analyses provided by the applicant were performed in accordance with applicable regulatory guidelines, such as RG 1.126, and utilizing applicable methods that have been previously reviewed and approved by the staff.

- The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading and has made a commitment to a post-irradiation surveillance program to monitor fuel performance and support product improvement.

- Most of the review of the U.S. EPR fuel mechanical design is performed by the staff as part of the review of ANP-10285P. The details of the open items tracked in **RAI 339, Question 04.02-17** and the resolution of these issues will be documented in the staff's Final Safety Evaluation Report on ANP-10285P.

4.3 Nuclear Design

4.3.1 Introduction

The objectives of the nuclear design of the fuel assemblies, control systems, and reactor core are to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core.

4.3.2 Summary of Application

FSAR Tier 1: The application identifies no FSAR Tier 1 items specifically related to the reactor core nuclear design. However, FSAR Tier 1, Section 2.2.6, "Chemical and Volume Control System"; Section 2.2.7, "Extra Borating System (EBS)"; Section 2.4.1, "Protection System (PS)"; Section 2.4.11, "Boron Concentration Measurement System"; Section 2.4.17, "Excore Instrumentation System"; and Section 2.4.19, "Incore Instrumentation System," describe requirements associated with the nuclear design.

FSAR Tier 2: The applicant has provided a description of the U.S. EPR nuclear design in FSAR Tier 2, Section 4.3, summarized here, in part, as follows:

A summary description of the U.S. EPR reactor design is contained in FSAR Tier 2, Section 4.1, "Summary Description."

FSAR Tier 2, Section 4.3, provides the design basis for the U.S. EPR, followed by a description of the nuclear design, including power distributions, reactivity coefficients, control requirements related to control rod position as a function of power, control rod worth, core stability considerations, vessel irradiation, and analytical methods applications used in the analysis.

The U.S. EPR reactor core is comprised of 241 fuel assemblies arranged in an approximate cylindrical pattern within the reactor pressure vessel. Each fuel assembly is made up 265 fuel rods plus 24 guide tubes configured in a 17x17 array. Fuel rods are loaded with enriched UO_2 and UO_2 + Gd_2O_3 ceramic fuel pellets. The fuel assembly is held together with 10 spacer grids and top and bottom nozzle fittings. The guide tubes are provided for insertion of control rods or incore instrumentation. The reactor has a total of 89 rod cluster control assemblies grouped into 4 control banks and 3 shutdown banks.

The U.S. EPR reactor also has a steel heavy reflector positioned between the exterior perimeter of the core and the core barrel. The purpose of the heavy reflector is to reduce fast neutron leakage and vessel irradiation, and to flatten radial core power distribution.

The applicant states that the nuclear design presented in FSAR Tier 2, Section 4.3 is for the initial core and is based on an 18-month cycle of operation. The maximum fuel rod burnup is stated to be 62 GWD/MTU.

A description of the power distributions, including hot channel factors, measurement, and operational limits is provided. The U.S. EPR core power distributions have been calculated by the applicant over a broad range of conditions. The applicant describes how the fixed incore instrumentation system is used to continuously monitor axial and radial core power distributions and how the aeroball movable incore detectors are used to calibrate the fixed incore detectors.

FSAR Tier 2, Section 4.3 also describes the various reactivity coefficients. The fuel temperature or Doppler coefficient provides negative reactivity feedback. The moderator temperature coefficient is negative at power. However, at the beginning of a cycle and below 50 percent power the moderator temperature coefficient may be slightly positive. A power coefficient is also defined as the combined effect of the moderator temperature coefficient and the fuel temperature coefficient. The power coefficient is negative throughout core lifetime.

Reactivity control measures are also described. The applicant states that soluble boron is added to the reactor coolant to reduce reactivity so that enough control rods remain available to meet shutdown margin (SDM) requirements. Control rod insertion is limited while operating in order to ensure shutdown capability. Control rod insertion limits are specified in limiting conditions for operation (LCO) in the plant Technical Specifications contained in FSAR Tier 2, Chapter 16 as Technical Specification 3.1.6, “Control Bank Insertion Limits.”

The applicant states that, because of the negative power coefficient (sum of Doppler and moderator coefficients), the reactor is inherently resistant to core power oscillations. However, a discussion of xenon-induced power distribution oscillations is provided in FSAR Tier 2, Section 4.3.2, “Description.”

FSAR Tier 2, Section 4.3.2 also describes the analytical methods used to generate nuclear cross section data, calculate three dimensional core power distributions, and infer reactor power distributions from incore neutron flux measurement. Reference is made to several licensing topical reports that document validation of the analytical methods.

ITAAC: As discussed above, no FSAR Tier 1 items explicitly related to reactor nuclear design were identified. The following ITAAC identified in FSAR Tier 1, however, relate to FSAR Tier 2, Section 4.3 and are listed below for reference:

- FSAR Tier 1, Section 2.4.17, ITAAC 4.2, Excore instrumentation output signal

- FSAR Tier 1, Section 2.4.19, ITAAC 4.2, Incore instrumentation output signal

The excore and incore instrumentation are tested during initial plant testing, as described in FSAR Tier 2, Section 14.2, “Initial Plant Test Program.”

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 4.3 are given in FSAR Tier 2, Section 16.2.0, “Safety Limits”; Section 16.3.1, “Reactivity Control Systems”; Section16.3.2, “Power Distribution Limits”; Section 16.3.3, “Instrumentation”; and Section 16.4.2, “Reactor Core.”

4.3.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 4.3 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 4.3 of the SRP.

1. GDC 10 requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences.

2. GDC 11, “Reactor Inherent Protection,” requires that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

3. GDC 12, “Suppression of Reactor Power Oscillations,” requires, in part, that the reactor core be designed to assure that power oscillations that could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

4. GDC 13, “Instrumentation and Control,” requires, in part, provision of instrumentation to monitor variables and systems that can affect the fission process over their anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and controls to maintain the variables and systems within prescribed operating ranges.

5. GDC 20, “Protection System Functions,” requires, in part, that the protection system be designed to provide automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to initiate operation of systems and components important to safety occurs under accident conditions. There are usually primary and secondary independent reactivity control systems.

6. GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” requires, in part, that no single malfunction of the RCSs (this does not include rod ejection or dropout) causes acceptable fuel design limits to be exceeded.

7. GDC 26, “Reactivity Control System Redundancy and Capability,” requires, in part, that two independent reactivity control systems of different design be provided, one of which uses control rods, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. The system that uses control rods must be capable of reliably controlling reactivity changes under anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

8. GDC 27 requires, in part, that the RCSs have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

9. GDC 28, "Reactivity Limits," requires, in part, that the reactivity control systems be designed with limits on the amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

10. GDC 14, "Reactor Coolant Pressure Boundary," as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the reactor coolant pressure boundary, in part, insofar as it considers calculations of neutron fluence.

11. GDC 31, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, in part, insofar as it considers calculations of neutron fluence.

12. Appendix G, to 10 CFR Part 50, as it relates to reactor pressure vessel material fracture toughness requirements, in part, insofar as it considers calculations of neutron fluence.

13. Appendix H, to 10 CFR Part 50, as it relates to reactor pressure vessel material surveillance program requirements, in part, insofar as it considers calculations of neutron fluence.

14. 10 CFR 50.61 as it relates to fracture toughness criteria for PWRs relevant to Pressurized Thermal Shock (PTS) events, in part, insofar as it considers calculations of neutron fluence.

SRP 4.3 and SRP 5.3 provide acceptance criteria adequate to meet the above requirements include:

1. There is a reasonable probability that the proposed design limits can be met for the expected range of reactor operation, taking into account analysis uncertainties and special problems such as fuel densification, core asymmetries, and misaligned control rods.

2. There is a reasonable probability that during normal operation the design limits will not be exceeded based on consideration of reactor core monitoring instrumentation, alarms, and calculation uncertainties.

3. NUREG-0085, "The Analysis of Fuel Densification," July 1976, for acceptance criteria for power spiking.

4. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," March 2007, as related to the confirmation of the nuclear design.

5. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

6. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

4.3.4 Technical Evaluation

The staff has reviewed FSAR Tier 2, Section 4.3, "Nuclear Design," including applicable TS, related ITAAC and the applicable portions of the initial test program, and the COL information items given in FSAR Tier 2, Table 1.8-2, to determine the acceptability of the design. An evaluation of the FSAR Tier 2 information was performed against the requirements of 10 CFR Part 50 Appendix A, GDC 10, GDC 11, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28, and 10 CFR 52.47(b)(1), "Contents of applications; technical information," as it pertains to ITAAC for design certification. The initial test program in FSAR Tier 2, Section 14.2, describes initial fuel loading and precriticality testing, initial criticality and low power physics testing, and power ascension testing, all of which are intended, in part, to confirm the nuclear design described in FSAR Tier 2, Section 4.3.

The FSAR Tier 1 sections identified in FSAR Tier 2, Section 4.3.2, "Description," were also reviewed. The staff confirmed that the applicable information contained in the FSAR Tier 1 sections conforms to the nuclear design descriptions contained in FSAR Tier 2, Section 4.3. There are no COL information items applicable to FSAR Tier 2, and none were determined to be necessary as a result of the staff's review.

The limiting conditions for operation specified in FSAR Tier 2, Chapter 16 TS 3.1 and FSAR Tier 2, Chapter 16 TS 3.2 propose requirements pertaining to reactivity and power distribution control, which establish the initial conditions for analyses described in FSAR Tier 2, Chapter 15, "Transient and Accident Analysis." These TS requirements would ensure that the plant can be operated, as designed, in compliance with the requirements of GDC 10, GDC 12, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28. The instrumentation specifications contained in FSAR Tier 2, Chapter 16 TS 3.3.1 provide the operability and setpoints requirements for the protection system that are taken in the analyses described in FSAR Tier 2, Chapter 15. These instrumentation specifications ensure compliance with GDC 10, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28. Technical Specification 4.2 specifies core-related design features which conform to the descriptions contained in FSAR Tier 2, Section 4.3.

In addition to the material provided in FSAR Tier 2, Section 4.3, the review was extended to the applicant's referenced licensing topical reports, as necessary to evaluate the design.

4.3.4.1 *Evaluation of Design Bases*

FSAR Tier 2, Section 4.3.1, "Design Bases," describes the bases for the nuclear design, including applicability of GDC 10, GDC 11, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28.

The design bases for the nuclear design are addressed in the following paragraphs.

The peak fuel rod burnup limit is 62 GWD/MTU, based on the information provided in BAW-10231P-A and BAW-10186P-A. In FSAR Tier 2, Section 4.3.1.1, "Fuel Burnup," the applicant states that the maximum fuel assembly average burnup is established such that the peak fuel rod burnup is within the limits described in ANP-10285P. The staff has not yet completed its review of ANP-10285P and, therefore, has made no finding regarding fuel assembly burnup limit methodology in ANP-10285P. **RAI 339, Question 04.02-17, which is**

associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item.

In addition, the staff notes that FSAR Tier 2, Table 4.3-1, "Core Design Criteria," lists design peak rod burnup of 55 GWD/MTU for gadolinia rods. The staff questioned the technical basis for this burnup limit and issued RAI 134, Question 04.03-7. In a January 29, 2009, response to RAI 134, Question 04.03-7, the applicant stated that the COPERNIC fuel rod design computer code is used to verify that the fuel rod designs for the U.S. EPR meet design and safety criteria. Page 1-2 of Topical Report BAW-10231P-A states that COPERNIC is applicable to a gadolinia fuel rod average burnup of up to 55 GWD/MTU. This criterion is the basis for the information in FSAR Tier 2, Table 4.3-1. Since the fuel rods and in particular the fuel rods containing gadolinia which will be used in the U.S. EPR are the same as the fuel rod design assessed during the COPERNIC assessment, the results of the analyses using COPERNIC are applicable to the U.S. EPR. The response is acceptable, since it references a previously reviewed and approved code. Therefore, the staff considers RAI 134, Question 04.03-7 resolved.

Peak fuel rod exposure will be limited to 62.0 GWD/MTU, for which the staff has approved the use of the COPERNIC Fuel Rod Design Code for operating PWRs. This limit is independent of the 55 GWD/MTU limit for gadolinia rods described above. The maximum assembly average burnup is chosen such that the peak rod burnup is within limits.

Core design lifetime is between 12 months and 24 months, depending on cycle design.

The design basis for the core reactivity coefficient is that the moderator temperature coefficient is less than or equal to 9 percent millirho (pcm)/°C (5 pcm/°F) at hot zero power conditions and less than or equal to 0 pcm/°C (0 pcm/°F) at or above 50 percent rated core thermal power for the entire cycle of operation. In addition, the fuel temperature, or Doppler coefficient of reactivity is always negative. This design basis is enforced through plant Technical Specification LCO 3.1.3, as described in TS Section 3 found in FSAR Tier 2, Chapter 16**.** The staff finds that the design basis satisfies GDC 11, since the negative moderator coefficient at power compensates for an increase in reactivity.

Core power distribution design calculations are performed to ensure that linear power density (LPD), fuel melting, departure from nucleate boiling ratio (DNBR), and peak rod burnup limits are not exceeded. The design calculations are performed with validated analytical methods, as described in ANP-10263P-A. Core monitoring instrumentation provides the capability for ensuring that power distribution-related Technical Specification LCOs are satisfied. The staff finds that this design basis meets the requirements of GDC 10 and GDC 13 in that specified acceptable fuel design limits are not exceeded during normal operation, and that instrumentation and controls have been provided to monitor LPD and DNBR to maintain the variables and systems within prescribed operating ranges.

GDC 25 requires that no single malfunction of the reactivity control systems (excluding the postulated control rod ejection accident) causes the specified acceptable fuel design limits to be exceeded. This requirement is met by limiting the worth and the speed of control rod withdrawal to values within the range assumed in the safety analysis. The control rod withdrawal event is evaluated in FSAR Tier 2, Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," which shows that the protection system responds to prevent specified acceptable fuel design limits from being exceeded for this event. Therefore, the staff finds that GDC 25 and GDC 20 are satisfied in that the protection system be designed to provide automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded and that single malfunction of the RCSs does not cause SAFDLs to be exceeded.

The design basis for shutdown margin is described by the applicant as being specified in the Core Operating Limits Report for all modes of reactor operation. Two independent reactivity control systems are described that ensure shutdown margin specifications are met, assuming that the maximum worth control rod remains stuck in the fully withdrawn position. The use of soluble boron for reactivity control of core-wide xenon transients and recognition of shutdown margin degradation due to crud buildup or boron deposition are also described. The staff determined that these design bases meet the requirements of GDC 26 in that two diverse means of controlling reactivity have been provided. In addition, the applicable requirements of GDC 27 are met, on the basis of the accident analyses presented in FSAR Tier 2, Chapter 15 and evaluated in Chapter 15 of this report which were performed with the most reactive RCCS stuck out which meets the provisions of GDC 27.

Compliance with GDC 28 is evaluated in FSAR Tier 2, Chapter 15, which demonstrates that the reactor can be shut down with a coolable geometry following a postulated reactivity accident such as a rod ejection or steam line break. Reactor core stability is ensured by the inherent resistance of the PWR design to core thermal-hydraulic power oscillations. Protection of the specified acceptable fuel design limits against the effects of spatial xenon transients is provided by core monitoring and protection system actions which will trip the reactor should oscillations result in unacceptable power distributions. The applicant states that because of the negative overall power coefficient, the U.S. EPR is inherently resistant to power oscillations, and only xenon-induced oscillations are considered. A description of the phenomenon is provided together with a discussion of how a xenon-induced oscillation is controlled, as exemplified in FSAR Tier 2, Figure 4.3-38, "Typical Damped Xenon Oscillation." The staff finds that the reactor core stability design basis described by the applicant meets the requirements of GDC 12.

Additional design bases for the fuel system, including material and mechanical properties, are described in FSAR Tier 2, Section 4.2.1 and evaluated in Section 4.2.4 of this report.

The staff finds that the design bases for the nuclear design described by the applicant in FSAR Tier 2, Section 4.3.1, "Design Bases," conform to the guidelines of SRP Section 4.3 and meet the associated GDC requirements relating to design bases.

4.3.4.2 *Description*

FSAR Tier 2, Section 4.3.2, "Description," provides the nuclear design of the reactor core, including a description of the fuel assembly and core design, core power distributions, reactivity coefficients, reactivity control, control rod patterns and reactivity worths, criticality of the reactor during refueling operations, core stability, and vessel irradiation. An overall description of the fuel and core design is summarized above in Section 4.3.2 of this report.

The applicant states that the reactor fuel and core design described in FSAR Tier 2, Section 4.3 is representative of a "typical" 18-month cycle initial core and is presented for illustration purposes. The staff accepts this representation on the premise that the nuclear design parameters provided in this section and evaluated by the staff are typical and that any significant deviations, such as increases in gadolinia weight percentage, fuel rod burnup levels, or design cycle lengths beyond 24 months, will require additional review by the staff. Further, the design and capability of permanent systems, such as reactivity control systems and the like may be evaluated in the context of such a typical one.

The applicant provides various data tables and figures to detail the nuclear design of the nuclear fuel assembly and reactor core. A general cross section of the U.S. EPR high performance fuel assembly is provided in FSAR Tier 2, Figure 4.3-1, "Cross Section of the U.S. EPR High Thermal Performance Fuel Assembly." FSAR Tier 2, Figures 4.3-6, "Fuel Assembly Designs A1 and A2," through 4.3-9, "Fuel Assembly Design C3," depict the radial pin-by-pin and axial distributions of enriched uranium and gadolinia for each of the seven fuel assembly types used in the initial core. FSAR Tier 2, Figure 4.3-3, "Typical Initial Core Loading Map," shows the fuel assembly core loading pattern.

Gadolinia Fuel Rods

As stated by the applicant, gadolinia neutron absorber is used in conjunction with soluble boron, because the use of soluble boron alone to hold down excess reactivity would cause the moderator temperature coefficient to be positive for power operating conditions at the beginning of cycle. The staff concurs with this assessmentThe staff has reviewed these design descriptions together with the resulting power distribution data provided in FSAR Tier 2, Figures 4.3-10, "Quarter Core Relative Assembly Radial Power Distribution (HFP at BOL, ARO, No Xenon)," through 4.3-21, "Typical Axial Power Shape at End of Life." Symmetric placement of gadolinia fuel rods is standard design practice to ensure proper burnup across the fuel assembly. However, placement of gadolinia fuel rods face-adjacent to the water-filled guide tubes could potentially result in preferential uneven depletion of the gadolinia which presents a challenge to the lattice physics model calculations. In RAI 134, Question 04.03-8, the staff expressed concern that such placement of gadolinia-bearing fuel rods could result in preferential uneven depletion of the gadolinia, thus posing a challenge to the lattice physics model calculations and, therefore, introduction of modeling error.

In a January 29, 2009, response to RAI 134, Question 04.03-8, the applicant stated that fuel assembly configurations with gadolinia-bearing fuel rods located face-adjacent to water-filled guide tubes had been used in operating plants with the applicant's supplied fuel assemblies for more than 25 years. According to the applicant, the plant measured data has shown that the critical boron, the power peaking, and the rod worths predicted by the applicant's design code system with MICBURN-3/CASMO-3G are acceptable for operation and safety-limit predictions. The applicant explained that in critical experiments, referenced in BAW-1810P-A, "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," April 1984, used to benchmark the applicant's methodologies, multiple configurations with gadolinium bearing fuel rods had been used and provided details.

The applicant states that the benchmark calculations for these critical experiments indicate that the accuracy of applicant's methodology for criticality is not sensitive to the amount of water or poison material surrounding the gadolinia-bearing fuel rods, the types of rods surrounding the gadolinia-bearing fuel rods (fuel, water, void, Ag-In-Cd, or B_4C), and the azimuthal arrangements of the rods surrounding the gadolinia-bearing fuel rods. This indicates that the approximate spectrum used to generate the multi-group gadolinia cross sections for the gadolinia-bearing fuel rods is adequate and the resulting multi-group cross sections for the gadolinia-bearing fuel rods are accurate enough for generating the homogenized two-group assembly cross-sections.

The applicant stated that the benchmark calculations for these power distributions indicate that the accuracy of the applicant's methodology on pin powers is not sensitive to the amount of water surrounding the gadolinia bearing fuel rods. They presented the results of sensitivity studies performed by the applicant with MICBURN-3/CASMO-3G to estimate the effects of using the approximate spectrum to generate the multi-group gadolinia cross section in the code MICBURN-3. These studies also illustrate the spectral effect on depletion. The applicant concludes that the azimuthal distribution is not important in criticality and peaking calculations. They also note that Gd bearing fuel rods are not limiting until depletion of the Gd, and by that time burnup has smoothed differences in peak pin powers.

The staff observes that these methods are widely used in the industry with excellent results. Considering that the mean free path length of a thermal neutron is greater than the geometric details and azimuthal orientation in the fuel assembly, it would not be expected that the results would be other than as stated by the applicant. Based on the operational experience, critical experiments, and sensitivity studies, the staff concludes that the calculational methods are acceptable. Therefore, the staff considers RAI 134, Question 04.03-8 resolved.

Core Loading Pattern

The core loading pattern shown in FSAR Tier 2, Figure 4.3-3 conforms to standard fuel loading schemes relative to power peaking and fuel economy considerations, for example, peripheral loading of low enriched fuel assemblies for reduced neutron leakage from the core and an internal ring of high enriched fuel for reactivity and power distribution control. The core radial power distributions shown in FSAR Tier 2, Figures 4.3-10 through 4.3-16 appear reasonable relative to the radial peaking factor and the corresponding core loading pattern. Analytical methods employed in these calculations are discussed below.

Power Dependent Control Rod Insertion Limits

The control rod group power-dependent insertion limits provided in FSAR Tier 2, Figure 4.3-2, “U.S. EPR Rod Group Insertion Limits Versus Thermal Power,” show partial insertion of control bank D and full withdrawal of the three other control banks, which conforms to proposed shutdown margin TS requirements. The three shutdown banks are fully withdrawn prior to withdrawal of any control bank during reactor startup, as stated in FSAR Tier 2, Section 16B.3.1. These power-dependent insertion limits meet GDC 10, GDC 20, and GDC 25 by providing adequate shutdown margin to ensure specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

Average Core Enrichment

FSAR Tier 2, Section 4.3.2.1, “Nuclear Design Description,” states that the core average enrichment is set by the fuel cycle length and total energy production. In RAI 134, Question 04.03-9, the staff requested that the applicant explain the rational for that statement and state if and how the fuel assemblies’ design burnup limit, which is based on the peak rod burnup limit, is considered in the determination of core average enrichment. In a January 29, 2009, response to RAI 134, Question 04.03-9, the applicant stated that, in order to generate the desired energy for the specified cycle length, a sufficient amount of U-235 must be loaded into the core. Without changing the physical configuration of the fuel rods (i.e., fuel radius, fuel stack height, or fuel density) the only option remaining that provides the desired U-235 loading is the enrichment. The overall core average enrichment (U-235 loading) establishes the capability of producing the desired energy based on the core power output for the specified cycle length. In this regard, the staff observes that peak fuel enrichment is limited to 5.00 weight percent U-235.

In FSAR Tier 2, Section 4.3, the applicant states, that for a fresh core the assembly enrichments are distributed to provide the desired core average enrichment. The enrichment of the assemblies, poison rod distribution within the assembly, and placement of the assemblies are determined such that the fuel performs within the specified design criteria including operation within the peak rod burnup limit. Evaluation against the peak rod burnup limit considers the potential for coastdown operation, which can extend operation of the cycle beyond the specified cycle length.

For reload cores, the core average enrichment (U-235 loading) also determines the ability of the core to operate delivering the desired energy to the specified cycle length. In this case, fuel irradiated in the previous cycle is carried over and the enrichment of the fresh fuel batch is determined to provide the desired core average enrichment for the cycle design. The number and enrichment of the fresh assemblies, along with poison rod loadings, are chosen such that all criteria are met.

The peak rod burnup limit does not impact the core average enrichment but has more of an impact on the assembly average enrichment and the number of fresh assemblies included in a reload core. By increasing the number of fresh assemblies in a reload core, the average enrichment of the reload fuel decreases. These assemblies operate at a lower power level in their first cycle. This translates into lower peak rod burnups at the end of the assembly life.

This explanation conforms to the underlying neutron physics, that is, the excess reactivity of the whole core determines the cycle length. The location of specific fuel assemblies in first and subsequent fuel cycles determines the core radial power distribution and fuel assembly burnup. The peak pin to assembly average burnup ratio, which is a function of the design, determines the peak pin burnup. Furthermore, the staff notes that the core, fuel assembly, and peak pin burnup are monitored with the incore instrument system to ensure that limits are not exceeded. Therefore, the staff considers RAI 134, Question 04.03-9 resolved.

Power Density

FSAR Tier 2 defines six quantitative means of expressing fuel rod or assembly power: Power density (kw/liter or w/cm^3); linear power density (kw/ft); average linear power density (kw/ft); local heat flux (btu/ft^2/hr); rod power (kW); and core average rod power (kw/ft). FSAR Tier 2, Table 4.3-5, “Nuclear Design Parameters (First Cycle),” lists the design core average linear power to be 17.126 kw/m (5.22 kw/ft) including gamma energy deposition. Utilizing other design parameters given in FSAR Tier 2, Table 4.3-3, “Reactor Core Description,” the staff’s calculation of design core average linear power matches the reported 17.126 kw/m (5.22 kw/ft), and the staff concludes that these values are internally consistent.

Fuel Densification

Hot channel factors for heat flux, F_Q and for enthalpy rise, $F_{\Delta H}$, together with their associated allowances and uncertainties are also defined. The treatment of fuel densification effects on F_Q is also described. Fuel pellet densification could result in gaps between the pellets within the rods, causing increased local power peaking in adjacent rods. The applicant states that the potential for significant fuel densification has been largely eliminated due to modern fuel manufacturing practices and, therefore, a power spike factor of 1.0 is used for the U.S. EPR fuel. Licensing Topical Report BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs," June 1989, is referenced by the applicant as the basis for not including an allowance for fuel densification.

During the review, the staff noted that a sufficient explanation on how the current fuel manufacturing practices have largely eliminated the occurrence of fuel densification was not provided in the FSAR. In RAI 134, Question 04.03-10, the staff requested an explanation of how fuel densification is largely eliminated due to modern fuel manufacturing practices, including a description of the pellet manufacture process, densification propensity, analytical predictions of fuel densification, and post-irradiation examinations results, if available.

In a January 29, 2009, response to RAI 134, Question 04.03-10, the applicant stated that the justification provided in BAW-10163P-A of a power spike factor of 1.0 is applicable to the U.S. EPR due to similarities of the U.S. EPR fuel assembly to a typical U.S. four-loop PWR fuel assembly and due to the applicant's fuel manufacturing practices.

Section 4.1.2.3 of BAW-10163P-A states that the "effects of fuel densification on the peak linear heat rate are included by determining the core average linear heat rate based upon densified fuel," and because "gaps are eliminated or reduced during power operation," and that "no explicit penalty is included to account for densification spike effects." The same section of BAW-10163P-A also states that gap measurements have been performed on irradiated fuel, and only very small gaps have been observed at cold conditions, concluding that the gaps are eliminated or reduced at power conditions.

The applicant further explained that specific density requirements are established for each PWR fuel design. These specific requirements reflect fuel thermal performance considerations, fuel densification, swelling at the end of design life, and typically result in pellet density, in a range between 95-96 percent (theoretical density). The U.S. EPR design uses 96 percent theoretical density. Rod prepressurization measures are established for each fuel rod design based on plant operating conditions, void volume, and fuel type, and range from about 1.69 MPa to 2.69 MPa (245 psia to 390 psia) for PWR designs. The U.S. EPR design calls for prepressurizing fuel pellets to 2.10 MPa (305 psia).

Based on the use of previously approved methods, post irradiation gap measurements reported in BAW-10163P-A, and the intended use of high density fuel in the U.S. EPR, the staff determined the response to be acceptable as operating experience has demonstrated that modern fuel designs have compensated for fuel densification issues that arose in earlier designs from the 1970s. Therefore, the staff finds RAI 134, Question 04.03-10 resolved.

Axial Power Distribution

The applicant states that axial power distributions are largely under the control of the plant control room operator through movement of control rods or through the automatic movement of control rods in response to power level or moderator temperature changes. Axial offset (AO), defined as the difference between the power in the top of the core and the bottom of the core, is displayed on a control room panel. The axial offset is calculated by the reactor control, surveillance, and limitation (RCSL) system from the fixed incore self-powered neutron detectors (SPNDs). FSAR Tier 2, Section 4.3, Figures 4.3-19, "Typical Axial Power Shape at Beginning of Life," through 4.3-21, "Typical Axial Power Shape at End of Life," show typical axial power distributions at the BOL, middle of life, and end of life (EOL). The representative axial power distributions appear reasonable in that skewed cosinal shapes are shown with the power shifting from the lower half to the upper half of the core as the lower half of the fuel bundles are depleted as is predicted from basic physics considerations.

FSAR Tier 2, Section 4.3 describes how the core power distribution is treated in establishing the initial conditions for the FSAR Tier 2, Chapter 15 safety analysis. Initial conditions are selected based on core design analyses such that the range of possible operating states is covered. Severe axial power shapes calculated for xenon transient conditions are used in the safety analysis.

Limits on the important initial conditions, such as power distribution parameters, are defined as LCOs in the plant TS. The TS LCOs are intended to ensure sufficient thermal margin is reserved to accommodate design basis events. The goal is verified by safety analysis shown in Chapter 15 of the FSAR and evaluated in Chapter 15 of this report which predicts that with TS LCOs as initial conditions in the analysis, that SAFDLs will not be exceeded.

Limiting Parameters

During reactor operation, continuous monitoring of core power distributions ensures that TS limits on DNBR, axial offset, quadrant power tilt, and linear power density are maintained.

The TS on AO and LPD ensure that hot channel factor F_Q remains below the LOCA analysis limit when operating at rated power. FSAR Tier 2, Figure 4.3-23, “Maximum F_Q as a Function of Core Height,” shows the design upper limit on F_Q versus core height to be 2.60. The design value of F_Q = 2.60 is given in FSAR Tier 2, Table 15.6-14, “SBLOCA - U.S. EPR System Analyses Parameters,” for the Small Break LOCA analysis. The applicant states that F_Q is continuously monitored using the fixed incore SPND detectors.

In RAI 134, Question 04.03-11, the staff requested an explanation of how F_q is continuously monitored using the fixed incore SPND detectors. The staff noted that there is no TS limiting condition for operation or surveillance requirement on F_q. FSAR Tier 2, Section 16 B.3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor ($F^N{}_{\Delta H}$),” states that F_Q is a direct input parameter to the LOCA analysis; however, the TS surveillance requirement is on $F^N{}_{\Delta H}$. The staff requested that the applicant explain why there is no TS surveillance requirement on F_q, and to clarify the relationship of F_q to $F^N{}_{\Delta H}$ relative to the LOCA analysis.

In a January 29, 2009, response to RAI 134, Question 04.03-11, the applicant stated that F_Q is indirectly monitored by linear power density which is continuously monitored. As stated in FSAR Tier 2, Section 4.3.2.2.1, “Definitions,” F_Q is defined as the maximum local heat flux divided by the average fuel rod heat flux. This is converted during the setpoint analysis to a LPD limit. Therefore, there is no limit on F_Q directly represented in the TS because of the core monitoring methodology for the U.S. EPR. There is a limitation on allowed axial offset in the COLR which was established to limit the maximum possible F_Q to a value of 2.6 with uncertainties.

The applicant explained that the safety analysis treats the F_Q corresponding to the LPD limit as if it were a technical specification upper limit. In realistic large break loss-of-coolant accidents (RLBLOCA), the F_Q for each case analyzed is randomly selected between a nominal F_Q (lower bound) and the F_Q corresponding to the LPD limit (upper bound). The nominal F_Q is determined from power history data. The staff accepts the explanation and limitations provided by the applicant, since there is a one-to-one relationship between LPD and F_Q. LPD is a surrogate for

F_Q in that the peak LPD divided by the core average linear heat rate equals F_Q. Therefore, the staff finds RAI 134, Question 04.03-11 resolved.

The FSAR also describes the application of hot channel factor $F^N_{\Delta H}$ as a design-basis criterion. $F^N_{\Delta H}$ is stated to be used for establishing control rod patterns and control rod bank sequencing, as well as fuel loading patterns. The staff noted that FSAR Tier 2, Section 16B.3.2, "Power Distribution Limits," describes the basis for the $F_{\Delta H}$ TS LCO to be also associated with the LOCA analysis. In RAI 134, Question 04.03-12, the staff requested that the applicant provide a description of the design basis for the hot channel factor $F^N_{\Delta H}$, including its relationship to hot channel factor F_Q, and how it is monitored during power operation. In a January 29, 2009, response to RAI 134, Question 04.03-12, the applicant referred to its response to RAI 134, Question 04.03-11 described above. The staff notes that $F^N_{\Delta H}$ is an input to safety analyses of anticipated transients and accidents which demonstrate acceptable consequences, as described in Chapter 15 of this report. The response provided by the applicant is acceptable based on the discussion above and, therefore, the staff finds RAI 134, Question 04.03-12 resolved.

Crud Deposition

The effect of crud deposition or boron buildup on core power distribution is also described in the FSAR, but the staff determined that it needed more information to complete its evaluation of the effects of these phenomena. In RAI 134, Question 04.03-13, the staff noted that FSAR Tier 2, Section 4.3.2.2.6, "Limiting Power Distributions," states that crud deposition or boron buildup on fuel rods can affect core power distribution and that continuous monitoring of DNBR, LPD, and axial offset would detect changes in power distribution caused by these phenomena. The staff requested an explanation of the crud and boron deposition phenomena and a description of the effects on core power distribution and core reactivity, how it is accounted for in the design analysis, and how it is detected during reactor operation.

In a January 29, 2009, response to RAI 134, Question 04.03-13, the applicant stated that a description of the effects of crud buildup and boron deposition on power distribution and shutdown margin is provided in NRC Information Notice 97-085, "Effects of Crud Buildup and Boron Deposition in Power Distribution and Shutdown Margin," December 11, 1987. As explained in Information Notice 97-085, crud buildup and boron deposition can result in an anomaly in the axial offset. The AO must remain within limits established in the TS to ensure that both shutdown margin and clad local peaking factors are not exceeded. The applicant explained that "[c]rud buildup and boron deposition on the upper portion of fuel assemblies on the reactor core results in a depression of the neutron flux at these locations and the power shifts toward the bottom of the core. This results in a reduction of SDM and an increase in local peaking factors."

The applicant stated that the potential for crud deposition and boron buildup is precluded through chemistry controls; therefore, it is not accounted for in the design analysis and noted that "the continuous monitoring of the DNB, the LPD, and the AO against LCO limits would detect changes in power distribution caused by these phenomena."

The staff accepts the explanation based on the planned continuous monitoring of the DNB, the LPD, and the AO in the U.S. EPR core and the limits in the TS. Therefore, the staff considers RAI 134, Question 04.03-13 resolved.

In view of the foregoing, the staff finds the applicant's treatment of core power distributions as described in FSAR Tier 2, Section 4.3, to reasonably bound the expected core operating states of the U.S. EPR reactor such that limits on LPD, DNBR, and peaking factors can be maintained during operation.

Verification of Power Distributions

Experimental verification of the power distribution analysis is also provided in this section of the FSAR. The applicant refers to proposed requirements of the plant TS for performing periodic comparisons of measured versus calculated power distributions; however, the staff was unable to identify such requirements in FSAR Tier 2, Chapter 16. In RAI 134, Question 04.03-14, the staff requested that the applicant provide reference to the FSAR Tier 2, Chapter 16 section that would require periodic comparison of measured versus calculated power distributions throughout cycle lifetime.

In a January 29, 2009, response to RAI 134, Question 04.03-14, the applicant stated that FSAR Tier 2, Chapter 16, Section 3.2.2 addresses power distribution limits. FSAR Tier 2, Section 16B.3.2.2 states that $F^{N}_{\Delta H}$ is verified by taking an AMS flux map. The AMS flux mapping process performs a comparison of measured versus calculated power distributions.

As stated in the FSAR, the movable incore AMS is used to periodically calibrate the fixed incore SPND detectors in accordance with TS surveillance requirement (SR) 3.3.1.2 and FSAR Tier 2, Section 16, Table 3.3.1-1, "Protection System Sensors, Manual Actuation Switches, Signal Processors, and Actuation Devices."

As further stated, uncertainties in the AMS readings are addressed through comparison of symmetric channels. The staff determined that the information contained in the FSAR was insufficient for the staff to complete its evaluation of how the symmetric aeroball measurements are utilized to address the AMS uncertainties. In RAI 134, Question 04.03-15 the applicant was requested to address AMS uncertainties and to explain how the uncertainties are treated in determining the measured power distributions and associated fuel thermal parameters such as DNBR, LPD, and hot channel factors.

In a January 29, 2009, response to RAI 134, Question 04.03-15, the applicant stated that power distribution related measurements and their uncertainties are discussed in ANP-10263P-A and ANP-10287P, Revision 0, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report," AREVA NP, Inc., November 2007. The applicant also provided a summary of the process by which the uncertainties are obtained. ANP-10263P-A was previously approved and ANP-10287P was approved in a staff SER June 16, 2009. These methods were approved for the U.S. EPR and have been used consistent with their approval. The staff finds the response to be acceptable, since it refers to previously approved methods. Therefore, the staff considers RAI 134, Question 04.03-15 resolved.

Initial Test Program

Initial plant testing described in FSAR Tier 2, Chapter 14, "Initial Test Program and ITAAC," includes a number of core physics tests. These tests provide confirmation of the values of certain key parameters and provide a check on the validity of the analytical methods. The staff has identified the following initial tests from FSAR Tier 2, Section 14.2, to be applicable to the nuclear design described in this section of the FSAR:

- Test No. 127 “Aeroball Measurement System”
- Test No. 141 “Incore Instrumentation”
- Test No. 142 “Excore Instrumentation”
- Test No. 188 “Post-Core Incore Instrumentation”
- Test No. 190 “Critical Boron Concentration: All Rods Out”
- Test No. 191 “Isothermal Temperature Coefficient”
- Test No. 192 “Rod Worth”
- Test No. 206 “Self Powered Neutron Detector Calibration”
- Test No. 207 “Steady-State Core Performance”
- Test No. 208 “Core Related Reactor Trips”
- Test No. 209 “Incore / Excore Cross-Calibration”
- Test No. 218 “HZP to HFP Reactivity Difference”

The staff has evaluated the above-listed initial tests against the nuclear design described in FSAR Tier 2, Section 4.3, the ITAAC requirements identified above in FSAR Tier 2, Section 4.3.2, the requirements of 10 CFR 52.47(b)(1), and the guidance contained in RG 1.68.

The staff determined that the proposed initial testing conforms to the guidance in the regulatory guide, and is therefore acceptable. However, the staff was unable to confirm that the following testing, as specified in RG 1.68, is included in the U.S. EPR initial test program described in FSAR Tier 2, Section 14.2.

- Determination of core linear power density and DNBR in order to verify the capability of the core monitoring system for use in complying with Technical Specification LCOs
- Performance of a pseudo-rod-ejection test to verify calculational models and accident analysis assumptions
- Demonstration of the capability of the incore neutron flux instrumentation to detect rod misalignment equal to or less than the Technical Specifications limits for control rod misalignment

The staff noted that FSAR Tier 2, Section 16B.3.1.7, “Rod Control Cluster Assembly (RCCA) Position Indication,” states that detection of a rod misalignment may be done through use of the incore aeroball measurement system.

In RAI 134, Question 04.03-16, the staff requested that the applicant explain why these tests are not included. In a January 29, 2009, response to RAI 134, Question 04.03-16, 2009, the applicant stated that:

- FSAR Tier 2, Section 14.2.12.18.9, Test No. 207 will be revised to reflect the Technical Specifications surveillances that must be completed following a full core flux map.

- The pseudo-rod-ejection test is performed during FSAR Tier 2, Section 14.2.12.15.3, Test No. 192 and meets Acceptance Criteria 5.2 of Test No. 192. FSAR Tier 2, Section 14.2.12.15.3, Test No. 192 will be revised to indicate those steps that are related to a pseudo-rod-ejection test.

- FSAR Tier 2, Section 16, LCO 3.1.4 states, “Individual indicated analog RCCA positions shall be within 8 steps of their group digital RCCA position indication.” FSAR Tier 2, Section 14.2.12.19.2, Test No. 213 will be revised to test the ability to detect a rod cluster control assembly that is misaligned so that the individual indicated analog RCCA position is within seven steps of the group digital RCCA position indication of other RCCAs in the same bank.

The staff has confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the response and, therefore, considers RAI 134, Question 04.03-16 resolved.

Reactivity Coefficients

The applicant describes the fuel temperature, moderator temperature, and power reactivity coefficients as conforming to standard practice. FSAR Tier 2, Figures 4.3-25, “Typical Doppler Temperature Coefficient,” through 4.3-27, “Typical Doppler-Only Power Defect at BOL and EOL,” provide typical ranges of the fuel temperature coefficient which show that the fuel temperature coefficient meets the design basis described in Section 4.3.4.1 of this report. Similarly, FSAR Tier 2, Figures 4.3-28, “Typical Zero Power Moderator Temperature Coefficient at BOL,” through 4.3-31, “Typical Hot Full Power Moderator Temperature Coefficient,” show the moderator temperature coefficient as a function of core lifetime and moderator temperature. Considering the critical boron concentrations given in Figure 4.3-5, “Boron Concentration Versus Burnup for a First Core,” the staff concludes that the moderator temperature coefficient meets its design basis as described in Section 4.3.4.1 above. That is, the peak boron concentration is of the order of 1000 ppm. At that level the increase in temperature of the water with concomitant decrease in boron density will not cause an increase in reactivity (a positive moderator coefficient) while at higher levels, for example 1200 ppm, it would. In addition, the power coefficient shown in Figure 4.3-32, “Typical Total Power Coefficient at BOL and EOL,” is consistent with the theoretical trend of moderator temperature coefficient as a function of boron concentration and moderator temperature. The calculated reactivity coefficients are verified during startup testing as described in FSAR Tier 2, Section 14.2.

Reactor kinetics parameters β_{eff} and l^* are also presented in FSAR Tier 2, Table 4.3-5. The staff notes that the β_{eff} range of .0074 at beginning of core life to .0052 at end of core life conforms to typical values for a light water moderated reactor using low enriched fuel. Similarly, the corresponding values given for l^* are typical for water moderated reactors.

Reactivity Control Features

FSAR Tier 2, Section 4.3 describes the reactivity control features of the U.S. EPR. Reactivity control is provided by use of the following:

- Soluble boron to control relatively slow changes in core reactivity, such as fuel depletion and xenon transients

- Fuel assembly burnable absorber rods using gadolinia to hold down excess reactivity in place of soluble boron in order to limit the positive moderator temperature coefficient

- Rod cluster control assemblies to control fast changes in core reactivity, such as reactor shutdown and load follow ramping

Enriched Boron: The abundance of B^{10} is 17 to 18 percent of natural boron. In FSAR Tier 2, Section 4.3.2.4, the applicant states that the soluble boron used for reactivity control is natural B^{10} abundance, whereas in FSAR Tier 2, Section 6.8.3, the applicant states that enriched boron is utilized in the extra borating system. FSAR Tier 2, Table 6.8-1 lists the extra boron system B^{10} enrichment to be 37 percent. In RAI 134, Question 04.03-17 the staff requested that the applicant confirm the use of natural boron for normal (chemical and volume control system) core reactivity control and to describe the plant controls for keeping the natural boron separate from the enriched boron, the station procedures for verifying the isotopic level of B^{10}, and the plant controls for ensuring the correct boron solutions are used in their respective systems.

In a January 29, 2009, response to RAI 134, Question 04.03-17, the applicant indicated that FSAR Tier 2, Section 4.3.2.4, "Control Requirements" states: "Boron concentrations (natural B^{10} abundance) for several core conditions are listed in FSAR Tier 2, Table 4.3-5." The applicant explained that the data was presented in terms of natural B^{10} and not enriched; it was not meant to indicate that the chemical and volume control system would have natural boron. By using a natural boron representation, the boron concentrations for a system using enriched boron are obtained by taking the ratio of the B^{10} content in the enriched boron to the B^{10} content in natural boron by weight. Therefore, the amount of B^{10} in the coolant is conserved.

The U.S. EPR uses only enriched B^{10} in the borated systems and monitors B^{10} isotopic abundance in accordance with the TS (e.g., FSAR Tier 2, Section 16.3.5, SR 3.5.1.6, SR 3.5.4.4, SR 3.5.5.6, and Section 16.3.7, SR 3.7.15.2). Since natural boron is not used, plant controls are not required for keeping the natural boron separate from the enriched boron. The response clarifies the use of enriched boron and is acceptable. Therefore, the staff considers RAI 134, Question 04.03-17 resolved.

Fuel Assembly Burnable Absorber Rods: The description of gadolinia integral burnable absorber rods is given above.

Control Rod Patterns: Further description of control rod patterns and worth is provided in the FSAR, as follows. The 89 control rods are grouped into 4 control banks or groups and 3 shutdown banks or groups.

Control groups are moved automatically or manually, and all 89 control rods drop into the core upon a reactor trip signal. Power dependent insertion limits are established which limit the insertion of the control banks during power operation in order to ensure sufficient shutdown reactivity worth. Shutdown banks are fully withdrawn when the reactor is critical. Reactivity control measures are given in FSAR Tier 2, Tables 4.3-5 and 4.3-6, "Reactivity Requirements for Rod Cluster Control Assemblies (First Cycle)."

The control rod insertion profile and reactivity insertion rate given by FSAR Tier 2, Figures 4.3-36, "Rod Position versus Time of Travel after Rod Release," and 4.3-37, "Reactivity Worth versus Rod Position," reasonably represent the expected characteristics of the reactor in that the shape of the curve conforms to a linear insertion multiplied by a flux squared importance weighting function. The staff determined there is reasonable assurance that the requirements of GDC 26 are satisfied.

4.3.4.3 *Analytical Methods*

FSAR Tier 2, Section 4.3.3 describes the analytical methods used in the nuclear design and the comparisons of calculated versus measured values of parameters.

The applicant references its licensing topical reports EMF-96-029P-A, "Reactor Analysis Systems for PWRs – Volume 1 – Methodology Description," and ANP-10263P-A. The methods described in EMF-96-029P-A have been approved by the staff for application to the U.S. EPR in ANP-10263P-A in a safety evaluation dated August 8, 2007. The staff notes, however, that the applicant's inferred power distribution reconstruction methodology is explicitly excluded in the staff's approval of ANP-10263P-A. This methodology is included in ANP-10287P. The applicant's licensing Topical Report ANP-10287P documents the analytical methodology used to determine the setpoints for the incore-based departure from nucleate boiling ratio and linear power density limiting safety-system settings (LSSS), and LCO functions in the U.S. EPR. The staff Safety Evaluation for AREVA Topical Reports ANP-10287P was issued on August 20, 2009.

FSAR Tier 2, Section 4.3.3.1.3, "Treatment of U.S. EPR Heavy Radial Reflector," describes the neutronic treatment of the heavy reflector and refers to ANP-10263P-A. The applicability of these methodologies is in the Safety Evaluation Report of ANP-10263P-A. The applicant explained that transport theory calculations were performed to generate a set of equivalent reflector cross-sections. The staff notes that this is a standard procedure. The staff has reviewed the information provided by the applicant and determined that the proposed treatment of boundary conditions follows the standard procedure and, accordingly, the methodology is acceptable. The staff also notes that future routine incore measurements will provide confirmatory data to validate this methodology.

Similarly, the applicant's reactor kinetics methodology described in BAW-10221P-A, "NEMO-K - A Kinetics Solution in NEMO," FRAMATOME COGEMA FUELS, September 1998, is approved for application to the U.S. EPR in ANP-10263P-A. The applicant states that NEMO-K is used to perform rod ejection analyses for the U.S. EPR.

In RAI 134, Question 04.03-18, the staff noted that FSAR Tier 2, Section 4.3.3.2.4, "NEMO Validation," states that the steady-state NEMO methodology has been benchmarked against the PRISM methodology, and refers to ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology," for comparison of results. The steady-state NEMO methodology was approved by the staff in BAW-10180A, "NEMO-Nodal Expansion Method Optimized," March, 1993. However, it is not explicitly included in the staff's approval of ANP-10263P-A. Moreover, ANP-10286P has not been approved by the staff. Thus, neither the steady-state nor the transient versions of the NEMO code have been approved by the staff for use in analyzing the U.S. EPR. Accordingly, the staff requested an explanation of the application of the NEMO methodology to the U.S. EPR and its validation basis. **RAI 344, Question 04.03-28, which is associated with the ongoing review of Topical Report ANP-10286P, is being tracked as an open item.**

In a January 29, 2009, response to RAI 134, Question 04.03-18, the applicant stated that the use of the term NEMO in FSAR Tier 2, Section 4.3.3.2.4 refers to NEMO-K run in a steady state mode. The NEMO-K code is used to analyze the kinetic conditions for the ejected rod accident analyses.

In order to set boundary conditions at the initial conditions of the event, NEMO-K needs to be run at steady state conditions. In steady state conditions, NEMO-K and NEMO yield the same results. The validation of NEMO-K is presented in BAW-10221P-A. The NEMO-K methodology in ANP-10263P-A uses adjustments to bound the neutronic parameters for the safety analysis. For each cycle, the neutronic parameters from PRISM are used to validate the applicability of the safety analysis performed with NEMO-K. Validation is performed on a cycle-by-cycle basis.

In conclusion, the analytic methods described above have been previously approved for U.S. EPR application with the exception of the open item listed above regarding the safety review of ANP-10286P. The inherent uncertainties in the analytical methods described in this section of the FSAR are addressed in the applicable licensing topical reports and are accounted for in design analyses and in the power distribution limiting conditions for operation.

4.3.4.4 *Changes*

The applicant states that, although the U.S. EPR is an evolutionary PWR, there are some changes in design from currently operating PWRs as follows:

- The use of a heavy reflector as described in FSAR Tier 2, Sections 4.3.2.1, “Nuclear Design Description,” and 4.3.3.1.3, “Treatment of U.S. EPR Heavy Radial Reflector.” The staff’s evaluation is contained in Section 4.3.4 and Chapter 5 of this report.

- The use of a pneumatic movable incore AMS utilizing vanadium-doped steel balls for activation as described in FSAR Tier 2, Section 4.4.6, “Instrumentation Requirements” The staff’s evaluation is contained in Section 4.4 of this report.

- The use of an incore-based protection system as described in FSAR Tier 2, Section 4.4.6. The staff’s evaluation is contained in Section 4.4 of this report.

- The use of annular control rods as described in FSAR Tier 2, Sections 4.2.1.6, “Rod Cluster Control and Neutron Source Assemblies,” 4.2.2.9, “Rod Cluster Control Assemblies Description,” and 4.3.2.6, “Criticality of Reactor During Refueling.” The staff notes that the use of annular control rods will not change macroscopic cross-sections since control rods are neutronically “black.” Hence their use will not affect control rod worths, shutdown margin, and power distributions and is, therefore, acceptable.

These features are described in the respective FSAR Tier 2 Sections and evaluated in the corresponding Sections of this report as identified above.

4.3.4.5 *U.S. EPR, Vessel Fluence Methodology and Calculation Review*

In accordance with SRP Section 4.3, this section presents the staff review of the U.S. EPR FSAR Revision 1, related to the pressure vessel fluence methodologies and fluence values associated with the development of the P-T limits and R_{PTS}. The staff’s evaluation of reactor vessel materials, P-T limits, PTS, and Charpy upper shelf energy (USE) is documented in Section 5.3 of this report.

The NRC-approved vessel fluence methodology described in BAW-2241P-A, "Fluence and Uncertainty Methodologies," April 2006, was used by the applicant for the estimation of the fluence values. The staff has determined that the calculation meets the guidance in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, and the requirements of 10 CFR Part 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements," regarding vessel SR with the exception of the benchmarking of the BAW 2241P-A methodology for the U.S. EPR design. The staff approved BAW-2241P-A based on surveillance capsule and dosimeter data from operating reactors. The approved version of BAW-2241P-A requires the use of surveillance capsule and dosimeter data points to verify the applicability of the methodology to any particular reactor vessel. No such data exists for any particular U.S. EPR reactor. Therefore, the staff has evaluated the applicability of BAW-2241P-A to the U.S. EPR design and concluded, as set forth below in more detail, that the methodology can be applied to the U.S. EPR design because the reactor vessel incorporates conservative design features and the proposed operating conditions reflect conservative assumptions.

The pressure-temperature limit curves were derived using linear elastic fracture mechanics and provide a margin of safety during: normal operation, heatup, cooldown, AOOs, hydrostatic testing, preservice leakage tests, and inservice leakage tests. The vessel fluence values provided by the applicant correspond to 60 EFPYs of operation.

The P-T limit methodology is in Revision 1 to Technical Report ANP-10283P, "U.S. EPR Pressure-Temperature Limits Methodology for RCS Heat-Up and Cool-Down," April 30, 2009, that is currently under staff review. The review of the P-T limit methodology is contained in Section 5.3 of this report.

As per 10 CFR 50.61, the applicant should provide all fluence values that are above 1.0×10^{17} n/cm^2 (6.45×10^{17} n/in.^{2}). The FSAR does not specifically state if the value of the fast neutron fluence in the eight inlet/outlet nozzles and their welds reach the minimum value fluence of 1.0×10^{17} n/cm^2 (6.45×10^{17} n/in.^{2}) to be considered in the embrittlement calculations. The staff verified that the fluence in the eight inlet/outlet nozzles was less than 1.0×10^{17} n/cm^2 (6.45×10^{17} n/in.^{2}).

The acceptance criteria are based on meeting the following GDC and other NRC requirements:

- GDC 14, "Reactor Coolant Pressure Boundary," as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the reactor coolant pressure boundary, in part, as neutron fluence relates to vessel embrittlement.

- GDC 31, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, in part, as neutron fluence relates to vessel embrittlement

- Appendix G, to 10 CFR Part 50, as it relates to reactor pressure vessel material fracture toughness requirements, in part, as neutron fluence relates to vessel material properties

- Appendix H, to 10 CFR Part 50, as it relates to reactor pressure vessel material surveillance program requirements, in part, as neutron fluence relates to vessel material properties,

- 10 CFR 50.61 as it relates to fracture toughness criteria for PWRs relevant to PTS events, in part, as neutron fluence relates to vessel material properties

In addition, RG 1.99 and RG 1.190 outline methods acceptable to the staff for performing the relevant calculations in satisfying the applicable criteria.

Technical Report ANP-10283P presents the results of the transport of neutrons with energies E > 1.0 MeV from core leakage to the inside surface of the vessel and through the vessel to ¼ T and ¾ T, where T is the radial thickness of the vessel. The calculation was performed using the methodology in BAW-2241P-A which has been approved by the NRC for conventional PWR geometries. The calculational methodology adheres to the guidance in RG 1.190 which includes a benchmarking process using a database from the same type of reactor geometry. The major difference of the U.S. EPR geometry with existing PWR plants (as far as neutron transport to the vessel is concerned) is the presence of the heavy-metal reflector inside the core barrel region. The application of BAW-2241P-A in analysis of the U.S. EPR ignored this benchmarking step. In RAI 270, Question 04.03-23, the staff questioned the applicant's lack of benchmarking for the application of the BAW-2241P-A methodology to the U.S. EPR vessel to calculate fluence. In an April 21, 2009, response to RAI 270, Question 04.03-23, the applicant stated that BAW-2241P-A has been benchmarked for current PWRs in addition to the pool critical assembly (PCA) and Davis Besse data. The applicant also recognized that the benchmarking called for by RG 1.190 is plant-specific and is especially applicable to the U.S. EPR because of its unique geometrical arrangement. The response stated, "This updated benchmark database will be developed when there is a lead factor of 10 or more in the embrittlement – fluence margin." This statement is unclear. The lead factors are given on FSAR Tier 2, Page 5.3-4, Revision 1, as clad/vessel = 1.6 and vessel ¼ T = 2.9, and are a function of geometry and not of exposure. The applicant clarified that the factor of 10 pertains to the fluence and not the embrittlement. The staff reviewed this response and finds the response acceptable in relation to the lead factor. However, the staff considers the benchmarking aspect of RAI 270, Question 04.03-23 as an open issue as discussed below.

The staff determined that the overall design of the U.S. EPR pressure vessel incorporates a number of conservatisms to ensure safe operation until such time as plant specific data becomes available to perform plant specific benchmarking of the fluence methodology. The conservativisms include: 1), the vessel plates and the weld materials contain a very small amount of copper (and phosphorus) the elements that are mainly responsible for high rates of embrittlement; 2) the vessel consists of ring forgings, thus, in the belt region there are no axial welds, which usually are the critical element for embrittlement in most conventional plants; 3) the material chosen for the U.S. EPR vessels has a low initial value of RT_{PTS}; therefore, a higher irradiation time is needed to reach RT_{NDT} values comparable to the screening criteria in 10 CFR 50.61; 4) the end of life peak vessel fluence values were calculated for 60 effective full power years of operation, which corresponds to 100 percent load, 100 percent of the time; and 5) the estimated RT_{PTS} values for EOL are very much lower than the screening criteria in 10 CFR 50.61, representing a very large margin to the screening criteria. The staff's evaluation of the reactor vessel integrity is documented in Section 5.3 of this report.

FSAR Tier 2, Section 5.3 provides generic P-T limit curves and material embrittlement properties for the purposes of the certification review. These quantities are based on fluence values projected to 60 EFPY. Since the estimated fluence level will be conservative since vessel fluence is low, compared to EOL fluence levels, in the first 10 EFPYs of operation and the use of BAW-2241P-A is acceptable without benchmarking. Accordingly, the staff finds that the fluence values provided in the FSAR are adequate for the initial quantification of conservative material properties and P-T limits. However, use of BAW-2241P-A to calculate fluence levels for use in constructing plant-specific PTLR after the initial 10 EFPYs of operation cannot be justified without benchmarking. Experimental measurements for use in benchmarking will become available with the removal and analysis of the pressure vessel surveillance capsules. For the U.S. EPR, this will occur about 7 and 17 EFPYs after initiation of operation. The staff determined that an additional COL information item is necessary in FSAR Tier 2, Section 5.3 to ensure that the fluence methodology is appropriately benchmarked upon receipt of surveillance capsule data. This approach is consistent with past practices. The staff has determined that a COL information item should be added to provide to the NRC a plant-specific fluence methodology benchmark promptly after the 7 and 17 EFPY surveillance capsule data become available. If the benchmark demonstrates that the method is not conservative, the licensee should recalculate the fluence values based on the benchmarked method. The staff has requested that the applicant provide a COL information item that addresses these concerns. **RAI 344, Question 04.03-27, which is associated with the above request, is being tracked as an open item.**

The justification for the acceptance of the results of the fluence calculation is based on the conservatisms previously identified and the staff's experience with previous methodologies adjusted due to benchmarking. Such adjustments were, at most, in the range of 30 percent for the operating fleet. In addition, the proposed material has a very low rate of embrittlement versus irradiation compared to existing pressure vessels. These conclusions are based on the satisfaction of Appendices G and H to 10 CFR Part 50. The staff's determination on satisfaction of Appendices G and H to 10 CFR Part 50 is contained in Section 5.3 of this report.

Based on the above information and with the exception of the issue identified in RAI 344, Question 04.03-27, the staff finds that the applicability of the generic fluence analysis for the U.S. EPR design is acceptable until benchmarking of the methodology based on measured data obtained from the surveillance capsules is complete.

4.3.5 Combined License Information Items

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

The staff has requested that the applicant provide a COL information item that addresses a plant-specific fluence methodology benchmark promptly after the first and second surveillance capsule data become available. **RAI 344, Question 04.03-27, which is associated with the above request, is being tracked as an open item.**

4.3.6 Conclusions

The nuclear design of the U.S. EPR reactor was reviewed and evaluated by the staff. The scope of the review included the design bases specification, design description, core power distributions, reactivity coefficients, reactivity control, control rod patterns and reactivity worth, reactor criticality during refueling, core stability, vessel irradiation, and analytical methods.

The applicant has described the analytical methods used in the nuclear design analysis and has provided sufficient examples of calculated parameters. Except for the methodology in

ANP-10286P, the staff concludes that the information presented adequately demonstrates the ability of these analyses to predict the reactor physics characteristics of the U.S. EPR.

The applicant has provided sufficient information relating to core reactivity for the initial core and has shown that excess reactivity can be controlled at all time during core lifetime through use of control rods, soluble boron, and burnable absorber fuel rods; and that sufficient control rod inventory is always available to shut the reactor down with a design minimum specified shutdown margin assuming the highest worth control rod remains stuck in the full out position. Therefore, staff concludes that the applicant’s analysis of the reactivity control systems shows adequate margin to assure shutdown capability will be available.

For the reasons set forth above, the staff further concludes that, except for the open items listed above, the nuclear design is acceptable and meets the applicable requirements of GDC 10, GDC 11, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28, in the following respects:

- The applicant has met the requirements of GDC 11 with respect to prompt inherent reactivity feedback in the power operating range by:
    - Using analytical methods that have been determined to be acceptable, except for ANP-10286P, “U.S. EPR Rod Ejection Accident Methodology”
    - Using an approved methodology to calculate a negative power coefficient of reactivity, based on conservative values of input parameters
- The applicant has met the requirements of GDC 12 with respect to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by showing that:
    - Short period thermal-hydraulic oscillations are not possible
    - Xenon-induced instabilities can be detected and mitigated by operation of the reactivity control systems
- The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and control to monitor variables and systems that can affect the fission process by:
    - Providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other processes such as reactor coolant temperature and pressure
    - Providing suitable alarms and/or control room indications for these monitored parameters
- The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by:
    - Having a system that uses control rods and can reliably control reactivity changes under anticipated operational occurrences
    - Having a system (the CVCS) that can hold the core subcritical under cold conditions

- Having two independent systems of different design (the control rod system and the CVCS) that can control rate of reactivity changes resulting from planned, normal power changes

- The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions by:

  - Providing a movable control rod system and a liquid poison system

  - Performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth control rod stuck fully out

- The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by:

  - The applicant has used calculational methods that have been determined to be acceptable for reactivity insertion accidents

  - Meeting the regulatory position in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974

  - Meeting the fuel enthalpy limit criteria in SRP Section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents"

  - Meeting the criteria on the capability to cool the core

- The applicant has met the requirements of GDC 10, GDC 20, and GDC 25 with respect to not exceeding specified acceptable fuel design limits by providing analyses demonstrating that:

  - Normal operations, including the effects of anticipated operational occurrences, have met fuel design criteria

  - The automatic initiation of the reactivity control systems assures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and assures the automatic operation of systems and components important to safety under accident conditions

  - No single malfunction of the reactivity control system causes the fuel design limits to be exceeded

The applicant has specified core physics-related testing which conforms to RG 1.68 and are adequate to verify FSAR Tier 1 commitments.

The staff reviewed the projected effects of fast neutron fluence on the mechanical properties of the vessel as described in SRP Section 4.3. For the reasons set forth above, the staff finds the proposed pressure vessel design to be acceptable and that predicted fluence values are conservative, except for the open items listed above in regard to fluence calculation methodology. The staff has requested that the applicant provide a COL information item that addresses a plant-specific fluence methodology benchmark promptly after the first and second surveillance capsule data become available.

4.4 Thermal-Hydraulic Design

4.4.1 Introduction

The thermal-hydraulic design of the U.S. EPR must provide adequate heat transfer for reactor fuel and core components such that fuel damage does not occur during normal operation or anticipated operational occurrences. The design must also ensure that the reactor can be safely shut down and kept subcritical with acceptable heat transfer following a postulated accident. In addition, the thermal-hydraulic design must be shown to be either not susceptible to thermal-hydraulic instability, or capabilities for detecting and suppressing instability must be provided such that specified acceptable fuel design limits are not exceeded.

4.4.2 Summary of Application

FSAR Tier 1: The application identifies no FSAR Tier 1 items specifically related to reactor thermal-hydraulic design.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 description of the thermal-hydraulic design in FSAR Tier 2, Section 4.4, summarized here, in part, as follows:

For normal operation and anticipated operational occurrences, the design assures that there is at least a 95 percent probability, at a 95 percent confidence level, that departure from nucleate boiling does not occur and the fuel melting temperature is not exceeded in any part of the core.

Fuel rod thermal evaluations are performed at steady state power and transient conditions for the full range of design burnup. The analyses verify that the fuel temperature and fuel cladding integrity design bases are satisfied. These design bases are presented in FSAR Tier 2, Section 4.2.1. The results of these analyses are used in the evaluation of FSAR Tier 2, Chapter 15 events.

According to the design basis, a minimum of 94.5 percent of the thermal design flow will pass through the active fuel region of the core for fuel rod cooling; a maximum of 5.5 percent is allotted as bypass flow. The bypass flow includes guide thimble cooling flow, head cooling flow, heavy reflector cooling flow, and leakage to the vessel outlet nozzle.

The design ensures that under normal operation and abnormal operating occurrences, hydrodynamic instability does not occur.

Reactor coolant pump head versus capacity curves are confirmed through test data. Operating restrictions are imposed on the reactor coolant pumps to ensure that a net positive suction head is maintained. The available net positive suction head of the reactor coolant pump is higher than the net positive suction head needed for the range of postulated system pressures and temperatures.

The core coolant flow distribution is calculated using LYNXT which has been approved by the staff in BAW-10156-A, Revision 1, “LYNXT Core Transient Thermal-Hydraulic Program.” FSAR Tier 2, Section 4.4.2.6.2, “Core Inlet Flow Mal-Distribution,” states that the core inlet flow distribution has been determined using a one-fifth scale-model testing of the RPV lower internals.

Uncertainties associated with fuel and cladding temperatures, pressure drops, inlet flow mal-distribution, critical heat flux correlation, flow, hydraulic loads, and mixing coefficient are calculated so that operating margins can be established.

The flux tilt, that is, the azimuthal imbalance of the radial power distribution in the core, is addressed. When the indicated azimuthal imbalance exceeds the limit, corrective action is necessary per FSAR Tier 2, Chapter 16 TS 3.2.5.

The three dimensional power distribution and peak power density are measured by the fixed incore self-powered neutron detection system. The AMS utilizes movable incore detectors for flux mapping, which in turn is used to calibrate the fixed incore detectors. An excore neutron flux instrumentation system is also provided for monitoring core power levels from the source range to full reactor power.

ITAAC: The following ITAAC identified in the FSAR Tier 1 relate to FSAR Tier 2, Section 4.4, and are listed below for reference:

- FSAR Tier 1, Section 2.2.1, ITAAC 7.2, which concerns reactor coolant system flow coastdown
- FSAR Tier 1, Section 2.2.1, ITAAC 7.3, which concerns reactor coolant system flow
- FSAR Tier 1, Section 2.4.17, ITAAC 4.2, which concerns excore instrumentation output signal
- FSAR Tier 1, Section 2.4.19, ITAAC 4.2, which concerns incore instrumentation output signal

Technical Specifications: The Technical Specifications applicable to the reactor thermal-hydraulic design can be found in the following FSAR Tier 2, Chapter 16, Section 2.0; Section 3.2; Section 3.4, Section 4.2, Section B 2.0, Section B 3.2; and Section B 3.4.

4.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 4.4 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 4.4 of NUREG-0800.

1. GDC 10, as it relates to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits are not exceeded during conditions of normal operation, including anticipated operational occurrences.
2. GDC 12, as it relates to whether the design of the reactor core and associated coolant, control, and protection systems assures that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed.

3. 10 CFR 50.34(f)(2)(xviii), as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions.

4. 10 CFR 52.47(b)(1), as it relates to ITAAC to ensure the as-built plant conforms to the certified design.

5. GDC 1, GDC 2, and GDC 4 as they relate to non-safety related nuclear instrumentation that are in the reactor coolant pressure boundary and are shared with other safety systems being classified as Seismic Category I, and meeting the applicable Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) Class 1E requirements.

Acceptance criteria adequate to meet the above requirements include:

1. A loose parts monitoring system is provided, consistent with the guidelines of RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

2. TMI Action Plan Item II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, as related to instrumentation such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples.

3. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," as related to the confirmation of the thermal hydraulic design.

4.4.4 Technical Evaluation

The staff has reviewed FSAR Tier 2, Section 4.4, "Thermal-Hydraulic Design," including applicable TS, related ITAAC and initial test requirements, and COL information items given in FSAR Tier 2, Table 1.8-2 to determine the acceptability of the design. An evaluation of the FSAR Tier 2 information was performed against the requirements GDC 10 and GDC 12 pertaining to fuel thermal margin, 10 CFR 50.34(f)(2)(xviii), as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions, and 10 CFR 52.47(b)(1) as it pertains to ITAAC for design certification.

The ITAAC from the FSAR Tier 1 given in Section 4.4.2 above are applicable to the thermal-hydraulic design and are determined to be acceptable by the staff because they will confirm the validity of assumptions made in the thermal-hydraulic design. There are no COL information items applicable to FSAR Tier 2, and none were determined to be necessary as a result of the staff's review.

The safety limits specified in FSAR Tier 2, Chapter 16 TS 2.0 meet the requirements of GDC 10 in that the reactor design has the appropriate margin to assure specified acceptable fuel design limits are not exceeded, and GDC 12 in that the design inherently suppresses power oscillations. The limiting conditions for operation specified in FSAR Tier 2, Chapter 16 TS 3.2 and FSAR Tier 2, Chapter 16 TS 3.4 impose limits on measured axial and radial power distributions. The LCOs require corrective action if the specified limits are exceeded. The three dimensional power distribution in the U.S. EPR is monitored on a real time basis. Thus, power oscillations and power shifts can be readily detected by the online monitoring system and will be suppressed by operator action if LCO limits are exceeded. The design is therefore in compliance with the requirements of GDC 12. The staff's evaluation of thermal-hydraulic instability is discussed later in this section.

Low power and shutdown operation is described in FSAR Tier 2, Section 19.1.6, "Safety Insights from the PRA for Other Modes of Operation," in which shutdown procedures including mid-loop operation and thermal-hydraulic characteristics of each operational mode are discussed. Thermal-hydraulic conditions during shutdown and lower power operation were not provided and the staff requested that the applicant provide the thermal-hydraulic conditions. In a January 29, 2009, response to RAI 134, Question 04.04-21, the applicant cited NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," September 1993, in stating that rapid boron dilution events can be prevented by use of appropriate procedures. In addition, FSAR Tier 2, Section 15.4.6.1.3, "Operation in Cold Shutdown with all RCPs Secured (Modes 5 and 6)," provides an evaluation of a boron dilution event in shutdown conditions, stating that the Engineered Safety Features (ESF) anti-dilution trip will automatically isolate the chemical and volume control system and terminate the event. Additional evaluations addressing loss of vessel inventory and loss of shutdown cooling are provided in FSAR Tier 2, Section 19.1.6, "Safety Insights from the PRA for Other Modes of Operation." The applicant proposed changing FSAR Tier 2, Section 4.4.4.4, "Core Thermal Response," to make reference to the additional evaluations. The staff has confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 134, Question 04.04-21 resolved. The adequacy of these ESF features to perform their specified functions is evaluated in Sections 15.4.6.4 and 19.1.6 of this report.

In FSAR Section 4.4.1, "Design Bases," the applicant states that "the reactor can be safely shutdown and kept subcritical with acceptable heat transfer following a design-basis postulated accident with only a small fraction of fuel rods damaged." In RAI 134, Question 04.04-22, the staff requested that the applicant define the phrase, "a small fraction of fuel rods damaged," and to explain how this criterion relates to the applicable regulatory requirements. In a January 29, 2009, response to RAI 134, Question 04.04-22, the applicant defined, "small fraction of fuel rods damaged," as the maximum number of damaged rods that can be tolerated without exceeding the regulatory dose acceptance criteria as defined in SRP Section 15.0.3. The applicant's response is acceptable to the staff. Therefore, the staff considers RAI 134, Question 04.04-22 resolved.

GDC 10 requires that "the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." The staff noted that no definitions of the SAFDLs were provided in the FSAR Tier 2 and requested that the applicant provide them. In a January 29, 2009, response to RAI 134, Question 04.04-23, the applicant stated that the SAFDLs for the U.S. EPR are that the departure from DNBR remain above 1.0 with 95 percent probability at a 95 percent confidence level and that fuel centerline melt (FCM) remain below 2705 °C (4901 °F), decreasing by 7.78 °C (14 °F) per 10,000 MWD/MTU of burnup. These are the safety limits given in Revision 1 of FSAR Tier 2, Section 16.2.1, "Safety Limits." The staff finds the applicant's definition of SAFDL in conformance with the definitions currently approved for use in operating plants and, therefore, the staff considers RAI 134, Question 04.04-23 resolved.

In licensing Topical Report ANP-10263P-A, the applicant states that the U.S. EPR is an evolutionary PWR similar in design to currently operating PWRs. The staff's final safety evaluation of ANP-10263P-A includes a comparison of the values of key design parameters for the U.S. EPR to those of a typical current generation U.S. 4-Loop PWR. The most notable differences are core power level, reactor coolant system loop flow, steam pressure and flow, primary system volume, and use of an in-vessel heavy reflector. Each of these differences is evaluated in the staff's SER on ANP-10263P-A. Based on the reactor system design similarities and the staff's evaluation of the design differences, the staff concluded in its final safety evaluation of ANP-10263P-A that there is reasonable assurance that the existing fuel analysis codes and methods identified in ANP-10263P-A are applicable to the U.S. EPR. The staff, therefore, concludes that the U.S. EPR is a justified extrapolation of a proven reactor design in this regard, and that the key differences have been identified and adequately assessed by the applicant.

Thermal Design

DNBR is evaluated by the applicant in FSAR Tier 2, Chapter 15 transient analyses. As discussed below, those analyses show that the design basis with respect to DNBR is met for normal operation, including AOOs. The validity of the applicant's transient analysis is evaluated in Chapter 15 of this report.

The linear power density LCO and the high linear power density reactor trip provide protection against fuel centerline melt and fuel cladding strain during normal operation and AOOs. Fuel centerline melt analyses are performed to determine LPD corresponding to fuel centerline melt. This LPD then used to establish the limiting LPD LCO and reactor trip setpoints. Information regarding fuel centerline melt during AOOs is provided by the applicant in FSAR Tier 2, Chapter 15 and evaluated by the staff in Chapter 15 of this report.

The DNBR is calculated on line using information from the RCSL system. The LCO DNBR must be set sufficiently large to ensure the DNBR SAFDL is not exceeded during an AOO. The applicant provides a calculated core minimum DNBR of 2.55 for nominal, rated power conditions, and a comparable DNBR LCO requirement of 2.50. These calculated DNBR values demonstrate adequate operating margin to the LCO. The DNBR LCO is claimed to provide protection against DNB-induced fuel rod cladding failure during normal operation and AOOs as evaluated under FSAR Tier 2, Chapter 15. The validity of this claim is evaluated in Chapter 15 of this report.

The DNBRs are calculated with Critical Heat Flux (CHF) correlations applicable to the high thermal performance fuel assemblies used in the U.S. EPR. The correlations are described in the applicant's licensing Topical Reports ANP-10269P-A, "The ACH-2 CHF Correlation for the U.S. EPR," December 31, 2007, and BAW-10199P-A, "The BWU Critical Heat Flux Correlations," December 6, 2000. The staff's approval of ANP-10269P-A concludes that the ACH-2 critical heat flux correlation is applicable to the U.S. EPR fuel with HTP grids and fuel rod parameters that are the same as given in FSAR Tier 2, Table 4.4-1, "Thermal and Hydraulic Design Data." The BWU-N critical heat flux correlation described in BAW-10199P-A is applicable to non-mixing vane grids such as the high mechanical performance grids employed at the top and bottom ends of the U.S. EPR fuel assembly as described in FSAR Tier 2, Section 4.2.1.

Licensing Topical Report ANP-10263P-A states that "the U.S. EPR will operate in the same thermal-hydraulic regime as current U.S. reactors which are being licensed with the LYNXT methodology. The major difference is in active core height. This will be accounted for in the development of the U.S. EPR CHF correlation." In light of this statement, the staff asked the applicant to confirm the applicability of the ACH-2 and BWU-N critical heat flux correlations to the U.S. EPR fuel. In a January 29. 2009, response to RAI 134 Question 04.04-24, the applicant stated that this statement was made prior to the submission and approval of ANP-10269P-A, and that the latter report accounts for the active core length. This was confirmed by the staff. The applicant's response also stated that since the BWU-N correlation is used for the bottom-most grid spacer, the bundle length used for development is not as important. The staff concurred with this assessment, since the BWU-N correlation is only used for the rod segment between the first and second grid spacers and rod power in this region is so low that this region is never limiting with respect to DNBR.

In order to obtain more information, the staff requested AREVA to explain the applicability of the BWU-N correlation to the upper-most spacer in the EPR fuel bundle. In a July 24, 2009, response to RAI 231 Question 04.04-48, the applicant stated that, while the correlation was applicable to the upper-most spacer, that spacer is above the top of the heated length and there is no need to apply the BWU-N correlation. The applicant proposed changing the FSAR to clarify the usage of the BWU-N correlation. The proposed change was made in Revision 1 of the U.S. EPR FSAR, dated May 29, 2009. Based on the above response and the associated revisions to the FSAR, the staff considers RAI 134, Question 04.04-24 and RAI 231, Question 04.04-48. resolved.

The applicant uses established industry-accepted correlations to calculate fuel rod surface convective heat transfer coefficients for bulk single phase and nucleate boiling conditions. RCS and vessel pressure drops are computed using the standard Darcy equation for frictional losses. The staff finds these procedures acceptable since they have been in wide use for many years for currently operating PWRs with acceptable results.

Fuel temperature calculations are performed utilizing the COPERNIC fuel rod design code. COPERNIC has been generically reviewed and approved by the staff as documented in licensing Topical Report BAW-10231P-A. The staff has determined its application to the U.S. EPR to be acceptable, per ANP-10263P-A. The staff's safety evaluation of the COPERNIC code cites a fuel rod average burnup limitation of 62 GWD/MTU for the peak rod, which conforms to the U.S. EPR fuel design as stated in FSAR Tier 2, Section 4.2. The applicant states that the bounding fuel temperature calculations at 95 percent probability and 95 percent confidence level account for uncertainties in the design code methodologies and fuel fabrication, plus the effects of fuel densification. In a January 29, 2009, response to RAI 134, Question 04.04-27, the applicant described how fuel fabrication uncertainties and tolerances are accounted for in COPERNIC calculations. In a May 8, 2009, response to RAI 231, Question 04.04-53, the applicant described how pellet chipping was treated. In essence, even though pellet chipping is not explicitly accounted for, the methodology accommodates uncertainties and tolerances by reducing the best-estimate calculated design parameters calculated by COPERNIC, as described in BAW-10231P-A. These responses are acceptable because they reference an approved methodology that is applicable to U.S. EPR fuel.

In a January 29, 2009, response to RAI 134, Question 04.04-34, as to the treatment of densification, the applicant again referred to BAW-10231P-A, where the approved fuel densification model for COPERNIC is described. This response references an applicable approved methodology, as approved by the staff in its SER on ANP-10263P-A. In May 8, 2009, responses to RAI 205, Questions 04.04-51 and 04.04-52, respectively, the applicant addressed the fuel manufacturing tolerances treated in the COPERNIC analysis and the calculated effect of each on key fuel rod performance factors. The response to Question 04.04-51 listed the uncertainties and tolerances treated in the calculation of fuel rod gas pressure and fission gas release. Since the list provided by the applicant addresses the appropriate inputs to COPERNIC, the staff finds the list to be appropriate and inclusive. The applicant noted also that the approved methodology (BAW-10231P-A) does not explicitly consider individual manufacturing tolerances for parameters other than rod pressure and fission gas release. The staff concurs. The applicant's response to Question 04.04-51 presented a table listing the quantitative effect of each manufacturing tolerance on fuel rod pressure. The staff reviewed the table and found the results to be reasonable and proper. For the above reasons, the staff finds the responses to Questions 04.04-51 and 04.04-52 acceptable.

Changes to the FSAR were proposed as part of the responses to RAI 134, Questions 04.04-27 and 04.04-34. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in these RAI responses.

Uncertainty associated with the DNBR critical heat flux correlations is described as being inherent in the critical heat flux test data on which the correlations are based. The ACH-2 and BWU critical heat flux correlations, including their data bases, were previously reviewed and determined to be acceptable as documented in the staff's safety evaluation of licensing Topical Reports ANP-10269P-A and BAW-10199P-A. The applicant stated that a statistical combination of uncertainties is utilized to treat the random nature uncertainties while a deterministic approach is used for others. Topical Report ANP-10287P, "Incore Trip Setpoint Methodology for US Evolutionary Power Reactor," August 20, 2009, is referenced as providing a description of the uncertainties used in the DNBR calculations. Reference to ANP-10287P for the applicant's treatment of DNBR calculation uncertainties is acceptable, for the reasons set forth in the Safety Evaluation Report on the applicant's Topical Report ANP-10287P, "Incore Trip Setpoint Methodology for US Evolutionary Power Reactor," August 20, 2009.

In RAI 134, Question 04.04-29, the staff noted that the effects of crud on DNBR and reactor system pressure drop were not addressed as called for in SRP Section 4.4. RAI 231, Questions 04.04-54 and 04.04-55 were follow-up questions to RAI 134, Question 04.04-29 to obtain more information about crud effects. In a January 29, 2009, response to RAI 134, Question 04.04-29, the applicant stated that the presence of crud on the fuel rod surface only slightly degrades surface heat transfer. In an August 3, 2009, response to RAI 231, Question 04.04-54, the applicant stated that the temperature at the surface of the fuel rod is not significantly altered by typical crud deposition; hence, the probability of the formation of a vapor film at the rod-coolant interface is not significantly altered. The applicant stated that formation of crud is not expected to be of concern based upon current oxide measurements in operating plants that are using the same cladding material that the U.S. EPR will use. The staff concurs with this assessment and notes that any abnormal buildup of crud sufficient to affect pressure drop would be detected by a reduction in RCS flow, occurrence of crud induced power shifts, or abnormal water chemistry readings and would result in plant shutdown. In an August 3, 2009, response to RAI 231, Question 04.4-55, the applicant noted that it is standard procedure to develop CHF correlations used to calculate DNBR from tests using electrically heated rods with no simulation of crud deposition. It also noted that deposition of crud on a fuel rod would slightly increase both the surface roughness and surface area, resulting in a slight improvement in CHF. The staff agrees with the applicant's responses to RAI 134, Question 04.4-29, and RAI 231, Questions 04.4-54, and 04.4-55. The responses are, therefore, acceptable.

Core azimuthal power imbalance is addressed through a bounding LCO, as provided for in FSAR Tier 2, Section 16.3.2. The applicant states that the enthalpy rise hot channel factor, $F_{\Delta H}$, is sufficiently conservative to account for azimuthal imbalance within the bounds of the LCO, and the design heat flux hot channel factor, F_Q does not include a specific allowance for azimuthal imbalance. This is not necessary since the conservative allowance for $F_{\Delta H}$ will ensure a conservative allowance for F_Q. In a January 29, 2009, response to RAI 134, Question 04.04-30, the applicant stated that the incore instrumentation is sufficient to detect azimuthal power tilts.

Inter-assembly coolant flow mixing is treated explicitly in the NRC-approved LYNXT code methodology as described in licensing Topical Report BAW-10156-A. The application of the LYNXT code to the U.S. EPR has been approved by the staff in its safety evaluation of ANP-10263P-A.

The applicant describes the two hot channel factors used for the U.S. EPR design. A heat flux hot channel factor (F_Q) defines the maximum heat flux in the core relative to the average heat flux. An enthalpy rise hot channel factor ($F_{\Delta H}$) defines the maximum power fuel rod in the core. The bias that is typically put on F_Q and $F_{\Delta H}$ is approximately four percent. Because the applicant's treatment of engineering tolerances and uncertainties results in margin that is added to the best estimate values of F_Q and $F_{\Delta H}$, the treatment is conservative. Therefore, the staff concluded that the treatment of fuel parameter variations and tolerances relative to the hot channel factors as described by the applicant is conservative and finds that this is acceptable.

The effects of fuel rod bow and assembly bow are addressed through reference to NRC-approved licensing Topical Report BAW-10147P-A, "Fuel Rod Bowing in Babcock & Wilcox (B&W) Fuel Designs," May 1983, and BAW-10186P-A. The application of this methodology to the U.S. EPR fuel is documented in NRC-approved licensing Topical Report ANP-10263P-A. The staff has verified the applicability of the referenced methodology to the U.S. EPR. As described in a January 29, 2009, response to RAI 134, Question 04.04-33, the applicant stated that the fuel rod bow correlation from BAW-10147P-A and BAW-10186P-A are applicable to the U.S. EPR fuel design. The staff's review of Topical Report ANP-10285P, which is ongoing, will evaluate the acceptability of the applicant's statement. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

The core power distribution used in the thermal margin analysis is a synthesis of the limiting radial power distribution and limiting axial power distribution. The applicant states that the design analysis is performed by setting the hot rod to the maximum predicted $F_{\Delta H}$, applying a conservative radial power distribution to the rest of the assembly, and combining that with a single bounding core axial power shape. The staff has determined that this treatment of core power distribution for DNBR analysis is conservative and hence acceptable. Core power distributions are discussed further in FSAR Tier 2, Section 4.3, and evaluated by the staff in Section 4.3 of this report.

FSAR Tier 2, Section 4.4.4.5.2, "Core Analysis" summarizes the application of the LYNXT computer code and LYNXT analysis uncertainties. The applicant states that the DNBR calculations account for measurement uncertainties and penalties associated with control bands on system conditions, fuel rod and assembly bow, core flow inlet mal-distribution, and conservative assembly thermal mixing. The treatment of fuel densification uncertainty is not mentioned. Yet, FSAR Tier 2, Section 4.4.2.9.1, "Uncertainties in Fuel and Cladding

Temperatures Uncertainties in Fuel and Cladding Temperatures," states that the effect of fuel densification is included in the total uncertainty in fuel and cladding temperature. The staff requested additional information from the applicant since consideration of fuel densification was unclear. In a January 29, 2009, response to RAI 134, Question 04.04-34, the applicant clarified that fuel densification is explicitly modeled, based on the guidance of RG 1.126, in the COPERNIC code, BAW-10231P-A and is not treated as an uncertainty. The applicant proposed revising FSAR Tier 2, Section 4.4.2.9.1 to delete reference to fuel densification in the context of treatment of uncertainties. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff determines that the applicant has adequately addressed, explained, and corrected the discrepancy between FSAR Tier 2, Section 4.4.4.5.2 and FSAR Tier 2, Section 4.4.2.9.1. Therefore, the staff considers RAI 134, Question 04.04-34 resolved. As discussed above, COPERNIC models the fuel densification phenomenon. The staff finds the use of applicant's method of core analysis to be acceptable since it uses and approved methodology which is applicable to the U.S. EPR core..

In the case of a post-scram main steam line break (MSLB) event with loss of offsite power (LOOP), the staff expressed concern that the local assembly flow in the stuck rod region may be outside the range of the CHF correlation used to compute the DNB. The staff requested that the applicant demonstrate that the thermal hydraulic conditions at the time of minimum DNBR in the MSLB with LOOP analyses are within the correlation ranges. Or, if the thermal-hydraulic conditions are beyond the correlation ranges, demonstrate that the number of rods experiencing CHF is conservatively calculated.

In a November 18, 2008, response to RAI 141 Question 04.04-36, the applicant identified the Biasi CHF correlation, which is included in the NRC-approved thermal hydraulic code LYNXT, as the correlation used for evaluating DNB for the post-scram MSLB with LOOP event rather than the BWU-N or ACH-2 correlations. The applicant states that the Biasi CHF correlation provides a conservative evaluation of DNB and has a much broader range of applicability for local mass flux, relative to either BWU-N or ACH-2. The applicant states that the Biasi CHF correlation used for evaluating MSLB events is consistent with the NRC approved ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR." Furthermore, the applicant states when the parameters are outside of the range of applicability of the Biasi CHF correlation, DNBR is conservatively calculated. The staff examined ANP-10263P-A to confirm the applicant's statements, and determined they were accurate. The staff, therefore, finds the applicant's response acceptable and considers RAI 141 Question 04.04-36 resolved. The staff evaluation of whether DNBR is conservatively calculated outside the range of applicability of the Biasi CHF correlation is set forth in Chapter 15 of this report.

The applicant provided an engineering basis for concluding that the U.S. EPR is not susceptible to thermo-hydrodynamic instability, including core configuration, subcooled and high pressure operation, and long time constant fuel pellet. The criteria are the same as those applied to operating PWRs with respect to the prevention of thermo-hydrodynamic instability and have been previously accepted by the staff. Therefore, the staff determined that the EPR design will not experience any thermo-hydrodynamic instability during normal operation, including AOOs.

The effects of flow blockage on fuel rod behavior are addressed in this section of the FSAR Tier 2 by describing the design obstructions in the flow path that would stop loose parts in the system from reaching the fuel rods, such as lower internals and core plate, the fuel assembly lower end fitting including the debris filter, and the lower spacer grids. The staff concurs with the applicant's conclusions that only insignificant small parts are capable of reaching the location of minimum DNBR in the core and that there is no adverse effect relative to the design basis because the presence of such small parts will not significantly alter the local thermal-hydraulic conditions.

Hydraulic Loads

The applicant states that hydraulic loads on vessel components and fuel assemblies were evaluated but did not provide any description of the evaluation. The staff requested that the applicant provide a description of the hydraulic loads analysis. In a January 29, 2009, response to RAI 134, Question 04.04-26, the applicant stated that the hydraulic loads were assessed and a reactor coolant pump over speed transient was identified as the limiting event. This event, in turn, was bounded by a calculation of hydraulic loads at a postulated 20 percent increase in flow above the mechanical design flow rate. The applicant did not provide the results of that analysis for the staff to review. Therefore, a follow-up RAI was issued to address the staff's concern. **RAI 325, Question 04.04-61, which is associated with the above request, is being tracked as an open item.**

Also, in a June 25, 2009, response to RAI 231, Question 04.04-49, the applicant referred to Section 5.1.9 of ANP-10285P which 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...." However, the pump overspeed is not considered a credible AOO for the U.S. EPR, but is selected as the reference condition, to demonstrate compliance to the criteria, since it is the bounding flow condition of all other AOOs. Except for the issues raised by the pump overspeed transient, the staff finds the response to RAI 231, Question 04.04-49 acceptable. The staff will evaluate the pump overspeed transient in the Safety Evaluation Report on ANP-10285P. **RAI 339, Question 04.02-17, which is associated with the ongoing review of Topical Report ANP-10285P, is being tracked as an open item**.

The staff requested an explanation of how the statistical hold down methodology addresses the condition of a relaxed hold down spring. In an April 8, 2009, response to RAI 205, Question 04.04-50, the applicant explained how its statistical assembly hold down methodology addresses the condition of relaxation of a hold down spring due to neutron fluence. The applicant's response is acceptable to the staff since it refers to a previously approved methodology, BAW-10243PA, "Statistical Fuel Assembly Hold Down Methodology," September 2005, which is applicable to the U.S. EPR fuel assemblies. This licensing topical report was approved by the staff for application to PWR designs and its applicability to the U.S. EPR has been verified in the Safety Evaluation Report on ANP-10263P-A.

The applicant states that uncertainties in the hydraulic loads analysis are covered by the use of a minimum value for the core bypass flow to conservatively bound nominal operation of the reactor. In addition, hydraulic loads on the fuel assembly are evaluated for normal cold shutdown operation and for a 20 percent pump over speed transient condition using a statistical combination of uncertainties as described in the referenced licensing Topical Report BAW-10243P-A. As stated above, the staff has approved the application of this methodology to the U.S. EPR design in the Safety Evaluation Report on ANP-10263P-A.

Hydraulic Design

The reactor coolant flow data presented in the FSAR Tier 2, Section 4.4, shows a non-uniform core inlet distribution. The inlet flow distribution is determined primarily by a flow distribution device located below, and attached to, the core lower support plate. The flow distribution device shapes the core inlet flow, producing higher flows in the central region of the core. The applicant states that a flow mal-distribution penalty is conservatively applied to the hot channel assembly. The staff requested a description of the flow mal-distribution penalty. In a January 25, 2009, response to RAI 134, Question 04.04-25, the applicant stated that the nominal flow distribution factors for the central (zone 1) and peripheral (zone 2) assemblies were derived from one-fifth scale mockup testing. The lower tolerances of the test data for each of the two zones were determined from a statistical analysis of the test data. The difference between the low tolerance value and the nominal value is the mal-distribution factor. Depending upon the location of the hot assembly, either the zone 1 or the zone 2 mal-distribution factor is applied as a flow penalty to the hot assembly only. The applicant agreed to revise the FSAR to incorporate the added information. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. The staff finds this information is conservative because it increases the flow to the center of the core where the hot, or limiting, assemblies are typically located.

The staff also requested that the applicant summarize the applicability of the Juliette one-fifth scale test loop to the U.S. EPR, including physical configuration, scale, coolant mass flow, coolant system temperature and pressure, and core representation. In a June 25, 2009, response to RAI 231, Question 04.04-56, the applicant provided a description of the one-fifth test facility and test parameter ranges. The staff reviewed the data provided by the applicant and concluded that the flow distribution obtained in the Juliette test loop is applicable to the U.S. EPR. The applicant conducted scaling analysis which showed the Juliette test loop was applicable to the U.S. EPR design. The staff reviewed the information provided by the applicant and concurred with the applicant's assessment that the Juliette test loop was applicable to the U.S. EPR design. Based upon its review of the test facility and the derivation of the mal-distribution factors, the staff finds that the applicant has derived and applied the hot assembly flow penalty appropriately. The penalty conservatively reduces the inlet flow to the hot assembly. Therefore, the staff accepts the responses to RAI 134, Question 04.04-25 and RAI 231, Question 04.04-56, and considers the issues addressed by them to be resolved.

The total reactor vessel pressure drop represents the unrecoverable losses from vessel inlet nozzles, downcomer and lower plenum, the lower core support plate, the fuel assemblies, and the upper plenum. The total calculated pressure drop for hot, best estimate reactor coolant system flow is given in FSAR Tier 2, Section 4.4.2.7.2, "Total Core and Vessel Pressure Drop," is 207.5 kPa (30.1 psid). The pressure drop calculation used standard textbook formulations and is acceptable to the staff.

Total vessel pressure drop estimates made at the design stage are primarily to assess reactor coolant system loop flows. Initial plant testing under Pre-critical Test No. 182, "Post-Core RCS

Temperature Cross Calibration," and Pre-critical Test No. 183, "Post-Core Reactor Coolant System Flow Baseline," described in FSAR Tier 2, Section 14.2.12.14, "Phase II: Initial Fuel Loading and Precritical Tests," provide confirmation of thermal-hydraulic design parameters, including reactor coolant flow.

The applicant states that the minimum net positive suction head for reactor coolant pump operation will be provided, as identified in FSAR Tier 2, Table 5.4-1, "Reactor Coolant Pump Design Data." FSAR Tier 1 specifies ITAAC 7.3 for testing and analyzing reactor coolant pump flows. Reactor coolant pump operation is verified during initial plant testing under Hot Functional Test No. 170, "Pre-Core Reactor Coolant System Flow Model Verification," Pre-critical Test No. 182, "Post-Core RCS Temperature Cross Calibration," and Test No. 183, "Post-Core Reactor Coolant System Flow Baseline," described in FSAR Tier 2, Section 14.2.12.14.

The effects of reduced core flow from inoperable pumps caused by unplanned events are described in FSAR Tier 2, Chapter 15. FSAR Tier 1, Section 2.2.1 ITAAC 7.2 requires a measurement of minimum four-pump coastdown flow. In a February 27, 2009, response to RAI 158, Question 14.02-91, the applicant stated that FSAR Tier 2, Section 14.2 Initial Plant Test No. 170 "Pre-Core Reactor Coolant System Flow Model Verification," and Initial Plant Test No. 183, "Post-Core Reactor Coolant System Flow Baseline," will be revised to require that the four-pump coastdown be measured for comparison to the FSAR Tier 2, Chapter 15 accident analysis assumptions. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff determined that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 158, Question 14.02-91 resolved.

Initial Plant Test # 188, "Post-Core Incore Instrumentation," includes testing of the core exit Thermocouples. However, the staff determined that no criteria were provided to confirm the functionality of the core exit thermocouples. The staff issued RAI 158, Question 14.02-90 to obtain more information on the criteria used to confirm the functionality of the core exit thermocouples. In a February 27, 2009, response to RAI 158, Question 14.02-90, the applicant clarified that U.S. EPR FSAR Tier 2, Section 14.2.12, Test #188 quantifies the cable resistance and verifies that the resulting temperature indication during hot zero power (HZP) corresponds to actual reactor coolant system (RCS) temperature. The applicant further stated that Test #188 would be revised to clarify the prerequisites and test methods used. The staff confirmed that Revision 1 of the U.S. EPR FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff determined that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 158, Question 14.02-90 resolved.

Power operation in Modes 1 and 2 with one or more reactor coolant pumps out of service is prohibited per FSAR Tier 2, Chapter 16, TS 3.4.4.

In FSAR Tier 2, Section 4.4.5, "Testing and Verification," the applicant identifies the initial plant testing pertaining to the thermal-hydraulic design. Reference is made to FSAR Tier 2, Section 14.2 for reactor coolant system testing and for core power distribution measurement.

The staff has identified the following initial tests from FSAR Tier 2, Section 14.2 to be applicable to the thermal-hydraulic design described in this section of the FSAR:

- Test No. 012 "Reactor Coolant System"

- Test No. 030 "Reactor Coolant System Hydrostatic Test"

- Test No. 127 “Areoball System”
- Test No. 133 “Loose Parts Monitoring System”
- Test No. 141 “Incore Instrumentation”
- Test No. 142 “Excore Instrumentation”
- Test No. 146 “PS”
- Test No. 147 “RCSL”
- Test No. 170 “Pre-Core Reactor Coolant System Flow Model Verification”
- Test No. 182 “Post-Core RCS Temperature Cross Calibration”
- Test No. 183 “Post-Core Reactor Coolant System Flow Baseline”
- Test No. 188 “Post-Core Incore Instrumentation”
- Test No. 207 “Steady State Core Performance”

The staff has evaluated the above-listed initial tests against the thermal-hydraulic design described in FSAR Tier 2, Section 4.4, and to the requirements of 10 CFR 52.47(b)(1) and the guidance contained in RG 1.68. The staff has determined that the testing identified in FSAR Tier 2, Section 14.2 adequately addresses the thermal-hydraulic design, including reactor coolant system flow verification, functionality of the associated instrumentation, monitoring, and trip systems, and verification of core power distributions, with inclusion of the following:

- Test No. 183 requires a measurement and evaluation of four-pump reactor coolant pump flow coastdown, as stated in the applicant’s February 27, 2009, response to RAI 158, Question 14.02-91.
- Test No. 188 requires an acceptance criterion to address the functionality of the core exit thermocouples, as stated in the applicant’s February 27, 2009, response to RAI 158, Question 14.02-90.

The staff confirmed that the test summaries for Test Nos.183 and 188 contained in Revision 1 of the FSAR, dated May 29, 2009, contain the changes committed to in the RAI responses. Accordingly, the staff finds that the applicant has adequately addressed these issues and, therefore, the staff considers RAI 158, Questions 14.02-90 and14.02-91 resolved.

Based on the foregoing, the staff concludes that the testing and verification program described in FSAR Tier 2, Section 4.4.5 is acceptable.

<u>Instrumentation</u>

FSAR Tier 2, Section 4.4.6, “Instrumentation Requirements,” describes the instrumentation requirements pertaining to the thermal-hydraulic design of the U.S. EPR. As described by the applicant, the incore neutron flux distribution is determined through the use of two separate systems: The fixed incore power distribution detector (PDD) system and the movable AMS.

The PDD system is comprised of 72 fixed self-powered neutron detectors (SPND) located at twelve radial locations in the core and six axial elevations per radial location. The PDD system provides continuous power distribution data to the RCSL system which utilizes on-line core monitoring software to continuously evaluate reactor core conditions. The information provided by the RCSL system is used by control room operators to monitor core parameters for compliance with Technical Specifications LCO requirements, as described in FSAR Tier 2, Section 16. Signals from the PDD system are also utilized by the protection system (PS), as described in FSAR Tier 2 Section 7.2. FSAR Tier 1, Section 2.4.19 ITAAC 4.2 requires testing be performed to verify the incore detectors provide signals as designed to the protection system.

The AMS is an on-line incore flux mapping system used to measure neutron flux density over the entire axial length of the core at 40 radial locations. The incore neutron flux measurements from the AMS are used to calibrate the SPND.

SRP Section 4.4, “Thermal and Hydraulic Design,” Part I, “Areas of Review,” provides that the core monitoring techniques that rely on incore or excore neutron sensor inputs be evaluated. The AMS includes pressure boundary penetrations and is used to calibrate the SPND safety system. The U.S. EPR AMS system is appropriately classified as a safety-related system with respect to its pressure boundary functions. The AMS is not relied upon to remain functional during and following design-basis events, nor is the AMS relied upon to prevent or mitigate a design basis accident. However, the AMS does constitute a portion of the RCPB, therefore the system design, as it relates to the pressure boundary, must meet the requirements of GDC 1, “Quality Standards and Records,” GDC 2, “Design Bases for Protection Against Natural Phenomenon,” GDC 4, “Environmental and Dynamic Effects Design Bases,” GDC 10, and 10 CFR 50.55a. The AMS is used to generate a flux map approximately every two weeks. There is no direct software or hardware link from the AMS computer to the PS.

The incore instrumentation system provides indication of reactor power levels that are independent and diverse from the reactor power levels indicated by the U.S. EPR excore instrumentation system. The SPNDs need periodic calibration; as do the excore detectors. This calibration is done approximately every 2 weeks using the AMS. The AMS allows the insertion and removal of alloy balls, also known as Aeroballs, to and from the core. The Aeroballs contain vanadium, which becomes activated upon irradiation in the core. The irradiated Aeroballs are counted at a measurement table and the activation rate data is supplied to the independent AMS computer. Based on these activation measurements, the AMS computer generates a snapshot in time of the core power distribution. The measured activation rates of the Aeroballs are used to construct a 3-D core power distribution map which identifies the position of both the minimum DNBR and the limiting LPD.

The staff reviewed the information contained in FSAR Tier 2, Section 4.4.6.2, and determined that the technical information provided on the AMS system did not address a variety of details as to the design and operation of the system. The staff requested additional information regarding the AMS. The RAIs and responses regarding the AMS system are summarized in the following text.

The staff was concerned about the containment isolation capability of the nitrogen supply used to transport the Aeroballs. Therefore, the staff requested that the applicant describe the containment isolation capability of the system. In a March 24, 2009, response to RAI 194, Question 04.04-01, the applicant explained the supply is through a containment penetration with an inner and outer isolation valve that close automatically upon a containment isolation signal. Since inner and outer containment isolation is employed, the response is acceptable and the staff considers RAI 194, Question 04.04-01 resolved. The capability of these containment isolation valves to perform their function is evaluated in Section 6.2 of this report.

Since the guide tubes are small diameter tubing, the staff was concerned about the impact of deformation and requested information regarding AMS guide tube deformation and the impact it would have operationally and neutronically. In a March 24, 2009, response to RAI 194 Question 04.04-2, the applicant explained that the AMS guide tubes are physically separated from the SPNDs. An individual AMS tube may be deactivated by an operator, thereby isolating that particular tube from the rest of the system. Due to the physical layout of the instrumentation lances, deformation of an AMS guide tube would have no neutronic impact on an SPND measurement. The response addresses the staff's concern and the staff considers RAI 194, Question 04.04-2 resolved.

In order to better understand the AMS the staff requested that the applicant provide detailed design information. In a March 24, 2009, response to RAI 194 Question 04.04-4, the applicant provided a description of the AMS addressing system design criteria, physical layout of the AMS, redundancy of key elements, moisture sensing, ability to isolate individual aeroball transport tubes, detector calibration, and the treatment of aeroball transport times in the counting process. The applicant also discussed the operational experience of the AMS in the twelve operating plants which use the system, and reported that the data obtained over the last 30 years of operation for 220 fuel cycles show that the last significant degradation of an AMS occurred in 1991. The AMS does not perform safety functions with respect to the features identified above. Rather, operation of a U.S. EPR would rely on the safety-related systems such as the in-core and ex-core nuclear instrumentation. Should the AMS be unavailable to calibrate the safety related instrumentation, operation of the U.S. EPR would be limited in accordance with TS. The response provided the detailed information and the staff considers RAI 194, Question 04.04-4 resolved.

Since the AMS is part of the reactor coolant pressure boundary, and is utilized to calibrate the safety-related SPNDs used for the core protection function, the staff requested that the applicant provide a description of the underlying design basis for the AMS, including compliance with applicable 10 CFR Part 50 GDC and conformance with applicable regulatory criteria and Institute of Electrical and Electronic Engineers (IEEE) Standards. In an April 8, 2009, response to RAI 205, Question 04.04-37, the applicant stated that the response to RAI 04.04-4 provides an operational overview of the AMS. The AMS is classified as non-safety related except for its pressure boundary components. The SPNDs and the core outlet thermocouples (COTs), which share the common lance yoke assembly, are classified as safety related. Consequently, these components are seismically qualified and meet IEEE Std Class 1E seismic requirements in accordance with IEEE Std 344-2004. Therefore, the instrumentation lances that house the SPNDs, the COTs, and the portions of the AMS located inside the core are seismically qualified as Seismic Category I. The structural analysis of the AMS instrumentation lance and associated components located within the reactor vessel and nozzle closure uses the analysis methods and strength criteria stipulated in ASME B&PV Code, 2004 Edition. The response clarifies the equipment classification and the staff considers RAI 205, Question 04.04-37 resolved.

The staff requested information regarding Lance Yoke Vibration Load. In an April 8, 2009, response to RAI 205, Question 04.04-40, the applicant stated that the lance yoke and lance shaft are not exposed to significant cross flow conditions. As a result, their response to flow excitations resulting from turbulence in the reactor vessel channel head is insignificant. This magnitude of flow represents less than 0.6 percent of the total mass flow rate through the reactor vessel. This flow rate coupled with the large volume of the upper head region will not produce cross flow conditions with the capability to create significant flow excitation to these structures. The response shows the concern to be negligible. The staff concurs and, therefore, considers RAI 205, Question 04.04-40 resolved.

The staff also sought to confirm the seismic design of the instrument lances and issued RAI 205, Question 04.04-41. In an April 8, 2009, response to RAI 205, Question 04.04-41, the applicant indicated that the instrumentation lances will be seismically qualified as Seismic Category I. The response clarifies the classification and the staff considers RAI 205, Question 04.04-41 resolved.

In order to better understand the operation of the AMS, the staff requested additional description of the sequence of operation of the AMS following irradiation of the Aeroballs. In an April 8, 2009, response to RAI 205, Question 04.04-44, the applicant clarified the procedure for transporting the aeroballs to the counting table after irradiation and their removal from the reactor. The applicant’s response emphasized that Aeroballs from only one core quadrant at a time are transported to the counting table. The measurement process described in the response is appropriately detailed and the staff considers RAI 205, Question 04.04-44 resolved.

The staff requested additional information about operational experience with the AMS. In an April 8, 2009, response to RAI 205, Question 04.04-47, the applicant explained that although the U.S. EPR will operate at a higher power level and an increased core flow relative to currently operating plants using the AMS, the performance of the AMS in the U.S. EPR is expected to be similar. The response adequately identifies the difference between the U.S. EPR and operating plants and the staff, therefore, considers RAI 205, Question 04.04-47 resolved.

Additionally, a public meeting on the AMS was held at the NRC headquarters on September 16, 2009; an audit of the SPNDs supporting documentation was held at the AREVA offices in Bethesda, Maryland on September 17, 2009; and conference calls were conducted with the applicant. As a result of these interactions with the applicant, the staff issued two questions to address remaining concerns with the operation and use of the AMS system. The staff requested that the applicant provide a testing plan to verify the accuracy of the correction algorithms that are applied to the raw AMS activation measurements. **RAI 308, Question 04.04-59, which is associated with the above request, is being tracked as an open item.** The staff also requested that the applicant provide the methodology to remove the ^{60}Co background from SPND measurements and how this background will be treated by the protection system and the AMS POWERTRAX/E calculations. **RAI 308, Question 04.04-60, which is associated with the above request, is being tracked as an open item.**

With the exception of the open items identified above, and for the reasons set forth above, the staff concludes that the AMS design and operation provides an acceptable methodology to calibrate the incore trip instrumentation (i.e., the SPNDs). Furthermore, the staff concludes that, based on this calibration, the EPR incore protection signals, which utilize the SPND readings to ensure that specified acceptable safety design limits, are not compromised during U.S. EPR operation as required by GDC 10.

The following items highlight the staff's specific findings regarding the AMS:

- The non-safety related portions of the AMS are of a quality commensurate with its importance in calibrating the safety-related nuclear instrumentation. Should the AMS be unavailable to calibrate the safety related instrumentation, operation of the U.S. EPR would be limited in accordance with TS. Therefore, it satisfies the requirements of GDC 1.

- The portions of the AMS that are within the reactor pressure vessel boundary and are shared with other safety systems are classified as Seismic Category I, and meet the applicable IEEE Std Class 1E requirements. Therefore, the safety-related portions of the AMS design satisfy the requirements of GDC 1, GDC 2, and GDC 4.

- The AMS system provides a technically acceptable methodology to calibrate the incore trip instrumentation (i.e., the SPNDs). Proper calibration of the incore trip system is necessary to satisfy the requirements of GDC 10, GDC 20, "Protection System Functions," and GDC 29, "Reactivity control system redundancy and capability."

- AMS measurements provide a full core power map, which, when properly analyzed, ensures that DNB and LPD limits are not exceeded during normal operation. In addition, AMS measurements provide a burnup tracking function that ensures that predictive core reload calculations are still applicable. Therefore, the AMS measurements satisfy the requirements of GDC 13, "Instrumentation and Control."

- The portions of the AMS that are within the reactor pressure vessel boundary are designed with an extremely low probability of leakage. Even if an AMS tube were to break, the leak flow would be small enough to be within the makeup capacity of a single charging pump. Therefore, the requirements of GDC 14, "Reactor Coolant Pressure Boundary," are satisfied.

The excore neutron monitoring system described by the applicant consists of three source range detectors, four intermediate range detectors, and four power range detectors located between the reactor vessel and the primary shield. The source range detectors are used to monitor subcritical neutron multiplication during shutdown and refueling operations. The intermediate range detectors are used for neutron flux monitoring during approach to criticality and during power escalation; the intermediate range detectors also provide input to protection system trips and the control rod withdrawal block function. The intermediate range detectors can also be used for post accident monitoring of core power level. The power range detectors are used for monitoring reactor power level and power shape, including azimuthal power imbalance, and they also provide input to protection system trips. FSAR Tier 1, Section 2.4.17, ITAAC 4.2 requires testing be performed to verify the excore detectors provide signals to the protection system in accordance with the design.

The details of the neutron detectors and design of the associated instrumentation are described in FSAR Tier 2, Chapter 7, "Instrumentation and Controls," and evaluated in Chapter 7 of this report.

The low DNBR and high LPD monitoring and trip function requirements are also described in this section of the FSAR Tier 2, Sections 4.4.6.4 and 4.4.6.5. During normal operation, the online calculated DNBR is kept above the TS LCO value. If the LCO threshold is crossed, countermeasures are initiated to alert the plant control room operator and terminate further increase in power level.

Online algorithms in both the core monitoring system and protection system utilize incore detector signals together with reactor inlet temperature, pressure, and core flow signals as input to the calculation of DNBR. The LPD monitoring and protection functions perform in similar fashion, utilizing incore detector signals to synthesize hot spot linear power density (kW/ft). The description of the DNBR and LPD monitoring and protection systems indicate they will satisfy the requirements of GDC 10 to ensure that SAFDL are not exceeded during normal operation and AOOs. Therefore, the descriptions are acceptable to the staff. The ability of the systems to perform as described is evaluated in Chapter 7 of this report.

Technical Specifications 2.0 and 3.2 provide the safety limits and LCOs for DNBR and LPD which are evaluated under FSAR Tier 2, Chapter 16. The staff has reviewed the bases for these TS and finds that they adequately reflect the functions described in FSAR Tier 2, Section 4.4.

In response to the TMI Action Plan Item II.F.2 of NUREG-0737 as related to instrumentation for primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, FSAR Tier 2, Section 4.4.6.1, "Fixed Incore Instrumentation," states that incore thermocouples are provided at each of the 12 SPND locations. FSAR Tier 2, Section 16B.3.2 describes the post-accident monitoring system instrumentation which allows a calculation and display of margin-to-saturation utilizing incore thermocouple readings, as well as reactor coolant system temperature and pressure as input. The information provided by the applicant meets the requirements of 10 CFR 50.34(f)(2)(xviii) and the general guidelines of Item II.F.2 of NUREG-0737 for primary coolant saturation monitoring, including number and spatial distribution of sensors, display and alarm, and TS. The detailed design of the post-accident monitoring system, including the margin-to-saturation instrumentation, is evaluated in Section 7.5 of this report.

The applicant notes that a loose parts monitoring system is provided as part of the U.S. EPR design. It is designed to detect, locate, and analyze loose parts or foreign bodies in the reactor coolant system and in the secondary side of the steam generators. The loose parts monitoring system is tested during initial plant startup, as described under Test No. 133 in FSAR Tier 2, Section 14.2. The applicant states that the U.S. EPR loose parts monitoring system conforms to the guidance of RG 1.133. The non-safety related loose parts monitoring system provides indication of the presence of larger loose parts to afford the operators to take corrective action they deem necessary. The primary protection against debris-related fuel damage is the fuel bottom nozzle debris filters described above. These filters ensure that only small parts are capable of reaching the location of minimum DNBR in the core, such that there is no adverse effect relative to the design basis.

4.4.5 Combined License Information Items

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

4.4.6 Conclusions

The thermal-hydraulic design of the U.S. EPR reactor core was reviewed and evaluated by the staff. The scope of the review included the design bases specification, the steady-state analysis of the core thermal-hydraulic performance, analytical methods, testing and verification, and instrumentation. In several instances, pertinent sections of the applicant's referenced licensing topical reports were reviewed by the staff in order to evaluate the design and to confirm applicability of the referenced methodologies.

Based on the staff's review and evaluation of the material provided in FSAR Tier 2, Section 4.4, and except for the open items as listed above, the staff concludes that the analytical methods previously reviewed and approved by the staff are applicable to the U.S. EPR, and that the U.S. EPR thermal-hydraulic design meets the requirements of GDC 10, GDC 12, 10 CFR 52.47(b)(1) and 10 CFR 50.34(f)(2)(xviii).

The applicant will also provide the methodology to remove the ^{60}Co background from SPND measurements and explain how this background will be treated by the Protection System and the AMS POWERTRAX/E calculations in order to address the associated open items described above.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 *Introduction*

FSAR Tier 2, Section 4.5.1, "Control `Rod Drive System Structural Materials," describes the materials used in the control rod drive mechanisms for both the reactor coolant system pressure boundary portion of the CRDM, and non-pressure boundary CRDM components.

4.5.1.2 *Summary of Application*

FSAR Tier 1: There are two FSAR Tier 1 requirements that are applicable to CRDM structural materials. These are given in FSAR Tier 1, Section 2.2.1 Paragraph 3. They are Items FSAR Tier 1, Sections 3.1 and 3.11. These relate to meeting the requirements of Section III of the ASME B&PV Code and the requirement for a fatigue analysis of Class 1 components that is also a requirement of the ASME Code.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in FSAR Tier 2, Section 4.5.1 summarized here, in part, as follows.

The application describes materials specifications, fabrication and processing of stainless steel components, contamination protection and cleaning of austenitic stainless steel, and provides information related to materials other than austenitic stainless steels.

The materials used in the CRDM pressure boundary that are in contact with reactor coolant include stabilized austenitic stainless steel, martensitic stainless steel, stainless steel weld filler metal, and nickel base alloy weld filler metal. Precipitation hardening austenitic iron-nickel-chromium bolting studs and martensitic stainless steel nuts are part of the pressure boundary but are not exposed to reactor coolant. These pressure boundary materials are selected based on their compliance with ASME Code, Section III. The applicant did not identify the use of any Code Cases given in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

The non-pressure retaining CRDM components are fabricated from materials meeting Deutsches Institut für Normung E.V. (DIN) specifications, Design and Construction Rules for Mechanical Components of PWR Nuclear Islands, RCC-M specifications (French ASME equivalent), and Society of Automotive Engineers/Aerospace Material Specifications (SAE/AMS). In addition to the standard specifications above, the applicant has what it refers to as ordering requirements which impose additional restrictions on these materials. These materials include stabilized austenitic stainless steel, martensitic stainless steel, nickel-based alloy, cobalt-chromium alloy, and cobalt based hard facing material. Non-pressure boundary materials are selected based on their compatibility with the reactor coolant and a proven German design with 30 years of operating experience.

Austenitic stainless steels are solution annealed. Martensitic stainless steels are quenched and tempered. Cobalt-chromium alloy is delivered in the solution annealed condition. Nickel-based alloy is solution annealed and thermally aged. Sliding surfaces of the latch unit are hard chromium plated.

ITAAC: FSAR Tier 1, Section 2.2.1, Table 2.2.1-5, "RCS Equipment Mechanical Design," contains ITAAC that are applicable to CRDMs. Entries in this table pertinent to CRDMs are FSAR Tier 1, Sections 3.1 and 3.11. These relate to meeting the requirements of Section III of the ASME B&PV Code and the requirement for a fatigue analysis of Class 1 components.

4.5.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 4.5.1 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 4.5.1 of NUREG-0800.

1. 10 CFR Part 50 Appendix A, GDC 1, and 10 CFR 50.55a, "Codes and Standards," require that structures, systems, and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety-functions performed. 10 CFR 50.55a also incorporates by reference applicable editions and addenda of the ASME B&PV Code. Application of 10 CFR 50.55a and GDC 1 to the control rod drive structural materials provides assurance that the control rod drive system will perform as designed.

2. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Application of GDC 14 assures that control rod drive materials are selected, fabricated, installed, and tested to provide assurance of an extremely low probability of significant degradation and, in the extreme, to minimize the potential for a gross RCPB failure that could substantially reduce the capability to contain reactor coolant inventory or to confine fission products.

3. GDC 26 requires, in part, that one reactivity control system use control rods and that this system be capable of reliably controlling reactivity changes.

Regulatory guidance used to meet the above requirements includes:

1. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," May 2007.

2. RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," October 2007.

3. ASME NQA-1-1994, "Quality Assurance Program Requirements for Nuclear Power Plants."

4.5.1.4 *Technical Evaluation*

The staff reviewed and evaluated the information included in FSAR Tier 2, Section 4.5.1, to ensure that the materials specifications, fabrication, and process controls are in accordance with the criteria of SRP Section 4.5.1.

Materials Specifications

The staff reviewed FSAR Tier 2, Section 4.5.1, to determine the suitability for service of the materials selected for CRDM components. FSAR Tier 2, Section 4.5.1 provides information on the specifications, types, grades, heat treatments, and properties used for the materials of the CRDM components.

The CRDM components that are part of the RCPB include materials used for the pressure housing and are described in FSAR Tier 2, Section 4.5.1.1, "Materials Specifications," and additional information is provided in FSAR Tier 2, Table 5.2-2, "Material Specifications for RCPB Components." The materials used for the pressure housing components include niobium stabilized stainless steel (Grade 347), martensitic stainless steel (Type F6NM), precipitation hardening austenitic iron-nickel-chromium (SA-453 Grade 660) bolting studs, and martensitic stainless steel nuts (SA-437 Grade B4C). The bolting studs and nuts are not exposed to reactor coolant. Welding filler materials Alloy 52/52M, Alloy 152, and Type 347 austenitic stainless steel will be used in the fabrication of the CRDM pressure housing. The staff reviewed the specifications and grades of the CRDM pressure housing materials and verified that the materials listed meet the requirements of ASME Code, Section III, Paragraph NB-2121, "Permitted Materials Specifications," which requires the use of materials listed in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B. The staff verified that the materials identified by the applicant are acceptable materials for use in ASME Code, Section III, Class 1 systems, except as noted below.

The RCPB materials specified in FSAR Tier 2, Table 5.2-2 for the CRDM pressure housing lists martensitic stainless steel materials SA-182 Grade F6NM (UNS S41500), and SA-479 (UNS S41500). SA-182 Grade F6NM is given in ASME Code, Section II, Part D, Subpart I, Table 2A, for use in Class 1 components in accordance with ASME Code Section III, NB-2121, "Permitted Material Specifications." The staff noted that SA-479 UNS S41500 is not given in Table 2A.

In RAI 88, Question 05.02.03-1, the staff requested that the applicant delete SA-479 UNS S41500 from FSAR Tier 2, Table 5.2-2, provide an alternative material, or take the appropriate steps to include SA-479 UNS S41500 in Table 2A of ASME Code, Section II. In a November 10, 2008, response to RAI 88, Question 05.02.03-1, the applicant stated that in August 2008, the applicant submitted a request to ASME to extend the properties currently provided in Section II Part D for SA-182 Grade F6NM (UNS S41500) to SA-479 (UNS S41500). The applicant stated that ASME is expected to issue a Code Case in the near future. This issue will remain an open item until ASME issues the applicant's Code Case, and the staff reviews its acceptability. **RAI 342, Question 04.05.01-7, which is associated with the above request, is being tracked as an open item.**

In RAI 88, Question 05.02.03-12, the staff requested information related to the fabrication of the CRDM pressure housing. In a November 10, 2008, response to RAI 88, Question 05.02.03-12, the applicant provided a description of fabrication of the CRDM pressure housing. In that response, the applicant provided a sketch of the CRDM housing including weld locations. The applicant's sketch shows that components are made from stainless steel grades TP347, F347, and F6NM. The staff noted that FSAR Tier 2, Table 5.2-2 does not list grade F347. The staff requested that the applicant clarify this discrepancy in a follow-up RAI. **RAI 342, Question 04.05.01-6, which is associated with the above request, is being tracked as an open item.**

Except for the above open items, the staff determined that the materials and materials specifications selected for the CRDM pressure housing meet the requirements of ASME Code Sections II and III and thus comply with GDC 1 and 10 CFR 50.55a and are, therefore, acceptable for use in the U.S. EPR design.

Application of GDC 26 to the control rod drive system materials ensures that the material selection and fabrication support reliable rod movement for reactivity control that preserves fuel and cladding integrity. Accordingly, components of the CRDM that are not part of the RCPB must be fabricated from materials that will assure that they function reliably in order to meet the requirements of GDC 26. U.S. EPR non-RCPB CRDM components are fabricated from materials meeting DIN specifications, RCC-M specifications, and SAE/AMS specifications. In addition to the standard specifications above, the applicant imposed additional restrictions in procuring these materials. FSAR Tier 2, Table 4.5-1 lists materials specifications and grades used for CRDM components. In addition, the FSAR Tier 2, Table 4.5-1 provides U.S. material specifications and grades comparable to the DIN and RCC-M specifications selected by the applicant where applicable.

The materials used for internal CRDM components include the following:

- Materials WNr. 1.4006 DIN EN 10088-3 (bars), DIN 17456 (Seamless tubes) or X12Cr13 RCC-M 3207. These materials are similar to ASME SA-479 type 410 (bars) and ASME SA-268 TP 410 (seamless tubes).
- Materials WNr. 1.4550, DIN EN 10088-3 or Z6CNDT18-12, RCC-M 3306. These materials are similar to ASME SA-479 Type 347.
- Materials WNr. 1.4571 DIN EN 10088-3 or Z8CNDT18-12 RCC-M 3306. These materials are similar to ASME SA-479 Type 316Ti.
- Cobalt-Chromium alloy Haynes 25, SAE/AMS 5759.
- Nickel based Alloy X-750, SAE/AMS 5698.

- Cobalt based Stellite 6 alloy meeting the AREVA ordering specification. The specified welding filler material is similar to ASME Classifications ERCoCr-A, ERCCoCr-A and ECoCr-A.

- Martensitic stainless steel weld filler material G(W) Z 17 Ti based on DIN EN 12072. This welding filler material is similar to ASME SFA 5.9 ER430.

The latch unit sliding surfaces are hard chrome plated. The applicant stated that these materials were selected based on a proven German design with 30 years of operating experience. In addition, the applicant stated that the materials are selected for their compatibility with reactor coolant, as required by ASME Code, Section III, Articles NB-2160 and NB-3120.

Given that there is no history regarding the use of materials meeting the DIN/RCC-M specifications in the current operating U.S. fleet, the staff requested, in RAI 15, Question 04.05.01-1, that the applicant provides a comparison of the given European material specifications to ASME Code material specifications and grades. In addition, the staff requested that the comparisons address differences in DIN and ASME specifications in chemistry, processing, final metallurgical condition, mechanical properties, and design allowable stresses. In an August 22, 2008, response to RAI 15, Question 04.05.01-1, the applicant provided a comprehensive comparison of the U.S. EPR materials procurement specifications to ASME Code Section II material specifications where applicable. The comparisons provided by the applicant include chemistry, mechanical properties, and heat treatment specifications.

The staff reviewed the information provided by the applicant in response to the staff's RAI regarding materials fabricated to DIN and RCC-M material specifications and applicable portions of the applicant's material procurement specifications. The staff determined that the materials specified, along with the applicant's procurement specifications, are essentially the same as the comparable ASME specifications that the applicant has given in FSAR Tier 2, Table 4.5-1. SAE/AMS materials given by the applicant are widely used in the U.S. and are common materials used in CRDM components in other reactor designs. In addition, the materials have been used in German designed CRDMs for over 30 years. The staff determined that these materials are of sufficient quality and reliability for use in non-RCPB CRDM components based on the suitability of these materials for their intended application and their successful operating history. In view of the foregoing, the staff determined that the materials and materials specification for the materials used in the CRDM are acceptable and meet GDC 1, GDC 14, GDC 26, and 10 CFR 50.55a.

Austenitic Stainless Steel Components

FSAR Tier 2, Section 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steels," describes the fabrication and processing of austenitic stainless steels to minimize susceptibility to stress corrosion cracking. The staff's evaluation of the applicant's fabrication and processing of austenitic stainless steels in CRDM RCPB components is discussed in Section 5.2.3 of this report.

For non-pressure boundary components, the applicant has selected stabilized grades of stainless steels similar to ASME SA-479 Type 347 (Niobium stabilized) and ASME SA-479 Type 316Ti (Titanium stabilized). These materials are used to fabricate latch unit and drive rod components and contain additions of niobium and titanium to prevent sensitization thus greatly reducing these materials susceptibility to intergranular stress-corrosion cracking (IGSCC). In addition, the applicant has specified corrosion testing to verify that its material processing procedures minimize the susceptibility of these materials to intergranular corrosion. Therefore, the staff considers the applicant's use of stabilized grades of stainless steels acceptable, because they provide an increased resistance to stress corrosion cracking when compared to unstabilized materials.

RG 1.44 provides staff guidance related to fabrication and processing of unstabilized austenitic stainless steels to avoid severe sensitization which can increase the susceptibility of stress corrosion cracking. Given that unstabilized grades of austenitic stainless steel are not used in the CRDM, staff guidance provided in RG 1.44, "Control of the Use of Sensitized Stainless Steel," is not applicable to the U.S. EPR CRDM components. Cold working can increase the susceptibility to stress corrosion cracking in stabilized and non-stabilized grades of stainless steels. In FSAR Tier 2, Section 4.5.1, the applicant stated that cold-worked stainless steels are not used in the control rod drive system. In addition, no austenitic stainless steel materials with yield strengths greater than 620,528 MPa (90,000 psi) are used in the control rod drive and drive systems, which conform to SRP 4.5.1.

Improper use of tools to perform abrasive work on austenitic steel can lead to contamination of surfaces which could promote stress corrosion cracking. RG 1.37 provides guidance by the staff on acceptable methods to control surface contamination caused by abrasive operations such as grinding. FSAR Tier 2, Section 4.5.1.2, "Austenitic Stainless Steel Components," states that abrasive work on austenitic stainless steel surfaces are controlled using the guidelines of RG 1.37.

Based on the above, the staff finds the applicant's fabrication, processing, and cleanliness controls for austenitic stainless steels acceptable.

Other Materials

Materials other than austenitic stainless steels are used to fabricate non-pressure boundary CRDM components. These materials include martensitic stainless steel (similar to Type 410), cobalt-chromium alloy (Haynes 25), nickel based alloy (Alloy X-750), cobalt based material (Stellite 6), and martensitic stainless steel weld filler material (ER 430 modified).

The FSAR identifies that the latch unit compression springs and drive rod compression springs (non-RCPB components) are fabricated from Alloy X-750. The FSAR indicates that this alloy will be supplied in the solution annealed (followed by quenching) and thermally aged condition for optimum resistance to stress corrosion cracking. Alloy X-750 is a commonly used material in CRDM components, as referenced in SRP 4.5.1, and has had a successful operational history in currently operating PWRs. In an October 28, 2008, response to RAI 15, Question 04.05.02-2, the applicant provided detailed information regarding its thermal treatment of Alloy X-750 springs used in the CRDMs. The specific details regarding the applicant's thermal treatment of Alloy X-750 are proprietary. The staff reviewed the thermal treatment methods and stress corrosion cracking testing results provided by the applicant and determined that the applicant's thermal treatment of Alloy X-750 provides reasonable assurance that stress corrosion cracking will not occur in these components, and is therefore acceptable.

FSAR Tier 2, Table 4.5-1 indicates that latch unit magnetic parts and some drive rod components are fabricated from W. Nr. 1.4006 or X12Cr13 (Type 410 martensitic stainless steel). The applicant stated that this material is delivered in a quenched-and-tempered condition. The minimum tempering temperature specified by the applicant conforms to SRP Section 4.5.1 and is, therefore, acceptable.

FSAR Tier 2, Table 4.5-1 indicates that latch-unit pins and the drive-rod spreader button are fabricated from Haynes 25 ordered to SAE/AMS 5759. Haynes 25 is a cobalt-nickel-chromium-tungsten alloy which has high strength and good oxidation resistance at elevated temperatures. In an August 22, 2008, response to RAI 15, Question 04.05.01-1, the applicant provided additional information related to the use of Haynes 25. The applicant stated that Haynes 25 has been used in approximately 15 German plants over a period of 30 years without any reported failures. In addition, Haynes 25 was also used in CRDMs in Babcock & Wilcox designed PWRs. Periodic inspections of B&W CRDMs removed from service have showed some expected wear but no corrosion-related degradation. Given the favorable performance history of Haynes 25, the staff has determined the applicant's use of this material to be acceptable.

Latch tips are hard faced with Stellite 6 which is a cobalt based material. Radiation buildup during plant operation can occur because of cobalt-60, which forms by neutron activation of cobalt-59. The staff understands that in some cases, the use of cobalt based materials is unavoidable due to service conditions and functional demands. The applicant stated that Stellite 6 was chosen for this application, because an alternative material will not perform satisfactorily in service. In addition, Stellite 6 is used in a very small portion of the CRDM. The staff determined this to be acceptable, because the material performs as needed, and the applicant has limited the use of cobalt based materials to the extent practical.

FSAR Tier 2, Section 4.5.1 indicates that the materials used to fabricate the CRDMs are selected based on a proven German design with 30 years of operating experience. The staff requested that the applicant verify that all materials given in FSAR Tier 2, Table 4.5-1 have significant (i.e., 30 years) operating experience in service and describe if evidence of general corrosion, stress-corrosion cracking or other forms of degradation has been discovered in operating foreign PWR nuclear plants. In an August 22, 2008, response to RAI 15, Question 04.05.01-1, the applicant stated that the CRDM design for the U.S. EPR is a proven German design used in Kraftwerk Union (KWU) PWRs since 1969. The materials noted in FSAR Tier 2, Table 4.5-1 have been used in the German design for the past 30 years. Of 937 CRDMs installed, 856 CRDMs remain in service. No evidence of general corrosion, stress-corrosion cracking, or other types of degradation has been noted in the operating CRDM KWU PWR fleet. The applicant did note that some units built before 1988 needed repair due to mechanical failure of Alloy X-750 compression springs. The springs were redesigned and have been in service since 1988 with no repair or replacement necessary. Based upon the information provided above, the staff finds the materials given in FSAR Tier 2, Table 4.5-1 to be acceptable in this regard.

Compatibility of Materials with Reactor Coolant

Materials selected for use in the CRDM must be compatible with reactor coolant, as described in NB-2160 and NB-3120 of the ASME Code, Section III. The information in the FSAR indicates that the materials used in the CRDM are compatible with reactor coolant. The materials selected for the CRDM are currently in use in nuclear power plants and have proven to perform satisfactorily under environmental conditions found in these plants. Therefore, the staff finds that the materials used for CRDM components comply with ASME Code, Subarticles NB-2160 and NB 3120 and are acceptable for use.

Cleaning and Cleanliness Controls

SRP 4.5.1 recommends that onsite cleaning and cleanliness controls for CRDMs should be in accordance with RG 1.37, which specifies the use of ASME NQA-1-1994 Edition. To meet RG 1.37 and NQA-1, the applicant uses procedures to provide cleanliness controls during all phases of manufacture and installation including final flushing. Controls minimize the introduction of harmful contaminants such as chlorides, fluorides, and low melting alloys on the surface of components. The applicant also controls abrasive work such as grinding, polishing, and wire brushing to prevent contamination that could contribute to stress corrosion cracking. In addition, FSAR Tier 2, Section 4.5.1.4 states that cleanliness of the CRDMs is controlled during manufacture and installation in accordance with ASME NQA-1-1994 and RG 1.37. On this basis, the staff finds the applicant's on-site cleaning and cleanliness controls for CRDM components acceptable.

4.5.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

4.5.1.6 *Conclusions*

Except for the above identified open items, the staff finds, for the reasons set forth above, that the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls are acceptable, because they satisfy regulatory requirements or positions described above (for RCPB materials), or because they have been demonstrated to be acceptable based on appropriate materials selections and extensive operating experience (for non-RCPB materials).

Based on the above, the staff concludes that the design of the CRDM structural materials is acceptable and meets the requirements of GDC 1, GDC 14, and GDC 26, as well as 10 CFR 50.55a.

4.5.2 Reactor Internals and Core Support Materials

4.5.2.1 *Introduction*

This section of the FSAR describes the materials, welding controls, non-destructive examination, and the fabrication and processing of austenitic stainless steels used for the reactor internals and core support structure.

The objectives of the staff's review are to confirm that the materials used for the reactor vessel internals and core support structure are acceptable and meet the requirements of 10 CFR Part 50, Appendix A, GDC 1 and 10 CFR 50.55a to ensure that the internal components are designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

4.5.2.2 *Summary of Application*

FSAR Tier 1: There are FSAR Tier 1 requirements that are applicable to reactor internals and core support materials. These are given in Paragraph 3 of FSAR Tier 1, Section 2.2.1, "Mechanical Design Features." These items are addressed in Section 3.9.5 of this report.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in FSAR Tier 2, Section 4.5.2 summarized here, in part, as follows.

The materials proposed by the applicant for the reactor vessel internals and core support structure are described in this section of the FSAR. The FSAR states that, with the exception of ASME SA-479 Type 316 Strain Hardened Level 1, which is included in ASME Code Case N-60-5, all materials are permitted by Section III of the ASME B&PV Code, Subsection NG-2120. ASME Code Case N-60-5 has been accepted by the NRC in RG 1.84. Weld filler metals for the reactor internals and core support structure are also stated in the FSAR as meeting the requirements of GDC 1 and 10 CFR 50.55a. The FSAR also states that the welding controls specified for austenitic stainless steel reactor pressure boundary components that are described in FSAR Tier 2, Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," apply to the reactor internals and core support structures. In addition, the controls provided for unstabilized austenitic stainless steel reactor coolant pressure boundary components in FSAR Tier 2, Section 5.2.3, to minimize the susceptibility of unstabilized austenitic stainless steel to intergranular stress corrosion cracking are also applicable to austenitic stainless steels used for reactor vessel internals and core support structures.

ITAAC: Table 2.2.1-5 of FSAR Tier 1, Section 2.2.1 contains ITAAC that are applicable to reactor vessel internals and core support structures. These items are addressed in Section 3.9.5 of this report.

Technical Specifications: There are no Technical Specifications for this area of review.

4.5.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 4.5.2 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections also can be found in Section 4.5.2 of NUREG-0800.

1. 10 CFR 50.55a which requires that structures, systems, and components (SSCs) shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR Part 50, Appendix A, GDC 1, which requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 1 also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

4.5.2.4 *Technical Evaluation*

The staff reviewed FSAR Tier 2, Section 4.5.2, which describes the materials used in the design of the reactor vessel internals and core support structures. The staff's review was performed using the guidelines in SRP Section 4.5.2, "Reactor Internals and Core Support Structure Materials." The following evaluation addresses the acceptance criteria outlined in SRP Section 4.5.2.

Material Specifications

FSAR Tier 2, Section 4.5.2.1, "Materials Specifications," and FSAR Tier 2, Table 4.5-2, "Reactor Vessel Internal Materials," provide the material specifications for the core support structures and reactor vessel internal components. The core support and reactor vessel internal materials consist of austenitic stainless steels, Grades 304LN and martensitic stainless steels for pins and inserts which are also coated with a cobalt alloy. Strain-hardened Type 316 material is also used for the reactor vessel internals.

To determine the material specification for the cobalt alloy and the use of this alloy on other components, the staff requested additional information in RAI 50, Questions 04.05.02-1 and 04.05.02-3b. In a September 18, 2008, response to RAI 50, Questions 04.05.02-1 and 04.05.02-3b, the applicant proposed to delete Type 304L from the weld material list in FSAR Tier 2, Table 4.5-2. The applicant also stated that the cobalt alloy is Stellite 6 or equivalent, and is used for hardfacing the Radial Key Inserts, upper core guide pins and inserts. The applicant noted that the Stellite 6 filler material specifications are ASME SFA 5.21, Classifications ERCCoCr-A and ERCoCr-A. However, the applicant did not include these filler metal specifications in FSAR Tier 2, Table 4.5-2. The staff determined this to be acceptable, since this material is used in operating reactors to minimize wear, but FSAR Tier 2, Table 4.5-2 should include the filler metal specifications. **RAI 339, Question 04.05.02-9, which is associated with the above request, is being tracked as an open item**.

The applicant also noted that Alloy 600/82/182 is not used in the design of the U.S. EPR reactor vessel core supports or reactor vessel internals. The staff determined this to be acceptable, since Alloy 600/82/182 can degrade components due to primary stress-corrosion cracking based on current operating experience.

To confirm that all of the materials used in the reactor vessel internals are given in FSAR Tier 2, Table 4.5-2, the staff requested that the applicant clarify its statement that, "there are no other materials used in the reactor vessel internals or core support structures that are not otherwise allowed under ASME Code, Section III, NG-2120," including whether Alloy 600/182/82 or Alloy 690/52/152 are used. In a September 18, 2008, response to RAI 50, Question 04.05.02-3a and -3c, the applicant stated that Alloy 690 is used to fabricate the radial key inserts, washers and locking bars; Alloy X750, Type 2 is used to fabricate the springs installed inside the irradiation specimen capsule access plugs; and stainless steel 304N is used to fabricate the heavy reflector slabs. The staff determined this acceptable, since the materials have been used with acceptable results in similar current nuclear applications. The applicable material specifications were included in FSAR Tier 2, Revision 1, Table 4.5-2 and verified by the staff.

In RAI 50, Question 04.05.02-6a, the staff requested information on the materials used for the radial keys, which are welded to the reactor pressure vessel and make up the core support structure. In a November 7, 2008, response to RAI 50, Question 04.05.02-6a, the applicant stated that the core support structure radial keys are fabricated from Alloy 690 and are welded to the Alloy 52/152/52M cladding of the reactor vessel using Alloy 52/152/52M filler metal. The reactor vessel is typically clad with stainless steel, except Alloy 52/152/52M cladding is used for the areas where the radial keys are attached. However, the staff noted that since two different types of cladding will be used, the application of the cladding should be discussed in the FSAR to assure fabrication defects are not introduced. For example, welding the stainless steel cladding on the NiCrFe cladding may introduce welding defects. In addition, since the weld attaching the radial key to the reactor vessel cladding is a structural weld, the effects on the low alloy steel reactor vessel should be discussed, including whether heat treatment of the reactor vessel is needed. Also, since the radial key attachment weld is a structural weld, the weld cladding should be qualified to the requirements of a structural weld and corrosion resistant cladding. However, per the ASME Code, Section III, Subsection NG-1130, the radial key and its attachment weld to the reactor vessel cladding is considered part of the reactor vessel, and further evaluation of this weld and the radial key is addressed in Section 5.3.1 of this report.

Except for the issue identified in the open item tracked in RAI 339, Question 04.05.02-9 above, the staff finds the material specifications for the core support and reactor vessel internals to be acceptable, because they meet the ASME B&PV Code (ASME Code), Section III, Division I, Subsection NG, and Code Case N-60-5, which is approved by RG 1.84.

Controls on Welding

The controls on welding for the core supports and reactor vessel internals appear to be the same as those controls for reactor coolant pressure boundary components provided in FSAR Tier 2, Section 5.2.3. However, in RAI 50, Question 04.05.02-2 the staff requested that the applicant clarify whether these controls apply to all reactor vessel internal components or only core-support structures.

In a September 18, 2008, response to RAI 50, Question 04.05.02-2, the applicant stated that the core supports classified in FSAR Tier 2, Table 3.9.5-1, "Component Classification," are welded in accordance with the ASME Code, Section III, Subsection NG. Therefore, as referenced in the ASME Code, Section III, Subsection NG-4320, the core supports are welded in accordance with ASME Section IX. In addition, the reactor internals are designed and constructed in accordance with Subsection NG-1122(c) of the ASME Code, Section III, to ensure that they do not adversely affect the integrity of the core support structures. The staff has determined this to be acceptable, since the core supports and the reactor internals will be designed and fabricated in accordance with the applicable subsections of the ASME Code, Section III, which specifies, among other things, heat treatment, weld joint geometries, weld procedures, qualification, and specification. In addition, the staff notes that the applicable RGs (i.e., RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," April 1978; RG 1.34, "Control of Electroslag Weld Properties," December 1972; RG 1.37; RG 1.44; and RG 1.71, "Welder Qualification for Areas of Limited Accessibility," May 2007) will also be used for welding of the core supports and the reactor internals to prevent stress corrosion cracking.

FSAR Tier 2, Section 4.5.2.1 specifies that the filler metal specifications for the core support structures and the reactor internals are given in FSAR Tier 2, Table 4.5-2. In a September 18, 2008, response to RAI 50, Question 04.05.02-1b, the applicant clarified that filler metal type 304L will be deleted from FSAR Tier 2, Table 4.5-2, and that the filler metals used are Types 308L, 309L, and 316L per filler metal specifications SFA 5.4, 5.9 and 5.22 of ASME Code, Section II. The staff has determined the use of filler metal Types 308L, 309L, and 316L per filler metal specifications SFA 5.4, 5.9, and 5.22 to be acceptable, since they are low carbon stainless steel weld materials that are more resistant to stress corrosion cracking than high carbon stainless steel weld materials. The staff also verified that FSAR Tier 2, Revision 1, Table 4.5-2 was revised as stated in the response to RAI No. 50, Question 04.05.02-1b, dated September 18, 2008 and is, therefore, acceptable.

Therefore, the staff determined the controls on welding on the core support and reactor vessel internals to be acceptable, because they meet the ASME Code, Sections III and IX, as well as the guidelines in the applicable regulatory guides (i.e., RG 1.31, RG 1.34, RG 1.37, RG 1.44, and RG 1.71) as discussed in Section 5.2.3 of this report.

Nondestructive Examination

FSAR Tier 2, Section 4.5.2.3, “Nondestructive Examination,” states that the base materials of the reactor vessel internals and core support structures will be inspected in accordance with ASME Code, Section III, Division 1, NG-2500, and that the welds will be inspected in accordance with ASME Code, Section III, Division 1, NG-5000. However, it was not clear to the staff which components will be fabricated and inspected to the ASME Code. Based on a September 18, 2008, response to RAI 50, Question 04.05.02-2, the core support structures identified in FSAR Tier 2, Table 3.9.5-1, are inspected to the ASME Code, Section III using the examination methods in Section V of the ASME Code. The staff determined this to be acceptable, since the applicable sections of the ASME Code will be used to perform the inspections of the core supports. The reactor internals are inspected to ensure the integrity of the reactor internals do not adversely affect the integrity of the core support structures. Therefore, the reactor internals are inspected based on the provisions in the ASME Code, Section III, Subsections NG-1100 and NG-5000. The staff determined that inspection of the reactor internals to ensure their integrity provides reasonable assurance that the reactor internals will not fail, thereby preventing the possibility of causing collateral damage to the core support structures.

Therefore, the staff determined the nondestructive examination for the core support and reactor vessel internals to be acceptable, because it meets the ASME Code, Section III, Division 1, Subsection NG, and the staff considers RAI 50, Question 04.05.02-2 resolved.

Fabrication and Processing of Austenitic Stainless Steel

The reactor vessel internals and core support structures are fabricated from low-carbon, austenitic stainless steel which are heat treated in accordance with RG 1.44 to minimize their susceptibility to stress-corrosion cracking. In addition, cold-worked Type 316 stainless steel has a maximum-specified room temperature yield strength of 620 MPa (90 ksi), which meets the guidance in SRP Section 4.5.2.

To determine whether the heat treatment of the radial keys is in accordance with RG 1.44, in RAI 50, Question 04.05.02-6, the staff requested information on how the radial keys are heat treated after they are welded to the reactor pressure vessel. In a November 7, 2008, response to RAI 50, Question 04.05.02-6b, the applicant stated that, in the area of the radial keys, the reactor vessel is clad with weld metal (filler metal specification SFA 5.14, Type 52/52M/152). The reactor vessel is then heat treated in accordance with ASME Section III, Subsection NB-4600. The radial keys are fabricated from a nickel alloy (ASME Code, material specification SB-564, Alloy 690) and welded to the reactor vessel cladding using filler metal specification SFA 5.14 Type 52/52M/152. Therefore, since the radial keys are not stainless steel, and not directly welded to the alloy steel reactor vessel, the guidance in RG 1.44 does not apply to this component, and the nickel alloy will not be sensitized. Further discussion of the radial key and its attachment weld is provided in Section 5.3.1 of this report. The staff has determined that the applicant’s proposal to meet the guidelines in RG 1.44 provides assurance that the methods for control of surface contamination, welding processes, heat treatment, and testing for susceptibility to IGSCC, among other things, will minimize sensitization of the stainless steel core support components and reactor internals. Therefore, the staff considers RAI 50, Question 04.05.02-6 resolved.

In addition, in order to prevent the introduction of contaminants that may promote stress-corrosion cracking, the guidelines of RG 1.37 are typically used for abrasive work on austenitic stainless steels. Therefore, the staff requested confirmation that the guidelines of RG 1.37 are followed for the reactor vessel internals and core support structures. In a September 18, 2009, response to RAI 50, Question 04.05.02-5, the applicant proposed to revise FSAR Tier 2, Section 4.5.1.2.1, "Austenitic Stainless Steel Pressure Boundary Components," to state that the guidelines of RG 1.37 will be used to ensure that abrasive work on austenitic stainless steel does not introduce contaminants that may promote stress-corrosion cracking. Among other things, RG 1.37 recommends that tools used for abrasive work on other material not be used on stainless steels. The staff determined that the applicant's proposed use of the RG 1.37 guidelines is acceptable to prevent stress-corrosion cracking. The staff also confirmed that Revision 1 to the FSAR was revised as stated by the applicant in the response to RAI No. 50, Question 04.05.02-5, dated September 18, 2008, and therefore is acceptable.

Except for the aforementioned confirmatory item, the staff finds the fabrication of the austenitic stainless steel core support and reactor vessel internals to be acceptable, because the applicant will use the guidelines in RG 1.44 and RG 1.37 to prevent material degradation, such as stress-corrosion cracking of stainless steel components.

Other Material Considerations

Other degradation mechanisms that are currently challenging the integrity of reactor vessel internals and core support structures include irradiation-assisted stress-corrosion cracking (IASCC) and void swelling. To assess how the design and use of the materials used in the FSAR will minimize the effect of these degradation mechanisms, in RAI 50, Questions 04.05.02-4a, 04.05.02-6c, and 04.05.02-7, the staff requested additional information concerning the design of the heavy reflector, core barrel, and radial key inserts, and whether these components would be susceptible to IASCC and void swelling which may cause these components to degrade such that they might not satisfactorily perform their intended safety function as required by GDC 1 to Appendix A of 10 CFR Part 50. In a November 7, 2008, response to RAI 50, Questions 04.05.02-4a, 04.05.02-6c, and 04.05.02-7, the applicant stated that it plans to use the criteria of Electric Power Research Institute MRP-175, "Materials Reliability Program [MRP]: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," and the latest findings of the Electric Power Research Institute (EPRI)/MRP reactor internals program to screen the heavy reflector, radial key inserts, and the intermediate core barrel shell to lower core barrel shell weld for IASCC and void swelling. The applicant also provided estimated peak 60 effective full power years neutron fluences of 8.56×10^{22} n/cm^2 (5.52×10^{23} n/in^2) (E>1.0 MeV), 7.65×10^{20} n/cm^2 (4.94×10^{21} n/in^2) (E>1.0 MeV) and 4.21×10^{21} n/cm^2 (2.716×10^{22} n/in^2) (E>1.0 MeV) for the re-entrant corners of the heavy reflector, the radial key inserts and the intermediate core barrel shell to lower core barrel shell weld, respectively.

The applicant stated that upon screening, if required, an augmented ASME Code, Section XI inspection program for these components would be developed to verify IASCC and void swelling do not impact the safety function of these components. The industry is addressing these ongoing issues through the EPRI MRP program. However, to adequately demonstrate that the susceptibility of these reactor internal components to these degradation mechanisms does not impact their safety function or other core support structures, the reactor internal components should be analyzed using the guidance in the material reliability program (i.e., MRP-175 and other appropriate documents). This analysis should be provided to the staff for review, or a COL action item should be included in FSAR Tier 2, Section 4.5.2 for the COL applicant to address this analysis. **RAI 339, Question 04.05.02-10, which is associated with the above request, is being tracked as an open item.**

In addition, columns and the upper-support plate use bolting that may by subjected to stress relaxation or loss of preload. Therefore, the staff requested that the applicant discuss the effects of stress relaxation or loss of preload on the bolting, and how it could impact the safety function of the corresponding components. In a September 18, 2008, response to RAI 50, Question 04.05.02-8, the applicant stated that stress analyses of bolted joints will take into account preload relaxation (loss of preload). Therefore, additional margin will be added to the preloads to account for the loss of preload degradation mechanism. The staff determined this to be acceptable, since preloads are developed to ensure that even with stress relaxation, sufficient design preload will exist so that the integrity of the bolting continues to perform its safety function (i.e., bolting that restrains the upper core plate to the upper support plate and tie rods that restrain the heavy reflector slabs). Therefore, the staff considers RAI 50, Question 04.05.02-8 resolved.

The U.S. EPR design uses a heavy reflector that replaces the thin baffle plates used in current operating reactors. The heavy reflector increases the neutron efficiency, protects the reactor pressure vessel from radiation-induced embrittlement, improves the long-term mechanical properties of the lower internals, and provides lateral support to maintain the geometry of the core. The heavy reflector consists of stacked slabs that rest on the lower-support plate and contains no welds or bolted connections close to the core. Keys are used to align the slabs and the slabs are axially restrained by tie rods bolted to the lower-core support plate.

Because this is a relatively new design concept, the staff requested, in RAI 50, Question 04.05.02c, additional information on the design, materials, and fabrication processes used to determine how the heavy reflector will maintain its intended safety functions as required by GDC 1 to Appendix A of 10 CFR Part 50. In a September 18, 2009, response to RAI 50, Question 04.05.02c, the applicant discussed the tie rods for the heavy reflector. The tie rods pass through the heavy reflector slabs and are bolted to the lower core support plate. The tie rods are used to axially restrain the heavy reflector thereby maintaining the integrity of the heavy reflector (which is a core support structure). Therefore, the tie rods perform a core support function. In FSAR Tier 2, Revision 1, the applicant reclassified the tie rods in FSAR Tier 2, Table 3.9.5-1, as “core support” in lieu of “reactor internals.” The staff finds this is acceptable.

Since the heavy reflector uses vertical keys and keyways in the forged heavy reflector slabs, the staff requested in RAI 50, Questions 04.05.02b and 04.05.02c, a discussion on the prevention of notches on the vertical keys and keyways that can act as stress concentrations and crack initiation sites, which could lead to the loss of function of the heavy reflector. In a September 18, 2009, response to RAI 50, Questions 04.05.02b and 04.05.02c, the applicant stated that the fabrication and installation details of the heavy reflector are proprietary information. This response does not address the staff’s concern of introducing stress concentrations and crack initiation sites that could affect the function of the heavy reflector. **RAI 339, Question 04.05.02-11, which is associated with the above request, is being tracked as an open item**

Except for the issues identified in the open items identified above, the staff finds that the applicant’s material selection, fabrication, nondestructive examination, and welding practices provide reasonable assurance that the materials used for the reactor vessel internals and core support structures will preclude operational inservice degradation, such as IASCC, stress relaxation, or loss of preload on bolting, and void swelling to ensure the structural integrity is maintained and that their corresponding safety functions are not adversely affected.

4.5.2.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

4.5.2.6 *Conclusions*

Except for the issues identified in as open items and tracked as RAI 339, Questions 04.05.02-9, 04.05.02-10, and 04.05.02-11, the staff finds that the material selection, fabrication practices, examination and testing procedures, and control practices will provide reasonable assurance that the materials used for the reactor internals and core support structures will preclude inservice deterioration and maintain their structural integrity. Therefore, the staff determined that FSAR Tier 2, Section 4.5.2 meets the requirements of Section 50.55a and GDC 1 of Appendix A to 10 CFR Part 50, and Section III of the ASME Code and is, therefore, acceptable.

4.6 Functional Design of Reactivity Control Systems

4.6.1 Introduction

The reactivity control systems for the U.S. EPR are comprised of two independent and diverse technologies that complement each other, but are each capable of unilaterally maintaining the reactor within specified design limits. One system, the CRDS is an electro-mechanical system capable of inserting and withdrawing fixed neutron poison rods (control rods) to control reactivity within the reactor core. The second means of reactivity control is through the use of soluble neutron poison boric acid coolant systems that include the CVCS, EBS, and the safety-injection system (SIS).

4.6.2 Summary of Application

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Sections 2.2.1 and 2.4.13, “Control Rod Drive Control System,” FSAR Tier 1, Sections 2.2.7, “Extra Borating System”; 2.2.3, “Safety Injection System and Residual Heat Removal System”; and 2.2.6, “Chemical and Volume Control System.”

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in FSAR Tier 2, Section 4.6, “Functional Design of Reactivity Control Systems,” summarized here in part, as follows:

In order to meet the requirements of GDC 26, the U.S. EPR design relies upon the CRDS and the three soluble neutron poison boric acid coolant systems.

The safety-related function of the CRDS is to bring the reactor to a subcritical condition by inserting control rods with or without electrical power. The U.S. EPR contains 89 electromagnetic jack designed control rod drive mechanisms, each consisting of a drive rod, pressure housing, latch unit, and coil housing assembly. The CRDMs are vertically mounted on the reactor vessel closure in a uniform quadrant array. The CRDMs are designed to use natural air circulation, convection cooling. Details of the CRDM components and operability are provided in FSAR Tier 2, Section 3.9.4, “Control Rod Drive System,” and a diagram of the

CRDM assembly is shown in FSAR Tier 2, Figure 3.9.4-1, “Control Rod Drive Mechanism Assembly.” The CRDMs are equipped with both a digital and analog position indication system, so the control rod position is measured over the height of the core by two diverse and independent methods.

The SIS is designed to limit fuel assembly damage during core flooding (via accumulators) and emergency core cooling events following a design-basis loss of coolant accident. The SIS pumps borated water from the in-containment refueling water storage tank (IRWST). The SIS is described in FSAR Tier 2, Section 6.3.

The EBS is designed to inject high-pressure borated water into the reactor coolant system to support reactor shutdown. The EBS is described in FSAR Tier 2, Section 6.8, “Extra Borating System.”

The CVCS is designed as a normal operational system used to maintain reactor coolant system boric acid concentrations during plant operations. The CVCS and the CRDS jointly maintain reactivity control during all normal modes of plant operations. The CVCS is described in FSAR Tier 2, Section 9.3.4, “Chemical and Volume Control System (Including Boron Recovery System).”

ITAAC: The ITAAC associated with FSAR Tier 2, Section 4.6 are given in FSAR Tier 1, Sections 2.2.1, 2.2.3, 2.2.6, 2.2.7, and 2.4.13.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 4.6 are given in FSAR Tier 2, Section 16.3.1, “Reactivity Control Systems,” Sections 16.3.1.8, “Boron Dilution Protection (BDP),” Section 16.3.9.1, “Boron Concentration,” and Section 3.5, “Emergency Core Cooling System (ECCS).”

4.6.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in Section 4.6 of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section 4.6 of NUREG-0800. These regulations include:

1. GDC 4, as it relates to the structures, systems, and components important to safety that shall be designed to accommodate the effects of and to be compatible with the environmental conditions during normal plant operation, as well as during postulated accidents.

2. GDC 23, “Protection system failure modes,” as it relates to the protection system failure modes such that the system shall fail into a safe state or into a state demonstrated to be acceptable on some other defined basis.

3. GDC 25, as it relates to the protection system’s capability to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

4. GDC 26, as it relates to the reactivity control system redundancy and capability such that two independent reactivity control systems of different design principles shall be provided and capable of reliably controlling reactivity changes under conditions of normal operation, including anticipated operational occurrences to assure acceptable fuel design limits are not

exceeded. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

5. GDC 27, as it relates to the reactivity control systems' combined capability, in conjunction with poison addition by the emergency core cooling system, to reliably control reactivity changes to assure that under postulated accident conditions the capability to cool the core is maintained.

6. GDC 28, as it relates to reactivity limits such that reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant boundary, nor disturb the core and its support structures to impair significant capability to cool the core.

7. GDC 29, as it relates to the protection system and reactor control systems being designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

Acceptance criteria adequate to meet the above regulatory requirements include:

1. To meet the requirements of GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents.

2. To meet the requirements of GDC 23, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure.

3. To meet the requirements of GDC 25, the design of the protection system should assure that a single malfunction of the CRDS will not result in exceeding acceptable fuel design limits.

4. To meet the requirements of GDC 26, the CRDS should be capable of providing sufficient operational control and reliability during reactivity changes during normal operation and anticipated operational occurrences.

5. To meet the requirements of GDC 27, the combined capability of CRDS and emergency ECCS should reliably control reactivity changes to assure the capability to cool the core under accident conditions.

6. To meet the requirements of GDC 28, the CRDS should be designed to assure that reactivity accidents do not result in damage to the reactor coolant pressure boundary, or result in sufficient damage to the core or support structures so as to significantly impair coolability.

7. The CRDS should be designed to ensure an extremely high probability of functioning during AOOs to be in conformance with GDC 29.

4.6.4 Technical Evaluation

The staff reviewed FSAR Tier 2, Section 4.6, “Functional Design of Reactivity Control Systems,” in accordance with Section 4.6 of the NUREG-0800.

The functional performance of the CRDS was reviewed to confirm that the system can provide a safe-shutdown response within acceptable limits during AOOs and prevent or mitigate the consequences of postulated accidents. The review covered the CRDS to ensure conformance with the requirements of GDC 4, GDC 23, GDC 25, GDC 26, GDC 27, GDC 28, and GDC 29.

Acceptance criteria adequate to meet the above regulatory requirements include:

Information for Control Rod Drive System

In FSAR Tier 2, Section 4.6.1, “Information for Control Rod Drive System,” information on the control rod drive system is given with several references to drawings and information in other parts of the FSAR. Natural convection cooling maintains the temperature of the control rod drive mechanisms below design operating temperature. The CRDM pressure housing is designed and tested in accordance with Section III of the ASME B&PV Code as presented in FSAR Tier 1, Section 2.2.1.

FSAR Tier 1, Section 2.2.1 also describes the design of the CRDM position and temperature sensors. FSAR Tier 1, Section 2.4.13 contains the system description, equipment design classifications, and ITAAC information deemed appropriate and necessary to reflect the information in FSAR Tier 2, Section 4.6. Stress limits and allowable deformations, the design temperature of 351 °C (664 °F), and an operating temperature of 250 °C (482 °F) are provided in FSAR Tier 2, Section 3.9.4.3, “Design Loads, Stress Limits, and Allowable Deformations,” on design loads.

The CRDM equipment is stated to be designed and qualified to operate in the reactor vessel cavity environment in FSAR Tier 2, Sections 4.6.1 and 3.9.4.3.

Details of the CRDS are given in FSAR Tier 2, Section 3.9.4, and the CRDM assembly is presented in FSAR Tier 2, Figure 3.9.4-1. Typical CRDM penetrations in the reactor pressure vessel closure head are shown in FSAR Tier 2, Figure 3.9.5-1, “Reactor Pressure Vessel General Arrangement,” and the RCCA pattern is illustrated in FSAR Tier 2, Figure 4.3-34, “Rod Cluster Control Assembly Pattern.” The RCCAs are described in FSAR Tier 2, Section 4.2. The CRDS as presented in the FSAR is based on similar proven technology and systems in use at many other PWR nuclear power plants. Additional important features of the CRDMs are contained in other sections of the FSAR, such as FSAR Tier 2, Section 4.2, concerning the design lifetime of 60 years for the CRDM pressure boundary portions, easy replacement of components as necessary, and thermal hydraulic analysis (FSAR Tier 2, Section 4.4) of the fuel assemblies with spacer grids that incorporate flow mixing nozzles (FSAR Tier 2, Figure 4.2-5, “HTP Spacer Grid Characteristics”).

The instrumentation and control systems have been described in FSAR Tier 2, Section 7.7, “Control Systems Not Required for Safety,” concerning the non-safety-related systems and the safety-related systems in FSAR Tier 2, Section 7.2. The RCCA position is measured with both a non-safety-related digital measurement system and a safety-related analog measurement system. These analog and digital control rod position measurement systems incorporate diverse and independent designs. The design of the two control rod position measurement instrumentation is acceptable with respect to the staff's review of this section because it provides two independent methods that can be used to indicate rod position. However, the staff has determined that more information is necessary to determine how the methods described in topical report ANP-10287P will be implemented and verified. In RAI 367, Question 04.06-14, the staff requested that the applicant provide an explanation on how these methods will be implemented and verified. **RAI 367, Question 04.06-14, which is associated with the above request, is being tracked as an open item.** The adequacy of this instrumentation to perform its safety function is evaluated in Sections 7.2 and 7.7 of this report.

The protection of control rod drive mechanisms from internally generated missiles is provided by concrete secondary shield walls and reinforced concrete missile shield slabs as stated in FSAR Tier 2, Section 4.6.1. The design and evaluation of the internally generated missile barriers for the CRDMs is performed in accordance with the procedures presented in FSAR Tier 2, Section 3.5.3, "Barrier Design Procedures." The design of the barriers to protect the control rod drive mechanisms from internally generated missiles is acceptable in regard to the staff's evaluation of this section. However, the adequacy of these barriers to perform their safety function is evaluated in Section 3.5.3 of this report.

The seismic protection of the CRDS is discussed in FSAR Tier 2, Section 4.6.1, which refers to FSAR Tier 2, Section 5.4.14, "Component Supports," on component supports. These supports restrict displacements during seismic events and design-basis accidents to within their design operable range of the CRDMs. The natural-based seismic events are part of the seismic analysis of the plant. However, the large missiles, such as those that could be generated by a tornado, can also cause seismic-like events inside the containment, where the direction of the force is different (from above and the side) than in the natural events (from below and the side). The staff's evaluation of seismic protection of the CRDS is described in Section 3.9.4 of this report.

ITAAC information included in FSAR Tier 1, Sections 2.2.1 and 2.14.13 was reviewed and the staff concluded that the description was accurate and of sufficient detail.

Evaluation of the Control Rod Drive System

FSAR Tier 2, Section 4.6.2, "Evaluation of the Control Rod Drive System," states that the safety-related function of the CRDS is to perform a rod drop (scram) and put the reactor into a sub-critical condition. In the rod drop, the RCCAs are inserted into the core by gravity.

The time for this insertion is explicitly given in order to be sure that the times used in the safety analysis, such as the 3.5 seconds shown in FSAR Tier 2, Figure 15.0-1, "RCCA Position as a Function of Time to Reach for Full Insertion," or in FSAR Tier 2, Table 15.4-9, "Dropped RCCA − Sequence of Events," will be conservative or best estimate (depending on the purpose of the analysis). Such an explicit maximum drop time for each RCCA is found in the FSAR Tier 2, Section 16.3.1.4, RCCA Group Alignment Limits, Surveillance Requirement 3.1.4.1, which requires verification of the maximum drop time for each RCCA after the reactor vessel head has been removed and before criticality. Since these technical specifications address all the assumptions in the Transient and Accident Analyses, FSAR Tier 2, Chapter 15, pertaining to Control Rods, the staff did not identify any additional necessary TS.

Failure of the CRDS in a safe condition to meet GDC 23 is fulfilled considering the failure in power supply, since the rods will then drop into the core by gravity. The CRDS is, according to FSAR Tier 2, Section 4.6.2, included in the environmental qualification (EQ) program and, thus, designed to operate in harsh environmental conditions (design pressure 17.48 MPa (2,535 psig) and design temperature 351 °C (664 °F) as given in FSAR Tier 2, Section 3.9.4.3 and, therefore, meet the criteria of GDC 4.

FSAR Tier 2, Section 3.1, "Compliance with Nuclear Regulatory Commission General Design Criteria," covers the protection of the CRDMs in the event of high and moderate energy pipe failures. FSAR Tier 2, Section 3.6.2.1.3, "Types of Breaks and Leakage Cracks," states that incore instrument and reactor vent lines do not represent a credible failure mode, and conforms to branch technical paper (BTP) 3-4 for lines less than one inch in diameter. Another failure mode consideration that is not credible is the CRDM pressure housing failure, which could potentially result in a control rod ejection event. The justification for this exclusion is discussed in FSAR Tier 2, Sections 3.5.1.2.2, "Non-Credible Internally Generated Missile Sources Inside Containment," and FSAR Tier 2, Section 15.4.8.

ANP-10281P, Revision 0, "U.S. EPR Digital Protection System Topical Report," March 2007, describes the design of the U.S. EPR protection system. The PS is a digital, integrated reactor protection system (RPS) and engineered safety features actuation system (ESFAS), and it is implemented using TELEPERM XS (TXS) technology described in EMF-2110(NP)(A), Revision 1, "TELEPERM XS: A Digital Reactor Protection System," July 2000. The TXS technology itself, as pointed out in the report, is applied at several power plants. The defense-in-depth, functional diversity, priority and redundancy of the control rod drive system are discussed in FSAR Tier 2, Section 7.1. According to FSAR Tier 2, Section 7.1.1.5.1, "Control Rod Drive Control System," the CRDCS is primarily classified as non-safety-related. The trip contactors are safety-related. The safety-related reactor trip system is described in FSAR Tier 2, Section 7.2 and the non-safety-related parts in FSAR Tier 2, Section 7.7.2.1.1, "Principles of RCCA Control." Principles of RCCA control and the rod drop limitation are presented in FSAR Tier 2, Section 7.7.2.3.6, "Rod Drop Limitation."

According to FSAR Tier 2, Section 4.6.2, the physical separation of the protection system is ensured by the physical separation into the four Safeguard Buildings. Since the four divisionally separated rooms containing the protection system equipment are in different fire zones, the consequences of internal hazards, such as fire, would impact only one protection system division.

Compliance with the single failure criterion of the reactor trip system is discussed in FSAR Tier 2, Section 7.2.2.3.1, "Compliance to the Single Failure Criterion (Clause 5.1 of IEEE Std 603-1998)," and in the FSAR Tier 2, Table 7.2-2, "FMEA Summary for Reactor Trip," which conclude that the protection system maintains the ability to perform the reactor trip function in the presence of any credible single failure of an input sensor, functional unit of the protection system, or reactor trip device. FSAR Tier 2, Table 7.2-2 includes as one failure type single failure of sensor-RCCA position measurement. The staff reviewed the information provided by the applicant in the above sections and finds the information to be acceptable as follows. The only instrumentation of the CRDM and supporting systems credited to safely operate is the position indicator coils, which provide input measurements to the PS. Based on the failure mode effects and analysis (FMEA) presented in FSAR Tier 2, Table 7.2-2, "FMEA Summary for Reactor Trip," the staff concludes that the failure of the position indicator coils to operate properly would not prevent the control rods from being inserted into the core or result in inadvertent withdrawal from the core since position input measurements are a passive, not active, component with respect to control of control rod movement. However, whether the

CRDS meets the acceptance criteria of GDC 4 and GDC 23, relative to instrumentation and control reactor protection systems is evaluated in Chapter 7 of this report.

FSAR Tier 2, Section 4.6.1 states that the CRDM equipment is designed and qualified to operate in the reactor vessel cavity environment. The staff requested additional information to obtain more information about the qualification of the CRDM in the reactor cavity. In an October 28, 2008, response to RAI 95, Question 03.09.04-1a, the applicant documented how the CRDMs will be qualified to operate in the reactor vessel cavity environment. However, the staff did not identify a reference in FSAR Tier 2, Section 4.6.1 to the FSAR section that provided the necessary information to demonstrate satisfaction of the GDC 4 requirement. The staff requested that the applicant include a statement in FSAR Tier 2, Section 4.6.1 that connects GDC 4 requirement to the discussion in other FSAR sections. Since ASME Code requirements do not apply to the non-pressure boundary components of the CRDM, the staff asked what effect the failure of non-pressure boundary components of the CRDM would have on the pressure boundary components of the CRDM. In a March 9, 2009, response to RAI 134, Question 04.06-3, the applicant referred to SRP Section 4.6 which states: "To meet the requirements of GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents."
FSAR Tier 2, Section 4.6.2 states that the control rod drive system is part of the environmental qualification program as described in FSAR Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." In addition, the staff confirmed that FSAR Tier 2, Section 4.6 was revised to include references to other FSAR Tier 2 sections that also demonstrate compliance with GDC 4. Based on the preceding discussion, the staff concludes the responses are acceptable and finds RAI 95, Question 03.09.04-1a and RAI 134, Question 04.06-3 resolved. Additional evaluation of the EQ program, which includes the CRDMs, is provided in Section 3.11 of this report.

In an August 22, 2008, response to RAI 15, Question 04.05.01-5, the applicant states that the U.S. EPR control rod drive mechanism design is a proven German design used in Kraftwerk Union pressurized water reactors. The only known mechanical failure involved the compression spring in the lifting armature, which has no impact on the release function of the latch unit for rod drop for the CRDM design. As shown in FSAR Tier 2, Table 3.2.2-1, "Classification Summary," the CRDM pressure boundary components are Seismic Category I. The impact of the failure of non-safety-related equipment or structures not designated as Seismic Category I Criteria structures, systems, and components on safety-related, Seismic Category I components is reflected in the classification summary in FSAR Tier 2, Table 3.2.2-1. Based on the CRDM operating experience and FSAR Tier 2, Table 3.2.2-1, the staff concludes that failure of the non-safety-related, non-seismic portions of the CRDMs will not prevent or degrade the safety function of any safety-related Seismic Category I component. For additional discussion on this subject, the staff's review of the system quality group classification is evaluated in Section 3.2.2 of this report.

The staff issued a question to obtain more information about the CRDM environmental qualification program. In a December 15, 2008, response to RAI 134, Question 04.06-6, the applicant indicated that according to FSAR Tier 2, Section 4.6.2, the control rod drive system is subject to the environmental qualification program, and thus designed to operate in a harsh environment. However, the staff requested that the applicant verify that the maximum design pressure and temperature of the CRDS is not exceeded in the safety analyses where the CRDS is assumed to remain operable. As described in FSAR Tier 2, Section 3.11, the environmental qualification program verifies that the control rod drive system is designed to be operable under the maximum design pressure and temperature environment (see FSAR Tier 2, Appendix 3D, "Methodology for Qualifying Safety-Related Electrical and Mechanical Equipment"). FSAR Tier 2, Table 3.11-1, "List of Environmentally Qualified Electrical/I&C Equipment," identifies the components of the CRDS that are environmentally qualified. The staff evaluation of the environmental qualification program is in Section 3.11 of this report.

Testing and Verification of the Control Rod Drive System

In FSAR Tier 2, Section 4.6.3, "Testing and Verification of the Control Rod Drive System," reference is made to the CRDS operability assurance program described in FSAR Tier 2, Section 3.9.4.4, "CRDS Operability Assurance Program." This section states that the ability of the pressure housing components to perform their function throughout their operating life of 60 years is confirmed by the primary stress analysis report. The stress analysis report is referenced in FSAR Tier 1, Section 2.2.1.

FSAR Tier 2, Section 3.9.4.4 states that to confirm the mechanical adequacy of the CRDS, a prototype testing program was created. According to FSAR Tier 2, Section 3.9.4.4, this prototype testing program comprises performance, stability and endurance tests. The performance tests verified the performance of the equipment under a broad range of conditions.

The staff requested additional information from the applicant on the CRDS prototype testing program. The staff noted that FSAR Tier 2, Section 4.6.3 refers to FSAR Tier 2, Section 3.9.4.4 where it states that to confirm the mechanical adequacy of the CRDS, a prototype testing program was created. According to FSAR Tier 2, Section 3.9.4.4, this program comprises performance tests, stability tests, and endurance tests. In an October 28, 2008, response to RAI 95, Question 03.09.04-1b, the applicant described two tests that have been completed for the U.S. EPR design, but did not provide a reference to a report or documentation of the type of tests and test results. In addition, the applicant's response to RAI 95, Question 03.09.04-1b stated that testing is currently underway but does not describe tests that are currently in progress. The staff requested that the applicant provide the NRC with the status and results of the CRDS prototype testing program and the range of environmental conditions that support the FSAR. The staff indicated that this information is necessary to evaluate the performance of the CRDS to ensure an extremely high probability of accomplishing its safety functions in accordance with GDC 29.

In a March 9, 2009, response to RAI 134, Question 04.06-9, in relation to the environmental qualification program, the applicant stated that the testing described above related to performance, stability, and endurance testing, and was related to the environmental conditions for the control rod drive system. The applicant indicated further that testing has been completed and that the test results are available for NRC inspection. The staff finds that the information provided by the applicant is acceptable and, therefore, considers RAI 134, Question 04.06-9 resolved.

The initial plant test program for the CRDS is described in FSAR Tier 2, Section 14.2 and is in accordance with the guidance provided in RG 1.68 Revision 3, as discussed in Section 14.2 of this report. The tests for the CRDS include performance evaluations of pre-core control system capability (Test No. 036); power system capability (Test No. 112); CRDM (Test No. 169); post-core CRDM (Test No. 184); and at power rod drop times (Test No. 222). In addition, the CRDM pressure housing is part of the initial ASME code hydrostatic test and non-destructive examination of the reactor coolant system. The staff evaluation of the above tests is in Section 14.2 of this report.

No COL information items were found in FSAR Tier 2, Section 4.6, given in FSAR Tier 2, Table 1.8-2, or in FSAR Tier 1 information related to the CRDMs, CRDS or the CRDCS. The staff's review did not identify the need for any COL information items.

Information for Combined Performance of Reactivity Systems

In FSAR Tier 2, Section 4.6.4, "Information for Combined Performance of Reactivity Systems," a discussion is presented on the U.S. EPR design of the two independent reactor control systems: The control rods and the soluble boron in the coolant from the CVCS, SIS, or EBS. FSAR Tier 2, Section 4.6.4 states that in the safety analyses in FSAR Tier 2, Chapter 15, except for the large break loss of coolant accident, no credit is taken for reactivity control systems other than reactor trip to mitigate the events to achieve a stable plant condition. The staff notes that in FSAR Tier 2, Section 15.1.5 appears to indicate that boron addition via the SIS is credited to mitigate large steam line breaks from hot zero power conditions. The staff issued RAI 366, Question 04.06-13 to address this apparent discrepancy. **RAI 366, Question 04.06-13, which is associated with the above request, is being tracked as an open item.**

FSAR Tier 2, Table 4.3.6 indicates that the minimum required SDM is 3,000 percent millirho (pcm). The calculated shutdown margins at BOL and EOL are 6,305 pcm and 6,480 pcm, respectively. Therefore, the excess SDM, the difference between the available SDM and the required SDM, at BOL and EOL are 3305 pcm and 3480 pcm, respectively. Thus, there is over 100 percent shutdown margin, as indicated in FSAR Tier 2, Section 4.6.4. When the most reactive RCCA is excluded, the SDM is still adequate, but the excess SDM above the required SDMs at BOL and EOL are reduced to 880 pcm and 1,724 pcm, respectively, based on the values given in FSAR Tier 2, Table 4.3-6.

The second independent reactivity control system (boron addition) consisting of the SIS, EBS and the CVCS is described in FSAR Tier 2, Sections 6.3, 6.8, and 9.3.4. These systems are briefly described in FSAR Tier 2, Section 4.6.4 without information regarding the boron content in the water. According to FSAR Tier 2, Table 6.3-1, "Accumulators Design and Operating Parameters," the maximum boron concentration is 1,900 ppm and the minimum boron concentration is 1,700 ppm for the SIS. The EBS injects borated water into the reactor coolant system to maintain the core subcritical for safe-shutdown. According to FSAR Tier 2, Table 6.8-1, "Extra Borating System Design and Operating Parameters," the boron concentration ranges from 7,000 ppm to 7,300 ppm.

The staff noted that the FSAR Tier 2, Section 4.6.4 states that the U.S. EPR contains two independent reactivity control systems in accordance with GDC 26: The control rods and the soluble boron in the coolant from the CVCS, SIS or EBS systems. SRP Section 4.6 under GDC 26 requirements refers to system redundancy and capability. SRP and FSAR Tier 2, Sections 4.6 refer to GDC 27 as it relates to the combined capability of the control rod system and the ECCS to reliably control reactivity changes to assure that under postulated accident conditions the capability to cool the core is maintained. A short summary of the SIS, EBS, and CVCS systems is provided in FSAR Tier 2, Section 4.6.4. In FSAR Tier 2, Section 4.6.5, "Evaluation of Combined Performance," the combined performance of the reactivity control systems is discussed. The applicant concludes that these analyses, as related to FSAR Tier 2, Chapter 15, demonstrates that the control rod drive system along with the combined systems, SIS and EBS, reliably control reactivity changes to cool the core under postulated accidents in accordance with GDC 27. Based on staff evaluation in Chapter 15 of this report, the staff finds that the combine reactivity control systems capability satisfies the criteria of GDC 27.

The staff requested clarification regarding the combined system performance and how the three independent systems (SIS, EBS, and CVCS) contribute to the redundancy of the reactivity control system. In a November 15, 2008, response to RAI 117, Question 04.06-2, the applicant revised U.S. EPR FSAR, Tier 2, Section 4.6.4 to clarify the combined system performance of the SIS, EBS, and CVCS, and to clarify how these systems contribute to the redundancy of the reactivity control system. The FSAR clarification provides the necessary information to support the staff's questions regarding compliance with GDC 26, which requires two independent reactivity control systems of different design principles. One system, the CRDS, satisfies the criterion that reactivity changes are controlled under normal and AOOs such that the fuel design limits are not exceeded. The CRDS also holds the reactor core subcritical under cold conditions. The other redundant system, the combination of CVCS, EBS, and SIS, relies upon injection of borated water into the reactor vessel to control the rate of reactivity changes of planned normal power changes such that fuel design limits are not exceeded. Therefore, the staff finds that RAI 117, Question 04.06-2 sufficiently addressed the two redundant systems and considers the RAI resolved. The staff evaluation of these systems is in Chapter 15 of this report, along with the description of SIS, EBS, and CVCS technical specifications addressed in FSAR Tier 2 Chapter 16 and evaluated in the Chapter 16 of this report.

As presented in FSAR Tier 2, Section 6.6.4, "Inspection Intervals," under normal operation and anticipated operational occurrences, control rods compensate for reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the control rod system provides a minimum shutdown margin during AOOs, and is capable of making the core subcritical to prevent exceeding acceptable fuel design limits, assuming that the highest worth control rod is in the fully withdrawn position. Soluble boron in the reactor coolant compensates for xenon burnout reactivity changes and maintains the core reactivity within the shutdown margin for the cold shutdown condition. The CVCS is described in FSAR Tier 2, Section 9.3.4 and is an operational system used to maintain RCS boron concentration during normal plant operating modes. The CVCS and CRDS work together to control reactivity during normal plant operations. As addressed in FSAR Tier 2, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant," during normal operation, administrative controls preclude dilution events through procedures that limit the rate and duration of dilution. In addition, the CVCS is designed to limit the rate of boron dilution, provide alarms, and perform certain protective actions (refer to FSAR Tier 2, Section 9.3.4) to mitigate an inadvertent dilution event. FSAR Tier 2, Section 15.4.6.2, "Method of Analysis," addresses analyses that show anti-dilution, safety-related protection channels provide effective protection by automatically eliminating the dilution source.

FSAR Tier 2, Section 4.6.4, states that that mechanical overheating of the CRDM causes failure of only one RCCA from inserting into the core by gravity, and the other CRDMs remain functional. The staff requested additional information concerning the potential overheating of the CRDMs. The staff requested that the applicant respond to two concerns. The first question was concerning the rationale for the above statement. The CRDM system is designed such that each CRDM is independent of the other CRDMs in regard to electrical and mechanical failure modes of individual CRDM components. However, under adverse environment conditions, there could be individual component failures of the CRDMs that may prevent the RCCA from inserting by gravity. The staff requested that the applicant identify the types of adverse conditions analyzed and the CRDM fail safe response results that satisfy the requirements of GDC 23.

In a March 9, 2009, response to RAI 134, Question 04.06-10, the applicant referred to an October 28, 2008, response to RAI 95, Question 03.09.04-1c, in which the applicant stated that the mechanical operation of each CRDM is independent of the mechanical operation of the other CRDMs. Overheating of the operating coil on an individual CRDM assembly does not prevent the other CRDMs from operating. If overheating of the operating coil occurred and electrical power was lost to the CRDM, the CRDM would fail in a safe condition. Once power is removed from the operating coils, the latches retract from the drive rod, and the RCCA inserts into the core by gravity. Failure to supply power to the operating coils of the CRDMs does not result in a condition that would prevent the rods from inserting into the core.

The applicant also noted the SRP Section 4.6 states: "to meet the requirements of GDC 23, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure." FSAR Tier 2, Section 4.6.2 states: "As described in FSAR Tier 2, Section 3.9.4, the CRDMs fail in an acceptable condition in accordance with GDC 23." Additionally, as noted in the response to RAI 95, Question 03.09.04-1, the applicant stated that the mechanical failure of any individual control rod drive mechanism does not cause failure of the other CRDMs. If a mechanical failure occurs (i.e., latch assembly failure, broken latch), the other CRDMs remain operational. The mechanical operation of each CRDM is independent of the mechanical operation of the other CRDMs. Therefore, the failure of a latch unit would not cause more than one CRDM rod cluster control assembly from dropping into the core. As noted in FSAR Tier 2, Section 15.0.0.3.4, "Rod Cluster Control Assembly Insertion Characteristics," the total negative reactivity inserted following reactor trip excludes the reactivity of the most reactive rod that is assumed to be stuck out of the core. Mechanical failures that could result in a single dropped RCCA or a dropped RCCA bank have been evaluated and the results presented in FSAR Tier 2, Section 15.4.3.1, "Dropped RCCA or RCCA Sub-Bank." The analysis results conclude that the plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage and that the core remains adequately cooled throughout these events. As set forth in Section 15.4.3 of this report, the failure of a CRDM to insert does not affect the capability to safely shutdown the plant in accordance with GDC 23 and GDC 25. Therefore, the staff considers RAI 134, Question 04.06-10 resolved.

Evaluation of Combined Performance

The evaluation of combined performance in FSAR Tier 2, Section 4.6.5 is strongly based on the safety analyses presented in FSAR Tier 2, Section 15.4; the evaluation of the safety-injection system in FSAR Tier 2, Section 6.3.1, "Design Bases"; and the information given on instrumentation and controls in FSAR Tier 2, Chapter 7 and, in particular, the reactor trip system in FSAR Tier 2, Section 7.2.

FSAR Tier 2, Section 4.6.5 states that following a reactivity accident, such as the rod ejection or steam line break, the core can be brought to the shutdown condition, and the core will maintain acceptable heat transfer geometry in accordance with GDC 28. The analyses described in FSAR Tier 2, Chapter 15, and evaluated in Chapter 15 of this report, indicate that the core can be safely shut down following a transient or accident and, hence, the design meets the criteria of GDC 28.

Fulfillment of the SRP Acceptance Criteria

1. To meet the requirements of GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents.

   Sufficient information was found in FSAR Tier 2, Section 4.6 or the other FSAR sections referred to in this report**.**

2. To meet the requirements of GDC 23, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure.

3. To meet the requirements of GDC 25, as it relates to the protection system's capability to ensure that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

   The fulfilment of this criterion is strongly based on the results of the safety analyses presented in FSAR Tier 2, Section 15.4. The requirements are met in all the analyses, as confirmed by the NRC review.

4. To meet the requirements of GDC 26, the CRDS should be capable of providing sufficient operational control and reliability during reactivity changes during normal operation and anticipated operational occurrences.

   The fulfilment of this criterion is strongly based on the results of the safety analyses presented in FSAR Tier 2, Section 15.4. The requirements are met in all the analyses, as confirmed by the NRC review.

5. To meet the requirements of GDC 27, as it relates to the reactivity control systems' combined capability to control reactivity changes to assure the capability to cool the core under accident conditions.

   The fulfilment of this criterion is strongly based on the results of the safety analyses presented in FSAR Tier 2, Section 15.4. The requirements are met in all the analyses, as confirmed by the NRC review.

6. To meet the requirements of GDC 28, the CRDS should be designed to assure that reactivity accidents do not result in damage to the reactor coolant pressure boundary, or result in sufficient damage to the core or support structures so as to significantly impair coolability.

   The fulfilment of these two criteria is strongly based on the results of the safety analyses presented in FSAR Tier 2, Section 15.4 concerning control rod withdrawal, control rod ejection and boron dilution. The requirements are met in all analyses, as confirmed by the NRC review.

7. The CRDS should be designed to ensure an extremely high probability of functioning during anticipated operational occurrences to be in conformance with GDC 29.

   In addition to the PRA results reported in FSAR Tier 2, Chapter 19, "PRA and Severe Accident," the applicant has stated that an extremely high probability of functioning is proven by the CRDM prototype testing program**.**

4.6.5 Combined License Information Items

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this Section.

4.6.6 Conclusions

The staff has reviewed the functional design of the control rod drive system to confirm that the system has the capability to shut down the reactor with appropriate margin during normal operation, anticipated operational occurrences, and accident conditions, including single failures. The CVCS augments the CRDS by supplying boric acid concentrates to the RCS to maintain safe-shutdown. The scope of review included process flow diagrams, layout drawings, piping and instrumentation diagrams, and descriptive information for the systems and for the supporting systems essential for operation of the CRDS system.

Except for the open items identified above, the staff has determined the adequacy of the applicant's proposed design criteria, design basis, and safety classification of the CRDS, and the methods for providing a safe shutdown during normal operation, anticipated operational occurrences, and accident conditions, including single failures. For the reasons set forth above, and with the exception of the open items identified above, the staff concludes that the design of the CRDS is acceptable and meets the requirements of GDC 4, GDC 23, GDC 25, GDC 26, GDC 27, GDC 28, and GDC 29.

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