4.0 REACTOR

This chapter of the United States - Advanced Pressurized-Water Reactor (US-APWR) design certification (DC) safety evaluation report (SER) describes the U.S. Nuclear Regulatory Commission (NRC) staff's review of the mechanical components of the reactor and reactor core for the Mitsubishi Heavy Industries, Ltd. (MHI), hereinafter referred to as the applicant, US-APWR. These mechanical components include the fuel system design (fuel rods and fuel assemblies), the nuclear design, and the thermal-hydraulic design.

4.1 Introduction

Section 4.1 of the US-APWR design control document (DCD) provides an overview of the reactor system. The analytical techniques used in Chapter 4 are summarized and definitions of the key technical terms used throughout Chapter 4 are also identified. The NRC staff's review of the information provided in DCD Section 4.1 is addressed, as appropriate, in the remaining sections of the SER for Chapter 4.

4.2 Fuel System Design

4.2.1 Introduction

The reactor fuel system is designed to provide assurance that:

- The fuel system is not damaged during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
- Fuel system damage is never so severe as to prevent control rod insertion when it is required.
- The number of fuel rod failures is not underestimated for postulated accidents, and coolability is always maintained.

The applicant followed Regulatory Guide (RG) 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants (LWR Edition)," to organize Section 4.2 of the DCD. Section 4.2 was split into five main parts (Sections 4.2.1-4.2.5), the first three of which deal with the design bases, description and design drawings, and the evaluation of the components that comprise the reactor fuel system. The fourth describes the inspection and testing involved throughout the manufacturing and fabrication process, while the fifth contains no information in this section as it is reserved for combined license (COL) information. Section 4.2 of this safety evaluation (SE) will present the applicable information by component, followed by the inspection and testing plan.

4.2.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with fuel system design is provided in Tier 1, Section 2.4.1 of the DCD. The Tier 1 design information submitted by the applicant includes the top-level design features and performance standards that pertain to the safety of the plant and includes descriptive text and supporting figures. The Tier 1 information has been derived from Tier 2.

Tier 1, Table 2.4.1-1, “Equipment Key Attributes,” specifies a total of 257 fuel assemblies, and 69 rod cluster control assemblies (RCCAs). Tier 1, Figure 2.4.1-1, “Reactor General Assembly,” shows the general location of the fuel assemblies, the lower core support plate, and the upper core plate within the US-APWR reactor vessel. Tier 1, Figure 2.4.1-2, “Arrangement of Fuel and Rod Cluster Control Assemblies,” shows the arrangement of fuel assemblies and RCCAs within the core.

DCD Tier 2: The applicant has provided a DCD Tier 2 description in Section 4.2, summarized here as follows:

DCD Section 4.2.1 summarizes the fuel system design bases and explains that more details can be found in Topical Report MUAP-07008-P, “Mitsubishi Fuel Design Criteria and Methodology,” [Reference 1] and Technical Report MUAP-07016-P, “US-APWR Fuel System Design Evaluation” [Reference 2]. The applicant states that the analysis methods presented in these reports have been established and justified for application up to the US-APWR rod design burnup of 62,000 megawatt days per metric ton uranium (MWD/MTU).

Topical Report MUAP-07008-P, “Mitsubishi Fuel Design Criteria and Methodology,” discusses in detail the methodology and material properties used by the applicant in its evaluation of fuel using the Fuel Rod Integrity Evaluation (FINE) computer code. The application of this code is referred to in Technical Report MUAP-07016-P evaluated in this SER. These reports provide a large portion of the information related to a variety of topics with respect to cladding, fuel performance, grid spacer response to various loads, fuel assemblies, and In-Core Control Components (ICCC).

Each fuel bundle (assembly) contains a 17x17 rod array composed nominally of 264 fuel rods, 24 control rod guide thimbles, and an in-core instrumentation guide thimble. To attain a desired radial power distribution, three batches of fuel assemblies contain rods of different enrichment. The central region of the core will consist of the higher enrichment, while the lower enriched assemblies will be placed on the periphery.

The following topics are discussed in more detail in this SE: fuel rod bowing consequences and penalty analysis, fuel and cladding temperature, and potential effects of transients are discussed in Section 4.4, while power coolant mismatch consequences, cladding overheating, energy release and potential effects of fuel burnout, energy released and pressure pulse effects, and fuel rod behavior during a coolant flow blockage event, are discussed in Chapter 15, “Transient and Accident Analyses.”

Fuel Cladding

The US-APWR fuel rod cladding material is ZIRLO™. The applicant states that ZIRLO™ is a zirconium alloy with improved corrosion resistance. A more detailed description of the material can be found in Appendix B of Technical Report MUAP-07008-P. DCD Section 4.2.1.1 includes the acceptance limits for cladding stress, strain, vibration and fatigue.

Fuel Material

DCD Section 4.2.1.2 specifies the melting temperature and density of the uranium dioxide (UO2) and gadolinia-uranium dioxide ((Gd, UO2) fuel material. Fuel densification, swelling, thermal-physical, and chemical properties are discussed in Topical Report MUAP-07008-P.

Fuel Rod

The fuel rod itself uses the material ZIRLO for its cladding, sintered UO2 pellets and/or Gd, UO2 pellets for its fuel, and helium gas to pressurize the rod as its protection against pellet/cladding interaction and collapse during operation. The fuel pellets are "dished" at the top and bottom in order to minimize the negative effects of pellet swelling during operation. The rod also contains both an upper and lower plenum to allow for fission product gases to collect within the rod itself, when they escape the pellet. A spring in the upper plenum holds the pellets in place during shipping and operation, while a stainless steel spacer adds stability in the lower plenum. The cladding thickness and gap between the pellets and the cladding is designed to accommodate thermal expansion during normal operation and all AOO's (including those where the pellet expands at a faster rate than the cladding).

Grid Spacers

The top and bottom grid spacers are fabricated from Inconel 718, while the nine intermediate grid spacers are fabricated from Zircaloy-4. The intermediate spacers also have mixing vanes to allow the coolant to be adequately mixed after passing through the spacers. DCD Section 4.2.1.4 describes the acceptance limits for specific mechanical properties in order to address the requirements with respect to maintaining control rod insertability and core coolability.

The fuel assemblies' response to a combined seismic and loss-of-coolant accident (LOCA) load is in Technical Report MUAP-08007-P, "Evaluation Reports of US-APWR Fuel System Structural Response to Seismic and LOCA Loads," which concludes that while the grid spacers will experience deformation, control rod insertability will not be compromised. This was analyzed using the Fuel IN-elastic Deformation under Seismic condition (FINDS) computer code, which is a modified form of the ANSYS modeling code, presented in Topical Report MUAP-07034-P, "FINDS: Mitsubishi PWR Fuel Assemblies," [Reference 3].

Fuel Assembly

The safety of the fuel assembly design was evaluated using the information in Topical Report MUAP-07008-P and Technical Report MUAP-07016-P. All the components that make up the fuel assembly are evaluated to ensure that they satisfactorily meet acceptance criteria and standards set by the NRC and the American Society of Mechanical Engineers (ASME). The top and bottom nozzles, control rod guide thimbles, grid spacers, and associated joints, are subject to acceptance criteria and standards with respect to stress and loads during normal operation, AOOs and postulated accidents. Another area of examination was the stresses and loads on components during certain non-operational conditions like shipping and handling.

The top and bottom nozzles that support and give structure to the fuel assemblies are made almost exclusively from 304 Stainless Steel. Inconel 718 is the material used to create the hold-down spring that prevents liftoff due to coolant hydraulic pressure in the top nozzle as well as the long strips inserted and welded into slits in the bottom nozzle to act as a filter for debris. Control Rod guide thimbles are locked into the top nozzle, using a three stage bulge joint in such a way that allows for the top nozzle to be removed and individual fuel rods to be removed in the case of damage or leaks in a specific rod. Both the top and bottom nozzle support the fuel assembly, guide coolant through the fuel assemblies, and have alignment holes to ensure proper placement in the core.

The final components that make up the structure of each fuel assembly are the control rod guide thimbles, the in-core instrumentation tube, and the grid spacers. The guide thimbles are attached to the top nozzle via an adapter plate that is shaped to promote flow while preventing fuel rods from upward motion in the fuel assembly during the lifetime of the assembly. The unique interlocking mechanism attaching the guide thimble and top nozzle allows for complete removal of the top nozzle as described previously. The guide thimble itself is made of Zircalloy-4. It is tapered and perforated to allow for a "buffer region" to protect the top nozzle from high impacts during a scram. The in-core instrumentation guide tube is also made of Zircalloy-4 and has a uniform diameter from the top nozzle to the bottom nozzle. The guide tube is attached to the top nozzle in a single stage bulge joint that is identical to the one used to connect the control rod guide thimbles. The grid spacers are summarized previously in this section.

In-Core Control Components

ICCCs include the rod cluster control assemblies, burnable absorber assemblies, neutron source assemblies, and the thimble plug assembly. These components are designed to satisfy safety requirements with respect to the thermal-physical properties of the absorber material, the chemical compatibility of the cladding and absorber, the cladding stress-strain limits, and the irradiation behavior of the absorber material. The cladding of all ICCCs must meet ASME Section III limits for stress intensity.

The ICCCs are typical in material and design (aside from overall length) to other MHI designs currently in use in Japan's operating fleet. They include the following components: the RCCA, a spider assembly with a mixture of 80 percent silver, 15 percent indium, and 5 percent cadmium, the burnable absorber rods made of borosilicate glass and stainless steel cladding, the primary and secondary neutron sources, composed of encapsulated californium and antimony-beryllium, respectively, while clad in stainless steel, and thimble plug rods used in place of ICCC to promote balanced core flow through those assemblies that do not require ICCC.

Testing and Inspection Plan

The US-APWR uses a 10 CFR Part 50, Appendix B compliant level of quality assurance throughout its component fabrication. Rod welds, rod leak tightness, and fuel pellet appearance are inspected individually for each fuel rod and pellet. Visual inspections and testing for defects are performed on samples of the fuel rods, fuel assemblies, and ICCC. A robust process control ensures the proper fuel pellets are in their proper place in each fuel assembly. All the tools performing these tests and verifications are calibrated at specific intervals based on authorized standards. The applicant includes a description of the necessary onsite inspections, coolant radiation monitoring (discussed in more detail in DCD Sections 9.3 and 11.5), and in-service surveillance required to ensure the safety of the fuel system. This entire plan has been used successfully in Japan's current operating fleet.

Technical Reports: Two technical reports, MUAP-07016-P, "US-APWR Fuel System Design Evaluation," and MUAP-08007-P [Reference 4], are evaluated as part of the DCD Section 4.2 review.

MUAP-07016-P, "US-APWR Fuel System Design Evaluation"

MUAP-07016-P describes the evaluation of fuel rod, fuel assembly and in-core component performance against the design acceptance criteria. Fuel rod evaluations include rod internal pressure, cladding stresses, cladding strain, rod growth, rod bow, chemical reaction, cladding fatigue and creep collapse. Fuel assembly evaluations performed include fuel assembly growth, assembly bow, grid spacer behavior, fuel clad fretting wear, and guide tube stress and strain. In-core component evaluations include control rodlets, insertable burnable poisons and neutron source rods. Also, included in MUAP-07016-P are four appendices, which provide the nuclear core design associated with a 24-month equilibrium fuel cycle, fuel assembly and ICCC material properties, fuel rod fretting wear methodology, and fuel cladding collapse evaluation.

MUAP-08007-P, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads"

MUAP-08007-P details the applicant's methodology for modeling non-linear vibration for the US-APWR using the FINDS code. The FINDS code is characterized by its ability to account for the non-linearity of the fuel assembly's vibration characteristics and the grid spacer impact behavior. The model uses nonlinear free vibration test data from pluck tests to act as a target nonlinear behavior. The FINDS model is tuned to reproduce the experimental findings within bands of uncertainty. The results of the experimentation are not direct inputs into the model, but used as a verification. The pluck tests are performed at room temperature and the assumption is that if the FINDS model can accurately produce matching results then the model can be scaled to operating temperatures without introducing significant error.

Inspection, Test, Analysis, and Acceptance Criteria (ITAAC): There are no ITAAC for this area of review.

Technical Specifications (TS): The TS associated with DCD, Tier 2, Section 4.2 are given in DCD, Tier 2, Chapter 16, "Technical Specifications," Section 4.2.1, "Fuel Assemblies," and Section 4.2.2, "Rod Cluster Control Assemblies." The related "Core Operating Limits Report (COLR)" is addressed in DCD, Tier 2, Chapter 16, Section 5.6.3.

4.2.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 4.2 of NUREG-0800, "Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Standard Review Plan (SRP), "Fuel System Design" and are summarized below. Review interfaces with other SRP sections can be found in Section 4.2 of NUREG-0800. These regulations include:

1. 10 CFR Part 50, Section 50.46 (10 CFR 50.46) and 10 CFR 50.34 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. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," as it relates to fuel assembly structure response to externally applied forces.

3. 10 CFR Part 50, Appendix A, 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.

4. GDC 27, “Combined Reactivity Control Systems Capability,” as it relates to the reactivity control system being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under post-accident conditions.

5. GDC 35, “Emergency Core Cooling,” as it relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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. SRP 4.2 Acceptance Criteria Section 1.C.

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 4.2 Acceptance Criteria Section 1.A provides a complete list of the damage criteria.

3. In order to meet the requirements of 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be provided for all known fuel rod failure mechanisms (10 CFR 50.34(a)(1) applies to fission product release for new reactors). SRP 4.2 Acceptance Criteria Section 1.B provides a complete list of the fuel failure criteria.

4. In order to meet the requirements of GDC 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. SRP 4.2 Acceptance Criteria Section 1.C provides a complete list of the criteria.

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 DCD:

1. RG 1.126, “An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification,” for acceptability of fuel densification models.

2. RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” for a description of models, correlations, data, and methods for ECCS and LOCA evaluations.

3. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (together with 10 CFR 50.34) for assumptions on fission product release.

The staff's review of the ECCS performance against the requirements of 10 CFR 50.46, 10 CFR 50.34, and 10 CFR Part 100 is provided in US-APWR DC SER Sections 6.3 and 15.6.5.

4.2.4 Technical Evaluation

Section 4.2 of the Mitsubishi US-APWR Design Certification Application discusses the Fuel System Design of the US-APWR. The applicant must ensure that it meets GDCs 10, 27, and 35 of Appendix A to 10 CFR Part 50, summarized below:

- Fuel design limits are not exceeded during normal operation or AOOs.
- Reactivity control systems are designed such that the core is coolable during a postulated accident, with appropriate margin regarding the number of failed or stuck control rods, and with the poison addition from the ECCS.
- Any fuel or cladding damage that affects core cooling is limited, and
- Clad metal-water interaction is limited to negligible amounts.

For this safety evaluation (SE), the staff reviewed the fuel design the applicant proposes to use in the US-APWR, by evaluating the information provided in the DCD, and the two Technical Reports summarized in Section 4.2.2 of this SE. Additional information supporting the fuel system can be found in Topical Reports MUAP-07008-P and MUAP-07034-P. While the staff's SE for MUAP-07008-P is complete, the staff has not yet completed its SE for MUAP-07034-P. The completion of this SE is being tracked as **Open Item 4.2-1**. The acceptability of the FINDS code as a model for fuel assembly response during combined loads presented in MUAP-08007-P is being tracked as **Open Item 4.2-2**.

Fuel Cladding Performance

The fuel cladding that the applicant proposes to use in the US-APWR is ZIRLO™, a Westinghouse product currently in use in the operating fleet. Its composition and material properties are presented in detail in Appendix B of MUAP-07008-P and the applicant has experience using this cladding material in its Japanese reactors as well. MUAP-08007-P provides additional details regarding the fuel cladding performance during seismic events, including the combined loads from seismic and LOCA forces.

Fuel rod failures can be caused by both clad failures and fuel pellet effects on cladding material. Cladding failure mechanisms include mechanical fracture caused by excessive stress and strain, hydriding, and ballooning/bursting. Fuel pellet induced failure mechanisms include overheating, excessive enthalpy, and pellet clad interaction (PCI) or pellet/cladding mechanical interaction (PCMI).

During its review of DCD Section 4.2, in Request for Additional Information (RAI) 129-1673, Question 04.02-13, the staff asked the applicant if the fuel cladding one-percent strain limit includes the effect of the maximum oxide thickness and/or maximum hydrogen pickup, since oxidation and hydrogen pickup reduce clad ductility especially at high fuel duties over an extended time. In document UAP-HF-09024, dated January 30, 2009, the applicant stated that

the cladding strain limit accounts for the degradation of cladding ductility due to hydrogen pickup during irradiation. The applicant explains that in MUAP-07008-P, the hydrogen limit is established based on irradiated and un-irradiated mechanical test data, and Appendix B of the same document gives the high-temperature tensile test results that show that the uniform elongation exceeds the one-percent criterion, without failure, for hydrogen concentrations up to the established limit. To compensate for the fact that the data apply to the inner diameter of the cladding, the hydrogen content limit was lowered to ensure adequate strain capability at the outer diameter where the one-percent criterion applies. Since the applicant showed that the strain limit includes the effect of maximum hydrogen pickup, and test data show that the clad is still functional at the hydrogen pickup limit given in MUAP-07008-P, RAI 129-1673, Question 04.02-13 is closed.

In RAI 877-5918, Question 04.02-48, the staff asked for the equation to define the specific wear ratio, S, in equation C.2.3-1 of MUAP-07016-P along with the basis for the coefficients used to define the S term including the dependency on oxide thickness. The applicant responded to Question 04.02-48 in document UAP-HF-12006, dated January 11, 2012, by providing the equation for the specific wear ratio, S, and the coefficients used. The applicant also provided the results of adjusted specific wear ratios from hydraulic tests for the fuel cladding wear. The applicant also provided a discussion of how the equivalent specific wear ratio is calculated for the wear volume calculation when both a growth of oxide layer and wear are occurring concurrently on the surface of cladding. The information presented here is part of the information that was included in the applicant's response and used in the staff's review. This information along with the additional information pertaining to grid spacers and fuel assemblies presented later in this section, resolved RAI 877-5918, Question 04.02-48.

In RAI 129-1673, Question 04.02-15, the staff asked the applicant to provide data that substantiates the 95/95 uncertainty values for the corrosion and hydrogen values provided in Table 3.1-2, "Fuel Performance Model Uncertainty Parameters," of Technical Report MUAP-07016-P. The applicant responded by referencing MUAP-07008-P, Sections 4.3.5, "Corrosion" and 4.3.6, "Hydrogen," which provides a summary of the uncertainty parameters in question. The applicant calculated the 95/95 uncertainty using only [ ] data points, which had the highest oxide thickness values. This is acceptable, as the high oxide thickness values are limiting based on meeting fuel cladding performance criteria. The applicant chose to use a 95/95 uncertainty value which bounds the calculated value using the highest oxide thickness data in Table 3.1-2. As the applicant is applying an uncertainty greater than that supported by the high oxide thickness data, the staff finds oxide thickness uncertainty in Table 3.1-2 acceptable. The hydrogen upper bound was removed from Table 3.1-2 based on past precedent from other staff evaluations of ZIRLO™ that hydrogen pick-up values are best estimate values. Therefore, the staff finds the response to RAI 129-1673, Question 04.02-15 acceptable.

Fuel Material

The fuel material for the US-APWR is a combination of both uranium dioxide pellets and Gd, UO2 pellets. The applicant states in DCD Section 4.2.1.2.1 that the fuel centerline temperature does not exceed the melting temperatures (5072°F/2800°C and 4892°F/2700°C, for uranium dioxide pellets and Gd, UO2 pellets, respectively) and that the nominal density is 97 percent theoretical density (TD). There is extensive operating experience with this fuel material, and the applicant has experience in the fabrication of these fuels for use in the current operating fleet in Japan. The fuel material is discussed in more detail in the Topical Report MUAP-07008-P, and

a more detailed evaluation, in which the NRC staff found the fuel material to be acceptable, can be found in the accompanying SE.

Fuel Rod Performance

The applicant employed the FINE code to model the fuel performance in the US-APWR design. MUAP-07016-P is the technical report that chronicles the application of this code in normal and AOOs. Chapter 4 of MUAP-07008-P describes the FINE code in detail. The staff's review of this topical report is documented in the SE. The following evaluation looks at both the information in the DCD and the information found in MUAP-07016-P. Section 4.2.2 of this SE lists those topics related to fuel rod performance, which are evaluated in more detail in other chapters or sections of this SE.

Fuel cladding stresses were evaluated based on ASME Boiler and Pressure Vessel Code (ASME Code), Section III [Reference 7], except PCMI, which is evaluated based on acceptance criteria for cladding strain and no fuel melting. Stress calculations include differential pressure across the cladding wall, thermal stresses, hydraulic and seismic vibration, fuel rod bowing, grid spacer contact and cladding ovality. The total stress was determined by summing the different contributions. Total stress of the limiting fuel rod for normal operation and AOOs is below the allowable stress intensity. For postulated accidents, cladding mechanical integrity is maintained if the applied stress is less than 90 percent of the irradiated yield stress at operating temperature. Evaluation of the US-APWR fuel rods found that applied clad stresses were less than 90 percent of the yield stress for combined seismic and LOCA loads.

Total clad strain for normal operations and AOOs is less than one percent, minimizing the potential failure due to PCI or PCMI. For power transients associated with AOOs, the strain evaluation statistically takes into account uncertainties in fuel fabrication and fuel performance models.

Cladding collapse is due to the differential pressure across the clad. The potential for cladding collapse is greatest early in life when the pressure differential between the coolant and rod internal pressure is greatest and fuel densification occurs. Cladding collapse is prevented by initially backfilling the rod with helium during manufacturing and limiting pellet densification.

Based on current operating experience, so long as the initial fuel pellet density is 95 percent of theoretical density or greater, any pellet-pellet gaps caused by fuel densification are too small to cause cladding collapse. Also, the helium backfill reduces the differential pressure between the rod internal and the system, which provides additional protection against cladding collapse.

If rod internal pressures are significantly greater than system pressure, clad lift-off can occur. Clad lift-off increases fuel temperatures and increases the chances of fuel rod ballooning during AOOs and postulated accidents. Therefore, rod internal pressure is kept to a value below which clad lift-off occurs.

While reviewing the potential for fuel clad lift-off, the staff asked in RAI 129-1673, Question 04.02-20, how the rod power histories in Section 3.3.2 of MUAP-07016 are used to determine rod internal pressure, and asked the applicant to clarify the methodology. The applicant responded to Question 04.02-20 in UAP-HF-09024 by establishing selection criteria for determining the peak rod internal pressure. The staff agrees that the approach will identify the limiting rod internal pressures as [ ] will be sampled every [ ], which are typical core physics code depletion steps. The applicant further stated

that the equilibrium cycle, fuel rod power histories used to calculate rod internal pressures are given in MUAP-07016-P, Appendix A, and the rod internal pressures are calculated using the FINE code as described in MUAP-07008-P. Detailed discussion regarding the methodology used to select and evaluate the limiting fuel rod internal pressures is given in MUAP-07008-P, RAI Responses 1a and 9b from UAP-HF-08299-P, dated December 19, 2008, and the staff's SE. From the staff's review of limiting rod internal pressure calculations given in MUAP-07008-P, and the corresponding SE, the staff finds that the limiting rod internal pressures will be calculated conservatively and compared with the acceptance criteria. Therefore, RAI 129-1673, Question 04.02-20 Revision 1 is closed.

Grid Spacers

In DCD Section 4.2.3.4, the applicant presents the grid spacers' performance in all observations, evaluations, and tests designed to ensure that core coolability and rod insertability are maintained for all postulated accidents in addition to normal operation and AOOs. These include observations and testing including, but not limited to, observations of irradiated assemblies, impact tests, corrosion and hydrogen absorption calculations, and fatigue tests. These results are documented in MUAP-07016-P and MUAP-08007-P.

Grid Spacers provide sufficient spring contact force to minimize fuel cladding fretting wear as evaluated in References 1 and 2. Based on review of Reference 2 and the supporting full-scale hydraulic assembly testing given in Technical Report MUAP-11017, "Hydraulic Test of the Full Scale-US-APWR Fuel Assembly," [Reference 5], the staff has not concluded that a conservative fretting wear estimate has been determined. **Therefore, fretting wear remains an open item and is being tracked as Open Item 4.2-3**. The vibration of the fuel rod is also limited to preclude fatigue of the grid spring. The fuel rod vibration measurements are documented in MUAP-11017-P but the staff could not conclude that vibration is limited to preclude grid spring fatigue. Therefore, **Open Item 4.2-4** has been created to track closure of this issue. Cladding stress and fatigue are evaluated accounting for cladding wear due to fretting.

The material properties of Inconel 718 upper and lower grid spacers limit oxidation and hydrogen pickup and allow the grids to perform their design functions. The Zircaloy-4 intermediate grid oxidation and hydriding are restricted in accordance with Reference 7, to prevent material degradation, thereby satisfying their design functions.

In reviewing Section 4.4.1 of MUAP-07016-P, the staff observed that the description of the oxide thickness at an assembly burnup of 60 GWd/MTU for grid spacers was based on data from guide tube oxidation thicknesses. The description states that the oxide behavior should be the same for both grid spacers and guide tubes since they are made of the same material. Consequently, RAI 129-1673, Question 04.02-3 was issued, since the staff was concerned that the hydrogen distribution will be impacted by stress which can significantly reduce ductility in the material. Since the spacer grids have different stress distributions than guide tubes, it could cause a different amount of hydrogen pickup and consequently different oxide thicknesses. The staff asked if the assumption of similar behavior has been verified by metallographic examination in terms of hydrogen distribution or through mechanical testing of the spacer grids. The staff also asked if the elongation data provided in Figure B.1.3.6-1, "Mechanical Properties of the Control Rod Guide Thimble," is total or uniform elongation and what the hydrogen levels were for the presented data.

The applicant responded to RAI 129-1673, Question 04.02-3 in UAP-HF-09024, dated January 30, 2009, by stating that examination and testing has not been done to investigate the hydrogen

distribution in the grid spacer. The applicant explains that the spring force of the spacer, which is the cause of the stress distribution, is significantly reduced early in life due to irradiation; Figures 4.4-2, “Grid Spacer Spring Relaxation Rate (US Reactor Data)” and 4.4-3, “Grid Spacer Spring Relaxation Rate (Japanese Reactor Data)” of MUAP-07016 are referenced to show this effect. The second part of the staff’s Question 04.02-3 was the type of elongation data shown in Figure B.1.3.6-1, “Mechanical Properties of the Control Rod Guide Thimble” of MUAP-07016 and the associated hydrogen values. In its response to Question 04.02-3 the applicant provided Figure 4.2-3-1, “Mechanical Properties of the Control Rod Guide Thimble,” which indicates the hydrogen level of some of the data, and Figure 4.2-3-2, “Total and Uniform Elongation of Irradiated Control Rod Guide Thimble,” of the same response, which distinguishes between total and uniform elongation data again for a subset of the data. Review of the uniform elongation data provided in Figure 4.2-3-2 shows low uniform elongations which indicate uniform strain below one percent as guide tube and grids have a fairly uniform distribution of hydrides as compared with fuel rods. The staff only used the uniform elongation data to make its safety determination, as total elongation is not a reliable measure of effective failure strain. With uniform strains greater than one percent, the guide tubes, and by inference, the grids remain ductile enough to maintain assembly integrity. Based on the response, the staff concludes that the grid spacer stress distribution is small enough that it would not significantly contribute to increased hydrogen pickup or redistribution and therefore the comparison with guide tube data is acceptable and RAI 129-1673, Question 04.02-3 is closed.

In RAI 129-1673, Question 04.02-4, the staff asked if the spacer grid impact test results will include the effect of oxide thickness and/or hydrogen pickup. As explained in its response to Question 04.02-3, the applicant stated that the effect of lower ductility on irradiated spacer grids is to be simulated by subjecting the spacer grid material to high levels of hydrogen in the previously discussed mechanical testing. Additionally, the relaxed spring force and oxidation of irradiated spacer grids will also be simulated during testing which is documented in MUAP-08007-P. The staff’s review of the mechanical testing report MUAP-08007-P determined that the oxide thickness and hydrogen pickup simulation during testing was implemented as stated and was appropriate to simulate the limiting mechanical state of an irradiated spacer grid. As the mechanical testing did include the effects of hydrogen pickup and oxide thickness, RAI 129-1673, Question 04.02-4 is closed.

In Appendix A of MUAP-08007-P Revision 2, the applicant discusses the conditions used during the testing which provided the input data for the FINDS code. The applicant subjected grid spacers to impact testing at room temperature and under the following three conditions: (1) as built (or beginning of life (BOL)), (2) relaxed grid springs (or end of life (EOL)), (3) hydrided (mid/end of life with increased yield strength due to irradiation). The applicant concluded that the most conservative input with respect to grid deformation would be the relaxed grid springs and as-built conditions. These conditions result in larger lateral displacements of the fuel assemblies, than the hydrided grid spacers. The hydrided grid spacers resulted in an increased buckling load due to hardening, as expected, resulting in lower grid deformations. The staff agrees that non-hydrided grids, with relaxed springs, yield the largest deformations, which have a direct effect on control rod insertability and coolability. Additionally, the staff review of the FINDS code is ongoing and being tracked as **Open Item 4.2-1**.

In RAI 477-3697, Question 04.02-27, the staff asked for information that would help the staff to understand the improvements made to the FINDS code. In its response in UAP-HF-09563, dated December 28, 2009, the applicant responded by saying that there are no differences between the modeled grid spacer design and the design to be used in the US-APWR. Additionally, more data was collected during the impact testing, which improved the accuracy of

the FINDS code. During impact testing, the applicant found that for two particular grid spacers, under relaxed grid spring conditions, the experimental data yielded larger deformations. The applicant used the FINDS model to predict the deformation for these grid spacers at operating conditions. The applicant then tested two grid spacers with relaxed grid springs at operating conditions (as opposed to room temperature) to confirm the accuracy of the FINDS model. The predicted deformation at operating conditions by the FINDS model sufficiently matched the results of the impact testing at operating conditions. Therefore, the staff found that the FINDS code was able to predict deformations at operating conditions adequately.

As stated above, when performing combined seismic and LOCA response analyses, the applicant decided to use the results from two of the eight tested grid spacers under relaxed spring conditions to verify the results of the FINDS code, as documented in Appendix A of MUAP-08007-P. The applicant found the impact tests of the grid spacers with relaxed springs yielded the largest deformation, so the inputs which described a relaxed grid spring were input into FINDS for the seismic plus LOCA loading analysis. The staff agrees that the applicant predicted the worst grid deformation. However, the staff requested in RAI 948-6383, that the applicant identify the BOL and EOL elastic limit loads used in the FINDS impact model, and the 95th percentile buckling load values used to count the number of buckled grids in the analysis. This is being tracked as **Open Item 4.2-5**.

The load limits are defined in Table 4.2-1(c), "Strength Evaluation for Shipping and Handling Loads," of Technical Report MUAP-07016-P for the top grid spacer joint and the intermediate grid spacer joint. In RAI 877-5918, Question 04.02-47, the staff asked how these load limits were determined, whether they were determined by testing, and if so, to provide a description. The applicant responded to Question 04.02-47 in UAP-HF-12006, dated January 12, 2012, by discussing how the load limit at the joints between the grid spacer and the control rod guide thimble, and the load limit at the joint between the intermediate grid spacer and the control rod guide thimble are determined. The load limit at the joints between the grid spacer and the control rod guide thimble is determined by the minimum value of four components, three of which are based on tensile tests and the other on a calculation described by the applicant. The load limit at the joint between the intermediate grid spacer and the control rod guide thimble is determined by the minimum value among three components, one of which was determined by tensile test, another by the manufacturer's minimum specification, and the other by a calculation described by the applicant. Based on the staff's review of the applicant's response, RAI 877-5918, Question 04.02-47 is closed.

In addition to the information presented in UAP-HF-12006 regarding fuel cladding, the applicant also provided a discussion of typical specific wear ratios results from wear tests of typical Japanese conventional fuel assemblies, and stated that there is no difference between the specific wear ratios measured for Inconel-718 and Zircaloy-4 grid spacers. The staff's review of the data supports the applicant's position that specific wear ratios between Inconel-718 and Ziraloy-4 are not significantly different and hence RAI 877-5918, Question 04.02-48 is closed.

Fuel Assembly

In DCD Sections 4.2.1.4.1 through 4.2.1.5.3, the applicant discusses the fuel assembly structural mechanical properties. The staff evaluated the fuel assembly structure against the following failure modes: (1) stress and strain on structural components such as guide tubes, spacers, bottom and top nozzles, (2) cumulative number of strain fatigue cycles are significantly less than design fatigue lifetimes, (3) fretting wear on structural members should be limited and accounted for in stress/strain analyses, (4) oxidation, hydriding and corrosion products (crud)

should be limited and within specified fuel system component values to maintain acceptable strength and ductility, (5) and dimensional changes such as fuel assembly growth should be limited in order to ensure control rod insertion, maintain adequate separation between fuel rod upper end caps and assembly upper nozzle, thus helping to limit fuel rod bow. These failure modes should also be evaluated accounting for the effects of changes induced by irradiation.

The US-APWR fuel assembly is two feet longer than the traditional Pressurized-Water Reactor (PWR) 12-foot assemblies used in the current operating fleet. This is the most obvious difference between the design and current assemblies. Others include the detachable nozzle described in the summary section, holddown spring mechanism to counteract hydraulic lift forces, and the inclusion of two extra grid spacers to help rod stability. The fuel assembly was described in more detail in MUAP-07008-P and its response to various loads was analyzed in MUAP-07016-P. Additionally, the applicant studied the fuel response to combined loads and presented that information in Reference 4.

For normal operation, AOOs, and postulated accidents, stresses in guide tubes and top/bottom nozzles are less than the acceptance limit based on ASME Code Section III. For Zircaloy-4 components (guide tubes), the allowable stress intensity, Sm, is the minimum of 2/3 the yield stress and 1/3 the ultimate stress, considering operating conditions. For stainless steel components, the allowable stress intensity is defined by ASME Code Section III. Spacer grids are designed to allow control rod insertion following a design-basis earthquake and LOCA. Grid spacer stiffness is obtained from impact tests at a 95 percent confidence level at operating conditions.

The results of the testing and application of the FINDS code found that there would be grid buckling in at least one grid spacer in all 257 fuel assemblies for the maximum combination of safe-shutdown earthquake (SSE) and LOCA loads. In RAI 1 of Reference 3, the staff requested that the applicant perform additional calculations to demonstrate how concurrent application of the SSE and LOCA accelerations would change the maximum combined bending stress in the fuel assemblies due to the deformation caused by the buckling grid spacers. The applicant responded in UAP-HF-11121-P that in the event that the grid spacers buckled and applied a stress to the components of the fuel assembly, the maximum control rod guide tube bending stress increased by less than [ ], which the staff agrees is not significant.

The fatigue design limits for guide tubes and top/bottom nozzles are less than one assuming a safety factor of [ ] on stress amplitude or a safety factor of [ ] on the number of cycles.

In order to calculate the loading during seismic and LOCA loads on the fuel assemblies, the applicant used beam and spring-friction elements. The lateral displacements found in the FINDS analysis are applied directly to the beam elements of the fuel assembly as described in Figure 4-1 of MUAP-08007-P, which allows the stresses of the LOCA and seismic load to be combined rather than the displacements. This method is consistent with RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.” In order to find the vertical loading, the applicant considered the loads to be constant throughout the core, and therefore, one assembly yields the results for all 257. The seismic and LOCA stresses produced by the maximum load are found independently and then combined using the square-root-of-the-sum-of-squares (SRSS) method, which yields a single stress intensity value for evaluation against the ASME code values. This method is consistent with SRP 4.2.

During its review of MUAP-07034-P, the staff noted that the applicant made several changes to the FINDS methodology in order to improve its accuracy in the estimates of maximum stress

and impact forces in the fuel assemblies. The original method of using FINDS was to combine the peak deformations in a row (X) of 17 fuel assemblies and a column (Z) of 17 fuel assemblies. This was conservative since the FINDS analysis typically predicts the maximum deformations and grid impact forces at the first or last fuel assembly in the longest row or column. However, the original analysis method incorrectly (and non-conservatively) performed the SRSS combination of the X and Z horizontal responses at each time step and then used the maximum combined response through the history. This incorrectly assumed that the X and Z seismic spectra were linked in time and that the X and Z LOCA histories were also linked in time. Later in MUAP-07034-P (R2), the applicant agreed that the SRSS combination should be performed on the individual X and Z maxima through time. This coupled with the conservatism of the lateral displacements of the longest row and column resulted in what were estimated to be overly conservative maximum lateral combined displacements. Independent calculations using the longest row method showed that the bending stress in the control rod guide tubes would have exceeded the ASME allowable stress. Therefore, the applicant refined its modeling approach to calculate the X and Z deformations and grid impact forces for each of the 257 fuel assembly locations in the core. This was accomplished by performing multiple FINDS analyses with rows and columns of 9, 13, 15, and 17 fuel assemblies. The maximum displaced shapes and corresponding grid impact forces were recorded for each of the fuel assemblies in the row and column positions (for both seismic and LOCA loads) and the SRSS combinations were performed for each X and Z location.

In addition, the typical analysis method models the response in two separate horizontal directions that are aligned with the rows and columns of the core. In RAI 477-3697, Question 04.02-36, the staff asked about the effect of considering motion and grid-to-grid impacts in the diagonal horizontal directions. The applicant responded to this in UAP-HF-09563, dated December 18, 2009, by stating that its method was conservative because it leads to the largest fuel assembly deflections, impact forces, and grid deformations. The NUPEC full-scale testing of a 7x7 fuel assembly core array investigated diagonal vibration and showed no significant increase in deflection or impact force. In an actual three-dimensional earthquake, fuel assembly motions in off-angle directions are expected to interfere with large deflections in the row or column directions. The applicant made a convincing argument that diagonal motion does not need to be specifically considered and the staff agrees.

The FINDS analyses for the different row and column lengths were also analyzed for two sources of uncertainty to estimate the maximum response for 1) the range of amplitude dependent damping and frequency measured from the pluck tests, and 2) +/-15 percent range in the seismic frequency.

The staff requested information in RAI 04.02-34 regarding uncertainty in FINDS analysis results due to uncertainty in the FINDS input data. The applicant responded to RAI 04.02-34 in UAP-HF-09563, dated December 18, 2009. The applicant listed the sources of potential uncertainty in the input variables (elastic modulus, grid spacer in-elastic response, input acceleration wave, and amplitude-dependent frequency and damping characteristics) and reviewed how they treat them. The staff agrees that its assumptions result in conservatively high vibrating deflection and grid plastic deformation. This RAI is further resolved in MUAP-08007-P, Revision 2 [Reference 4] which introduced a new method for dealing with uncertainty. It includes bounding analyses considering BOL and EOL conditions and upper and lower bound frequency and damping factors to find the load combination and location within the core with the highest stresses and grid deformations. This final method for dealing with modeling uncertainty implemented by the applicant is a more robust method of finding the maximum stress condition in the fuel rods and the control rod guide tubes than other methods

considered previously during this review process. The process used to determine uncertainty is provided by the applicant in Figure 4-1 of MUAP-08007-P.

In RAI 572-4583, Question 04.02-44, the staff requested the applicant to either justify its existing method or change to one of the established methods described in the RG and recalculate the stresses reported in MUAP-08007-P. The applicant described, in UAP-HF-10136, dated May 19, 2010, how its response combination method is consistent with RG 1.92 methods. The applicant also described the consideration of input frequency variation in the final calculations. SRP 4.2, Appendix A Section II.3 calls for a check of model sensitivity to input amplitude and frequency variations, and the applicant provides this check in Appendix F, Section F.5 of MUAP-08007-P, Revision 2 [Reference 4]. By adding input frequency variation to the final calculations, the applicant is adding an additional margin of conservatism to the results. The applicant has also changed its analysis methodology to find the bounding stress results for all combinations of the Lower Bound, Best Estimate, and Upper Bound amplitude dependent frequency and damping factors in the interest of conservatism. The staff finds that these measures are reasonable in that they will help ensure the conservatism of the calculated results.

The staff examined the results in MUAP-08007-P and found that the maximum combined stresses analyzed for the top nozzle, bottom nozzle, fuel cladding, and control rod guide thimbles were within the acceptance limits. The stress acceptance limits defined in Table 2.3.1, "Acceptance Criteria during SSE and LOCA events," of MUAP-08007-P are based on the ASME Code Section III, Appendix F for Level D service loads (normal operation + SSE + LOCA loads). Additional requirements in Table 2.3.1 ensure that the fuel assemblies satisfy the performance requirements of SRP 4.2. Specifically, fuel cladding stress is limited to 90 percent of yield stress to ensure that fuel fragmentation does not occur. Meeting the ASME Code Section III, Appendix F for Level D service ensures control rod guide thimble buckling shall not occur so that control rod insertability is assured. Table 4.4.3-2, "Stress for the Fuel Assembly under SSE (560-100) and LOCA (CLB 8B 102%LF) Events at Most Limiting Conditions (EOL with Uncertainties)," of MUAP-08007-P lists the applied stresses and the stress limits for the various components. The staff checked the stress limit values based on material strength data given in MUAP-07016-P and MUAP-07008-P. It was noted that the control rod guide thimble acceptance limit is approximately equal to the lower-bound value of the ASME allowable stress intensity at temperature (Sm) for primary membrane stresses (Pm) and 1.5 Sm for primary membrane plus bending stress (Pm + Pb). These are conservative stress limits that ensure the control rod guide thimbles will remain elastic throughout the combined SSE and LOCA loads. If these limits are satisfied during combined seismic and LOCA loads, then it also follows that they will remain acceptable at normal operation and during other less challenging AOO's. The applicant's analysis of combined loads from seismic and LOCA events showed that the top and bottom nozzles reached only [ ] and [ ] of their limits respectively, while the fuel cladding came to 68 percent of the applicant's [ ] irradiated yield stress criteria. The control rod guide thimble reached [ ] of the acceptance limit. [

]. Therefore, each of the fuel assembly component stresses are less than their respective acceptance limits.

Section 3.3 of MUAP-08007-P discusses the applicant's methodology for evaluation of accident (LOCA) loads. In a LOCA event caused by failure of a coolant pipe, a sudden decompression wave is generated at the rupture location which propagates around the core barrel vertically and circumferentially. This causes an impact load on fuel assemblies due to core barrel movement resulting from the pressure differential. The detailed explanation of the evaluation of the core's

response during a LOCA event is provided in Technical Report MUAP-09002-P, Revision 2, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping." The staff has not completed its review of this document. Any concerns identified during the staff's review potentially could impact the evaluation documented in MUAP-08007-P. Consequently, this is being tracked as **Open Item 4.2-6**.

The fuel assembly structural analysis documented in MUAP-08007-P, Revision 2, is based on the 560-100 SSE (560 ft/sec shear velocity at 30 m depth, 100 feet to bedrock) because it gave the maximum lateral fuel assembly deflection. When comparing Revisions 1 and 2 of MUAP-08007-P, it was noted that the LOCA and SSE acceleration histories had changed. Section 4 of MUAP-08007-P, Revision 2, references the seismic Technical Report MUAP-10001-P, Revision 1, "Seismic Design Bases of the US-APWR Standard Design Plant," as the source of the SSE input to the FINDS code. The staff reviewed MUAP-10001-P and found that its latest revision (Revision 4) had deleted the soil profiles 560-100 and 560-200. Further investigation determined that MUAP-10001-P is currently undergoing additional revision by the applicant. In RAI 948-6383, Question 04.02-62, the staff requested the applicant to provide a revised structural analysis in MUAP-08007-P, which utilizes the appropriate soil profiles. This is being tracked as **Open Item 4.2-7**.

Fuel assembly structural dimensional changes are limited to prevent compressive loads which could excessively distort or bend the assembly. The US-APWR fuel assembly has been designed to reduce the possibility of Incomplete Rod Insertion (IRI). This has been accomplished by increasing the number of intermediate grids such that lateral stiffness is similar to that of a 12 foot assembly and the guide tubes have a longer region with an enlarged inner diameter in the dashpot region, providing increased assembly bending resistance.

Section 4.6, "Fuel Assembly Liftoff," of MUAP-07016-P, states that the holddown spring undergoes plastic deformation under cold conditions. In RAI 129-1673, Question 04.02-6, the staff asked the applicant to verify the observation. In its response to Question 04.02-6 found in UAP-HF-09024, dated January 30, 2009 the applicant verified the staff's observation and added that at normal operating temperatures, the spring operates at a lower deflection due to the thermal expansion properties of the spring material, which brings it in the linear regime of the material load-deflection curve. The holddown spring has two design functions, 1) hold the assembly against the lower core support plate under normal operation and AOOs and 2) demonstrate that the spring will still be able to perform its design function after a pump overspeed. The second criterion is met if no additional (after the initial cold startup) plastic deformation during a pump overspeed event.

As the spring undergoes plastic deformation on cold startup conditions, the available spring force is reduced. Therefore, an evaluation must be performed demonstrating that the assembly remains seated (no lift-off) under normal operation and AOOs. In Table 4.6-1, "Holddown Spring Evaluation at Cold and Hot Conditions," of MUAP-07016-P, a comparison between the upward forces of hydraulic lift and buoyancy against the downward assembly dead weight and holddown spring forces is presented.

As shown in Table 4.6-1, "Holddown Spring Evaluation at Cold and Hot Conditions," assembly liftoff does not occur but there is no mention of which AOOs were evaluated or if the cold startup condition bounds the respective AOOs; therefore, **Open Item 4.2-8** has been created to address this issue. The assembly hydraulic lift force used in Table 4.6-1 was validated by full-scale hydraulic testing documented in MUAP-11017-P dated May 2011.

In Table 4.6-2, “Holddown Spring Evaluation at Pump-over-speed Condition,” of MUAP-07016-P, the resulting pump overspeed spring force is less than the force at cold startup conditions and hence no additional plastic deformation will occur. To demonstrate that the overspeed spring force is less than at cold startup, a series of full scale hydraulic tests were performed and documented in MUAP-11017. From the review of MUAP-11017 the staff could not determine if additional plastic deformation occurs at pump overspeed conditions. Therefore, **Open Item 4.2-9** has been created to address how the hydraulic testing confirms that no additional plastic deformation occurs. Therefore, the staff agrees that the following two design criteria: 1) no assembly lift-off during normal operation or AOOs, and 2) no additional plastic deformation following a pump overspeed event, are satisfied, and RAI 129-1673 can be closed following successful resolution of **Open Items 4.2-8 and 4.2-9**.

The applicant performed a series of hydraulic vibration tests of a mockup US-APWR fuel assembly to verify that no excessive flow induced vibration of the fuel rods and the fuel assembly occur over a range of flow velocities and temperatures. These tests are documented in Technical Report MUAP-11017-P. The applicant concluded that acceptable flow induced vibration of the fuel rod and fuel assembly will occur in the US-APWR fuel assembly over the entire range of expected operating conditions. In RAI 953-6437, Question 04.02-64, the staff requested the basis for this conclusion, including the acceptance criteria and source reference. This is being tracked as **Open Item 4.2-4**.

A 1000-hour long term flow test was used to verify the computational models for flow induced vibration and fretting wear of the US-APWR fuel assembly. MUAP-11017 contains the 1000 hour fuel fretting test results used to support the grid-to-rod fretting model in MUAP-07016-P, Appendix C. The staff did not review the fretting model, as historical precedence has been to perform the test and then conservatively extrapolate the measurement data to a fuel assembly’s end of operational lifetime. Therefore, RAI 929-6380 was written requesting a conservative extrapolation of the measured fretting data and an evaluation of the stress and fatigue limits. This is being tracked as part of **Open Item 4.2-3**.

In RAI 129-1673, Question 04.02-5, the staff asked if guide tube stresses are evaluated accounting for oxidation and/or hydrogen pickup and what are the assumed oxide thickness and hydrogen pickup values. The applicant responded in UAP-HF-09024, dated January 30, 2009, by stating that the guide tube stresses are evaluated at BOL conditions and compared to the acceptance limit which is defined in terms of the unirradiated yield strength and ultimate tensile strength. The applicant states that this methodology is conservative since the yield strength and ultimate tensile strength increases drastically and early on as the guide tubes undergo irradiation. This is shown in Figure 4.2-5-1, “Mechanical Properties of the Guide Tube,” in UAP-HF-09024. Furthermore, it is explained that this increase in material strength greatly outweighs the relatively small increase in stress on the guide tubes as corrosion occurs and the oxide layer builds to the limiting value occurring at EOL. Hydrogen pickup is not factored into the methodology since data is presented showing that the yield strength and ultimate tensile strength is insensitive to hydrogen pickup at high levels; Figure 4.2-5-2, “Effect of Hydrogen pickup on Material Property of Zircaloy-4,” in UAP-HF-09024 is referenced to show the effect.

The staff finds the methodology to determine the limiting guide tube stresses to be acceptable, since the applicant has demonstrated that the limiting stress evaluation approach at BOL is conservative. Consequently, RAI 129-1673, Question 04.02-5 is closed. However, the applicant’s response to Question 04.02-5, does not address the guide tube ductility. In the review of Question 04.02-5, the staff noted that an [ ] hydrogen limit is used as a guide tube design value. Figure 4.2-3-1, “Mechanical Properties of the Control Rod Guide,” of the

applicant's response to Question 04.02-3, UAP-HF-09024, dated January 30, 2009, only provided uniform elongation data up to a hydrogen concentration of [ ]. As hydrogen concentration has an effect on Zr-4 ductility it is not clear to the staff that cladding ductility is maintained up to [ ] of hydrogen. Therefore, **Open Item 4.2-10** has been created for the applicant to justify a guide tube cladding hydrogen concentration limit greater than the [ ] provided in response to RAI 129-1673, Question 04.02-3.

During the staff's review of DCD Section 4.2.3.5.2, "Analysis of Combined Shock (Including LOCA) and Seismic Loading," it was noted that the guide tube design has an enhanced region with an enlarged inner diameter in the dashpot region of the fuel assembly in order to preclude IRI. In RAI 129-1673, Question 04.02-7, the staff commented that this design improvement appears to only move the weak spot lower in the assembly since there is still a small transition from the large diameter dashpot region back to the small diameter dashpot region for a small axial portion of the guide tube that fits into the bottom nozzle. The staff asked if a comprehensive analysis for the potential for IRI has been performed which accounts for assembly growth, holddown spring force, and assembly lateral stiffness. The staff also asked if the design has been compared to other assembly designs, which have experienced IRI. In its response in UAP-HF-09024, dated January 30, 2009, the applicant responded to RAI 129-1673, Question 04.02-7 by describing the benefit of extending the enlarged diameter dashpot region in terms of reduced drag force on the RCCA. It is also explained that the transition from smaller to larger guide tube diameter in the dashpot region can lead to local assembly distortion and the smaller diameter also reduces the local bending resistance. This implies that the enhanced guide tube design with the larger diameter throughout most of the dashpot region will increase the stiffness and consequently have a larger resistance to bending. The response did not indicate whether comprehensive analysis for the potential for IRI was performed or not, or, whether or not the US-APWR fuel design was compared to other assembly designs which have experienced IRI, but the applicant stated that Japanese PWR plants using the applicant conventional fuel have not experienced IRI. The staff finds the guide tube design in the dashpot region acceptable, since this design enhancement will increase the overall stiffness of the assembly in the dashpot region and will also reduce the drag force on a RCCA when inserting it, further reducing the chance of IRI occurrence. Consequently, RAI 129-1673, Question 04.02-7 is closed.

In RAI 129-1673, Question 04.02-12, the staff asked if an evaluation had been performed examining the potential for guide tube water boiling when control rods are inserted and the possibility of boiling induced corrosion on the guide tube inner diameter. The applicant responded by stating that an ICCC, such as an insertable boron poison rod, bounds a control rod, as the ICCC has a higher heating generation rate. The ICCC is evaluated at the highest coolant sub-channel temperature which is determined assuming all the surrounding fuel rods are at the maximum design hot channel enthalpy rise factor ($F\Delta H$) value. The analysis then assumes the neutron flux, which determines the ICCC heat flux, is based on all surrounding fuel rods at the TS maximum hot channel factor (F_q) value of 2.6. The staff finds the analysis to be conservative, as it considers the maximum sub-channel, outlet coolant temperature and the maximum neutron driven heat flux. As a result, the staff agrees that no guide tube boiling will occur with an inserted control rod or ICCC component, and consequently RAI 129-1673, Question 04.02-12 is closed.

In-Core Control Components (ICCC)

The applicant states in Section 4.2.2.3, "In-core Control Components," of the DCD that apart from their overall geometry and length, the ICCC in the US-APWR do not have any major

differences from ICCC in use in the applicant's current operating fleet. The materials used in all of the ICCC are presented in Table 4.2-2, "In-core Control Component Assembly Specification," of the DCD. The applicant evaluated the ICCC response to normal operation and AOO's and presented the results of these evaluations in MUAP-07008-P. The staff's evaluation of the ICCC covers the information in both the DCD and the applicant's technical report.

The Ag-In-Cd control rods are of a standard design with extensive operating experience. The upper absorber section, which is 151.3 inches long, has an outer diameter of 0.341 inches, while the lower absorber section of [ ] has an outer diameter of [ ]. Cladding material is Type 304 stainless steel. The projected lifetime is 15 calendar years.

In RAI 129-1673, Question 04.02-8, the staff asked if the control rod lifetime of 15 years in Section 4.2.2.3.1 of the DCD, "Control Rods and Rod Cluster Control Assemblies," is in calendar years or effective full power years. The staff also asked the applicant to discuss the data and assumptions used to arrive at this lifetime. The applicant responded that the limiting factor is the irradiation swelling of the control rod cladding and data from operational experience was used to establish the maximum control rod diameter increase over the control rod lifetime. In its response found in UAP-HF-09024, the applicant did not provide enough information for the staff to make a safety conclusion, so follow-up RAI 869-6139, Question 04.02-45 was submitted requesting details regarding how the control rod lifetime was determined. The applicant stated that a maximum fluence is determined over 15 Effective Full Power Years (EFPY) at a core average kW/ft of [ ] (core average power). This fluence value is then multiplied by a safety factor of [ ] to yield the projected end of duty fluence and maximum cladding swelling. In addition to the maximum fluence induced swelling, the maximum manufacturing tolerances are used to predict the final maximum control rod diameter. The staff finds the analysis assumptions to be conservative, as the control tip will not experience a constant, core average neutron flux, as the rods are usually mostly withdrawn in PWR operation. Also, the analysis assumes EFPY while the DCD control lifetime is given in terms of 15 calendar years. This is conservative, as it assumes a capacity factor of [ ], which does not account for any refueling/outage time. Finally, using a safety factor of [ ] provides enough margin to account for any reasonable uncertainty in clad swelling rate versus fluence data provided in the applicant's letter UAP-HF-12037 dated February 9, 2012. Therefore, RAI 129-1673, Question 04.02-8 and RAI 869-6139, Question 04.02-45 are acceptable and closed, as a gap will remain between the guide tube inner diameter and maximum cladding outer diameter over the 15 year calendar control rod lifetime.

In RAI 129-1673, Question 04.02-9, the staff asked the applicant to elaborate on the reduction in reactivity worth of the control rod over its lifetime and to provide the conservative assumptions used regarding the rod's neutron absorbing capabilities. The applicant's response found in UAP-HF-09024, explains that a maximum fluence limit is set for the bottom of the control rod which minimizes the impact on reactivity control and shutdown margin. It is explained that the selected fluence limit applies to the bottom portion of the control rod only and corresponds to a reduction in rod worth that, when averaged over the entire rod, is minimal. Since the fluence limit criterion provides for such a small reduction in overall control rod reactivity worth with respect to reactivity control and shutdown margin, the staff finds the fluence limit to be acceptable and RAI 129-1673, Question 04.02-9 is closed.

During the review of MUAP-07016-P, the staff questioned a statement in Section 5.1.6.2, "Evaluation," of the report claiming that there is no significant control rod cladding wear over its lifetime. In RAI 129-1673, Question 04.02-10, the staff asked if there is measured data to support the statement. The staff also asked if the control rod tips had the potential to wear

against the softer, guide tube inner diameter, and if so, has any guide tube wear been observed. In its response to UAP-HF-09024, the applicant states that there will be no significant control rod cladding wear over the RCCA design lifetime based on wear data spanning several years. In a follow-up to RAI 129-1673, Question 04.02-10, Question 04.02-10-1, was issued asking for more details regarding the type and numbers of observations performed including any guide tube wear measurement data. The staff also asked if there were any design differences between the US-APWR and the designs from which the data originated, that might induce control rod drive vibration leading to the control rod tips wearing down the guide tubes. The applicant explained in document UAP-HF-11427, dated December 14, 2011, that the upper head and plenum geometries are the same between the US-APWR and the plants used to provide control rod wear measurements and guide tube wear observations. In addition to upper head and plenum geometry similarities, the applicant also noted that US-APWR core mass flow rates are lower than the plants used to collect the historical data and hence, the driving force for control rod vibration will be less. The control cladding wear data provided demonstrated that sufficient cladding wall thickness will be maintained for the 15 year calendar lifetime. No observed guide tube wear was observed for the 25 fuel assemblies visually examined using a fiberscope. Based on the data provided to Question 04.02-10, Revision 1, the staff agrees that excessive control cladding and guide tube wear will not occur. Therefore, RAI 129-1673, Question 04.02-10-1 is closed.

In RAI 877-5918, Question 04.02-49 the staff asked the applicant to provide the swelling values at EOL and include fast fluence and temperatures for 304 stainless steel and Ag-In-Cd rod control cluster (RCC) assemblies, details on how the maximum control rod diameters will be tracked to EOL, and to summarize past experience with control rods similar to the US-APWR design. In its response in UAP-HF-12006, dated January 11, 2012, the applicant responded to Question 04.02-49 by providing the swelling values for the cladding material, 304 stainless steel, at EOL, as well as the corresponding fast fluence and temperature used in the evaluation. The applicant also summarized experience with similar RCC assemblies in 17x17 fuel type plants including the maximum fluence level, and the number of control rods that have experienced this fluence level. The applicant also stated that the US-APWR RCC design is the same as conventional 17x17 fuel type plants except for overall length. In regard to Ag-In-Cd, swelling the applicant stated that, since there has been no increase in outer diameter in RCC cladding observed in almost 15 years of operation, there is no significant contact between the outer diameter of the Ag-In-Cd and the inner diameter of the RCC cladding. Therefore, the applicant concluded that Ag-In-Cd swelling is not a concern for the performance of the RCC for up to 15 years of operation. The RCC rods for conventional 17x17 fuel assemblies have been measured by the UT method at every periodic inspection in Japanese plants. Since Ag-In-Cd swelling has not been observed over the expected US-APWR fluence values there is no need for individual tracking of control rod tip fluence values. Therefore, the response to RAI 877-5918, Question 04.02-49 is acceptable and this question is closed.

In RAI 877-5918, Question 04.02-50, the staff asked for more information on the basis for the softening point for the borosilicate glass, how the rod pressures are determined for 15 years of operation, and the resultant fluence level and B-10 neutron captures, and to discuss the applicant's experience with borosilicate glass as a burnable absorber, including experience with similar absorber rods. In its response to Question 04.02-50 in UAP-HF-12006, the applicant stated that the borosilicate glass softening point is given by the manufacturer's specification. The applicant also cited conservatisms in the calculation for the maximum temperature for the borosilicate glass such as a cladding surface temperature above coolant boiling temperature and no heat transfer in the axial direction. The applicant stated that since the expected function for the burnable absorber is reactivity control during one cycle of operation, it is assumed that

the released gas reaches its maximum value within one cycle of operation. The staff needs additional information to determine if this assumption is correct. Burnable absorbers can, and have been, inserted in multiple cycles. As such the staff cannot reach the conclusion that one cycle of operation is limiting. The applicant summarized past experience with burnable absorbers including the number of burnable absorber assemblies irradiated for one, two and three cycle, and provided the maximum neutron fluence level for the burnable absorber assemblies. The applicant also stated that the only difference between US-APWR burnable absorber and the design for the Japanese conventional 17x17 fuel type plants is that the overall length is 14 feet, instead of 12 feet. The staff needs more information on the maximum calculated pressure for burnable absorber assemblies. **RAI 877-5918, Question 04.02-50 remains open and is Open Item 4.2-11**.

Table 5.2.3-1 of MUAP-08007-P, "Stress Evaluation Results for Control Rod," [Reference 4] summarizes the rod cluster control assembly stress evaluation for combined SSE and LOCA loads. The control rod cladding and the top end plugs are made of 304 SUS stainless steel. Comparing the acceptance limits and the yield strengths, it appears that the control rod cladding limit is based on the material strengths in the cold worked condition. However, not enough information was provided in MUAP-07016-P to confirm this observation. A RAI was submitted requesting the detailed calculations of the acceptance limit stresses, including the assumed material condition (cold worked or annealed) and the material strengths (Sy, Su, and Sm) at the 350°C (662°F) operating temperature. The acceptance limit stresses for the top end plug evaluation are consistent with the strength properties of annealed 304 stainless steel at 325°C (617°F). This is being tracked as **Open Item 4.2-12**.

MUAP-08007-P also discusses the issue of insertability and coolability during seismic loading. The applicant determined that there would be grid spacer deformation in at least one grid spacer in every fuel assembly. This required them to provide reasonable assurance that insertability and coolability was maintained during such an accident. Testing of control rod insertability at room temperature concluded that, at most, the delay time was less than [ ] for all fuel assembly deformations that were tested. The staff found the use of room temperature testing to be acceptable, because the main factor which has a negative effect on the insertability at operating temperatures (the increased fluid drag forces by upward core flow), is mitigated by other factors that are only an issue at lower temperatures (i.e. increased density, increased friction due to higher elastic moduli, and a higher dynamic viscosity).

Additional rod insertion tests were performed with the most distorted grid patterns observed from grid impact tests. Similar grid distortion patterns were observed in the full scale shaker tests reported in Reference 9. The control rod guide thimble misalignment is imposed using thimble array fixtures that are more rigid than the actual buckled grids and result in conservatively high friction drag forces during insertion tests. The control rod guide thimbles used in the tests are also bulged to the grid spacer guide sleeves in four places along the length compared to one bulge per grid in the US-APWR fuel assemblies. This increases the local bending stiffness and concentrates the bending curvature to further increase the friction resistance during insertion. Therefore, the rigidity of the test fixture increases the drag forces and the insertion delay time of the tests compared to the actual fuel assemblies. The tests were also performed at room temperature, where the elastic moduli of the control rods and the guide tubes is higher, further increasing the bending stiffness and the friction forces compared to the operating temperature conditions.

The grid impact tests were performed for the most limiting grid deformations identified from an assessment of the possible combinations of buckled grid rows. It was also determined that the

stiffness of the fuel rods and control rod guide tubes tends to enforce the same grid deformation patterns on axially adjacent grids. Therefore, up to five similarly distorted thimble array fixtures were installed at the central grid spacers of the insertion test fixture. In addition to the distorted thimble arrays, the guide thimbles were also bowed statically or subjected to forced vibrations during the control rod drop tests. Figure D.2.3.4-1 in Reference 4, "Insertion Delay Time from Start of Control Rod Drop to Dashpot Entrance," shows that the maximum insertion delay time was less than [ ] for the highest static and dynamic bowing amplitudes. Because the highest bowing amplitude with five buckled grids increased the insertion time by less than [ ], the applicant chose that as their acceptance limit. So as long as the maximum number of buckled grids does not exceed five, then the staff agrees with this decision. Table 4.4.2-6 in Reference 4, "Buckled Grid Spacer Number for Each Fuel Assembly Position Including Uncertainties," shows that the maximum number of buckled grids is six and it occurs in only four fuel assemblies at the periphery of the core where there are no control rods. Since the buckling of five grids could lead to a control rod delay time of up to [ ] the staff has requested that the applicant evaluate the scram reactivity versus position or time curve for the most limiting event analyzed in Chapter 15. The effect of the [ ] control rod insertion delay is being tracked as **Open Item 4.2-13**.

The specific coolability requirements for a LOCA are addressed by 10 CFR 50.46(b)(4), "Coolable geometry," and (b)(5), "Long-term cooling." This is interpreted by the staff to mean that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. This requirement must be met regardless of whether a seismic event occurs concurrently with LOCA. Appendix D of MUAP-08007-P describes the applicant's approach for evaluating coolability with both LOCA and seismic loads combined. The applicant's method described in Appendix D is consistent with SRP 4.2, Appendix A and GDC 2 for the combination of loads resulting from natural phenomena and accident conditions. Since the analyses performed by the applicant show significant deformation of grid spacers throughout the core with SSE loads alone, the staff requested the results of analyses considering the limiting Large Break LOCA (i.e., double-ended cold leg guillotine) loads only. This request is RAI 953-6437, Question 04.02-65. This is being tracked as **Open Item 4.2-14**.

Testing and Inspection Plan

The applicant states in the DCD that its testing and inspection procedures are compliant with 10 CFR Part 50 Appendix B criteria for Quality Assurance Programs. Both the DCD and the staff's evaluation cover the process used to ensure quality during fuel fabrication and the inspection plan used during refueling to ensure that the fuel being used in the plant remains safe to use. Additional monitoring and surveillance takes place while the plant is in operation that helps ensure the fuel assemblies remain adequately safe.

In RAI 129-1673, Question 04.02-16, the staff asked the applicant if manufacturing moisture limits have been set which eliminate internal hydriding, and if so, what limits and what manufacturing controls, or quality assurance sampling, are in place to limit fuel moisture. The applicant responded in UAP-HF-11427 dated December 14, 2011, by stating that the manufacturing control on pellet moisture content assures that the moisture content is kept sufficiently low compared with the limit of 1.3 ppm as specified in "Standard Specification for Sintered Uranium Dioxide Pellets from American Standards and Testing Methods (ASTM) C776-06," [Reference 8] and as also stated in Section 3.3.1 of MUAP-07008-P, "Hydriding."

The staff finds that the criteria for fuel pellet lot rejection, with respect to moisture content, is acceptable as described in the response to RAI 129-1673, Question 04.02-16 since it is conservative with respect to the ASTM limit, and consequently, RAI 129-1673, Question 04.02-16 is closed.

In Section 4.2.4.5 of the DCD, the applicant states that during normal refueling outages some assemblies will be dimensionally checked. In RAI 129-1673, Question 04.02-18, the staff asked the applicant to discuss the type of dimensional checks to be performed and the criteria to be used to determine re-insertion acceptability. The applicant responded to RAI 129-1673, in UAP-HF-11427, dated December 14, 2011, two types of fuel inspections will occur. The first is a visual inspection performed during every refueling. If the visual inspection discovers any abnormalities, the fuel will undergo a detailed inspection. The results of the detailed inspection will determine if the fuel assembly can be reinserted or other fuel assemblies should be inspected. The type of detailed inspection will be a function of the abnormality observed.

The second type of inspection involves the initial fuel assemblies loaded into the first US-APWR. This more detailed inspection is composed of fuel mechanical and clad oxide thickness measurements as described in the applicant's response to RAI 893-6232, Question 04.02-51, Table 4.2-19-1, "Surveillance Program (Minimum Scope) Measurements and Inspections for Fuel Assemblies loaded into the initial Core (Close examination)". The fuel mechanical inspection comprises: assembly growth, assembly bow, total gap and rod bow at the end of Cycle 2, for a 24 month fuel cycle, or at the end of Cycle 3, for an 18 month fuel cycle. The fuel mechanical inspection results will be compared to the design criteria given in MUAP-07016-P. A report describing the results of these fuel inspections will be made available for possible NRC review 90 days following the last assembly inspection. The staff finds the detailed fuel mechanical inspections for the first US-APWR fuel to reach either the end of Cycle 2, for a 24 month fuel cycle, or Cycle 3, acceptable as the fuel cladding, guide tube and grid materials are the same for current operating PWR designs and have demonstrated acceptable performance characteristics.

As noted in the staff's MUAP-07008-P SE, the projected US-APWR fuel rod power histories are different than those used to the develop the oxide thickness growth model in MUAP-07008-P. Therefore, the staff asked in RAI 129-1673, Question 04.02-19 what type of detailed inspections would be performed to ensure that 1) the [ ] oxide thickness limit would not be violated and, 2) that the oxide thickness model in MUAP-07008-P is conservative for the expected US-APWR rod power histories. In its response in UAP-HF-11427 (and in UAP-HF-12046) the applicant committed to modifying DCD Section 4.2.4.5, "Inservice Surveillance," to include oxide thickness fuel inspections at the end of Cycles 1 and 2, for a 24 month fuel cycle, or at the end of Cycles 2 and 3, for an 18 month fuel cycle. This is **Confirmatory Item 4.2-1**. The oxide inspection plan sets a criterion to ensure the [ ] limit is not exceeded for re-inserted fuel assemblies and documents the comparison between predicted and measured oxide thickness values. An interim report for the first fuel inspections will be available for possible NRC review before the start of the next operating cycle. A complete report, including the relevant portions of the interim report, will be made available for possible NRC review 90 days following the off-load of the last fuel assemblies following Cycle 2, for a 24 month fuel cycle, or Cycle 3 for an 18 month fuel cycle. The oxide thickness measurements and corresponding reports need only to be performed for the first US-APWR to reach the above refueling milestones. The details of the oxide thickness inspections can be found in UAP-HF-11427, Revision 1, Table 4.2-19-1, "Surveillance Program (Minimum Scope) Measurements and Inspections for Fuel Assemblies loaded into the initial Core (Close examination)." The final DCD Section 4.2.4.5 wording change, including identification that the revised wording is Tier 2*, can be found in the response

to RAI 893-6232, Question 04.02-51, UAP-HF-12153 dated June 18, 2012. The staff finds the oxide fuel inspection plan acceptable, as US-APWR power densities (kW/ft) are low and oxide thickness builds slowly during the first cycle of operation.

Therefore, performing inspections at the end of the first cycle for a 24 month fuel cycle or second cycle for an 18 month fuel cycle is acceptable, as the [ ] limit will not be violated.

Confirmatory Item

Confirmatory Item 4.2-1. Confirm that DCD Section 4.2.4.5 is modified to include the oxide fuel inspections as described in RAI response 129-1673. Confirm the oxide fuel inspections are Tier 2*.

Open Items

Open Item (OI) 4.2-1, MUAP-07034-P, Safety Evaluation by the Office of New Reactors, Topical Report MUAP-07034-P, Revision 3, "FINDS: Mitsubishi PWR Fuel Assemblies Seismic Analysis Code," Mitsubishi Heavy Industries, Ltd.

OI 4.2-2, The staff's evaluation of MUAP-08007-P, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads."

OI 4.2-3, RAI 929-6380. Provide a conservative estimate of grid-to-rod-fretting wear and the effect on guide tube stresses and fatigue.

OI 4.2-4, RAI 953-6437. The staff requested the basis for the applicant's conclusion that acceptable flow induced vibration of the fuel rod and fuel assembly will occur in the US-APWR fuel assembly over the entire range of expected operating conditions, including the acceptance criteria and source reference.

OI 4.2-5, RAI 948-6383. The staff requested that the applicant clarify the BOL and EOL elastic limit loads used in the FINDS impact model, and the specific buckling load values used to count the number of buckled grids in the analysis.

OI 4.2-6. The staff's review of MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping."

OI 4.2-7. RAI 948-6383. The staff requested a revised structural analysis in MUAP-08007-P which utilized the appropriate soil profiles.

OI 4.2-8, RAI 929-6380. Table 4.6-1of MUAP-07016-P, "Holddown Spring Evaluation at Cold and Hot Conditions," provides data which supports the conclusion that assembly liftoff does not occur during normal operation and AOOs, but does not describe the limiting AOO or if the cold startup condition bounds the limiting AOO. Provide a description of the AOOs evaluated, the limiting AOO and details of how the limiting AOO was determined.

OI 4.2-9, RAI 929-6380. From the review of MUAP-11017-P, the staff could not determine if additional plastic deformation of fuel assembly holddown springs occurs at pump overspeed conditions. The staff wants to know how the hydraulic testing confirms that no additional plastic deformation occurs.

OI 4.2-10, RAI 928-6378. In its response to RAI 129-1673, Question 04.02-5, dated January 30, 2009, an [ ] hydrogen guide tube design limit is stated. The data presented in response to RAI 129-1673, Question 04.02-3 provides data up to hydrogen concentrations of [ ]. Justify why a guide tube hydrogen limit of greater than [ ] is acceptable.

OI 4.2-11, RAI 929-6380. In its response to RAI 877-5918, Question 04.02-50, dated January 11, 2012, the applicant makes an assumption that the burnable absorber (BA) released gas reaches its maximum value within one cycle of operation. From the data provided the staff could not reach that conclusion as removal burnable absorbers are not prohibited from insertion into multiple duty cycles. Therefore, the staff is requesting that additional information be provided regarding the limiting BA internal pressure and whether internal or external pressure is limiting from a cladding strain perspective.

OI 4.2-12, RAI 948-6383. The staff requested detailed calculations of the acceptance limit stresses including the assumed material condition (cold worked or annealed) and the material strengths (Sy, Su, and Sm) at the 350°C (662°F) operating temperature for control rod cladding.

OI 4.2-13, RAI 929-6380. The staff requested the effect of additional [ ] of rod insertion delay.

OI 4.2-14, RAI 953-6437. The staff requests analysis that ensures coolability during worst-case LOCA loading.

Conditions and Limitations

The staff's approval of the oxide thickness model in MUAP-07008-P, "Mitsubishi Fuel Design Criteria and Methodology," is predicated based on fuel inspections described in the response to DCD Chapter 4.2, RAI 129-1673, Question 04.02-19, Revision 1, Table 4.2-19-1, "Surveillance Program (Minimum Scope) Measurements and Inspections for Fuel Assemblies loaded into the initial Core (Close examination)". In its response to RAI 893-6232, Question 04.02-51, dated February 23, 2012, the applicant committed to revising DCD Section 4.2.4.5.3, "Cladding Oxide Thickness Inspections for Additional Design-Basis Verification," to identify the oxide inspection commitment as a Tier 2* item. This is Confirmatory Item 4.2-1.

4.2.5 Combined License Information Items

There are no COL information items from DCD, Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapter 1-19," that affect this section.

4.2.6 Conclusions

As a result of the open items, the staff is unable to finalize its conclusion on Section 4.2 in accordance with the requirements of NRC regulations.

4.2.7 References

1. Mitsubishi Heavy Industries, US-*APWR Topical Report:* "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P, Revision 2, July 2010.

2. Mitsubishi Heavy Industries, *US-APWR Technical Report:* "US-APWR Fuel System Design Evaluation", MUAP-07016-P-P, Revision 2, December 2009.

3. Mitsubishi Heavy Industries, US-*APWR Topical Report:* FINDS: Mitsubishi PWR Fuel Assemblies Seismic Analysis Code, MUAP-07034-P, Revision 3, July 2010.

4. Mitsubishi Heavy Industries, *US-APWR Technical Report*: "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads," MUAP-08007-P Revision 2, December 2010.

5. Mitsubishi Heavy Industries, *US-APWR Topical Report.* "Hydraulic Test of the Full-Scale US-APWR Fuel Assembly" MUAP-11017.

6. American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III.

7. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.

8. "Standard Specification for Sintered Uranium Dioxide Pellets from American Standards and Testing Methods (ASTM) C776-06."

9. NUPEC. "Proving Test on the Seismic Reliability for Nuclear Power Plant, PWR Reactor Core Internals," Nuclear Power Engineering Test Center.

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 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.

The staff based its review of the nuclear design on information contained in the DCD, responses to staff RAIs, and technical reports referenced by the applicant. The staff conducted its evaluation in accordance with the guidelines provided in SRP Section 4.3, "Nuclear Design" (Reference 1).

4.3.2 Summary of Application

4.3.2.1 Nuclear Design

The DCD contains the description of the first cycle fuel loading, which consists of a specified number of fuel bundles. Each fuel bundle (assembly) contains a 17x17 rod array composed nominally of 264 fuel rods, 24 rod cluster control thimbles, and an in-core instrumentation thimble. There are 257 fuel assemblies. The active axial core length and equivalent radial diameter are 13.78 ft (4.2 m) and 12.75 ft (3.9 m), respectively. To attain a desired radial power distribution, three batches of fuel assemblies contain rods of different enrichment. The central region of the core will consist of the higher enrichment, while the lower enriched assemblies will

be placed on the periphery. To reduce pressure vessel embrittlement and increase neutron economy a stainless steel reflector is placed at the periphery of the active core. Reload cores are anticipated to operate approximately 24 months between refueling, accumulating a cycle burnup of approximately 23,000 MWD/MTU. Refueling outage length is 0.5 months.

US-APWR DCD, Table 4.3-1, “Reactor Core Description (Initial Core),” Table 4.3-2, “Main Nuclear Design Parameters for the Initial Core,” Table 4.3-4, “Typical Neutron Flux inside the Reactor Vessel,” and Table 4.3-5, Fast Neutron Fluence at the Reactor Vessel,” contain summaries of the reactor core design parameters, including critical soluble boron concentrations and worths, reactivity coefficients, delayed neutron fraction, neutron lifetimes, and neutron spectrum.

The US-APWR has a heavy steel reflector positioned between the exterior perimeter of core and core barrel. The purpose of the heavy reflector is to reduce fast neutron leakage and vessel irradiation, and to flatten radial core power distribution.

A description of the power distributions, including hot channel factors, measurement, and operational limits is provided. The applicant described how the movable in-core instrumentation system is used to monitor axial and radial core power distributions. The core power distributions have been calculated by the applicant over a broad range of conditions. Section 4.3 describes the various reactivity coefficients. The fuel temperature, or Doppler, coefficient provides negative reactivity feedback with an increase in fuel temperature. The moderator temperature coefficient is generally negative; however, under certain beginning of cycle (low power) core conditions the moderator temperature coefficient may be zero or negative. A power coefficient is also defined as the combined effect of the moderator temperature coefficient and the fuel temperature coefficient. The total power coefficient is shown to be negative throughout core lifetime and in the power operating range.

Reactivity control requirements are also described. The applicant states that soluble boron is added to the reactor coolant to hold reactivity down so that enough control rods remain available to meet shutdown margin requirements. Control rod insertion is limited in order to ensure shutdown capability. Control rod insertion limits are specified limiting conditions for operation in the plant TS contained in Chapter 16, “Technical Specifications.”

Section 4.3 also describes the analytical methods used to generate nuclear cross section data, calculate three dimensional core power distributions, infer reactor power distributions from incore neutron flux measurement, and calculate neutron fluence at the pressure vessel. Reference is made to several licensing reports that document validation of the analytical methods.

4.3.2.2 Analytical Methods

4.3.2.2.1 Neutronics

In DCD, Section 4.3.3.1, “Nuclear Design Methods,” the applicant described the PARAGON and ANC (Reference 2) computer programs and methods used to calculate the nuclear characteristics of the reactor design. The applicant has submitted Technical Report MUAP-07019-P, “Qualification of Nuclear Design Methodology using PARAGON/ANC,” to justify its use of the codes. The methodology was developed by Westinghouse Electric Co., LLC. (WH) for calculating PWR core parameters and had been approved by the NRC for use by WH. PARAGON is an assembly code and is used to generate few group cross sections, which are

used in ANC for a full core simulation. The applicant's qualification process consists of analyzing various critical experiments, analysis of post-irradiated fuel, and application to operating reactors.

PARAGON is a lattice physics code that models a two-dimensional heterogeneous assembly. It solves the coupled collision probability equations, using a nuclear data library based on Evaluated Nuclear Data File, version B, (ENDF/B-VI). The multi-group nuclear data library for the nuclides in each of the various regions within a heterogeneous assembly are generated by collapsing a fine group library using a spectrum calculated using a B_1 leakage calculation (assuming linearly anisotropic scattering). The input to ANC (few group cross sections), is obtained by collapsing the assembly group constants using the normalized assembly wide group spectrum. In PARAGON depletion is accomplished by solving the relevant equation using the Laplace technique.

ANC is a core simulator code that solves the 3-D two-group diffusion theory equations using the nodal expansion method. The few group cross sections are created by PARAGON.

The output from ANC is a variety of core parameters such as critical boron concentration, power distribution, fuel burnup, reactivity coefficients, etc. Pin-by-pin power re-construction is also possible, using discontinuity factors. Special attention is paid to the buildup and burnout of nuclides that have a significant effect on the group constants e.g. Xenon and Samarium (Reference 3).

- Critical Experiments and Post-Irradiation Examination Analysis.

The PARAGON code is applied to the analysis of a variety of domestic and overseas critical experiments, and the analysis of post-irradiation examination (PIE) analysis of exposed fuel. In addition, critical experiments that included a significant thickness of iron in the reflector were analyzed to validate the capability to predict the effect of iron reflector thickness.

A comparison between PARAGON's predicted results and measured results from over one hundred critical experiments are described in the report. These experiments varied the water to uranium ratios, uranium enrichments, experimental buckling, pellet diameter, and boron concentration. In addition, a comparison was made with experiments carried out for a range of above ambient temperatures. Furthermore, a series of experiments in which part- and full-length burnable poison and control rods are included were analyzed. Finally, a set of international critical experiments that include mixed oxide (MOX) fuel were analyzed (References 4 – 8). Generally, the agreement between PARAGON and the measured parameters was good, and there did not seem to be any particular trend with any of the variables measured. The difference between PARAGON calculated and measured rod powers is within [ ] but a majority of the pin powers is within [ ]. The one exception being the MOX based experiments in which differences as high as [ ] for maximum rod power were observed. The standard deviation of the relative difference of fuel rod power distribution is [ ] for MOX fuel and [ ] for UO_2 fuel, respectively.

Using PIE the ability of PARAGON to predict the isotopic variation with depletion was confirmed. Post-Irradiation analysis of samples retrieved from two operating reactors was compared with predicted results. The samples were obtained from the Saxton and Yankee power plants following a burnup of approximately 56 GWD/T (Reference 2, 9, 10). The PIEs were performed for uranium oxide and mixed oxide fuel. In addition, fuel containing Gd in the European test reactor Belgian Reactor 3 (BR3) were also analyzed in order to verify the variation of the

gadolinium isotopic content with depletion (Reference 11). The agreement between PARAGON predictions of isotopic vectors and measured results was good. However, some trends with plutonium and uranium content in the case of MOX based fuel did manifest themselves. These results indicate that the agreement for UO_2 based fuels is approximately [ ], but for MOX based fuels the deviations can be twice as large. This point is discussed further in the evaluation Section 4.3.4.6 of this SE.

The US-APWR configuration includes a thick stainless steel reflector, rather than having the usual thinner iron structures in the reflector region. In order to validate the ability of PARAGON to correctly predict the behavior of the neutron flux at the core/reflector interface a series of relevant critical experiments were carried out, which were then analyzed using PARAGON (Reference 12,13). The measured and calculated thermal and epithermal neutron flux distributions show good agreement. The results of reactivity effect (percent) with iron reflector thickness (mm) essentially fall on top of each other up to a thickness of 160 mm.

Furthermore, the measured epithermal and thermal fluxes deviate from the calculated fluxes by fractions of a percent at the core reflector interface. However, the fueled region was limited to 2.6 percent enriched fuel, and no MOX fueled experiments were carried out.

- Validation and Benchmarking of PARAGON/ANC code package.

A series of startup physics test data and operating plant measurements taken on WH and Combustion Engineering type PWR plants was used to validate the entire PARAGON/ANC code package. In addition, the applicant has applied the code package to predicting Japanese PWR performance. The results presented in the technical report only apply to the Japanese PWR data; the other startup comparisons are only referred to by reference (Reference 2, 14).

The startup physics test analyses presented in the Technical Report MUAP-07019-P is for two PWR plants. Both have 17x17 fuel assemblies and enrichments varying from 2.0 percent to 4.8 percent, with Gd burnable poison rods (6.0 wt percent and 10.0 wt percent). The one core has 157 assemblies corresponding to a three loop plant, and the other 193 assemblies corresponding to a four loop plant. The hot zero power startup tests included measurements of critical boron concentration, isothermal temperature coefficients, and control rod worth. In general the agreement with the startup test data with PARAGON/ANC code package predictions is good. There are instances where the difference for the critical boron concentration is high [ ], but on average the critical boron concentration is predicted with good accuracy (average difference [ ]). There do not seem to be any particular trends with plant type and core. In addition, a xenon induced power tilt was induced as part of the startup tests. The resulting power oscillation, its damping and final amplitude were measured and predicted by PARAGON/ANC. The code package managed to predict both the amplitude and period of the perturbation with good accuracy.

The PARAGON/ANC code package was used to predict the critical boron concentration as a function of burnup, assembly power distribution, and the axial power shape for operating reactors and for selected cycles. The critical boron concentration as a function of burnup dependency is predicted with good agreement with measured values. The assembly average power distribution agrees to within approximately five percent. In all 30 flux maps were analyzed for beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) conditions. Finally, the axial power distribution was predicted using PARAGON/ANC, and the agreement was again good with the predicted and measured points essentially falling on top of each other.

4.3.2.2.2 Power Distribution Monitoring

The applicant submitted Technical Report MUAP-07021-P, "US-APWR Incore Power Distribution Evaluation Methodology," to describe the power distribution evaluation methodology for the US-APWR. The incore power distribution is determined by taking a flux map of the core, using incore detectors, and interpolating between the measured points using an appropriate core simulator. Included in the discussion is an analysis of the uncertainties associated with the power peaking factors.

In the following sections, the incore power distribution methodology is described by using a flux map of the core and incore detectors. An interpolation method is used between the measured points. Included in the discussion is an analysis of the uncertainties associated with the power peaking factors.

- Summary of Incore Nuclear Instrumentation System.

Movable fission chamber detectors (MDs) are used to determine a neutron flux distribution along an axial path through the core at selected assembly positions. The axial path is determined by the location of a thimble that guides and houses the fission chamber. The fission chamber is attached to a cable that can be driven into the core. Each drive unit has a motor, a take up spool for the cable, and a path selector. In this manner the path selector can insert any detector into any of the thimble locations. Calibration of the fission chambers is achieved by inserting them into a common path and cross correlating the responses. The thimbles extend from the detector drive unit in the top of the pressure vessel to the bottom of the core.

Flux maps are recorded in accordance with the TS surveillance requirements. Selecting a thimble location and driving the fission chamber through it into the core creates a flux profile. The resulting signal generated by the fission chamber is recorded and converted to a neutron flux value. Following this, another thimble is selected and the process repeated. This eventually results in a neutron flux map representative of the core at the locations at which the measurements were made. A total of 37 instrumentation thimbles are distributed, nearly uniformly, in the core of the US-APWR.

- Incore Power Distribution Determination Method.

A method is outlined to determine the power distribution within the US-APWR core using the flux maps determined by the MDs. This method is based on that used in conventional PWRs. The point-wise recordings created by the MDs are converted to power distributions, and are corrected to compensate for reactor power fluctuations while the flux trace is being taken. A relationship is discussed in the Technical Report MUAP-07021-P that relates reaction rate in the fission chamber to detector readout, background, and reactor power from ex-core detectors. To correct for differences between fission chambers the common thimble is used as a calibration pass.

The measured data are integrated with data predicted using the 3-D core simulator code ANC. The predicted data file from the 3-D simulator code is known as the "source deck" in the topical report. The source deck data takes the form of a three-dimensional flux or power map. The calculations are carried out to simulate the condition of the core at the time of the incore measurements.

The evaluated assembly average power, or fuel rod power in assembly “i” at axial location “k” is given by summing over the nearest measured reaction rates multiplied by the ratio of measured to predicted powers, and by a weight factor that is a function of distance. The distance referred to is between the point of interest and the nearest measured point. Power peaking is determined based on the relative fuel rod power distribution. The hot channel factor for an arbitrary fuel rod is defined as the ratio of the integrated fuel rod power to the average fuel rod power in the core.

This ratio is modified by multiplying it with the “engineering heat flux hot channel factor” that allows for engineering uncertainties.

- INCORE-M Code

The INCORE-M code integrates the measured and predicted data described in the above section, and determines the power at locations intermediate to the measured locations, and is based on the WH INCORE code (Reference 15). Input to the code consists of the following options – plant description and specifications for the calculational model to be used, core conditions, cycle number and [ ], movable detector data, which are reaction rate data measurements, and “Source deck,” which is the predicted data generated by PARAGON/ANC core simulator.

The output consists of:

1. Echoes back input data – MD data and “source deck” data, in addition [ ].

2. Evaluated incore power distributions – [ ].

- Uncertainty Evaluation for Power Peaking Factors

The uncertainty analysis is based on an approved analysis submitted by WH (Reference 16). However, there is a major difference between the WH system and the current system, and that involves the number of radial detector locations in the core. In the original WH document approximately 30 percent of the fuel assemblies in the core had an instrumentation thimble. In the US-APWR that fraction is reduced to 14.4 percent, which means that proportionately there are only half as many instrument thimbles available to acquire data. The effect of this reduction on the uncertainty is that the distance between measurements is larger implying that more reliance will be placed on interpolation. In addition, if there is any thimble/instrumentation failures the effect will be proportionately higher.

The uncertainty analysis of the peaking factors is divided into the following six components:

1) Radial measurement reproducibility;
2) Predicted power-to-reaction rate;
3) Extrapolation between measurements;
4) Predicted local peaking;
5) Predicted elevation-dependent, power-to-reaction rate, and

6) Axial measurement reproducibility

Components 1), 2), 4), 5), 6) all are controlled by predicted code calculations (PARAGON/ANC), or the accuracy of the detectors, leaving only number 3 as the uncertainty component specific to the current case. In order to estimate the magnitude of the uncertainty introduced by extrapolation the powers determined with and without extrapolation is compared. This comparison was carried out on an actual four loop PWR, which has 193 fuel assemblies and a total of 58 instrumented assemblies. In this experiment [ ] instrumented assemblies were retained, while the remaining [ ] were treated as values obtained from extrapolation between the measured points.

The standard deviation between the actual measured values at the [ ] locations compared to the value obtained by extrapolation from the [ ] retained instrumented locations for [ ] different combinations of [ ] varied from [ ]. These standard deviations are combined to yield a standard deviation of [ ] in predicted assembly power based on a total of [ ] data points. Thus, [ ] is the value used for component three mentioned above.

An estimate of the overall uncertainty was made by combining the above six components as determined for a WH PWR (Reference 16), with the exception of number three, which used the value of [ ] as determined above. The applicant found that the overall uncertainty estimate for the hot channel factor for a fuel rod was [ ], and the uncertainty for the heat flux hot channel factor for a fuel rod was [ ]. These two values have a one sided 95 percent probability and a 95 percent confidence level. In addition, these values are slightly higher than the corresponding values for a WH PWR, since the uncertainty associated with the number of instrumented fuel assemblies is higher. Finally, it was pointed out that in the actual analysis of the US-APWR core the values used for these two uncertainties are six percent and eight percent respectively.

- A Study on the Effect of Instrumentation Thimble Failure for the US-APWR.

This study was performed by the applicant to evaluate the effect of reducing the number of available instrumentation thimbles. Instrumentation thimbles could be unavailable due to blockage or damage to the thimble tubes. An effect on the peaking factor must be estimated by deleting selected thimbles for a variety of core conditions. The core conditions vary with burnup, control rod alignment, and accident conditions associated with dropped control rods and power tilts. It is not possible to gain data on these perturbations from operating reactors, thus a series of core calculations were carried out (using PARAGON/ANC) to simulate these conditions. Thus, by creating the data from a control (normal operating condition) calculation, and using it as input to the INCORE-M code to give simulated MD data. The same sequence of calculations was carried out for the perturbed core. In the case of the perturbed cores "instrumentation thimbles" were selectively deleted in the determination of various peaking factors. The deletion of the thimbles was not entirely random, since it was attempted to avert adjacent thimble failures. A condition that there be no thimble failure within $(10)^{1/2}$ assembly pitches was imposed on the selection of the failed thimble locations.

The applicant found that even following five deleted thimbles subject to the condition described above, including the one closest to the hot spot the maximum change in the uncertainty is [ ] for F_Q and [ ] for $F_{\Delta H}$. This result seems to be essentially independent of core condition and accident scenario assumed, and the deviations need to be added to the uncertainties determined above i.e. [ ] for $F_{\Delta H}$ and [ ] for F_Q.

4.3.2.2.3 Fluence Calculation

The neutron flux evaluation methodology is selected in accordance with, “neutron fluence calculational methods,” of RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” (Reference 17). The evaluation methodology is as follows.

The DORT code is used to evaluate neutron flux distributions (Reference 18). DORT is widely used in the nuclear industry for flux-distribution evaluations of reactor vessels. DORT is a discrete ordinates S_n code, and can perform calculations in (X, Y), (R, θ) and (R, Z) geometry. Each of these geometric capabilities is used as described below.

To determine the reactor vessel neutron flux distribution, (R, θ) geometry is selected, modeling the circular shape of the reactor vessel and core barrel. Likewise, the irregular shapes of the core and the neutron reflector are “smoothed” and are modeled as circular regions. Calculations of the perpendicular (vertical) neutron flux distribution in the reactor vessel are performed with an (R, Z) geometric model. On the other hand, in the evaluation of the neutron reflector neutron flux distribution, greater local spatial accuracy is desired. Therefore, (X, Y) geometry is selected for this region, and the “polygon” shape of the side facing the reactor core of the neutron-reflector is modeled by straight lines.

For the DORT calculations, the BUGLE-96 cross section library is used (Reference 19). This library is generated from ENDF/B-VI data collapsed to 47 neutron energy groups.

Fuel assembly and pin-wise power distributions are obtained by standard reactor core calculations and are used as the fission source terms for the neutron flux distribution calculations outside the reactor core. The average power distribution in a representative reactor core operating at full power is used. This distribution is considered by the applicant to be representative of the core average power distribution during the plant lifetime. This approach is considered appropriate for reactor vessel irradiation calculations, because the integrated flux (neutron fluence) during operation is considered most important.

4.3.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are given in Section 4.5.3 of NUREG-0800, “Review of Safety Analysis Reports for Nuclear Power Plants,” and are summarized below. Review interfaces with other SRP sections can be found in Section 4.5.3 of NUREG-0800.

DCD, Section 4.3, “Nuclear Design,” (Reference 20) presents the design bases for the US-APWR nuclear design. The nuclear design must ensure that the specified acceptable fuel design limits will not be exceeded during normal operation, including 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. To meet these objectives, the nuclear design must conform to the following GDCs:

1. 10 CFR Part 50, Appendix A, GDC 10, “Reactor Design,” requiring the reactor design (reactor core, reactor coolant system, control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.

2. GDC 11, “Reactor Inherent Protection,” requiring a net negative prompt feedback coefficient in the power operating range.

3. GDC 12, “Suppression of Reactor Power Oscillations,” requiring that power oscillations that can result in conditions exceeding SAFDLs are not possible, or can be reliably and readily detected and suppressed.

4. GDC 13, “Instrumentation and Control,” requiring a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

5. GDC 20, “Protection System Functions,” requiring, in part, a protection system that automatically initiates a rapid control rod insertion to assure that fuel design limits are not exceeded as a result of AOOs.

6. GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” requiring protection systems designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

7. GDC 26, “Reactivity Control System Redundancy and Capability,” requiring, in part, a reactivity control system capable of holding the reactor subcritical under cold conditions.

8. GDC 27, “Combined Reactivity Control Systems Capability,” requiring, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system (ECCS).

9. GDC 28, “Reactivity Limits,” requiring, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding.

The acceptance criteria in SRP 4.3 applicable to core power distribution are that:

- 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, and

- 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.

Acceptance criteria for power spiking are discussed in RG 1.126, “An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification.”

There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady state or load following. These limits are determined from an integrated consideration of fuel limits (DCD Section 4.2 “Fuel System Design”), thermal limits (DCD Section 4.4 “Thermal-Hydraulic Design”), scram limits (DCD

Chapter 7 “Instrumentation and Controls”), and transient and accident analyses (DCD Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204°C (2200°F) peak cladding temperature allowed for LOCAs, are not exceeded during design-basis accidents.

There are no criteria that explicitly establish acceptable ranges of reactivity coefficient values or preclude the acceptability of a positive moderator temperature coefficient such as may exist in PWRs at beginning of core life. The criterion stated in GDC 11 is applied directly to design reactivity coefficients, and acceptability is determined in the reviews of the analyses in which they are used.

The acceptability of control rod patterns and control rod worth are determined based on an evaluation of reactivity insertion rates relative to the transient and accident analyses. In addition, equipment, operating limits, and procedures should be capable of limiting the reactivity insertion rates, consistent with transient and accident analyses.

The acceptability of the analytical methods described is determined based on the validation results presented or cited, and a determination of the applicability of the analytical methods to the design of the US-APWR fuel system and nuclear design.

Acceptance criteria adequate to meet initial testing guidelines are discussed in RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” as related to the confirmation of the nuclear design.

4.3.4 Technical Evaluation

4.3.4.1 Power Distribution

The acceptance criteria in the area of nuclear design, specifically power distributions, are based on meeting the relevant requirements of the GDC (particularly GDC 13) related to the reactor core and the reactivity control systems.

Three dimensional power distributions are calculated using the approved NRC codes PARAGON and ANC. The validation of these codes is detailed in MUAP-07019-P, “Qualification of Nuclear Design Methodology Using PARAGON/ANC,” issued May 2007, evaluated in Section 4.3.4.6 below.

The upper limit total initial core peaking factor, F_Q, for the US-APWR is 2.60, corresponding to 12.1 kW/ft. The average linear power for the US-APWR is 4.65 kW/ft. The uncertainties for the power distribution calculations are discussed in WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” issued by Westinghouse in June 1988, and Technical Report MUAP-07021-P, issued December 2007.

As required by GDC 10, the design bases affecting power distribution of the US-APWR include the following parameters:

- The peaking factor in the core will not be greater than 2.60 during normal operation or AOOs at full power.

- Under abnormal conditions the peak linear heat rate will not cause fuel melting.

- The core will not operate, with a power distribution that will cause a departure from nucleate boiling (DNB).

GDC 13 provides the required criteria to evaluate online and ex-core monitoring. The online core monitoring system will be employed to continuously monitor important reactor characteristics and establish margins to operating limits. The incore instrumentation devices are movable fission chamber detectors which are inserted into the instrumentation thimbles from the top of the core. The methods for converting the output signal of the fission chamber to a usable power distribution are discussed in MUAP-07021-P, discussed in Section 4.3.4.7, below.

The axial power distribution is measured by ex-core detectors. The power in the top and bottom halves of the core is obtained by four ex-core detectors placed in 90 degree symmetry. The axial power distribution is verified by calibrating the ex-core detectors measurements against the in-core detectors measurements. The calibrated flux difference is used determine the shape penalty function to the over-temperature delta T (OTΔT) DNB protection and the overpower delta T (OPΔT) overpower protection.

On the basis of the design information provided in DCD Tier 2, Section 4.3.2.2, "Power Distribution," regarding the power distributions and core monitoring, and the review of the Technical Report MUAP-07021-P, the staff concludes that this section of the DCD is acceptable because it meets the requirements of GDC 13.

4.3.4.2 Reactivity Coefficients

The reactivity coefficients express the effects of changes in the core conditions, such as power, fuel and moderator temperature, moderator density, and boron concentration, on core reactivity. These coefficients vary with fuel burnup and power level. The applicant has provided calculated values and trends of the coefficients throughout the anticipated cycle length in US-APWR DCD (Reference 2), Table 4.3-2, Figure 4.3-21, Figure 4.3-22, Figure 4.3-23, Figure 4.3-24, and Figure 4.23-25. The applicant used the NRC-approved two dimensional transport lattice physics code PARAGON and core simulator ANC to determine these reactivity coefficient calculations. In addition, moderator and Doppler coefficients, along with boron worth, will be measured as part of the startup physics testing to assure that the actual values for these parameters are within the range of those used in these analyses.

As required by GDC 11 the US-APWR Doppler coefficient is negative through the fuel cycle. The predicted moderator temperature coefficient (MTC) values are negative for the full range of expected operating conditions during the initial cycle. DCD Section 15.0.0.2.4, "Reactivity Coefficients," states that the minimum MTC is zero (equivalent to having a zero or negative MTC). The value of the MTC is a function of the concentration of the soluble boron; this value becomes more positive as the boron concentration increases. A negative MTC at beginning of life is ensured by using burnable absorbers and reducing the boron concentration.

The staff finds these values for the reactivity coefficients to be acceptable since it meets the requirements of GDC 11.

4.3.4.3 Control Requirements

As set forth above, GDC 20, 25, 26, and 27 specify the requirements for the reactivity control systems.

Chemical neutron poisons dissolved in the reactor coolant system (RCS), RCCAs, and burnable absorbers, control core reactivity. The reactivity control systems are designed to automatically initiate reactivity control, thereby meeting the requirements of GDC 20. To allow for changes in reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of positive reactivity is built into the core. As described below, the US-APWR DCD provides information about the reactivity balance for the first core, and shows that the design incorporates methods to control excess reactivity at all times. This meets the requirements of GDC 27.

The RCCAs and/or adjusting the boron concentration in the reactor and burnable absorbers can control excess reactivity and power level. The DCD describes the boron concentration for several key configurations, including the unit boron worth for the initial cycle. The combination of control systems satisfies the requirements of GDC 25 and 26 because there are two independent systems of different design. Even with a single malfunction of the system, the control systems will still assure that the fuel design limits are not exceeded.

At a steady state normal power operations minor variations in reactivity will be compensated using the boron compensation adjustments. Reactivity changes due to changes in power level and axial xenon oscillations will be controlled by the RCCA. To minimize the distortion of core power distribution, insertion limits are placed on the control rod motion; therefore, when the limits are met, boron concentration is adjusted to compensate for additional changes in reactivity.

The RCCAs are classified into two groups: shutdown group and control group. The shutdown group is always out of the core during power operations. The control group is withdrawn from the core during the cycle to compensate for the loss in reactivity. US-APWR DCD Section 4.3.2.5, “Control Rod Patterns and Reactivity Worths,” discusses the control rod pattern and reactivity worths. Conforming to GDC 25, the control rod group withdrawal is limited by the control rod drive mechanism, which only allows a specific withdrawal sequence, therefore preventing accidental withdrawal.

Soluble boron is used to compensate for slow reactivity changes, including changes associated with fuel burnup, changes in xenon and samarium concentrations, buildup of long-life fission products, and depletion of burnable absorber rod, as well as the large moderator temperature change from cold shutdown to hot standby.

The staff reviewed the US-APWR calculated rod worths and the uncertainties in those worths. The staff has determined that the assessment of the reactivity control system is suitably conservative, and that the control system has adequate negative reactivity worth to ensure shutdown capability, assuming that the most reactive control rod is assumed stuck in the fully withdrawn position. Therefore, the RCCAs and soluble boron worths are acceptable for use in the accident analysis.

On the basis of its review of the information provided in DCD Tier 2, Section 4.3, as described above, the staff concludes that the functional design of the US-APWR reactivity control systems meet the requirements of GDC 20, 25, 26, and 27 and, therefore, is acceptable.

4.3.4.4 Stability—Xenon-Induced Spatial Oscillations

GDC 12 requires that power oscillations, which could result in exceeding of the specified acceptable fuel design limits, be prevented or be readily detected and suppressed.

US-APWR DCD Section 4.3.2.7, "Stability," discusses the stability of the reactor to xenon-induced power distribution oscillations and the control of such transients.

MHI investigated core stability using the modal perturbation method. The results showed the core had radial stability through the cycle length. The US-APWR exhibits axial stability at BOC but the axial stability index becomes zero at MOC and oscillations are possible from MOC to EOC. The applicant states that it has had no incidents of uncontrolled power oscillations from its PWRs. The applicant states the axial oscillation period is long enough to permit control rod movement to compensate for deviations from permitted axial offset. Reactivity changes due to control rod movement would then be compensated by boron dilution/boration. Ex-core detectors are used to detect power distributions beyond the axial offset band.

Accordingly, the staff finds that the information provided in the DCD Section 4.3 has properly addressed the concern of xenon-induced spatial oscillations in accordance with the guidelines of the SRP and has satisfied GDC 12. Therefore, the design is acceptable with respect to power oscillations.

4.3.4.5 Vessel Irradiation

The neutron flux generated in the reactor core is decreased by the neutron reflector, core barrel and reactor coolant which exist between core and reactor vessel. In the evaluation of neutron flux distribution in the reactor vessel, the evaluation technique is selected to consider the attenuation in each region.

The calculated fast neutron (E>1MeV) fluence at the inside surface of the pressure vessel for 60 EFPY was 9.8×10^{18} (n/cm^2). Calculation uncertainty of this flux estimation methodology is less than one percent and is within RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," guidance for fluence calculational uncertainty.

The requirements for analytic calculations of pressure vessel fluence are set forth in RG 1.190 (Reference 17), "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The applicant's method for calculating the vessel fluence for US-APWR reactors is described in Revision 1, issued October 2009, of Technical Report MUAP-09018, "Calculation Methodology for Reactor Vessel Neutron Flux and Fluence." The review of the methodology has focused on: (1) the details of the fluence calculation methods and (2) the qualification of the methods for application to the US-APWR reactors. The results of this review are summarized in the following.

The applicant's vessel fluence calculations are performed with the DORT neutron transport code using the BUGLE-96 cross section library based on ENDF/-VI nuclear data and generally follow the approach described in RG 1.190. However, the staff found that there are several aspects of the proposed methodology that are either outside the approach or do not meet the requirements of RG 1.190. In order to evaluate these issues the RAIs were prepared and forwarded to the applicant.

These issues include: (1) the treatment of the heterogeneous geometry of the steel and water neutron reflector on the core periphery (RAI 545-4290, Questions 04.03-48, 04.03-49, and 04.03-64 to 04.03-66), (2) the need for qualification fluence calculation-to-measurement benchmark comparisons (RAI 545-4290, Questions 04.03-50, and 04.03-60 to 04.03-63), (3) the determination of the fluence calculational uncertainty and bias (RAI 545-4290, Question 04.03-52), (4) the determination of the core neutron source (RAI 545-4290, Questions 04.03-54, 04.03-57, and 04.03-59), (5) the DORT model of the geometry (RAI 545-4290, Questions 04.03-55, 04.03-58), and (6) the inclusion of the baffle/barrel former plates in the DORT calculational model (RAI 545-4290, Question 04.03-56). A brief summary of the RAIs and the applicant's responses are given below.

RAI 545-4290 Question 04.03-48 requested additional justification for [ ].

In its response to this concern, dated April 28, 2010, the applicant has generated a new cross section set using an independent processing system. Comparisons of calculations performed with the two independent cross section sets indicate agreement to within approximately one percent for the vessel peak inner-wall > one-MeV fluence.

This approximately one percent effect is negligible compared to the 20 percent RG 1.190 guidance for fluence calculational uncertainty and is acceptable resolving the concern raised in the RAI.

RAI 545-4290, Question 04.03-49 requested additional justification for [ ].

[ ].

This <one percent effect is negligible compared to the 20 percent RG 1.190 guidance for fluence calculational uncertainty and is acceptable resolving the concern raised in the RAI.

RAI 545-4290, Question 04.03-50 requested additional calculation benchmark comparisons as qualification for the applicant's fluence methodology.

The applicant has provided additional benchmarking, in Response 04.03-50, with its calculations of the recommended pressure vessel benchmark problems of NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions." The comparisons of the applicant's predictions of the vessel inner-wall and surveillance capsule > one-MeV fluence with the NUREG/CR-6115 PWR results indicate good agreement [ ] and provide further detailed qualification of their methodology.

However, because of the limited operating reactor data-base (viz., the single H.B. Robinson-2 capsule measurement), lack of as-built plant measurements, and the special features of the US-APWR (e.g., the neutron reflector), as recommended in RG 1.190 when the applicant's fluence methodology is applied to a specific US-APWR, the methods qualification must be updated by comparing the applicant's predictions with available capsule fluence measurements. These comparisons must be incorporated into the operating reactor measurements data-base, and the methodology calculational biases and uncertainties must be updated as necessary.

The staff requested that a COL information item be added to Table 1.8-2 of the DCD, "Compilation of All Combined License Applicant Items for Chapter 1-19," in regard to collection of plant specific surveillance capsule data to be used to benchmark and adjust the US-APWR fluence predictions as necessary (RAI 874-6101, Question 04.03-68). This is listed in Section 4.3.5 of this SE.

RAI 545-4290, Question 04.03-51, requested justification for [

].

[

].

Since this <one percent effect is negligible compared to the 20 percent RG 1.190 guidance for fluence calculational uncertainty. This is acceptable and resolves the group-dependence concern raised in the RAI.

[

].

RAI 545-4290, Question 04.03-52, requested that calculational input that is known to be biased (e.g., because of differences between the design assumptions and the as-built and operating data) be identified and the effect of the bias on the calculated fluence be determined. This input includes, for example, the plant configuration geometry, coolant temperatures and core neutron source. As given by RG 1.190, when this additional data, beyond that in Technical Report MUAP-09018, becomes available, the effect of these biases must be incorporated in the fluence determination in a conservative manner. A COL item has been added per RAI 874-6101, Question 04.03-68.

RAI 545-4290, Question 04.03-53, requested that [

].

In its response to RAI Question 04.03-53, dated April 28, 2010, the applicant has provided additional calculations which indicate that the effect of these simplifications is approximately one percent.

Since this approximately one percent effect is negligible compared to the 20 percent RG 1.190 guidance for fluence calculational uncertainty. This is acceptable and resolves the concern raised in the RAI.

RAI 545-4290, Question 04.03-54 requested [

].

RAI 545-4290, Question 04.03-55 requested an explanation for the difference between the core boundary in the cylindrical (r, θ) DORT calculational model and the core boundary in the rectangular (x, y) model.

In its response to RAI 04.03-55, dated April 2, 2010, the applicant indicated that the rectangular (x, y) calculation is used to verify the cylindrical (r, θ) calculation at selected locations (which are far removed from the location of the core boundary differences) and that the (r, θ) calculation (which includes an accurate model of the core boundary) is used to determine the fluence at the vessel.

The applicant uses the (x, y) model for evaluation and verification of special effects and the (r, θ) model is used for the determination of the vessel fluence, as described in RG 1.190. This is consistent with standard industry practice and acceptably resolves the concern rose in the RAI.

RAI 545-4290 Question 04.03-56 requested a complete description of the structures/supports (such as former-plates) that are located between the core and the vessel (other than the reflector and barrel) and how they are incorporated in the calculation model.

In its response to RAI 04.03-56, dated April 2, 2010, the applicant has indicated that the US-APWR design does not include former-plates. However, the design does include alignment pins and tie rods associated with the neutron reflector which are included in the model and represented as reflector stainless steel.

The effect of all structures and supports between the core and the vessel are included in the fluence calculational model and, consequently, this acceptably resolves the concern rose in the RAI.

RAI 545-4290, Question 04.03-57 requested that [

].

Since this approximately one percent effect is negligible compared to the 20 percent RG 1.190 guidance for fluence calculational uncertainty, this is acceptable resolving the concern raised in the RAI.

Since the measurements are (1) used to verify/bias fluence predictions and (2) very sensitive to the capsule modeling, RAI 545-4290, Question 04.03-58 requested additional information concerning the DORT calculation modeling of the surveillance capsule in the H. B. Robinson-2, Cycle-9 fluence benchmark measurement.

In its response to RAI 04.03-58, dated April 2, 2010, the applicant has provided a detailed description of the measurement capsule geometrical model and the fine-mesh spatial representation used in the fluence DORT transport benchmark calculation.

The applicant's fine-mesh model of the measurement capsule is sufficiently detailed to insure that the capsule modeling uncertainty is well within the RG 1.190 guidance of 20 percent fluence uncertainty, this is acceptable resolving the concern raised in the RAI.

RAI 545-4290, Question 04.03-59 requested justification for [

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RAI 545-4290, Question 04.03-60 inquired as to the definition of the "equivalent fission-flux" used in the analysis of the Venus-1 pressure vessel simulator measurement.

In its response to RAI 04.03-60, dated April 2, 2010, the applicant includes the specific definition that was used to interpret the Venus-1 fluence measurements and indicates a consistent use of the fission-flux in analyzing the Venus-1 measurements. This is acceptable resolving the concern raised in the RAI.

[

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4.3.4.6 Analytical Methods for Core Neutronics

The PARAGON/ANC code package has already been approved by the NRC for use in the analysis of PWRs (Reference 2). The technical evaluation and the code review concentrated on the agreement between calculated and measured quantities. This agreement is the measure that is being used to gauge the ability of the applicant to correctly use the PARAGON/ANC code package. In addition, issues that have arisen regarding the applicability of the methodology are discussed in the following section, in response to RAI 202-1846, Question 04.03-9.

4.3.4.7 Description of Code Package

Five requests for additional information (RAI 04.03-1, RAI 04.03-2, RAI 04.03-3, RAI 04.03-4, and RAI 04.03-5) were submitted concerning the code package architecture. A brief description of the responses is given below.

RAI 202-1846, Question 04.03-1 required the reference regarding NRC approval of codes mentioned in the submittal. The response to the RAI gives the required reference, thus satisfying the request.

RAI 202-1846, Question 04.03-2 requested clarification regarding the few group structure generated by the PARAGON code.

The PARAGON code starts with an energy group structure that has 70-groups. The 70 groups in PARAGON are divided as follows:

[

]

When using ANC, the resonance range is extended down in energy to 0.625 eV, and the thermal range spans the range from 0.0 eV to 0.625 eV. [
].

This more detailed explanation of the few group structure satisfactorily addressed the RAI.

RAI 202-1846, Question 04.03-3 requested information on the treatment of resonance cross sections in the PARAGON code.

The applicant's response to the RAI indicated that the resonance calculation for MOX fuel is the same method as used in PHEONIX-P code (Reference 21). Those pins that contain Gd as a burnable poison use a spatially dependent Dancoff method for determining the self-shielding in the calculation of the radial flux distribution within the pin.

The above response to the RAI satisfies the request.

RAI 202-1846 Question 04.03-4 requested information regarding the creation of few group cross sections in the reflector core interface, since no modal shape exists at this location.

The determination of few group reflector cross-sections presents particular questions, since no modal shape exists at the core/reflector inter-face. The applicant pointed out that a one-dimensional calculation is carried out using PARAGON that includes core and reflector zones. The resulting spectra are used to determine the respective few group cross sections on either side of the core/reflector interface.

The response to the RAI described in detail the one-dimensional calculation used to create the cross sections, which satisfied the request.

RAI 202-1846, Question 04.03-5 requested a detailed list of the parameters calculated by the core simulator ANC.

The response to the RAI explains that the ANC core simulator code calculates a variety of parameters. These include:

1. power distributions,
2. fuel burnable absorber depletion,
3. reactivity feedback coefficients,
4. boron, xenon, and control rod worth, and
5. xenon stability.

The response contained the above list of parameters calculated by the core simulator ANC, which satisfied the RAI request.

In summary, the applicant's responses to RAI Questions 04.03-1, RAI 04.03-2, RAI 04.03-3, RAI 04.03-4, and RAI 04.03-5, dated March 27, 2009, provide sufficient description of the code architecture. The staff considers the responses to these RAIs acceptable.

4.3.4.8 Analysis of Critical Experiments and Post-Irradiation Measurements

The application to critical experiments, post-irradiation analyses of exposed fuel, and special critical experiments that validate the application of the code package to systems with thick iron reflectors has been addressed. Eight RAIs (RAI 04.03-6, RAI 04.03-7, RAI 04.03-8, RAI 04.03-9, RAI 04.03-10, RAI 04.03-11, RAI 04.03-12, and RAI 04.03-13) were submitted concerning the analysis of critical experiments and post-irradiation measurements. A brief description of the responses is given below.

RAI 202-1846, Question 04.03-6 requested details of generating the few group cross sections for the steel reflector, since there is a significant resonance structure in the iron (a major component of steel) cross section.

The response is partially covered by RAI 04.03-4, dated March 27, 2009, which indicated that the cross sections generated in 70-group by PARAGON was collapsed to a few-group structure for ANC. Additionally a description was included of a series of steel thickness dependent calculations to generate cross sections with varying degrees of geometric self-shielding. The RAI response shows that the few group cross sections for the steel neutron reflector are

generated using a one dimensional PARAGON model. The few group cross sections for ANC are generated by collapsing the PARAGON 70 energy group cross sections and applying flux discontinuity factors at the fuel/reflector interface. The one-dimensional calculation used to create the cross sections, is sufficient and resolves the staff questions.

This response satisfies the RAI request.

RAI 202-1846, Question 04.03-7 requested a description of the critical experiments implied by Reference 1 of the submittal.

A list of the critical experiments that were used in the validation of the PARAGON code is given below:

> One hundred and one critical experiments based on uranium oxide/metal fuel, room temperature water, and a wide range of lattice parameters (Reference 4), KRITZ high temperature critical experiments carried out to simulate operating conditions (Reference 22), Babcock and Wilcox critical experiments that include burnable poison, guide thimbles, and various fuel types (Reference 5, 6), Monte Carlo assembly comparative benchmark analyses for systems containing burnable poisons and a wide variety of fuel types.

The list of critical experiments given in the RAI response satisfies the request.

RAI 202-1846, Question 04.03-8 requests details of the Vermont Yankee and Saxton results.

The post-irradiation fuel analysis covers data determined at the Saxton and Vermont Yankee plants. The data include a variety of measurements made on both MOX and uranium oxide fuel. Both stainless steel and Zircaloy clad were included in the fuel designs, and burn up values up to [ ] were considered.

For each major isotope of uranium or plutonium the measured data were presented in the following ways:

[

]

The data was provided in graphical form using the above ratios as a function of fuel burnup. The isotopic comparisons between measured and calculated results agree well for these experiments in the case of both uranium and MOX based fuel. The calculations were reported in Reference 1 of the report WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON."

The description of the results given in the RAI, and summarized above satisfies the request.

RAI 202-1846, Question 04.03-9 requested magnitudes of the errors involved with the data presented in the submittal, and a reason for the systematic trend in the data.

The data shown for uranium isotopes in Figure 3.1-8, “PIE Analysis (Uranium Isotope)” of Technical Report MUAP-07019, “Qualification of Nuclear Design Methodology using PARAGON/ANC,” agree well with analysis for uranium oxide, except at a burnup of

approximately 50 GWD/T. The agreement between experiment and analysis are provided below:

[

].

[

]. In the case of plutonium isotopes, shown in Figure 3.1-9, "PIE Analysis (Plutonium Isotope)" of Technical Report MUAP-07019.

Even though MOX is not used in the US-APWR, the trends that are apparent are:

1. The agreements for the uranium isotopes are significantly better than those for the plutonium isotopes in both UO_2 and MOX based fuels.

2. The data for plutonium isotopes shows much better agreement in UO_2 than in MOX based fuels.

The detailed response to the RAI, summarized above, satisfies the RAI.

RAI 202-1846, Question 04.03-10 requested information regarding the space dependent burnup of Gd containing rods.

The space-energy variation of the Gd variation with time is treated in the following manner. The space dependent Dancoff factor method is used to determine the space dependent depletion of the Gd.

The key parameters that the BR3 reactor (Reference 11) and the US-APWR have in common are:

1. reactor operating conditions,
2. fuel rod dimensions and pitch,
3. fuel pellet uranium enrichment, dimensions, and Gd loading, and
4. Gd rod environment.

The agreement between calculated and measured data and the similarity between the BR3 and the US-APWR confirms that the gadolinium behavior should be predicted with confidence. The detailed response to the RAI, summarized above, satisfies the RAI.

RAI 202-1846, Question 04.03-11 requested information on the applicant's history in treating high burnup rods using the PARAGON code.

The ability of the applicant to use the PARAGON code to high burn up was validated against experiments that had an average maximum burn up in excess of [

]. The RAI response summarized above answers the request for information.

RAI 202-1846, Question 04.03-12 requested details of the tank type critical experiment (TCA) used to validate the predictions of the thick steel reflector.

A description of the PARAGON TCA critical experiment model is given in Reference 12. The PARAGON model of the TCA included a fuel-reflector-water configuration. The actual TCA fuel region consisted of a [ ] array of fuel rods.

The response to the RAI summarized above, and the graphical comparison presented in MUAP-07019 in Figures 3.1-12 to 3.1-16 of measured and calculated fluxes and reactivity effects as a function of different steel reflector thickness, satisfied the request for information.

In summary, the staff finds that the applicant's responses to RAI 202-1846, Questions 04.03-6, RAI 04.03-7, RAI 04.03-8, RAI 04.03-10, RAI 04.03-11, and RAI 04.03-12, dated March 27, 2009, resolve its questions and are acceptable.

4.3.4.9 Analysis of Operating Plant Data

The ability of the applicant to analyze operating plant data using the PARAGON/ANC code package has been validated (Reference 14).

Two RAIs (RAI 202-1846 Questions 04.03-13 and 04.03-14, and RAI 253-1467 Questions 04.03-15, 04.03-16, and 04.03-17) were issued concerning the analysis of operating plant data. A brief description of the responses is given below.

RAI 202-1846, Question 04.03-13 requested a definition of the stability index.

Among parameters that were analyzed is the "stability index." The time dependent perturbation in the power distribution consists of an exponential term and an oscillatory term. The exponential term either grows or dies with time depending on the sign of the exponent. The magnitude of the exponent is known as the "stability index" and if it is negative, a stable xenon oscillation results. Positive values result in an unstable xenon oscillation. The response summarized above included analytic relationships of the stability index, and satisfied the RAI.

RAI 202-1846, Question 04.03-14 requested information as to which Paragon cross section sets were adjusted, and which sets were unadjusted for input into ANC.

In its response to this RAI dated March 27, 2009, the applicant explained that the few group data that is used in ANC and created by PARAGON is based on unadjusted nuclear data libraries. The cross sections are not modified, except for one case. This case involves the U-238 resonance cross sections, which are adjusted to yield a resonance integral that is approximately [ ], than that determined without adjustment.

The response given above explains that only one cross section set was adjusted, and satisfies the request for information.

[

]

In summary, the applicant's responses to RAI 04.03-14, RAI 04.03-15, RAI 04.03-16, and RAI 04.03-17 resolve these questions, and should be considered closed. The larger difference between measured and calculated data for plutonium isotopes for the PIE analysis suggests that only UO_2 based fuels should be used. This is consistent with the DCD which only requests the use of UO_2 based fuels.

The primary conclusion from this review is that the applicant has demonstrated that it is fully qualified to use the PARAGON/ANC code package to analyze APWR cores. The methodology should be re-visited if MOX based fuels or any other fuel type other than that based on low enriched UO_2 are to be used in the core. This conclusion is based on the seemingly larger discrepancy for the PARAGON/ANC predictions of plutonium isotopes of minus to plus 10 percent in MOX based fuels compared to UO_2 based fuels.

Based on the calculated core parameters, and the applicant's use of PARAGON/ANC, the staff concludes that the information presented adequately demonstrates the ability of this analytical method to calculate the reactor physics characteristics of the US-APWR core.

- Power Distribution Monitoring

The INCORE code package has already been approved by the NRC for use in the analysis of PWRs (Reference 15). The appropriate sections of the SRP (NUREG-0800) were applied to this review.

- Evaluation of Incore Instrumentation System

RAI 257-1613, Questions 04.03-20, 04.03-21, 04.03-22, and 04.03-23, was issued concerning the evaluation of incore instrumentation system. A brief description of the responses is given below.

RAI 257-1613, Question 04.03-20 requested the number of detectors available for concurrent measurement.

[

]

The RAI response provided the information requested, which satisfies the request.

RAI 257-1613, Question 04.03-21 requested how the planes defining the quadrants were arranged.

In its response to this RAI dated March 30, 2009, the applicant provided a figure outlining the planes was supplied. The planes follow the conventional directions i.e. north-south, and east-west and hence the staff finds it acceptable.

The response with the figure satisfied the request for information.

RAI 257-1613, Question 04.03-22 requested information regarding the distribution of the thimbles in each quadrant.

The response indicates that the instrumentation thimbles and RCCA locations are distributed essentially uniformly in each quadrant. The table below shows the number located in each quadrant, and those located on the symmetry planes, which overlap quadrants.

Quadrant	Non-axis locations	Axis locations	Total in quadrant
1 (NW)	8	4	12
2 (NE)	7	4	11
3 (SW)	7	3	10
4 (SE)	8	3	11

It is seen that each of the quadrants has a similar number of instrumentation thimbles. In addition, each of the instrumentation thimbles is located next to a control rod that is distributed in a symmetric manner.

The response to the RAI embellishes the discussion in the submittal and thus satisfies the request for information.

RAI 257-1613, Question 04.03-23 requested the rational for choosing the thimble locations.

The response to this RAI, dated March 30, 2009, shows that [

].

The response to the RAI explains the rational for choosing the location of the thimbles and thus satisfies the request for information.

In summary, the applicant's responses to RAI 257-1613, Questions 04.03-20, 04.03-21, 04.03-22, and 04.03-23 resolve these questions and should be considered closed.

- Incore Power Distribution Determination Method.

The description of the incore power distribution determination method includes relevant equations, and measurement protocol. Eight RAIs (RAI 04.03-24, RAI 04.03-25, RAI 04.03-26, RAI 04.03-27, RAI 04.03-28, RAI 04.03-29, RAI 04.03-30, and RAI 04.03-31) were submitted concerning the incore power distribution determination method. A brief description of the responses is given below.

RAI 257-1613, Question 04.03-24 requested additional information regarding the types of PWRs that are considered in applications of the methodology.

The methodology used in this report is similar to the methodology that has been used for currently operating plants in Japan, and domestic "Westinghouse-type" PWRs. These plants include two, three, and four-loop plants with 14x14, 15x15, and 17x17 fuel assembly arrangements.

The response summarized above outlined the various types of PWRs for which this methodology applies. The response satisfies the request for information by explaining the rational.

RAI 257-1613, Question 04.03-25 requested a justification for applying the methodology to the US-APWR.

The US-APWR is essentially identical to current four loop plants with 17x17 fuel assemblies and a central instrumentation thimble.

The movable fission chambers are in common usage in Japanese PWRs. Thus, the incore power distribution determination method proposed for the US-APWR is the same as in current use in other Japanese plants.

The response of the applicant to the RAI satisfies the RAI by explaining the rational.

RAI 257-1613, Question 04.03-26 requested the frequency with which data is recorded by the movable detectors.

[

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The response to the RAI, dated March 30, 2009, summarized above, explains the intervals over which data is recorded. This information satisfies the request for information.

RAI 257-1613, Question 04.03-27 requested an explanation of the units used in the relation used in determining the detector response.

The units in equation 3-1 are either in [ ].

The response to the request explained which of the units were [
]. The response satisfies the RAI.

RAI 257-1613, Question 04.03-28, requested the number of detectors normally used for a flux map was submitted.

[

]. The response summarized above satisfies the RAI.

RAI 257-1613, Question 04.03-29 requested clarification regarding the method and frequency of calibrating a detector was requested. To correct the difference in sensitivity between detectors, each detector is routed through a calibration thimble once for each flux map determination.

In addition, if detector drift check is necessary the detector might be routed through the calibration thimble twice. Finally, either of the detectors could be designated as a reference detector.

The response of the applicant to the RAI satisfactorily explains the method of calibration. This satisfies the RAI.

RAI 257-1613, Question 04.03-30 requested information regarding the definition of a reference detector.

The designation “reference detector” is purely arbitrary and may be either one of the detectors for a given flux map.

The response to the RAI is satisfactory, in that it explains the use of the term “reference detector.”

In summary, the applicant’s responses to RAI 257-1613, Questions 04.03-24, 04.03-25, 04.03-26, 04.03-27, 04.03-28, 04.03-29, and 04.03-30, resolve these questions and should be considered closed.

- INCORE-M Code

The description of the INCORE-M code is in the form of an outline with attached figures from the computer output pages of the INCORE-M code. Seven requests for additional information (RAI 04.03-31, RAI 04.03-32, RAI 04.03-33, RAI 04.03-34, RAI 04.03-35, RAI 04.03-36, and RAI 04.03-37) were submitted concerning INCORE-M. A brief description of the responses is given below.

RAI 257-1613, Question 04.03-31 requested the axial zone size.

[

]. This response satisfies the RAI.

RAI 257-1613, Question 04.03-32: requested a definition of the engineering hot channel factor was requested.

The engineering heat flux hot channel factor (F_Q^E) in this case accounts for increases in heat flux due to manufacturing tolerance effects. Local variations in the pellet density, diameter, fuel enrichment, and fuel rod surface area are included.

The definition supplied by the applicant satisfied the RAI.

RAI 257-1613, Question 04.03-33 asked if INCORE had been approved by the NRC. The response was “yes,” and a reference was provided in the report. As part of the staff evaluation the version of INCORE used in these calculations was reviewed. The response to the RAI was satisfactory.

RAI 257-1613, Question 04.03-34 requested a description of the differences between INCORE and INCORE-M codes.

The INCORE-M code differentiates itself from INCORE (Reference 15) by the addition of the following modifications:

[

].

The response supplied by the applicant outlined the salient differences. The differences are minimal, and have primarily to do with the weight function that accounts for the extrapolation uncertainty. The response satisfies the RAI.

RAI 257-1613, Question 04.03-36 requested clarification of Incore-M fuel assembly nodalization.

The fuel assemblies are nodalized axially with [ ] radial node. [
].

The response to the RAI clarified the nodalization that is acceptable and satisfied the RAI.

RAI 257-1613, Question 04.03-37 requested a sample output for a typical case.

A sample output is given for the case with an extreme power tilt, and at MOC. Case number 8 in Table A-1, "Summary of Simulation Cases," of the Technical Report MUAP-07021.

The response to the RAI was the submission of an output, which satisfied the RAI.

In summary the applicant's responses to RAI 257-1613, Questions 04.03-31, 04.03-32, 04.03-34, 04.03-36, and 04.03-37, dated March 30, 2009, resolve these questions and are considered closed.

- Uncertainty Evaluation for Power Peaking Factors.

This section provides a reasonable method of determining the uncertainty evaluation for power peaking factors, by dividing the uncertainty into six components, five of which are identical to a WH PWR used in the U.S. The sixth factor is an extrapolation factor based on the number and percentage of core locations that contain instrumented thimbles.

The sixth uncertainty factor was determined for a Japanese PWR that had a 15x15 assembly core with 193 assemblies and [ ] of the assemblies were instrumented. However, the US-APWR has a 17x17 assembly structure and a core with 257 assemblies and with [ ] of the assemblies instrumented, there is a larger distance between instrumented positions. Four RAIs (RAI 04.03-39, RAI 04.03-40, and RAI 04.03-41) were submitted concerning the uncertainty evaluation for peak power factors. A brief description of the responses is given below.

RAI 257-1613, Question 04.03-39 requested a more detailed discussion of the uncertainty analysis.

A detailed description is presented on the six different contributors to the uncertainties for $F_{\Delta H}^{N}$ and F_{Q}^{N}. These are based on a data base summarized in Reference 16, and are applicable primarily to WH plants. Those uncertainties that do not apply explicitly to the US-APWR are discussed in more detail. Briefly, all uncertainties that apply to the WH plants carry over to the US-APWR, except for the extrapolation uncertainty. This uncertainty is referred to as uncertainty number three in the submittal.

The uncertainty associated with extrapolation from instrumented to un-instrumented fuel assemblies is determined by analyzing data determined for a Japanese 15X15 PWR core. In this experiment selected, instrumented locations were treated as extrapolation positions and the remainder as actual instrumented positions. The analytically extrapolated readings were then compared to the measured readings to determine the corresponding uncertainty. Although this is not an exact simulation of a US-APWR, the number of simulated positions and measured positions are kept invariant. The sample standard deviation was evaluated to be [ ] where the total number of data points used was [ ].

The uncertainty of [ ] includes both the measured and extrapolation errors. [ ].

A multi-page response by the applicant is summarized above. All aspects of the uncertainty are elaborated upon. This response impacts other RAIs, as will be pointed out below. The response satisfies the RAI.

RAI 257-1613, Question 04.03-40 requested the units used in uncertainty equations.

The terms in equations 5-1 and 5-2 are [ ].

The response to the RAI satisfactorily explains the units.

RAI 257-1613, Question 04.03-41 requested a list of symbols used in MUAP-07021.

A list of symbols was supplied in alphabetical order, with their definitions.

The response to the RAI was satisfactory.

In summary the applicant's responses to RAI 257-1613, Questions 04.03-38, 04.03-39, 04.03-40, and 04.03-41, dated March 30, 2009, resolves these questions and they should be considered closed.

- A Study on the Effect of Instrumentation Thimble Failure for the US-APWR.

The 3D core simulation code ANC was used to compare a reference design to perturbed designs to simulate single and multiple thimble failures, and the uncertainties were determined to be a maximum of [ ] on $F_{\Delta H}^N$ and F_Q^N. When added to the measurement uncertainties of [ ] on $F_{\Delta H}^N$ and F_Q^N these are less than the factors for safety analysis and core surveillance which were set to six percent and eight percent for $F_{\Delta H}^N$ and F_Q^N respectively. However, in the event of multiple core thimble failures the safety margins are reduced to such low levels that may be unacceptable. It is not clear in the application what action would be taken in the event of a single thimble failure or multiple thimble failures. Four requests for additional information (RAI 04.03-42, RAI 04.03-43, RAI 04.03-44, and RAI 04.03-45) were submitted concerning the uncertainty evaluation for peak power factors. A brief description of the responses is given below.

The request for a detailed discussion on uncertainties is referred to the response to Question 04.03-39.

RAI 257-1613, Question 04.03-42 requested the applicant to elaborate on the uncertainty analysis regarding the extrapolation between assemblies.

The response by the applicant referred back to the response given for RAI 04.03-39, which had a detailed analysis and discussion of uncertainties.

The response given by the applicant is satisfactory in that the cited response covers the request in great detail.

RAI 257-1613, Question 04.03-43 requested the source data used in the power peaking factors.

The response by the applicant referred back to the response given for RAI 04.03-39, which had a detailed analysis and discussion of uncertainties.

The response given by the applicant is satisfactory in that the cited response covers the request in great detail.

RAI 257-1613, Question 04.03-44 requested the perturbations introduced due to misaligned control rods.

An estimate of the uncertainty associated with a misaligned control rod is performed by the applicant assuming two normal perturbations and two situations that result in extreme quadrant tilt, providing a severe global power perturbation. These are designated types A – D perturbations. In the type A perturbation [

].

RAI 257-1613, Question 04.03-45 enquired as to the level of conservatism implied by various perturbations introduced in the core.

The type B perturbation is similar to the type A perturbation, except that [ ]. The type C perturbation is brought about by a dropped control rod at [ ].

[ ]. The type D perturbation is brought about [

].

Of the above perturbations type A and B are chosen to be consistent with typical plant operations. Types C and D represent abnormal conditions. In particular, type C provides a severe local power distribution perturbation, while type D provides a severe global power perturbation. Thus, types C and D result in the more conservative estimates.

In summary, the applicant's responses to Questions 04.03-42, -04.03-43, -04.03-44, and -04.03-45, dated March 30, 2009, resolve these questions and are considered closed.

The methodology outlined in the report for determining incore power distributions is acceptable provided that the maximum number of failed instrumentation thimbles is five with no adjacent failed thimbles within the square root of 10 assembly pitches from each other when performing incore flux measurement. Greater than five failed thimbles or failed thimbles within the square root of 10 assembly pitches of each other are considered a "change to the approved methodology."

RAI 874-6101, Question 04.03-69 was issued regarding the assumed failed thimble conditions in MUAP-07021. In its response to RAI 04.03-69, dated January 10, 2012, the applicant indicated that it will revise the DCD Section 4.3 to include a statement which specifies that the thimble failure conditions exceeding the assumed conditions used in the MUAP-07021, Appendix A, are considered a change to the approved methodology. The response is satisfactory. Confirmatory Item 4.3-1 has been created to track the DCD modification.

4.3.5 Combined License Information Items

The staff requests that a COL information item be added to Table 1.8-2 of the DCD, "Compilation of All Combined License Applicant Items for Chapter 1-19," in regard to collection of plant specific surveillance capsule data used to benchmark and adjust the US-APWR fluence predictions as necessary (RAI 874-6101, Question 04.03-68).

4.3.6 Conclusions

4.3.6.1 Nuclear Design

To allow for changes in reactivity from reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, the applicant has designed a significant amount of excess reactivity into the core. The applicant has provided substantial information about core reactivity balances for the first cycle, and has shown that the design incorporates methods to control excess reactivity at all times in the cycle life. The applicant has shown that sufficient control rod worth would be available at any time during the cycle to shut down the reactor, assuming that, at least a 1.0 percent delta-k/k subcritical margin exists in the hot shutdown condition.

The applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and the control system has adequate negative worth to ensure shutdown capability.

With respect to the requirements applicable to the nuclear design of the US-APWR, the staff finds the following:

- The applicant has satisfied the requirements of GDC 10, 20, and 25 with respect to SAFDLs by demonstrating that the US-APWR design meets the following objectives:
  - No fuel damage occurs during normal operation, including the effects of AOOs (GDC 10).
  - Automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs, and that systems and components important to safety will automatically operate under accident conditions (GDC 20).
  - No single malfunction of the reactivity control system will violate the fuel design limits (GDC 25).
- The staff reviewed the results of the applicant's calculations to demonstrate that the Doppler and moderator coefficients of reactivity are negative and will prevent a rapid, uncontrolled reactivity excursion. The staff has determined that they are suitably conservative, were performed with NRC-approved physics methods, and use appropriate US-APWR-specific inputs. Accordingly, the applicant has satisfied the requirements of GDC 11, with respect to nuclear feedback characteristics.

- The staff reviewed the applicant's analysis of power oscillations and has determined that they are suitably conservative, were performed with NRC-approved physics methods, and use appropriate US-APWR-specific inputs.

  Accordingly, the applicant has satisfied the requirements of GDC 12 by showing that power oscillations are not possible, or can be reliably and readily detected and suppressed.

- The applicant has satisfied the requirements of GDC 13 by providing instrumentation and controls to monitor the following variables and systems that can affect the fission process:

  1. Reactor power distribution.

  2. Control rod positions and patterns.

- The US-APWR design has RCCA and boron concentration control to provide the following capabilities:

  1. Reliable shutdown of the reactor during normal operation conditions and during AOOs.

  2. Adequate boration to establish and maintain safe-shutdown conditions accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 26 by providing two independent reactivity control systems of different design.

- Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 28 with respect to postulated reactivity accidents.

For the reasons set forth above, the staff concludes that the US-APWR nuclear design satisfies the requirements of GDC 10, 11, 12, 13, 20, 25, 26, and 28 and, therefore, is acceptable.

4.3.6.2 Analytical Methods

The applicant described the computer programs and calculation techniques used to predict the nuclear characteristics of the reactor design and provided examples to demonstrate the ability of these methods to predict experimental results. The information presented adequately demonstrates the ability of these analyses to predict the reactivity and physics characteristics of the US-APWR design.

4.3.6.2.1 Neutronics Analysis

The primary conclusion from this review is that the applicant has demonstrated that it is fully qualified to use the PARAGON/ANC code package to analyze PWR cores for UO_2 based fuels only.

Based on the calculated core parameters, and the applicant's use of PARAGON/ANC, the staff concludes that the information presented adequately demonstrates the ability of this analytical method to calculate the reactor physics characteristics of the US-APWR core.

4.3.6.2.2 Power Distribution Monitoring

The methodology outlined in the report for determining incore power distributions is acceptable subject to:

The methodology outlined in the report for determining incore power distributions is acceptable provided that the maximum number of failed instrumentation thimbles is five with no adjacent failed thimbles within the square root of 10 assembly pitches from each other when performing incore flux measurement. Greater than five failed thimbles or failed thimbles within the square root of 10 assembly pitches of each other are considered a "change to the approved methodology." In its response to RAI 04.03-69, dated January 10, 2012, the applicant indicated that it will revise the DCD Section 4.3 to include a statement which specifies that the thimble failure conditions exceeding the assumed conditions used in the MUAP-07021 Appendix A are considered a change to the approved methodology. **Confirmatory Item 4.3-1 has been created to track the DCD modification**.

4.3.6.2.3 Fluence Methodology

The applicant employs a calculational methodology for determining the pressure vessel >1-MeV neutron fluence that generally conforms to the guidance of the NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The methodology uses the DORT transport code together with the ENDF/B-VI nuclear data to determine the attenuation of the neutron flux between the core and the vessel. This code and nuclear data set are consistent with standard industry practice and conform to the recommendations of RG 1.190. The fluence calculational uncertainty is determined by calculation-to-measurement benchmark comparisons and by an independent analytic uncertainty analysis.

Based on a detailed review of the applicant's fluence methodology, the staff concluded that the methodology is acceptable for determining the US-APWR pressure vessel fluence subject to the following:

(a) Because of the limited operating reactor data-base (viz., the single H. B. Robinson-2 capsule measurement) and the special features of the US-APWR (e.g., the neutron reflector), as recommended in RG 1.190 when the applicant's fluence methodology is applied to a specific US-APWR, the methods qualification must be updated by comparing the applicant's predictions with available capsule fluence measurements.

These comparisons must be incorporated into the operating reactor measurements data-base, and the methodology calculational biases and uncertainties must be updated as necessary. (RAI 04.03-50)

(b) The applicant's fluence analysis is based on assumed design and operating data (e.g., configuration geometry, fuel design/loading pattern, core power distribution, coolant temperatures). However, when the actual as-built and operating data become available, the US-APWR fluence predictions and uncertainty analysis must be updated and incorporated in the plant fluence analysis as necessary. (RAIs 04.03-51, 04.03-52, 04.03-54, 04.03-59)

A COL item has been added to address the above two issues.

4.3.7 References

1. NUREG-0800, Section 4.3, “Nuclear Design,” March 2007 (ML070740003).

2. WCAP-16045-P, "Qualification of the Two-Dimensional Transport Code PARAGON" (Proprietary) March 7, 2003 (ML030760081).

3. WCAP-10965-P-A, “ANC – A Westinghouse Advanced Nodal Computer Code,” (Proprietary) September, 1986.

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4.4 Thermal-Hydraulic Design

4.4.1 Introduction

The thermal-hydraulic design of the US-APWR must provide adequate heat transfer for reactor fuel and core components such that fuel damage does not occur during normal operation or AOO’s. The design must also ensure that the reactor can be safely shutdown 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 SAFDL are not exceeded.

4.4.2 Summary of Application

DCD Tier 1: The application identifies no Tier 1 items specifically related to reactor thermal-hydraulic design. However, Design Control Document, Tier 1, Section 2.4.2, “Reactor Coolant System,” Section 2.4.3, “Loose Parts,” Section 2.4.4, “Emergency Core Cooling System,” and Section 2.5.1, “Instrumentation and Controls” describe requirements associated with the thermal-hydraulic design.

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

For normal operation and AOOs, the design assures that there is at least a 95 percent probability, at a 95 percent confidence level, that DNB 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 Section 4.2.1, “Design Bases.” The results of these analyses are used in the evaluation of Chapter 15 events.

According to the design basis, a minimum of 91 percent of the thermal design flow will pass through the active fuel region of the core for fuel rod cooling; a maximum of nine percent is allotted as bypass flow. The bypass flow includes guide thimble cooling flow, reactor vessel head cooling flow, neutron reflector cooling flow, leakage to the core cavity, and leakage to the vessel outlet nozzle.

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

The core coolant flow distribution is calculated using the thermal hydraulics computer code VIPRE-01M [Reference 2], **(OI 4.4-2) which is undergoing review by the NRC in conjunction with its review of Topical Report MUAP-07009, "Thermal Design Methodology."**

VIPRE-01M [Reference 2] is the applicant's version of the VIPRE-01 [Reference 1] code, which is an approved subchannel analysis code developed by Battelle Pacific Northwest Laboratories to perform thermal-hydraulic analyses in reactor cores.

Using the original VIPRE-01 [Reference 1] code as the basis, the applicant incorporated added functions for more flexible design applications. VIPRE-01M [Reference 2] is used to evaluate reactor core thermal limits related to the minimum departure-from-nucleate-boiling ratio (DNBR), reactor core coolant conditions, and fuel temperature and heat flux in normal and off-normal conditions. The core inlet flow distribution calculation and the calculation of DNB have been validated by means of test data, as documented in Topical Report MUAP-07009-P [Reference 2], "Thermal Design Methodology."

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 to obtain conservative evaluations.

The flux tilt, that is, the azimuthal imbalance of the radial power distribution in the core, is addressed by the applicant in Section 4.4.2.10, "Flux Tilt Considerations." When the indicated azimuthal imbalance exceeds the limit, corrective action is required as provided in the TS, such as reducing core power level depending on the magnitude of the unexpected tilt.

The core-related safety limits, limiting safety system settings, and limiting conditions for operation are provided in the DCD, Tier 2, Chapter 16, "Technical Specifications."

The core power distribution is periodically monitored by using movable in-core neutron detectors and is constantly surveyed by fixed neutron ex-core detectors. The in-core instrumentation is top-mounted, and detectors are inserted through the in-core instrumentation guide tube to monitor the entire fuel assembly active length. The in-core detectors provide sufficient information to reconstruct detailed three-dimensional power distributions. The ex-core detectors provide on-line global axial and radial power distribution data and power changes, and provide input to automatic control functions. Thermocouples at the outlets of a subset of fuel assemblies also provide core performance data.

ITAAC: As discussed in Tier 1 above, no items explicitly related to reactor thermal-hydraulic design were identified. The following ITAAC identified in the DCD, Tier 1, however, relate to DCD, Tier 2 Section 4.4, "Thermal-Hydraulic Design," and are listed below for reference:

- DCD, Tier 1, Table 2.4.2-5, "Reactor Coolant System Inspections, Tests, Analyses, and Acceptance Criteria," Design Commitment 10.d, Reactor Coolant System flow

- DCD, Tier 1, Table 2.4.4-5, "Emergency Core Cooling System Inspections, Tests, Analyses, and Acceptance Criteria," Design Commitment 7.b, ECCS makeup, boration, and safety injection.

- DCD, Tier 1, Table 2.4.4-5, Design Commitment 7.d, Safety Injection Pump NPSH.

TS: The TS applicable to the reactor thermal-hydraulic design can be found in DCD, Tier 2, Chapter 16, Sections 2.0, 3.4, 3.5, 4.2, and B 2.0, B 3.2, and B 3.4.

4.4.3 Regulatory Basis

The relevant requirements of the Commission regulations for this area of review, and the associated acceptance criteria, are given in Section 4.4 of the SRP, NUREG-0800 [Ref. 4.4.10], henceforth SRP 4.4, and are summarized below. Review interfaces with other SRP sections also can be found in SRP 4.4.

1. 10 CFR Part 50, Appendix A, GDC 10 [Reference 3], as it relates to whether the design of the reactor core includes appropriate margin to assure that SAFDL are not exceeded during either normal operation or during AOO's.

2. GDC 12 [Reference 3], 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) [Reference 4], as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions.

4. 10 CFR 52.47(b)(1) [Reference 5], as it relates to proposed inspections, tests, analyses, and acceptance criteria to ensure the as-built plant is consistent with the certified design.

Acceptance criteria from SRP 4.4 adequate to meet the requirements of GDC 10 and GDC 12 [Reference 3] include:

1. There is at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB during normal operation or during AOOs. This includes consideration of uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculation methods used in the assessment of thermal margin.

2. Experimental or analytical demonstrations account for problems affecting fuel performance such as fuel densification, rod bow, core oscillations, and thermal-hydraulic instabilities.

3. A loose part monitoring system is provided, consistent with the requirements of RG 1.133, "Loose-Part Detection program for the Primary System of Light-Water-Cooled Reactors," [Reference 6].

4. TS are included to ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters, including safety limits and limiting safety settings for parameters, or combinations of parameters, to satisfy the fuel design acceptance criteria described above.

Acceptance criteria adequate to meet 10 CFR 50.34(f)(2)(xviii) [Reference 4] are contained in Three Mile Island Action Plan Item II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," [Reference 7] as related to instrumentation such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples.

Acceptance criteria adequate to meet initial testing guidelines are contained in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," [Reference 8] as related to the confirmation of the thermal hydraulic design.

4.4.4 Technical Evaluation

The staff has reviewed the US-APWR DCD, Tier 2, Section 4.4, "Thermal-Hydraulic Design," including applicable TS, related ITAAC and initial test requirements, and COL information items listed in DCD, Tier 2, Table 1.8-2, "Compilation of All Combined License Applicant Items for Chapter 1-19," to determine the acceptability of the design. An evaluation of the DCD, Tier 2, information was performed against the requirements of 10 CFR Part 50, Appendix A, GDCs 10 and 12 [Reference 3] pertaining to fuel thermal margin, 10 CFR 50.34(f)(2)(xviii) [Reference 4], as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions, and 10 CFR 52.47(b)(1) [Reference 5] as it pertains to inspections, tests, analyses, and acceptance criteria for the EC.

The DCD, Tier 1, sections identified in Section 4.4.2 above were also reviewed. The applicable information contained in the DCD, Tier 1, sections is confirmed to be consistent with the thermal-hydraulic design descriptions contained in DCD, Tier 2, Section 4.4. The ITAAC from the DCD Tier 1, listed in Section 4.4.2 above are applicable to the thermal-hydraulic design and were found to be acceptable by the staff.

Revision 1 of the DCD, Tier 2, Section 4.4.7, "Combined License Information," identified one COL information item, COL 4.4(1). The COL applicant is to confirm whether the design limits of minimum DNBR described in DCD, Tier 2, Section 4.4 are valid based on the relevant plant-specific instrumentation uncertainties, or the safety analysis limit of minimum DNBR value covers the new design limits of minimum DNBR and other DNBR penalties, such as rod bow penalty, transition core geometry and/or reserving more core operational flexibilities. In later revisions of the DCD this COL item was removed. In RAI 845-6116, Question 04.04-41, the staff requested additional information regarding what requirement would replace the COL item. In its response to letter UAP-HF-11379, dated November 11, 2011 [Reference 9], the applicant submitted a response to RAI 845 6116, Question 04.04-41 and stated that it would add the

requirement in DCD Section 4.4.5.4. Based on the applicant's response, RAI 845-6116, Question 04.04-41 is resolved pending verification of changes made to the DCD; this is **Confirmatory Item 4.4-2.**

The TS identified in Section 4.4.2 above were reviewed by the staff and found to be consistent with the descriptions and requirements provided in DCD, Tier 2, Section 4.4. The safety limits specified in TS 2.0 meet the requirements of GDC 10 and GDC 12 [Reference 3]. The limiting conditions for operation specified in TS 3.2 and TS 3.4 provide requirements pertaining to power distribution control and reactor coolant system, consistent with the requirements of GDC 10 and GDC 12 [Reference 3]. TS 4.2 specifies core-related design features consistent with the descriptions contained in DCD, Tier 2, Section 4.4.

The acceptance criteria used in this evaluation are contained in NUREG-0800 SRP 4.4 [Reference 10], "Thermal and Hydraulic Design," as summarized above in Section 4.4.3 of this SE.

Completeness of the content of the DCD, Tier 2, section was reviewed against SRP 4.4 "Thermal and Hydraulic Design" and the applicable portions of RG 1.206 [Reference 11], "Combined License Applications for Nuclear Power Plants."

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

In addition to the above-described evaluations, a comparison of the US-APWR thermal-hydraulic design, located in Table 4.1-1, "Comparison of Principle Reactor Design Parameters" of the DCD, with that of a previously approved PWR design is provided. The staff evaluated Table 4.1-1 and found that the US-APWR thermal-hydraulic design parameters are analogous with 14 ft (4.3 m) conventional PWR values listed as expected.

The thermal-hydraulic performance of the reactor under transient and accident conditions is demonstrated in DCD, Tier 2, Chapter 15, "Transient and Accident Analyses."

4.4.4.1 Design Bases

This section of the DCD, Tier 2 describes the bases for the thermal-hydraulic design, including applicability of GDC 10 and GDC 12 [Reference 3].

The applicant states that the objective of the thermal-hydraulic design is to provide adequate heat transfer for the fuel rods and control components such that: fuel damage is not expected to occur during normal operation and AOOs, except for a small number of fuel rod failures within the capability of the plant cleanup system; and 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.

This design objective is consistent with GDC 10 [Reference 3], which 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." DCD, Tier 2, Section 3.1.2, "Protection by Multiple Fission Product Barriers," provides a description of the reactor coolant, control, and protection systems that helps ensure

compliance with GDC 10 [Reference 3]. The effects of AOOs and postulated accidents on the SAFDLs are analyzed in DCD, Tier 2, Chapter 15.

The applicant further states that the hydrodynamic stability design basis in modes of normal operation and AOOs does not lead to instability events that could cause SAFDLs to be exceeded. This design basis is consistent with GDC 12 [Reference 3], which requires that "The reactor core and associated coolant, control, and protection systems shall be designed to assure 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."

The design basis for the DNBR is stated to be a 95-percent probability and 95-percent confidence level that DNB will not occur during the limiting transient. DNBR is evaluated in the DCD, Tier 2, Chapter 15, transient analyses, and the design basis is shown to be met.

The design basis for fuel temperature is stated to be a 95 percent probability and 95 percent confidence level that fuel centerline melt temperature will not be exceeded during normal operation and AOOs. The means for ensuring that this design basis is met is described in DCD, Tier 2, Section 4.4.1.1.1, "Design Basis." Fuel centerline melt analyses are performed to determine the local linear heat rates corresponding to fuel centerline melt, which are utilized to establish the limiting conditions for operation (LCOs) and reactor trip setpoints. Fuel centerline melt during AOOs is evaluated in DCD, Tier 2, Chapter 15. For example, DCD, Tier 2, Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," states that fuel centerline temperatures do not exceed the melting point.

The linear power density (LPD) LCO and high linear power density reactor trip function provide protection against fuel centerline melt and fuel cladding strain during normal operation and AOOs.

DCD, Tier 2, Section 4.4.1.3, "Core Flow," describes the core flow design basis in terms of active flow through the fuel region of the core required for fuel rod cooling and bypass flow for component cooling and leakage to the reactor vessel outlet nozzle. A minimum of 91 percent of the design flow is specified as active core flow and a maximum nine percent of the design flow is allocated to bypass flow.

The reactor coolant pumps are sized to meet or exceed the design flow requirements, as evaluated in DCD, Tier 2, Section 5.4.1.4, "Design Evaluation;" the associated DCD Tier 1, Section 2.4.1.2., specifies ITAAC 10.c and ITAAC 10.d for testing and analyzing reactor coolant pump flows. The staff finds the requirements of ITAAC 10.c and ITAAC 10.d to adequately address reactor coolant system flow verification. The initial test program described in DCD, Tier 2, Section 14.2, "Initial Plant Test Program," includes a Hot Functional Test for reactor coolant system flow verification, a Pre-critical Test for reactor coolant system temperature cross calibration, and a Pre-critical Test for post-core reactor coolant flow baseline.

Additional design bases for the fuel system, including material and mechanical properties, are evaluated under DCD Tier 2, Section 4.2.1.

Overall, the design bases for the thermal-hydraulic design described by the applicant in DCD, Tier 2, Section 4.4.1, meet GDC requirements and the guidelines of NUREG-0800 [Reference 10].

4.4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core

This section of the DCD, Tier 2, describes the thermal-hydraulic design of the reactor core, including calculations of DNBR and linear power density, reactor core void fraction, coolant flow, hydraulic loads, and pressure drop. A discussion of analysis uncertainties is also provided.

The applicant provides a calculated core minimum DNBR of 2.05 for nominal, rated power conditions, and a comparable DNBR LCO requirement of ≥1.35. These calculated DNBR values demonstrate adequate operating margin to the LCO.

The DNBR LCO provides protection against DNB-induced fuel rod cladding failure during normal operation and AOOs as evaluated under DCD, Tier 2, Chapter 15.

The DNBRs are calculated with Critical Heat Flux (CHF) correlations applicable to high thermal performance fuel assemblies used in the US-APWR. The correlations are described in the applicant's licensing Topical Report MUAP-07009P, "Thermal Design Methodology."

The staff is reviewing Topical Report MUAP-07009 [Reference 2] **(OI 4.4-2) to ensure that the WRB-1 and WRB-2 DNB correlations are applicable to US-APWR fuel with thermal mixing vane grids and fuel rod parameters that are the same as listed in DCD , Tier 2, Table 4.4-1, "Thermal-Hydraulic Comparison between US-APWR and Other Designs**."

DCD, Tier 2, Table 4.4-1, provides a comparison of the design parameters for the US-APWR with those of typical U.S. 4-loop plants (12-ft (3.6 m) and 14-ft (4.3 m) cores). The most notable differences include core power level, reactor coolant system loop flow, steam pressure and flow, and primary system volume.

Based on the reactor system design similarities and the staff's evaluation of the design differences, the staff is reviewing MUAP-07009-P to ensure there is reasonable assurance that the VIPRE-01M code and analysis methods are applicable to the US-APWR.

The applicant provides a calculated fuel rod average LPD of 4.65 kW/ft (15,866 Btu/hr-ft). The reported 4.65 kW/ft represents the fraction of heat generated within the fuel rod. The maximum allowable peak LPD is 12.1 kW/ft (41,285 Btu/hr-ft). The LPD data presented by the applicant appears reasonable by comparison to typical PWR data.

The total reactor 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 given in the DCD, Tier 2, is 32.1 psid. The pressure drop correlation presented by the applicant is consistent with standard textbook formulation (e.g., Tong, Weisman, "Thermal Analysis of Pressurized Water Reactors") was previously accepted for similar applications.

In RAI 377-2629, Question 04.04-3, the staff requested additional description of the calculation method, assumptions, and results on hydraulic loads on vessel components and fuel assemblies. In its response to letter UAP-HF-09336 [Reference 12] dated June 25, 2009, the applicant submitted its response to Question 04.04-3 and provided reference to MUAP-07016-P [Reference 13] for the analysis of hydraulic loads on the fuel, and reference to Technical Report MUAP-09004-P [Reference 14], "Summary of Stress Analysis Results for the US-APWR Core Support Structures," for the analysis of hydraulic loads for the core support structures. The staff reviewed the referenced methodology, MUAP-09004 and MUAP-07016, in UAP-HF-09336

and found the response acceptable based on acceptance limits less than the ASME code Section III. Therefore, RAI 377-2629, Question 04.04-03 is resolved.

Fuel rod surface convective heat transfer coefficients for bulk single phase and nucleate boiling conditions are described, with reference made to established industry-accepted documentation [Reference 21].

Uncertainties in the calculated estimates of the limiting thermal-hydraulic conditions are also described in this section of the DCD, Tier 2. Fuel temperature calculations are performed utilizing the VIPRE-01M [Reference 1] code. The applicant stated 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, but additional information was needed for the staff to complete its evaluation of uncertainties associated with fuel fabrication. [RAI 377-2629, Questions 04.04-02, 04.04-04, 04.04-05, and 04.04-06]

The applicant responded to RAI 377-2629, Question 04.04-02 in UAP-HF-09336 [Reference 12] dated June 25, 2009. However the staff needed additional clarification regarding the one percent rod bow penalty and a follow up RAI, **OI 4.4-1**, was asked**.** The applicant should explain why a rod bow penalty, δ_{bow}, of one percent is acceptable.

The staff in RAI 377-2629, Question 04.04-04 requested a tabulation of all the numerical uncertainties considered in the statistical evaluation of DNBR, since the applicant had previously provided a qualitative discussion of the uncertainties for DNB analysis. Based on the information provided in the applicant's response in letter UAP-HF-09546 [Reference 15] dated December 2, 2009, RAI 377-2629, 04.04-04 resolution is located in the SE for MUAP-07009.

The applicant responded to RAI 377-2629, Question 04.04-05 in UAP-HF-09336 [Reference 12] dated June 25, 2009, with a tabulation of all the numerical uncertainty values considered in the evaluation of fuel center-line temperature. The applicant also provided a discussion for not explicitly including the uncertainty of fuel cladding temperatures. Based on the applicant's response, the RAI 377-2629, Question 04.04 05 resolution is located in the SE for MUAP-07008.

The total vessel pressure drop estimates made at the design stage are primarily to assess reactor coolant system loop flows. Initial plant testing is described in DCD, Tier 2, Section 14.2. The SE of Section 14.2 will provide confirmation of thermal-hydraulic design parameters, including reactor coolant flow.

The 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 WRB-1 and WRB-2 critical heat flux correlations, including their data bases, were previously reviewed and found to be acceptable as documented in the staff's SE of Westinghouse Electric Corporation licensing topical reports WCAP-8762-P-A [Reference 16] and WCAP-10444-P-A [Reference 17].

The applicant utilized the Revised Thermal Design Procedure document (Reference 3 of the DCD) in the DNBR calculation method, which has been previously approved by the NRC staff.

The uncertainties in the process parameters used in the DNBR calculations are also described in the "Revised Thermal Design Procedure" document. The applicant states that a statistical

combination of uncertainties is utilized to treat the random nature uncertainties, while a deterministic approach is used for others. The referenced licensing Topical Report MUAP-07009-P [Reference 2], provides added description of the uncertainties used in the DNBR calculations. The applicability of WRB-2 and WRB-1 are located in the SE for MUAP-07009.

The effects of crud on DNBR and reactor system pressure drop are addressed in Topical Report MUAP-07009-P [Reference 2].

Although the VIPRE-01M [Reference 1] code does not consider the effects of crud buildup on fuel rods, sensitivity studies described in Appendix A of VIPRE-01M demonstrate conservative predictions by the VIPRE-01M [Reference 1] code compared to other codes which do consider the effect of crud buildup.

Core azimuthal imbalance is addressed through a bounding LCO, as provided for in DCD, Tier 2, Chapter 16.

4.4.4.3 Description of Thermal and Hydraulic Design of the Reactor Coolant System

This section of the DCD, Tier 2, describes the thermal and hydraulic design of the reactor coolant system, including an overall description of the plant configuration, reactor coolant pump operating restrictions, reactor coolant temperature-power regime, and load following characteristics.

The applicant provides reference to various DCD, Tier 2, sections where the plant reactor coolant system and emergency core cooling system information may be found. This design information is evaluated under the respective DCD, Tier 2 section.

In Section 4.4.3.2, “Operating Restrictions on Pumps,” the applicant states that net positive suction head requirements for reactor coolant pump operation will be met. DCD, Tier 1, specifies ITAAC 10.d for testing and analyzing reactor coolant pump flows. Reactor coolant pump operation is verified during initial plant testing by “Hot Functional Test, Reactor Coolant System Flow Model Verification, RCS Temperature Cross Calibration,” and Reactor Coolant System Flow Baseline tests described in DCD, Tier 2, Section 14.2, “Initial Plant Test Program.”

The effects of reduced core flow from inoperable pumps caused by unplanned events are described in Chapter 15 of the DCD, Tier 2. DCD, Tier 1, Table 2.4.2-5, ITAAC 10.c requires tests will be performed to determine the RCP flow coastdown.

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

4.4.4.4 Technical Evaluation

This section of the DCD, Tier 2, provides the applicant’s evaluation of the thermal-hydraulic design of the reactor relative to critical heat flux, core hydraulics, core power distributions, core thermal response, and analytical methods.

The applicant describes the two hot channel factors used for the US-APWR 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 staff finds the treatment of fuel parameter variations and tolerances relative to the hot channel factors as described by the applicant to be consistent with other PWR applications.

In RAI 378-2672, Question 04.04-07, the staff asked if DCD Reference 9 was applicable to US-APWR fuel and why no F_Q^E penalty is needed to conservatively calculate DNBR. The applicant responded in UAP-HF-09337 [Reference 18] dated June 25, 2009, that the factor $FdelatH^E$, $_1$ is used to account for overall rod power uncertainty due to manufacturing deviations and no penalty on F_Q^E is needed. The applicant further pointed out that small local power spikes caused by F_Q^E have a negligible effect on DNBR calculations. The staff agrees as tests documented in Reference 9 show that the calculated DNBR is negligibly affected as it was within the repeatability of the DNB measurements and is applicable to US-APWR fuel as both designs have mixing vane grids. The test results in Reference 9 were performed as significant fuel densification could occur based on then current manufacturing processes. With present day manufacturing processes fuel densification is much less an issue with theoretical pellet densities equal to or greater than 95 percent. As such, the effect on local heat flux due to fuel densification will be significantly less than the 20 percent increase used to evaluate DNBR in DCD Reference 9. Therefore, the response to RAI 378-2672, Question 04.04-07 is acceptable and this question is closed.

However, the staff requested justification for the specific values utilized for US-APWR fuel in RAI 378-2672, Question 04.04-07. The applicant responded, in UAP-HF-09337 [Reference 18] dated June 25, 2009, to RAI 378-2672, Question 04.04-07 with a justification for the specific values used for US-APWR fuel and references to applicable documents. The staff's evaluation of F_Q has been evaluated in MUAP-07009.

The core hydraulics evaluation provided in the DCD, Tier 2, addresses inlet flow distribution, vessel flow paths, and the empirical friction factors used in the analyses. Inlet flow distribution is evaluated in Section 4.4.4.2 above. The vessel bypass flow paths considered in the thermal-hydraulic evaluation are vessel-to-upper internals gap flow to the upper head, vessel-to-core barrel gap flow into the hot leg, fuel assembly thimble flow, and flow between heavy reflector and core barrel. The maximum total bypass flow is calculated to be nine percent of the total reactor coolant flow. These design characteristics are consistent with similar PWR designs.

4.4.4.5 Analytical Methods

This section of the DCD, Tier 2, provides a description of the analytical methods used in the thermal-hydraulic analyses.

The applicant states that standard steady-state hydraulics calculations are utilized to arrive at predictions of the reactor coolant system flows. Hydraulic resistances of the vessel, excluding the core, are determined from scale model testing. Form and friction losses through the core are determined from hydraulic testing of fuel assemblies.

DCD , Tier 2, Table 4.4.-1 lists three calculated reactor coolant system flows, as follows: the minimum measured flow rate (MMF), given as 460,000 gpm (1,741,285 Liters/Minute), for the

actual plant conditions; the vessel thermal design flow rate, given as 448,000 gpm (1,695,860 L/M), is a conservatively low flow for the core cooling analysis; and the vessel effective flow rate, given as 407,680 gpm (1,543,233 L/M), is a conservatively high design for the mechanical analysis of the reactor coolant system. The staff notes that these vessel volumetric flows are consistent with the reactor coolant system parameters listed in DCD, Tier 2, Table 5.1-1, "Reactor Coolant System Design and Operating Parameters." DCD, Tier 1, Section 5.1.1 describes the bases for determining these three reactor coolant flow parameters.

ITAAC 10.d requires that tests and analyses to measure RCS flow with the as-built four reactor coolant pumps operating at no-load RCS pressure and temperature conditions will be performed.

DCD, Tier 2, Section 14.2, describes the testing that will be performed to confirm the bounds of the reactor system flows. The staff finds the use of best estimate and bounding reactor coolant system flows and the plans for testing acceptable.

TS 3.4 provides LCOs applicable to reactor coolant system performance which are evaluated under DCD, Tier 2, Chapter 16.

The core analysis description provided in DCD, Tier 2, Section 4.4.4.5.2, summarizes the application of the VIPRE-01M computer code and analysis uncertainties. The applicant states that the DNBR calculations account for measurement uncertainties or 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 uncertainties described by the applicant meets the acceptance criteria listed in Section 4.4.3 above.

This section of the DCD, Tier 2, also addresses thermo-dynamic stability. The applicant provides a qualitative engineering basis for concluding that the US-APWR is not susceptible to thermo-dynamic instability, including core configuration, subcooled and high pressure operation, and transient fuel pellet heat transfer. DCD, Tier 2, Section 4.3.2.7, "Stability," provides additional analysis of thermo-dynamic stability. The staff finds that the NUREG-0800 SRP [Reference 10] Section 4.4 acceptance criterion pertaining to core oscillations and thermal-hydraulic instability is met.

The effects of flow blockage on fuel rod behavior are addressed in this section of the DCD, 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 debris filter, and the lower spacer grids. The staff concurs with the applicant's conclusions that only 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 stated in Section 4.4.4.1 above.

The staff reviewed the downstream in-vessel effects of a LOCA-generated debris blockage in Section 6.3 of the DCD, "ECCS."

The staff concludes that the content of DCD, Tier 2, Section 4.4.4.5 meets the thermo-hydraulic methods acceptance criteria of NUREG-0800 [Reference 10] SRP Section 4.4. The staff notes, **(OI 4.4-2)** however, that licensing Topical Report MUAP-07009-P [Reference 2], "Thermal Design Methodology," referenced by the applicant, has not yet been approved by the staff.

4.4.4.6 Testing and Verification

In this section of the DCD, Tier 2, the applicant identifies the initial plant testing pertaining to the thermal-hydraulic design. Reference is made to DCD, Tier 2, Section 14.2, for reactor coolant system testing and for core power distribution measurement.

The staff has evaluated the initial tests described in DCD, Tier 2, Section 14.2, against the thermal-hydraulic design described in DCD, Tier 2, Section 4.4, and to the requirements of 10 CFR 52.47(b)(1) [Reference 7] and the guidance contained in RG 1.68 [Reference 8].

The staff finds that the testing identified in DCD, 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.

The applicant states that testing and inspection of fuel assembly components are performed following fabrication, as described in DCD, Tier 2, Section 4.2.4.3, "Onsite Inspection." Section 4.2.4.3 also provides discussion pertaining to new fuel testing and inspection relevant to the thermal-hydraulic performance of the fuel.

The testing and verification program described in DCD, Tier 2, Section 4.4.5 is pending acceptable resolution of DCD Section 14.2 RAIs, **(OI 4.4-3)**.

4.4.4.7 Instrumentation Requirements

This section of the DCD, Tier 2, describes the instrumentation requirements pertaining to the thermal-hydraulic design of the US-APWR.

Both ex-core and in-core instrumentation are used to measure the power distribution and quadrant power tilts, and the in-core instrumentation system (ICIS) measurements are compared to predictions. The widely-used ANC [Reference 19] computer code is used for the generation of predicted values. The in-core instrumentation consists of movable neutron detectors (MDs) and core exit thermocouples (TCs).

There are three power ranges of detectors; source-range, intermediate-range, and power-range. These ex-core neutron flux detectors are used to limit the maximum power output of the reactor within their respective power ranges. The detectors are also used to monitor neutron flux from a completely shutdown condition to the overpower condition at 120 percent. The power range channels are intended to protect the core against the consequences of rod ejection accidents, to protect the core against the consequences of adverse power distributions resulting from dropped control rods, to provide the control function of rod movements, and to alert the operator to excessive quadrant power tilt.

TS 2.0 and TS 3.2 provide the safety limits and LCOs for DNBR and LPD which are evaluated under DCD, Tier 2, Chapter 16. The staff has reviewed the bases for these TS and finds them consistent with the functions described in DCD, Tier 2, Section 4.4. The applicant also describes the requirements for a loose parts monitoring system.

The loose parts monitoring system 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 in DCD, Tier 2, Section 14.2.

The applicant states that US-APWR loose parts monitoring system conforms to the guidance of RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." The loose parts monitoring system is evaluated under DCD, Tier 2, Section 7.1.

For the monitoring of the post-accident inadequate core cooling (ICC), the degree of subcooling, Reactor Vessel water level, and core exit temperature will be measured as described in DCD, Tier 2, Section 4.4.6.4, "Inadequate Core Cooling," and Section 7.5, "Information Systems Important to Safety."

Monitoring the degree of liquid subcooling indicates loss of subcooling, occurrence of saturation and recovering to subcooled condition. The Reactor Vessel water level provides information on the decreasing liquid inventory in the reactor. The core exit temperature indicates the increasing steam temperatures associated with inadequate core cooling and the decreasing steam temperatures associated with recovery from inadequate core cooling. The staff concluded that intent of SRP [Reference 10], Section 4.4 has been met.

Initial testing of the instrumentation described in DCD, Tier 2, Section 4.4, is evaluated under Section 4.4.4.6 above.

4.4.4.8 Confirmatory Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCD:

Confirmatory Item 4.4-1: RAI 845-6116. In Revision 2 of Section 4.4.7 of the DCD, the applicant removed a COL item that was in Revision 1. Given the importance of verifying that the design limits and the safety analysis limit of minimum DNBR described in Section 4.4 will remain valid for the plant-specific instrumentation uncertainties using the revised thermal design procedure (RTDP) methodology [Reference 2] prior to fuel loading, the applicant agreed to modify the DCD Section 4.4.5.4 to include wording which would evaluate the minimum DNBR limit used in the safety analysis based on actual plant instrument uncertainties.

Confirmatory Item 4.4-2: Based on the applicant's response, RAI 845 6116, Question 04.04-41 is resolved pending verification of changes made to the DCD.

4.4.4.9 Open Items

OI 4.4-1: RAI 952-6333, Question 04.04-42 Applicant should explain the δ_{bow} calculation and why the penalty is appropriate for less than one percent rod bow.

OI 4.4-2: MUAP-07009-P [Reference 2] has not been approved by the staff (of which some of the calculations in Section 4.4 are based on).

OI 4.4-3: DCD, Tier 2, Section 4.4.5 is pending acceptable resolution of DCD Section 14.2 RAIs.

4.4.5 Combined License Information Items

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

4.4.6 Conclusions

As a result of the open and confirmatory items, the staff is unable to finalize its conclusion on Section 4.4 in accordance with the requirements of NRC regulations.

4.4.7 References

1 Stewart, C. W., et al., VIPRE-01: A Thermal-Hydraulic Code for Reactor Core, Volume 1-4 (Revision 4, February 2001), Volume 5 (March 1988), NP-2511-CCM-A.

2 Thermal Design Methodology, MUAP-07009-P (Proprietary) and MUAP-07009-NP (Non-Proprietary) May 2007.

3 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, *Code of Federal Regulations*, 10 CFR Part 50, Appendix A.

4 NRC Regulations Title 10, *Code of Federal Regulations*, 10 CFR 50.34f (2).

5 NRC Regulations Title 10, *Code of Federal Regulations*, 10 CFR 52.47 (b).

6 Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors, NRC Regulatory Guide 1.133, November 2009.

7 TMI Action Plan Item II.F.2, NUREG-0737, December, 2011.

8 Initial Test Programs for Water-Cooled Nuclear Power Plants, NRC Regulatory Guide 1.68, March 2007.

9 UAP-HF-11379, “MHI’s Response to US-APWR DCD RAI No. 845-6116,” November 2011.

10 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 4.4, March 2007.

11 Combined License Application for Nuclear Power Plants, NRC Regulatory Guide 1.206, November 2009.

12 UAP-HF-09336, “MHI’s Response to US-APWR DCD RAI No. 377-2629,” June 2009.

13 US-APWR Fuel System Design Evaluation, MUAP-07016-P Rev.3 (Proprietary) and MUAP-07016-NP Rev.3 (Non-Proprietary), August 2010.

14 Summary of Stress Analysis Results for the US-APWR Core Support Structures, MUAP-09004-P (Proprietary) and MUAP-09004-NP (Non-Proprietary) March 2009.

15 UAP-HF-09546, “MHI’s Amended Response to US-APWR DCD RAI No. 377-2629,” December 2009.

16 Motley, F. E., Cadek, F. F., New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids, WCAP-8762, Addendum 1-P-A (Proprietary) (Non-Proprietary), July 1984.

17 Davidson, S. L. and Kramer, W. R. (Ed.), Reference Core Report VANTAGE 5 Fuel Assembly, WCAP-10444-P-A (Proprietary) and WCAP-10445-NP-A (Non-Proprietary), September 1985.

18 UAP-HF-09337, “MHI’s Response to US-APWR DCD RAI No. 378-2672,” June 2009.

19 Liu, Y.S., et al., ANC - A Westinghouse Advanced Nodal Computer Code, WCAP-10965-P-A (Proprietary), and WCAP-10966-A (Non-Proprietary), September 1986.

20 Hydraulic Test of the Full Scale US-APWR Fuel Assembly, MUAP-11017-P Rev.0 (Proprietary), May 2011.

21 El-Wakil, M.M., Nuclear Heat Transport, 1971

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.1 Introduction

This section of the DCD describes the materials used in the control rod drive mechanisms (CRDMs) for both the RCS pressure boundary portion of the CRDM, and non-pressure boundary CRDM components.

4.5.1.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1 Section 2.4.1, “Reactor System.”

DCD Tier 2: The applicant has provided a Tier 2 design description in DCD Section 4.5.1, “Control Rod Drive System Structural Materials,” 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.

Material specifications for CRDM components are listed in Table 4.5-1, “Summary of Control Rod Drive System Structural Materials.” The materials used to fabricate the CRDM pressure housing are solution annealed austenitic stainless steel and austenitic stainless steel weld filler material. CRDM pressure housing materials meet ASME Code Section III requirements for Class 1 components. 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.”

CRDM non-pressure boundary components that are in contact with reactor coolant include solution annealed austenitic stainless steel, quenched and tempered martensitic stainless steel, solution annealed and thermally aged nickel base alloy and cobalt-nickel-chromium-tungsten alloy. Some latch assembly components are clad with cobalt alloy, chrome plated or coated with chrome carbide to resist wear. All materials used in the CRDMs are selected for their compatibility with the reactor coolant water, as described in NB-2160 and NB-3120 of the ASME Code, Section III. The proposed materials have been successfully used in operating plants in the U.S. and Japan. The manufacturing and process controls for preventing intergranular corrosion of stainless steel components follow the guidance provided in RG 1.44, "Control of the Use of Sensitized Stainless Steel."

Cleaning and cleanliness control comply with the description provided in DCD Subsection 5.2.3.4.1, "Cleaning and Contamination Protection." Onsite cleaning and cleanliness control will be carried out in accordance with the requirements of ASME NQA-1 and follow the guidance of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

ITAAC: The ITAAC associated with DCD Tier 2, Subsection 4.5.1 are discussed in DCD Tier 2, Subsection 14.3.4.4, "ITAAC for Reactor Systems," and delineated in DCD Tier 1, Subsection 2.4, Items 5, 6 and 7 in Table 2.4.1-2, "Reactor System Inspections, Tests, Analyses, and Acceptance Criteria."

COL information or action items: There are no COL information items or action items for this area of review.

4.5.1.3 Regulatory Basis

The relevant requirements of the Commission's 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 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 (SSCs) 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 Boiler and Pressure Vessel (B&PV) Code. Application of 10 CFR 50.55a and GDC 1, "Quality Standards and Records," to the control rod drive structural materials provides assurance that the control rod drive system will perform as designed.

2. GDC 14, "Reactor Coolant Pressure Boundary," 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.”

2. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

3. RG 1.44, "Control of the Use of Sensitized Stainless Steel."

4. 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 DCD Tier 2, Section 4.5.1 “Control Rod Drive System Structural Materials,” 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 DCD Section 4.5.1, to determine the suitability for service of the materials selected for CRDM structural components. 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 for the pressure housing. The pressure housing consists of the rod travel housing and the latch housing that are butt welded. DCD Revision 3, Table 4.5-1, “Summary of Control Rod Drive System Structural Materials,” lists SA-182 Grade F316 and Grade F316LN stainless steel as the materials used to fabricate the pressure housing. The staff reviewed the specification 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.” The staff verified that the materials identified by the applicant are acceptable materials for use in ASME Code, Section III, Class 1 systems and that they are included in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B.

Based on the above review, 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 the requirements of GDC 1 and 10 CFR 50.55a.

Compliance with the requirements of GDC 26 as it relates 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 do not perform a pressure retaining function must also be fabricated from materials that will assure that they function reliably in order to meet the requirements of GDC 26. Non-pressure retaining CRDM component materials exposed to reactor coolant include SA-479 Type 410, Alloy X-750 (SB637 N07750), SA-268 TP410, SA-479 Type 403 and SA-479

Type 304 materials. The aforementioned materials specifications and grades are listed in ASME Code, Section II, Part D, Subpart 1,Tables 2A and 2B, and are therefore acceptable materials specifications for use in non-pressure boundary applications. In addition, these materials are commonly used in currently operating plants and have a successful operating history. Haynes 25 and Stellite 6 are also listed for use in non-RCPB portions of the CRDMs; however, equivalent materials may be used. Procurement specifications will require the same chemistry and material properties for equivalent materials as those for Haynes 25 and Stellite 6. The use of Stellite 6 and Haynes 25 materials is acceptable because these materials are commonly used for CRDM components in operating plants and have a successful operating history. The use of equivalent materials is acceptable because equivalent materials will have the same chemistry and material properties as the preferred materials. The staff determined that the materials listed above are of sufficient quality, have reliably performed as CRDM components and have a successful operating history. In view of the foregoing, the staff determined that the materials and materials specification for the materials used in the non-pressure retaining CRDM components are acceptable and meet GDC 1, GDC 14, GDC 26, and 10 CFR 50.55a.

Austenitic Stainless Steel Components

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. DCD Section 4.5.1.2 states that austenitic stainless steel base materials for CRDM applications are used in the solution annealed condition and are not heated above 427°C (800°F) after solution annealing, other than locally by welding operations. Using solution annealed materials prevents stress corrosion cracking and is consistent with RG 1.44. Table 4.5-1 lists SA-182 Grade F316 and Grade F316LN stainless steels as materials for the fabrication of the pressure housing. Grade F316LN has a maximum carbon content of 0.03 percent, meets the recommendations of RG 1.44 and is, therefore, acceptable. SA-182 Grade F316 has a maximum carbon content of 0.08 percent; however, Table 4.5.1-1 shows that if Grade F316 is used, the maximum carbon content will be controlled under 0.05 percent heat analysis and 0.06 percent product analysis. RG 1.44, Section 4, recommends that low carbon grade (≤ 0.03 percent) austenitic stainless steels be used unless the controlled concentration of dissolved oxygen has a limiting value of 0.10 ppm to prevent intergranular stress corrosion cracking (IGSCC). DCD Section 4.5.1.1 states that the dissolved oxygen level inside the CRDM pressure housing is the same as the primary coolant shortly after startup and a stagnant primary coolant environment that could result in dissolved oxygen content greater than 0.10 ppm during plant operation does not exist in any portion of the CRDM. The staff notes that the standard value of dissolved oxygen of reactor coolant water is less than or equal to 0.005 ppm as listed in DCD Table 5.2.3-2, “Recommended Reactor Coolant Water Chemistry Specification.” The staff finds this acceptable because stagnant flow conditions will not be present in the CRDM, thus making the use of standard grades of stainless steel acceptable. In addition, the staff notes that as a result of the applicant’s compliance with the recommendations in RG 1.44, the applicant will perform intergranular corrosion tests for each welding procedure to be used for welding material having carbon content greater than 0.03 percent, thus providing additional assurance that materials will not be susceptible to stress corrosion cracking.

The applicant stated, in DCD Section 4.5.1.2, “Austenitic Stainless Steel Components,” that manufacturing process controls for preventing intergranular corrosion of stainless steel components are used in accordance with the guidance in RG 1.44. The applicant also stated that furnace sensitized material is not allowed and methods described in RG 1.44 are followed for cleaning and protecting austenitic stainless steels from contamination during handling,

storage, testing, and fabrication and for determining the degree of sensitization during welding. The staff finds this acceptable because the applicant will follow the guidance in RG 1.44 to reduce the susceptibility of components to stress corrosion cracking.

The staff requested, as part of RAI 457-3305, Question 04.05.01-10, that the applicant list DCD Section 4.5.1 in DCD Table 1.9.1-1, "US-APWR Conformance with Division 1 Regulatory Guides," under the line item for RG 1.44 to provide clarity. In its response dated October 30, 2009, the applicant provided a proposed revision to Table 1.9.1-1, which lists DCD Section 4.5.1 under the line item for RG 1.44. The staff will verify that the appropriate modifications are made to Table 1.9.1-1 in DCD Revision 4. **[Confirmatory Item 04.05.01-10]**

Cold working can increase the susceptibility of stress corrosion cracking in austenitic stainless steels. DCD Subsection 4.5.1.1, "Material Specifications," states that strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.

RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," contains staff guidance pertaining to the delta ferrite content in austenitic stainless steel welds to minimize the presence of microfissures, which could have an adverse effect on the integrity of components. RG 1.31 recommends a minimum delta ferrite level of 5 FN (ferrite number). DCD Subsection 4.5.1.2, "Austenitic Stainless Steel Components," states that weld metals for the pressure boundary and non-pressure boundary CRDM components are used in accordance with the guidance in RG 1.31 and the minimum acceptable delta ferrite is 5 FN. In addition, the staff notes that DCD Table 1.9.1-1 identifies RG 1.31 as being applicable to DCD Section 4.5.1 with no exceptions. The staff finds this acceptable because the applicant will follow the guidance in RG 1.31 to minimize the presence of microfissures in austenitic stainless steel welds.

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, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," provides guidance by the staff on acceptable methods to control surface contamination caused by abrasive operations such as grinding. Since DCD Section 4.5.1.2 states that process controls during abrasive work are consistent with the guidelines of RG 1.37, the staff finds this to be acceptable.

Other Materials

Materials other than austenitic stainless steels are used to fabricate non-pressure boundary CRDM components. These materials include Type 410 martensitic stainless steel, cobalt-chromium alloy (Haynes 25), nickel based alloy (Alloy X-750), cobalt based material (Stellite No. 6), and Type 403 martensitic stainless steel. Latch assembly sliding surfaces are chrome plated. Latch assembly latch arm tips are coated with chrome carbide.

The DCD Sections 4.5.1.1.2, "Latch Assembly," and 4.5.1.1.3, "Drive Rod Assembly," identify that portions of the latch assembly and drive rod assembly are fabricated from Type 410 martensitic stainless, which is annealed where strength is not an issue. Using annealed Type 410 improves magnetic properties and reduces the risk of stress corrosion cracking and is therefore acceptable. For those Type 410 components that are quenched and tempered, the applicant states, in Subsection 4.5.1.1, "Material Specifications," that the tempering temperature is over 565°C (1050°F). The staff finds this acceptable because utilizing a tempering temperature over 565°C (1050°F) prevents temper embrittlement and is consistent with the guidance provided in SRP Section 4.5.1.

The drive rod assembly coupling is fabricated from Type 403 martensitic stainless steel. The applicant did not provide any information related to the final condition of Type 403. The staff requested, in RAI 268-2181, Question 04.05.01-5, that the applicant provide its heat treatment requirements, including a basis for its requirements, for Type 403 martensitic stainless steel and modify the DCD accordingly. In its response dated May 1, 2009, the applicant provided its proprietary heat treatment for Type 403 martensitic stainless steel. The staff reviewed the applicant's heat treatment requirements and finds that they are sufficient to preclude material conditions that could promote stress corrosion cracking and therefore finds them acceptable.

Latch assembly latch arm tips and pin holes are clad with cobalt alloy Stellite No. 6 and the drive rod locking button and latch assembly pins are fabricated from cobalt based alloy Haynes 25. Radiation buildup during plant operation can occur because of cobalt-60, which forms by neutron activation of cobalt-59. 10 CFR 20.1406, "Minimization of Contamination," requires, in part, that applicants for standard design certification describe how facility design will minimize, to the extent practical, contamination of the facility and the environment, and minimize, to the extent practical, the generation of radioactive waste. The staff understands that in some cases, the use of cobalt based materials is unavoidable due to service conditions and requirements. The use of Stellite No. 6 is typical for these types of components in various US and foreign reactor designs that have been approved by the staff. Haynes 25 is a cobalt-nickel-chromium-tungsten alloy that has been successfully used in U.S. and foreign reactor designs and has had a favorable operating history. Given the limited application of these materials in the US-APWR CRDM and the satisfactory performance of these materials in currently operating plants, the staff finds the applicant's use of Stellite No. 6 and Haynes 25 acceptable. A detailed description of the applicant's compliance with 10 CFR 20.1406, "Minimization of Contamination," is located in final safety evaluation report (FSER) Section 12.3, "Radiation Protection Design Features."

Alloy X-750 (SA-637 N07750) is used for latch assembly and drive rod assembly springs. The staff notes that the resistance of Alloy X-750 to stress corrosion cracking (SCC) is dependent on adequate processing and heat treatment. Table 4.5-1, "Summary of Control Rod Drive System Structural Materials," indicates that additional requirements of MIL-S-23192 will be applied. In its response to RAI 268-2181, Question 04.05.01-6, which requested additional information regarding the thermal processing of Alloy X-750, dated May 1, 2009, the applicant stated that solution heat treatment and thermal aging will be performed in accordance with MIL-S-23192 Class A for coil springs in the latch assembly and Class D for coil springs in the drive rod. In addition, the applicant stated that springs fabricated to MIL-S-23192 have no failure experience in Japan. The staff reviewed the applicable processing and heat treatment specifications described in MIL-S-23192 and finds them acceptable because the final material condition will provide adequate resistance to SCC. In addition, the favorable operating history of these materials in Japanese plants provides further assurance that these materials will not be susceptible to SCC.

Compatibility of Materials with Reactor Coolant

Materials selected for use in the CRDM must be compatible with the reactor coolant, as described in Subsubarticles NB-2160 and NB-3120 of the ASME Code, Section III. In the DCD, the applicant indicates that the materials used in the CRDM are compatible with the reactor coolant and, thus, comply with the ASME Code, Subsubarticles NB-2160 and NB 3120.

Furthermore, the materials selected for the CRDM are currently used in nuclear power plants, and have performed satisfactory under environmental conditions found in these operating plants. On this basis, the staff finds the selected materials acceptable.

Cleaning and Cleanliness Controls

Cleaning and cleanliness controls for the CRDM during manufacture and assembly are discussed in DCD Subsection 4.5.1.4, "Cleaning and Cleanliness Control." DCD Subsection 4.5.1.4 also references Subsection 5.2.3.4.1, "Cleaning and Contamination Protection," which provides additional information and is also applicable to the CRDMs. 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 imposes strict process controls for cleaning and protection against contamination of materials during all stages of component manufacture and installation. Exposure to contaminants, such as low melting point materials, chlorides and fluorides, and their compounds are controlled during manufacture and installation. Water used for final cleaning or flushing is demineralized water with chloride and fluoride concentrations less than 0.15 ppm. Tools used in abrasive work on austenitic stainless steel, such as grinding or brushing do not contain and are not contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or stress corrosion cracking. In addition, the applicant states, in DCD Subsection 4.5.1.4, that onsite cleaning and cleanliness control will be carried out in accordance with ASME NQA-1 and provisions of RG-1.37. On this basis, the staff finds the applicant's on-site cleaning and cleanliness controls for CRDM components acceptable.

ITAAC

The ITAAC associated with DCD Tier 2, Section 4.5.1 are discussed in DCD Tier 2, Section 14.3.4.3 and delineated in DCD Tier 1, Section 2.4. Items 5, 6 and 7 in Table 2.4.1-2, "Reactor System Inspections, Tests, Analyses, and Acceptance Criteria." The staff's evaluation of ITAAC associated with CRDMs are located in FSER Section 14.3.4.

4.5.1.5 Combined License Information Items

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

4.5.1.6 Conclusions

As a result of the confirmatory item the staff is unable to finalize its conclusion on Section 4.5.1 in accordance with the requirements of NRC regulations.

4.5.2 Reactor Internal and Core Support Structure Materials

4.5.2.1 Introduction

This section of the DCD 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. The staff's review also verifies that its positions regarding fabrication and heat treatment practices meet the requirements of the ASME Boiler and Pressure Vessel Code.

4.5.2.2 Summary of Application

DCD Tier 1: There are several Tier 1 requirements that are applicable to reactor internals and core support materials. These are listed in Table 2.4.1-2, "Reactor System Inspections, Tests, Analyses, and Acceptance Criteria." Item 5.a relates to the fabrication, installation, and inspection of ASME Code System III components. Item 5.b relates to the design of ASME Code System III components. Item 8 relates to Seismic Category I which Table 2.4.1-1, "Equipment Key Attributes," identifies to include core support structures. Item 9 relates to the design of the reactor internals to withstand the effects of flow-induced vibration.

DCD Tier 2: The applicant has provided a Tier 2 system description in 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 DCD. The DCD states that materials used in reactor internals and core support structures comply with the ASME Code, Section II supplemented by ASME Code Case N-4, as approved in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." The DCD also states that the welding controls specified for austenitic stainless steel reactor coolant pressure boundary components that are described in DCD 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 DCD Section 5.2.3 to minimize susceptibility to intergranular stress corrosion cracking are also applicable to austenitic stainless steels used for reactor vessel internals and core support structures.

ITAAC: Table 2.4.1-2 of Tier 1 contains ITAAC that are applicable to reactor vessel internals and core support structures. Entries in this table pertinent to reactor vessel internals and core support structures are Items 5.a, 5.b, 8, and 9. These ITAAC are related to verifying the design, fabrication and inspection of ASME Code System III requirements, the design of Seismic Category I equipment and the design of internals to withstand flow-induced vibration.

TS: There are no TS for this area of review.

4.5.2.3 Regulatory Basis

The relevant requirements of the Commission 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, "Codes and Standards," which requires that SSCs 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, "Quality Standards and Records," 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 US-APWR DCD Section 4.5.2 describing 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

US-APWR DCD, Section 4.5.2.1 and Table 4.5-2, "Summary of Reactor Internals and Core Support Materials," provide the material specifications for the core support structures and reactor vessel internal components. The core support and reactor vessel internal materials consist mostly of austenitic stainless steels, Grade/Type 304, 304L and 316. Strain-hardened Type 316 material with chrome plated threads is used for the neutron reflector tie rods. Type 304 material with Stellite 6 (cobalt alloy) cladding is used for alignment pins and clevis inserts. Other materials used are Type 403 Modified stainless steel (martensitic) for hold-down spring, nickel alloy X-750 for radial support clevis insert bolts and spring (and as an option for the guide tube support pins), and nickel alloy 690 for the radial support clevis inserts. ASME Code Case N-4-11, which is approved by RG 1.84, is applied for the AISI Type 403 Modified hold-down spring. All other materials meet the requirements of ASME Code Sections II and III. The staff requested the applicant provide answers to the following questions to clarify or correct the materials information summarized in Table 4.5-2 of the DCD. This was necessary to make a determination of the acceptability of materials used for reactor internals and core support structures in accordance with the requirements of 10 CFR 50.55a and 10 CFR Part 50 Appendix A GDC 1.

To correctly specify the material used for welding, the staff requested in RAI 269-2155, Question 04.05.02-1 that the weld material type be listed in Table 4.5-2. The applicant adequately responded to the staff's RAI by revising Table 4.5-2 of the DCD to include the weld material type.

Since the materials for the neutron reflector components, other than the tie rods, were not explicitly identified in the DCD, the staff requested additional information in RAI 269-2155, Question 04.05.02-2a and RAI 414-3102, Question 04.05.02-6. The applicant's response, dated August 7, 2009, documents that Type 304 stainless steel (specification SA-336 Grade F304) will be used for the neutron reflector ring blocks. The staff finds the applicant response to these questions acceptable because this material meets ASME Code Section II materials specifications and is appropriate for this application.

The staff requested additional clarifications to the reactor internals and core support materials list in Table 4.5-2 of the DCD. (See RAI 269-2155, Question 04.05.02-3b and RAI 414-3102, Questions 04.05.02-8, 04.05.02-9 and 04.05.02-16.) The applicant's response to these questions clarified the material specifications used for the lower core support plate, the radial support keys and other reactor internals. The staff finds the applicant response to these questions acceptable because these materials meet ASME Code Section II materials specifications and are appropriate for these applications.

Therefore, the staff finds the material specifications for the core support and reactor vessel internals to be acceptable because they meet the requirements of the ASME Code Sections II and III, and Code Case N-4-11 that is approved by the NRC staff in RG 1.84.

Controls on Welding

Section 4.5.2.2 of the US-APWR DCD states that welding of reactor internals and core support structures follows the provisions of Section III, Articles NG-2000, NG-4000, and NG-5000 and Section IX of the ASME Code. Additional discussion in Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," verifies conformance of welding with RGs 1.31, 1.34, 1.37, 1.44 and 1.71. The staff requested the applicant to provide answers to several welding questions discussed in the following paragraphs. These questions were necessary to make a determination of the acceptability of welding controls used for reactor internals and core support structures in accordance with the requirements of 10 CFR 50.55a and 10 CFR Part 50 Appendix A GDC 1.

Since a previous revision of the DCD specified a lower delta ferrite limit (FN3) for temperatures exceeding 427°C (800°F), the staff requested technical justification for this in RAI 269-2155, Question 04.05.02-5b. The applicant adequately responded that the design temperatures for the reactor internals would not exceed 427°C (800°F) and by revising the DCD to delete the delta ferrite requirements for design temperatures exceeding 427°C (800°F).

The staff requested additional information related to core barrel welding. (See RAI 269-2155, Question 04.05.02-4a, RAI 414-3102, Question 04.05.02-13, and RAI 502-3979, Question 04.05.02-18.) The applicant's responses to these questions provided a list of core barrel welds not listed in the DCD, documented the welding processes to be used, and stated that all welding processes would be controlled and qualified in accordance with the ASME Code Sections III and IX. The applicant will use both electron beam welding (with no added filler metal) and gas tungsten arc welding (with added filler metal) in the manufacture of the core barrel and attachments to the core barrel. The staff finds the applicant response to these questions acceptable because the welding processes meet the requirements of the ASME Code Sections III and IX and are appropriate for these applications.

In RAI 527-4203, Question 04.05.02-21, the staff requested the applicant to clarify how the irradiation specimen guides are attached to the core barrel. In its response dated March 2, 2010, the applicant clarifies that pads are welded to the outer surface of the core barrel and that threaded fasteners are used to connect the irradiation specimen guides to the weld pads. The staff finds the applicant's response to this question acceptable because these processes meet the requirements of the ASME Code Sections III and IX and are appropriate for these applications.

Reactor Vessel Attachments

The following questions were necessary determine the acceptability of the controls on any welding used in the attachment of the core barrel to the reactor vessel.

In RAI 414-3102, Question 04.05.02-11, the staff requested the applicant to discuss various aspects of the radial support to reactor vessel attachment including radial support material, welding process, welding material, welding qualification, cladding material, and cladding sequence. This question explored an important interface between core support and reactor coolant pressure boundary. The controls on welding in this interface are critical to ensure integrity of the reactor coolant pressure boundary. In its response dated August 7, 2009, the applicant confirmed the materials and processes used to make the radial support welds and reactor vessel cladding. The applicant's response also clarified that nickel based alloy buttering, qualified as structural welds, is used to clad the reactor vessel base material in the vicinity of the radial supports. Finally, the applicant's response clarified the sequence of applying stainless steel cladding and nickel based alloy cladding, and this sequence is critical to prevent cracking in the reactor vessel.

In RAI 502-3979, Question 04.05.02-17, the staff requested the applicant to discuss welding controls imposed to ensure that welding of the radial supports to the reactor vessel does not embrittle the reactor vessel. In its response dated January 18, 2010, the applicant replied that the qualification procedures, in accordance with ASME Code Section IX, will control the essential parameters and that the reactor vessel will be post weld heat treated after welding of the radial supports. The staff notes that the provisions of ASME Code Section IX effectively prohibit an increase in heat input without requalification for structural welds to base materials with impact requirements (such as the reactor vessel (RV)) and that post weld heat treatment of the RV would effectively mitigate stresses potentially induced by the welding process.

Therefore, the staff finds the controls on welding on the core support and reactor vessel internals to be acceptable because they meet the requirements of the ASME Code, Sections III and IX.

Nondestructive Examination

Section 4.5.2.3 of the US-APWR DCD states that nondestructive examination of wrought seamless tubular products and fittings is in accordance with ASME Code, Section III, Division 1, NG-2500, and that acceptance criteria of the nondestructive examination are in accordance with ASME Code, Section III, Division 1, NG-5300.

Therefore, the staff finds the nondestructive examination for the core support and reactor vessel internals to be acceptable because they meet the provisions of the ASME Code, Section III, Division 1, Subsection NG.

Fabrication and Processing of Austenitic Stainless Steel

The austenitic stainless steel components from the reactor vessel internals and core support structures are fabricated in accordance with RG 1.44, "Control of the Use of Sensitized Stainless Steel," to minimize their susceptibility to stress-corrosion cracking. Additional discussion in Subsection 5.2.3 verify conformance of reactor internals and core support structures with RG 1.44 and RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." In addition, strain hardened austenitic stainless steel has a maximum-specified room

temperature yield strength of 90 ksi, which meets the guidance in SRP Sections 4.5.2 and 5.2.3. Contamination protection and cleaning of austenitic stainless steels conform to ASME Code NQA-1. As discussed in SRP Section 4.5.2, the acceptance criteria for this area of review are given in SRP Section 5.2.3, Subsections II.2 and II.4.a, b, d, and e.

In Table 4.5-2, "Summary of Reactor Internals and Core Support Materials," of the US-APWR DCD, cast austenitic stainless steel alloy SA-351 CF8 is identified as an option for the guide funnel of the CRDM thermal sleeve. In RAI 620-4886, Question 04.05.02-23, in accordance with the recommendations of the SRP Section 4.5.2 Subsection III.4, staff requested that the applicant verify that this material will provide adequate fracture toughness over its design life considering thermal aging due to exposure to reactor coolant operating temperatures. In its response dated September 14, 2010, the applicant stated that DCD Section 5.2.3 requirements (referenced by note to Table 4.5-2) will control the delta ferrite content and will limit this content to 20 percent or less. The staff finds the applicant response acceptable because it provides reasonable assurance that this component will not experience in-service degradation due to thermal aging that could compromise the long-term integrity of this component.

Neutron Reflector Design and Fabrication

The US-APWR design uses an austenitic stainless steel neutron reflector that replaces the thin baffle plates used in current operating reactors. The neutron 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 neutron reflector consists of stacked blocks that rest on the lower-support plate and contains no welds or bolted connections close to the core. Alignment pins are used to align the slabs and are axially restrained by tie rods bolted to the lower-core support plate. Because this is a relatively new design concept, the staff requested additional information on the design, materials and fabrication processes used to determine how the neutron reflector will maintain its intended safety functions.

In RAI 269-2155, Question 04.05.02-2a, the staff requested the applicant to provide a detailed drawing of the neutron reflector and its components. In its response dated May 13, 2009, the applicant provided a sketch of the neutron reflector which was to be included in Section 3.9.5, "Reactor Pressure Vessel Internals," of the DCD. **Therefore the staff identifies Confirmatory Item 04.05.02-1 for the applicant to revise Section 3.9.5 of the DCD as stated in the response to RAI 269-2155, Question 04.05.02-2a**.

In RAI 269-2155, Question 04.05.02-2b, the staff requested the applicant to discuss the presence or absence of sharp notches in neutron reflector components, since notches in the design could act as stress concentrators and crack initiators which could lead to loss of function of the neutron reflector. In its response dated May 13, 2009, the applicant stated that the DCD would be modified to state that sharp notches will be avoided in the neutron reflector and other reactor internal structures. **Therefore the staff identifies Confirmatory Item 04.05.02-2 for the applicant to revise Section 3.9.5 of the DCD as stated in the response to RAI 269-2155, Question 04.05.02-2b**.

Other Degradation Mechanisms (Irradiation-Assisted Stress-Corrosion Cracking (IASCC) and Void Swelling)

Other degradation mechanisms that are currently challenging the integrity of the reactor vessel internals and core support structures include IASCC and void swelling. To assess how the

design and use of the material used in the US-APWR DCD will manage these degradation mechanisms, the NRC staff requested additional information concerning the design of the neutron reflector and core barrel 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.

In RAI 269-2155, Question 04.05.02-2c, the staff requested the applicant to discuss plans to address void swelling, if any, on the neutron reflector components. In its response dated May 30, 2009, the applicant provided detailed proprietary calculations of void swelling, along with a description of methodology to determine acceptability. The applicant's methodology involves calculating void swelling predictions in accordance with Electric Power Research Institute (EPRI) MRP-175, Equation G-2, with no further functionality evaluation if the prediction is less than 2.5 percent. Since the prediction is significantly less than 2.5 percent for the neutron reflector, the applicant considers that sufficient evaluation has been performed. The staff finds the applicant response acceptable because it provides reasonable assurance that these components will not experience in-service degradation due to void swelling that would compromise the integrity of these components.

In order to determine the possible impact of IASCC on the neutron reflector components, the staff requested in RAI 269-2155, Question 04.05.02-2d that the applicant discuss the end of life peak neutron fluence at the neutron reflector. In its response dated May 13, 2009, the applicant provided a proprietary calculation of the maximum fluence at the corner of the neutron reflector at the end of the 60 year design life. The applicant's response also stated that this maximum fluence region can be inspected and that the neutron reflector can be repaired or replaced if damage is observed. As a follow up to the applicant's response, the staff requested in RAI 414-3102, Question 04.05.02-7 that the applicant discuss the inspection plan for the neutron reflector to assure that IASCC will not affect the integrity of the neutron reflector. The applicant responded that visual inspections of the inside surface of the neutron reflector at 10 year intervals with acceptance criteria being no structural damage or change. As additional follow up to the applicant's response, the staff requested in RAI 502-3979, Question 04.05.02-19 that the applicant provide technical bases for concluding that the effects of IASCC in the US-APWR neutron reflector will be less than found in core-baffle structures in existing PWR designs and thus not significant. The applicant responded that although the fluence level in the US-APWR design may be on the same order of magnitude as existing PWR designs, there is no high stress on the ring blocks as there would be on threaded fasteners of existing core baffle designs. The applicant further responded that although the US-APWR design uses tie rods with some tensile stresses applied, the fluence level on the tie rods is much lower than the fluence level on the baffle fasteners in existing PWR designs. The staff finds the applicant's response to RAI 269-2155, Question 04.05.02-2d, dated May 13, 2009, as supplemented by the applicant response to RAI 414-3102, Question 04.05.02-7, dated August 7, 2009, and RAI 502-3979, Question 04.05.02-19, dated January 18, 2010, acceptable because it provides reasonable assurance that these components will not experience in-service degradation due to IASCC that would compromise the integrity of these components.

In order to determine the possible impact of IASCC on the core barrel components, the staff requested in RAI 269-2155, Question 04.05.02-4b that the applicant discuss the maximum fluence on the core barrel at the end of a 60 year design life. The applicant responded, May 13, 2009, with a proprietary calculation of the end of life fluence, and a statement that the core barrel can be inspected. As a follow up to the applicant's response, the staff requested in RAI 414-3102, Question 04.05.02-14 that the applicant discuss the inspection plan of the core barrel to ensure that IASCC will not affect the integrity of the core barrel. In its response, dated

August 7, 2009, the applicant stated that 10-year interval visual inspections of the whole surface of the core barrel except the inside surface of the core region represented by the inspection of the neutron reflector will be performed to ensure no structural damage or change observed. As additional follow up to the applicant's response, the staff requested in RAI 502-3979, Question 04.05.02-20 that the applicant discuss the technical bases for determining that there will be no adverse effects of IASCC on the core barrel. The applicant responded, January 18, 2010, that the shielding effect of the neutron reflector would significantly reduce the fluence on the core barrel of the US-APWR design as compared to the fluence on existing PWR designs. The applicant further responded that any failure of the core barrel due to IASCC has not been identified in PWRs in U.S., Europe and Japan. The staff finds the applicant's response to RAI 269-2155, Question 04.05.02-4b as supplemented by the applicant response to RAI 414-3102, Question 04.05.02-14 and RAI 502-3979, Question 04.05.02-20 acceptable because it provides reasonable assurance that these components will not experience in-service degradation due to IASCC that would compromise the integrity of these components.

In its evaluation of the possible impact of IASCC regarding the neutron reflector and core barrel, the staff reviewed the applicant's response to RAI 374-2446, Question 03.09.05-9 relative to IASCC. In its response dated July 17, 2009, the applicant stated that its assessment of IASCC was performed in accordance with EPRI's MRP-175 guidelines. The applicant also provided end of life neutron exposure for the US-APWR neutron reflector block alignment pin, neutron reflector tie-rod, and core barrel. However, the applicant did not clearly state the operating stresses for these components or otherwise give a clear statement that these components would be operate below the stress threshold specified in MRP-175 for evaluating IASCC. Therefore, the staff requested in RAI 653-5208, Question 04.05.02-24 that the applicant discuss the anticipated operational stress for the US-APWR neutron reflector alignment pin, neutron reflector tie-rod, and core barrel and whether this operational stress is predicted to be below the MRP-175 stress threshold for IASCC. In its response dated December 14, 2010, the applicant provided an analysis in accordance with the guidelines of MRP-175 which showed the anticipated operational stress to be below the threshold level for IASCC. The staff finds the applicant analysis to be conservative and the applicant conclusion to be acceptable. The staff notes that a finding that the integrity of the reactor internals will not be compromised by IASCC is based not only on an evaluation of the susceptibility of internal components to IASCC but also on the planned inspection of these components at regular intervals during the operating life of the reactor (see discussion on RAI 502-3979, Questions 04.05.02-19 and -20 in previous paragraphs of this SER). Therefore, the staff finds the applicant response to RAI 653-5208, Question 04.05.02-24 acceptable because it provides reasonable assurance that these components will not experience in-service degradation due to IASCC that would compromise the integrity of these components.

Therefore, upon resolution of **Confirmatory Items 04.05.02-1 and 04.05.02-2**, the staff finds the fabrication of the austenitic stainless steel core support and reactor vessel internals to be acceptable because they satisfy the guidelines in RG 1.44 and acceptance criteria in SRP Section 4.5.2, including the acceptance criteria in SRP Section 5.2.3, Subsections II.2 and II.4.a, b, d, and e, which is referenced by SRP Section 4.5.2.

Other Material Considerations

In Table 4.5-2 of the US-APWR DCD, Alloy X-750 is specified for clevis insert bolts and springs, and as an option for the guide tube support pins. As discussed in EPRI MRP-175, the susceptibility of Alloy X-750 to stress corrosion cracking is dependent on processing and heat treating variables. Since the resistance of Alloy X-750 to stress corrosion cracking is dependent

on processing and heat treating requirements, the staff requested in RAI 573-4592, Question 04.05.02-22 that the applicant discuss the operating history of Alloy X-750 in these applications and to discuss how the processing and heat treatment will minimize the possibility of stress corrosion cracking. In its response dated May 19, 2010, the applicant provided proprietary solution treatment and age hardening data, and stated that guide tube support pins with this heat treatment have been used since 1982 and are applied for more than 20 PWR plants in Japan with no cases of SCC identified. The staff finds the applicant's response acceptable because it provides reasonable assurance that these components will not experience in-service degradation due to SCC that would compromise the integrity of these components and because the material meets the requirements of the ASME Code Sections II and III.

Therefore, the staff finds the material selection, fabrication, nondestructive examination and welding practices meet the requirements of the ASME Code Sections II, III and IX and provide reasonable assurance that the materials used for the reactor vessel internals and core support structures will preclude operational in-service degradation, such as SCC, IASCC, thermal aging, and void swelling to ensure the structural integrity is maintained and that their corresponding safety functions can be performed. **The staff identifies Confirmatory Items 04.05.02-1 and 04.05.02-2 for the applicant to revise Section 3.9.5 of the DCD as stated in the response to RAI 269-2155, Questions 04.05.02-2a and 04.05.02-2b.**

4.5.2.5 Combined License Information Items

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

4.5.2.6 Conclusions

As a result of the confirmatory item the staff is unable to finalize its conclusion on Section 4.5.2 in accordance with the requirements of NRC regulations.

4.6 Functional Design of Reactivity Control Systems

4.6.1 Introduction

The reactivity control systems for the US-APWR are comprised of two independent and diverse technologies that complement each other, but each are capable to effect a safe shutdown maintaining the reactor within specified design limits during AOOs, and prevent or mitigate the consequences of AOOs and postulated accidents. One system, the control rod drive system (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 the use of the chemical reactivity control of the ECCS. No credit for the reactivity control capabilities of the chemical and volume control system (CVCS) is taken for AOOs and postulated accidents.

4.6.2 Summary of Application

DCD Tier 1: The Tier 1 information associated with this section is found in Tier 1 sections as it relates to the functional control of the reactivity control systems:

- Section 2.4.1, “Reactor System,”
- Section 2.4.4, “Emergency Core Cooling System,”
- Section 2.5.1, “Reactor Trip System and Engineered Safety Feature Systems,” and
- Section 2.5.3, “Diverse Actuation System.”

DCD Tier 2: The applicant has provided a DCD Tier 2 system description in Section 4.6, summarized below.

- The reactivity control systems are designed to perform the following functions:
  - Provide the capability to effect a safe shutdown,
  - Respond within acceptable limits during AOOs, and
  - Prevent or mitigate the consequences of AOOs and postulated accidents.

- Two independent reactivity control systems of different design mechanisms:
  - Mechanical reactivity control of the control rod absorbers, and
  - Chemical reactivity control of the ECCS.

- Mechanical Design
  - CRDS has 69 RCCAs with each containing 24 absorber rods composed of Ag-In-Cd alloy.
  - CRDM is of the magnetically operated jacking type design.
  - CRDM consists of a pressure housing, latch assembly, drive rod assembly, and coil stack assembly.
  - CRDS rod insertion in the core is by gravity when electrical power to the CRDM coils is lost.

- Chemical Design
  - ECCS uses boron injection into the reactor coolant for chemical reactivity control through the safety-related safety injection system (SIS)

- Each system is capable of reliably controlling reactivity changes under conditions of normal operation, including AOO to assure acceptable fuel design limits are not exceeded

- Each system is provided with safety-related equipment redundancy to deal with single failure, environmental qualification, and protection from external hazards

- In conjunction, CRDS and SIS provides protection in the following events:
  - LOCA.
  - Ejection of a control rod cluster assembly.
  - Secondary steam system piping failure.
  - Inadvertent opening of main steam relief or safety valve.
  - Steam Generator tube rupture.

ITAAC: The ITAAC associated with Tier 2 Section 4.6 are given in Tier 1 sections as it relates to the functional control of the reactivity control systems:

- Section 2.4.1, “Reactor System,”

- Section 2.4.4, “Emergency Core Cooling System,”
- Section 2.5.1, “Reactor Trip System and Engineered Safety Feature Systems,” and
- Section 2.5.3, “Diverse Actuation System.”

TS: The TS associated with DCD Tier 2 Section 4.6 are given in Tier 2 Chapter 16:

- Section 3.1, “Reactivity Control Systems” and
- Section 3.5, “Emergency Core Cooling Systems (ECCS).”

Initial Plant Testing: DCD Tier 2 Chapter 14 identifies the reactivity and system tests related to Section 4.6 to be performed as part of the initial plant testing program including:

- 14.2.12.1.9, “Reactor Control, Rod Control, and Rod Position Indication Preoperational Test”
- 14.2.12.1.10, “CRDM Motor-Generator Set Preoperational Test,”
- 14.2.12.1.11, “CRDM Initial Timing Preoperational Test,”
- 14.2.12.1.54, “Safety Injection System (SIS) Preoperational Test,”
- 14.2.12.1.65, “CRDM Cooling System Preoperational Test,”
- 14.2.12.2.1.5, “Rod Drop Time Measurement Test,”
- 14.2.12.2.1.6, “CRDM Operational Test,”
- 14.2.12.2.1.7, “Rod Control System Test,”
- 14.2.12.2.1.8, “Rod Position Indication Test,”
- 14.2.12.2.2.1, “Initial Criticality Test Sequence,”
- 14.2.12.2.2.2, “Initial Criticality,”
- 14.2.12.2.3.4, “RCCA Bank Worth Measurement at Zero Power Test,”
- 14.2.12.2.3.5, “Pseudo Rod Ejection Test,” and
- 14.2.12.2.4.8, “Automatic Rod Control System Test.”

Also, system accessibility will be provided for periodic testing and in-service inspection of the CRDS and SIS.

4.6.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for this area of review, and the associated acceptance criteria, are given in the SRP 4.6 ,”Functional Design of Control Rod Drive System,” of NUREG-0800 and are summarized below. Review interfaces with other SRP sections can be found in Section I of SRP 4.6. The applicable regulations include:

- GDC 4, “Environmental and Dynamic Effects Design Bases,” found in Appendix A to 10 CFR Part 50, as it relates to the SSCs 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,

- 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,

- GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” as it relates to the fuel design such that the specified limits are not exceeded for any single malfunction of the reactivity control system,

- GDC 26, “Reactivity Control System Redundancy and Capability,” 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 AOOs 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,

- GDC 27, “Combined Reactivity Control Systems Capability,” as it relates to the combined reactivity control systems’ capability such that the reactivity control system design shall have a 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,

- GDC 28, “Reactivity Limits,” 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,

- GDC 29, “Protection Against Anticipated Operational Occurrences,” as it relates to protecting systems against AOOs such that the design of the protection and reactor control systems should assure an extremely high probability of accomplishing its safety functions in the event of AOOs, and

- 10 CFR 52.47(b)(1), as it relates to the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, and the NRC's regulations.

Note: Although the requirements related to the reduction of risk from anticipated transients without scram (ATWS), 10 CFR 50.62(c)(1) and (2), are not part of SRP 4.6, the staff reviewed these requirements in regard to the functional design of the reactivity control systems because the applicant addressed these in Section 4.6.2, “Evaluations of the CRDS.” In addition, the applicant included in its design an independent system which performs the functions that satisfy the 10 CFR 50.62(c)(2) requirements although US-APWR is a WH type reactor design and not required to satisfy 10 CFR 50.62(c)(2). ATWS is described in DCD Sections 7.8, “Diverse Instrumentation and Control Systems,” and 15.8, “Anticipated Transients without Scram.”

- 10 CFR 50.62(c)(1), as it relates to protecting the system against ATWS. Specifically for PWRs, the alternate monitoring system from sensor output to final actuation device must be diverse from the reactor trip system, to automatically

initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS and

- 10 CFR 50.62(c)(2), as it relates to protecting the system against ATWS. Specifically for Combustion Engineering or Babcock and Wilcox PWRs, the alternate monitoring system must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

Acceptance criteria in SRP 4.6 adequate to meet the above requirements include for:

- GDC 4, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents,

- 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,

- GDC 25, the design of the reactivity control systems should assure that a single malfunction of the CRDS will not result in exceeding acceptable fuel design limits,

- GDC 26, the functional reactivity control systems should be redundant and composed of two independent reactivity control systems of different design principles with the capability of providing sufficient operational control and reliability during reactivity changes during normal operation and AOOs,

- GDC 27, the combined capability of CRDS and emergency core cooling system should reliably control the reactivity changes to assure the capability to cool the core under accident conditions,

- 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,

- GDC 29, the CRDS should be designed to ensure an extremely high probability of functioning during anticipated operational occurrences, and

- 10 CFR 52.47(b)(1), the DC application should contain the proposed ITAAC that are necessary and sufficient to assure that the as-built facility meets the requirements set forth in the DCD.

4.6.4 Technical Evaluation

The staff reviewed the US-APWR DCD, Tier 2, Section 4.6, “Functional Design of Reactivity Control Systems,” in accordance with SRP 4.6 including applicable portions of other SRP sections identified in Section I of SRP 4.6 that interface with the reactivity control systems.

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 GDCs 4, 23, 25, 26, 27, 28, 29 and 10 CFR 52.47(b)(1).

GDC 4 - Environmental and Dynamic Effects Design Bases

GDC 4 requires that the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions. In DCD Section 4.6, the applicant states that failure in the CRDM cooling system will, in the worst case, result in an individual control rod trip or a full reactor trip since the CRDM coil stack temperature will increase until the coil stack assembly movement/stationary position function is loss which results in the release of the RCCA into the core by gravity. However, the staff was unable to find a discussion or reference that included a consideration of other environmental conditions, such as humidity, vibration, or possible pipe fracture releasing fluid onto the CRDMs that could affect the CRDS functional capabilities to provide a safe reactor shutdown. Therefore, the staff issued RAI 316-2296, Question 04.06-1 to request that the applicant provide a discussion or reference that supports the CRDS capabilities to operate and perform its design function in the reactor vessel cavity under adverse environmental conditions. By letter dated May 20, 2009, the applicant provided a response, to RAI 316-2296, Question 04.06-1, that the CRDS design is qualified to operate under adverse environmental conditions and perform its intended safety function by referencing DCD sections in Tier 2 Chapter 3, “Design of Structures, Systems, Components, and Equipment,” that support the operability of the CRDS through various analysis, quality assurance, and in-service testing programs. Since the pressure housing is part of the pressure boundary of the reactor primary coolant system, it is categorized as a Class 1 component which is designed in accordance with requirements in 10 CFR 50.55a and ASME Code, Section III. Therefore, the pressure housing and internal components are designed for a temperature of 343°C (650°F) and a pressure of 2500 psia (17.24 MPa). Some of the CRDM components, such as, the latch mechanism, the drive rod, and the coil assembly are classified as non-pressurized components which do not required compliance with ASME Code requirements. However, these components come under the quality assurance requirement regarding design, fabrication, inspection, and testing of safety components in 10 CFR 50.55a. In addition, the staff reviewed and found acceptable DCD Sections 3.9.4.4 and 3.9.6 with respect to the CRDS operability assurance program and in-service testing, respectively. Also, in DCD Section 3.9.4.3, the applicant provides a description of the seismic qualification of the CRDS. Therefore, in this respect, the staff considers the response and the applicable DCD sections mentioned above adequately address the CRDS functionality concerns in an adverse environment during normal and postulated events acceptable in compliance with GDC 4. A review of the CRDS is presented in Chapter 3 of this report.

Potential for Moderate and High Energy Line Breaks and Generated Missiles

In DCD Section 4.6.2, the applicant states that the protection of the essential components of the CRDS from the effects of postulated moderate and high energy water line breaks and associated generated missiles is described in Subsection 3.5.1.2. Primary missile protection is provided by locating the credible missile sources behind concrete walls and floors, and/or locating the SSCs outside the zones of postulated missile strikes. However, the discussion was a general description of the protection features without directly addressing the specific features incorporated to protect the CRDS components. Therefore, the staff submitted RAI 316-2296, Question 04.06-4 to the applicant to address specific features incorporated to protect the CRDS

components from generated missiles. In its response dated May 20, 2009, the applicant states that the load of postulated moderate and high energy water line breaks are considered in the design of the CRDM housing and a description of the CRDM stress analysis and results are provided in the Technical Report MUAP-09009 Revision 0, "Summary of Stress Analysis Results for the US-APWR Control Rod Drive Mechanism." Upon review of the additional information, the staff concludes that the concerns were adequately addressed and finds the response acceptable. Moderate and high energy water line breaks and associated generated missiles are further discussed in Chapter 3 of this report.

GDC 23 - Protection System Failure Modes

To comply with GDC 23, the CRDS must fail in a safe condition in the event of adverse conditions or environments during all modes of operation, including all plant conditions from the cold shutdown condition to the full-load condition under normal operation, AOOs, and accidents. The CRDS primary function of RCCA movement and stationary position control is performed by the CRDM coil stack assembly. However, Section 4.6 does not provide a description of the CRDS/CRDM or identify another DCD section or reference a report which discusses the CRDS/CRDM in respect to GDC 23. The staff submitted RAI 316-2296, Question 04.06-5 to the applicant to address the fail safe features of the CRDM with respect to GDC 23. In its response dated May 20, 2009, the applicant states that the CRDM fails safe under various failure modes which, in each failure mode, results in the insertion of the RCCA into the core by gravity. For instance, the coil stack assembly consists of three coils that control the movement and stationary position function of the RCCA assemblies and in the event of loss of electrical power or cooling of the CRDM, this control function will be lost; thereby, releasing the RCCA assemblies to insert into the core by gravity. In addition, in the response, the applicant refers to RAI 107-1293-03.09.04-03 (ML090130258) that included a revision to DCD Section 3.9.4.2.3 to clarify the requirements applicable to the non-pressurized components of the CRDM which are classified as non-safety components but are designed, fabricated, inspected, and tested under the quality assurance requirement regarding safety components in 10 CFR 50.55a. Based on the response with regard to the review of DCD Sections 3.9.4 and 4.6, the staff concludes that the requirements of GDC 23 are satisfied; therefore, the staff finds DCD Section 4.6 acceptable with respect to GDC 23.

GDC 25 - Protection System Requirements for Reactivity Control Malfunctions

The compliance of the design to the single failure criterion of the reactor trip system is discussed in DCD Section 7.2, "Reactor Trip System," Table 7.2-8, "FMEA for Reactor Trip in PSMS," and Technical Report MUAP-07015(R0), "FMEA of Control Rod Drive Mechanism Control System," which concludes 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, a functional unit of the protection system, or a reactor trip device. Based on the Control Rod Drive Mechanism Control System (CRDMCS) analyzed in detail in the failure mode and effects analysis report, the analyses described in Section 7.2, "Reactor Trip System," the technical report, and the results of the safety analyses presented in Section 15.4, "Reactivity and Power Distribution Anomalies," the staff concludes that the CRDS performs its intended safety function, a reactor trip, by placing the reactor in a subcritical condition such that the fuel design limits are not exceeded. Therefore, the staff concludes that GDC 25 is satisfied as it relates to the fuel design such that the specified fuel design limits are not exceeded for any single malfunction of the reactivity control system. A review of the ECCS with respect to fuel design limits are presented in Chapters 6 and 15 of this report.

GDC 26 - Reactivity Control System Redundancy and Capability

The functional design of the reactivity control systems include two totally independent, redundant, and diverse designed systems each capable of bringing the reactor to a subcritical condition.

The primary independent reactivity control system is the CRDS which is extremely reliable in controlling reactivity changes under conditions of normal operation, including AOOs to assure acceptable fuel design limits are not exceeded. CRDS is composed of 69 RCCAs with each containing 24 absorber rods which are moved by the CRDM. The CRDM is an electro-mechanical device of the magnetically operated jacking type design. The CRDM consists of a pressure housing, latch assembly, drive rod assembly, and coil stack assembly that provide vertical movement and position latching control of the RCCA assembly. When electrical power to the CRDM coils is lost, the ability to hold the RCCA assembly is removed allowing the assembly to insert into the core by gravity.

The second independent reactivity control system is the safety injection system, of the ECCS, which includes the high head injection system, the accumulator system, and the emergency letdown system. The safety-related SIS is designed with redundancy to deal with single failure, environmental qualification, and protection from external hazards. Boron is injected into the reactor coolant for chemical reactivity control through the SIS. The primary safety-related function is the removal of stored and fission product decay heat from the reactor core following an accident. The SIS is evaluated in Chapter 6 of this report.

Therefore, the staff finds the functional design of the reactivity control systems satisfy GDC 26 requirements because the systems are diverse with each capable of bringing the reactor to a subcritical condition.

GDCs 27 and 28 - Evaluation of Combined Performance

In Section 4.6.4, the applicant describes the two independent reactor control systems: the control rods and the SIS of the ECCS in respect to postulated events. Furthermore, the applicant states that in the safety analyses, as discussed in Chapter 15, there are only two postulated events which assume credit for two reactivity control systems to render the plant subcritical. The events are the steam line break (SLB) and LOCA with the CRDS via the reactor trip system and the ECCS are the reactivity control systems for which credit is taken in these accidents. The CVCS is another reactivity control system with water boration capabilities but is not considered in the Chapter 15 transients' analysis other than as a malfunctioning system or improper operator maneuver. Reactivity control is also discussed in DCD Sections 4.2, "Fuel System Design," 4.3, Nuclear Design," 6.3, "Emergency Core Cooling Systems," and 7.2, Reactor Trip System." Upon review of the DCD sections above, the staff finds that the combined performance of the reactivity control systems are acceptable and complies with the requirements of GDC 27 and GDC 28.

GDC 29 - Reactivity Control Systems Probability of Achieving Safety Functions

Based on the evaluation of the failure mode and effects analysis, in-service testing programs, quality assurance programs, TS, and redundancy in system trip mechanisms, the results demonstrate that the CRDM design complies with GDC 29 to ensure an extremely high probability of functioning during AOOs. In addition, the common cause failure impairing the ability of the reactor trip system to perform its safety function is extremely low because of the

defense in depth philosophy that ensures that a backup monitoring system will initiate a redundant trip which results in the loss of electrical power to the CRDM coil stack assembly and the release of the RCCAs into the core by gravity.

Initial Plant Test Program – RG 1.68

The initial plant test programs for water-cooled nuclear power plants are described in RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants." For this technical evaluation, the staff reviewed the guidance for ITAAC to be completed prior to fuel load, system preoperational testing after system construction is completed, and startup power accession testing after fuel load. The staff identified and performed an overview of DCD Tier 1 Sections 2.4.1, "Reactor System," 2.44, "Emergency Core Cooling System," 2.51, "Reactor Trip System and Engineered Safety Feature Systems," and 2.53, "Diverse Actuation System," related to ITAAC tests and preoperational/startup power accession tests described in DCD Tier 2 Section 14.2 to confirm that the tests are consistent with the information provided in Section 4.6. However, for a detailed review, the staff provided a thorough evaluation of the tests as described in Chapter 14, "Verification Programs."

10 CFR 52.47(b)(1) - ITAAC Information

The staff identified the reactivity control systems ITAAC information in the following DCD Tier 1 sections:

- Section 2.4.1, "Reactor System"
  - CRDM
- Section 2.4.4, "Emergency Core Cooling System"
  - ECCS (SIS Actuation)
- Section 2.5.1, "Reactor Trip System and Engineered Safety Feature Systems"
  - Reactor Trip
  - Engineered Safety Feature (ESF) systems
  - ESF systems (diverse path – manual actuation)
- Section 2.5.3, "Diverse Actuation System"
  - Diverse Reactor Trip (M/G set trip)
  - Turbine Trip
  - Emergency Feedwater Pump
  - Safety Injection Pump (manual activation – MCR).

As applicable, the staff confirmed that ITAAC exist and are sufficient to verify automatic and manual actuation of the equipment identified in each of the above sections. Therefore, the staff finds the tests to be acceptable and satisfy the requirements of 10 CFR 52.47(b)(1). The staff provided additional information in Chapter 14 of this report.

Initial Plant Test Program - Preoperational/Power Accession

The preoperational and power accession tests directly related to the CRDS and ECCS were reviewed by the staff to verify that the reactivity control systems accomplish their functional design as described in DCD Section 4.6. The preoperational tests are performed following the

completion of the equipment construction phase and the operational tests are performed following the fuel loading and power accession phase. Of the tests listed in Section 4.6.2 of this report, the staff performed an overview while focusing on the prerequisites, test methods, and acceptance criteria and found each test providing sufficient information to perform a creditable check of the reactivity control functional requirements. Therefore, the staff considers the tests to be adequate and sufficient for their intended purposes and did not identify the need for further tests. These tests are described in Chapter 14 of this report.

ATWS - 10 CFR 50.62(c)(1) and (2)

In the event that the reactor trip system fails, the US-APWR is designed with an independent non-safety related backup system, diverse instrumentation and control system (DAS) that provides automatic and manual actuation of the reactivity control systems. The DAS is totally independent such that the plant parameters are processed from the sensor output to final signal actuation device without an interface with the safety-related systems of the reactor trip system. Some DAS functions include the automatic initiation of the emergency feedwater system, a turbine trip, and a scram of the reactivity control system which satisfies the requirements of 10 CFR 50.62(c)(1) and (2). Therefore, the staff finds the DAS acceptable with respect to reactivity control system. In addition, the staff reviewed the DAS Actuation Test 14.2.12.1.20 and determined that necessary test steps are performed to confirm compliance with 10 CFR 50.62(c)(1) and (2). A description of the ATWS, DAS, and initial startup tests is provided in Chapters 7, 14, and 15 of this report.

Performance Assurance Program of the Control Rod Drive System

CRDS performance testing is categorized in several phases of the operability assurance program: (1) Prototype component testing, (2) Production testing of components, (3) Initial/Startup testing, and (4) Periodic in-service testing. In Section 4.6.3, reference is made to the CRDS operability assurance programs, described in Section 3.9.4.4, which were developed to confirm the functional performance of the CRDMs both statically and dynamically. In each phase, from production to in-plant operation, a series of tests are performed to confirm that the design specification requirements to the operational requirements are satisfied. Also, this section states that the structural integrity of the RCS pressure boundary is confirmed by stress analysis performed in accordance with ASME Code, Section III as described in MUAP-09009, “Summary of Stress Analysis Results for Control Rod Drive Mechanism.” The staff performed an overview of the assurance program and found it sufficient in checking the CRDS performance requirements. A description and staff evaluation is presented in Chapter 3 of this report.

Reactivity Control Systems - TS

The US-APWR CRDS is designed to independently provide the reactivity capability to achieve a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of AOOs and postulated accidents. To accomplish the safety functions, the RCCAs are designed with 24 rods, which act as a neutron absorber in the reactor core. In the rod drop, 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 will be conservative or best estimate (depending on the requirements of the analysis). Such an explicit maximum drop time for each RCCA is found in the TS 3.1.4, “Group Alignment Limits,” Surveillance Requirement (SR) 3.1.4.3, which requires verification of the maximum drop time for each RCCA after the reactor vessel head has been removed and before criticality. Also, SR 3.1.4.2 requires

verification of control rod freedom of movement by moving each rod not fully inserted in the core in either direction. In addition, TS Section 3.5, "Emergency Core Cooling Systems (ECCS)" verifies the amount and concentration of the accumulators' borated water and the operability of the SIS. The staff did not identify any additional TS requirements in regard to the reactivity control systems and similar to the approved standard TS. Therefore, the staff considers the TS adequately address the operational aspects of the reactivity control systems. However, the TS are described and evaluated in detail in Chapter 16 of this report.

- **Control Rod Drive Mechanism Cooling System**

As discussed in DCD Section 9.4.6, "Containment Ventilation System," the CRDM cooling system is part of the containment ventilation system which is classified as a non-safety related and non-seismic Category I system. The CRDM cooling system maintains the temperature of the CRDMs below design operating temperature by satisfying the following design bases: (1) Removal of heat dissipated by the CRDMs, (2) Continuity and reliability of operation with 100 percent standby capacity for system fans, and (3) During a loss-of-offsite power (LOOP) condition, the CRDM cooling fans are served by the alternate alternating current (AC) power source.

However, references were not provided in this section to document the specific analysis and qualification of the CRDM cooling system to satisfy the design bases.

The staff submitted RAI 316-2296, Question 04.06-3, to the applicant to provide references and/or discussion of the specific analysis and qualification of the CRDM cooling system. In its response dated May 20, 2009, the applicant states that the design of the non-safety related CRDM cooling system is based on data from existing plants. From this data, it was determined that the required cooling rate for each CRDM is approximately [ ]. Considering the space in the CRDM cooling shroud and the total of 69 CRDMs, the applicant calculated the total required cooling flow rate to be approximately [ ]. The CRDM cooling system is designed with a capacity of 71,000 cfm which provides sufficient excess cooling capacity. However, the staff noted that a cooling rate of [ ] per CRDM with a total of 69 CRDMs requires a minimum cooling rate of [ ] but the applicant did not address the assumptions and factors used to account for this difference and how it relates to the CRDM cooling space. Therefore, the staff submitted follow-up (RAI 913-6331, Question 04.06-12) to the applicant to address the difference in the required CRDM cooling system flow rates including an explanation of the assumptions and factors used in this calculation. In their response, dated April 26, 2012, the applicant explains that the CRDM cooling shroud configuration has 101 CRDM locations of which CRDMs are installed in 69 of these locations for this reactor design. In the remaining 32 empty locations, ventilation dummies, which are equivalent in cross section area of CRDM coil stack, are installed to provide uniform ventilation flow through the CRDMs. Therefore, the total flow rate of [ ] is based on 101 locations at [ ]. The staff finds the RAI response acceptable; therefore, RAI 913-6331, Question 04.06-12 is resolved.

In Section 4.6, it is stated that this section describes how the design of the CRDS complies to the requirements of GDC 4, 23, 25, 26, 27, 28, and 29, as well as to the requirements of 10 CFR 50.62. In addition to the review of Section 4.6, the staff also relied upon the review of Tier 2, Chapters 3, 4, 7, 9, 14, 15 and 16 to confirm that the CRDS complies with the requirements and regulations stated in Section 4.6.3 of this SE.

4.6.5 Combined License Information Items

There are no COL information items identified DCD Tier 2, Table 1.8-2, as applicable to the Functional Design of Reactivity Control Systems and none were identified as a result of the staff's review.

4.6.6 Conclusions

The staff reviewed the functional design of the CRDS and SIS to confirm that the systems have the capability to shut down the reactor with appropriate margin during normal operation, AOOs, and accident conditions. The scope of review included process flow diagrams, layout drawings, piping and instrumentation diagrams, topical reports, and system descriptions of the CRDS and SIS along with the interfacing and supporting systems that provide essential functions in the operation of the CRDS system. All RAIs have been resolved and closed. Therefore, the staff finds that the functional design of the reactivity control system is acceptable because the relevant requirements of the Commission's regulations are otherwise satisfied.