4.0 REACTOR

4.1 Summary Description

This chapter describes the results of the U.S. Nuclear Regulatory Commission (NRC) staff's review of the Advanced Power Reactor 1400 (APR1400) reactor, as described in Chapter 4, "Reactor," of the APR1400 design control document (DCD), Revision 3 (ML18228A681), submitted by the design certification (DC) applicant, Korea Hydro & Nuclear Power Co., Ltd. (KHNP), a wholly owned subsidiary of Korea Electric Power Corporation (KEPCO).

The reactor design includes the fuel system design, reactor nuclear design, thermal-hydraulic design, reactor materials, and functional design of the reactivity control systems. The reactor is designed to withstand normal operations, anticipated operational occurrences (AOOs), and postulated accidents. Independent systems provide reactivity control, and the core design and reactor protection system (RPS) prevent fuel damage from exceeding limits.

4.2 Fuel System Design

4.2.1 Introduction

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

4.2.2 Summary of Application

DCD Tier 1: In Revision 3 of the DCD there were no DCD Tier 1 entries for this area of review. Request for additional information (RAI) 556-9154 (ML17291B308), requested that the applicant provide Tier 1 information describing the fuel system. In response (not publicly available), the applicant provided a DCD Tier 1 fuel system description. The staff reviewed the text provided in Tier 1 of the APR1400 DCD, Revision 3, and confirmed that the text in the revision matched what was expected based on the response to RAI 556-9154 (not publicly available). Based on this comparison, the staff finds that Confirmatory Item 4.2-2 is now closed.

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

4.2.2.1 *Fuel Assembly Description*

The APR1400 fuel assembly contains 236 fuel rods and burnable absorber rods, 4 guide thimble tubes, and 1 instrument tube in a 16x16 array held together by a bottom and top nozzle and guide tubes welded to 12 spacer grids. The guide thimble tubes serve as channels to guide control element assemblies (CEAs) over their entire length of travel.

In-core instrumentation is inserted in the central guide tube of selected fuel assemblies.

4.2.2.2 *Fuel Rod Description*

The applicant stated that the fuel rods consist of slightly enriched uranium dioxide (UO_2) cylindrical ceramic pellets, and a round wire Type 302 stainless steel compression spring located at each end of the fuel column, all encapsulated within a ZIRLO^{TM} tube. The fuel rods are internally pressurized with helium during assembly. The applicant stated that the internal pressurization, by reducing the stresses from differential pressure, postpones creep collapse beyond the required service life of the fuel and improves the thermal conductivity of the pellet-to cladding gap.

The fuel cladding is cold-worked and stress-relief-annealed ZIRLO^{TM} tubing with a nominal wall thickness of 0.5715 millimeters (mm) (0.0225 inches [in]). The applicant stated that the UO_2 pellets are concave at both ends to better accommodate thermal expansion and fuel swelling. The nominal density of the UO_2 in the pellets is 10.44 grams per cubic centimeter (g/cc) (0.3771 pounds per cubic in [lb/in^3]), which corresponds to 95.25 percent theoretical density.

The plenum spring at the top of the fuel pellet column keeps the column in its proper position during handling and shipping. The applicant stated that the fuel rod plenum, which is located above the pellet column, allows space for axial thermal differential expansion of the fuel column and accommodates the initial helium loading and evolved fission gases.

4.2.2.3 *Burnable Absorber Rod Description*

To reduce the beginning-of-life moderator coefficient, the applicant included fixed burnable neutron absorber rods in selected fuel assemblies, replacing fuel rods at selected locations. The burnable absorber rod is mechanically similar to fuel rods but consists of gadolinium oxide (Gd_2O_3) admixed in enriched UO_2 in the central rod portion (axially) and enriched UO_2 at the top and bottom. The total column length is the same as the column length of the fuel rods.

4.2.2.4 *Control Element Assembly Description*

The CEAs consist of either 4 or 12 neutron absorber elements arranged to engage the peripheral guide tubes of fuel assemblies. The applicant stated that these elements are connected using a spider structure that couples to the control element drive mechanism (CEDM) drive shaft extension. The neutron absorber elements of a four-element CEA engage the four guide thimbles in a single fuel assembly. The four-element CEAs will control power distribution and core reactivity in the power operating range. The twelve-element CEAs engage the four corner guide tubes in one fuel assembly and the two nearest corner guide tubes in adjacent fuel assemblies. The twelve-element CEAs provide the core with strong shutdown rods.

The applicant stated that the control elements of a four-element or twelve-element full-strength CEA consist of an Inconel 625 tube loaded with a stack of cylindrical boron carbide (B_4C) pellets, with the exception of the lower portions of the elements, which contain reduced-diameter B_4C pellets wrapped in a sleeve of Type 347 stainless steel (felt metal).

4.2.2.5 *Design Evaluation*

The applicant summarized the design evaluations of the fuel rod, fuel assembly, and in-core control components. The fuel rod design considers all events expected during normal operations, AOOs, and postulated accidents. The in-core control components design considers events during normal operations, AOOs, and postulated accidents. Some postulated accident events such as reactivity-initiated accident (RIA) events, loss-of-coolant accident (LOCA) events, and anticipated transients without scram events are evaluated in DCD Tier 2, Chapter 15, “Transient and Accident Analyses.”

The applicant summarized the design evaluations for each component and event and stated a conclusion indicating that the appropriate specified acceptable fuel design limits (SAFDLs) are met. Each analysis includes a pointer to a more detailed analysis contained in the referenced fuel design analysis Topical Report, APR1400-F-M-TR-13001-P, Revision 1, “PLUS7 Fuel Design for the APR1400” (ML17237A023).

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC items for this area of review.

Technical Specifications (TS): No TS associated with DCD Tier 2, Section 4.2, are given in DCD Tier 2, Chapter 16, “Technical Specifications.”

4.2.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Revision 6,, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition” (hereafter referred to as NUREG-0800 or SRP), Section 4.2, “Fuel System Design,” and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 4.2. These regulations include:

1. Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and 10 CFR 50.34, “Contents of applications; technical information,” as they relate to the cooling performance analysis of the emergency core cooling system (ECCS), using an acceptable evaluation model, and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

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

3. GDC 2, “Design bases for protection against Natural phenomena,” and GDC 27, “Combined reactivity control systems capability,” as it relates to control rod insertability under postulated accident conditions.

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

5. 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are met, a facility that incorporates the DC is built and operated in accordance with the DC, the

provisions of the Atomic Energy Act (AEA) of 1954, as amended, and the NRC's regulations.

4.2.4 Technical Evaluation

The staff followed the guidance provided in NUREG-0800 Section 4.2 to ensure that (1) the fuel system is not damaged during normal operations and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained.

4.2.4.1 *Design Bases*

(a) Fuel System Damage

Section 4.2 of the APR1400 DCD summarizes the analyses which cover fuel system damage, fuel rod damage, and core coolability. The analyses cover each failure mechanism and provide the applicable SAFDLs and a concluding summary in regards to the ability of the APR1400 DCD fuel system design, based on the PLUS7 fuel assembly, to meet these limits. The fuel system damage criteria addressed by the topical report are:

- Stress/strain limits
- Fuel assembly component fatigue
- Fuel fretting
- Oxidation and hydriding
- Dimensional changes (bowing/growth)
- Rod internal pressure
- Fuel assembly liftoff
- Reactivity control assembly insertability

The APR1400 DCD references Topical Report APR1400-F-M-TR-13001-P, Revision 1, for additional detailed analyses regarding the method of evaluation and the analyses results. The staff's review of the fuel system damage related SAFDLs and detailed analyses for PLUS7 fuel is addressed in the review of APR1400-F-M-TR-13001-P, Revision 1 (ML17237A023).

DCD Tier 2 Section 4.2 identifies FATES as the primary code for analyzing key fuel pin parameters. The staff noted that FATES does not include a burnup dependent thermal conductivity model and, therefore, can under-predict temperatures for burnt fuels. The referenced Topical Report APR1400-F-M-TR-13001-P, Revision 1, addresses this lack of a burnup dependence in FATES by establishing a methodology for calculating fuel temperatures which includes a penalty to cover any underprediction, therefore establishing a conservative result. The staff documented its review of this thermal conductivity degradation (TCD) analysis methodology in the safety evaluation report (SER) for APR1400-Z-M-TR-13001-P, Revision 1 (ML17348A152).

The applicant established a basis for the requested burnup limit of 60,000 megawatt days per metric ton of uranium (MWD/MTU) from the results of the analyses. The staff reviewed the information provided in the APR1400 DCD, Section 4.2, and confirmed that the data provided from the referenced PLUS7 fuel design Topical Report APR1400-F-M-TR-13001-P, Revision 1, covers the requested burnup range.

Based on the staff's review of APR1400-F-M-TR-13001-P, Revision 1, as detailed in the staff's SER, the staff finds that the requested maximum burnup limit is acceptable and that the applicant has met the requirements of GDC 10 in terms of the fuel system damage criteria by ensuring that SAFDLs are not exceeded for normal operation and AOOs.

(b) Fuel Rod Failure

Section 4.2 of the APR1400 DCD summarizes the analyses which cover fuel system damage, fuel rod damage, and core coolability. The analyses cover each failure mechanism and provide the applicable SAFDLs and a concluding summary with regards to the ability of the APR1400 DCD fuel system design, based on the PLUS7 fuel assembly, to meet these limits. The fuel rod failure criteria addressed by the Topical Report APR1400-Z-M-TR-13001-P, Revision 1, are:

- Hydriding
- Clad collapse
- Overheating of cladding
- Overheating of fuel pellets
- Excessive fuel enthalpy
- Pellet/Cladding Interaction
- Fuel rod bursting
- Mechanical fracturing

Per the staff's review of APR1400-F-M-TR-13001-P, Revision 1, as detailed in the staff's SER, the staff finds that the PLUS7 fuel system design meets GDC 10 as it relates to SAFDLs for normal operation and AOOs, and 10 CFR Part 100 as it relates to fission product releases for postulated accidents.

(c) Fuel Assembly Structural Response to External Loads

In July 2017, the applicant submitted Technical Report APR1400-Z-M-NR-14010-P, Revision 2, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading" (ML17228A782), which is based on a new test database and revised analyses. This technical report covers an analysis of (1) the loads imparted onto the fuel assemblies and (2) the determination of fuel assembly strength. The staff followed the guidance provided in Appendix B of SRP Section 4.2 while reviewing the technical report.

(i) Analysis of Loads

The staff reviewed the loading calculation provided in APR1400-Z-M-NR-14010-P, Revision 2, to determine acceptability of the calculated loads on the fuel assembly. The APR1400 methodology is based on the Combustion Engineering methodologies detailed in CENPD-42, which is used to determine the dynamic response of the reactor vessel internals (RVI) including the core plate, and CENPD-178-P, Revision 1, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Loading," which is used for the structural analysis of fuel assemblies based on the RVI dynamic response.

In order to calculate the loads imparted on the fuel assemblies, the applicant performed various tests to determine the structural response characteristics of the PLUS7 fuel assembly. These tests considered the effects of burnup by testing at beginning of life (BOL) conditions and simulated end of life (EOL) conditions. These tests are described in Chapter 5 and Appendices A and B of APR1400-Z-M-NR-14010-P, Revision 2. The fuel assembly tests performed included:

1. lateral stiffness (to verify the static lateral characteristics of the fuel assembly models);

2. forced vibration (to obtain the BOL and EOL fuel assembly higher mode natural frequencies and mode shapes);

3. pluck vibration (to provide the BOL and EOL 1st mode natural frequencies and critical damping ratios for the pluck impact model);

4. lateral impact (simulates assembly to core shroud impact); and

5. axial stiffness (to determine guide thimble and holddown spring stiffness values).

The tests covered air, still water, and flowing water conditions. The results of these tests were used to define the fuel assembly structural response in the analysis code.

The staff reviewed the testing methods and results as presented in the technical report and found that the referenced methodology was followed and that the results would cover both BOL and EOL conditions. By covering the effects of burnup, the applicant addressed any potential impacts as described in information notice IN-2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength."

In Section 5.2.5 of APR1400-Z-M-NR-14010-P, Revision 2, the applicant presented a discussion of flowing water damping tests. The core coolant acts to dampen the fuel assembly structural response and the amount of damping is proportional to the coolant velocity. The applicant credited the damping caused by flowing coolant to offset the margin lost caused by EOL conditions. Appendix C of the technical report presents the flowing water damping test matrix and results.

The staff reviewed the tests as presented in Appendix C of the technical report to determine if the proposed damping credit was justified. The staff first reviewed the water velocity being credited for flow damping. [The chosen flow rate is based on the velocity at the time of control rod insertion. This is the minimum flow rate before the rods are fully inserted and corresponds to the minimum damping credit.] This is conservative and the staff finds it to be acceptable. The staff then reviewed the testing conditions including the various initial displacements and temperatures. The PLUS7 fuel assembly test results showed no strong temperature dependence, however, the tests could not be performed at normal operating temperatures. The applicant used test data from a similar fuel assembly which had been tested at the same temperature as one of the PLUS7 fuel assemblies and also at the normal operating temperature. The staff reviewed the similarities and differences between the PLUS7 fuel assembly and the related fuel assembly, and concluded that there were no design differences which would make the higher temperature test data invalid for use in developing a temperature dependence curve for the PLUS7 fuel assembly.

Based on the staff's review (as detailed above) of the assumed flow rate, damping curve, and temperature correction, the staff finds the flow damping credit to be acceptable for use as an input in calculating the loads imparted on the fuel assemblies due to seismic or LOCA excitations.

The seismic analyses of RVI were performed separately for the two horizontal and one vertical directions and the results were then combined by the square root of the sum of the squares (SRSS) method for structural evaluation. This method of combining spatial components is consistent with the guidance in NRC Regulatory Guide (RG) 1.92, Revision 3, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." The staff reviewed the APR1400 RVI analysis for seismic excitation as presented in the technical report, and found that the applicant followed NRC guidance, and the results are, therefore, acceptable.

Blowdown loads from pipe ruptures are analyzed in Chapter 4 of APR1400-Z-M-NR-14010-P, Revision 2. The method described in the technical report follows the approved Topical Report CENPD-252-P-A, "Method for the Analysis of Blowdown Induced forces in a Reactor Vessel," July 1979.

Tables 6-1 and 6-2 of APR1400-Z-M-NR-14010-P, Revision 2, identify the safe shutdown earthquake (SSE) and blowdown load combinations used in the stress intensity analyses for PLUS7 fuel assembly components. These load combinations follow the guidelines presented in NUREG/CR-1018, Revision 3, and are therefore acceptable.

(ii) Determination of Strength

APR1400-Z-M-NR-14010-P, Revision 2, presents the determination of strength for grids and other fuel assembly components.

Two types of grid crush testing are used to determine grid buckling strength. The one-sided drop test is used to determine the one-sided crush strength (representing grid-to-barrel impacts), and the through-grid long pulse test is used to determine the through-grid crush strength (representing grid-to-grid impacts). These tests follow the methodology outlined in CENPD-178-P, Revision 1. The staff audited the grid testing program, including results from both through-grid and one-sided drop tests. Additionally, the staff audited the test plan and reports and confirmed that the tests followed the methodology provided in CENPD-178-P, Revision 1.

Section 2.2.2 of Topical Report APR1400-F-M-TR-13001-P, Revision 1, provides the acceptance criteria for fuel assembly components other than grids. These criteria are based on ASME [American Society of Mechanical Engineers] Boiler Pressure Vessel (BPV) Codes Section III. The fuel assembly components other than guide tubes are based on Service Level D limits. Additionally, the fuel assembly components for CEA insertion are evaluated using normal operation and AOO stress limits to ensure CEA insertability, even for SSE and LOCA conditions.

The use of these limits follows the guidance provided in SRP Section 4.2, Appendix A, and the staff therefore finds that the chosen limits support compliance with GDC 2.

(iii) Analysis of PLUS7 Fuel Assembly Structural Response

Technical Report APR1400-Z-M-NR-14010-P, Revision 2, Section 6, presents the structural analysis of the PLUS7 fuel assembly due to external forces based on the PLUS7 specific fuel assembly characteristics generated through the testing program evaluated in parts (a) and (b) of this section. The evaluations cover fuel rods, guide thimbles, bottom nozzle, top nozzle (outer guide post), top nozzle (adapter plate), and top nozzle (holddown plate). The staff reviewed the stress results for each component, which demonstrate that no stress limits were exceeded.

(iv) Summary of the Review of the Fuel Assembly Structural Response to Externally Applied Loads Analysis

Based on the staff's review of the load calculations, strength limits, and calculation results, as detailed above, the staff finds that the applicant has complied with GDC 2 by demonstrating that the fuel system is designed to withstand the effects of natural phenomena without the loss of capability to perform its safety functions.

(d) Core Coolability

Sections 3.2.13 through 3.2.16 of Topical Report APR1400-F-M-TR-13001-P, Revision 1, cover the following core coolability criteria:

- Cladding embrittlement
- Violent expulsion of fuel
- Generalized cladding melting
- Fuel rod ballooning

The staff's review of the core coolability criteria for the PLUS7 fuel assembly is contained in the SER for Topical Report APR1400-F-M-NR-13001-P, Revision 1.

Technical Report APR1400-Z-M-NR-14010-P, Revision 2, addresses the core coolability criteria of structural deformation as discussed previously in Section 4.2.4.1 of this SER. As detailed in the staff's review of the fuel assembly structural response to externally applied loads, the analysis demonstrates that no structural deformation of the PLUS7 fuel assembly is calculated to occur for combined SSE and blowdown loads.

Based on the staff's safety evaluation for Topical Report APR1400-F-M-TR-13001-P, Revision 1, and the staff's review of Technical Report APR1400-F-M-NR-14010-P, Revision 2, the staff finds that the applicant meets the requirements of GDC 27 in terms of control rod insertability and GDC 35 and 10 CFR Part 50.46 in terms of core coolability.

4.2.4.2 *Description and Design Drawings*

The staff reviewed the fuel system description and design drawings provided by the applicant in the APR1400 DCD, Section 4.2. Additional fuel assembly design information is available in the referenced Topical Report APR1400-F-M-TR-13001-P, Revision 1. The staff found that the applicant followed the guidance provided in NUREG-0800, Section 4.2, in terms of providing an acceptable fuel system description.

4.2.4.3 *Design Evaluation*

(a) Operating Experience

As noted in the referenced Topical Report APR1400-F-M-TR-13001-P, Revision 1, PLUS7 fuel assemblies have already been in use in South Korea. The applicant used this operating experience in the topical report to justify the models used to analyze the PLUS7 fuel assembly for use in the APR1400 plant design. The review of the models is part of the staff's SER for APR1400-F-M-TR-13001-P, Revision 1.

(b) Analytical Predictions

The analytical predictions used to analyze certain design bases and related parameters are covered in Topical Report APR1400-F-M-TR-13001-P, Revision 1. The staff's evaluation of these analytical predictions is covered in the staff's associated SER.

(c) Testing, Inspection, and Surveillance Plans

NUREG-0800, Section 4.2.I, Item 4, provides review guidance regarding testing, inspection, and surveillance plans. The PLUS7 fuel design has been used in Korean plants for several years and the applicant used this operational experience and associated post irradiation testing as a basis for the limits and analyses presented in Topical Report APR1400-F-M-TR-13001-P, Revision 1. The staff's SER for the topical report addresses this review area. APR1400 DCD, Section 4.2, discusses the APR1400 inspection and surveillance plans that would cover the fuel assemblies. Because the applicant's referenced PLUS7 fuel design is already in use at other plants, the staff finds that the visual examination program is sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. However, the staff noted that DCD Tier 2 Section 4.2.4 does not refer to a continuing generic PLUS7 surveillance program. Accordingly, based on the guidance provided in NUREG-0800, Section 4.2.I, Item 4, the staff issued RAI 275-8294, Question 04.02-1 (ML15300A492), requesting the applicant to provide descriptions of any ongoing generic fuel surveillance program for the PLUS7 fuel that would preclude the need for a more detailed plant-specific program.

In its response to RAI 275-8294, Question 04.02-1 (ML16035A517), the applicant provided further clarification regarding the surveillance program, including the in-service inspection surveillance plan. The response included markups of the draft DCD Tier 2 Section 4.2, which incorporated the additional description of the visual examinations performed on a cycle-by-cycle basis. The staff reviewed the RAI response and draft markups and determined that the information provided in the response followed the guidance provided by SRP Section 4.2

regarding post-irradiation fuel inspection and is, therefore, acceptable. The staff reviewed DCD Revision 1, Section 4.2.4.7 "Inservice Surveillance," and confirmed that the changes indicated in the response to RAI 275-8294 were implemented.

4.2.5 Combined License Information Items

There are no combined operating license (COL) information items associated with Section 4.2 of the APR1400 DCD.

4.2.6 Conclusions

The staff concludes that the fuel system of the APR1400 plant has been designed so that: (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35 in Appendix A to 10 CFR Part 50; and 10 CFR 50.34. This conclusion is based on the following:

1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response and fuel densification have been performed in accordance with: (1) methods that the staff has reviewed and found to be acceptable and (2) the guidelines in Appendix A to SRP Section 4.2. Those analytical predictions dealing with control rod ejection have been performed in accordance with the interim criteria for RIAs in Appendix B to SRP Section 4.2.

2. The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform online fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and the application thereby meets the related requirements of 10 CFR 50.34.

4.3 Nuclear Design

4.3.1 Introduction

The staff reviewed DCD Tier 2 Section 4.3, "Nuclear Design," to verify that the design of the fuel assemblies, reactivity control systems, and reactor core ensures that design limits will not be exceeded during normal operations or AOOs and that the effects of postulated accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The staff's review of the nuclear design focused on the design bases, core power distributions, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality during refueling, stability, pressure vessel irradiation, and the applicant's analytical methods.

4.3.2 Summary of Application

DCD Tier 1: The applicant did not identify any DCD Tier 1 items specifically related to the reactor core nuclear design. However, DCD Tier 1, Section 2.4.6, "Chemical and Volume Control System," and Section 2.5.1, "Reactor Trip System and Engineered Safety Features Initiation," describe requirements associated with the nuclear design.

DCD Tier 2: The applicant provided a description of the APR1400 nuclear design in DCD Tier 2 Section 4.3 that is summarized as follows:

The APR1400 is a pressurized-water reactor (PWR) with a design power output of 3,983 megawatts thermal. The reactor core consists of 241 fuel assemblies. The fuel assembly is a 16x16 square array containing 236 fuel or burnable poison rods, four control rod guide tubes, and one instrumentation guide tube. The fuel rods contain UO_2 pellets of various enrichments, and the burnable poison rods contain gadolinia-urania (Gd_2O_3-UO_2) pellets. The active fuel length is 381 centimeters (12.5 feet).

The applicant's design refueling interval is 18 months with replacement of one-third of the fuel assemblies during each refueling outage. The maximum fuel rod average burnup is 60 gigawatt days per metric ton uranium (GWd/MTU).

DCD Tier 2 Subsection 4.3.2.2, "Power Distribution," describes the reactor core power distributions, including hot channel factors, associated measurements, and operational limits. The applicant presented calculated core power distributions over a broad range of conditions and the axially integrated radial peaking factors used as part of the bases for determining the departure from nucleate boiling ratio (DNBR) limit. The applicant also discussed power level and power distribution control for safe operation and safe shutdown of the reactor during normal operations or mitigation of accident conditions. In addition, the applicant described how the core operating limits supervisory system (COLSS) and the core protection calculator system (CPCS) of the RPS monitor core conditions and help to ensure that the core power distribution is acceptable and that the associated limits are not exceeded.

DCD Tier 2 Subsection 4.3.2.3, "Reactivity Coefficients," identifies the core reactivity coefficients related to fuel and moderator temperature changes and associated moderator density and void variations. These include the fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), moderator density coefficient, moderator nuclear temperature coefficient, moderator void coefficient, and power coefficient. Although the MTC is slightly positive for cold conditions, it is negative under full-power operating conditions at all times in cycle. The power coefficient is negative throughout core life.

The applicant also discussed reactivity control requirements. The APR1400 reactivity control mechanisms are the CEAs, soluble boron, and burnable poison, and the applicant provided calculated reactivity worths for each of these mechanisms. The CEAs control short-term reactivity changes and are the primary nuclear safety control mechanisms. The APR1400 uses two types of CEAs. The full-strength CEAs (FSCEAs) are the main means to bring the reactor to a subcritical state, and the part-strength CEAs (PSCEAs) may be used to regulate the core power distribution during normal operations. The APR1400 uses soluble boron from the chemical and volume control system (CVCS) as a flexible means for long-term reactivity control. Soluble boron and burnable poison control the excess reactivity loaded into the core at beginning of cycle (BOC) to keep the reactor critical or subcritical as required.

The CVCS is also used to maintain reactor criticality safety during refueling. In addition, DCD Tier 2 Section 14.2.10, “Initial Fuel Loading and Initial Criticality,” provides safe loading criteria and procedures meant to ensure that refueling operations do not result in unacceptable subcritical neutron multiplication.

DCD Tier 2 Subsection 4.3.2.7, “Stability,” discusses the stability issue associated with APR1400 reactor operations. Because the negative overall power coefficients in PWRs are an inherently stabilizing factor with respect to power oscillations, the applicant discussed only xenon-induced power distribution oscillations and means to detect and control such oscillations.

The applicant discussed its analytical methods in DCD Tier 2 Section 4.3.3, “Analytical Methods.” The applicant used the DIT and ROCS code suite for its nuclear design calculations. In addition to the discussions on code validation in the NRC-approved topical reports for the DIT and ROCS codes, the applicant summarized further validation benchmarks using measurement data from operating reactors in the DCD. The applicant propagated the code biases and uncertainties identified in the code benchmarking analyses into the core nuclear design parameters. The applicant also discussed analytical methods used to infer reactor power distributions from ex-core neutron flux measurements.

DCD Tier 2 Subsection 4.3.2.8, “Vessel Irradiation,” discusses reactor pressure vessel (RPV) irradiation. Technical Report APR1400-Z-A-NR-14015, Revision 1, “Neutron Fluence Calculation Methodology for Reactor Vessel,” (ML17094A121) provides detailed calculations, including methodology and results.

The initial test program in DCD Tier 2 Section 14.2, “Initial Plant Test Program,” describes initial fuel loading and pre-criticality testing, initial criticality and low power physics testing, and power ascension testing intended, in part, to confirm the nuclear design described in DCD Tier 2 Section 4.3.

ITAAC: As discussed above, no DCD Tier 1 items explicitly related to reactor nuclear design were identified; therefore, no ITAAC were identified. However, the ITAAC in DCD Tier 1, Table 2.4.6-4, “Chemical and Volume Control System ITAAC,” and Table 2.5.1-5, “Reactor Trip System and Engineered Safety Features Initiation ITAAC,” relate to requirements associated with the nuclear design.

TS: The TS associated with DCD Tier 2 Section 4.3 are in DCD Tier 2 Chapter 16, Sections 2.0, “Safety Limits”; 3.1, “Reactivity Control Systems”; 3.2, “Power Distribution Limits”; 3.3, “Instrumentation”; and 4.2, “Reactor Core.”

4.3.3 Regulatory Basis

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

1. GDC 10, “Reactor design,” requires, in part, that SAFDLs are not exceeded during normal operations, including the effects of AOOs.

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

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

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

5. GDC 20, “Protection system functions,” requires, in part, that the protection system be designed to provide automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded during AOOs and to initiate operation of systems and components important to safety under accident conditions.

6. GDC 25, “Protection system requirements for reactivity control malfunctions,” requires, in part, that no single malfunction of the reactivity control systems (this does not include rod ejection or dropout) causes SAFDLs to be exceeded.

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

8. GDC 27, “Combined reactivity control systems capability,” requires, in part, that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions with appropriate margin for stuck rods.

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

10. 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are met, a facility that incorporates the DC is built and operated in accordance with the DC, the provisions of the AEA of 1954, as amended, and the NRC’s regulations.

11. NUREG-0800 Section 4.3 provides additional detail on acceptance criteria related to power distribution, reactivity coefficients, control rod patterns and reactivity worths, and analytical methods and data. Acceptance criteria for analytical calculations of RPV fluence are discussed in RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”

12. RG 1.68, Revision 4, "Initial Test Programs for Water-Cooled Nuclear Power Plants," provides guidance on appropriate initial test programs.

4.3.4 Technical Evaluation

The staff reviewed DCD Tier 2 Section 4.3 and the applicant's referenced licensing topical and technical reports in accordance with SRP Section 4.3 to ensure compliance with the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. The staff also reviewed applicable portions of the TS and the initial test program and confirmed that the relevant design information in DCD Tier 1, including the ITAAC, is consistent with the nuclear design descriptions in DCD Tier 2 Section 4.3. In addition, the staff performed confirmatory analyses for the initial cycle nuclear design using the Polaris code within the SCALE code system for lattice physics calculations and the PARCS/PATHS code package for nodal core neutronic and thermal-hydraulic calculations.

4.3.4.1 *Evaluation of Design Bases*

DCD Tier 2 Section 4.3.1, "Design Bases," describes the applicant's bases for the nuclear design, including applicability of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. The following paragraphs provide the staff's evaluation of the design bases.

The applicant based its core design lifetime on an 18-month refueling interval and replacement of one-third of the fuel assemblies per refueling outage. The applicant stated that it chose the desired average burnup, listed in DCD Tier 2 Table 4.3-1, "Nuclear Design Characteristics," such that the peak burnup is consistent with fuel design limits. Per APR1400-F-M-TR-13001, the maximum fuel rod average burnup is 60 GWd/MTU. The staff's evaluation of the fuel design is in Section 4.2 of this SER.

The applicant stated that the core power distribution and other core operating parameters are controlled such that TS limiting conditions for operation (LCO) will not be violated, ensuring that SAFDLs and other criteria are not exceeded for accidents. The applicant used the results of transient and accident analyses in DCD Tier 2, Chapter 6, "Engineered Safety Features," and Chapter 15, "Transient and Accident Analyses," to determine the LCOs and limiting safety system settings.

Instrumentation and controls are provided to monitor core power distributions and other parameters and maintain them within prescribed operating ranges. The nonsafety-related COLSS and safety-related CPCS continuously monitor the power distribution and other core parameters. Because the TS bases are demonstrated by analyses, and because instrumentation and controls (I&C) provided, the staff finds the fuel burnup and core power distribution design bases provide reasonable assurance that SAFDLs will not be exceeded during normal operations, consistent with GDC 10 and GDC 13. In addition, as discussed in DCD Tier 2 Chapter 7, "Instrumentation and Controls," the RPS uses information from multiple inputs and sensors to provide an automatic reactor trip and engineered safety features actuation when required, which is consistent with GDC 20.

As discussed below in Subsection 4.3.4.2 of this SER, the net effect of the reactivity coefficients is negative and tends to compensate for rapid reactivity increases during power operation. This is supported by the acceptable results of the transient and accident analyses. Therefore, the staff finds the reactivity coefficient design bases consistent with GDC 11.

Xenon-induced power distribution oscillation is a phenomenon of PWR reactor operation that must be controlled and suppressed to prevent exceeding SAFDLs. The APR1400 is equipped with I&C systems to detect and suppress xenon-induced power distribution oscillations. The COLSS and CPCS provide monitoring capability, and FSCEAs and/or PSCEAs are used to dampen oscillations. Therefore, the staff finds this design basis consistent with GDC 12.

Fuel and burnable absorber depletion characteristics and desired burnup determine the excess reactivity for each cycle, for which a combination of burnable poison, soluble boron, and CEAs is used to compensate. The reactivity worth of the burnable absorbers is intended to ensure net reactivity feedback in accordance with GDC 11. As discussed in Subsection 4.3.4.2 of this SER, the CVCS controls the soluble boron concentration in the reactor coolant system (RCS) to account for long-term reactivity changes due to burnup and xenon transients resulting from reactor load changes. In addition, the CVCS controls reactivity changes due to temperature changes and xenon decay during shutdown and provides adequate shutdown margin during refueling. The CEA reactivity worth under all operating conditions provides margin after cooldown to hot zero power (HZP), accounting for the failure of the most reactive CEA to insert into the core and any additional shutdown reactivity requirements assumed in the accident analyses. Because the CVCS provides long-term reactivity control and can hold the core subcritical under cold conditions, and because the CEAs provide short-term reactivity control with margin for a stuck rod, the staff finds the design bases related to the reactivity control systems, together with those described for the safety injection system (SIS) in DCD Tier 2 Section 6.3, to be consistent with GDC 26 and GDC 27.

The applicant also stated that the maximum reactivity insertion rate from the reactivity control systems, including the CEAs, reactor regulating system, and the boron charging portion of the CVCS, is controlled. The limits include maximum CEA speeds described in DCD Tier 2 Section 15.4.2, "Uncontrolled Control Assembly Withdrawal at Power," and TS 3.1.8, "Charging Flow," which limits CVCS charging flow. These design bases are consistent with GDC 25 and GDC 28 because they provide reasonable assurance that the potential rate and amount of reactivity insertion from normal operation and postulated reactivity accidents do not result in violation of the SAFDLs for any single malfunction of the reactivity control systems, damage to the RCPB, or damage to the core or core internals sufficient to impair core cooling capability.

4.3.4.2 *Nuclear Design Description*

DCD Tier 2 Section 4.3.2, "Description," describes the nuclear design, including the fuel assembly and core design, core power distributions, reactivity coefficients, reactivity control, control rod patterns and reactivity worths, criticality of the reactor during refueling operations, core stability, and vessel irradiation. Section 4.3.2 of this SER summarizes this information.

Operating Strategy

The staff noted that DCD Tier 2, Revision 0, Sections 4.3 and 4.4, referred to load-follow operations and transients. The staff interprets the term "load following" as meaning that the reactor's power output follows the grid demand. Given that definition, the staff determined that additional information would be necessary to approve the APR1400 design for load following, such as the ranges of allowed load-follow power maneuvers and analysis of core and system transients associated with load following. Moreover, the transient and accident analyses do not explicitly consider transient load-follow operating conditions in determining the most limiting initial conditions. The staff issued RAI 293-8332, Question 04.03-4 (ML15314A018), requesting the applicant to either provide the information necessary for the consideration of load following or state that it is seeking approval of the APR1400 design only for baseload operations.

In its responses to RAI 293-8332, Question 04.03-4 (ML16109A196 and ML17107A407, respectively), the applicant clarified that it is requesting approval only for baseload operation, not for load following. In addition, the applicant provided markups of several sections in DCD Tier 2 that revise or eliminate terminology related to load following. The staff finds the responses to RAI 293-8332, Question 04.03-4, and related markups, acceptable because they clarified that the APR1400 is to be operated as a baseload plant, so no additional information for load following is necessary. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 293-8332, Question 04.03-4, is resolved and closed.

Fuel Assemblies and Core Loading Patterns

The applicant provided various data tables and figures to detail the nuclear design of the fuel assembly and reactor core. DCD Tier 2 Table 4.2-1, “Mechanical Design Parameters,” provides key parameters of the fuel and control rod designs, and DCD Tier 2 Figure 4.2-1, “Fuel Assembly,” shows a typical APR1400 fuel assembly. DCD Tier 2, Figures 4.3-1, “First-Cycle Fuel Loading Pattern,” and 4.3-2, “First Cycle Assembly Fuel Loadings with Water Hole and Burnable Absorber Rod Placement,” provide a quadrant-symmetric core loading pattern and the fuel assembly pin-wise fuel enrichment and burnable poison loading patterns for the first cycle, respectively. DCD Tier 2 Figure 4.3-3, “Equilibrium-Cycle Typical Fuel Loading Pattern and Fuel Enrichments,” shows this loading information for a typical equilibrium cycle. The loading patterns consist of a three-batch scheme with a mixed central zone and generally higher-enriched fuel on the periphery, which reduces radial peaking factors. These core loading patterns are the starting point for the nuclear design calculations discussed subsequently in this SER, which show that the presented fuel assembly and core design does not result in exceeding fuel design limits.

The applicant stated that the reactor fuel and core design described in APR1400 DCD Tier 2 Section 4.3 is representative of a typical 18-month cycle initial core and is presented for illustration purposes. The staff accepts this representation on the premise that the nuclear design parameters provided in this section and evaluated by the staff are typical and that any significant deviations, such as increases in the gadolinium content per fuel pin or fuel assembly, fuel rod burnup levels, maximum soluble boron concentration, or design cycle lengths beyond 18 months, will require additional review and approval by the staff.

Power Distributions

The applicant calculated the core power distributions at various core operating conditions using the DIT and ROCS code suite, which is discussed below in Subsection 4.3.4.3 of this SER. The objective of the power distribution calculations is to ensure that the power distribution limits will be met during operation. The APR1400 is designed with the following power distribution limits:

1. The limiting three-dimensional heat flux peaking factor, F_q^n, for hot full power conditions is 2.43, corresponding to the LOCA peak linear heat rate (LHR) of 446.2 watts per centimeter (W/cm) (13.6 kilowatts per foot [kW/ft]) described in DCD Tier 2 Subsection 15.6.5.3.1, “Evaluation Model.” For reference, the APR1400 average LHR is 183.7 W/cm (5.60 kW/ft).

2. A minimum DNBR of 1.29, as determined using the KCE-1 critical heat flux (CHF) correlation described in DCD Tier 2 Section 4.4, “Thermal-Hydraulic Design.”

The applicant presented several figures showing calculated power distributions for a typical first cycle at various burnup levels. These include planar average radial power distributions for different control rod insertion configurations, unrodded core average axial power distributions, and fuel pin power distributions. The applicant also provided representative planar average radial power distributions for an equilibrium cycle. DCD Tier 2 Table 4.3-10, “Comparison of Rodded and Unrodded Peaking Factors for Various Rodded Configurations,” compares the maximum radial peaking factors without rods and with various rod insertion configurations.

The staff noted that the applicant assumed only unrodded operation for the axial core power distributions in the DCD, which is not fully representative of all possible operating conditions. However, the DCD states that CEA insertion is mainly for power maneuvers and will be limited during normal operation. The staff notes that power dependent insertion limits (PDILs) in the TS restrict the amount of CEA insertion during operation. Therefore, the unrodded axial power distributions represent expected power distributions for most of the operating cycle. Furthermore, the COLSS and CPCS monitor the core power distribution during operation. If use of CEAs were to cause the power distribution limits to be exceeded, operators would be notified and/or an automatic reactor trip would occur. Finally, the CPCS overall uncertainty analysis accounts for the effects of rodded operation and transient conditions, such as xenon effects, by comparing DIT and ROCS-simulated power distributions assuming these effects against CPCS algorithm calculations and taking the difference as an uncertainty.

The figures and tables mentioned above show that the applicant considered different times in cycle, different rod configurations for radial power distributions, and both initial and equilibrium cycles in its power distribution analyses. The staff therefore concludes that the applicant has performed a comprehensive set of expected power distribution calculations. The staff notes that the radial power distributions are reasonable considering the fuel loading patterns, CEA locations, and peaking factors. The staff also notes that the representative axial power distributions are reasonable based on comparison with those for similar designs and because the axial power shape shifts as expected from an initial cosine shape to a high saddle-shaped curve due to the change in fission density resulting from fuel burnup in the center of the core. In addition, the staff’s confirmatory analysis shows reasonable-to-excellent agreement between the applicant- and staff-calculated power distributions. The root mean square difference between the applicant and staff radial power distributions was within 2 to 5 percent depending on time in cycle, and minimal differences were observed between the staff and applicant prediction of the highest-powered assemblies. The differences in axial power distribution peaking were up to 10 percent depending on time in cycle, which is consistent with the PARCS assessment against plant data. Furthermore, the presented power distributions and peaking factors remain within the aforementioned power distribution limits. Therefore, the staff considers the representative power distribution data acceptable.

The COLSS, discussed in DCD Tier 2 Section 7.7, “Control Systems Not Required for Safety,” continuously evaluates the margin to core power operating limits, the core axial shape index (ASI), and azimuthal flux tilt against TS and notifies operators via control room alarms if the LCOs are exceeded. The inputs to the COLSS include in-core neutron flux, RCS data, and CEA positions based on the digital rod control system (DRCS) pulse counters. The in-core neutron flux measurement system includes 61 in-core instrumentation assemblies, each with five self-powered rhodium detectors.

The CPCS of the RPS, discussed in DCD Tier 2 Section 7.2, "Reactor Trip System," continuously monitors the core power distribution and DNBR. The CPCS computes DNBR and local power density (LPD) based on several input parameters, including ex-core neutron flux (as measured using three axial levels of detectors at four radial core locations), RCS data, and information from reed switch position transmitters that indicate CEA position. If the low DNBR or high LPD setpoints are exceeded, the CPCS generates a trip signal.

The applicant accounted for uncertainties in power level measurement as well as COLSS and CPCS calculations when monitoring margin to power distribution limits. These uncertainties are addressed below in Subsection 4.3.4.3 of this SER. COLSS alarm setpoints are discussed in Technical Report APR1400-F-C-NR-14002, Revision 1 "Functional Design Requirements for a Core Operating Limit Supervisory System for APR1400," (ML17094A132) and CPCS setpoint methodology is discussed in Technical Report APR1400-F-C-NR-14001, Revision 2 "CPC Setpoint Analysis Methodology for APR1400." (ML18044B044) Sections 7.2, "Reactor Trip System," and 7.7, "Control Systems Not Required for Safety," of this SER provide the staff's evaluations of the CPCS and COLSS, respectively.

The APR1400 provides various methods of controlling the power distribution. The core loading pattern and burnable absorber distribution are determined prior to operation. During operation, regulating FSCEAs or PSCEAs may be used to control power shapes during power maneuvers and to suppress xenon-induced axial power oscillations. Soluble boron may also be used to decrease CEA insertion and create a more uniform radial power distribution. Finally, the COLSS provides a means for operators to become aware of conditions that cause degradation of power-distribution-related margins so they may take appropriate actions.

DCD Tier 2 Subsection 4.3.2.2.2, "Nuclear Design Limits on the Power Distribution," discusses how the power distribution design limits are used as design inputs in the transient and accident analysis initial conditions. These limits are enforced in TS and ensure sufficient margin to accommodate design basis events. The TS relevant to the power distribution are in Section 3.2 of the TS and cover LHR, planar radial peaking factors, azimuthal power tilt, DNBR, and ASI. Safety Limit 2.1.1.1 also specifies the minimum DNBR.

The staff reviewed the power distribution data together with the design limits, means of monitoring and controlling the power distribution, and the associated TS. The staff notes that most of this information is similar to that presented for the staff-approved Combustion Engineering (CE) System 80+. As described above, the staff concludes that the representative power distributions and peaking factors are acceptable. In addition, the APR1400 has instrumentation to monitor the power distribution and methods to control it. For these reasons, the staff has reasonable assurance that the power distributions will remain within design limits throughout normal operations. Therefore, the staff finds that the APR1400 power distribution information meets GDC 10 and GDC 13.

Reactivity Coefficients

The applicant calculated the fuel and moderator reactivity coefficients using the DIT and ROCS code suite. DCD Tier 2 Table 4.3-4, "Reactivity Coefficients," presents the applicant's calculated reactivity coefficients at various states of reactor operations.

The FTC, or fuel Doppler coefficient, is the change in reactivity per unit change in fuel temperature. DCD Tier 2 Figure 4.3-30, "Fuel Temperature Coefficient vs. Effective Fuel Temperature," shows that the calculated FTC is negative for the range of fuel temperatures, as the staff expects; an increase in fuel temperature will cause resonance broadening, reducing reactivity.

The MTC is the change in reactivity per unit change in moderator temperature. DCD Tier 2, Figures 4.3-31, “Moderator Temperature Coefficient vs. Moderator Temperature at BOC HFP [Hot Full Power] Equilibrium Xenon and Fuel Temperature,” and 4.3-32, “Moderator Temperature Coefficient vs. Moderator Temperature at EOC HFP Equilibrium Xenon and Fuel Temperature,” show the MTC at BOC and end of cycle (EOC), respectively, for different operating conditions as well as bounding values used in the transient and accident analyses. Although the MTC is slightly positive at BOC for cold conditions, it is negative for all power operating conditions. The staff notes that the MTC values are consistent with theoretical trends: for example, that the MTC should be more negative at EOC due to reduced soluble boron concentration.

The applicant also discussed the moderator density coefficient, moderator nuclear temperature coefficient, moderator pressure coefficient, and moderator void coefficient. The moderator density coefficient is the change in reactivity per unit change in moderator density. As shown in DCD Tier 2 Figure 4.3-33, “Moderator Density Coefficient vs. Moderator Density at BOC,” it is generally positive and becomes less positive as the boron concentration increases. The staff expects this behavior because increasing density leads to increased moderation and higher reactivity; however, as density increases, boron solubility increases and contributes a slightly negative component. The moderator pressure and moderator void coefficients are closely related to the moderator density coefficient. The applicant described the moderator nuclear temperature coefficient, defined as the change in reactivity per unit change in core average moderator temperature at constant moderator density, as the difference of the MTC and moderator density coefficient.

Finally, the applicant discussed the power coefficient, which is the change in reactivity due to change in core power level. The applicant calculated the power coefficient using the FTC and MTC; the staff finds this appropriate since these coefficients are the primary contributors to the power coefficient. The applicant used the FATES3B code, which is discussed in DCD Tier 2 Section 4.2 and Topical Report APR1400-F-M-TR-13001Revision 1, to calculate the fuel temperature dependence on the LHR for subsequent use in the power coefficient calculation.

The staff notes that FATES3B does not account for the effect of TCD as a function of burnup. The fuel temperature increase due to TCD could affect the FTC and overall core reactivity predictions. However, the staff notes that the applicant has implicitly captured the effects of TCD by benchmarking the nuclear design methodology against operating plant data at various burnups. Therefore, the nuclear design calculations, including inputs to the transient and accident analyses, account for the effect of TCD.

DCD Tier 2 Figure 4.3-34, “Fuel Temperature Contribution to Power Coefficient,” shows the fuel temperature contribution to the power coefficient to be negative at all times, as expected. As listed in DCD Tier 2 Table 4.3-4, and as verified by the staff’s confirmatory analysis, the power coefficient is negative throughout the cycle as required by GDC 11.

DCD Tier 2 Table 4.3-1 presents the reactor kinetics parameters; i.e., effective delayed neutron fraction and neutron lifetime. The staff notes that the presented values are consistent with the typical values for a light-water reactor using low enriched fuel.

DCD Tier 2 Table 4.3-3, “Comparison of Core Reactivity Coefficients with Those Used in Various Accident Analyses,” shows that the reactivity coefficient values used in the transient and accident analyses bound the most positive expected values at BOC and the most negative expected values at EOC, as appropriate, which is conservative with regard to maximizing reactivity feedback. This, in part, satisfies GDC 28 regarding reactivity insertion rates.

The MTC and power coefficient are verified during startup testing as described in DCD Tier 2 Section 14.2. In addition, TS 3.1.3 prescribes that the MTC shall be within the limits in the core operating limits report (COLR) as well as a maximum positive value for the MTC.

Based on the characteristics of the reactivity coefficients and the staff’s confirmatory analysis, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity in the power operating range. In addition, the calculation methodology covers full ranges of operating conditions. Finally, the applicant assigned suitably conservative reactivity coefficient values that will be confirmed during operation and at startup. Therefore, the staff has reasonable assurance that the APR1400 nuclear design meets GDC 11.

Reactivity Control Provisions and Requirements

DCD Tier 2 Section 4.3 describes the APR1400 reactivity control features, which consist of:

1. CEAs to control fast changes in core reactivity, such as reactor shutdown.

2. Soluble boron to control relatively slow changes in core reactivity, such as fuel depletion and xenon transients, and to hold the core subcritical under cold conditions.

3. Fuel with gadolinia burnable absorber pellets to hold down excess reactivity in place of soluble boron; this limits the positive MTC at BOC.

The APR1400 is equipped with 81 FSCEAs and 12 PSCEAs. The neutron absorber in the FSCEAs is B_4C. The PSCEAs are made of Inconel steel. The applicant designated 36 of the FSCEAs as shutdown CEAs and grouped them into two banks. The remaining FSCEAs are grouped into five regulating banks. DCD Tier 2 Figure 4.3-36, “Control Element Assembly Group Assignment,” shows the CEA locations in the core and their group assignments.

The CEAs are the primary nuclear safety control mechanisms, particularly the FSCEAs. Although a major function of the PSCEAs is to assist in the control of core power distribution, the PSCEAs can also provide reactivity control to compensate for minor variations in moderator temperature and boron concentration during power operations and to assist in compensating for changes in reactivity from power level and xenon during power maneuvers.

Because the regulating CEAs may be inserted into the reactor for extended periods of time and at various depths during power operations, the staff became concerned that FSCEAs used as regulating rods may lose their worth over time due to boron-10 (B-10) depletion. The loss could ultimately reduce the safe shutdown margin or impair the ability of the CEAs to safely shut down the reactor. Therefore, on June 23, 2015, and December 28, 2015, staff issued RAI 48-7943, Question 04.03-2 (ML15174A082), and its follow-up RAI 293-8332, Question 04.03-5 (ML15362A588), requesting additional information on CEA worth loss estimates, impacts on safety analyses and CPCS and/or COLSS setpoints, and the potential need for service limits for regulating CEAs.

In its response to RAI 48-7943, Question 04.03-2 (ML15203A536), the applicant provided B-10 burnout estimates and a summary of physics test results from operating Optimized Power Reactor 1000 (OPR1000) cores that are similar to APR1400. The estimates showed substantial B-10 burnout near the regulating bank CEA tips over the lifetime of the CEA, and the CEA insertion history assumptions were not shown to be conservative or bounding. The physics tests results showed that total CEA bank worths measured after multiple operating OPR1000 fuel cycles still matched predictions within allowed uncertainty limits even though the predictions neglected B-10 depletion. The staff also audited related calculation notes that further detailed the applicant's bases for the provided B-10 burnout estimates.

In its response to RAI 293-8332, Question 04.03-5 (ML15362A588), the applicant concluded that service limits are not necessary to address B-10 burnout. The applicant first demonstrated that the CEA insertion history assumptions for the burnout estimate mentioned previously are more limiting than the actual operational practice and PDILs and are therefore conservative.

To address the staff's concern about CEA worth and service lifetimes, the applicant stated that the rod worth uncertainty implicitly accounts for B-10 burnout. The staff notes that the rod worth uncertainty is much greater than the B-10 burnout estimate that the applicant provided in response to RAI 48-7943, Question 04.03-2, so the staff has reasonable assurance that the rod worth uncertainty bounds the potential loss of reactivity worth. In addition, startup testing ensures that actual rod worth is consistent with predictions. The staff also notes that DCD Tier 2 Subsection 4.2.3.6.1,"Control Element Assembly," states that CEAs are designed for a 10-year lifetime based on internal pressure, allowable plastic strain of the cladding, and neutron absorber burnup estimates. The staff concludes that this CEA lifetime, in addition to the PDILs, help to limit the loss of CEA worth. The staff also notes that the large rod worth uncertainty and startup testing provide reasonable assurance that loss of CEA worth will not affect the vital reactivity control function of the FSCEAs in the APR1400. For these reasons, the staff finds that the applicant adequately addressed potential loss of control rod worth. Therefore, RAI 293-8332, Question 04.03- 5, is resolved and closed.

The staff reviewed the applicant's calculated FSCEA and PSCEA bank worths in DCD Tier 2 Table 4.3-6, "Worths of CEA Groups (% $\Delta\rho$)," and concludes they are reasonable based on the APR1400 design descriptions and comparison to similar NRC-approved reactor designs. The staff notes that DCD Tier 2, Figures 4.3-39, "Typical Integral Worth vs. Withdrawal at Zero Power, EOC," and 4.3-40, "Typical Integral Worth vs. Withdrawal at Hot Full Power, EOC, Equilibrium Xenon Condition," are consistent with the rod worth data presented in DCD Tier 2 Table 4.3-6. In addition, the staff's confirmatory analysis showed generally excellent agreement between the applicant- and staff-calculated bank worth values. Although the staff's calculations under-predicted total rod worth relative to the applicant, the bank worth differences were well under 10 percent, which is considered satisfactory agreement for PWR core design applications. The staff also reviewed the CEA reactivity allowances and available CEA worth in DCD Tier 2, Tables 4.3-8, "CEA Reactivity Allowances," and 4.3-9, "Comparison of Available CEA Worths and Allowances." The staff verified that the available CEA worth for the most limiting time in the cycle (EOC), excluding the worth of the most reactive CEA and uncertainty, is sufficient to provide safe shutdown capability during normal operations and postulated accidents. The staff confirmed that the transient and accident analyses in DCD Tier 2 Chapter 15 conservatively incorporate CEA worth values and assumptions.

The PDILs will be part of the COLR submitted by a COL applicant. Based on the typical PDILs in DCD Tier 2 Figure 4.3-38, "Typical Power-Dependent CEA Insertion Limit," the shutdown banks are fully withdrawn prior to withdrawal of any control bank during reactor startup, as specified in TS 3.1.5. This figure also illustrates the regulating group insertion order and the 40 percent fixed overlap between successive regulating groups. In addition, TS 3.1.6 and TS 3.1.7 restrict regulating FSCEA and PSCEA groups, respectively, to the withdrawal sequence, insertion limits, and associated time restraints specified in the COLR. The staff finds that the PDILs conform to the proposed shutdown margin in TS 3.1.1.

DCD Tier 2 Table 4.3-1 provides the critical boron concentration at BOC for various power levels as well as the soluble boron content required for refueling and holding the core 5-percent subcritical under cold and hot conditions. DCD Tier 2 Figure 4.3-47, "Critical Boron Concentration vs. Core Average Burnup Unrodded Full Power, Equilibrium Xenon," shows the decrease in critical boron concentration with burnup. The staff's confirmatory critical boron concentration calculation agrees well with the applicant's at BOC and EOC and exhibits the same general trend throughout core life. The staff's calculation predicts a significantly higher critical boron concentration during mid-cycle, likely due to the different handling of isotopes such as gadolinia or to fuel and moderator conditions within the respective computer codes. However, these differences do not warrant a safety concern since mid-cycle is not limiting in safety analysis, and good agreement is observed at the limiting times in cycle (BOC and EOC).

DCD Tier 2 Section 4.2 and Topical Report APR1400-F-M-TR-13001 describe the gadolinia integral burnable absorber rods, and DCD Tier 2 Table 4.3-1 lists burnable absorber worth at BOC for cold and hot conditions. The burnable absorber worth becomes negligible at EOC.

Based on the preceding discussion on reactivity control provisions and requirements, the staff finds that the applicant clearly identified control needs at various operating conditions and times in cycle and the available negative reactivity insertion when needed. The staff finds the predicted reactivity worths reasonable and their use in the transient and accident analyses appropriate. The staff also concludes that the APR1400 is designed with adequate shutdown margin considering uncertainties and single malfunctions of the reactivity control systems.

Therefore, the staff has reasonable assurance that the APR1400 meets GDC 25 and GDC 26. Compliance with GDC 27 is mainly based on the results of the transient and accident analyses in DCD Tier 2, Chapter 15. The results show that the reactivity control systems, in conjunction with the ECCS, can reliably control reactivity changes to assure that core coolability is maintained assuming the highest-worth CEA is stuck in the fully withdrawn position.

In addition, the applicant discussed limits on the maximum reactivity insertion rate to demonstrate compliance with GDC 28. The applicant imposed limits on CEA speeds, as discussed in DCD Tier 2 Section 15.4.2; the speed and sequence of regulating rod movement; and dilution of soluble boron in the charging portion of the CVCS, as enforced by TS 3.1.8. The applicant used appropriate reactivity worth and reactivity coefficient values in the transient and accident analyses. Therefore, the staff has reasonable assurance that the maximum reactivity insertion rates meet GDC 28.

Reactor Criticality During Refueling

DCD Tier 2 Table 4.3-1 provides the soluble boron concentrations during refueling, and the applicant stated that these concentrations assure that the core effective multiplication factor does not exceed 0.95. In addition, DCD Tier 2 Subsection 14.2.10.1.1, "Safe Loading Criteria," provides specific criteria for safe loading of fuel. These criteria require fuel loading to stop immediately if, during any single loading step, the neutron count rate from either temporary channel unexpectedly doubles or the neutron count rate from any individual nuclear channel increases by a factor of 5. Furthermore, DCD Tier 2, Subsections 14.2.12.2.1, “Initial Fuel Loading,” and 14.2.12.3.1, “Initial Criticality,” prescribe safe methods for initial fuel loading and initial criticality. Because the core effective multiplication factor will be held under 0.95 during refueling and because of the safe loading criteria and methods, the staff has reasonable assurance that the core will remain subcritical during refueling, consistent with the requirement in GDC 26 that one of the reactivity control systems shall be capable of holding the reactor core subcritical under cold conditions.

Stability Criteria

DCD Tier 2 Subsection 4.3.2.7 discusses stability. The applicant stated that the APR1400 nuclear design provides inherent negative reactivity feedback to counter core power perturbations that may lead to local and global instability and therefore discussed control of only xenon-induced power distribution oscillations. Xenon-induced oscillations occur because of rapid perturbations to the power distribution, which cause the xenon and iodine distributions to be out of phase with the perturbed power distribution.

The applicant discussed xenon-induced oscillations of the radial, azimuthal, and axial power distributions and calculated the stability index for each using the ROCS code, which is discussed below in Subsection 4.3.4.3 of this SER. The applicant stated that the radial oscillatory mode is the most stable and demonstrated that a radial oscillation calculation for a System 80 plant, which is similar in design to APR1400, showed that radial instabilities would not occur. The staff notes that the calculated stability index is highly negative, which corresponds to decaying oscillations and system stability.

As shown in DCD Tier 2 Figure 4.3-42, “Expected Variation of the Azimuthal Stability Index, Hot Full Power, Unrodded,” the azimuthal stability index is negative at all times during the first cycle for the expected mode of reactor operation. Therefore, the staff concludes that the APR1400 core is stable against azimuthal xenon oscillations. In addition, the COLSS and CPCS monitor, and TS 3.2.3 enforces, azimuthal power tilt; should an improbable azimuthal xenon oscillation occur, it could readily be detected and addressed.

The applicant stated that axial oscillations could be unstable during the first cycle. DCD Tier 2 Table 4.3-11, “Calculated Variation of the Axial Stability Index during the First Cycle (hr^{-1}),” shows that the axial stability index during the first cycle may vary from -0.0097 to 0.0090 at 100 percent power. Based on the COLSS measurement of the axial power distribution, the operator may move the FSCEAs or PSCEAs to control any axial power oscillations. DCD Tier 2 Figure 4.3-43, “PSCEA Controlled and Uncontrolled Xenon Oscillation,” depicts a simulated axial xenon oscillation both uncontrolled and controlled with PSCEAs during the expected most unstable time in the first cycle (EOC). The figure shows that the ASI oscillates between -0.9 to 0.8 for the uncontrolled oscillation but remains at an almost steady value of -0.1 for the controlled oscillation. Therefore, the staff has reasonable assurance that the PSCEAs are effective in controlling axial oscillations. The CPCS also monitors the axial power shape and generates a reactor trip if necessary to prevent the fuel from exceeding SAFDLs and to limit the consequences of postulated accidents.

Because the design of the reactor and associated systems are demonstrated to be capable of detecting and suppressing power oscillations, the staff finds that the APR1400 meets GDC 12.

4.3.4.3 *Analytical Methods*

DCD Tier 2 Section 4.3.3 describes the analytical methods used in the APR1400 nuclear design and the benchmarking of those methods against measurements from operating reactors.

DIT and ROCS Code Suite

The applicant used the DIT and ROCS code suite for its nuclear design calculations. The DIT lattice code uses two-dimensional discrete integral transport methods to generate few-group neutron cross sections. The ROCS code then performs two- or three-dimensional neutronics calculations using the nodal expansion method (NEM) to solve the two-group diffusion equations using the DIT-generated cross sections. The applicant stated that the MCXSEC code processes the DIT cross sections for use in ROCS.

The use of DIT and ROCS for nuclear design calculations is consistent with the System 80+ methodology that the NRC staff approved in NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," and the staff notes that the APR1400 nuclear design is similar to that of the System 80+. The applicant referenced the NRC-approved 1983 Topical Report CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," as well as the DIT and ROCS code revisions and modeling refinements described in the NRC-approved 1988 Topical Report CENPD-275-P, Revision 1-P-A, "CE Methodology for Core Design Containing Gadolinia-Urania Burnable Absorbers" (*Legacy Library* Accession No. 9905140145).

CENPD-266-P-A describes the models, analytical methods, capabilities, verification, and level of accuracy of the DIT and ROCS codes, which were intended for use in initial and reload core designs for CE-type reactors. The report also details the logic structure, transport theory methods, and nuclear data used by the codes. To verify the codes, the report demonstrates that physics parameter predictions using DIT and ROCS compare satisfactorily with actual plant measurements. The staff at the time found the DIT and ROCS codes as described in CENPD-266-P-A acceptable for core design and neutronics calculations for licensing actions with the recommendation that, since ROCS was originally based on a higher-order difference (HOD) method, further verification be performed when the NEM is incorporated into the ROCS code to ensure that equivalent calculational biases and uncertainties are obtained with ROCS-NEM compared to ROCS-HOD.

CENPD-275-P, Revision 1-P-A, details the CE analysis methods for gadolinia-urania fuel containing up to 8 weight percent gadolinia, including models, analytical methods, and analysis procedures. It emphasized the changes made relative to the accepted methodology in CENPD-266-P-A for modeling the behavior of gadolinia-urania fuel, such as accounting for self-shielding and spectral changes in the calculation methods and adding physical properties for gadolinia-urania fuel. CENPD-275-P, Revision 1-P-A, also included comparisons of predictions to actual measurements and isotopic analyses. The staff at the time found the methodology in CENPD-275-P, Revision 1-P-A, acceptable for licensing applications for up to 8 weight percent gadolinia and recommended continued comparisons of calculated physics parameters with measured data.

The staff was unable to ascertain further verification of ROCS-NEM, as recommended by the staff. The staff issued RAI 387-8485, Question 15.00.02-3 (ML16032A111), requesting the applicant to verify the ROCS-NEM solver and its impacts relative to the ROCS-HOD solver.

In its response to RAI 387-8485, Question 15.00.02-3 (ML16068A284), the applicant stated that comparisons of NEM and HOD method results in CENPD-266-P-A showed that use of NEM does not adversely affect calculation results and uncertainties. In addition, the applicant compared predicted core design parameters based on NEM against data from critical experiments and operating reactors. These actions both verified the method and provided a set of biases and uncertainties. The applicant stated that the biases and uncertainties in the bias and uncertainty manual, CE-CES-129, Revision 9-P, Methodology Manual – Physics biases and Uncertainties," (March 2003), were determined for ROCS-NEM using the same statistical methods that were used for ROCS-HOD. The applicant provided a table comparing the biases and uncertainties of the NEM and HOD methods, and the staff confirmed that they are similar.

The staff concludes that the applicant's use of ROCS-NEM is acceptable because the applicant generated the biases and uncertainties in CE-CES-129, Revision 9-P, and applied them to the nuclear design calculations by comparing results from ROCS-NEM against experimental and operating reactor data, thereby validating use of the NEM method. Furthermore, these bias and uncertainty values are similar to those derived for the HOD method. Therefore, RAI 387-8485, Question 15.00.02-3, is resolved and closed.

The staff noted that the applicant did not provide adequate details on the MCXSEC code and its use with the DIT and ROCS codes. The staff issued RAI 387-8485, Question 15.00.02-4 (ML16032A111), requesting the applicant to provide details on how the MCXSEC code generates macroscopic cross sections for ROCS.

In its response to RAI 387-8485, Question 15.00.02-4 (ML16068A284), the applicant provided details on the MCXSEC code as quoted from CE-CES-65, Revision 4-P, "User's Manual for MCXSEC" (January 1977). In summary, the DIT code performs multigroup transport assembly calculations and produces an output file for MCXSEC containing two-group mesh region-cross sections. MCXSEC uses these files to generate fine mesh macroscopic cross-section interpolation tables, called tablesets, for ROCS. MCXSEC simply interpolates the DIT cross sections and prepares multidimensional cross-section tablesets for the ROCS code, allowing for functionalization on several key parameters. If tablesets are dependent on more than two parameters (e.g., enrichment, burnup, and control rod insertion), then multiple DIT restart calculations are required. MCXSEC organizes this cross-section data into the order needed by the calculation. The staff concludes that the applicant's use of MCXSEC is acceptable because it is a tool to process cross sections, not a calculational method itself. Therefore, RAI 387-8485, Question 15.00.02-4, is resolved and closed.

Use of ENDF/B-IV Nuclear Data Library in DIT and ROCS

The DIT code uses a nuclear data library containing multigroup cross sections, fission spectra, fission product yields, and other supplemental data. The principal source of data for the library is the evaluated nuclear data files in ENDF/B-IV, which was issued in 1974. CENPD-266-P-A describes three adjustments to the library data to reflect changes to ENDF/B-IV recommended by the Cross Section Evaluation Working Group for ENDF/B-V at the time. Those reported adjustments include an approximately 3 percent reduction in the shielded resonance integral for uranium-238 (U-238), the use of the harder Watt fission spectra for uranium-235 (U-235) and plutonium-239 (Pu-239), and approximately 0.1 percent increases in the fission neutron yield data for U-235 and Pu-239.

The staff noted that ENDF/B-IV and ENDF/B-V were published in 1974 and 1978, respectively, and that many revisions and refinements to the evaluated nuclear data files have been developed since that time. The staff therefore sought to estimate the impacts of subsequent ENDF/B refinements and use resulting insights to help inform its assessment of the applicant's ability to use the DIT and ROCS codes to accurately predict APR1400 core power distributions, control rod worths, and reactivity coefficients.

To assess the overall impact of the cross-section data, the staff first performed a basic criticality benchmark analysis using the SCALE 6.1 computer code for the low enrichment critical experiment LEU-COMP-THERM-001-001 from the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The resulting reactivity using ENDF/B-IV data was about 825 percent millirho (pcm) lower than the results using ENDF/B-VII continuous-energy data, or about \$1.32 in terms of the effective delayed neutron fraction, β_{eff}. The staff also performed a preliminary SCALE confirmatory criticality analysis on arrays of unpoisoned PLUS7 fuel bundles. Based on the analysis results, the staff found the calculated infinite multiplication factor values using ENDF/B-IV and ENDF/B-V compared to ENDF/B-VII libraries to be significantly different (about 1,520 pcm and 590 pcm, respectively).

In addition, the staff performed confirmatory SCALE/TRITON calculations on two-dimensional infinite-array models of each APR1400 fuel assembly type at various burnup levels to estimate differences between MTCs and other reactivity effects calculated with data libraries based on ENDF/B-IV versus the latest release of ENDF/B-VII. The differences were substantial; the largest ones involved the lowest-enriched, unpoisoned PLUS7 fuel assemblies and showed MTCs calculated with ENDF/B-IV that were up to about 40 percent lower (i.e., more strongly negative) at zero burnup than those calculated with ENDF/B-VII. This was of particular concern to the staff because the MTC is least negative at BOC.

DCD Tier 2 Section 4.3.3 states that the applicant used updated DIT/ROCS biases and uncertainties based on core design improvements and additional measurement data from critical experiments and operating reactors. However, based on the presented information, it was unclear to the staff whether the applicant adequately benchmarked the codes for the APR1400 design and PLUS7 fuel.

As a result, the staff sought to further assess the adequacy of DIT and ROCS code predictions as applied to the APR1400 nuclear design by discussing the cross-section data and benchmarking concerns in a public meeting with the applicant held April 28, 2015, and by subsequently asking the applicant for two additional information items. The staff issued RAI 47-7959, Question 04.03-1 (ML15174A080), and RAI 55-7940, Question 04.03-3 (ML15189A488), requesting the applicant to provide additional information concerning the adequacy of DIT/ROCS benchmarking to determine the bias and bias uncertainty specific to the APR1400 and the code's reliance on the ENDF/B-IV-based data library, respectively.

In its responses to RAI 47-7959, Question 04.03-1 (ML15203A999), and RAI 55-7940, Question 04.03-3 (ML15210A141), the applicant explained that the DIT/ROCS bias and bias uncertainties are developed by comparing DIT/ROCS calculation results to measurements from plants with similar core designs, particularly eight CE plants in the United States. The staff audited several of these DIT/ROCS code bias and bias uncertainty calculations (see audit summary at ML17230A260). In addition, the applicant stated that the APR1400 bias and uncertainty manual is identical to that for operating OPR1000 plants in South Korea except for a reactivity bias at the end of reload cycles. The applicant provided comparisons between the APR1400 and OPR1000 to demonstrate applicability of OPR1000 biases and uncertainties to the APR1400 design. The applicant also showed that DIT/ROCS predictions including bias and uncertainty for the OPR1000 plants compared well against plant measurements. These predictions used

ENDF/B-IV cross-section data. Finally, the applicant included a markup of DCD Tier 2 Section 4.3 that further clarifies the benchmarking.

The staff concludes that the applicant adequately assessed the bias and bias uncertainty in the reactivity coefficients, reactivity data, control rod worth, and core power distributions computed with DIT/ROCS and the ENDF/B-IV cross-section library and demonstrated that they are applicable to the APR1400 design. Because this benchmarking accounts for use of the ENDF/B-IV cross-section library, and because a large portion of the operating fleet uses the ENDF/B-IV library in their nuclear design analyses, the staff concludes that the use of this library is acceptable. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 47-7959 , Question 04.03-1, and RAI 55-7940, Question 04.03-3, are resolved and closed.

Based on the available cross-section data in the SCALE code package, the staff believed the ENDF/B-IV cross-section library did not contain data on gadolinium isotopes. The staff noted that the variance in isotopic burnout rates could have non-conservative impacts on nuclear design calculations. Therefore, the staff issued RAI 419-8517, Question 04.03-8 (ML16054A802), requesting that the applicant clarify the source of gadolinium data that were used.

In its response to RAI 419-8517, Question 04.03-8 (ML16214A323), the applicant explained that DIT/ROCS uses ENDF/B-IV isotopic gadolinium data from Brookhaven National Laboratory and included documentation showing the existence of data tapes for separate gadolinium isotopes. The applicant also showed that the isotopic gadolinium thermal absorption cross sections are very similar between ENDF/B-IV and ENDF/B-V. The staff finds the use of the ENDF/B-IV gadolinium data acceptable because it includes the different gadolinium isotopes and, as previously discussed, calculations using it are benchmarked against plant data. Therefore, RAI 419-8517, Question 04.03-8, is resolved and closed.

The applicant also used the HERMITE code to generate axial power shapes that correspond to TS limits from which it ran transient calculations described in DCD Tier 2, Chapter 15. Section 15.0.2 of this SER discusses HERMITE in greater detail.

Methodology for the CPCS

The applicant described a number of calculational methods in DCD Tier 2 Subsection 4.3.3.1.1.4, “Other Analysis Methods,” to determine correction factors that enable the CPCS to effectively use information from ex-core detectors by approximating the detector responses had the detectors been placed at the core periphery. To account for radial power distribution effects of CEA insertion, the applicant described calculating CEA shadowing factors using ROCS-generated power distributions and the DORT code. Each ex-core detector has three sub-channels at different axial locations, so the applicant described using the Monte Carlo N-Particle (MCNP) code to perform adjoint calculations to determine shape-annealing functions (SAF). The SAFs are then used to calculate the ex-core detector response for each sub-channel. Finally, the applicant described use of the ANISN one-dimensional transport code to calculate the temperature shadowing effect due to neutron attenuation differences at different moderator temperatures.

The staff considers DORT and ANISN to be industry standard codes due to their wide availability and longstanding history of use for neutron flux calculations; in addition, the System 80+ design used ANISN for the same purpose. Therefore, the staff finds the use of these two codes for the above-described applications acceptable. The staff notes that MCNP is a commonly used code for radiation transport calculations; however, the staff sought further clarification on how the applicant uses MCNP to determine SAFs at an audit held on January 20 - 21, 2016 (ML16060A413). The staff determined that the information presented at the audit regarding the MCNP methodology and validation in determining SAFs needed to be docketed. The staff issued RAI 419-8517, Question 04.03-7 (ML16054A802), requesting that the applicant provide this information.

In its response to RAI 419-8517, Question 04.03-7 (ML16110A449), the applicant provided Technical Report KEPCO E&C/ND/TR/16-001, "Validation of MCNP Application on SAF Using Test Results from SKN-1," dated March 2016 (ML16110A453), to document the use of MCNP to calculate SAFs. The Shin-Kori Nuclear Power Plant, Unit 1 (SKN-1) is an OPR1000 reactor that originally used the two-dimensional (2-D) DORT code for SAF calculations. However, when SKN-1 updated its ex-core detector design to use significantly smaller detectors, the DORT SAF calculations were shown to be inadequate during startup physics testing due to modeling limitations of the 2-D DORT code. APR1400 will use the same, smaller ex-core detector design. Technical Report KEPCO E&C/ND/TR/16-001 demonstrates that SAF calculations using the 3-D MCNP code for the updated SKN-1 ex-core detectors provide results that agree well with startup physics testing results when comparing ex-core detector responses to in-core power distributions.

In addition, the applicant provided a markup of DCD Tier 2 Subsection 4.3.3.1.1.4 that further defines SAFs, briefly discusses the successful use of the MCNP code for SAF calculations in operating plants, and references Technical Report KEPCO E&C/ND/TR/16-001. The staff finds that Technical Report KEPCO E&C/ND/TR/16-001 and the markup adequately document how MCNP is used and verified in calculating SAFs. In addition, the staff notes that the SAFs, along with the temperature and CEA shadowing factors, are verified at plant startup. Therefore, the staff concludes that the applicant adequately addressed how SAFs are determined and verified. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 419-8517, Question 04.03-7, is resolved and closed.

Additional information on how the ex-core detector responses are converted to axial power distributions for use in the CPCS is discussed in Section 4.4 of this SER.

In conclusion, the NRC previously approved the DIT and ROCS code suite for nuclear design calculations for designs similar to APR1400, and these codes remain valid for use in APR1400 design calculations. The applicable licensing topical reports and the applicant's RAI responses discussed above address the inherent uncertainties in the nuclear design analytical methods. The staff concludes that the applicant appropriately accounted for method biases and uncertainties in the nuclear design analyses and in the relevant TS. The staff also concludes that the applicant's methodologies related to ex-core neutron flux processing are appropriate for such applications. Therefore, the staff finds the applicant's nuclear design analytical methods acceptable.

4.3.4.4 *Reactor Vessel Fluence Methodology and Calculation*

In accordance with SRP Section 4.3, the staff reviewed RPV fluence methodologies and fluence values associated with the development of the pressure-temperature (P-T) limits and the reference temperature (RT) based on the fast neutron fluence at end of life (RT_{PTS}) for the APR1400. Section 5.3 of this SER documents the staff's evaluation of reactor vessel materials, P-T limits, pressurized thermal shock (PTS), and Charpy upper shelf energy.

The staff reviewed the applicant's methodology for calculating neutron fluence to the RPV described in Technical Report APR1400-Z-A-NR-14015, Revision 1. As summarized under the following headings, the staff examined the fluence calculation method, the fluence bias determination method, the fluence parameter uncertainties, the overall fluence uncertainties, vessel surveillance, and the calculated vessel fluence.

Fluence Calculation Method

To obtain a 3-D (an r-θ-z) neutron flux distribution, the applicant performed fixed-source 2-D (r-θ) multigroup DORT transport calculations and combined the resultant r-θ flux distribution with an axial profile. The applicant obtained the axial profile by applying long-term axial power distributions with a bounding axial peaking factor of 1.15. The applicant calculated a multigroup fixed-source fission spectrum by considering the hottest fuel assemblies over many cycles and using an average source spectrum determined by mixing the fission spectra of U-235, U-238, Pu-239, and Pu-241. To prepare multigroup cross sections, the applicant used the BUGLE-93 multigroup cross-section library, which includes 20 gamma groups and 47 neutron groups.

The staff notes that DORT is widely used in the nuclear industry to evaluate RPV neutron flux distributions. Although the applicant's method does not employ the standard method of 3-D flux synthesis, it does use a conservative predicted axial power distribution and accounts for uncertainty in the methodology.

The staff reviewed the nuclide compositions of the modeled reactor materials, the fission spectra data, and the neutron source spectrum data provided in Tables 2-1 and 2-2 of APR1400-Z-A-NR-14015, Revision 1, respectively, by performing hand calculations and comparing the source spectrum with published independent data (Haghighat, Mahgerefteh, and Petrovic, 1995). The staff found no erroneous data but was not able to discern and confirm the nuclide fractions that the applicant used for mixing the fission source spectra. The staff issued RAI 293-8332 (ML15314A018), Question 04.03-6, and in part (a), requested the applicant to state the fission nuclide fractions used in calculating the source spectrum.

In its response to RAI 293-8332, Question 04.03-6, part (a) (ML16119A442), the applicant provided a table listing the number of neutrons per fission and the cycle-averaged relative fission rates for the uranium and plutonium isotopes in its source spectrum. The staff found this data consistent with measurements from the ENDF/B-VII library. For this reason, the staff finds that the applicant used adequate nuclide fractions for calculating the source spectrum. Therefore, RAI 293-8332, Question 04.03-6, part (a), is resolved and closed.

To assess the adequacy of the 2-D mesh used in the applicant's r-θ DORT calculations, the staff compared the applicant's mesh with the various mesh models evaluated in an independent published study (Petrovic and Haghighat, 1996). Based on this comparison, the staff confirmed that the applicant's 2-D mesh is sufficiently detailed.

With regard to the choice of multigroup nuclear cross-section library, the applicant examined the effect of using the newer BUGLE-96 library in place of BUGLE-93 by performing calculations at the inner reactor vessel surface using each library. As shown in Figure 4-1, "Comparison of Fast Neutron Fluxes Using BUGLE-93 and BUGLE-96," of APR1400-Z-A-NR-14015, the resulting fast neutron fluxes obtained from using the two libraries are nearly identical. Therefore, the staff concludes that the use of the older BUGLE-93 library is acceptable.

The staff noted that APR1400-Z-A-NR-14015 does not identify the numerical options (e.g., differencing schemes) the applicant used in its DORT calculations. The staff issued 293-8332, Question 04.03-6, part (b), (ML15314A018) requesting that the applicant indicate what DORT numerical options it used.

In its response to RAI 293-8332, Question 04.03-6, part (b) (ML16119A442), the applicant listed the major DORT calculation options it used. The staff notes that the applicant's selected options are consistent with RG 1.190 and are therefore acceptable. For this reason, RAI 293-8332, Question 04.03-6, part (b), is resolved and closed.

Fluence Bias Determination Method

The applicant used the VENUS-1 benchmark problem to determine the bias due to the nuclear cross sections and the discrete ordinates method used in the DORT code. To determine the 3-D multigroup flux distribution in the VENUS-1 benchmark, the applicant used the standard synthesis approach by performing x-y calculations to obtain a radial distribution and performing r-z and r-dependent DORT calculations to obtain an axial distribution.

The applicant found ratios of measured data to calculation results and used the measured-to-calculated (M/C) ratios to determine the bias in the calculated flux and fluence at the APR1400 pressure vessel. The staff compared the applicant's reported M/C ratios in Table 3-3, "Bias of Each Region for Equivalent Fission Flux," of APR1400-Z-A-NR-14015 to those from a report by the Organisation for Economic Co-operation and Development Nuclear Energy Agency on the VENUS-1 benchmarking study, "Prediction of neutron embrittlement in the pressure vessel: VENUS-1 and VENUS-3 Benchmarks," dated 2000. The staff confirmed that both results are consistent. However, APR1400-Z-A-NR-14015 does not explain how the reported total bias of 6 percent was calculated. The staff issued RAI 239-8332, Question 04.03-6, part (c), requesting that the applicant explain the calculated bias.

In its response to RAI 239-8332, Question 04.03-6, part (c) (ML16119A442), the applicant committed to correct the footnote to Table 3-3 in APR1400-Z-A-NR-14015, to state the biases are M/C values, not calculated-to-measured values. In addition, the applicant clarified that the reported 6 percent bias came from rounding the M/C value for the neutron pad in Table 3-3 of APR1400-Z-A-NR-14015 in the conservative (higher bias) direction. The applicant determined the bias using the region with most deviation from the measurement (the neutron pad region), which is conservative. Based on the conservative approach, the staff concludes that the applicant's choice of 6 percent bias is appropriate. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 239-8332, Question 04.03-6, part (c), is resolved and closed.

Fluence Parameter Uncertainties

APR1400-Z-A-NR-14015 indicates that the applicant performed a detailed uncertainty analysis addressing two main areas: reactor modeling (geometry discretization and material composition, excluding the numerical method and cross-section data already included in the bias calculation) and neutron source determination. In each area, the applicant identified key parameters and determined their impact on the calculated uncertainty.

The staff reviewed the reported ranges of uncertainties caused by different parameters and compared the results with those presented in three published studies (Petrovic and Haghighat, 1996; Wagner, Haghighat, and Petrovic, 1996; and Petrovic, Haghighat, and Wagner, 1997).

The staff found that the selected parameters are consistent, and the resulting uncertainties are conservative. The staff therefore finds the applicant's parameter uncertainty analysis to be complete and appropriately conservative.

Overall Fluence Uncertainty

Table 3-4, "Uncertainty of Vessel Fluence Calculation for the APR1400," of APR1400-Z-A-NR-14015 provides calculated uncertainties for key parameters and a maximum combined uncertainty of 11.6 percent. The applicant chose an overall pressure vessel uncertainty value of ±20 percent, which is the sum of the 11.6 percent uncertainty, the 6 percent bias, and 2.4 percent for extra conservatism. The staff finds that the applicant's overall approach to fluence uncertainty is conservative and therefore acceptable.

Vessel Surveillance

APR1400-Z-A-NR-14015 includes a brief description of the use of in-vessel surveillance capsules. However, the staff was not able to discern from the report the modeled axial locations of the surveillance capsules. The staff noted that the capsule placement should be in the region of highest flux and consistent with the design information in DCD Tier 2 Section 5.3, "Reactor Vessel," for accurate modeling and appropriate fluence and uncertainty values. The staff issued RAI 239-8332, Question 04.03-6, part (d), (ML15314A018) requesting that the applicant provide the locations of the capsules.

In its response to RAI 239-8332, Question 04.03-6, part (d) (ML16119A442), the applicant stated that the surveillance capsule assembly length is about 114 in. (289.86 cm), and the mid-plane of the capsule and active core are aligned. The staff notes that this configuration ensures placement of the surveillance capsule in the high-flux region of the reactor, and the information is consistent with DCD Tier 2 Section 5.3. As a result, the staff finds that the applicant has modeled the axial locations of the surveillance capsules appropriately, and part (d) of RAI 239-8332, Question 04.03-6, is resolved. Section 5.3 of this SER documents the staff's evaluation of the reactor vessel surveillance program.

Calculated Vessel Fluence

The staff reviewed the calculated flux and fluence values presented in Section 5, "Results of Neutron Transport Calculations," of APR1400-Z-A-NR-14015 as well as the information in DCD Tier 2, Table 4.3-5, "Typical Neutron Flux Inside the Reactor Vessel," and Table 4.3-7, "Fast Neutron Fluence at the Reactor Vessel," and found them to show relative behaviors consistent with those reported for similar designs in three independent published studies (Wagner, Haghighat, and Petrovic, 1996; Petrovic, Haghighat, and Wagner, 1997; and Haghighat, Aït Abderrahim, and Sjoden, 2001). The staff notes that the applicant assumed 60 years of operation with a 93 percent capacity factor (or 55.8 effective full power years [EFPYs]) to calculate the limiting calculated fast neutron fluence, 9.5×10^{19} neutrons/cm^2. The assumed 60 year operation time is conservative since a plant license duration is 40 years, but the staff's analysis assumes a license period of 40 years and does not imply that a 60-year operating period is acceptable. The applicant then used the fast neutron fluence to calculate the reference temperature for nil-ductility transition (RT_{NDT}) and RT_{PTS}. As discussed in DCD Tier 2 Subsection 5.3.2.3, "Pressurized Thermal Shock," the applicant calculated RT_{PTS} to be well within the limits prescribed in 10 CFR 50.61. Section 5.3 of this SER presents the staff's evaluation of the calculated RT_{PTS}.

DCD Tier 2 Section 5.3.2, "Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper Shelf Energy Data and Analyses," presents a typical P-T limit curve based on fluence values projected for the assumed 60 year/55.8 EFPY operation time. The staff notes that neutron fluence values are low at beginning of life compared to end of life; therefore, the fluence methodology in APR1400-Z-A-NR-14015 is adequate to quantify conservative RPV material properties and P-T limits for the first 10 EFPYs of operation without plant-specific benchmarking. However, use of the methodology in APR1400-Z-A-NR-14015 to calculate plant-specific P-T limits after the initial 10 EFPYs of operation cannot be justified without benchmarking. Pressure vessel surveillance capsules are to be used for benchmarking and, in accordance with DCD Tier 2 Table 5.3-7, "Capsule Assembly Removal Schedule," the capsules will be removed and analyzed at about 6, 15, and 32 EFPY of operation. To ensure that the fluence methodology is applicable to a specific plant, the staff identified the need for an additional COL information item in DCD Tier 2 Section 5.3 to benchmark the fluence methodology based upon plant-specific surveillance capsule data. Should the benchmark demonstrate that the method is nonconservative, the licensee should recalculate the fluence values based on the benchmarked method. This approach is consistent with past practices.

In a supplemental response to RAI 293-8332, Question 04.03-6, part (e) (ML16119A442), the applicant committed to add a COL Information Item to DCD Tier 2 for a COL applicant to provide plant-specific surveillance data for benchmarking purposes. This adequately addresses the staff's concern regarding plant-specific benchmarking. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 293-8332, Question 04.03-6, part (e), is resolved and closed.

In conclusion, the staff finds that the applicant has followed the guidance in RG 1.190 and used suitably conservative methods and modeling assumptions that are consistent with industry practice to calculate neutron fluence to the RPV.

4.3.4.5 *Initial Test Program*

DCD Tier 2 Chapter 14.2, "Initial Test Program," prescribes a number of core physics tests as part of the initial plant test program. These tests confirm the values of certain key parameters and provide a check on the validity of the analytical methods. The staff identified the following initial tests from DCD Tier 2 Section 14.2 to be applicable to confirmation of the APR1400 nuclear design.

- 14.2.10.1 Initial Fuel Loading
- 14.2.10.2 Initial Criticality
- 14.2.12.1.5 Chemical and Volume Control System Letdown Subsystem Test
- 14.2.12.1.7 Chemical and Volume Control System Charging Subsystem Test
- 14.2.12.1.24 Plant Protection System Test
- 14.2.12.1.25 Ex-Core Neutron Flux Monitoring System
- 14.2.12.1.26 Fixed In-Core Nuclear Signal Channel Test

- 14.2.12.1.27 Digital Rod Control System
- 14.2.12.1.28 Reactor Regulating System Test
- 14.2.12.1.31 Core Operating Limit Supervisory System Test
- 14.2.12.1.32 Reactor Power Cutback System Test
- 14.2.12.2.1 Initial Fuel Loading
- 14.2.12.2.9 Post-Core In-Core Instrumentation Test
- 14.2.12.3.1 Initial Criticality
- 14.2.12.3.3 Isothermal Temperature Coefficient Test
- 14.2.12.3.4 Shutdown and Regulating Control Element Assembly Group Worth Test
- 14.2.12.3.5 Differential Boron Worth Test
- 14.2.12.3.6 Critical Boron Concentration Test
- 14.2.12.3.7 Control Element Assembly Symmetry Test
- 14.2.12.4.1 Variable T_{avg} (Isothermal Temperature Coefficient and Power Coefficient) Test
- 14.2.12.4.3 Control Systems Checkout Test
- 14.2.12.4.7 Shutdown from Outside the Main Control Room Test
- 14.2.12.4.10 Steady-State Core Performance Test
- 14.2.12.4.11 Intercomparison of Plant Protection System, Core Protection Calculator, Information Processing System, and Qualified Information and Alarm System Inputs
- 14.2.12.4.12 Verification of Core Protection Calculator Power Distribution Related Constants Test
- 14.2.12.4.14 Core Protection Calculator Verification
- 14.2.12.4.16 In-Core Detector Test
- 14.2.12.4.17 Core Operating Limit Supervisory System Verification
- 14.2.12.4.24 Pseudo-Ejected CEA Test
- 14.2.12.4.25 Pseudo-Dropped CEA Test

The staff evaluated the above tests against the nuclear design described in DCD Tier 2 Section 4.3 and the guidance in RG 1.68, Revision 4, "Initial Test Programs for Water-Cooled Nuclear Power Plants." As discussed in DCD Tier 2 Subsection 14.2.7.1.8, the APR1400 initial test program does not include a test to demonstrate the capability of the in-core and ex-core neutron flux instrumentation to detect control rod misalignment equal to or less than the TS limits, which is suggested as a power ascension test in Appendix A-5 of RG 1.68. The staff finds this acceptable because the RPS does not rely on neutron flux instrumentation to detect CEA misalignment; instead, there are two independent sets of reed switches that indicate CEA position to the CPCS, which imposes penalties for CEA misalignment on the margin to reactor trip.

The staff finds that the components of the initial test program related to the nuclear design are consistent with DCD Tier 2 Section 4.3 and the guidance in RG 1.68, Revision 4, and are therefore acceptable. Section 14.2 of this SER provides additional evaluation of the above tests.

4.3.5 Combined License Information Items

There are no COL information items directly associated with Section 4.3 of the APR1400 DCD. However, as discussed in Section 4.3.4.4 of this SER, COL Information Item 5.3(2) requires a COL applicant to benchmark the fluence methodology based upon plant-specific surveillance capsule data.

4.3.6 Conclusions

The staff reviewed the following aspects of the APR1400 nuclear design: design bases, design descriptions, core power distributions, reactivity coefficients, reactivity control, control rod patterns and reactivity worth, reactor criticality during refueling, core stability, vessel irradiation, and analytical methods. The staff also performed confirmatory analyses for the first-cycle core design. For the reasons discussed above, the staff finds that the nuclear design is acceptable and meets the applicable requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

4.4 Thermal and Hydraulic Design

4.4.1 Introduction

The staff reviewed the DCD Tier 2 Section 4.4, “Thermal-Hydraulic Design,” to verify that the thermal-hydraulic design of the APR1400 provides adequate heat transfer for reactor fuel and core components such that fuel damage does not occur during normal operations or AOOs. The design must also ensure that the reactor can be safely shut down and kept subcritical with acceptable heat transfer following a postulated accident. In addition, the applicant must show that the thermal-hydraulic design is not susceptible to thermal-hydraulic instability or provide capabilities for detecting and suppressing instability such that the reactor does not exceed the SAFDL. The staff’s review focused on the thermal-hydraulic analytical methods for the reactor core, experiments to validate those methods, and the acceptability of the APR1400 thermal-hydraulic design based on the analysis results.

4.4.2 Summary of Application

DCD Tier 1: The applicant identified no Tier 1 items explicitly associated with the reactor thermal-hydraulic design. However, DCD Tier 1, Sections 2.4.1, "Reactor Coolant System"; 2.4.3, "Safety Injection System"; 2.4.4, "Shutdown Cooling System"; 2.5.1, "Reactor Trip System and Engineered Safety Features Initiation"; and 2.5.3, "Qualified Indication and Alarm System," describe requirements related to the thermal-hydraulic design.

DCD Tier 2: The applicant provides thermal hydraulic design description in DCD Tier 2 Section 4.4 that is summarized here in part, as follows:

The APR1400 thermal-hydraulic design ensures at least a 95 percent probability at a 95 percent confidence (95/95) level that departure from nucleate boiling (DNB) and CHF phenomena will not occur during normal operations and AOOs. The heat flux required for DNB for the fuel design to be used in the APR1400 (PLUS7 fuel) has been calculated using the KCE-1 DNB correlation with coolant condition information provided by the thermal margin analysis using the TORC code, "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core" (CENPD-161-P-A, 1986). The applicant determined that a DNBR limit of 1.29 satisfies the probability and confidence criteria. Topical Report APR1400-F-C-TR-12002-P-A, "KCE-1 Critical Heat Flux Correlation for PLUS7 Thermal Design" (ML17115A556), describes the development and use of the KCE-1 correlation. CENPD-206-P-A, "TORC Code: Verification and Simplified Modeling Methods," documents verification and usage of the TORC code. The applicant modified the TORC code and its fast running derivative, CETOP code, "CETOP Code Structure and Modeling Methods for Arkansas Nuclear One Unit 2" (CEN-214(A)-NP, 1982), to replace the CE-1 correlation with the KCE-1 correlation.

APR1400 thermal margin analyses consider the effects of coolant flow, velocity, and void fraction. Technical Report APR1400-F-C-NR-12001-P, Revision 2, "Thermal Design Methodology" (ML13317A141), recently updated to reflect staff approval of the KCE-1 DNB correlation, describes the applicant's overall thermal design methodology. The primary codes the applicant used are TORC and CETOP, which the NRC previously reviewed and approved for the CE System 80+ DC. The staff safety evaluation for the system 80+ design is documented in NUREG-1462, "Final Safety Analysis Report, Related to the Certification of the System 80+ Design," Vol 1, 1994.

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 DCD Tier 2 Section 4.2.1 "Design Bases." The results of these analyses are used in the evaluation of DCD Tier 2 Chapter 15 accident events.

The applicant used a fuel rod design code (FATES3B) described in Topical Report CEN-161(B) Supplement 1-P that was previously approved by the NRC in 1987 to analyze thermal performance of fuel rods of various burnups and LHRs. The fuel Topical Report APR1400-F-M-TR-13001-P provides a detailed description of the code and the method used in performing these analyses. The applicant's analyses, which account for uncertainties in the model and fuel fabrication, show that the APR1400 fuel centerline temperature will not exceed the fuel melting temperature when accounting for fuel TCD due to burnup and the addition of burnable absorbers.

In addition to the DNB and fuel centerline temperature considerations, DCD Tier 2 Section 4.2.1, describes the design basis that ensures the fuel cladding and fuel assembly integrity are maintained. In DCD Tier 2 Section 7.2, the applicant stated that the digital CPCS in the APR1400 will generate an automatic trip or take other actions to prevent the violation of the design limits. Technical Report APR1400-F-C-NR-14003-P, “Functional Design Requirements for a Core Protection Calculator System for APR1400” (ML15008A440), describes the CPCS. A summary description of the CPCS hardware is provided in DCD Tier 2 Section 7.2, “Reactor Trip System.”

The CPCS infers maximum LPD and makes use of ex-core neutron detectors and CEA position indication to synthesize a “hot” pin and “hot” channel power distribution. The applicant determined the minimum DNBR using power distribution information and other sensor data (inlet temperature, pressure, and core mass flow rate) sent by the CPCS to the plant protection system (PPS).

The COLSS consists of process instrumentation and algorithms used to continually calculate the DNBR margin, LHR margin, total core power, core average ASI, and azimuthal tilt magnitude. The COLSS also continuously compares the calculated values to the LCO on the parameters. If a LCO is exceeded for any of the parameters, COLSS alarms are initiated and operator action is taken as required by the TS.

The APR1400 design limits specify that at least 97 percent of the thermal design flow will pass through the active fuel region of the core to cool the fuel rods. The applicant allotted the remaining up to three percent as bypass flow, which includes flow to the reactor outlet nozzle clearances, alignment keyways, the core shroud annulus, and guide tubes and thimbles. The applicant based the core inlet flow and exit pressure distributions input to the TORC code on System 80 and Hanbit Nuclear Power Plant, Units 3 and 4, reactor flow test data documented in KEPCO E&C/ND/TR/12-020, “APR1400 Core Inlet Flow and Outlet Pressure Distributions Based on Flow Model Test.” The core design of the APR1400 is essentially identical to the System 80 core design. The applicant stated that it used the PLUS7 fuel assembly components testing results to predict core pressure drops and hydraulic loads on internal components. The applicant verified the pressure drops with Palo Verde System 80 startup flow test data.

The applicant evaluated uncertainties associated with pressure drop, hydraulic loads, fuel and clad temperatures, fuel fabrication, DNBR calculation inputs, analytical models, and correlations. The applicant statistically combined the uncertainties and used them to establish operating margins. Technical Report APR1400-Z-J-NR-14004-P, Revision 1, “Uncertainty Methodology and Application for Instrumentation,” (ML17094A177) provides a detailed description of the statistical treatment of uncertainty.

To estimate the total uncertainty in an instrument or measurement, the applicant algebraically combined various random and bias errors. The applicant combined random errors using the SRSS method, and summed bias errors together, before combining the sum with the SRSS result. Similar treatment of the uncertainties was applied to the previously certified System 80+ design.

Thermal-hydraulic instability does not occur during normal operations or AOOs for the APR1400 due to the inherent thermal-hydraulic characteristics of PWRs as indicated in NUREG-1462. NUREG-1462 states that the expected pressures and flows are sufficiently large to suppress fluctuations that could cause flow instability.

To ensure that required net positive suction head (NPSH) is available to the reactor coolant pumps (RCP) under all operating conditions, an operating curve that gives acceptable RCS pressure as a function of temperature is used. The applicant stated that the APR1400 does not allow for power operation with inoperative pumps. Additionally, DCD Tier 2 Chapter 16, TS B 3.4 "REACTOR COOLANT SYSTEM," specifically requires two RCS loops for Modes 1 and 2 with both RCPs in operation in each loop.

The applicant calculated core coolant flow distribution based on flow test programs with geometrically scaled models performed for the System 80 and the Hanbit Nuclear Power Plant, Units 3 and 4. Flow blockage in the vessel core is assumed to occur in the determination of the core DNBR limits. This assumption of blockage helps address the effect of crud buildup in the primary coolant system. Furthermore, the calculation of hydraulic loads account for pressure losses due to core crud.

The applicant stated that a licensee will continuously monitor the flux tilt, that is, the azimuthal imbalance of the radial power distribution in the core, using the COLSS during operation. The azimuthal power tilt is also determined by the COLSS and it is included in the COLSS determination of core margin. Protection limit setpoints allow a certain amount of flux tilt. The operator will monitor these margins and take corrective action if the limits are approached.

The APR1400 design uses the COLSS and CPCS to help ensure that the design basis limits are not exceeded for the DNBR and fuel temperature. The applicant precluded fuel melting by limiting the LPD. These limits are maintained by LCO, which are provided in DCD Tier 2 Chapter 16.

APR1400 thermal-hydraulic-related instrumentation includes CEA position indication, ex-core neutron flux detectors, and reactor coolant flow indicators for determination of thermal-hydraulic conditions. In-core instrumentation assemblies permit three-dimensional flux mapping of the core and include core-exit thermocouples. The ex-core neutron flux monitoring system measures reactor power level from startup levels to 200 percent of full power.

The inadequate core cooling monitoring system monitors hot and cold leg temperatures, pressurizer pressure, core exit temperature, and reactor vessel water level, to identify inadequate core cooling conditions.

The loose parts monitoring system detects and evaluates metallic loose parts that could impact the thermal-hydraulic performance of the RCS through analysis of transient acoustic impact signals caused by the presence of loose parts impacting against the inner wall of the reactor vessel or piping.

The internal vibration monitoring system monitors time dependent ex-core neutron flux signals that indicate changes in neutron absorption path lengths. The changes indicate vibration of reactor internals.

Testing and verification for the thermal-hydraulic design includes RCS flow measurement tests and inspections on manufactured fuel. DCD Tier 2 Section 14.2 describes these tests and inspections.

DCD Tier 2 Table 4.4-1, "Thermal and Hydraulic Parameters," provides the thermal-hydraulic parameters for the reactor. DCD Tier 2 Section 5.1, "Summary Description," provides a summary description of the thermal-hydraulic design of the RCS. DCD Tier 2

Subsection 4.4.3.1, “Plant Configuration Data,” summarizes the drawings and plant configuration data from various sections of the DCD.

ITAAC: No ITAAC associated with DCD Tier 2 Section 4.4 are identified.

TS: The TS associated with DCD Tier 2 Section 4.4 are given in DCD Tier 2 Chapter 16, and in Sections 2.0; 3.2; 3.4, “Reactor Coolant System (RCS)”; 3.5, “Emergency Core Cooling Systems”; and 4.2.

Initial Plant Test Program: The initial test program elements associated with the APR1400 thermal hydraulic design are described in the following subsections of Tier 2, DCD Section 14.2:

Post core loading Hot Functional Tests

- 14.2.12.2.3 Integrity Monitoring System
- 14.2.12.2.4 Reactor coolant system flow measurements
- 14.2.12.2.9 In-core instrumentation test

Power Ascension Tests

- 14.2.12.4.10 Steady-state core performance test
- 14.2.12.4.12 Verification of core protection calculator power distribution related constants test
- 14.2.12.4.14 Core protection calculator verification
- 14.2.12.4.16 In-core detector test
- 14.2.12.4.17 Core operating limit supervisory system verification

4.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are in NUREG-0800, Section 4.4, “Thermal and Hydraulic Design,” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 4.4. The requirements governing ITAAC related to the reactivity control systems are provided in NUREG-0800, Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria.”

1. GDC 1, “Quality standards and records,” GDC 2, “Design bases for protection against natural phenomena,” and GDC 4, “Environmental and dynamic effects design bases,” as they relate to non-safety-related nuclear instrumentation that are in the reactor coolant pressure boundary and shared with other safety systems being classified as Seismic Category I and meeting the applicable Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) Class 1E requirements.

2. GDC 10, “Reactor design,” as it relates to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits are not exceeded during conditions of normal operation, including anticipated operational occurrences.

3. GDC 12, “Suppression of reactor power oscillations,” 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.

4. 10 CFR 50.34(f)(2)(xviii), as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions.

5. 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are 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, as amended, and the NRC’s regulations.

Acceptance criteria adequate to meet the above requirements 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 operations 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 that account for problems affecting fuel performance, such as fuel densification, rod bow, core oscillations, and thermal hydraulic instabilities.

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

4. RG 1.68, Revision 4, as related to the confirmation of the thermal hydraulic design.

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

4.4.4 Technical Evaluation

The staff reviewed APR1400 DCD Tier 2 Section 4.4 including applicable TS, related ITAAC and initial test requirements, and COL information items listed in DCD Tier 2 Table 1.8-2, "Combined License Information Items," 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, GDC 10 and GDC 12 pertaining to fuel thermal margin; 10 CFR 50.34(f)(2)(xviii), as it relates to instrumentation requirements for the detection and recovery from inadequate core cooling conditions; and 10 CFR 52.47(b)(1), as it pertains to ITAAC. Section 4.4 of the SRP provides the review guidance.

The staff reviewed the applicable TS listed in Section 4.4.2 of this SER and found them to be consistent with the descriptions and requirements provided in DCD Tier 2 Section 4.4. The staff also examined the safety limits specified in TS 2.0 to determine if they meet the requirements of GDC 10 and GDC 12. The LCOs specified in TS 3.2 and TS 3.4 provide requirements pertaining to power distribution control and the RCS parameters for the APR1400.

The acceptance criteria used in this evaluation are in SRP Section 4.4. The staff reviewed the completeness of the content of the DCD Tier 2 Section 4.4 against SRP Section 4.4 and the applicable portions of RG 1.206, "Combined License Applications for Nuclear Power Plants." In addition to the material provided in this DCD Tier 2 Section 4.4, the staff reviewed supporting technical reports as necessary to evaluate the design.

The applicant provided a comparison of the APR1400 thermal hydraulic design in DCD Tier 2 Table 4.4-1 with that of the previously-certified C-E System 80+ design. The staff evaluated Table 4.4-1 and found that the APR1400 thermal hydraulic design parameters are comparable to the System 80+, with small differences due to the slight increase, i.e., less than 2 percent, in power of the APR1400.

Thermal-Hydraulic Design of the Reactor Core

The staff reviewed Technical Report APR1400-F-C-NR-12001-P, Revision 2, which summarizes the applicable design computer codes and methodologies supporting the APR1400 DCD thermal-hydraulic design. The applicant used the previously approved codes, TORC and CETOP, to calculate the minimum DNBR in the core that defines the core thermal margin. Both codes are steady-state sub-channel analysis codes. They are also used for potential two-phase regions in the core that could develop during AOOs. The staff noted that previous NRC approval of the TORC and CETOP codes were limited to steady-state applications with single phase flow or homogeneous two-phase flow (such as bubbly flow regime). However, in the staff's SER in CENPD-183-A, "Loss of Flow, C-E Methods for Loss of Flow Analysis" (ML16224A358), which was developed as part of the licensing basis for Palo Verde Nuclear Generating Station (PVNGS), the TORC and CETOP codes were found to produce conservative core thermal margins for quasi-steady state conditions.

The staff noted that the previously approved version of the TORC code required a simplified representation of the core due to limitations on number of flow channels, and further simplification to represent the hot channel. The staff issued RAI 326-8408, Question 04.04-4 (ML15337A168), requesting that the applicant clarify whether the TORC and CETOP codes have been modified by the applicant to eliminate the limitation on number of channels. In its response to RAI 326-8408, Question 04.04-4 (ML16035A007), the applicant stated that the TORC code has been modified over time to increase the dimensional array capacity. However, the methodology is consistent with the thermal-hydraulic formulations described in CENPD-161-P-A. The CETOP code is a simplified version of the TORC code as described in CENPD-206-P-A and is the same code that has been applied at currently licensed U.S. nuclear power plants for predicting minimum DNBR. The applicant used parametric studies to determine the value of the CETOP hot assembly adjustment factor to ensure that the CETOP results are conservative with respect to the TORC results as documented in CEN-214 (A)-NP. The CETOP factor was used for analysis of DCD Chapter 15 AOOs as well as for core protection calculator (CPC) and COLSS core monitoring algorithms. The staff reviewed documentation describing the TORC code modifications during a software quality assurance inspection during February 29, 2016, through March 4, 2016 (ML16081A081). The staff had no findings as a result of the inspection and the staff therefore concluded that use of the TORC and CETOP code versions for APR1400 applications to model core thermal-hydraulics is appropriate.

The applicant’s Technical Report APR1400-F-C-NR-12001-P, Revision 2, provides the statistical thermal margin methodology used to combine system parameter uncertainties in DNBR calculation results to ensure that SAFDLs are not violated at or above a 95 percent probability at 95 percent confidence level. The applicant utilized the previously approved methodology for CE reactors in CEN-356(V)-P-A, Revision 01-P-A “Modified Statistical Combination of Uncertainties,” which is Reference 2 of APR1400-F-C-NR-14001-P, Revision 1.

The staff reviewed Topical Report APR1400-F-C-TR-12002-P, which presents the test data analyses and results for the KCE-1 CHF correlation development for the safety limit of DNBR. The KCE-1 CHF correlation is applied within the TORC sub-channel code to perform these analyses. The staff's evaluation (ML16230A224) concluded that the information presented in the topical report satisfactorily supports the applicant’s proposed DNBR safety limit of 1.124 for APR1400.

The applicant used the methodology from CEN-356(V)-P-A Revision 01-P-A to convert the DNBR safety limit of 1.124 to the DNBR analytical limit of 1.29 for the APR1400 design. The analytical limit is used in the CPCS and it statistically accounts for I&C system uncertainties. The methodology used to convert the DNBR safety limit to the analytical limit uses statistical combination of uncertainties to address uncertainties in both system and state parameters to generate CPCS and COLSS limits that ensure that reactor operation does not result in DNB for normal operations and AOOs with a 95 percent probability at the 95 percent confidence level. The methodology consists of a number of steps, including: (1) verifying that the CETOP code provides conservative DNBR values when compared to the TORC code results; (2) benchmarking CETOP results against reactor operating data over a spectrum of operating conditions for Arkansas Nuclear One, Unit 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; and San Onofre Nuclear Generating Station, Units 2 and 3, to demonstrate conservatism as part of the TORC code verification; (3) applying statistical methods to ensure the 95 percent probability at the 95 percent confidence statistical criteria are satisfied; and (4) applying a penalty for rod bow.

The staff reviewed the applicant’s response to RAI 301-8280, Question 07.01-45, (ML17214A218), and found it to be acceptable, since it clarified that “the application of the setpoints for the CPC is inherently one-sided so that the value of 1.645 is the 95/95 one-sided tolerance limit factor for an infinite number of data points” and staff determined that the setpoint methodology satisfies the regulatory acceptance criteria, guidelines, and the necessary performance requirements. The applicant’s revisions included changes to Technical Report APR1400-F-C-NR-14001-P, Revision 1, “CPC Setpoint Analysis Methodology for APR1400,” and DCD Table 1.9 -1 (12 of 35), with clarifications in DCD Chapters 7 and 15 for exceptions to compliance with Revision 3 of RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," for the CPC Setpoint Analysis Methodology and an update to DCD Table 7.1-1 (3 of 6). The staff has confirmed incorporation of the changes discussed above in Revision 3 to DCD Tier 2 and in Revision 3 to Technical Report APR1400-F-C-NR-14001-P and, therefore, RAI 301-8280, Question 07.01-45 is resolved and closed.

Core Thermal-Hydraulic Calculations

The staff questioned use of the Maurer method in WAPD-BT-19, "A Method of Predicting Steady State Boiling Vapor Fractions in Reactor Coolant Channels," to calculate the core average void fraction and the maximum void fraction. This correlation was developed in 1960, but had not been explicitly reviewed or approved by the staff. The staff reviewed this reference and concluded that the amount of uncertainty could be significant for some transients. The staff issued RAI 326-8408, Question 04.04-1 (ML15337A168), requesting the applicant to justify that the use of the Maurer method results in conservative calculation of thermal hydraulic parameters for core design. Additionally, the staff requested clarification on which computer codes use the Maurer method. In its response to RAI 326-8408, Question 04.04-1 (ML16011A259), the applicant stated that the Maurer model is not used to assess the APR1400 margin to SAFDLs since it is not used in the current core thermal design using the TORC code. The core average void fraction and the maximum void fraction were calculated using the Maurer model to meet RG 1.206 requirements. The Maurer correlation in this application was used to compute supplemental design information as suggested in RG 1.206 (Section C.I.4, "Reactor") outside of licensing calculation use, where the results are primarily for illustrative purposes. Because this correlation is not used for safety margin determination and has no impact on license calculations, the staff considers use of the Maurer correlation acceptable for illustrating general trends of local quality and void fraction in the reactor core per DCD Tier 2 Figures 4.4-4, "Average Void Fractions and Qualities in Core Region," and 4.4-5, "Axial Distribution of Void Fraction and Quality in the Matrix Subchannel with Minimum DNBR."

Irrecoverable pressure losses from the inlet to the outlet nozzles are calculated by the applicant using the classical fluid mechanistic relations and data from the System 80+ reactor flow tests documented in KEPCO E&C/ND/TR/12-020. To calculate pressure drop, either for single phase heating or for subcooled boiling, the friction factor for isothermal flow is modified through the use of the multipliers given by Pyle in WAPD-TM-213, "A Program for the Thermal Analysis of a Pressurized Water Nuclear Reactor during Steady-State Operation, STDY-3." The multipliers were considered on the effects of subcooled voids on the acceleration and elevation components of the pressure drop, as well as the effect on the friction losses. This approach had not been explicitly reviewed and approved by the staff. The staff has previously approved the use of a curve fit to the Martinelli-Nelson two-phase friction factor data in CENPD-161-P-A, but has not documented an evaluation of the Pyle multipliers to account for the effects of subcooled voids on the acceleration and elevation components of the pressure drop. The staff issued RAI 326-8408, Question 04.04-2 (ML15337A168), requesting that the applicant provide justification that the Pyle multipliers on the two-phase friction factors resulted in conservative prediction of thermal hydraulic parameters used in core design. In its response to RAI 326-8408, Question 04.04-2 (ML16011A259), the applicant stated that the two-phase friction factor multiplier for subcooled boiling conditions given by Pyle (as described on page 8 of WAPD-TM-213) is already included in the previously reviewed and approved TORC code. The staff concluded that TORC two-phase friction factor multipliers for subcooled boiling conditions are based on the methods of Sher-Green, modified Martinelli-Nelson, and Pyle, so that the Pyle correlation multipliers were already within the CE-approved licensing methodology and, therefore, are acceptable for use in the APR1400 thermal-hydraulic design.

The DCD Subsection 4.4.2.6.2, "Reactor Vessel and Core Pressure Drops," refers to pressure losses also calculated using classical fluid mechanistic relationships and information from the System 80 reactor flow tests. The staff compared the reactor designs in Table 4.3 of NUREG-1462 and found that reactor internals and the flow geometry are essentially the same for both the System 80 and the APR1400 designs, therefore, the Reynolds number representing the reactor flow condition is almost the same for the two designs, with slight differences in coolant temperature and flow rate. Therefore, the staff concludes that the application of the System 80 reactor flow tests to the APR1400 design is a reasonable use of the prior flow test data and that the methods used are appropriate and acceptable.

Reactor vessel internal component hydraulic loads are provided in DCD Tier 2 Table 4.4-5, "Steady-State Hydraulic Loads on Vessel Internals and Fuel Assemblies." Subsection 4.4.2.6.3 of the DCD, "Hydraulic Loads on Internal Components," refers to load determination from analytical methods based on results of reactor flow model and component test programs. To evaluate the applicant's approach, the staff issued RAI 326-8408, Question 04.04-6 (ML15337A168), requesting that the applicant provide further discussion on: (1) the code or codes used to develop vessel internal component design hydraulic loads, (2) the nodalization model, (3) assumptions and boundary conditions (including the treatment of crud buildup), and (4) the results of any sensitivity analyses. Additionally, staff requested that if the method combined loads resulting from seismic events and LOCA, the applicant should provide a discussion of the treatment of uncertainties in the analyses. In its response to RAI 326-8408, Question 04.04-6 (ML16019A483), the applicant stated that no code was used for this analysis. The applicant indicated that the design hydraulic loads consist of static drag, impingement loads and dynamic loads resulting from pump-induced pressure pulsations, vortex shedding, and turbulence; and that the arrangement, design, and operating conditions of the APR1400 are substantially the same as those for the System 80 design. Therefore, the APR1400 design data for the hydraulic loads were taken from the flow model and tests in KEPCO E&C/ND/TR/12-020 for the System 80 design. The hydraulic loads are calculated from the System 80 data, including the System 80 flow test, with an additional pressure loss of 6 pounds per square inch (psi) (0.041 mega Pascal [MPa]) applied to account for the effect of crud buildup. As stated earlier, the staff reviewed the data and concluded that the RVIs and the flow geometry are essentially the same for the System 80 design and the APR1400. Therefore, previous test measurements on System 80 for static loads, pump-induced pressure pulsation loads, vortex shedding loads, and random turbulence loads may be applied to the calculation of similar flow loads for the APR1400. The staff concluded that the application of the System 80 flow data and model tests are a reasonable use of the prior hydraulic load data and that the methods used are, therefore, appropriate and acceptable.

Reactor Core Instrumentation Requirements

Methods of controlling the power distribution within the core are an important part of confirming that the thermal hydraulic limits are being maintained. The staff reviewed the applicant's technical reports related to the APR1400 CPCS and the PPS high LPD and low DNBR trip setpoint methodology, which are all based on the CE System 80 reactor design. The methodology is described in Technical Reports APR1400-F-C-NR-14003-P, Revision 1, and APR1400-Z-J-NR-14004-P, Revision 1.

The COLSS and the CPC (described in DCD Tier 2 Section 7.2) continually monitor the core power distribution and the DNBR by processing reactor coolant data, neutron flux signals from ex-core detectors, and the input from redundant reed switch assemblies that indicate CEA position. The applicant's treatment of various uncertainties in the methods and analyses is discussed in Technical Report APR1400-Z-J-NR-14004-P, Revision 1. To supplement the description of the CPCS provided in DCD Tier 2, Sections 4.3, 4.4, and 7.2, the applicant submitted two Technical Reports: APR1400-F-C-NR-14003-P, Revision 1 and APR1400-F-C-NR-14001-P, Revision 1. The latter report was not referenced in DCD Tier 2, Sections 4.3, 4.4, or 7.2, and it was not listed in DCD Tier 2 Table 1.6-2, "List of Technical Reports," to be incorporated by reference in the DCD. Because this report contains the description of methods applied in CPC DNBR and LPD calculations for every cycle, the staff considered this report necessary to reach a safety finding on the design. The staff issued RAI 328-8422, Question 04.04-7 (ML15341A120), requesting the applicant to provide a clear explanation of the design basis, and to reference documents appropriately for the design basis of the APR1400 CPCS. In its response to RAI 328-8422, Question 04.04-7 (ML16232A567), the applicant indicated that the report APR1400-F-C-NR-14001-P would be referenced in DCD Tier 2 Section 7.2, and provided a draft markup of the DCD changes. The staff considers the proposed changes provided in the response sufficient to clarify the CPCS design basis, and based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above. Therefore RAI 328-8422, Question 04.04-7, is resolved and closed.

In its response to RAI 328-8422, Question 04.04-7 (ML16007A213), the applicant also committed to provide an additional COL information item in DCD Table 1.8-2 requiring the preparation of the specific procedures necessary to develop the CPCS constants. The staff considers these procedures necessary to make sure CPC constants are correctly and consistently implemented into the CPCS per requirements of IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and RG 1.105.

The staff issued RAI 328-8422, Question 04.04-8 (ML15341A120), requesting that the applicant provide a clearer basis for the CPCS constants in the Technical Report APR1400-F-C-NR-14003-P. The staff noted that the CE Topical Report, CENPD-170-P, "Assessment of the Accuracy of PWR Safety System Actuation as Performed by the Core Protection Calculators," July 1975, provided a much more detailed description of the CPCS constants. In its response to RAI 328-8422, Question 04.04-8 (ML16007A213), the applicant provided its supplemental response (ML16242A429), after the audit, indicating that CENPD-170 was published in 1975 prior to implementation of the CPC Improvement Program. The CPC Improvement Program included modifications and methodology improvements to reduce future reload efforts and to reduce unnecessary plant trips. These CPC methodology changes were described in Topical Report CEN-310-P-A, "CPC and Methodology Changes for the CPC Improvement Program," and approved by the NRC in the SER, "'Safety Evaluation of CE Topical Reports CEN-308-P, CPC\CEAC Software Modifications for the CPC Improvement Program,' and CEN-310-P, 'CPC and Methodology Changes for the CPC Improvement Program,'" and were adopted by several System 80 plants. In its response, the applicant indicated that consequently several parts (sections) of CENPD-170 are not applicable to the APR1400 CPCS. The sections in CENPD-170 that are applicable to the APR1400 CPC are power distribution synthesis, thermal margin technique, and CPC uncertainty analysis. Power distribution synthesis is discussed in DCD Subsection 4.3.2.2, "Power Distribution." Thermal margin technique is discussed in DCD Subsection 4.4.4.5.2, "Thermal Margin Analysis," and CPC uncertainty analysis is documented in Technical Report APR1400-F-C-NR-14001-P. Also, in its response to RAI 435-8541, Question 07.02-16 (ML16145A527), the applicant provided a draft markup of DCD changes that reference CEN-310-P-A. The previously approved Topical

Report CEN-310-P-A provides the post CPC Improvement Program details on the intended function and operation of the CPC. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore RAI 435-8541, Question 07.02-16, is resolved and closed.

In January 2016, the staff conducted an audit (ML16060A413) to review additional documentation and calculations related to the CPCS and the development and verification of the analytical methods used. The audit also covered other subjects including core design and the COLSS. The staff asked questions about the CPCS regarding the sensitivity of the ex-core detectors to fuel burnup, CEA position, and cycle-to-cycle variation in loading pattern. Additionally, the procedures for development and implementation of the shape annealing functions, rod shadowing factors, and addressable constants into the CPC were not fully described. Accordingly, the staff notified the applicant that there were several remaining information items that the staff would develop into RAIs. The applicant opted to address the issues by attaching responses to the outstanding audit concerns to a supplement for an updated response for the existing RAI 328-8422, Question 04.04-8, in a letter dated August 3, 2016 (ML16216A631), so the staff did not issue any follow-up RAIs. The staff's review of the remaining items is summarized in the following subparts.

1. CPCS Items

In CPCS Item 2, the staff requested clarifications on the analytical methods and computer codes from CENPD-170-P that were used for the APR1400, and whether the selected codes had been reviewed and approved by the NRC. In its supplemental response attached to the response to RAI 328-8422, Question 04.04-8 (ML16216A631), the applicant provided its response to this item stating that the codes described in Chapter 3 of CENPD-170 are not used for the APR1400, and that the CPC and methodology changes for the CPC Improvement Program are those implemented at several U.S. nuclear power plants, including San Onofre Nuclear Generating Station and PVNGS. The applicant also stated that the specific CPCS design for APR1400 comes from that of PVNGS, and thus its functional design is based on those described in the staff's SER on CEN-308-P and CEN-310-P. Based on previously approved SER of CEN-308-P and CEN-310-P and the applicant’s implementation of the setpoint methodology in Technical Report APR1400-F-C-NR-14001-P, Revision 1, the staff concluded that CPC and methodology changes for the CPC Improvement Program are acceptable pending the incorporation of CEN-308-P in Technical Report APR1400-F-C-NR-14001-P by reference and the incorporation of CEN-310-P in the DCD by reference. The staff has confirmed incorporation of the changes discussed above in Revision 3 to DCD Tier 2 and in Revision 3 to Technical Report APR1400-F-C-NR-14001-P and, therefore, RAI 321-8422, Question 04.04-8 is resolved and closed.

In CPCS Item 3, the staff noted in APR1400-F-C-NR-14001-P that CPC calculations are verified using a large number of power distributions at BOC, middle-of-cycle, and EOC. However, the report does not describe how the CPC constants used to determine core power distribution limits (e.g., planar radial peaking factor, F_{xy}, and three dimensional power peaking factor, F_q) are calculated so that the peak LHR and DNBR are within operating limits. This is especially important when new fuels are introduced (i.e., mixed cores). Also, the staff requested an explanation of how multiple DNBR uncertainties would be implemented in the CPC algorithms for mixed cores. In its response to this item (ML16216A638), the applicant stated that the CPCS constant F_{xy} is calculated for the fuel cycle using the ROCS code. The maximum cycle F_{xy} calculated by the ROCS code is tentatively set as the F_{xy} value for the entire cycle. Then during startup testing, F_{xy} is measured and compared to the calculated F_{xy} for a time range early in the fuel cycle. If the ratio of the measured F_{xy} at startup to the calculated BOC F_{xy} is greater than 1.0, then this ratio is applied as a penalty factor to the tentative maximum cycle F_{xy} calculated by ROCS and used for the entire cycle. If the ratio is less than 1.0, then the maximum cycle F_{xy} calculated by ROCS is used for the entire cycle as calculated. Using a penalty factor based on measured data, or using a maximum calculated value in lieu of a lesser measured value makes the F_{xy} conservative. The F_q is based on the maximum cycle F_{xy} installed in the CPCS and the measured ex-core detector signals. The staff concluded that the use of a cycle-specific penalty factor applied to the calculated maximum F_{xy}, based on the startup physics testing that confirms use of the maximum F_{xy} value for the entire cycle, is conservative. The F_q value is determined based on actual ex-core detector output and the maximum F_{xy} value input into the CPCS. Therefore, the staff concluded that the F_q calculation is also conservative.

In a mixed core scenario, the applicant indicated that DNBR uncertainty would be calculated independently for each fuel type, and that the calculation yielding the largest uncertainty would be used by the CPCS. An uncertainty difference would also be added to applicable DNBR penalty factors for added conservatism. For a mixed core, which would not include Cycle 1, the maximum DNBR uncertainty is applied for both fuel types in the core and an additional penalty factor would be applied to the DNBR uncertainty. The staff concluded that the implementation of multiple DNBR uncertainties would be conservative in a mixed core.

Although the methodology for mixed cores is appropriate, the staff concluded that the documentation in the DCD was not adequate. Therefore, in a NRC-KHNP public teleconference held on July 27, 2016, the staff requested that the applicant add a mixed core discussion in the CPCS Technical Report APR1400-F-C-NR-14001-P and in DCD Tier 2 Section 4.4. The staff reviewed the applicant's response to RAI 328-8422, Question 04.04-8, dated August 29, 2016 (ML16242A430), and found it to be acceptable since it clarified the CPC calculation of the DNBR penalty factor for mixed cores and staff determined that the uncertainty methodology was conservative in regard to preserving DNB margin. Based on the review of DCD Tier 2 and Technical Report APR1400-F-C-NR-14001-P Revision 2, the staff has confirmed incorporation of the changes described above; therefore RAI 328-8422, Question 04.04-8, is resolved and closed.

The staff notes that a license amendment request would be required to update the Plant TS if a new fuel type was introduced into the APR1400 core. Although cycle-specific core operating parameters have generally been relocated to the COLR, an update to the analytical methods used to determine the core operating limits with the new fuel would be necessary in TS Section 5. This would include neutronics calculation code results and other calculations supporting the fuel transition, including a vendor mechanical fuel design report and an updated LOCA break spectrum analysis.

In CPCS Item 4, the staff inquired whether the design-basis AOO events are included in the database of power shapes used by the core simulator 3D power distributions to determine the CPC power distribution uncertainties. In its response to this item (ML16216A638), the applicant stated that approximately 4,800 power shapes are generated for a given fuel cycle to evaluate a combination of changes related to burnup, power level, rod (CEA) position, and xenon (based on a variety of transients). Although specific AOOs are not modeled, extreme shapes within the analysis range of the CPC hot pin ASI are generated among the 4,800 power shapes. The applicant expected that these extreme shapes will bound any hot pin ASI shapes that could occur during an AOO event. Numerous transients will be evaluated among the 4,800 power shapes that are generated, leading to numerous variable xenon conditions, and since these transient variations are coupled with variable power levels and rod shadowing conditions, the staff concludes that the power shapes under variable conditions bound the possible hot pin conditions that could be generated during AOOs.

In CPCS Item 5, staff asked the applicant to describe the reload process for CPCS setpoint calculations, define what calculations are performed generically for each fuel type and which ones are cycle-specific, and identify the codes used and provide a reference to the approval SER for each. In its response to this item (ML16216A638), the applicant provided a list of reload setpoint calculations and processes to determinate the CPC constants. However, these calculations are explained and addressed via the RAI 435-8541, Question 07.02-16. The staff concluded that methods of implementing the CPCS constants are similar to and consistent with those of the previously approved methods for the CE plants and are, therefore, acceptable. However, the calculation processes should be documented for each COL license application.

In CPCS Item 6, the staff noted that CENPD-170 discusses conversion of ex-core detector responses into up to eight algorithm constants (α_1 through α_8) for the core average axial power distribution. These are pre-calculated to represent flat-, saddle-, top-, or bottom-peaked axial shapes for beginning-of-life (BOL) and end-of-life (EOL) conditions. The CPC uses some degree of pattern recognition on the 3-ring axial power distribution to determine which of the four power shape types are present and then uses a cubic spline fit to data. The staff requested an explanation of how the constants are developed, whether they are cycle dependent or burnup-dependent within the cycle, and how they will be verified against plant data. In its response to this item (ML16216A638), the applicant stated that only parts of CENPD-170-P are accurate relative to the APR1400 design. The applicant indicated that the CPC uses the measured boundary point power correlation coefficients (BPPCC) constants, which are determined during the startup tests at the site. The BPPCC constants are combined using a least square fitting of startup test data, and are used for the entire cycle. The staff concluded that the algorithm constants methods are identical to processes that have previously been approved for operating Combusting Engineering plants documented in CEN-356(V)-P-A Revision 01-P-A, and that the processes included in the applicant response provide adequate margin.

In CPCS Item 7, staff noted that no detail was provided on the method used to define radial power distribution, planar radial peaking factors, or axial augmentation factors used to define the hot pin power as a function of the CEA configuration. The applicant was requested to explain how the planar radial peaking factors will be calculated, whether cycle- or burnup dependent, what codes are used in the calculations, and whether the codes have been approved. In its response to this item (ML16216A638), the applicant stated that F_{xy} is calculated for each fuel cycle using the ROCS code. The applicant indicated that the ROCS code approval is documented in CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design." The maximum cycle F_{xy} is then calculated as discussed previously under the response for CPCS Item 3.

In CPCS Item 8, staff asked the applicant to explain how the axial augmentation factors are calculated, whether cycle- or burnup-dependent, what codes are used, and whether the codes have been approved. In its response to this item (ML16216A638), the applicant stated that for the current CPC algorithms, the axial augmentation factors, as described in CENPD-170, were deleted because the 3-D power peaking factor increase due to fuel densification was demonstrated to be insignificant in the staff's SER for CEN-310-P-A. The staff concluded that the deletion of axial augmentation factors was satisfactory based on the demonstration provided in CEN-310-P-A.

In CPCS Item 9, staff asked the applicant to explain whether the CPC constants will be adjusted during a cycle to reduce conservatism, and if so, how the changes are verified. In its response to this item (ML16216A638), the applicant stated that the CPC constants related to power distribution are planar radial peaking factor (F_{xy}), rod shadow factor (RSF), shape annealing matrix (SAM), and BPPCC; and that the constants are installed using the measured values during reactor startup tests. For the reactor startup, these constants are pre-calculated and loaded into the CPC prior to the tests, and the pre-calculated value include additional penalties to ensure that the CPC remains conservative prior to the installation of the measured values. The additional penalties in the CPC constants are deleted after startup tests by installation of the measured F_{xy}, RSF, SAM, and BPPCC values into the CPC. The CPC constants related to power distribution are adjusted to reduce conservatism only after startup tests, and are not changed during a cycle. The staff concludes that the applicant's approach for setting the CPC constants related to power distribution is appropriate given the conservative penalties applied, and consistency with approved methods for the CE System 80 plants currently in operation.

In CPCS Item 10, the staff asked the applicant to explain the safety analysis time delay that the CPC will utilize before reverting to the "pre-determined penalty factor" (PF), as discussed in its response dated December 29, 2015 to RAI 274-8277, Question 07.01-37 (ML15363A340), and explain how the PF will be a large value that will ensure a CPC initiated DNBR reactor trip and/or a LPD reactor trip. In its response to this item (ML16216A638), the applicant stated that there is no design-basis event (DBE) discussed in DCD Chapter 15 that assumes a CEA misoperation concurrent with both CEA calculators (CEACs) failed. Therefore, there is no CPC penalty factor time delay considered in the DCD Chapter 15 analyses. The applicant indicated that a PF of 8 is applied to ensure that a reactor trip occurs after a 30 second time delay if both CEACs are determined to be inoperable. Based on a minimum power level from the POWER module of approximately 20 percent, the POWER module output would be 160 percent power. Even considering worst case system uncertainties, this PF is adequate to ensure a high power reactor trip is generated.

The staff's review of the PF confirmed that the range of the CEA PFs is 0 to 8.0 per Table A2.1.2-1 of CENPD-396-P, "Common Qualified Platform Core Protection Calculator System," indicating that the applicant had selected the most conservative input for CPC design. The staff therefore concludes that the PFs for both DNBR and LPD (set at 8) provide a conservative safety margin to ensure that a reactor trip will occur if both CEACs are inoperable.

2. COLSS ITEMS

In order to better understand the APR1400 COLSS design, the staff noted that Topical Reports CENPD-169 and CEN-312 were used to describe the COLSS in the System 80+ SER. In RAI 328-8422, Question 04.04-8, COLSS Item 1, the staff inquired whether these documents also supported the APR1400 design, and if so, the applicant should make these documents available for review. In its response to this item (ML16216A638), the applicant stated that CENPD-169-P, "COLSS-Assessment of the Accuracy of PWR Operating limits as Determined by the Core Operating Limit Supervisory System," August 1975, has been changed and improved by CE such that several parts are not appropriate to refer for the APR1400 design. The COLSS design for APR1400 comes from those used at PVNGS and thus its functional design is based on those described in CEN-312-P, "Overview Description of the Core Operating Limit Supervisory System (COLSS)." The applicant indicated that CEN-312-P and the COLSS functional design requirements described in APR1400-F-C-NR-14002-P, Revision 1, contain the design bases information for the COLSS that is sufficient to reference.

The staff reviewed the previous NRC evaluations of CEN-312-P and concluded that it, along with the applicant's Technical Report APR1400-F-C-NR-14002-P, provide an adequate basis for the COLSS design; however, they both need to be appropriately referenced in the DCD Tier 2 Table 1.6-2. The applicant has committed to add CEN-312-P per its response to RAI 435-8541, Question 7.2-16 (ML16145A530), but no commitment was made for APR1400-F-C-NR-14002-P. The staff determined that APR1400-F-C-NR-14002-P is needed as reference due to system interfaces with the CPCS, to confirm the regulatory basis for operation of the CPCS. The staff reviewed the applicant's response to RAI 328-8422, Question 04.04-7 (ML162324A567), and found it to be acceptable since it added the needed reference to Table 1.6-2 of the DCD. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore for RAI 328-8422, Question 04.04-7, is resolved and closed.

3. CORE DESIGN ITEMS

In Core Design Item 3, the staff observed that the applicant's axial core power distributions at various core depletion states assumed only unrodded operation. Since the APR1400 is designed to use regulating rods to control power shapes, staff requested the applicant to clarify whether presented unrodded axial power shapes are meant to bound those cases where regulating rods are partially inserted in the core. In its response to this item (ML16216A638), the applicant stated that the axial core power shapes presented are typical axial power distributions expected during normal unrodded base-load operation of the APR1400 initial core. Operating margin, monitored by the COLSS, may direct the use of regulating rods to control axial power distribution.

The staff concluded that the rodded axial core power distributions were considered and the methodology is validated during startup physics testing, which is conservative. The F_q is based on the maximum cycle F_{xy} installed in the CPCS and the measured ex-core detector signals.

In Core Design Item 4, staff asked the applicant to justify not performing (1) demonstrated CEA positions and misalignment tests and (2) demonstration of the capability of the in-core neutron flux instrumentation to detect rod misalignment equal to or less than the TS limits for control rod misalignment. In its response to this item (ML16216A638), the applicant stated that the in-core neutron flux instrumentation is not safety-grade and therefore cannot be used to detect CEA misalignment. The applicant noted that per DCD Subsection 7.2.1.1, the position of each CEA is measured by two independent reed switch assemblies that provide safety grade outputs. Because incore neutron flux instrumentation is not used to detect CEA misalignment, and independent reed switch assemblies are used for this purpose instead, the staff concludes that there is no need to test in-core instrumentation for CEA position and misalignment.

4.4.5 Combined License Information Items

There are no COL information items associated with Section 4.4 of the APR1400 DCD. However, in response to RAI 435-8541, Question 07.02-16, the applicant committed to add COL Information Item 7.2(1) to DCD Tier 2 Table 1.8-2 for a COL applicant to provide procedures to develop CPCS setpoint constants. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 435-8541, Question 07.02-16, item 3, is resolved and closed.

4.4.6 Conclusions

The scope of the review included the design bases specification, the steady-state analysis of the core thermal-hydraulic performance, analytical methods, testing and verification, and instrumentation. The staff found that the applicant provided sufficient information relating to core thermal hydraulic requirements to ensure that the reactor SAFDLs are not violated, in accordance with GDCs 10 and 12. The staff reviewed the KCE-1 CHF correlation and found that the formulation and methodologies used contain adequate margins and adequately account for uncertainties to predict DNB within at least a 95-percent probability at a 95-percent confidence level and that the reactor core and associated control and protection systems have been designed with appropriate margin to ensure that DNB limits are not exceeded.

The staff concludes that the APR1400 thermal hydraulic design parameters are very comparable to those of the System 80+. The APR1400 thermal hydraulic design methodology depends heavily upon codes and methods previously approved by staff for domestic CE plant applications. The thermal hydraulic requirements which were in place at the time of the System 80+ DC, and which were in place at the time the staff approved the CE codes and methodologies, remain applicable for this review. The APR1400 thermal hydraulic design also depends heavily upon design calculations and startup test data from similar domestic U.S. and foreign reactors. Therefore, the staff determined that the design provides adequate assurance that the reactor will perform its related safety functions under all modes of operation. The staff further concludes that the codes and methods previously approved for CE plant applications remain valid for the APR1400. The staff also concludes that the application meets GDC 1, GDC 2, GDC 4, GDC 12, 10 CFR 50.34(f)(2)(xviii) and 10 CFR 52.47(b)(1).

4.5 <u>Reactor Materials</u>

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 Coolant System.”

DCD Tier 2: The applicant has provided a Tier 2 design description in DCD Tier 2 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, materials other than austenitic stainless steels, and cleanliness control.

The materials used to fabricate the CRDM pressure housing are austenitic stainless steel, martensitic stainless steel, and nickel-based alloy (Alloy 690, thermally treated). CRDM pressure housing materials meet ASME Code Section III requirements for Class 1 components. The applicant identified the use of Code Case N-4-13 given in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 36.

The CRDM non-pressure boundary components that are in contact with reactor coolant include solution annealed austenitic stainless steel that conform to RG 1.44, "Control of the Processing and Use of Sensitized Stainless Steel," quenched and tempered martensitic stainless steel, solution annealed and thermally treated nickel-based alloy and cobalt-based alloys. Some latch assembly components are fabricated from, or clad with, cobalt-based alloy, chrome plated, or coated with chrome oxide to resist wear. The proposed materials have been successfully used in operating plants in the U.S. and Korea. The manufacturing and process controls for preventing intergranular corrosion of stainless steel components follow the guidance in RG 1.44.

Cleaning and cleanliness control comply with the description provided in DCD Subsection 5.2.3.4.2, "Cleaning and Contamination Protection." Cleaning and cleanliness control during shipment and storage will be carried out in accordance with the requirements of ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," and follow the guidance of RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," Revision 4, dated June 2010.

ITAAC: The ITAAC associated with DCD Tier 2 Section 4.5.1 are discussed in DCD Tier 2, Subsection 14.3.2.4, "ITAAC for Reactor Systems," and delineated in DCD Tier 1 Section 2.4.1 and in Items 2.a, 3.a, and 4.a in Table 2.4.1-4, "Reactor Coolant 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 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. GDC 1, "Quality standards and records," and 10 CFR 50.55a, "Codes and Standards," as they 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 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," as it 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, “Reactivity control system redundancy and capability,” as it requires, in part, that one reactivity control system use control rods and that this system be capable of reliably controlling reactivity changes.

4. Title 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are met, a facility that incorporates the DC is built and operated in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC’s regulations.

Regulatory guidance used to meet the above requirements includes:

1. RG 1.28, “Quality Assurance Program Criteria (Design and Construction).”

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

3. RG 1.44, “Control of the Processing and Use of Sensitized Stainless Steel,” and RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

4. ASME NQA-1, 2008 Edition with 2009 Addenda.

4.5.1.4 Technical Evaluation

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

Materials Specifications

The staff reviewed DCD Tier 2 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 materials used for the pressure housing components include stabilized stainless steel (Grades 347 and 348), martensitic stainless steel (ASME Code Case N-4-13 modified Type 403), nickel-based alloy SB-166 (alloy 690, thermally treated) and austenitic stainless steel (SA-213, Type 316 and SA-479, Type 316). Welding filler materials Alloy 52/52M, Alloy 152, and Types 308 and 316 austenitic stainless steel will be used in the fabrication of the CRDM pressure housing. The staff reviewed the specifications and grades of the CRDM pressure housing materials and verified that the materials listed meet the requirements of ASME Code, Section III, Paragraph NB-2121, “Permitted Materials Specifications,” which requires the use of materials listed in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B. The staff also verified that modified Type 403 stainless steel per the ASME Code Case N-14-3 is approved for use in RG 1.84, Revision 36, and is therefore acceptable for use.

The staff verified that the materials identified for each of the specific pressure boundary components are acceptable materials for use in ASME Code, Section III, Class 1 systems, as noted below.

The staff issued RAI 303-8391, Question 04.05.01-1 (ML15314A597), requesting the applicant to provide the following information to determine whether the APR1400 design meets these criteria in GDC 14 and GDC 26 with regard to pressure-retaining and internal components of its control rod drive (CRD) system:

> Determine if there are components of the APR1400 design that would be considered RCPB components, or components in contact with reactor coolant in APR1400 DCD Sections 4.5.1.1(a) and 4.5.1.2(b) which have been omitted, but which should be included in these sections along with their ASME Code material specifications. The staff notes, for example, that APR1400 DCD Figure 3.9-7 identifies the housing nut and vent stem, which are reactor coolant pressure retaining components, but which have been omitted from DCD Section 4.5.1.1(a).
>
> Revise DCD Sections 4.5.1.1(a) and 4.5.1.1(b) to include any omitted components and their ASME Code material specifications.

In its response to RAI 303-8391, Question 04.05.01-1 (ML15356A554), the applicant provided the revised DCD Subsection 4.5.1.1(a), "Material Specifications," to include all pressure retaining components and the associated material specifications. The materials used include ASME SA-182, Grades F347/348; SB-166, Alloy 690; SA-479, Type 316; and Modified Type 403 (ASME Code Case N-4-13) which have previous operating experience. The staff found the revised section in Revision 1 of the DCD acceptable, because it includes all pressure retaining components and the materials used have previous operating experience.

APR1400 DCD Subsections 4.5.1.1(a) and 4.5.1.2(b), "Austenitic Stainless Steel Components," specify the use of Type 304 and 316 stainless steel for the CRD system. Type 304 and 316 stainless steel, which have higher carbon content than Type 304L and 316L, and their associated Type 308 and 316 weld filler metals which are designated for use in the APR1400 CRD system are less resistant to sensitization due to heat treatment or welding than low carbon Types 304L and 316L stainless steel. NRC Information Notice 2006-27, "Circumferential Cracking in Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors," and operating experience with leaking CRD canopy seal welds document the potential of stress corrosion cracking (SCC) in Type 304 and 316 SS. These instances of SCCs are occurring in stagnant or dead end PWR coolant environments prone to increased levels of oxygen.

Since Type 304 and 316 stainless steel materials are more susceptible to SCC than low carbon Type 304L, 304LN, 316L, and 316LN SS, the use of Type 304 and 316 materials may affect the integrity of the CRD system components, including portions of the RCPB. Specifically, the use of these materials can affect the structural integrity of CRD components that are subjected to stagnant water, dead legs, or areas prone to increased levels of oxygen.

The staff issued RAI 303-8391, Question 04.05.01-2 (ML15314A597**)**, requesting that the applicant revise Subsection 4.5.1.1 of APR1400 DCD to specify the use of Types 304L and 316L stainless steel and their applicable filler metals (Types 308L and 316L) or provide further justification (e.g., discussing how design features of the APR1400 ensure these areas are not prone to increased levels of oxygen to minimize the likelihood of SCC of in Type 304 and/or 316 grades) addressing the acceptability of the use of normal carbon Type 304 and 316 materials and their applicable filler metals (Types 308L and 316L).

In its response to RAI 303-8391, Question 04.05.01-2 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.2, to include the venting design to minimize increased levels of oxygenated water in stagnant or dead end PWR coolant areas of the CRD components. The magnetic drive CRDM design of the APR1400 incorporates the Versa Vent™ to eliminate the air trapped in the top of the CRDMs. The Versa Vent™ has been used for the OPR1000 and current APR1400 plants. In addition, the revised DCD Subsection 4.5.1.2 specifies that the Versa Vent™ is substituted for the CRD housing nut, which is a pressure boundary. However, the Versa Vent™ is not considered a pressure retaining boundary, and therefore is made of non-code and non-nuclear safety-related material. Since the Versa Vent™ is substituted for the CRDM housing nut (which is a pressure boundary), the Versa Vent™ should also be a pressure boundary. Therefore, the staff issued RAI 436-8538, Question 04.05.01-11 (ML16110A454), requesting that the applicant revise DCD Subsection 4.5.1.1 to include the material specification and type for the Versa Vent™.

In its response to RAI 438-8538, Question 04.05.01-11 (ML16110A454), the applicant stated the following:

> The Versa Vent™ is substituted for the CEDM housing nut and used for venting as long as the vent stem does not leak. In this case the pressure boundary consists of the upper pressure housing and vent stem, and the Versa Vent™ is non pressure boundary. When the vent stem leaks excessively, the Versa Vent™ is removed and CEDM housing nut is installed. In this leaking case the pressure boundary is changed to consist of the upper pressure housing and CEDM housing nut and omega seal weld. Venting is not possible when the CEDM housing nut and omega seal weld is applied. This is why the Versa Vent™ is not pressure boundary.

The staff does not consider that the Versa Vent™ is not a pressure boundary component, because the Versa Vent™ replaces and provides the same function (with the addition of venting capability) as the CRD housing nut. Only the parts of the Versa Vent™ that provide pressure boundary function needs to be classified as RCPBs. Therefore, the staff issued a follow-up RAI 523-8684, Question 04.05.01-15 (ML16275A003), requesting that the applicant revise DCD Subsection 4.5.1.1 to include the material specifications and types for the Versa Vent™.

In addition, the staff issued follow-up RAI 523-8684, Question 04.05.01-16, (ML16275A003), requesting that the applicant provide data/operating experience that justifies venting during refueling outages to keep oxygen levels low (as stated in RG 1.44) so that non-L grade stainless steels can be used as previously discussed in RAI 303-8391, Question 04.05.01-2. RAI 523-8684, Questions 04.05.01-15 and 04.05.01-16 were tracked as open items.

In its response to RAI 523-8684, Question 04.05.01-15 (ML16309A071), the applicant stated that only the vent stem and the ball are pressure boundary components. The Versa Vent™ is not credited for providing a pressure boundary function, and is not needed if the vent stem and ball are installed. If the vent stem and ball cannot provide the pressure boundary function (i.e., leaks), the housing nut is installed and welded onto the CRD vent stem, and in this scenario the housing nut provides the pressure boundary function. Therefore, the staff finds that the ball and vent stem, and the housing nut if installed and welded, are pressure boundary components, and the Versa Vent™ is not credited as a pressure boundary component. The staff finds the materials for the pressure boundary components acceptable, since the materials are specified in Section II of the ASME Code, and have satisfactory operating experience.

In its response to RAI 523-8684, Question 04.05.01-16 (ML16335A463), the applicant provided operating experience of the Versa Vent™ which has approximately 20 years of satisfactory operation, with no history or cracking in the Versa Vent™ or the CRD components. The Versa Vent™ has been used in operating reactors to reduce dissolved oxygen levels. In addition, the CRD upper housing and vent stem in the APR1400 design are at low temperatures, 57 degrees C (135 degrees F), which also minimize the occurrence of SCC. The pressure boundary components, as stated previously, are solution annealed, and welding procedures are qualified in accordance with RG 1.44 to minimize the occurrence of sensitization. The filler metal used for welding is Type 316L, which is more resistant to stress corrosion cracking. Therefore, the staff finds that the Type 304 and 316 stainless steel materials and Type 316L filler metal material are acceptable, based on operating experience of low oxygen levels provided by the Versa Vent™, low operating temperatures, and by using the guidance of RG 1.44 to control sensitization of the stainless steel materials to minimize SCC. Therefore, these open items, associated with RAI 523-8684, Questions 04.05.01-15 and 04.05.01-16, are resolved and closed.

APR1400 DCD Subsection 4.5.1.1(a)(1) specifies austenitic stainless steel, martensitic stainless steel, and Alloy 690 as materials of construction for the CRD motor housing assembly which is a component of the RCPB. However, it was not clear if the motor housing is one part, and the specific application of each material was not given. The staff issued RAI 303-8391, Question 04.05.01-3 (ML15314A597), requesting that the applicant revise DCD Subsection 4.5.1.1(a)(1) to clearly identify the different parts of the CRD motor housing assembly and specify the applicable ASME Code material specification for each part, emphasizing the staff's concern raised in RAI 303-8391, Question 04.05.01-2, above.

In its response to RAI 303-8391, Question 04.05.01-3 (ML15314A597), the applicant revised DCD Tier 2 Subsection 4.5.1.1(a)(1) to identify the different parts of the CRD motor housing assembly and the applicable ASME Code material specification for each part. The staff finds this information, which was included in Revision 1 of the DCD, to be acceptable because the various parts of the CRD motor housing assembly are identified with the applicable material specifications and types. In addition, 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.

APR1400 DCD Tier 2 Subsections 4.5.1.1(b)(9), 4.5.1.1(b)(13), and 4.5.1.1(c)(4) specify Type 300 series stainless steel as the material of construction for the CRD system dowel pin, locking cap and screws, and extension shaft pins. The specification of "Type 300 Series SS" was too generic for the staff to make a clear determination regarding the acceptability of the material to be selected for each of these applications. The staff issued RAI 303-8391, Question 04.05.01-4 (ML15314A597), requesting the applicant revise DCD Subsections 4.5.1.1(b)(9), 4.5.1.1(b)(13) and 4.5.1.1(c)(4) to give a more definitive ASME Code material specification for each part, emphasizing the staff's concern raised in RAI 303-8391, Question 04.05.01-2, above.

In its response to RAI 303-8391, Question 04.05.01-4 (ML15356A554), the applicant revised DCD Tier 2 Subsections 4.5.1.1(b)(9), 4.5.1.1(b)(13), and 4.5.1.1(c)(4) to include the material specifications and types, which included ASTM A276, Type 304; ASTM A479, Type 304; and ASTM A193, Grade B8. The functions for the CRD system dowel pin, locking cap and screws, and extension shaft pins are to provide anti-rotation of alignment taps, guiding moving parts, and positive locking for threaded parts and are exposed to reactor coolant. These parts are not considered RCPB and do not experience loading conditions conducive to SCC, and are common material used in operating plants. The staff finds these material specifications specified in Revision 1 of the DCD are identical to the ASME Code, Section II materials and are acceptable for use in non-pressure boundary applications exposed to reactor coolant.

Compliance with the requirements of GDC 26 as it relates to the CRD 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:

- austenitic stainless steels (ASTM A269, Type 316; ASTM A276, Type 316; ASTM A276, Type 321; ASTM A276, Type 304; ASTM A269, Type 304),
- martensitic stainless steels (ASTM A276,Types 410, 410 condition T and 440C),
- nickel-based alloys (AMS 5698, Alloy X-750 and AMS 5699, Alloy X-750),
- nickel-chromium-molybdenum-columbium alloy (ASTM B446, Alloy 625),
- cobalt-based alloys (Haynes No. 36 and Stellite No. 6B),
- SAE AMS 2460 chrome plating, and
- chrome oxide plasma spray treatment.

APR1400 DCD Tier 2 Subsection 4.5.1.3, “Other Materials,” specifies a cobalt-based alloy (Stellite No. 6B) for the pins and another cobalt-based alloy (Haynes No. 36) for the latch and links. The staff understands that these materials will be used as structural materials for the fabrication of these components (not hardfacing material). The staff issued RAI 303-8391, Question 04.05.01-5 (ML15314A597), requesting that the applicant provide the following additional information to ensure that the material properties of these components (e.g., toughness, etc.) will be appropriate for these applications.

- Revise APR1400 DCD Subsection 4.5.1.3 to specify the applicable material specifications, including heat treatment to be applied, etc., which will ensure appropriate component material properties when utilizing Haynes No. 36 or Stellite No. 6B material.
- Also provide operating experience which supports the use of Haynes No. 36 and Stellite No. 6B material of the given specifications for these applications.

In its response to RAI 303-8391, Question 04.05.01-5 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.3 to specify the pins are a cobalt alloy of AMS 5894 (i.e., Stellite No. 6B as bars), and are ordered in the vacuum-annealed condition, and the latch and links are AMS 5979 (i.e., Haynes alloy No. 25 as bars or 36 as castings). The staff issued RAI 436-8538, Question 04.05.01-12 (ML16110A454), requesting that the applicant provide the appropriate material specifications and include them in Subsection 4.5.1.3 for Haynes alloy Nos. 25 and 36, because there is no material specification AMS 5979. The staff also requested that DCD Tier 2 Subsection 4.5.1.3 include the heat treatment (temperature, time and cooling, if applicable) to be applied to the Stellite No. 6B, Haynes alloy No. 25 and Haynes No. 36 materials, as previously requested.

In its response to RAI 436-8538, Question 04.05.01-12 (ML16110A454), the applicant revised DCD Tier 2 Subsection 4.5.1.3 that included the correct material specification AMS 5759 for Haynes alloy Nos. 25 and 36, and the appropriate heat treatment for the Stellite No. 6B, Haynes 25 and Haynes No. 36. The use of these materials are commonly used in operating plants and have satisfactory operating experience, and therefore the material specifications and heat treatments in Revision 1 of the DCD are acceptable to the staff.

APR1400 DCD Tier 2 Subsection 4.5.1.1(d) specifies the use of Types 308 and 316 stainless steel, and Alloys 52, 52M, and 152 welding filler metal. This statement is ambiguous and unclear for which components and applicable base material each weld filler metal is to be used.

The staff issued RAI 303-8391, Question 04.05.01-6 (ML15314A597), requesting that the applicant delete APR1400 DCD Tier 2 Subsection 4.5.1.1(d) and specify the applicable weld metal, including the filler metal specification and type of weld, for each component in APR1400 DCD Tier 2 Subsections 4.5.1.1(a), (b) and (c), so that the staff can make its determination that the material is compatible with its environment and that each component will perform its design function, emphasizing the staff's concern raised in RAI 303-8391, Question 04.05.01-2, above.

In its response to RAI 303-8391, Question 04.05.01-6 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.1(d) and added Table 4.5-1, "Weld Filler Materials for Reactor Coolant Pressure Boundary of CEDM," which specified the applicable filler metal type for each weld. However, the applicant did not provide the filler metal specification for each type of filler metal. The staff issued RAI 436-8538, Question 04.05.01-13 (ML16110A454), requesting that the applicant provide the filler metal specification, as previously requested, in order to determine the quality of the material and compatibility to the reactor coolant.

In its response to RAI 436-8538, Question 04.05.01-13 (ML16110A454), the applicant revised Table 4.5-1 that provided the applicable filler metal specifications and types, which included ASME Section II, SFA-5.9, Type ER316L and SFA-5.14, Type ERNiCrFe-7A. The staff finds these ASME Code filler metals that are included in Revision 1 of the DCD acceptable because the austenitic stainless steel filler metal has low carbon which increases the resistance to SCC and the nickel-chromium alloy filler metal has higher chromium content which also increases the resistance to SCC.

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.

APR1400 DCD Tier 2 Subsection 4.5.1.3 states that, "The material used in CEDM [CRDM] that are not included in ASME Section III, Appendix I, Division 1 are identified in Sections 4.5.1.1 b and c." It is not clear to the staff what materials this statement refers to since the austenitic stainless steels and nickel-based alloys identified in APR1400 DCD Tier 2 Subsections 4.5.1.1(b) and 4.5.1.1(c) are included in Appendix I of the ASME Code, Division 1. The staff issued RAI 303-8391, Question 04.05.01-7 (ML15314A597), requesting that the applicant explain which specific materials the statement above from DCD Subsection 4.5.1.3 applies to or revise DCD Subsection 4.5.1.3 to delete the statement.

In its response to RAI 303-8391, Question 04.05.01-7 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.3 that deleted the statement, as requested above. The staff finds this material specified in Revision 1 of the DCD acceptable, because the austenitic stainless steels and nickel-based alloys are included in Appendix I of the ASME Code, Section II. Materials other than stainless steel and nickel-based alloys are discussed in "Other Material" Section of this SER below.

APR 1400 DCD Tier 2 Subsection 4.5.1.1 states that the steel ball in the vent valve on top of the CRDMs, bearing inserts, and alignment tab in the motor assembly are made of martensitic stainless steel, which conforms to American Society for Testing and Materials (ASTM) A276, Type 440C. This material has a yield strength greater than 620 MPa (90 ksi). However, its usage is limited to the items above and the steel ball is used as a seal and is not a primary load-bearing member of the pressure boundary. In addition, the inserts and alignment tab, which are Type 440C (used for surface hardness), operate under low stress, and are not part of the safety release mechanism in the motor assembly. This material was tested for the design life of the CRDM component and has favorable operating experience in operating reactors in the U.S. and Korea, and therefore, the staff finds this use of this material for these limited items acceptable.

Austenitic Stainless Steel Components

DCD Tier 2 Subsection 4.5.1.2 states that the use of austenitic stainless steel base materials for CRDM applications is consistent with the recommendations of RG 1.44, and that only procedures that have been demonstrated to not sensitize CRDM stainless steel components are used. 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 SCC. The staff concludes that the use of austenitic stainless steel base materials for CRDM applications is acceptable because furnace-sensitized material is not used and methods described in RG 1.44 will be 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, and to reduce susceptibility of components to SCC.

In addition, DCD Tier 2 Subsection 4.5.1.2 states, "[f]abrication and processing of austenitic stainless steel are applicable to the CRDM stainless steel as addressed in Subsection 5.2.3.4." The staff issued RAI 303-8391, Question 04.05.01-8 (ML15314A597), that requesting that the applicant confirm this means "the fabrication and processing requirements in Subsection 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steel," are applicable to the austenitic stainless steel materials for the CRDMs," and revise the DCD accordingly.

In its response to RAI 303-8391, Question 04.05.01-8 (ML15356A554), the applicant revised DCD Subsection 4.5.1.2, which states that the fabrication and processing requirements in Subsection 5.2.3.4 of the APR1400 DCD are applicable to the CRDMs. The applicant also proposed to include that all unstabilized austenitic stainless steels in the CRDMs are supplied in the solution annealed condition, which ensures a homogenous and unsensitized material. Therefore, the staff finds the fabrication and processing requirements in Revision 1 of the DCD acceptable since the fabrication of stainless steel components will be consistent with RG 1.44 in regards to solution heat treatments, avoiding sensitization, and using ASTM A262 practices to determine susceptibility to intergranular corrosion.

As previously stated above, Subsection 4.5.1.1 of APR1400 DCD specifies the use of Types 304 and 316 stainless steel and their applicable filler metals (Types 308 and 316 SS), but did not provide further justification (e.g., discussing how design features of the APR1400 ensure these areas are not prone to increased levels of oxygen to minimize the likelihood of SCC of in 304 and/or 316 grades) addressing the acceptability of the use of normal carbon Type 304 and 316 materials and their applicable filler metals (Types 308 and 316), especially if they become sensitized. 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 SCC. In its response to RAI 523-8684, Question 04.05.01-16 (ML16335A463), the applicant provided 20 years of satisfactory operating experience of using the Versa Vent™ in operating reactors to reduce the dissolved oxygen levels so that no SCC has been identified. Therefore, the staff finds that the Type 304 and 316 stainless steel materials and Type 316L filler metal material are acceptable, based on operating experience of low oxygen levels provided by the Versa Vent™, low operating temperatures, and by using the guidance of RG 1.44 to control sensitization of the stainless steel materials to minimize SCC. Therefore, RAI 523-8684, Question 04.05.01-16, is resolved and closed.

Cold working can increase the susceptibility of SCC in austenitic stainless steels. DCD Tier 2 Subsection 4.5.1.2 states that strain hardened Type 300 austenitic stainless steel bolting or pin materials, if used, are controlled to have a 0.2 percent offset yield strength that is no greater than 620 MPa (90 ksi) to reduce the probability of SCC. This practice is consistent with SRP Section 4.5.1 when using strain hardened stainless steels, and is, therefore, acceptable.

DCD Tier 2 Subsection 4.5.1.2 states that the recommendations of RG 1.31 are used for the primary pressure retaining welds in the CRDM components to preclude microfissuring. RG 1.31 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). 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 SCC. RG 1.28 provides guidance on acceptable methods to control surface contamination caused by abrasive operations such as grinding. Since DCD Tier 2 Subsection 4.5.1.2 states that controls for abrasive work and cleaning on austenitic stainless steel surfaces are used to prevent cold work and contamination as specified in RG 1.28 and ASME NQA-1, the staff finds this acceptable.

Other Materials

Materials other than austenitic stainless steels are used to fabricate pressure boundary and non-pressure boundary CRDM components are listed below:

- martensitic stainless steels (ASTM A276,Types 410, 410 condition T and 440C),
- nickel-based alloys (AMS 5698, Alloy X-750 and AMS 5699, Alloy X-750),
- nickel-chromium-molybdenum-columbium alloy (ASTM B446, Alloy 625),
- cobalt-based alloys (Haynes No. 36 and Stellite No. 6B),
- SAE AMS 2460 chrome plating, and
- chrome oxide plasma spray treatment.

These materials include Type 410 martensitic stainless steel, cobalt-based 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 guide tubes are coated with chrome oxide.

DCD Tier 2 Subsection 4.5.1.1(b) identifies that portions of the latch magnet and center spacer assembly, lift magnet and latch spacer assembly, and the lower lift stop assembly are fabricated from Type 410 martensitic stainless. Typically, martensitic stainless steels are annealed where strength is not an issue. Using annealed Type 410 improves magnetic properties and reduces the risk of SCC and is, therefore, acceptable. However, for those Type 410 components that are quenched and tempered, the applicant does not specify the tempering temperature. SRP Section 4.5.1, Paragraph II.4 states that the tempering temperature of martensitic stainless steels should be specified for assurance that these materials will not deteriorate from SCC in service. Acceptable heat treatment temperature for Type 410 stainless steel is over 565 degrees Celsius (C) (1050 degrees Fahrenheit [F]). The staff issued RAI 303-8391, Question 04.05.01-9 (ML15314A597), requesting that the applicant provide the tempering temperature for Type 410, martensitic stainless steel, and confirm that other Type 410 martensitic stainless steel is provided in the annealed condition.

In its response to RAI 303-8391, Question 04.05.01-9 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.1(b), which included the heat treatment conditions for Type 410, martensitic stainless steel. However, the staff noted that the response did not include the tempering temperature of martensitic stainless steels in order to ensure that these materials will not deteriorate from SCC in service. Therefore, on March 8, 2016, the staff issued RAI 436-8538, Question 04.05.01-14 (ML16110A454), requesting that the applicant provide the tempering temperature of martensitic stainless steels.

In its response to RAI 436-8538, Question 04.05.01-14 (ML16110A454), the applicant revised DCD Tier 2 Subsection 4.5.1.1(b) that included the tempering temperature is higher than 565 degrees C (1050 degrees F) for the ASME A276, Type 410 stainless steels material. The staff finds the martensitic stainless steel acceptable, because the specified heat treatment condition including the heat treatment temperature are in accordance with the guidance in SRP Section 4.5.1, Paragraph II.4 to provide assurance that these martensitic stainless steels will not deteriorate form SCC in service.

Alloy X-750 (AMS 5698 and AMS 5699) is used for latch assembly and extension shaft assembly springs. The staff notes that the resistance of Alloy X-750 to SCC is dependent on adequate processing and heat treatment. APR1400 DCD Tier 2 Subsection 4.5.1.3 states that these springs are drawn from hot-finished rod (cleaned to remove surface imperfections and scale), which is heat treated to 1149 degrees C (2100 degrees F) 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 Korean plants provides further assurance that these materials will not be susceptible to SCC. Haynes 6B, 25 and 36 were discussed above in the "Materials Specification" section of this SER, concerning material in contact with the reactor coolant and were found to be acceptable.

As stated in APR1400 DCD Tier 2 Subsection 4.5.1.3, thermally treated Alloy 690 and Alloys 52/52M and 152 weld metals have shown favorable operating experience both in the U.S. and Korea, and have higher resistance to primary water SCC than Alloy 600 and its associated weld metals (Alloy 82 and 182). Therefore, based on the favorable operation experience, the staff finds the use of thermally treated Alloy 690 and Alloys 52/52M and 152 weld metals acceptable for use in the CRDM system, and is consistent with SRP Sections 4.5.1 and 5.2.3. In addition, APR1400 DCD Tier 2 Subsection 4.5.1.3 specifies that Alloy 625 is used for the gripper assembly and this material was tested to exceed the design life as described in APR1400 DCD Tier 2 Section 3.9.4. This material also has favorable operating experience, both in the U.S. and Korean nuclear plants. Therefore, the staff finds this material acceptable for use in the CRDM system based on the testing and favorable operating experience in current operating nuclear plants.

APR1400 DCD Tier 2 Subsection 4.5.1.1(b)(1) specifies chrome oxide (plasma spray treatment) as a material of construction (surface treatment) for the latch guide tubes. This material is understood by the staff to be intended to promote the reliable functioning of the CRD system and the design's ability to meet the requirements of GDC 26. The staff issued RAI 303-8391, Question 04.05.01-10 (ML15314A597), requesting that the applicant revise DCD Tier 2 Subsection 4.5.1.1(b)(1) to document a material specification for this material, in order to determine its acceptability in meeting the requirements of GDC 26.

In its response to RAI 303-8391, Question 04.05.01-10 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.1(b)(1), which provided the material composition as chromium (III) oxide (Cr_2O_3) and is applied using the plasma spray coating process onto the Type 316 stainless steel latch guide tubes. The material composition was included in Revision 1 of the DCD and this material has been used to enhance the surface finish and wear in the latch guide tubes in 12 plants in Korea with successful operating experience for over 20 years. Given the limited application of this material in the latch guide tubes for surface finish and wear, and the satisfactory performance of this material in operating plants, the staff finds the applicant's use of Cr_2O_3 acceptable.

Cleaning and Cleanliness Controls

Cleaning and cleanliness controls for the CRDM during manufacture and assembly are discussed in DCD Tier 2 Subsection 4.5.1.4, "Cleaning and Cleanliness Control." DCD Tier 2 Subsection 4.5.1.4 also references Subsection 5.2.3.4.2, "Cleaning and Contamination Protection," which provides additional information and is also applicable to the CRDMs. However, the staff needed clarification on cleaning and contamination protection requirements.

The staff issued RAI 303-8391, Question 04.05.01-8 (ML15314A597), requesting the applicant verify the NRC staff's understanding of DCD Tier 2 Subsection 4.5.1.4, which states, "[d]iscussions of the fabrication and processing of austenitic stainless steel provided in Subsection 5.2.3.4.2 are applicable to the cleaning and cleanliness control of the CRDMs." Therefore, the applicant was requested to confirm that this means "the cleaning and contamination protection requirements in Subsection 5.2.3.4.2 are applicable to the CRDMs," and revise the DCD accordingly.

In its response to RAI 303-8391, Question 04.05.01-8 (ML15356A554), the applicant revised DCD Tier 2 Subsection 4.5.1.4, which stated that the cleaning and contamination protection requirements in Subsection 5.2.3.4.2 of the APR1400 DCD are applicable to the CRDMs. Therefore, the staff finds the cleaning and contamination protection included in Revision 1 of the DCD acceptable, because the cleanliness of the stainless steel components will be consistent with RG 1.28 in regards to providing cleanliness and contamination protection during fabrication, shipment, and storage.

SRP Section 4.5.1 recommends that onsite cleaning and cleanliness controls for CRDMs should be in accordance with ASME NQA-1. To meet NQA-1, the applicant imposes strict process controls for cleaning and protection against contamination of materials during all stages of component manufacture and installation. Tools used in abrasive work on austenitic stainless steel, such as grinding do not contain, and are not contaminated with, ferritic carbon steel or other materials that could contribute to intergranular cracking or SCC. In addition, in DCD Subsection 4.5.1.4 the applicant states that onsite cleaning and cleanliness control will be carried out in accordance with ASME NQA-1 and provisions of RG 1.28. On this basis, the staff finds the applicant's onsite cleaning and cleanliness controls for CRDM components acceptable and consistent with SRP Section 4.5.1.

ITAAC: The ITAAC associated with DCD Tier 2 Section 4.5.1 are discussed in DCD Tier 2 Subsection 14.3.2.4 and delineated in DCD Tier 1, Section 2.4.1 and in Items 2.a, 3.a, and 4.a in Table 2.4.1 4, "Reactor Coolant System ITAAC." The ITAAC Items include the following:

- ITAAC Item 2.a specifies that the as-built ASME Code components identified in Table 2.4.1-2, including the CRDM, are designed and constructed in accordance with the ASME Code, Section III requirements.

- ITAAC Item 3.a specifies that the pressure boundary welds in the as-built ASME Code components identified in Table 2.4.1-2, including the CRDM, meet the ASME Code, Section III requirements.

- ITAAC Item 4.a specifies that the ASME Code components identified in Table 2.4.2-2, including the CRDM, retain their pressure boundary integrity at their design pressure based on the results of a hydrostatic test of the as-built components in accordance with ASME Code, Section III requirements.

Because the ASME Code Data Reports will provide the required information that verifies the as-built CRDM assemblies will be designed, constructed, inspected, and tested in accordance with Section III of the ASME Code, the staff finds ITAAC Items 2.a, 3.a, and 4.a acceptable.

Additional evaluation of ITAAC associated with CRDMs is in Section 5.4.1.2.4 of this SER.

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

The staff concludes that the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls to be acceptable since they satisfy regulatory requirements (10 CFR 50.55a, and 10 CFR Part 50, Appendix A, GDC 1, 14, and 26) and regulatory positions described above (for RCPB materials), including acceptable demonstration of appropriate materials selections and acceptable operating experience (for non-RCPB materials). Based on the above, the staff concludes that the design of the CRDM materials is acceptable and meets the requirements of GDC 1, 14, and 26, as well as 10 CFR 50.55a.

4.5.2 Reactor Internals and Core Support Materials

4.5.2.1 *Introduction*

This section of the DCD describes the RVIs and core support materials.

4.5.2.2 *Summary of Application*

DCD Tier 1: The Tier 1 information associated with this section is found in DCD Tier 1 Section 2.2.6, "Reactor Vessel Internals."

DCD Tier 2: The applicant has provided a Tier 2 design description in DCD Tier 2 Section 4.5.2, "Reactor Internals Materials," which is summarized here in part, as follows:

DCD Tier 2, Section 4.5.2, describes the materials used to fabricate RVIs and core support structures. The DCD provides information about the materials specifications, controls on welding, nondestructive examination (NDE), fabrication, and processing of austenitic stainless steel components, items concerning materials other than austenitic stainless steel, and consideration of additional degradation mechanisms. Each of these topics is discussed below.

Materials Specifications

The DCD requires that all RVIs and core support structures must satisfy the criteria of ASME Code, Section III, Article NG-2000. In addition, ASME Code Case N-60-5 is cited as approved in RG 1.84. The applicant provided detailed component-level material specification tables in Tables 4.5-2, "Reactor Internals and Core Support Structures Materials," and 4.5-3, "Reactor Internals and Core Support Structures Weld Materials." The subject components are primarily made from Type 304 austenitic stainless steel with Type 308L welding materials. Several components such as bolts, screws, pins, and similar components are made from a variety of stainless steels and nickel-based superalloy. Cold-worked austenitic stainless steel is only to be used in bolting or pin applications. Chrome plating and hardfacing is specified in several locations, as clearly stated in DCD markup provided in the supplemental letter. The applicant stated that the materials used in RVI and core support structures were selected for compatibility with the reactor coolant, consistent with ASME Code, Section III, Subsubarticles NG-2160 and NG-3120.

Controls on Welding

The DCD requires that welds in RVIs and core support structures be fabricated in accordance with the criteria of ASME Code, Section III, Article NG-4000, and the examination and acceptance criteria included in Article NG-5000. The control of welding is performed in accordance with ASME Code, Sections III and IX as appropriate. Welding is to be conducted consistent with RGs 1.31 and 1.44 as well.

Nondestructive Examination

The DCD requires that NDE of RVI and core support structure materials be in accordance with the requirements of ASME Section III, Subsection NG.

Fabrication and Processing of Austenitic Stainless Steel Components

The DCD requires that the recommendations of RG 1.44 be applied to control the use of sensitized austenitic stainless steel. Processes demonstrated not to produce sensitized structures are to be used in the fabrication of RVIs and core support structures. The DCD references DCD Tier 2 Subsection 5.2.3.4 for additional information pertaining to the use of austenitic stainless steel. The DCD specifies that raw austenitic stainless steel, both wrought and cast, is to be supplied in the solution-annealed condition as specified by the material specification. Solution treatment is not to be performed on completed or partially-fabricated components, rather chromium carbide precipitation is to be controlled during all stages of fabrication.

The DCD then requires conformance with RGs 1.31; 1.28; and 1.71, "Welder Qualification for Areas of Limited Accessibility," by citing DCD Tier 2 Subsections 5.2.3.4.4, 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steel," and 5.2.3.3, "Fabrication and Processing of Ferritic Materials," respectively. Conformance with these RGs covers ferrite control for RVI and core support materials; welder qualification for areas of limited accessibility; and quality assurance requirements for cleaning fluid systems and associated components of water-cooled nuclear plants.

Other Materials

The DCD Tier 2 Subsection 4.5.2.5, "Other Materials," states that the precipitation-hardened stainless steel used in the RVIs and core support structures is to be SA-638 Grade 660. The DCD states that SA-479, S21800 material is to be supplied in the annealed condition. The DCD identifies that SA-182 F6NM is to be solution heat treated at a specified temperature and then air cooled. Finally, the DCD identifies thermally-treated nickel-based Alloy 690 and Alloys 52/52M and 152 weld metals are to be used, and provides references supporting the primary water stress corrosion cracking (PWSCC) resistance of said superalloys.

Other Degradation Mechanisms

The DCD Tier 2 Subsection 4.5.2.6, "Other Degradation Mechanisms," provides a description of the methodology by which RVIs are assessed for susceptibility to irradiation-assisted stress corrosion cracking (IASCC) and void swelling. Components for which neutron fluence higher than 5×10^{19} n/cm^2 ($E > 1.0$ MeV) is estimated are modeled using computer techniques including the MCNP STAR-CCM+ computational fluid dynamics, and USERMAT codes. Based on the results of these assessments the effective stresses and volumetric changes of components of the APR1400 RVI are deemed to be below IASCC susceptibility stress and 2.5 volume percent change for the licensing term.

The staff reviewed DCD Tier 2 Section 4.5.2 in accordance with SRP Section 4.5.2, Revision 3. The design, fabrication, and testing of the materials used in the RVIs and core support structures are acceptable if they meet codes and standards commensurate with the safety functions to be performed.

ITAAC: No ITAAC associated with DCD Tier 2 Section 4.5.2 are identified.

4.5.2.3 *Regulatory Basis*

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

1. GDC 1, "Quality standards and records," and 10 CFR 50.55a as they require that SSCs important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are met, a facility that incorporates the DC is built and operated in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and the NRC's regulations.

The following acceptance criteria of SRP Section 4.5.2 are an acceptable way to meet the requirements of 10 CFR 50.55a and GDC 1:

Materials Specifications, Selection, and Heat Treatment

For core support structures and reactor internals, ASME Code, Section III, Article NG-2000 identifies and describes the permitted materials specification. ASME Code Cases approved for use identify additional permitted materials and their applications, as described in RG 1.84.

All materials used for RVIs and core support structures must be compatible with the reactor coolant, as described in ASME Code, Section III, Subsubarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels should be specified to provide assurance that these materials will not deteriorate in service.

Controls on Welding

Methods and controls for welding core support structures and RVIs must conform to ASME Code, Section III, Article NG-4000. The welds must be examined and meet the acceptance criteria specified in ASME Code, Section III, Article NG-5000.

Nondestructive Examination

NDE shall conform to the requirements of ASME Code, Section III, Subarticle NG-2500. The acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Subarticle NG-5300.

Austenitic Stainless Steels

SRP Section 5.2.3, Subsections II.2 and II.4.a, b, d, and e provide the acceptance criteria for the RVI materials. RG 1.44 describes acceptance criteria for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, as well as for determining the degree of sensitization that occurs during welding. RG 1.31 describes acceptable criteria for ensuring the integrity of welds in stainless steel components.

Other Materials

All materials used for RVIs and core support structures must be selected for their compatibility with the reactor coolant, as described in ASME Code, Section III, Subsubarticles NG-2160 and NG-3120. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are aging at 565 degrees C to 595 degrees C (1,050 degrees F to 1,100 degrees F) for Type 17-4 PH and tempering at 565 degrees C (1,050 degrees F) for Type 410 stainless steels. Other materials shall have similarly appropriate heat treatment and fabrication controls in accordance with strength and compatibility requirements.

4.5.2.4 *Technical Evaluation*

The staff divided its evaluation of the RVI and core support materials discussion in DCD Tier 2 Section 4.5.2, as supplemented by responses to various RAI questions in and letters dated November 12, 2015 (ML15316A780), and March 14, 2016 (ML16096A274), into six topics mapped to those described in SRP Section 4.5.2 plus “Other Degradation Mechanisms.” These topics are materials specifications, controls on welding, NDE, fabrication, and processing of austenitic stainless steel components, other materials, and other degradation mechanisms.

Materials Specifications

The DCD specifies that RVI and core support materials will satisfy the requirements of ASME Code, Section III, Article NG-2000. The staff finds this to be acceptable because applying ASME Code, Section III, Article NG-2000 requirements, which are for core support materials, is also appropriate for RVIs. The staff reviewed the specifications listed in DCD markup Tables 4.5-2 and 4.5-3 and confirmed that all specifications were to ASME Code, Sections II or III; or NRC-approved ASME Code Case specifications.

The staff noted that all unstabilized austenitic stainless steel materials are to be fabricated consistent with RG 1.44 to prevent sensitization to SCC. This is acceptable as conformance to RG 1.44 provides adequate assurance of the prevention of sensitization.

Additionally the staff noted that cold-worked materials are to be controlled to have yield strength less than 620 MPa (90 ksi) to avoid SCC; that the superalloy pins are designed to have no more than 275.79 MPa (40 ksi) load; and that these materials have performed adequately in operating reactors in the U.S. and Korea. Because the cold-worked materials are to be controlled to have yield strength less than 620 MPa (90 ksi) to avoid SCC, the loadings on superalloy pins will not exceed 275.79 MPa (40 ksi) load, and operating experience in the U.S. and Korea, the staff concludes the materials used are acceptable.

Controls on Welding

SRP Section 4.5.2 specifies that the methods and controls for core support structure and RVI welds must be performed in accordance with ASME Code, Section III, Division 1, Article

NG-4000, and the welds must be examined and meet acceptance criteria as specified in Article NG-5000. DCD Tier 2 Subsection 4.5.2.2, “Controls on Welding,” confirms that welds in the RVIs and core support structures will be fabricated and meet acceptance standards in accordance with the above. Welding is also to conform to the recommendations of RGs 1.31 and 1.44 as described in DCD Tier 2 Subsection 4.5.2.4, “Fabrication and Processing of Austenitic Stainless Steel Components.” Because the methods and controls conform to the recommendations of RGs 1.31 and 1.44, the staff concludes they are acceptable.

Nondestructive Examination

SRP Subsection 4.5.2.II.3 specifies that the acceptance criteria for NDE shall be in accordance with the requirements of ASME Code, Section III, Subarticle NG-5300. DCD Tier 2 Subsection 4.5.2.3, “Nondestructive Examination,” specifies that the NDE for RVI and core support materials will be performed in accordance with ASME Code, Section III, Subsection NG. Because using ASME Code Section III meets the acceptance criteria for GDC 1 and 10 CFR 50.55a, the staff finds this acceptable.

Fabrication and Processing of Austenitic Stainless Steel Components

The staff reviewed DCD Tier 2 Subsection 4.5.2.4 for its conformance to regulatory requirements. The staff confirmed that environmental conditions are controlled and welding procedures are developed such that the probability of sensitization and microfissuring are minimized. This is achieved by following the guidance of RGs 1.44 and 1.31, respectively. The staff confirmed the RVI and core support material compatibility with coolant through a review of the selection of materials for each component; commitment to RGs and ASME Code requirements; the topics detailed in DCD Tier 2, Subsections 4.5.2.4 and 4.5.2.6 and 5.2.3.4; and the water chemistry requirements regarding oxygen content in DCD Tier 2 Section 5.2, “Integrity of the Reactor Coolant Pressure Boundary,” Table 5.2-5, “Reactor Coolant Preoperation Operating Specifications.” The oxygen concentration requirement of 0-0.1 parts per million (ppm) ($T_{coolant}$ > 121.1 degrees C [250 degrees F]) is known to inhibit SCC in particular. The staff reviewed the fabrication and cleaning controls imposed on stainless steel components. These controls are principally enforced through adherence to RG 1.28 and ASME Code NQA-1 in addition to the extended discussion referenced in DCD Tier 2 Subsection 5.2.3.4 concerning cleaning chemicals, cleaning water chemistry, and halides. Because the fabrication and cleaning controls conform to the recommendations of RGs 1.31 and 1.44, staff concludes they are acceptable.

Other Materials

DCD Tier 2 Subsection 4.5.2.5 lists several materials as “Other Materials.” Precipitation-hardened stainless steels and SA-638 Grade 660 are identified with corresponding heat treatment requirements. In addition SA-479, S21800, a chromium-manganese-nickel type stainless steel, and SA-182, Grade F6NM, a martensitic stainless steel, are listed with heat treatment notes. Finally Alloy 690, 52/52M, and 152 superalloys are listed along with a supporting discussion of the resistant nature of these alloys regarding SCC. The staff evaluated the identified materials and associated heat treatments and found the discussion of “Other Materials” acceptable as they are consistent with precedent and regulatory requirements, specifically those of 10 CFR 50.55a.

Other Degradation Mechanisms

DCD Tier 2 Subsection 4.5.2.6 addresses “Other Degradation Mechanisms,” specifically IASCC and void swelling. The staff confirmed that the criteria used by the applicant for evaluating the internals with regards to IASCC and void swelling were appropriate. The staff reviewed the proprietary report upon which Subsection 4.5.2.6 was based, Technical Report APR1400-Z-M-NR-14017-P, Revision 0, “Evaluation of Irradiation Assisted Stress Corrosion Cracking and Void Swelling on Reactor Vessel Internals” (ML16096A281). This report provides substantial detail regarding the susceptibility determination process and results used by the applicant to ensure that IASCC and void swelling had been adequately accounted for. The staff reviewed both the process and the results in to confirm that the applicant had followed good practices and performed calculations of sufficient depth and quality to ensure that application of the above noted criteria would be adequate. The applicant performed radiation transport analysis, computational fluid dynamics analysis, and structural analysis to a granularity appropriate to identify peak locations in radiation and temperature in order to adequately apply the criteria. The staff determined that the codes and methods used, such as MCNP transport code, are consistent with good practice for this type of evaluation. The staff then compared the results with results for similar designs to confirm that the simulations were accurately implemented. Because the degradation mechanisms were confirmed to be appropriately addressed, the staff finds that the reactor internal materials meet the requirements of 10 CFR 50.55a with regards to the potential for IASCC and void swelling.

4.5.2.5 *Combined License Information Items*

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

4.5.2.6 *Conclusions*

On the basis of the information submitted, the staff concludes that the APR-1400 design of the RVI and core support materials satisfies the relevant requirements of 10 CFR 50.55a and GDC 1, and therefore, is acceptable. This conclusion is based on the fact that the APR-1400 RVI and core support structure materials satisfy ASME Code, Section III; RGs 1.28, 1.31, 1.44, 1.71, and 1.84; and conform to the guidance in SRP Section 4.5.2. Because there are no applicable ITAAC for this section, the requirements of 10 CFR 52.47(b)(1) are met.

4.6 Functional Design of Reactivity Control Systems

4.6.1 Introduction

The staff reviewed DCD Tier 2 Section 4.6, “Functional Design of Reactivity Control Systems,” to ensure that the reactivity control systems are capable of reliably controlling reactivity under both normal and accident conditions. For the APR1400, these systems are comprised of the safety-related CEDMs and SIS, and the nonsafety-related CVCS. The staff’s review focused on the functional performance of the CEDMs and the combined performance of the CEDMs and SIS, including consideration of single failure and common-cause failures.

4.6.2 Summary of Application

DCD Tier 1: The DCD Tier 1 information associated with this section is in DCD Tier 1, Section 2.4.1, “Reactor Coolant System”; Section 2.4.3, “Safety Injection System”; and Section 2.7.3.6, “Reactor Containment Building HVAC [Heating, Ventilation, and Air

Conditioning] System and Reactor Containment Building Purge System." The DCD Tier 1 information associated with the CVCS is in DCD Tier 1 Section 2.4.6, "Chemical and Volume Control System."

DCD Tier 2: The applicant provided a DCD Tier 2 system description in DCD Tier 2 Section 4.6 that is summarized as follows:

The APR1400 design includes three reactivity control systems: CEDMs, SIS, and CVCS. The APR1400 relies on the CEDMs or combination of CEDMs and SIS to prevent and mitigate DBEs. While the CVCS is not required for DBE mitigation, the applicant stated that the CVCS provides redundancy and enhances the reliability of the reactivity control systems.

The CEDMs and the DRCS comprise the CRD system. The safety-related function of the CRD system is to insert the CEAs into the reactor core via gravity when a reactor trip causes removal of motive power from the CEDM power bus. DCD Tier 2 Section 3.9.4, "Control Element Drive Mechanisms," provides the CEDM component descriptions, characteristics, and a diagram. DCD Tier 2 Section 7.7, "Control Systems Not Required for Safety," provides the functions and description of the DRCS. DCD Tier 2 Subsection 3.9.4.4, "CEDM Operability Assurance Program," discusses the testing and verification of the CEDMs. Finally, DCD Tier 2 Section 14.2, "Initial Plant Test Program," addresses the initial startup testing for the CEDMs.

DCD Tier 2 Table 4.6-1, "Design Basis Events," identifies all DBEs in DCD Tier 2 Chapter 15, "Transient and Accident Analyses," that require reactivity control system operation to prevent or mitigate each event as well as the associated system(s). DCD Tier 2 Chapter 15 describes the analyses for postulated accidents that assume actuation of the CRD system and/or SIS.

The SIS and CVCS are discussed in more detail in DCD Tier 2, Sections 6.3, "Safety Injection System," and 9.3.4, "Chemical and Volume Control System," respectively.

ITAAC: ITAAC associated with DCD Tier 2 Section 4.6 are given in DCD Tier 1, Table 2.4.1-4, "Reactor Coolant System ITAAC"; Table 2.4.3-4, "Safety Injection System ITAAC"; and Table 2.7.3.6-1, "Reactor Containment Building HVAC System and Reactor Containment Building Purge System ITAAC."

TS: The TS associated with DCD Tier 2 Section 4.6 are given in DCD Tier 2 Chapter 16, Sections 3.1, "Reactivity Control Systems," and 3.5, "Emergency Core Cooling System (ECCS)."

4.6.3 Regulatory Basis

The relevant requirements of NRC regulations for the functional design of reactivity control systems, and the associated acceptance criteria, are in NUREG-0800 Section 4.6, "Functional Design of Control Rod Drive System," and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800 Section 4.6. The requirements governing ITAAC related to the reactivity control systems are provided in NUREG-0800 Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria."

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

2. GDC 23, "Protection system failure modes," as it relates to the protection system failing into a safe state or into a state demonstrated to be acceptable on some other defined basis.

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

4. GDC 26, "Reactivity control system redundancy and capability," as it relates to the requirement that two independent reactivity control systems of different design principles shall be provided and capable of reliably controlling reactivity changes under conditions of normal operation, including anticipated operational occurrences, to assure that acceptable fuel design limits are not exceeded. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

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

6. GDC 28, "Reactivity limits," as it relates to the reactivity control systems being 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 pressure boundary nor disturb the core and its support structures to impair significantly the capability to cool the core.

7. GDC 29, "Protection against anticipated operational occurrences," as it relates to the protection system and reactivity control systems being designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

8. 10 CFR 52.47(b)(1), which requires that a DC application contain 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 are met, a facility that incorporates the DC is built and operated in accordance with the DC, the provisions of the AEA of 1954, as amended, and the NRC's regulations.

The related acceptance criteria are as follows:

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

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

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

4. To meet the requirements of GDC 26, two independent reactivity control systems of different design principles with the capability of providing sufficient operational control and reliability during reactivity changes during normal operations and AOOs should be provided.

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

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

7. To meet the requirements of GDC 29, the CRD system should be designed to ensure an extremely high probability of functioning during AOOs.

4.6.4 Technical Evaluation

The staff reviewed DCD Tier 2 Section 4.6 in accordance with NUREG-0800 Section 4.6. The staff evaluated the functional performance of the CRD system 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 CRD system and its combined performance with other reactivity control systems to ensure conformance with the requirements of GDC 4, 23, 25, 26, 27, 28, and 29.

In addition to RAIs addressing specific technical issues related to DCD Tier 2 Section 4.6, discussed in detail below, the staff issued RAI 136-8081, Questions 04.06-1 and 04.06-5, to clarify certain descriptive statements made by the applicant related to DCD Tier 2 Section 4.6. This technical evaluation does not discuss these editorial RAIs because they do not alter the substantive technical information provided by the applicant.

4.6.4.1 *Information for Control Rod Drive System*

DCD Tier 2 Section 4.6.1, “Information for the Control Rod Drive System,” describes the CRD system. The system consists of the CEDMs, which insert or withdraw the CEAs, and the DRCS, which actuates the CEDMs. The safety-related function of the CRD system is to insert the CEAs into the core when the reactor trip switchgear that receives the reactor trip signal removes electrical power from the DRCS, which in turn removes power from the CEDM coils. DCD Tier 2 Section 3.9.4 provides further description of the CEDMs. DCD Tier 2 Figure 3.9-7, “Control Element Drive Mechanism,” depicts a CEDM, and DCD Tier 2 Figure 3.9-8, “Reactor Internals Arrangement,” shows the CEDMs in relation to the reactor.

The CRD system should remain functional under adverse environmental conditions and after postulated accidents. The applicant stated that the pressure boundary components of the CEDMs are designed to withstand normal operating conditions, transient conditions, pipe breaks, and seismic and other loads, in accordance with ASME code requirements. As presented in DCD Tier 2 Table 3.2-1, “Classification of Structures, Systems, and Components,” the CEDMs and pressure housing assemblies are designed and tested in accordance with Section III of the ASME Code, Safety Class 1, are Quality Group A, and are designed to be seismic Category I. The staff notes that life cycle testing, production testing, and initial startup testing further ensure CEDM reliability. Section 3.9.4 of this SER includes the staff’s evaluation of the adequacy of the CEDMs to perform their mechanical functions, including the testing program and consideration of design loads, stress limits, and allowable deformations.

DCD Tier 2 Section 9.4.6, “Reactor Containment Building HVAC System and Purge System,” states that the CEDM cooling system is a subsystem of the nonsafety-related reactor containment building HVAC system and consists of three 50 percent capacity fans located on the integrated head assembly, two of which run during normal operation. Therefore, in the event that a single fan fails, the CEDM cooling system maintains its capability for adequate cooling. In accordance with the guidance in SRP Section 4.6, the staff confirmed that the CEDM cooling system meets the design requirements by auditing a design specification, as described in a staff audit report (ML16340A147).

DCD Tier 2 Section 7.2, “Reactor Trip System,” discusses the information regarding I&C related to reactor trip, and DCD Tier 2 Section 7.7 describes I&C for control systems, including the DRCS. Sections 7.2 and 7.7 of this SER evaluate the adequacy of these respective DCD sections.

4.6.4.2 *Evaluation of the Control Rod Drive System*

A single failure in the CRD system should not prevent the system from performing its safety-related function. Subsection 4.6.2.1 of DCD Tier 2, “Single Failure,” states that a failure in the DRCS does not prevent a reactor trip, as verified by a failure modes and effects analysis (FMEA), because the reactor trip switchgear removes motive power from the DRCS upon receipt of the reactor trip signal. This causes the CEAs to be inserted into the core by gravity. The staff audited the FMEA (see audit summary at ML17230A260), which demonstrated that no single failure in the DRCS could prevent a reactor trip or cause a reactivity transient through the dropping or uncontrolled stepping of control rods. In addition, DCD Tier 2 Section 7.2 provides an FMEA of the RPS that demonstrates that a single failure in the RPS does not prevent an RPS reactor trip.

Subsection 4.6.2.2 of DCD Tier 2, “Evaluation of the Control Rod Drive System,” briefly discusses isolation of the RPS from the nonessential I&C components. DCD Tier 2 Figure 7.7-2, “Digital Rod Control System - Reactor Protection System Interface Block Diagram,” shows the isolation of the motive power from the logic control signal at the DRCS power switches. This ensures DRCS operation has no adverse impacts on the essential function of reactor trip. Chapter 7 of this SER contains the staff’s evaluation of single failures regarding the RPS and DRCS.

The staff notes that the failure of a single CEDM would not prevent other CEDMs from inserting CEAs into the core because CEDMs operate independently. Therefore, the staff concludes that the safety-related reactor trip function is available in the event of a single failure in the CRD system. In addition, the staff notes that sufficient shutdown margin exists should a CEA fail to insert, as demonstrated in the transient and accident analyses. For these reasons, the staff concludes that the CRD system meets the requirements of GDC 23 with respect to the CRD system failing safe and GDC 25 with respect to a single malfunction of the CRD system not resulting in exceeding SAFDLs. However, CRD system requirements of GDC 23 and GDC 25 with respect to the I&C aspects of the protection system are evaluated in Chapter 7 of this SER.

DCD Tier 2 Subsection 4.6.2.3, “Protection from Common-Cause Failure,” states that the APR1400 uses physical separation, pipe whip restraints, and other means to physically isolate essential systems from postulated pipe breaks and associated missiles, as described in DCD Tier 2 Section 3.6, “Protection against Dynamic Effects Associated with the Postulated Rupture of Piping.” Section 3.6 of this SER contains the staff's evaluation of the protection of essential SSCs important to safety against postulated pipe breaks inside containment.

The staff concludes that the CRDS meets GDC 4 because its design and qualification described in Section 4.6(D)(a) of this SER, along with physical protection provisions, provide reasonable assurance that the CRDS will remain functional and provide safe shutdown capability under adverse environmental conditions and after postulated accidents.

4.6.4.3 *Testing and Verification of the Control Rod Drive System*

DCD Tier 2 Section 4.6.3, “Testing and Verification of the Control Rod Drive System,” refers to DCD Tier 2 Subsection 3.9.4.4 for the testing and verification of the CEDMs. DCD Tier 2 Subsection 3.9.4.4 states that the APR1400 CEDM is almost identical to the System 80 CEDM and describes the tests for the System 80 CEDM, including a life cycle test, scram test, and production tests. Section 3.9.4 of this SER contains the staff’s evaluation related to these tests. DCD Tier 2 Section 4.6.3 also refers to the initial startup test program for the CRD system in DCD Tier 2 Section 14.2. The following subsections contain the tests applicable to the CRD system and are evaluated in Section 14.2 of this SER:

1. 14.2.12.1.27, “Digital Rod Control System.”

2. 14.2.12.1.36, “Control Element Drive Mechanism Cooling System Test.”

3. 14.2.12.1.54, “Pre-Core Control Element Drive Mechanism Performance Test.”

4. 14.2.12.2.5, “Post-Core Control Element Drive Mechanism Performance.”

In addition, the hydrostatic test of the RCS described in DCD Tier 2 Subsection 14.2.12.1.35, “Reactor Coolant System Hydrostatic Test,” includes the CEDM pressure housing.

The staff notes that the above tests are comprehensive in that they assess CEDM operation during RCS heatup, CEA insertion and withdrawal at hot shutdown and HZP conditions, and CEA drop time at HZP. However, the staff felt the test acceptance criteria were not specific enough to verify some of the test objectives. The staff issued RAI 136-8081, Question 04.06-2 (ML15227A013), requesting the applicant to provide additional detail in the test acceptance criteria for each of the tests.

In its response to RAI 136-8081, Question 04.06-2 (ML15257A308), the applicant committed to update the initial startup test information and DCD sections the tests reference to include more clear and complete acceptance criteria. The updated tests and acceptance criteria meet the intent of RG 1.68, Revision 4, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” for the CRD system tests; therefore, the staff considers the initial startup test acceptance criteria acceptable. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 136-8081, Question 04.06-2, is resolved and closed.

Initial startup test 14.2.12.2.5 verifies proper operation and drop times of the CEDMs and CEAs, and TS SR 3.1.4.5 requires verification after reactor vessel head removal and prior to criticality that the maximum drop time from the fully withdrawn position to 90 percent inserted is at most 4 seconds. This specific drop time, also specified in DCD Tier 2 Subsection 3.9.4.1, "Descriptive Information of Control Element Drive Mechanism," ensures the appropriateness and conservatism of the drop times used in the safety analyses, as described in DCD Tier 2 Subsection 15.0.0.2.4, "CEA Insertion Characteristics."

According to SRP Section 4.6, the test program for the CRD system should include experimental verification of system operation where a single failure, such as a stuck rod, has been assumed. This was not addressed in the DCD. The staff issued RAI 136-8081, Question 04.06-3 (ML15227A013), requesting that the applicant show how system operation is experimentally verified in the event of a single failure. In its response to RAI 136-8081, Question 04.06-3 (ML15257A308), the applicant stated that an electrical malfunction in the DRCS would not disturb the reactor trip function, and a mechanical failure of a CEDM would not affect the other CEDMs because they operate independently. In addition, the applicant noted that the surveillance requirements (SR) in TS 3.1.4, "Control Element Assembly (CEA) Alignment," can detect a stuck rod. The applicant also committed to insert text into DCD Tier 2 Section 4.6.3 to reference this SR. Although the applicant does not propose tests with imposed single failures, the staff concludes that this test is not necessary because the previously discussed FMEA shows that possible single failures would not prevent the reactor trip function, and operators can identify single failures through TS SR or other indications, such as control room alarms. . Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 136-8081, Question 04.06-3, is resolved and closed.

The staff concludes that the CRD system meets the requirements of GDC 29 because the tests addressed above, along with the design of the CRD system previously discussed, ensure an extremely high probability that the CRD system will accomplish its safety function in the event of an AOO.

4.6.4.4 *Information for Combined Performance of the Reactivity Control Systems*

The reactivity control systems for the APR1400 design include the CEDMs, the SIS, and the CVCS. Only the CEDMs and SIS are required to prevent or mitigate DBEs, as identified in DCD Tier 2 Table 4.6-1. The CVCS is a nonsafety-related system that is not required for any accident mitigation or safe shutdown function. The CEDMs are completely independent from, and are based on different design principles and methods of reactivity control than, the SIS and CVCS. In addition, the CEDMs, SIS, and safety-related portions of the CVCS are protected from missiles, pipe breaks, and their effects. Therefore, the staff finds that the systems are not subject to common-mode failures.

DCD Tier 2 Section 4.6.5, "Evaluation of Combined Performance of the Reactivity Control Systems," states that the CVCS is designed for a high degree of redundancy and reliability. However, DCD Tier 2 Section 4.6 did not explain how the CVCS or the other reactivity control systems provided in the APR1400 design contribute to the redundancy and capability of controlling reactivity changes required by GDC 26. The staff issued RAI 136-8081, Question 04.06-4 (ML15227A013), requesting that the applicant clarify how the function of these systems provides redundancy in reactivity control. In its response to RAI 136-8081, Question 04.06-4 (ML15257A308), the applicant clarified that the two independent reactivity control mechanisms for normal operations, AOOs, and accidents are control rods and soluble boron. The SIS and CVCS use soluble boron. In addition, the CVCS provides long-term reactivity control during normal operations. Though the CVCS does not have the same degree of redundancy as the CEDMs and SIS, it has some redundant components to enhance system availability and reliability. DCD Tier 2 Section 9.3.4 describes this redundancy and states that the CVCS is capable of holding the reactor core subcritical under cold conditions. The applicant also included markup of DCD Tier 2 Section 4.6 that discusses compliance with GDC 26. The staff considers this response acceptable because it clarified how the APR1400 complies with GDC 26. Based on the review of DCD Tier 2, the staff has confirmed incorporation of the changes described above; therefore, RAI 136-8081, Question 04.06-4, is resolved and closed.

Because the CEDMs and CVCS are independent, operate on different design principles (control rods and soluble boron), and are designed with redundant components to ensure the systems can reliably control reactivity changes during normal operations, the staff finds that the design demonstrates the redundancy required by GDC 26. In addition, the SIS may be used to augment the CEDMs to control reactivity changes during AOOs or postulated accidents.

4.6.4.5 *Evaluation of Combined Performance of the Reactivity Control Systems*

GDC 27 states that the reactivity control systems must be capable of controlling reactivity changes during DBEs to maintain core cooling. DCD Tier 2 Section 4.6.5 states that the steam line break and CEA ejection accidents assume the combined actuation of the CRD system and SIS and refers to the analyses for these accidents in DCD Tier 2 Chapter 15 as demonstration that the CRD system and SIS reliably control reactivity under postulated accident conditions.

However, DCD Tier 2 Table 4.6-1 shows that feedwater line break (FLB), LOCA, letdown line break (LDLB), and steam generator tube rupture also require both the CEDMs and SIS for event mitigation. Furthermore, some of the required reactivity control systems for the DBEs listed in DCD Tier 2 Table 4.6-1 appear to be inconsistent with the transient and accident analyses. In particular, DCD Tier 2 Table 4.6-1 shows that the FLB, LDLB, and CEA ejection events require use of SIS to mitigate the event, while the Chapter 15 analyses for these events do not mention SIS actuation. The staff issued RAI 136-8081, Question 04.06-6 (ML15227A013), requesting that the applicant address the apparent inconsistencies.

In its response to RAI 136-8081, Question 04.06-6 (ML15257A308), the applicant clarified that the steam line break and CEA ejection accidents are just two examples that require combined reactivity control system actuation; DCD Tier 2 Table 4.6-1, contains the complete list. Regarding the apparent discrepancies for the FLB, LDLB, and CEA ejection events, the applicant explained that the most limiting cases assumed in the DCD Tier 2 Chapter 15 analyses do not describe all possible accident scenarios. For the FLB, different break sizes and locations can trigger a heatup or cooldown of the RCS in the initial stages of the accident. Although the case analyzed in Chapter 15 involved an initial RCS heatup and did not require SIS actuation, an accident with initial cooldown may require SIS actuation. For the Chapter 15 LDLB, the applicant assumed pressurizer heater actuation to maximize break flow, which maintains RCS pressure high enough to prevent a reactor trip or SIS actuation. Without the pressurizer heaters on, RCS inventory and pressure would decrease sufficiently to actuate the reactivity control systems. Similarly, the Chapter 15 CEA ejection analysis does not consider RCS depressurization or leakage that would occur in reality; such depressurization would actuate the reactivity control systems. With this clarification in mind, the staff notes that DCD Tier 2 Table 4.6-1 captures the reactivity control systems that, depending on the accident scenario, may be required. Therefore, RAI 136-8081, Question 04.06-6, is resolved and closed.

DCD Tier 2 Section 4.3 describes the shutdown reactivity requirements for the CRD system and states that the combined CEA worth, excluding the most reactive CEA and accounting for uncertainty, is sufficient for shutdown at all times during the cycle. DCD Tier 2 Section 6.3 describes the SIS and states that the safety injection function maintains the core subcritical during the extended period after a LOCA. In addition, DCD Tier 2 Section 6.3 states that the SIS safe shutdown function is capable of providing sufficient boron to maintain subcriticality during cold shutdown assuming the most reactive CEA is out of the core. The DCD Tier 2 Chapter 15 analyses for the DBEs identified in DCD Tier 2 Table 4.6-1 show that the reactivity control systems bring the core to a shutdown condition and maintain fuel integrity, consistent with the design information in DCD Tier 2, Sections 4.3 and 6.3. Based on the staff evaluation in Chapter 15 of this SER, the staff concludes that the reactivity control systems meet the requirements of GDC 27, in that they are capable of controlling reactivity changes during DBEs to maintain core cooling.

However, DCD Tier 2 Section 4.6 does not address how the reactivity control systems meet the requirements of GDC 28 such that the effects of postulated reactivity accidents can neither result in damage to the RCPB nor disturb the core and its support structures. The staff issued RAI 136-8081, Question 04.06-7 (ML15227A013), requesting that the applicant describe compliance with GDC 28.

In its response to RAI 136-8081, Question 04.06-7 (ML15257A308), the applicant stated that the design and operational requirements of the reactivity control systems limit reactivity changes due to system malfunctions or accidents. The CRD system design limits include the control rod material, structure, and group assignment. The applicant also referred to DCD Tier 2 Section 3.1.24, “Criterion 28 – Reactivity Limits,” which describes CEA set and group assignments, administrative procedures, and interlocks that ensure proper sequencing of CEA insertion and withdrawal. These measures prevent the rate of reactivity addition and individual CEA worth from exceeding limits. The applicant stated that TS 3.1.4, “Control Element Assembly (CEA) Alignment,” and TS 3.1.5, “Shutdown Control Element Assembly (CEA) Insertion Limits,” limit CEA insertion. The staff notes that TS 3.1.6, “Regulating Control Element Assembly (CEA) Insertion Limits,” and TS 3.1.7, “Part Strength Control Element Assembly (CEA) Insertion Limits,” specify additional CEA insertion limits.

In addition, DCD Tier 2 Section 3.1.24 states that the maximum rate of reactivity addition from CVCS operation is too small to threaten the RCPB or reactor internals. TS 3.1.8, “Charging Flow,” limits CVCS charging flow. DCD Tier 2 Section 3.1.24 also states that the applicant designed the RCPB and reactor internals to meet appropriate codes and standards and to accommodate loads associated with sudden releases of energy without rupture and with limited deformation so core cooling can be maintained.

The staff notes that the APR1400 reactivity control measures described in the applicant’s response to RAI 136-8081, Question 04.06-7, and the DCD are consistent with those utilized in the current operating fleet of PWRs, and together with TS 3.1.4 through 3.1.8, provide reasonable assurance that reactivity insertion due to CEAs and CVCS charging flow will be sufficiently limited during normal operations and AOOs. In addition, the Chapter 15 analyses demonstrate that the reactivity control systems perform acceptably and prevent damage that would inhibit core cooling capability, and all transient and accident acceptance criteria are met. Therefore, the staff finds that the APR1400 meets GDC 28.

4.6.5 Combined License Information Items

There are no COL information items associated with Section 4.6 of the APR1400 DCD.

4.6.6 Conclusions

The staff reviewed the functional design of the CRD system to confirm that the system has the capability to shut down the reactor with appropriate margin during normal operation, AOOs, and accident conditions, considering single failures. The staff also reviewed the combined performance of the CRD system and SIS to ensure capability to cool the core under AOO and postulated accident conditions. In addition, although it is not credited for accidents, the CVCS provides another means of reactivity control.

The staff has determined the adequacy of the applicant’s proposed design criteria, design bases, and safety classification of the CRD system and the requirements for providing a safe shutdown during normal operation, AOOs, and accident conditions, including single failures. For the reasons set forth above, the staff concludes that the design of the CRD system is acceptable and meets the requirements of GDC 4, 23, 25, 26, 27, 28, and 29.