# 4.0 Reactor

## 4.1 Introduction

In Revision 17 to the AP1000 design control document (DCD), Westinghouse Electric Company, LLC (Westinghouse or the applicant) proposed changes related to the reactor core and fuel design. By letter dated October 31, 2006, to the U.S. Nuclear Regulatory Commission (NRC), Westinghouse submitted Westinghouse Commercial Atomic Power (WCAP)-16652-NP, Revision 0, APP-GW-GLR-059, "AP1000 Core & Fuel Design Technical Report" (Technical Report (TR)-18), to justify the proposed changes. Various sections of DCD Chapter 4, Revision 17, incorporate the majority of these proposed changes. The staff evaluation of the proposed changes for each section of Chapter 4 of the AP1000 DCD is addressed in the corresponding section of this safety evaluation. This particular section describes the staff's evaluation of the proposed changes to DCD Section 4.1, "Summary Description."

In its letter of May 26, 2007 (DCP/NRC1912), regarding its application to amend the AP1000 design certification rule, Westinghouse referred to the criterion of Title 10, Section 52.63(a)(1)(vii), of the *Code of Federal Regulations* (10 CFR 52.63(a)(1)(vii)) and noted that these proposed changes contribute to increased standardization of the certification information.

### 4.1.1 Evaluation

General Design Criterion (GDC) 10, "Reactor Design," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that the reactor core and associated coolant, control, and protection systems be designed to assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). GDC 13, "Instrumentation and Control," requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. GDC 26, "Reactivity Control System Redundancy and Capability," requires that two independent reactivity control systems of different design principles be provided to ensure that SAFDLs are not exceeded and that the reactor core can be held subcritical under cold conditions. GDC 27, "Combined Reactivity Control Systems Capability," requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes to ensure that, under postulated accident conditions and with appropriate margins for stuck rods, the capability to cool the core is maintained. GDC 28, "Reactivity Limits," requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents will not result in damage to the reactor coolant pressure boundary (RCPB) or sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Various sections of Chapter 4 of the AP1000 DCD pertaining to the reactor core, fuel design, nuclear design, thermal-hydraulic design, and reactivity control system design describe compliance with these GDC.

DCD Section 4.1, “Summary Description,” describes the AP1000 reactor core and fuel design. The staff reviewed the proposed changes to DCD Section 4.1 and other sections in Chapter 4 for compliance with these GDC. In DCD Revision 17, Westinghouse proposed the following changes to DCD Tier 2, Section 4.1, related to the AP1000 reactor core and fuel design: (1) clarification of the fuel protective grid design and modification of the top nozzle design nomenclature, (2) modification of the gray rod cluster assembly (GRCA) design, (3) modification of the values of several parameters in DCD Section 4.1.1 and Table 4.1-1 for consistency, and (4) addition of four neutron panels on the thermal shield.

The current DCD Section 4.1 states that the top and bottom grids of the fuel assembly design do not contain mixing vanes. In DCD Revision 17, the applicant revised this sentence to state that the top and bottom grids and the protective grid do not contain mixing vanes. This clarification specifically states that the protective grid also does not contain mixing vanes. In DCD Table 4.1.1, the applicant revised the number of grids per fuel assembly to specifically identify one protective grid. This is a clarification of the design, not a design change, and is, therefore, acceptable.

In DCD Revision 17, Westinghouse proposed to change “integral clamp top nozzle (ICTN)” to “Westinghouse integral nozzle (WIN)” in the fuel assembly design. In TR-18, Westinghouse stated that the change from a reconstitutable ICTN to WIN in the fuel assembly is a nomenclature change. The WIN design is a proven enhancement to the ICTN design and is currently in use in the Westinghouse fleet. Both the ICTN and WIN designs eliminate the need for the top nozzle spring screws and spacer clamps. There is no significant design difference between the ICTN and WIN. Westinghouse described the design refinements of WIN, as compared to the ICTN, in a letter dated April 19, 2004, regarding the fuel criterion evaluation process (FCEP) notification of the WIN design (proprietary and nonproprietary) (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML041120331 and ML041120332). The staff finds that Westinghouse developed the WIN design using the fuel criteria evaluation process described in WCAP-12488. Therefore, the change as presented in DCD Revision 17, is acceptable.

In DCD Revision 17, Westinghouse proposed to change the GRCA design from 4 rodlets (out of 24) containing silver-indium-cadmium (Ag-In-Cd) to 12 rodlets containing Ag-In-Cd with reduced diameter, as described in DCD Section 4.2.2.3.2. The applicant revised DCD Table 4.1-1 to reflect this change. As discussed in Section 4.2 of this report, the staff finds the change to the GRCA design acceptable.

In DCD Section 4.1.1, the applicant changed the calculated core average power limit in the principal design requirement from 5.71 kilowatts per foot (kW/ft) to 5.718 kW/ft. The value of 5.71 kW/ft is a truncation of the actual value of 5.718 kW/ft. DCD Tables 4.1-1 and 4.3-2 round the core average linear power to 5.72 kW/ft. Therefore, the staff finds the change in Section 4.1.1 to 5.718 kW/ft acceptable. The applicant also revised several parameters in DCD Table 4.1-1. The effective reactor coolant flow area of heat transfer, the average velocity along the fuel rods, and the average mass velocity were revised slightly to be consistent with DCD Table 4.4-1 and the revised definition of core flow area and the core bypass flow described in DCD Section 4.4. Section 4.4 of this report addresses the revision to the core flow area and bypass.

The applicant revised DCD Table 4.1-1 to include four neutron panels on the thermal shield. Westinghouse made this revision to be consistent with the reactor internals design changes, which call for the addition of four neutron panels to reduce the reactor vessel fluence at the

circumferential locations that have the highest fluence values, along with the addition of a flow skirt in the lower reactor vessel head, as described in WCAP-16716-NP, Revision 2, "AP1000 Reactor Internals Design Changes" (TR-29). Since these changes could affect the core inlet flow distribution and the flow area and flow resistance in the reactor vessel downcomer and lower plenum, the staff requested that Westinghouse provide an evaluation of the impacts of these proposed changes on the analysis results of each of the transients and accidents described in DCD Chapter 15, "Accident Analyses."

In response to Request for Additional Information (RAI)-TR29-SRSB-01, Westinghouse presented the results of the evaluation and safety analyses of the reactor vessel internals design changes. This evaluation also included the pressurizer changes described in APP-GW-GLR-016, Revision 0, "AP1000 Pressurizer Design" (TR-36). (In its letter of July 18, 2008, in response to RAI-SRP10.3-SBPA-02, Westinghouse also stated that the evaluation of the limiting Chapter 15 event analyses provided in RAI-TR29-SRSB-01 included the revised main steam safety valve setpoints and capabilities.) The applicant concluded that these reactor internals changes have minimal impact on the fluid volume, metal masses, pressure drop through the reactor vessel, and the design reactor coolant system (RCS) flow rates used in the safety analyses. The applicant performed the evaluation or analyses on the limiting event of each of the event categories discussed in DCD Chapter 15. The analysis results for these events demonstrate that the applicable acceptance criteria for each event are met or that the existing analysis is bounding.

In its June 30, 2008, letter, Westinghouse provided APP-GW-GLE-026, Revision 0, "Application of ASTRUM Methodology for Best Estimate Large Break Loss of Coolant Accident Analysis for AP1000." Subsequently, Westinghouse in a letter dated February 3, 2009 (ADAMS Accession No. ML090410367), submitted APP-GW-GLE-026, Revision 1. This technical report describes a reanalysis of a large-break loss-of-coolant accident (LOCA) using the NRC-approved uncertainty treatment method, Automatic Statistical Treatment of Uncertainty Method (ASTRUM). This large-break LOCA reanalysis includes the reactor vessel internals design changes discussed above. The results of this large-break LOCA analysis with the design changes to date show compliance with the acceptance criteria of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." In the review of large-break LOCAs, Section 15.2.6.5 of this report addresses the staff evaluation of this technical report.

DCD Section 4.1.1, "Principal Design Requirements," lists the criteria, or principal design requirements, that must be met by the mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems. These principal design criteria are designated Tier 2* information, which indicates that NRC staff approval will be required before implementation of any change to this information. Various sections of Chapter 4 of this report describe other Tier 2* information related to the reference core design. In Section 4.1 of the final safety evaluation report (FSER), the staff states that the following sections in DCD Tier 2, Chapter 4, include Tier 2* information:

- 4.1 WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," issued October 1994
- 4.1.1 Principal Design Requirements

- 4.3.1.1 Maximum Fuel Rod Average Burnup of 62,000 Megawatt-Days per Metric Ton of Uranium (MWDF/MTU)

- Table 4.3-1 Reactor Core Description (First Cycle)

- Table 4.3-2 Nuclear Design Parameters (First Cycle)

- Table 4.3-3 Reactivity Requirements for Road Cluster Control Assemblies

In DCD Revision 16, Westinghouse proposed to reclassify all Tier 2* information throughout Chapter 4 as Tier 2 information to allow future changes to this information to be implemented without prior NRC staff approval. However, in response to RAI-SRP4.2-SRSB-01, Westinghouse stated that it will withdraw the request that Tier 2* items be reassigned to Tier 2 status in Chapter 4. Therefore, all Tier 2* items in DCD Chapter 4, including the principal design requirements, remain Tier 2* items. In DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15 is restored to Tier 2*. In DCD "Introduction," Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change," these Tier 2* items, which were included in DCD Revision 15 but deleted in Revision 16, have been restored. The staff finds this acceptable.

### 4.1.2 Conclusion

Based on this evaluation, the staff concludes that the changes to DCD Section 4.1, as described in DCD Revision 17, continue to meet the requirement of GDCs 10, 13, 26 and 28 as described above. In addition, the results of the large break LOCA analysis with design changes to date show compliance with the acceptance criteria of 10 CFR 50.46. Therefore, the proposed changes are acceptable.

## 4.2 Fuel System Design

Revision 17 of the AP1000 DCD contains the following changes to Section 4.2 (as compared to Revision 15): (1) reference changes, (2) densification value, (3) control rod descriptions, (4) burnable absorber rod design, (5) debris protection package description, (6) top nozzle nomenclature, and (7) grid fabrication description.

The staff based its review of the AP1000 fuel design on the information contained in the DCD and the TRs referenced by the applicant. The review was limited in scope to the changes to DCD Revision 15, as presented in Revision 17. The staff conducted its review in accordance with the guidelines provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), Section 4.2, "Fuel System Design," which prescribes acceptance criteria to ensure that certain requirements of 10 CFR Part 50 are met. In particular, the AP1000 fuel design must meet the following GDC:

- GDC 10, "Reactor Design"
- GDC 27, "Combined Reactivity Control Systems Capability"
- GDC 35, "Emergency Core Cooling"

The fuel design must also meet the requirements of 10 CFR Part 100, "Reactor Site Criteria." Thus, in reviewing the AP1000 fuel system design, the staff's objective was to ensure that the design fulfills the following criteria:

- The fuel system will not be damaged during any condition of normal operation, including the effects of AOOs.
- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.
- The number of fuel rod failures for postulated accidents is not underestimated.
- Coolability is always maintained.

The term "will not be damaged" means that the fuel rods will not fail, the fuel system's dimensions will remain within operational tolerances, and the functional capabilities will not be reduced below those assumed in the safety analysis. These objectives address GDC 10, and the design limits that accomplish these objectives are referred to as SAFDLs. In a "fuel rod failure," the fuel rod leaks and the first fission product barrier (i.e., the fuel cladding) is breached. The applicant must account for fuel rod failure in the dose analysis for postulated accidents required by 10 CFR Part 100. The radiological dose consequences criteria given in 10 CFR 50.34(a)(1) are referenced in 10 CFR 100.21 "Non-Seismic Siting Criteria." Compliance with dose consequence criteria in 10 CFR 50.34(a)(1), with the site parameters postulated for the design, is discussed in Section 15.3 of this report.

"Coolability," which is sometimes termed "coolable geometry," is the ability of the fuel assembly to retain the geometrical configuration of its rod bundle with adequate coolant channel spacing for removal of residual heat. GDC 27 and 35 specify the general requirements for maintaining control rod insertability and core coolability. In addition, 10 CFR 50.46 establishes specific requirements for the performance of the ECCS following postulated LOCAs. Compliance with Section 50.46 is discussed in section 15.2.6.5 of this report.

### 4.2.1 Evaluation

The applicant made various reference changes in Revision 17. When needed, Westinghouse included additional references. In other cases, the applicant replaced references to correct erroneous references found in Revision 15. For example, the applicant deleted an erroneous reference for the integral fuel burnable absorber design in Section 4.2.2.1. In RAI-SRP4.2-SRSB-04, the staff requested that the applicant include the correct reference instead of simply removing the incorrect reference. In its response, Westinghouse committed to add a reference to WCAP-12610-P-A to the conclusion of Section 4.2.2.1. DCD Section 4.2.6 includes this report as Reference 5. The NRC staff has reviewed these changes and finds them acceptable.

Throughout Chapter 4.2, Westinghouse revised the description of the control rod assemblies to clarify that the control rod clusters consist of both rod cluster control assemblies (RCCA) and gray rod cluster assemblies (GRCA). The term gray rod refers to clusters of neutron absorber rods with reduced rod worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for

changes to the concentration of soluble boron. Additionally, the applicant changed the GRCA design. Specifically, the applicant increased the number of rodlets from 4 to 12 and correspondingly reduced the diameter such that the overall worth of the GRCA is essentially the same, as explained in the applicant's response to RAI-SRP4.2-SRSB-05, but the assembly power profile is more even. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation. DCD Revision 17 does not present the impact of these GRCAs on determining the worst-case scenarios for accident analyses. The applicant's response to RAI-SRP4.2-SRSB-05 indicated that the Chapter 15 analyses remain bounding because the use of the newer GRCA design results in a more uniform rod power throughout the affected assemblies and does not affect the worst-case assumptions used in the accident analyses. The NRC staff finds this acceptable based on the RAI response and the nonsafety status of the GRCAs.

Westinghouse changed the burnable absorber rod design from wet annular burnable absorbers (WABAs) to borosilicate glass. However, the reference for the borosilicate design presented (WCAP-7113) is from 1967 and does not appear to have been previously approved by the NRC. Additionally, the DCD continues to list the WABA design as an alternative. The staff issued RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB-03, and RAI-SRP4.2-SRSB-07 regarding these items. The applicant's response pointed out previous NRC approval of Reference Safety Analysis Reports (RESARs), that used the same reference (WCAP-7113), covering borosilicate glass RCCAs, but Westinghouse did not provide a specific RESAR reference. The applicant also referred to operational experience with the borosilicate glass rodlets, including a single failure of a rodlet that became detached and remained in the guide tube. In response to RAI-SRP4.2-SRSB-07, the applicant states that borosilicate burnable absorbers were used previously in a Westinghouse core design with a 14-ft core and a feed fuel enrichment of 4.4%. This is very similar to the AP1000 core. The response further explained that there are no changes to rodlet diameter and no changes to boron concentration for the AP1000 design as compared with the previously approved design. Based on the responses to RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB03, and RAI-SRP4.2-SRSB-07, the staff concludes that the use of borosilicate glass burnable absorbers is acceptable.

The applicant provided further description of the debris mitigation package in DCD Sections 4.2.2 and 4.2.2.1. Specifically, the applicant explicitly listed the parts of the debris mitigation package that includes a zirconium oxide coating on the bottom section of the fuel cladding.

Westinghouse changed the top nozzle design in Section 4.2 (and other sections) from "Integral Clamp Top Nozzle (ICTN)" to "Westinghouse Integral Nozzle (WIN)." As described in Section 4.1.1 of this SER, the NRC staff found this change to be acceptable.

The applicant further explained the grid fabrication process in Section 4.2.2.2.4. This information did not appear to change the grid design from that presented in DCD Revision 15 and is therefore acceptable.

In accordance with APP-GW-GLR-119 (TR-119), the applicant edited DCD Revision 16 to remove Tier 2* designation from various fuel-related items. The applicant argued, in TR-119, that these items would be provided in a future "core reference report" and as such they could be reclassified as Tier 2. In RAI-SRP4.2-SRSB-01, the staff expressed concern that the reclassification could theoretically be interpreted to allow changes to the affected items after the

core reference report was approved. In response, Westinghouse stated that it would withdraw the request for reclassification of all items presented in TR-119 and restore, in DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15. The staff verified that Section 4.2 of Revision 17 contains all of the items designated as Tier 2* in Revision 15. In addition, Revision 17 for Table 1-1 of the DCD "Introduction" also contains these Tier 2* items, which require NRC approval for change. This staff finds this acceptable.

In Revision 15 of the DCD, Sections 4.2.5, 4.3.4, and 4.4.7 state that combined license (COL) applicants referencing the AP1000 certified design will address changes to the reference design presented in the DCD to the fuel burnable absorber rods, RCCAs, or initial core design. DCD Tier 2, Table 1.8-2, lists these COL applicant actions as COL Information Items 4.2-1, 4.3-1, and 4.4-1. In DCD Revision 17, Westinghouse proposed to revise Sections 4.2.5, 4.3.4, and 4.4.7.1 to state that (1) APP-GW-GLR-059 completely addressed the COL information requested in these sections, (2) the DCD incorporated applicable changes, and (3) no additional work is required by the COL applicant to address the COL information requested in this section. In DCD Revision 17, the applicant revised Table 1.8-2 to require no COL applicant action for COL Information Items 4.2-1, 4.3-1, and 4.4-1. These COL information items intend for the COL applicant to provide information regarding the changes to the referenced reactor core design in the AP1000 design certification. To increase standardization of the certified design, Westinghouse submitted APP-GW-GLR-059, which provides information that addresses changes to the reference design of the fuel burnable absorber rods, RCCAs, or initial core design in Revision 15 of DCD. DCD Revision 17 incorporates the applicable changes.

Based on the evaluation discussed in this safety evaluation report (SER), the staff concludes that COL Information Items 4.2-1, 4.3-1, and 4.4-1 associated with DCD Revision 15 have been completed, and Revision 17 of DCD Sections 4.2.5, 4.3.4, and 4.4.7.1 is acceptable. However, it should be noted that the referenced design of the fuel burnable absorber rods, RCCAs, and initial core design parameters described in DCD Section 4 are classified as Tier 2* information. It is likely that future advancements in the design of these items could occur and COL applicants might desire these improved designs. Any change to the Tier 2* information will require prior approval by the NRC staff.

### 4.2.2 Conclusion

The staff concludes that the AP1000 fuel system, as defined by the DCD Revision 17 changes to the approved design, 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; and 10 CFR Part 50.34, "Contents of Construction Permit and Operating License Applications; Technical Information."

The applicant provided sufficient information indicating that the changes to the approved design, as detailed in DCD Revision 17, meet the guidance provided in SRP Chapter 4.2.

## 4.3 Nuclear Design

The staff reviewed Section 4.3 in Revision 17 of the AP1000 DCD. The staff conducted its evaluation in accordance with the guidelines provided in SRP Section 4.3, "Nuclear Design."

In DCD Revision 17, Section 4.3, “Nuclear Design,” Westinghouse proposed changes to the following areas related to the AP1000 nuclear design: (1) online monitoring of power distribution, (2) gray rod assembly design, (3) criticality design method outside the reactor, including soluble boron credit methodology, (4) deletion of a specific value for the moderator temperature change that accounts for the control system deadband, and (5) typical control bank worth for the initial cycle. In support of these changes, Westinghouse submitted APP-GW-GLR-059. Westinghouse originally proposed in DCD Revision 16 to reclassify the Tier 2* information in DCD Section 4.3 to Tier 2. However, in DCD Revision 17, Westinghouse rescinded this proposal and restored this information to its original Tier 2* designation.

### 4.3.1 Evaluation

DCD Tier 2, Section 4.3, “Nuclear Design,” presents the design bases for the AP1000 nuclear design. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including AOOs, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core.

Section 4.3 of the SRP outlines relevant requirements of Commission regulations for this area of review and the associated acceptance criteria, which include the following:

- GDC 13 requires a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

- GDC 26 requires, in part, a reactivity control system capable of holding the reactor subcritical under cold conditions.

- GDC 28 requires, in part, that the reactivity control systems be designed to limit reactivity accidents so that the RCS boundary is not damaged beyond limited local yielding.

In Section 4.3.2.2.6, Westinghouse modified the text to state, “Online monitoring system is not a required element for a short term reactor operation,” and added the statement, “Limits are placed on the axial flux difference so that the heat flux hot channel factor (FQ) is maintained within acceptable limits.” In Section 4.3.2.2.9, Westinghouse modified the text to state, “the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service.”

During the review, the staff requested additional information on why online monitoring system outage is not a required element for a short-term operation. In response to RAI-SRP4.3-SRSB-01, Westinghouse stated the following:

> in the unlikely event that the Online Power Distribution Monitoring System (OPDMS) should become inoperable, reactor operation can continue with shutdown margin and power distribution controls established by bounding analyses and implemented by Technical Specification Limiting Conditions for Operations (LCOs) 3.1.5, 3.1.6, 3.2.1, 3.2.2, 3.2.3 and 3.2.4. These LCOs become applicable immediately or as otherwise specified in the associated Technical Specification applicability statements, when the OPDMS is inoperable.

The staff agrees with the RAI response and finds the proposed change acceptable. In Section 4.3.2.2.6 of the DCD, the phrase “short-term” refers to the time needed to restore the online power distribution monitoring system to operable status.

In the event that the online monitoring system is out of service for a short time, the in-core and ex-core detectors provide the operator with the necessary information regarding the power distribution based on the bounding and precalculated analysis. Therefore, the online monitoring system is not a required element. The staff finds the proposed change acceptable. Westinghouse provided a clarification change to Sections 4.3.2.2.4, 4.3.2.2.6, and 4.3.2.2.9 regarding the axial, limiting power distribution, and monitoring instrumentation. The clarification states that, in the event the online monitoring system is out of service, limits placed on axial flux differences are designed so that the heat flux hot channel factor and departure from nuclear boiling ratio (DNBR) are maintained within acceptable limits. The staff finds this clarification to be acceptable.

In Section 4.3.2.4.13, Westinghouse proposed to change the number of GRCA Ag-In-Cd rodlets from 4 to 12 reduced-diameter rodlets and to change the stainless steel rodlets from 20 to 12. The total number of rodlets is unchanged at 24. The applicant made a related change to the description of the GRCAs in DCD Section 4.3.2.2.2, Figures 4.3-8 and Figure 4.3-11. Specifically, the applicant changed the gray bank from M0 to MA+MB. This change also applies to the corrections made to the titles of Figures 4.3-8 and 4.3-11. In Section 4.3.2.4.16, Westinghouse added the statement, “Gray rod operation is a Condition I event which includes the periodic exchange of gray rod banks.”

The term “gray rod” refers to the reduced reactivity worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation.

This change affects information contained in Sections 4.3 and 4.3.2.4.13, as well as Table 4.3.1. This change is not expected to impact the conclusion of Chapter 15 and is therefore acceptable. Condition I comprises normal operation and operational transients that are accommodated with margin between any plant parameter and the value of that parameter requiring either automatic or manual protective action. The staff agrees with categorizing GRCA operation as a Condition I event and finds the proposed addition of the statement, “gray rod operation is a Condition I event which includes the periodic exchange of gray banks,” to Section 4.3.2.4.16 acceptable.

In Section 4.3.2.6.1, “Criticality Design Method Outside the Reactor,” the applicant deleted the formula describing the total uncertainty of the criticality calculation, and added the soluble boron credit methodology to Section 4.3.2.6.2. The methodology used in soluble boron credit analysis references WCAP-14416-P (Reference 63). This WCAP is not approved by the NRC; therefore this proposed change is not acceptable and requires further review by the NRC staff. The staff identified the issue in RAI-SRP4.3-SRSB-03. Since this issue was also identified in RAI-SRP9.1.1-SRSB-05 pertaining to the spent fuel storage rack criticality analysis, the staff designated this as Open Item OI-SRP9.1.1-SRSB-01. In response to RAI-SRP9.1.1-SRSB-05, which also serves as the response to RAI-SRP4.3-SRSB-03, Westinghouse indicated that it has completely revised the AP1000 Spent Fuel Pool (SFP) criticality analysis in APP-GW-GLR-029, Revision 0, “Spent Fuel Storage Racks Criticality Analysis,” with a new methodology that meets

10CFR 50.68 requirements. The new analysis is APP-GW-GLR-029, Revision 1 (ADAMS Accession No. ML091660338), entitled "AP1000 Spent Fuel Racks Criticality Analysis" which is a complete rewrite to supersede Revision 0. In addition Westinghouse has stated that the new method and models replace specific shortcomings that the NRC identified as being no longer reliable as "approved methodology." In a letter dated September 29, 2009 (ADAMS Accession No. ML093170675), Westinghouse submitted APP-GW-GLR-029, Revision 2, which is identical to Revision 1, except for minor updates to include pyrex insert burnable absorber design among other burnable absorbers. These new efforts include evaluation of soluble boron. Westinghouse also proposed to revise DCD Sections 4.3.2.6.1 and 4.3.2.6.2 to be consistent with the methodology described in APP-GW-GLR-029, Revision 2. The staff evaluation of spent fuel criticality analysis of APP-GW-GLR-029, Revision 2, will be addressed in Section 9.1.2.2.4 of this SER. The staff identifies Confirmatory Item **CI-SRP9.1.1-SRSB-01** to verify that APP-GW-GLR-029, Revision 2 has been approved by the staff and that DCD Sections 4.3.2.6.1 and 4.3.2.6 are revised as stated by the applicant.

In Section 4.3.2.4.2, Westinghouse proposed to delete the specific value of the 4-degree-Fahrenheit (°F) moderator temperature increase that accounts for the control system deadband. The shutdown margin control requirements calculation accounts for the control system deadband and measurement uncertainties by assuming that the moderator temperature is at its maximum possible value before plant trip. This conservatively increases the change in moderator temperature when going from hot full power to hot zero power after plant trip, thereby increasing the shutdown margin control requirement. The uncertainty is based on conservative engineering judgment and includes both instrument errors and deadband. The AP1000 analyses used a preliminary $T_{avg}$ uncertainty, which is provided in the DCD. The applicant revised Section 4.3.2.4.2 to reflect that the allowance for deadband and measurement errors is not set at a fixed value of 4 °F. This change does not affect the conclusion of Chapter 15, and, therefore, it is acceptable.

Westinghouse revised Tables 4.3-1 and 4.3-2 and Sections 4.3.2.2.7, 4.3.4, and 4.3.5 to reflect the DCD Chapter 15 accident analysis input assumptions and its results. The applicant revised the fuel assemblies' diameter of guide thimbles and lower part to be consistent with the other dimension values given in the table. The applicant revised the nuclear design parameters, reactivity coefficient, and doppler coefficients in Table 4.3-2 to correctly refer to Figure 15.0.4-1 and to be consistent with the DCD Chapter 15 accident analysis. The applicant revised Section 4.3.4 to reflect COL information updates. The applicant also updated Sections 4.3.2.2.7 and 4.3.5 to correct, delete, and add new references. These changes provide consistency with other DCD sections and do not impact the conclusion of Section 4.3 or Chapter 15; therefore, they are acceptable.

In Table 4.3-2, Westinghouse provided first-cycle values for the typical hot channel factors and bank worth from beginning of life to end of life. The changes to these values represent first-cycle updated results of the in-core fuel management scheme and are acceptable.

In Section 4.3.2.4.16, Westinghouse revised mechanical shim load follow and base load operations (including the gray rod bank insertion sequence exchanges) to establish a more negative value than the axial offset associated with all-rods-out condition. The staff considers this to be a conservative change and it is, therefore, acceptable.

The applicant revised Section 4.3.4 to state that APP-GW-GLR-059 (Reference 64) completely addresses the COL information requested in this section, and the DCD incorporates applicable

changes. No additional work is required by the COL applicant to address the COL information requested in this section. This change is acceptable as discussed in Section 4.2 of this report.

Revision 17 of DCD Section 4.3 includes numerous editorial and clarification changes concerning specifications that are covered in the core operating limits report, general formatting changes to the references, and clarifications to the variable units used in Section 4.3 of the DCD. The staff finds these clarifications to be acceptable.

### 4.3.2 Conclusion

The staff reviewed the changes to DCD Revision 17 that demonstrate that sufficient control rod and burnable poison worth exist to provide safe shutdown of the plant. Furthermore, the control rod system is designed to ensure that reactivity accidents do not result in damage to the RCPB. Therefore, the staff finds that the proposed DCD changes comply with the requirements of GDC 13, 26, and 28 and are acceptable. The changes to Sections 4.3.2.6.1, and 4.3.2.6.2, as discussed above, are considered as Confirmatory Item **CI-SRP9.1.1-SRSB-01** pending resolution. The staff expects the above changes to apply to all COL applications referencing the AP1000 design certification.

## 4.4 Thermal-Hydraulic Design

In Revision 17 to the AP1000 DCD, Westinghouse proposed changes related to the reactor core thermal-hydraulic design. By letter dated October 31, 2006, Westinghouse submitted WCAP-16652-NP, Revision 0, APP-GW-GLR-059 (TR-18), which provides rationale and justification for the proposed changes related to the reactor core and fuel design. In APP-GW-GLN-019, "AP1000 Standard Combined License Technical Report, Fluid System Changes" (TR-108), Westinghouse proposed revisions to the AP1000 fluid systems, including the digital metal impact monitoring system (DMIMS). DCD Revision 17, Section 4.4, includes the proposed changes that affect the reactor core thermal-hydraulic design.

In its letter of May 26, 2007, regarding its application to amend the AP1000 design certification rule (DCP/NRC1912), Westinghouse referred to the criterion of 10 CFR 52.63(a)(1)(vii) and stated that these proposed changes contribute to increased standardization of the certification information.

### 4.4.1 Evaluation

DCD Section 4.4, "Thermal and Hydraulic Design," describes the AP1000 reactor core thermal-hydraulic design to ensure adequate heat removal to prevent fuel damage during normal operation and transients. In Revision 17 of the AP1000 DCD, Westinghouse proposed the following changes to Section 4.4 related to the AP1000 core thermal-hydraulic design: (1) revision of the core bypass flow, (2) change of the maximum rod bow penalty for DNBR calculation from less than 1.5 percent DNBR to less than about 2 percent DNBR, (3) change of the hydraulic loads calculation from the mechanical design flow to the best estimate flow, (4) change of the peak linear power resulting from overpower transient/operator errors from 22.5 kW/ft to less than or equal to 22.45 kW/ft, (5) change of the term "canned motor pump" to "reactor coolant pump," (6) revision of the description of the DMIMS, (7) changes of the values of several parameters in Table 4.4-1, (8) addition of WCAP-15063-P-A, Revision 1, (9) addition of Reference 87 (WCAP-16652-NP), and (10) division of Section 4.4.7, "Combined License Information," into Sections 4.4.7.1 and 4.4.7.2, with revisions to each section.

GDC 10 specifies that the reactor core and the associated coolant, control, and protection systems must be designed with appropriate margin to ensure that SAFDLs are not exceeded

during any condition of normal operation, including the effects of AOOs. In its review of the AP1000 DCD changes related to DCD Section 4.4, the staff used the guidance of SRP Section 4.4, "Thermal and Hydraulic Design," which sets forth the acceptance criteria used by the staff to evaluate the thermal-hydraulic design of the reactor core for compliance with the relevant requirements of GDC 10. The following discusses the evaluation of these changes.

DCD Section 4.4.1.3.1, "Design Basis," identifies the core bypass flow (which is not considered effective for heat removal) as the coolant flow through the thimble tubes and the leakage from the core barrel shroud region into the core. Revision 17 revised the core bypass flow as the coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles. In TR-18, Westinghouse indicated that it made this change to accurately describe the core cavity flow area caused by a change from the core baffle-former design to a welded core shroud design, which eliminates the need for bolts in the high fluence regions immediately adjacent to the reactor core. The revision to DCD Section 4.4.1.3.1 clarifies the regions of the core bypass flow that are not effective for core heat transfer. The shroud core cavity flow is considered to be active flow that is effective for fuel rod cooling. Therefore, the applicant deleted Item E in Section 4.4.4.2.1 from a core bypass flow path to be consistent with core bypass flow design basis detailed in Section 4.4.1.3.1. This revision results in slight changes in the effective core flow area, average velocity along the fuel rods, and average mass velocity, as reported in Tables 4.1-1 and 4.4-1, from 41.5 square feet ($ft^2$), 15.9 feet per second (ft/s), and $2.41 \times 10^6$ lbm/hr-$ft^2$ to 41.8 $ft^2$, 15.8 ft/s, and $2.55 \times 10^6$ lbm/hr-$ft^2$, respectively.

DCD Section 4.4.4.2.1 states that calculations using drawing tolerances in the most conservative direction and accounting for uncertainties in the pressure losses show the core bypass to be no greater than the 5.9-percent design value. The maximum value of 5.9 percent allotted as bypass flow and the thermal design flow of 94.1 percent of the thermal flow rate assumed for the core cooling evaluations, as stated in DCD Sections 4.4.1.3.2 and 4.4.1.3.1, respectively, remain unchanged. Thus, the change in the core bypass flowpaths has insignificant effects on the core cooling calculation and is, therefore, acceptable. The change in the average mass velocity reflects a correction to the coolant temperature used in the calculation of the core average flow rate from the core average temperature to the core inlet temperature. This is acceptable because the calculation with higher water density of the core inlet temperature results in a higher mass velocity with the same volumetric flow rate.

In DCD Revision 17, the applicant added Reference 88 to Sections 4.4.1.2.1 and 4.4.8 (the reference section). Reference 88 addresses the NRC approval of the maximum burnup limit of 62,000 MWD/MTU for WCAP-10444-P-A, WCAP-12610-P-A, WCAP-12488-A, and WCAP-15063-P-A, all of which are referenced in Section 4.4.8. The applicant added Reference 88 for completeness, and the staff finds this addition acceptable.

In DCD Revision 17, the applicant added Reference 82a to Sections 4.4.2.2.1 and 4.4.8. Reference 82a addresses the application of an adjustment factor to the WRB-2M critical heat flux correlation described in Reference 82, which is used in the AP1000 thermal-hydraulic

design calculations. The applicant added Reference 82a for completeness, and the staff finds this addition acceptable.

Section 4.4.2.2.5 states that the maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU and indicated that the maximum rod bow penalty is less than 1.5 percent DNBR. In Revision 17, the applicants changed the maximum rod bow penalty of "less than 1.5 percent DNBR" to "less than about 2 percent DNBR." In TR-18, Westinghouse stated that it made this change for consistency with the basis used for 3-D FAC analysis. Because the change of the maximum penalty from 1.5 percent to about 2 percent DNBR is a more conservative change, and the actual value of the rod bow penalty used in the safety analysis will be calculated based on actual fuel design and assembly burnup of 24,000 MWD/MTU, the staff concludes that the change is acceptable.

Section 4.4.2.6.2 states that the hydraulic loads at normal operating conditions are calculated considering the mechanical design flow and account for the minimum core bypass flow based on manufacturing tolerances. In Revision 17, the applicant changed the mechanical design flow to the best estimate flow. As stated in TR-18, the applicant changed the mechanical design flow to the best estimate flow for the hydraulic calculation to achieve consistency with the current design procedures. Therefore, the staff finds the change to be acceptable.

In DCD Revision 17, Section 4.4.2.7.1, the applicant corrected a typographic error in the Dittus-Boelter correlation for the force convection heat transfer coefficients calculation. However, the staff found that the correction still contains editorial errors. The applicant stated, in response to a telecom of November 17, 2008, that it will correct these errors in a followup version of DCD Section 4.4.2.7.1. This is Confirmatory Item **CI-SRP4.4-1-SRSB-01**.

In Section 4.4.2.11.6, the applicant changed the peak linear power resulting from overpower transient or operator errors from 22.5 kW/ft to less than or equal to 22.45 kW/ft. This change ensures consistency with the reactor design value specified in Tables 4.1-1 and 4.4-1, and is, therefore, acceptable.

In Section 4.4.4.6, the applicant changed the term "canned motor pump" to the more generic term, "reactor coolant pump." This editorial change does not affect the negative slope for the pump head-capacity curve as a generic reactor coolant pump characteristic and is therefore acceptable.

In Section 4.4.6.4, the applicant made changes to the description of the DMIMS. In APP-GW-GLN-019, Revision 2, "AP1000 Standard Combined License Technical Report, Fluid Systems Changes" (TR-103), Westinghouse stated that it made these changes to correct a misinterpretation of the required number of sensors and to correct and delete incorrect information. The requirement for loose parts monitoring system sensors was incorrectly interpreted as requiring four, rather than two, sensors per collection region. This correction results in the removal of the term "redundancy" regarding sensors at each RCS location and instrumentation channel. The applicant made other changes to the descriptions of the DMIMS performance tests method and the technique used to minimize false impact detection. The

applicant made these changes to accurately represent the DMIMS design, and they do not result in a change of the design. The DMIMS design continues to conform to system design aspects of Regulatory Guide (RG) 1.133, Revision 1, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

In Table 4.4-1, the applicant changed the values of several parameters to be consistent with Table 4.1-1. In Footnote j, the applicant revised the theoretical density of the fuel from 95 percent to 95.5 percent for AP1000 (95 percent for others). The applicant modified Footnote f, which is associated with heat transfer, by changing the phrase, "based on 157 fuel assemblies and hot densified fuel length" to "based on densified active fuel length." The applicant rounded the value for AP1000 to 5.72 kW/ft. The changes regarding the theoretical density are acceptable as they are consistent with DCD Section 4.2.3.2.

In Section 4.4.2.11, the applicant added Reference 85, WCAP-15063-P-A, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)." This report presents an NRC-approved fuel rod design methodology. The addition of a reference to this topical report reflects the NRC's approval of the methodology. Therefore, this addition is acceptable.

DCD Revision 16, Section 4.4, proposed to change Reference 9, WCAP-12488-A, from Tier 2* information to Tier 2. As discussed in Section 4.1 of this report, in response to RAI-SRP 4.2-SRSB-01, Westinghouse stated that it would withdraw the request that this Tier 2* item be reassigned to Tier 2 status in Chapter 4. In DCD Revision 17, the applicant restored all Tier 2* information identified in DCD Revision 15, including WCAP-12488-A, to Tier 2*. The staff finds this acceptable.

In Revision 17, the applicant revised Section 4.4.7.1 to state that APP-GW-GLR-059 (Reference 87) completely addressed the COL information requested in this section and the DCD incorporated applicable changes; therefore, no additional work is required by the COL applicant to address the COL information requested in this section. The staff finds this revision to be acceptable, as discussed in Section 4.2 of this report.

In DCD Revision 17, Section 4.4.7.2 states the following:

> Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, and prior to fuel load, the Combined License holder will calculate the design limit DNBR values. The calculations will be completed using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty.

This is a change from DCD Revision 15, Section 4.4.7, "Combined License Information." The applicant has changed the "combined license applicant" who will calculate the design limit DNBR values to the "combined license holder" who will calculate the design limit DNBR values before fuel load. In response to RAI-SRP 4.4-SRSB-02, Westinghouse stated that the design limit DNBR depends on the selection of the actual plant operating instrumentation, as well as on calculation of the instrumentation uncertainties of the operating plant parameters. The applicant further stated the following:

> The actual calculated instrumentation uncertainties will be used, when available, to recalculate the RTDP [revised thermal design procedures] DNBR design limits. If these recalculated DNBR design limits are less than those given in

DCD 4.4.1.1.2, then the DCD values are conservatively high and will not be revised. If the calculated values are higher than the values in the DCD, then the design limits will be revised. Note that since conservatively high values of the instrumentation uncertainties were used in the calculations, we do not expect the design limits to change.

The NRC staff agrees that the actual design limit DNBR can only be calculated after the selection of actual plant operating instrumentation, and therefore the change from "the combined license applicant" to the "combined license holder" who will calculate actual design limit DNBR values is acceptable. It should be noted that in Appendix F, "Combined License Action Items," to the FSER related to certification of the AP1000 standard design (NUREG-1793), FSER Item 4.4-1 reiterated DCD COL Information Item 4.4-2. Therefore, this same change in COL Information Item 4.4-2, as described in DCD Section 4.4.7.2, is applied to FSER Item 4.4-1. Also, in Revision 17 of the DCD, the applicant revised Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," to be consistent with the changes to Section 4.4.7.2 to indicate that COL Information Item 4.4-2 is an action required by the COL holder. This is acceptable to the staff.

### 4.4.2 Conclusion

The staff has reviewed the changes to DCD Section 4.4, as described in DCD Revision 17. Based on the above evaluation and contingent upon an acceptable resolution to the Confirmatory Item **CI-SRP4.4-1-SRSB-01** discussed above, the staff concludes that the core thermal-hydraulic design continues to meet the requirements of GDC 10 and is, therefore, acceptable.

## 4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.2.1 Material Specification

In Revision 16 to the AP1000 DCD, Westinghouse proposed changes to Section 4.5.1 to include austenitic SS Types 304, 304L, 316, and 316L for parts of the control rod drive mechanisms (CRDMs) and control rod drive (CRD) line exposed to reactor coolant, including pressure-boundary components. APP-GW-GL-009 (TR-33), Revision 1, submitted by letter dated May 24, 2007, identified and justified these changes.

In addition, the applicant revised DCD Tier 2, Section 4.5.1, to include the use of a cobalt alloy or qualified substitute to fabricate CRDM latches and links. The applicant also modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs from Aerospace Material Specification (AMS) 5698E and AMS 5699E to AMS 5698 and AMS 5699. APP-GW-GLN-106 (TR-106), Revision 1, submitted by letter dated September 28, 2007, identified and justified the aforementioned changes.

4.5.1.2.2 Evaluation

4.5.1.2.2.2 Changes to Control Rod Drive Mechanism Stainless Steel Materials (TR-33)

GDC 1, “Quality Standards and Records,” requires structures, systems, and components (SSCs) important to safety to be designed, fabricated, erected, and tested to quality standards to ensure a quality product in keeping with the required safety function. GDC 4, “Environmental and Missile Dynamic Effects Design Bases,” requires that SSCs important to safety be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff review of the AP1000 DCD changes related to this section ensures that the materials for the CRD system meet the American Society of Mechanical Engineers (ASME) Code requirements and are compatible with the reactor coolant environment to ensure a quality product commensurate with its importance to safety.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the CRDM and CRD line. The parts of the CRDMs and CRD line exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the control CRDM. The parts of the CRDM exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment. Currently, DCD Section 4.5.1.1 and the corresponding Table 5.2-1 in Section 5.2.3 include SS Types 304LN and 316LN, which have high resistance to sensitization. Therefore, these materials are more resistant to stress-corrosion cracking (SCC) because of their low carbon content. The applicant revised DCD Tier 2, Section 4.5.1, to include austenitic SS Types 304, 304L, 316, and 316L, as discussed in APP-GW-GL-009 (TR-33), Revision 1, submitted by letter dated May 24, 2007. TR-33, which provides the basis for the change, states that the addition of these materials will enhance manufacturing flexibility, reduce costs, and reduce risk relative to material availability. The applicant submitted changes to the AP1000 DCD, as proposed in TR-33, pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that the proposed changes contribute to increased standardization of the certification information.

SS Types 304 and 316 (higher carbon content) are less resistant to sensitization due to heat treatment or welding. In addition, NRC Information Notice 2006-27, “Circumferential Cracking in Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors,” and numerous

requests for relief from the ASME Code concerning repairs to leaking CRDM canopy seal welds discuss emerging issues involving SS Types 304 and 316. These instances of SCC are occurring in stagnant or deadend pressurized-water reactor (PWR) coolant environments. Since Type 304 and 316 SS materials are more susceptible to intergranular stress-corrosion cracking (IGSCC) and transgranular stress-corrosion cracking than the low-carbon SS Types 304L, 304LN, 316L, and 316LN, the use of Types 304 and 316 materials may affect the integrity of the CRD components (including the RCPB portions of the latch housing and rod travel housing). Specifically, the use of these materials can affect the structural integrity of CRD components that are subjected to stagnant water (trapped oxygen), dead legs, or areas prone to increased levels of oxygen.

Therefore, the NRC staff requested, in RAI–SRP4.5.1-CIB1-01, that Westinghouse delete the proposed addition of SS Types 304 and 316 from the AP1000 DCD or provide further justification addressing the acceptability of the proposed addition of these materials.

In a letter dated May 30, 2008, Westinghouse provided further justification concerning the acceptability of the proposed addition of SS Types 304 and 316 in components subjected to stagnant water. Westinghouse stated that all austenitic SSs are procured in the solution annealed condition and that controls are established in Section 5.2.3.4 of the AP1000 DCD to

avoid sensitization due to heat treatment or welding and prevent susceptibility to intergranular attack as addressed by RG 1.44, "Control of the Use of Sensitized Stainless Steel," issued May 1973. In addition, cleaning procedures and contamination preventative measures are implemented to prevent the presence of detrimental impurities that could contribute to SCC. The AP1000 CRDM design uses only one canopy seal (CRDM to the rod travel housing) in lieu of the three canopy seals in current designs. In addition, this canopy seal weld was redesigned with a larger radius and thicker wall to reduce stress levels, and the design includes a vent and drainpath for the canopy seal volume to prevent fully stagnant conditions. The number of occurrences of cracking in SS components exposed to PWR operating environment is small considering the number of SS components used in PWR applications. Most of these failures have occurred as a result of adverse conditions (trapped oxygen in stagnant water) and high stresses.

The NRC staff agrees that using the guidance in RG 1.44 and proper cleaning techniques limits the amount of sensitization of Types 304 and 316 SS, which in turn reduces the susceptibility to SCC. In addition, the redesign of the CRDM reduces the stresses and adverse environment (vent and drainlines to minimize the presence of an oxygenated environment) that is a major contributor to the susceptibility of the SS to SCC. Therefore, the staff finds that Types 304 and 316 SS can be used, in addition to Types 304L, 304LN, 316L, and 316LN (which are less susceptible to sensitization) previously approved by the staff in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004, because the Westinghouse design changes will reduce the number of canopy seal welds to one per CRDM, reduce the stresses in the canopy seal weld, eliminate the presence of an oxygenated environment using vent and drainlines, and follow the guidance in RG 1.44. In a letter dated May 13, 2009, Westinghouse provided a marked-up copy of the DCD which included the design changes and the use of RG 1.44. However, these design changes (reduced number of welds, reduced stresses, and use of vent and drainlines) and the use of RG 1.44 are critical in preventing the occurrence of SCC in these components. In the Chapter 4 SER with OIs the staff concluded that Westinghouse should add these design changes and the use of RG 1.44 for the CRDM components, including the canopy seal welds, to the DCD. It should be noted that DCD Section 4.5.1.2 applies the controls on preventing SCC, including the guidance in RG 1.44 for the austenitic SS pressure-housing components of the CRDM. The canopy seal welds may not be considered a pressure-housing component since they only provide a leakage barrier. The NRC staff identified this issue as **Open Item OI-SRP4.5.1-CIB1-01**.

The staff identifies **Confirmatory Item CI-SRP4.5.1-CIB1-01** to confirm that the DCD is revised as stated by the applicant in the marked-up copy of the FSAR provided in the Westinghouse letter dated May 13, 2009 (ADAMS Accession No. ML091350197). The identified added materials and the additional information on grades, types, or classes meet the requirements of GDC 1, GDC 4, and the ASME Code, Sections II and III, in accordance with the guidance of SRP Section 4.5.1. Therefore, the addition of the identified materials and the additional information on grades, types, or classes presented in TR-33, Revision 1, are acceptable.

The NRC staff reviewed the proposed material changes as they relate to Revision 16 to the AP1000 DCD. Westinghouse has incorporated the proposed changes, as identified in TR-33, Revision 1, into the AP1000 DCD and TR-134, Revision 0. Therefore, the staff finds that the DCD changes that address material changes, as proposed by Westinghouse in TR-33, Revision 1, are acceptable.

Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD and are acceptable.

4.5.1.2.2.3 Changes to Control Rod Drive Mechanism Latches, Links, and Springs (TR-106)

GDC 26, "Quality Standards and Records," requires that one of the reactivity control systems use control rods, preferably with a positive means for inserting the rods, and be capable of reliably controlling reactivity changes for assurance that fuel design limits are not exceeded under conditions of normal operation, including AOOs.

The staff reviewed the AP1000 DCD changes related to this section to ensure that the materials for the CRDMs will perform adequately throughout the design life of the plant. The CRDM latches, links, and springs are not RCPB components and are, therefore, not required to be fabricated from materials meeting ASME Code, Section III. DCD Tier 2, Section 4.5.1, describes the materials used to fabricate CRDM components.

The applicant revised DCD Tier 2, Section 4.5.1, to include a cobalt alloy or qualified substitute to fabricate CRDM latches and links. Previously, the applicant intended to fabricate latches and links from SS hard faced with stellite. The applicant also deleted its requirement that all cobalt alloy material be ordered in the solution-treated, cold-worked condition. In addition, Westinghouse modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs. Westinghouse also changed the specification from AMS 5698E and AMS 5699E to AMS 5698 and AMS 5699.

In TR-106, the applicant stated that the previous design of the CRDM in Section 4.5 of Revision 15 of the AP1000 DCD was based on a design utilized in past Westinghouse plants to achieve a design life of 3 million steps. The applicant modified the design to meet a design life of 8 million steps, which required a material change for CRDM latches and links. The redesigned latches have also been modified to a double tooth design constructed of solid stellite in lieu of the previous design which utilized a single tooth design and was constructed of SS hard faced with stellite.

In response to RAI-TR106-CIB1-02, dated October 5, 2007, the applicant provided additional information regarding the use of solid stellite components in PWR CRDMs. The applicant stated that cast single-tooth latches constructed from solid stellite have been used extensively in the Combustion Engineering (CE) fleet and all CE units in Korea. CE has used single-tooth cast stellite to construct latches for the past 30 years in its units. Electricite de France EDF has used double-tooth hardfaced latches. The AP1000 cast latches will be based on AMS 5387.

AMS 5387 (similar to Stellite 6) is a cast cobalt-based alloy that provides adequate resistance to wear and corrosion in the reactor coolant environment. The staff is unaware of any failures or degradation issues associated with the use of cast stellite in CRDM components. Based on the corrosion resistance and mechanical properties of this material, as well as the favorable operating experience in currently operating plants, the staff finds the applicant's use of AMS 5387 to fabricate latches and links acceptable. The applicant's elimination of required solution heat treatment and cold-working for cobalt alloys used in the CRDMs is acceptable because the AMS 5387 used for latches and links requires that components fabricated to this specification are used in the as-cast condition. It is the staff's understanding that cobalt alloy link pins will still be delivered in the solution treated, cold-worked condition because these components are not cast.

In addition to the modifications to DCD Tier 2, Section 4.5.1.3, discussed above, the applicant changed the specification for CRDM springs (Alloy 750) from AMS 5698E or AMS 5698 to AMS 5698 or AMS 5699. This modification does not change the materials that the staff previously approved. The modification to specifications listed for CRDM springs allows current versions of AMS 5698 and AMS 5699 to be used and is, therefore, acceptable.

The staff finds that the proposed modifications are acceptable and meet the requirements of GDC 26 and the acceptance criteria of SRP Section 4.5.1. In addition, the NRC staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. Revision 16 of the DCD incorporates the proposed changes identified in TR-106. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

4.5.1.2.3 Conclusion

Based on the above evaluation and upon resolution of **Confirmatory Item CI-SRP4.5.1-CIB1-01**, the staff finds that the revisions proposed by Westinghouse to DCD Section 4.5.1 meet the requirements of 10 CFR 50.55a, "Codes and Standards," and ASME Code, Section III, and are, therefore, acceptable. Revision 16 of the AP1000 DCD and TR-134, Revision 0, incorporate the proposed changes. Furthermore, the staff finds that the TR-33, Revision 1, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 design certification. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

In addition, based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to AP1000 DCD, Section 4.5.1, meet the requirements of GDC 26 and, therefore, are acceptable. Revision 16 of AP1000 DCD incorporates the proposed changes, as identified in TR-106, Revision 1. Furthermore, the staff finds that the TR-106, Revision 0, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 design certification. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

### 4.5.2 Reactor Internal and Core Support Materials

In Revision 16 of the AP1000 DCD, Westinghouse proposed design changes related to the material specifications for AP1000 reactor internal components. This resulted in changes to Section 4.5.2.1. By letter dated July 31, 2007, Westinghouse submitted TR-31 (APP-GW-GLN-015), Revision 1, which included WCAP-16624-P, Revision 1, "Reactor Internals Materials Changes for the AP1000 Plant," to provide the technical justification for the proposed design changes related to the material specifications for reactor internal components.

In TR-31, Westinghouse proposed the following changes to the AP1000 reactor internal and core support material specifications: (1) the addition of American Iron and Steel Institute Types 304, 304H, and 304L SSs to the material specifications for reactor internal core structure components, (2) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS, (3) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs, (4) the addition of Stellite 6 to the hardfacing materials for the

radial keys, clevis inserts, and alignment pins, and (5) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS.

Other related changes identified in TR-31 include (1) an additional specification in DCD Section 4.5.2.1 for qualification of welding procedures in accordance with the staff guidance in RG 1.44, (2) revisions to the language in DCD Section 4.5.2.1 pertaining to the susceptibility of reactor internal components to irradiation-assisted stress-corrosion cracking (IASCC) and void swelling, and (3) an additional statement in DCD Section 4.5.2.1 indicating that Alloy 600 would not be used in the AP1000 reactor internal components.

Revision 16 to DCD Section 4.5.2.1 implements all of the above changes proposed in TR-31 as follows:

> The major core support structure material for the reactor internals is SA-182, SA-479, or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel and for the clevis insert-to-vessel bolts either Unified Numbering System for Metals and Alloy UNS N07718 or N07750. Remaining internals parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, alignment pens (Stellite™ 6 or 156 or low cobalt hardfaces); dowel pins (Type 316); hold down spring (Type 403 stainless steel (modified)); clevis inserts (UNS N06690); and irradiation specimen springs (UNS N07750). Core support structure and threaded structural fastener materials are specified in the ASME Code, Section III, Appendix I as supplemented by Code Cases N-60 and N-4. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in Subsection 4.5.1.3.
>
> The use of cast austenitic stainless steel (CASS) is minimized in the AP1000 reactor internals. If used, CASS will be limited in carbon (low carbon grade: L grade) and ferrite contents and will be evaluated in terms of thermal aging effects.
>
> The estimated peak neutron fluence for the AP1000 reactor internals has been considered in the design. Susceptibility to irradiation-assisted stress corrosion cracking or void swelling in reactor internals identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs. Ni-Cr-Fe Alloy 600 is not used in the AP1000 reactor internals.

4.5.2.1 Evaluation

GDC 1 and 10 CFR 50.55a 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. The staff's review of the AP1000 DCD changes related to this section covers material, components design, fabrication, and inspection to ensure structural integrity in compliance with 10 CFR 50.55a and GDC 1.

TR-31 proposes changes to the material specifications for the AP1000 reactor internal components. DCD Tier 2, Section 4.5.2.1, identifies the reactor internal material specifications. In Revision 15 of the DCD, the major materials of construction for the reactor internal core support structure components, excluding threaded structural fasteners, are Type 304LN SS. Revision 15 of the DCD specifies strain-hardened Type 316 SS for threaded structural fasteners. Other reactor internal materials specified in DCD Revision 15, Section 4.5.2.1, include Stellite 156 or low cobalt hardfacing on the radial keys, clevis inserts, and alignment pins; Type 316 SS for the dowel pins; modified Type 403 SS for the holddown spring; and Type 302 SS for irradiation specimen springs.

The proposed material specification changes primarily involve the addition of Types 304, 304H, and 304L to the 304-series SSs specified for the reactor internal core support structure components. Other material changes include (1) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS, (2) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs, (3) the addition of Stellite 6 to the above hardfacing materials for the radial keys, clevis inserts, and alignment pins, and (4) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS. DCD Revision 16, Section 4.5.2.1, implements the changes to these material specifications.

Section 3 of TR-31 provides a technical description and justification for the proposed material specification changes. The addition of Types 304 and 304H SSs is significant because these SSs have a higher carbon content than Type 304LN. Types 304 and 304H have a maximum carbon content of 0.08 percent (by weight) and 0.10 percent, respectively, whereas Type 304LN is a low-carbon SS with a maximum carbon content of 0.03 percent. Type 304H, in particular, is notable as a high-carbon SS with a specified minimum carbon content of 0.04 percent. None of the other SS grades specified in DCD Revision 16, Section 4.5.2.1 (Types 304, 304L, 304LN), has a specified minimum carbon content. The 304H grade was developed to ensure better resistance to high-temperature creep by maintaining at least 0.04-percent carbon. The higher allowable carbon content in Types 304 and 304H austenitic materials can potentially result in a significant degree of sensitization to intergranular corrosion and IGSCC at elevated temperatures. In the temperature range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F), chromium carbides, $(Fe,Cr)_{23}C_6$, are insoluble and precipitate at grain boundaries through the diffusion of carbon. Precipitation of chromium carbides at grain boundaries results in the depletion of chromium in the surrounding matrix alloy immediately adjacent to the grain boundaries. The chromium-depleted alloy at the grain boundaries is much less corrosion resistant than the rest of the bulk alloy (i.e., away from the grain boundaries). The galvanic coupling of chromium-depleted alloy at the grain boundaries with bulk alloy in the passive state (due to undepleted chromium) can result in significant intergranular corrosion. High-carbon SSs subjected to temperatures in the range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F) for a sufficient time to allow for the formation of chromium carbides at the grain boundaries are sensitized to intergranular corrosion and IGSCC. When temperatures exceed 815.6 °C (1,500 °F), the chromium carbides are soluble, and below 426.7 °C (800 °F) the diffusion rate of carbon is not sufficient to permit the formation of chromium carbides. Therefore, it is specifically in the intermediate temperature range that sensitization is a significant concern for high-carbon SSs. Welding is known to produce sensitization in weld heat affected zones (HAZs), located on either side and, at times, somewhat removed from the actual weld bead. It is in these HAZs that the welding process can produce temperatures in this intermediate range for a sufficient time to allow for carbon diffusion and the formation of chromium carbides and chromium-depleted zones at the grain boundaries. The carbon content of low-carbon SS grades (i.e., SSs with a specified maximum carbon content of 0.03 percent) is not high enough for sensitization to be a

significant concern because a sufficient quantity of carbon does not exist to cause significant chromium depletion at grain boundaries within a practical timeframe for the welding process.

High-carbon austenitic SSs in nuclear reactor structural components that have become locally sensitized are potentially susceptible to IGSCC at these sensitized locations. While significant intergranular corrosion is generally not an issue for unstressed parts, the presence of tensile stresses has been known to produce IGSCC in components where sensitization has occurred. For a given high-carbon SS component, the effects of sensitization can be minimized by controlling weld parameters, such as heat input and cooling rate. RG 1.44 describes acceptable methods for controlling the processing of SSs to avoid sensitization that could lead to IGSCC. The RG specifies that, for a given material composition, welding practices should be controlled to avoid excessive sensitization of base metal HAZs adjacent to welded joints. The RG also specifies an intergranular corrosion test for the qualification of welding procedures to be used for welding SSs having a carbon content of greater than 0.03 percent. DCD Revision 16, Section 4.5.2.1, includes language requiring that welding procedures be qualified in accordance with RG 1.44.

DCD Revision 16, Section 4.5.2.1, also added Type 304L, a low-carbon grade of SS (0.03-percent maximum) similar to Type 304LN, to the list of permissible SS grades for reactor internal core support structure components. Type 304L has a lower nitrogen content (0.10-percent maximum) than Type 304LN (0.10 percent to 0.16 percent). Because of its lower nitrogen content, Type 304L is not as strong as Type 304LN and high-carbon SS Types 304 and 304H. Relative to Type 304L, the higher nitrogen content in Type 304LN results in a strengthened material that is both resistant to sensitization (due to the low carbon content) and possesses the higher tensile and yield strength properties of Type 304 and 304H materials. Types 304LN, 304, and 304H all possess an ASME Code minimum tensile strength of 517.1 megapascals (MPa) (75 kilogram per square inch (ksi)) and a minimum yield strength of 206.8 MPa (30 ksi). Type 304L possesses an ASME Code minimum tensile strength of 475.7 MPa (69 ksi) and a minimum yield strength of 172.4 MPa (25 ksi). Type 304L SS may be used for reactor internal core support structural applications where its lower strength properties are permitted, in accordance with ASME Code, Section III.

According to Westinghouse, the primary justification for adding Types 304, 304H, and 304L SSs to DCD Section 4.5.2.1 is the application of these materials in currently operating Westinghouse plants. Westinghouse indicated that SCC has not been experienced in reactor internal core support structure components fabricated with any of these three materials. The use of Type 304L, the lower strength grade, is unconditionally approved for currently operating Westinghouse reactors where its lower strength properties are permitted. Westinghouse stated that it previously implemented a change from Type 304 to Type 304H for certain reactor internal components in operating plants as the reactor design evolved over time; the applicant provided a list of plants to demonstrate the extensive application of Type 304H SS in reactor internal core support structure components for these later designs. Westinghouse noted that, for many of the later plants, carbon content in several Type 304 components was limited to a specified range of 0.04 percent to 0.08 percent. This was stated as being the equivalent of a Type 304H SS with a more restrictive 0.08-percent upper limit on carbon content. Westinghouse stated that, to be consistent with these later plants, the carbon content in Type 304H SSs should be limited to a maximum of 0.08 percent for reactor internal core support structure components in the AP1000 plant. The Westinghouse Utilities Requirements Document requires the use of RG 1.44. Accordingly, all fabricators will be required to establish maximum heat inputs for each welding process with respect to the maximum carbon content for each SS type.

Overall, the staff found that the applicant provided sound justification for the addition of Types 304, 304H, and 304L SSs to the material specifications in DCD Revision 16, Section 4.5.2.1. In particular, the staff noted that the currently operating Westinghouse plants have not experienced problems with IGSCC in reactor internal core support structure components fabricated from these materials. Furthermore, the staff noted that DCD Revision 16, Section 4.5.2.1, specifies that fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44.

In an RAI dated March 11, 2007, the staff requested that Westinghouse clarify or elaborate on several issues. In RAI-TR31-001 Question 1, Part a, the staff noted an inconsistency in the language in Section 1.2, "Introduction and Brief Description of Change," of TR-31. Specifically, the staff noted that Section 1.2 of TR-31 states that DCD Revision 15, Section 4.5.2.1, currently specifies Type 304LN SS for reactor internal core support structure components. However, Section 1.2 also states that reactor internal components were designed using Types 304, 304H, and 304L SSs. Therefore, the staff requested that Westinghouse clarify whether it changed the AP1000 design for the reactor internal components after the issuance of DCD Revision 15 to include these additional SS grades. In its response to RAI-TR31-001 Question 1, Part a, Westinghouse indicated that it did change the design of the AP1000 reactor internal components after the issuance of Revision 15 of the DCD to include Types 304, 304H, and 304L SSs in addition to Type 304LN SS as potential materials of construction for AP1000 reactor internal components. The staff found that this response resolved RAI-TR31-001 Question 1, Part a because Westinghouse adequately clarified the statement made in Section 1.2 of TR-31 pertaining to reactor internal component design.

In RAI-TR3-001 Question 1, Part b, the staff requested that Westinghouse list the materials of construction for each reactor internal component based on the newly proposed reactor internal material specifications identified in TR-31. In its response to RAI-TR31-001 Question 1, Part b, Westinghouse provided a table depicting the materials of construction for each of the reactor internal components. The staff evaluated this table and determined that the predominate materials of construction for all major core support structures, excluding bolting, are essentially limited to Types 304, 304H, 304L, and 304LN SSs. Furthermore, the staff confirmed the exceptions to the use of the 304-series SSs identified previously, specifically the use of nickel-based Alloy 690 for the clevis inserts, nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts, nickel-based alloy 750 for the irradiation specimen springs, and the addition of Stellite 6 hardfacing for wear surfaces on the radial keys, clevis inserts, and alignment pins. These exceptions were verified to be applicable only to these specific components. The high yield and tensile strength properties and corrosion resistance of nickel-based alloys justify their use for these specific components. These alloys are all acceptable in accordance with ASME Code, Section III. Therefore, the staff identified no safety-related issue associated with their use in these instances. DCD Revision 15, Section 4.5.1.3, previously addressed the qualification of Stellite 6 hardfacing for use in RCSs. The staff found that Westinghouse adequately addressed RAI-TR31-001 Question 1, Part b because it provided a comprehensive list of material specifications for each reactor internal component that is consistent with DCD Revision 16, Section 4.5.2.1.

In RAI-TR31-002 Question 2, the staff requested that Westinghouse elaborate further on how it addressed the susceptibility of Type 304 and 304H SSs to various forms of corrosion and SCC, where welding on components fabricated using these materials could result in sensitization due to chromium depletion at grain boundaries. In its response to RAI-TR31-002 Question 2, Westinghouse reiterated its assertion that these SS grades have been used extensively for

reactor internal components for currently operating Westinghouse plants. Westinghouse further stated that the available technical data on environmental degradation applicable to the currently operating Westinghouse plants are also applicable to the design of AP1000 reactor internal components. In addition, these materials have been assessed for a reactor internal component design life of 60 years, with respect to known mechanisms of IASCC and void swelling. DCD Revision 16, Section 4.5.2.1, states that internal material reliability programs are addressing the susceptibility to IASCC and void swelling in reactor internal components identified in the currently operating Westinghouse fleet, and the selection of materials for AP1000 reactor internal components considers information developed by these programs. TR-12, which was provided in WCAP-16620-P, Revision 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant," dated July 31, 2006, addresses the evaluation of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling over the 60-year design life and the application of the IASCC and void swelling criteria established by the above material reliability programs. The staff found that Westinghouse's response to RAI-TR31-002 Question 2 was acceptable because it demonstrated that the applicant had adequately addressed the susceptibility of reactor internal components fabricated from Types 304 and 304H SSs to known corrosion and SCC phenomena for reactor internal components.

Based on the above discussion, the staff determined that Westinghouse provided an appropriate technical justification for the reactor internal material specification changes proposed in TR-31 because these proposed changes meet the requirements 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff determined that DCD Revision 16, Section 4.5.2.1, fully represents these material changes. Therefore, the staff determined that the DCD changes proposed by Westinghouse in TR-31 are acceptable.

The NRC staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. Revision 16 of the DCD incorporates the proposed changes identified in TR-31. Accordingly, these changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD.

#### 4.5.2.2 Conclusion

The staff finds that the changes to the material specifications for the reactor internal components proposed in TR-31 are technically acceptable because these changes meet the requirements of 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff finds that the TR-31 conclusions regarding design changes related to the material specifications for AP1000 reactor internal components are generic and are expected to apply to all COL applications referencing the AP1000 design certification. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

### 4.5.3 Changes to In-Core Instrument Guide Tubes

In DCD Revision 17, the applicant proposed a change related to the replacement of in-core instrument guide tubes with QuickLoc assemblies, as well as several editorial modifications to Tier 1, Section 2.1.3, "Reactor System."

4.5.3.1 Evaluation

The technical change involved the replacement of in-core instrument guide tubes with QuickLoc assemblies, as supported by APP-GW-GLE-016. Also, this change included specifying the incore instrumentation QuickLoc assemblies as ASME Code Section III Classification and Seismic Category I classification in Tier 1 Table 2.1.3-1, and the QuickLoc assemblies as the pressure boundary components in Table 2.1.3-2. The staff reviewed APP-GW-GLE-016 and issued RAI-SRP15.4.8-SRSB-01 related to the rod ejection analysis in DCD Section 15.4.8. The staff determined the QuickLoc-related changes, as described in DCD Revision 17, Tier 1, Section 2.1.3, to be acceptable.

The first editorial modification is found in Table 2.1.3-2, which includes Design Commitment 2.a, and stipulates an ITAAC for the reactor upper internals rod guide arrangement. The applicant made an editorial change to refer to the correct figure describing the reactor upper internals rod guide arrangement. Specifically, Revision 15 listed Figure 2.3.1-1 instead of the correct figure (Figure 2.1.3-1).

The applicant made additional editorial modifications throughout Section 2.1.3 to correct the spelling of "rod cluster control assemblies" (from "rod control cluster assemblies") and to clarify that the fuel assemblies are located in the containment location only after fuel loading (located in auxiliary building prior to fuel loading). The staff finds these changes to be acceptable.

4.5.3.2 Conclusion

Based on the above evaluation, the staff concludes that the changes in Revision 17 of DCD Tier 1, Section 2.1.3, are acceptable.

## 4.6 Functional Design of Reactivity Control Systems

The reactivity control systems for the AP1000 facility are the control rod drive system (CRDS), the reactor trip system, and the passive core cooling system which can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents.

### 4.6.2 Evaluation

In Revision 17 of DCD Section 4.6, Westinghouse proposed the following revisions to Section 4.6.1, "Information for Control Rod Drive System": (1) a clarification that DCD Figure 4.2-8 provides the configuration of the driveline, including the CRDM, not the layout of the CRDS, (2) the deletion of the statement that the CRDM outer shroud is an integral portion of the head lifting system, and (3) the deletion of the "conduits for the in-core instrumentation" from the components located among the CRDM and supported by the integral head package.

The staff reviewed these revisions and concluded that they are editorial in nature. The applicant made the latter two revisions for the purpose of accuracy and consistency with the DCD Section 3.9.4.1.1 modifications to the integrated head package and redesign of the in-core instrumentation. These changes do not alter the functional design of the reactivity control systems and are therefore acceptable.

### 4.6.2 Conclusion

The NRC staff concludes that revisions to AP1000 DCD Section 4.6 are acceptable because the changes are editorial and do not alter the functional design of the reactivity control systems.