# 4. REACTOR

## 4.1 Introduction

AP1000 Design Control Document (DCD) Tier 2, Chapter 4, “Reactor,” describes the mechanical components of the AP1000 reactor and reactor core, including the fuel system design (fuel rods and fuel assemblies), the nuclear design, and the thermal-hydraulic design. DCD Tier 2, Section 4.1.1, “Principal Design Requirements,” specifies the principal design criteria with which the mechanical design, the physical arrangement of the reactor components, and the capabilities of reactor control, protection, and emergency cooling systems (when applicable) must comply.

DCD Tier 2, Chapter 4, also identifies certain areas as Tier 2* information, departures from which require prior U.S. Nuclear Regulatory Commission (NRC) staff approval. DCD Introduction, Section 3.5, “Plant-Specific Changes to Designated Information in the Tier 2, Information,” provides a definition of and the criteria governing Tier 2* information.

The following sections in DCD Tier 2, Chapter 4, include Tier 2* information:

- 4.1 Westinghouse Commercial Atomic Power (report) (WCAP)-12488-A, “Westinghouse Fuel Criteria Evaluation Process,” issued in October 1994

- 4.1.1 Principal Design Requirements

- 4.3.1.1 Maximum Fuel Road Average Burnup of 62,000 megawatt-days per metric ton of uranium (MWD/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 Rod Cluster Control Assemblies

## 4.2 Fuel System Design

The staff based its review of the AP1000 fuel design on the information contained in the DCD and the topical reports referenced by the applicant. 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,” (also referred to as the SRP), Section 4.2, “Fuel System Design,” which prescribes acceptance criteria to ensure that certain requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” are met. In particular, the AP1000 fuel design must meet the following general design criteria (GDC) found in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”:

- 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 anticipated operational occurrences (AOOs).

- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.

- The number of fuel rod failures is not underestimated for postulated accidents.

- Coolability is always maintained.

The term “will not be damaged,” used above, means that the fuel rods will not fail, the fuel system’s dimensions will remain within operational tolerances, and their 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 called specified acceptable fuel design limits (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 its 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.” As discussed in Section 15.3 of this report, the AP1000 design complies with the dose consequences criteria in 10 CFR 50.34(a)(1), given the site parameters postulated for the design. Therefore, the AP1000 design also meets the requirements of 10 CFR Part 100.

“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, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” establishes specific requirements for the performance of the emergency core cooling system following postulated loss-of-coolant accidents (LOCAs). As set forth in Section 15.2.6.5 of this report, the AP1000 design complies with the requirements of 10 CFR 50.46.

### 4.2.1 Fuel Assembly Description

Each of the AP1000 reactor fuel assemblies consists of 264 fuel rods in a 17x17 square array. The assemblies are very similar to the 17x17 robust fuel assemblies (RFAs) discussed in Westinghouse letters dated October 13, 1998, and March 25, 1998, and the 17x17 XL RFAs discussed in a Westinghouse letter dated June 23, 1998, which evolved from NRC-approved Westinghouse fuel designs, such as VANTAGE 5, VANTAGE 5 Hybrid, and VANTAGE+. All of these designs have substantial design and operating experience associated with them. The 17x17 RFAs have an active fuel length of 3.7 m (12 ft) and three intermediate flow mixing (IFM) grids in the top mixing vane grid spans. The 17x17 XL RFAs have an active fuel length of

4.3 m (14 ft) with no IFM grids. The AP1000 fuel assemblies are the same as the 17x17 XL RFAs, except that they have four IFM grids in the top mixing vane grid spans.

Each AP1000 fuel assembly consists of a total of ten structural grids, including six low-pressure-drop intermediate grids and four IFM grids. Each fuel assembly has a reconstitutable top nozzle and a debris filter bottom nozzle (DFBN) to minimize the potential of fuel damage due to debris in the reactor coolant. The AP1000 fuel design also includes a protective grid adjacent to the DFBN for enhanced debris resistance.

Some spaces of the 17x17 fuel rod array contain guide tubes in place of fuel. These guide tubes house instrumentation and accommodate either rod cluster control assemblies (RCCAs) or gray rod cluster assemblies (GRCAs), both of which provide in-core reactivity control, as discussed below.

### 4.2.2 Fuel Rod Description

The AP1000 fuel rods consist of cylindrical, ceramic pellets of slightly enriched uranium dioxide ($UO_2$). These pellets are contained in cold-worked and stress-relieved ZIRLO tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. ZIRLO is an advanced zirconium-based alloy. The $UO_2$ pellets are slightly dished to better accommodate thermal expansion and fuel swelling, and to increase the void volume for fission product release. The void volume will also accommodate the differential thermal expansion between the clad and the fuel as the pellet density increases in response to irradiation.

The AP1000 fuel rod is designed with two plenums (upper and lower) to accommodate fission gas release. A holddown spring keeps the upper plenum in place, while a standoff assembly holds the lower plenum in position. A stainless steel compression spring, located at the top of the fuel pellet column, restrains the column in its proper position during shipping and handling. The solid bottom end plug has an internal grip feature and tapered end to facilitate fuel rod loading during fuel assembly fabrication and reconstitution. The end plug extends through the bottom grid. This precludes any breach in the fuel rod pressure boundary as a result of clad-fretting wear, which is induced by debris trapped at the bottom of the grid location.

The fuel rods are internally pressurized with helium during fabrication. This internal pressurization minimizes clad stresses from differential pressure and prevents clad flattening under reactor coolant operating pressures.

The AP1000 fuel rod design may also include axial blankets consisting of fuel pellets of reduced enrichments at each end of the fuel rod pellet stack. Axial blankets help to reduce axial neutron leakage and enhance fuel utilization. The presence of these axial blankets will not impact the operation of the AP1000 source-range neutron detectors because the expected reduction in neutron flux is limited to the top and bottom 20.3 cm (0.67 ft) of the core, while the source-range detectors are typically located 91.4 cm (3 ft) from the bottom of the core.

The AP1000 design also includes a second type of fuel rod, which uses an integral fuel burnable absorber containing less than a 0.03-mm (0.001-in.)-thickness boride coating on the surface of the fuel pellets. The use of these integral fuel burnable absorber rods within individual fuel assemblies will vary, depending on the specific application.

### 4.2.3 Burnable Absorber Assembly Description

Discrete burnable absorber rods, inserted into selected thimbles within the fuel assemblies, reduce the beginning-of-life moderator temperature coefficient (MTC). The burnable absorber rods consist of pellets of alumina-boron carbide material contained within zirconium alloy tubes. The tubes are plugged, pressurized with helium, and seal-welded at each end to encapsulate the stack of absorber material. The burnable absorber rods in each fuel assembly are grouped and attached together, at the top end of the rods, to a holddown assembly by a flat perforated retaining plate. This forms the burnable absorber assembly. The burnable absorber assemblies are held down and restrained against vertical motion through a spring pack, which is attached to the retaining plate. The upper core plate compresses the spring pack when the reactor upper internals assembly is lowered into the reactor.

### 4.2.4 Rod Cluster Control Assembly/Gray Rod Cluster Assembly Description

The AP1000 reactivity control design has two types of rod control assemblies known as RCCAs and GRCAs. Both consist of neutron-absorbing rods fastened at the top end to a common spider assembly. The various components of the spider assembly are made of 304- and 308-type stainless steel. The assembly retainer is made of 17-4 PH material, and the impact springs are made of a nickel-chromium-iron (Ni-Cr-Fe) alloy, known as Alloy 718.

The AP1000 reactor uses 53 RCCAs and 16 GRCAs. The RCCA absorber material is a very high thermal neutron absorber silver-indium-cadmium alloy, with additional resonance absorption to enhance rod worth. Bullet-shaped tips are used as plugs at the bottom of the rods to reduce hydraulic drag during reactor trip and to help guide the rods smoothly into the dashpot of the fuel assembly.

Typically, the GRCAs are used in load-follow maneuvering. These assemblies provide a mechanical shim reactivity mechanism (versus a chemical shim, which is achieved by means of changing the concentration of soluble boron) in the reactor coolant. Each GRCA has 24 rodlets fastened at the top end to a common hub or spider. Of the 24 rodlets, 20 are made of stainless steel, while the remaining 4 contain the same silver-indium-cadmium alloy absorber material as is used in the RCCAs. The mechanical design of the GRCAs and the gray rod drive mechanisms, as well as the interface with the fuel assemblies and guide thimbles, is identical to the RCCA design.

### 4.2.5 Design Basis

The applicant established the AP1000 fuel rod and fuel assembly design bases to satisfy the general performance and safety criteria presented in SRP Section 4.2, “Fuel System Design”. The NRC-approved WCAP-10125-P-A, “Extended Burnup Evaluation of Westinghouse Fuel,” issued in December 1985, describes the fuel rod burnup limit, design criteria, methods, and evaluation. In addition, WCAP-12488-A describes the design bases and acceptance limits used by the applicant to analyze the AP1000 fuel rods and assemblies. WCAP-12488-A, which is categorized as Tier 2* information, specifies a set of fuel design criteria which must be satisfied by new fuel designs. Any departure from the fuel design criteria specified in WCAP-12488 will require NRC staff approval prior to its implementation.

Fuel integrity is ensured by design limits imposed on various stresses and deformations resulting from nonoperational loads (i.e., shipping), normal loads (as defined for Westinghouse Condition I and Condition II, which are normal operation and operational transients and events of moderate frequency, respectively), and abnormal loads (as defined for Westinghouse Condition III and Condition IV, which are infrequent incidents and limiting faults, respectively). The overall fuel rod and fuel assembly analysis, including analysis of the performance of the limiting rod with appropriate consideration for uncertainties, is evaluated to ensure that the limits specified by the design bases are not exceeded. Moreover, a combined license (COL) applicant or holder will evaluate future changes to the in-core components (including control rods, burnable absorber rods, and neutron source rods) using the criteria defined in WCAP-12488-A. The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, is used as a general guide in the structural design of these components.

### 4.2.6 Design Evaluation

DCD Tier 2, Chapter 4, and associated topical reports (including WCAP-12488-A) present a variety of methods to demonstrate that the AP1000 fuel rods, fuel assemblies, and control assemblies meet the established design criteria. These methods include operating experience, prototype testing, and analytical predictions.

4.2.6.1 Fuel Rod Performance Evaluation

The applicant analyzed the fuel rod performance during steady-state operations in terms of the various design limits for stress, strain, vibration and wear, creep collapse, and strain fatigue. The applicant performed most of the analyses using the PAD fuel performance code described in WCAP-15063-P-A, “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),” Revision 1, issued in July 2000, and WCAP-10851-P-A, “Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations,” issued in August 1988.

SRP Section 4.2 states that stress limits should be obtained using methods that are consistent with the ASME Code, Section III and a strain limit less than 1 percent. Thermal expansion of the fuel pellets, fission gas release, and reactor coolant pressure affect cladding stress and strain. The AP1000 fuel rod design analyses using the PAD code confirmed that the stress limits are not exceeded and that the strain remains below 1 percent under normal operating conditions.

Flow-induced fuel rod vibrations could result in significant wear. The effect of vibration on the fuel rods was determined through extensive flow tests on prototypical fuel elements. No significant wear of the clad or grid supports has been observed during the life of the fuel assembly, based on out-of-pile flow tests and observations of similar fuel designs for other reactors. In addition, design analysis, using industry-accepted methods, has not predicted such wear.

Creep collapse is a phenomenon that occurs when axial gaps in the fuel pellet column appear due to densification of the fuel pellets and subsequent collapse of the cladding into the gap. Collapsing cladding is considered a fuel failure. The applicant's analyses (using the approved methodology described in WCAP-13589-A, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," issued in March 1995) show that significant axial gaps do not form in the fuel stack, thus preventing clad collapse.

SRP Section 4.2 states that the cumulative number of strain fatigue cycles on the structural components should be significantly less than the design fatigue lifetime. An acceptable fatigue analysis is based on the O'Donnell and Langer model (O'Donnell, 1964). The applicant's fatigue analysis using the O'Donnell and Langer model shows that the cumulative fatigue life is significantly below the design fatigue lifetime.

Based on the results of the applicant's analyses under normal operating conditions for the AP1000, performed using approved methodologies, including the PAD code, the staff concludes that the fuel rod performance for the AP1000 fuel design during steady-state operations is acceptable.

#### 4.2.6.2 Fuel Assembly Performance Evaluation

The applicant evaluated the structural performance of the fuel assemblies, including the grid spacers and the IFM grids, during seismic and LOCA events. The applicant performed its analyses using the approved methodologies described in WCAP-9401-P-A, "Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly," issued in August 1981, and WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," issued in September 1985. SRP Section 4.2, Appendix A, states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during seismic and LOCA events.

For grid spacer components, the maximum grid impact force induced by either a seismic or a pipe break event must be less than the maximum grid crushing load. Based on the use of the approved leak-before-break criteria, the applicant demonstrated that a pipe rupture induced by a safe-shutdown earthquake is highly unlikely, precluding the need to combine both seismic and LOCA loads for grid analysis. Using the methodology described in WCAP-9401-P-A, the applicant determined that the grid loads from either a seismic or pipe break event will not cause unacceptable grid deformation, thereby maintaining the coolable geometry for the AP1000 fuel design. The staff finds the applicant's analyses to be acceptable because they were performed using approved methodology.

The applicant assessed the stresses induced in the various fuel assembly IFM grid components using the most limiting seismic condition. The seismic-induced stresses were compared with the allowable stress limits for the major components of the fuel assembly. The results showed that the component stresses are below the established allowable limits. Based on the above, the staff concludes that the fuel assembly IFM grid components in the AP1000 fuel design are acceptable for the design-basis seismic event.

### 4.2.7 Testing and Inspection Plan

The AP1000 fuel is subject to a quality assurance (QA) program similar to those associated with earlier Westinghouse fuel designs. This QA program ensures that the fuel is fabricated in accordance with the design bases, reaches the plant site undamaged, and is correctly loaded into the core without damage. Online fuel rod failure monitoring and postirradiation surveillance will be performed to detect anomalies or confirm that the fuel system is performing as expected. The QA program is described in the Westinghouse Electric Company's Quality Management System (QMS), Revision 5, issued on October 1, 2002, which was approved by the NRC in a safety evaluation dated September 13, 2002. Based on the above, the staff found that the Westinghouse QMS meets the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

### 4.2.8 Conclusions

On the basis of the information discussed above, the staff determined that the AP1000 fuel system is designed to meet the following objectives:

- The fuel system will not be damaged by 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 is not underestimated for postulated accidents.
- Core coolability will always be maintained during design-basis transients and accidents.

Accordingly, the fuel system conforms to the acceptance criteria of SRP Section 4.2. Therefore, the staff concludes that the AP1000 fuel system design (including the control assembly design) satisfies the requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR Part 100.

In DCD Tier 2, Sections 4.2.5, 4.3.4, and 4.4.7, "Combined License Information," Westinghouse stated that COL applicants referencing the AP1000 certified design will address any changes to the reference design of the fuel, burnable absorber rods, and RCCAs from that presented in the DCD. The staff finds this to be acceptable. This is COL Action Item 4.2.8-1.

## 4.3 Nuclear Design

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

### 4.3.1 Design Basis

DCD Tier 2, Section 4.3, "Nuclear Design," presents the design bases for the AP1000 nuclear design. The nuclear design must ensure that the specified acceptable fuel design limits will not be exceeded during normal operation, including anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core. To meet these objectives, the nuclear design must conform to the following GDC:

- GDC 10, requiring the reactor design (reactor core, reactor coolant system, control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.

- GDC 11, "Reactor Inherent Protection," requiring a net negative prompt feedback coefficient in the power operating range.

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

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

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

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

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

- GDC 27, requiring, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the emergency core cooling system (ECCS).

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

As discussed in the following sections, the staff finds that the design bases presented in the DCD comply with the requirements of the above GDC and, therefore, are acceptable.

### 4.3.2 Description

4.3.2.1 Nuclear Design Description

The DCD contains the description of the first cycle fuel loading, which consists of a specified number of fuel bundles. Each fuel bundle (assembly) contains a 17x17 rod array composed nominally of 264 fuel rods, 24 rod cluster control thimbles, and an in-core instrumentation thimble. The fuel rods within a given assembly have the same uranium enrichment in both the radial and axial planes. To attain a desired radial power distribution, three batches of fuel assemblies contain rods of different enrichment. The central region of the core will consist of the lower enrichment, while the higher enriched assemblies will be placed on the periphery. Axial blankets are included in the design to reduce neutron leakage and to improve fuel utilization. Reload cores are anticipated to operate approximately 18 months between refueling, accumulating a cycle burnup of approximately 21,000 MWD/MTU.

DCD Tier 2, Table 4.3-1, “Reactor Core Description (First Cycle),” DCD Tier 2, Table 4.3-2, “Nuclear Design Parameters (First Cycle),” and DCD Tier 2, Table 4.3-3, “Reactivity Requirements for Rod Cluster Control Assemblies,” contain summaries of the reactor core design parameters, including critical soluble boron concentrations and worths, reactivity coefficients, delayed neutron fraction, neutron lifetimes, and plutonium buildup. Values presented for the delayed neutron fraction and prompt neutron lifetime at the beginning and the end of the cycle are neutronic parameters typically used in a Westinghouse fuel design, such as VANTAGE+ fuel, and are usually included as part of the standard reload design procedure. The reactor core design parameters contained in DCD Tier 2, Tables 4.3-1 through 4.3-3, are designated as Tier 2* information. Any departure from these tables, including the fuel and reactivity controls information, will require prior NRC approval.

4.3.2.2 Power Distribution

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

The accuracy of power distribution calculations has been confirmed through approximately 1000 flux maps using the methods documented in WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” issued in June 1988. The total peaking factor, $F_Q$, for the AP1000 is 2.60, corresponding to 15.0 kW/ft. The average linear power for the AP1000 is 5.72 kW/ft.

The design bases affecting power distribution of the AP1000 include the following parameters:

- The peaking factor in the core will not be greater than 2.60 during normal operation at full power to meet the initial conditions assumed in the LOCA analysis.

- Under abnormal conditions (including maximum overpower), the peak linear heat rate will not cause fuel melting.

- The core will not operate, during normal operation or AOOs, with a power distribution that will cause the departure from nucleate boiling ratio (DNBR) to fall below the DNBR limit using the WRB-2M departure from nucleate boiling (DNB) correlation and the corresponding statistical uncertainties described in WCAP-15025-P-A, “Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids,” issued in April 1999.

GDC 13 provides the required criteria to evaluate online and ex-core monitoring. The online core monitoring system will be employed to continuously monitor important reactor characteristics and establish margins to operating limits. The online core monitoring system will provide, on demand, the operator-detailed power distribution information in both the radial and axial direction. This system, which consists of software executed on the plant computer, will utilize the output of the fixed in-core detector system to synthesize the core average power distribution. The processing algorithms contained within the online monitoring system are identical to those historically used for the evaluation of power distribution measurements in Westinghouse pressurized-water reactor (PWRs). WCAP-12472-P-A, “BEACON: Core Monitoring and Operations Support System,” issued in August 1994, describes these algorithms, which have been approved for use by the staff in a safety evaluation dated February 16, 1994.

Ex-core detectors register signals which are then processed and calibrated against in-core measurements to derive the power at the top and bottom of the core. These calibrated measurements (referred to as the flux difference, ΔI), are displayed on a panel in the control room. These data determine the shape penalty function to the overtemperature delta T (OPΔT) DNB protection and the overpressure delta T (OPΔT) overpower protection.

The online monitoring system also evaluates the power distribution based on the conditions prevalent in the reactor at that time. It provides the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment, and operational recommendations to manage and maintain required thermal margins. As such, the online monitoring system provides the primary means of managing and maintaining required operating thermal margins during normal operations.

On the basis of the design information provided in DCD Tier 2, Section 4.3.2.2, “Power Distribution,” regarding the power distributions and core monitoring, the staff concludes that this section of the DCD is acceptable because it meets the acceptance criteria of GDC 13.

### 4.3.2.3 Reactivity Coefficients

The reactivity coefficients express the effects of changes in the core conditions, such as power, fuel and moderator temperature, moderator density, and boron concentration, on core reactivity. These coefficients vary with fuel burnup and power level. The applicant has provided calculated values of the coefficients in DCD Tier 2, Table 4.3-2. The applicant used NRC-approved physics methods to determine these reactivity coefficient calculations. In addition, moderator and Doppler coefficients, along with boron worth, will be measured as part of the startup physics testing to assure that the actual values for these parameters are within the range of those used in these analyses.

The AP1000-predicted MTC values are negative for the full range of expected operating conditions during the initial cycle. The value of the MTC is a function of the concentration of the soluble boron; this value becomes more positive as the boron concentration increases. The AP1000 design uses burnable absorbers to reduce the required boron concentration, thus ensuring that the MTC remains negative over the range of power operation. The effect of the burnable rods is to make the moderator coefficient more negative.

The staff finds these values for the reactivity coefficients to be acceptable because they are negative and meet the requirements of GDC 27.

### 4.3.2.4 Control Requirements

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

A chemical poison dissolved in the coolant, RCCAs, and GRCAs controls core reactivity. The reactivity control systems are designed to automatically initiate reactivity control, thereby meeting the requirements of GDC 20. To allow for changes in reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of positive reactivity is built into the core. As described below, the DCD provides adequate information about the reactivity balance for the first core, and shows that the design incorporates methods to control excess reactivity at all times. This meets the requirements of GDC 27.

Moving control rod drive (CRD) assemblies or adjusting the boron concentration in the reactor coolant and the thermal-hydraulic conditions of the core can control both excess reactivity and power level. The addition of soluble boron to the coolant and the burnable absorbers can control the excess reactivity, when necessary. The DCD describes the boron concentration for several AP1000 core configurations, including the unit boron worth for the initial cycle. The combination of control systems satisfies the requirements of GDC 25 and 26 because there are two independent systems of different design. Even with a single malfunction of the system, the control systems will still assure that the fuel design limits are not exceeded.

AP1000 plants will likely operate at steady-state full power. RCCAs and/or GRCAs permit operators to compensate for fast reactivity changes (e.g., changes in power level and the effects of minor variations in moderator temperature and boron concentration) without impairing shutdown capability.

Gray rods and control rods assist primarily in controlling core power distribution, including xenon-induced axial power oscillations during operation, and axial power shape during load-following transients. The rod control system automatically modulates the insertion of the axial offset control bank, which controls the axial power distribution, simultaneous with the mechanical shim gray and control rod banks to maintain programmed coolant temperature. Gray rods and control rods can also control reactivity to compensate for minor variations in moderator temperature and boron concentration during power operations. They can also assist in compensating for reactivity changes caused by power level and xenon changes during load-following transients. The total reactivity worth of these rods will enable licensees to control load-following transients without changing boron concentration.

The power-dependent insertion limits given in DCD Tier 2, Chapter 16, “Technical Specifications,” control rod insertion. These limits ensure that (1) sufficient negative reactivity is available to quickly shut down the reactor with ample margin, and (2) if a control rod were ejected (an unlikely event), the worth of the ejected rod would be no more than the rod worth assumed in the accident analysis.

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

The staff reviewed the AP1000 calculated rod worths and the uncertainties in these worths. The applicant based these calculations on many reactor-years of startup test data for pressurized-water reactor (PWR) critical experiments. The calculations show that the rod values are typical. On this basis, the staff has determined that the assessment of the reactivity control system is suitably conservative, and that the control system has adequate negative reactivity worth to ensure shutdown capability, assuming that the most reactive control rod is assumed stuck in the fully withdrawn position. Therefore, the RCCAs and soluble boron worths are acceptable for use in the accident analysis.

On the basis of its review of the information provided in DCD Tier 2, Section 4.3, as described above, the staff concludes that the functional design of the AP1000 reactivity control systems meets the requirements of GDC 20, 25, 26, and 27 and, therefore, is acceptable.

#### 4.3.2.5 Stability—Xenon-Induced Spatial Oscillations

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

DCD Tier 2, Section 4.3.2.7, “Stability,” discusses the stability of the reactor with respect to xenon-induced power distribution oscillations and the control of such transients. Because the AP1000 core is 0.6 m (2 ft) taller than the typical Westinghouse 3.66 m (12 ft) cores, analysis has shown that the AP1000 is expected to be slightly less stable axially, with respect to axial xenon oscillations. However, the online monitoring system is designed as an integral component of the AP1000 reactor and will provide monitoring of power distribution (axially and radially) and guidance to the plant operator as to the timing and appropriate actions to be taken to maintain a stable core. Also, ex-core detectors provide the plant operator with additional indication in the event of axial xenon-induced spatial oscillations.

In analyzing the xenon stability issue for the AP1000, the applicant drew on its experience with other 4.3 m (14 ft) cores, such as those at South Texas Units 1 and 2, Tihange Unit 3, and Doel Unit 4. Using the industry-accepted and NRC-approved computer code, Panda (see WCAP-7084-P-A, “The Panda Code,” issued in February 1975), the applicant performed computational comparisons for typical 3.66 m (12 ft) and 4.3 m (14 ft) cores, at beginning-of-cycle life and at end-of-cycle life. The analysis showed that the axial oscillation period is comparable for both 3.66 m (12 ft) and 4.3 m (14 ft) cores. The analysis also showed that at beginning-of-cycle life, a 3.66 m (12 ft) core has a period of approximately 27 hours; the 4.3 m (14 ft) core has a period of approximately 28 hours. At end-of-cycle life, periods of about 32 and 34 hours were obtained for the 3.66 m (12 ft) and 4.3 m (14 ft) cores, respectively. These values are plant specific and depend heavily on specific core design and burnup.

The rod control system will automatically react to changes in the power distribution that fall outside very tight axial bands. Axial offset control rod banks are designed specifically to maintain a constant axial offset over the entire operating range of the core. In addition, the same control system can be operated manually to maintain an axial offset within prescribed operating bands, or core protection limits. If the axial offset exceeds prescribed operating power bands, the turbine is automatically reduced, or a reactor trip is generated, or both actions are taken. In summary, the staff determined that the AP1000 design incorporates reliable systems (1) to monitor power distribution and detect oscillations, and (2) to suppress axial power oscillations automatically. Accordingly, the staff finds that the applicant has properly addressed the concern of xenon-induced spatial oscillations in accordance with the guidelines of the SRP and has satisfied GDC 12. Therefore, based on the above analysis, the design is acceptable with respect to power oscillations.

### 4.3.3 Analytical Methods

In DCD Tier 2, Section 4.3.3.2, “Macroscopic Group Constants,” the applicant described the PHOENIX-P and LEOPARD/CINDER computer programs and calculation methods used to calculate the nuclear characteristics of the reactor design. The applicant used the NRC-approved computer code, PHOENIX-P (see WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” issued in June 1988), in place of LEOPARD/CINDER to generate typical core parameters. Based on the calculated core parameters, and the applicant’s use of PHOENIX-P, the staff concludes that the information presented adequately demonstrates the ability of this analytical method to calculate the reactor physics characteristics of the AP1000 core.

### 4.3.4 Summary of Evaluation Findings

To allow for changes in reactivity from reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, the applicant has designed a significant amount of excess reactivity into the core. The applicant has provided substantial information about core reactivity balances for the first cycle, and has shown that the design incorporates methods to control excess reactivity at all times. The applicant has shown that sufficient control rod worth would be available at any time during the cycle to shut down the reactor, assuming that, with at least a 2.0-percent delta k/k subcritical margin in the hot shutdown condition, the most reactive control rod is stuck in the fully withdrawn position.

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

The applicant described the computer programs and calculation techniques used to predict the nuclear characteristics of the reactor design and provided examples to demonstrate the ability of these methods to predict experimental results. The information presented adequately demonstrates the ability of these analyses to predict the reactivity and physics characteristics of the AP1000 design.

With respect to the requirements applicable to the nuclear design of the AP1000, the staff finds the following:

- The applicant has satisfied the requirements of GDC 10, 20, and 25 with respect to SAFDLs by demonstrating that the AP1000 design meets the following objectives:
    - No fuel damage occurs during normal operation, including the effects of AOOs (GDC 10).
    - Automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs and that systems and components important to safety will automatically operate under accident conditions (GDC 20).
    - No single malfunction of the reactivity control system will violate the fuel design limits (GDC 25).
- The staff reviewed the results of the applicant's calculations to demonstrate that the Doppler and moderator coefficients of reactivity are negative and will prevent a rapid, uncontrolled reactivity excursion. The staff has determined that the calculations are suitably conservative, were performed with NRC-approved physics methods, and use appropriate AP1000-specific inputs. Accordingly, the applicant has satisfied the requirements of GDC 11 with respect to nuclear feedback characteristics.

- The staff reviewed the applicant's analysis of power oscillations and has determined that the analysis is suitably conservative, was performed with NRC-approved physics methods, and used appropriate AP1000-specific inputs. Accordingly, the applicant has satisfied the requirements of GDC 12 by showing that power oscillations can be reliably and readily detected and suppressed.

- The staff reviewed the applicant's core monitoring system, and found that the applicant has satisfied the requirements of GDC 13 by providing instrumentation and controls to monitor the following variables and systems that can affect the fission process:

  - reactor coolant system (RCS)
  - steam and core power conversion systems
  - containment
  - engineered safety systems
  - auxiliary systems
  - reactor power distribution
  - control rod positions and patterns
  - process variables, such as temperatures and pressures

- The AP1000 design includes RCCAs and GRCAs, as well as a chemical shim (boric acid), which provide the following capabilities:

  - reliable shutdown of the reactor during normal operation conditions and during AOOs

  - adequate boration to establish and maintain safe-shutdown conditions

  Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 26 by providing two independent reactivity control systems of different design.

- The AP1000 design provides reactivity control systems, in conjunction with absorber addition by the ECCS, to reliably control reactivity changes under the following postulated accident conditions:

  - The design provides a movable rod reactivity control system and a liquid reactivity control system.

  - The applicant has performed calculations to demonstrate that the core has margin sufficient to shut down the reactor, assuming the highest-worth RCCA is stuck, as discussed in Section 4.3.2.4 of this report.

  Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 27.

- The applicant has followed the methodology described in WCAP-7588-A, "An Evaluation of Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial

> Kinetics Methods," Revision 1, issued in January 1975. This NRC-approved topical report analyzes the assumptions used in evaluating a control rod ejection accident for PWRs. Moreover, the criteria and results presented in WCAP-7588 are within the criteria and limits prescribed by Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 28 with respect to postulated reactivity accidents.

For the reasons set forth above, the staff concludes that the AP1000 nuclear design satisfies the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28 and, therefore, is acceptable.

## 4.4 Thermal-Hydraulic Design

In its review of the AP1000 thermal-hydraulic design, the staff considered information contained in the DCD, responses to the staff's RAIs, and the topical reports referenced by the applicant. In addition, the staff conducted its review in accordance with the guidelines provided by SRP Section 4.4, "Thermal and Hydraulic Design." As described in the following sections, the thermal and hydraulic design of the reactor core provides adequate heat transfer compatible with the heat generation distribution in the core.

### 4.4.1 Thermal-Hydraulic Design Bases

The principal thermal-hydraulic design basis for the AP1000 reactor core is to ensure adequate heat removal to prevent fuel damage during any conditions of normal operation, including the effects of anticipated operational transients. GDC 10 specifies that the reactor core and associated coolant, control, and protection systems must be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. SRP Section 4.4, "Thermal and Hydraulic Design," sets forth the acceptance criteria used by the staff to evaluate the thermal-hydraulic design of the reactor core. The acceptance criteria are based on the relevant requirements of GDC 10.

#### 4.4.1.1 Departure From Nucleate Boiling

The DNB design basis is one of the reactor core thermal-hydraulic design bases for complying with the SAFDLs. As stated in SRP Section 4.4, the DNB design basis requires at least a 95 percent probability, at a 95 percent confidence level, that the limiting fuel rods in the core will not experience DNB during normal operation, any transient conditions arising from faults of moderate frequency, or AOOs. To this end, a limit for the DNBR (defined as the predicted critical heat flux that would result in a DNB (or DNB heat flux) divided by the actual heat flux) was established. This limit requires at least a 95 percent probability, at a 95 percent confidence level, that the hot fuel rod in the core will not experience a DNB when the calculated DNBR is higher than the DNBR limit. The AP1000 DNBR calculation is performed with the VIPRE-01 reactor core thermal-hydraulic analysis computer code and the WRB-2M critical heat flux correlation. The VIPRE-01 reactor core thermal-hydraulic analysis computer code is described in the Electric Power Research Institute (EPRI) NP-2511-CCM-A, "VIPRE-01: A Thermal-

Hydraulic Code for Reactor Core," Volumes 1–3, issued in August 1989, and Volume 4, issued in April 1987, and in WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," issued in October 1999. DCD Tier 2, Section 4.1.1, specifies that the minimum DNBR calculated using the WRB-2M correlation must be greater than or equal to 1.14 during normal operation and anticipated transient conditions. This principal design requirement is Tier 2* information; thus, any departure from this criterion requires prior NRC approval.

In calculating the DNBR, uncertainties in the values of process parameters, core design parameters, and the calculation methods used in the assessment of thermal margin should be treated with at least a 95 percent probability at a 95 percent confidence level.

The applicant performed the AP1000 thermal-hydraulic design analyses using the revised thermal design procedure (RTDP) described in WCAP-11397-P-A, "Revised Thermal Design Procedure," issued in April 1989. The RTDP is a statistically based methodology whereby uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are statistically combined to determine DNB uncertainty factors. Section 4.4.2 of this report provides a more detailed discussion on this subject.

To maintain a DNBR margin, and thus offset DNB penalties such as those attributable to fuel rod bow, the applicant performed safety analyses using DNBR limits higher than the design-limit DNBR values. The difference between the design-limit DNBRs and the safety analysis DNBRs is the available DNBR margin.

4.4.1.2 Fuel Temperature Design Basis

Another SAFDL is that fuel melting will not occur at the overpower limit for American Nuclear Society (ANS) Condition I (normal operation and operational transients) and Condition II (events of moderate frequency) events, as specified in DCD Tier 2, Section 4.1.1. This fuel melting design basis requires, during modes of operation associated with ANS Condition I and Condition II events, at least a 95 percent probability, at a 95 percent confidence level, that the peak centerline temperature of the fuel rods will not exceed the $UO_2$ melting temperature. The melting temperature of unirradiated $UO_2$ is assumed to be 2804.4 °C (5080 °F), decreasing by 14.4 °C (58 °F) per 10,000 MWD/MTU. By precluding $UO_2$ melting, the AP1000 design preserves the fuel geometry and eliminates the possible adverse effects of molten $UO_2$ on the cladding. The applicant performed fuel rod thermal evaluations for Condition I and Condition II events and verified that, even at high local powers, the fuel centerline temperature is calculated to be below the $UO_2$ melting temperature limit, thereby meeting the fuel temperature design basis. It should be noted that the applicant has chosen 2593 °C (4700 °F), which is the $UO_2$ temperature at a burnup of 62,000 MWD/MTU, for the calculated fuel centerline temperature limit for all burnups. This is acceptable because the NRC-approved method described in WCAP-12488-A limits the evaluation to a maximum fuel rod average burnup of 62,000 MWD/MTU. In DCD Tier 2, Section 4.3.1.1.1, "Basis," this fuel rod burnup limit is designated as Tier 2* information, requiring NRC approval prior to any departure.

4.4.1.3 Core Flow Design Basis

This section addresses the minimum coolant flow through the fuel rod regions at the entrance of the reactor vessel. Core cooling evaluations are dependent on the thermal flow rate (minimum flow) entering the reactor vessel. The AP1000 core flow design basis requires that a minimum of 94.1 percent of the thermal flow rate passes the fuel rod region of the core and is effective for fuel rod cooling. A maximum of 5.9 percent core bypass flow is not considered effective for core heat removal. The core bypass flow includes coolant flow through the rod cluster control guide thimble tubes, core shroud region, head cooling spray nozzles, outlet nozzles, and baffle plate-core cavity gap, as listed in DCD Tier 2, Figure 5.1-3, “Reactor Coolant System—Loop Layout.”

The maximum bypass flow fraction of 5.9 percent assumes the use of thimble-plugging devices in the rod cluster control guide thimble tubes that do not contain any other core components.

4.4.1.4 Hydrodynamic Stability

In accordance with the acceptance criteria provided in SRP Section 4.4, the reactor should have sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation and AOOs. The hydrodynamic stability design basis for the AP1000 reactor specifies that modes of operation associated with ANS Condition I and Condition II events do not lead to hydrodynamic instability.

In DCD Tier 2, Section 4.4.4.6, “Hydrodynamic and Flow Power Coupled Instability,” the applicant stated that the AP1000 is thermal-hydraulically stable. The potential for hydrodynamic instability exists in steady-state, two-phase, heated flow in parallel channels. Boiling flows may also be susceptible to thermodynamic instabilities. These instabilities are undesirable in reactors because they may cause a change in thermal-hydraulic conditions, which may lead to a reduction in the DNB heat flux, relative to that observed during a steady-flow condition, or to undesired forced vibrations of core components. Therefore, the applicant developed a thermal-hydraulic design criterion that states that modes of operation under ANS Condition I and Condition II events must not lead to thermal-hydrodynamic instabilities.

The AP1000 reactor design considers two specific types of flow instabilities. Specifically, these are the Ledinegg, or flow excursion type of static instability, and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady-state to a lower value. This instability occurs when the slope of the RCS pressure drop-flow rate curve (internal characteristic of the channel) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve (external characteristic of the channel).

Therefore, the flow excursion instability does not occur if the partial derivative of the pressure drop, with respect to the flow rate of the RCS, is greater than or equal to the derivative of the head with respect to the flow of the reactor coolant pump head-capacity curve. The Westinghouse pump head curve has a negative slope, whereas the RCS pressure drop-flow curve has a positive slope over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The applicant also considered the dynamic density wave instability. DCD Tier 2, Section 4.4.4.6, provides a brief description of the mechanism of density wave oscillations in a heated boiling channel. In a heated boiling channel, an inlet flow fluctuation produces an enthalpy perturbation. This, in turn, perturbs the length and the pressure drop of the single-phase region, and causes quality or void perturbations in the two-phase regions that travel up the channel with the flow. These quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, because the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

The applicant assessed the density wave instability of typical Westinghouse reactor designs, such as South Texas Units 1 and 2, under Condition I and Condition II operation. The assessment was performed using the simplified stability criterion of Ishii (Saha, 1976), which was developed for parallel closed-channel systems to evaluate whether a given condition is stable with respect to the density-wave-type of dynamic instability. The results indicate that a large margin-to-density wave instability exists (e.g., increases on the order of 150 percent of rated reactor power would be required for the predicted inception of this type of instability).

The application of Ishii's method to Westinghouse PWR designs with open-lattice cores is conservative. For such open-lattice cores, there is little resistance to lateral flow leaving the flow channels of high-power density. There is also energy transfer from channels of high-power density to channels of lower-power density. This coupling with cooler channels has led to the conclusion that an open-channel configuration is more stable than the above, closed-channel analysis under the same boundary conditions. Moreover, boiling flow density wave instability tests performed by Kakac, et al. (Kakac, 1974) in a cross-connected, four-parallel-channel upflow system showed that boiling in a cross-connected system is more stable than a boiling system without cross connection or a system having a smaller number of channels. The PWR open-lattice cores with less cross-flow resistance than the cross-connected parallel channels would be even more stable.

Observed flow instabilities have occurred almost exclusively in closed-channel systems operating at low pressure relative to the Westinghouse PWR operating pressures. Kao, et al. (Kao, 1973) analyzed parallel closed-channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 15.2 MPa (2200 psia). The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 8.3 MPa (1200 psia).

Moreover, the DNB tests performed for many Westinghouse rod bundles over wide ranges of operating conditions show no evidence of premature DNB or inconsistent data that might indicate flow instabilities in the rod bundle. The data from these tests provide additional evidence that flow instabilities do not adversely affect thermal margin.

Based on the above evaluation, flow excursion and density wave instabilities will not occur under Condition I and Condition II modes of operation for Westinghouse PWR reactor designs. There is a large power margin to the predicted inception of these instabilities. Minor plant-to-plant differences in Westinghouse reactor designs, such as fuel assembly arrays, core power-to-flow ratios, and fuel assembly length, will not result in gross deterioration of the above power margins.

As set forth above, the staff concludes that past operating experience, flow stability experiments, and the inherent thermal-hydraulic characteristics of Westinghouse PWRs provide a basis for accepting the AP1000 stability evaluation.

### 4.4.2 Thermal-Hydraulic Design of the Reactor Core

The AP1000 reactor core contains 157 fuel assemblies. Each assembly consists of 264 fuel rods in a 17x17 square array with a guide thimble in the center position for in-core instrumentation and 24 guide thimbles for the RCCA. Section 4.2.1 of this report describes the AP1000 17x17 XL RFA design. DCD Tier 2, Table 4.4-1, “Thermal and Hydraulic Comparison Table,” compares the design parameters for the AP1000, the AP600, and a Westinghouse-designed plant using XL RFAs.

#### 4.4.2.1 Thermal-Hydraulic Analyses Methods

The applicant performed the AP1000 core thermal-hydraulic analysis using the VIPRE-01 computer code and the WRB-2M critical heat flux correlation.

VIPRE-01 is a subchannel, thermal-hydraulic computer code used to analyze the reactor core of a reactor system. Battelle Pacific Northwest Laboratories, under the sponsorship of EPRI, developed VIPRE-01 and submitted it to the NRC for generic review in 1984. The NRC approved VIPRE-01 for application to PWRs in 1985, with the condition that each VIPRE-01 user submit documentation describing the proposed use for the code, other computer codes with which it will interact, the source of each input variable, and the selected correlations and their justification. WCAP-14565-P-A, issued in October 1999, documents the applicant’s use of VIPRE-01 for Westinghouse-designed PWRs. The staff approved this topical report in 1999. The staff has determined that use of VIPRE-01 for the AP1000 core thermal-hydraulic analysis is acceptable because the AP1000 is a Westinghouse-designed PWR for which the VIPRE-01 modeling is qualified, as described in WCAP-14565-P-A.

In the thermal-hydraulic analysis to calculate the DNBR, uncertainties in the values of process parameters, core design parameters, and the calculation methods used in the assessment of thermal margin should be treated with at least a 95 percent probability at a 95 percent confidence level. The applicant performed the AP1000 thermal-hydraulic design analyses using the RTDP; for those analyses in which the RTDP was not applicable, the applicant used the standard thermal design procedure.

In the standard thermal design method, the parameters used in the analysis are treated in a conservative way in terms of the DNBR. The parameter uncertainties are applied directly to the input values to the plant safety analyses. This gives the lowest minimum DNBR. The DNBR limit for the standard thermal design procedure is the DNB correlation limit plus the appropriate DNB margin to cover any DNBR penalties associated with the analysis.

The RTDP, described in WCAP-11397-P-A, is a statistically based methodology whereby uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and the DNB correlation predictions are statistically combined to determine the RTDP design-limit DNBR, which is higher than the DNB correlation limit. Because the derivation of the RTDP design DNBR limits accounts for the uncertainties of these parameters, the safety analyses input uses nominal values for these parameters. The staff has approved WCAP-11397-P-A for generic application of the RTDP methodology to PWRs, subject to certain restrictions, including the use of plant-specific uncertainties and sensitivity factors. The RTDP methodology is acceptable for application to the AP1000 design, with the same restrictions, because the AP1000 is a Westinghouse PWR design for which the RTDP methodology is qualified, as discussed in WCAP-11397-P-A.

4.4.2.2 Departure from Nucleate Boiling

For the AP1000 reactor, the applicant calculated the DNBRs using the WRB-2M critical heat flux (CHF) correlation, described in WCAP-15025-P-A. The applicant also used the VIPRE-01 code for the core subchannel analysis to determine the flow distribution in the core and the local conditions in the hot channels for use in the WRB-2M correlation for the CHF calculation.

The staff has approved the WRB-2M correlation for predicting CHF in the modified 17x17 Vantage 5H fuel, with or without modified IFM grids. The WRB-2M correlation has a DNBR limit of 1.14, with the use of the THINC-IV code or the VIPRE-01 code. Table 1 in the staff's safety evaluation for WCAP-15025-P-A specifies the WRB-2M correlation applicability ranges of various parameters, including pressure, local mass velocity, local quality, heated length, grid spacing, and equivalent hydraulic and heated diameters. Because the AP1000 17x17 RFA fuel assemblies use the same modified V5H mixing vane grid design and the same IFM grid design that was used to develop the WRB-2M correlation, and the parameters are applied within the correlation's specified range of applicability, the staff concludes that a WRB-2M correlation with a DNBR limit of 1.14 is acceptable for the AP1000 DNBR calculation using the VIPRE-01 code.

The WRB-2M correlation is used for the analysis of the AP1000 RFA fuel within its ranges of applicability. In response to RAI 440.024, the applicant stated that the local mass velocity does not fall below the lower limit of the WRB-2M local mass velocity range of applicability for any of the AP1000 design-basis transients. In addition, in the VIPRE-01 core thermal-hydraulic analysis, the output local conditions at the location of minimum DNBR are checked to determine whether the conditions are within the range of applicability. Whenever conditions are outside the applicability ranges of the WRB-2M correlation, the WRB-2 or W-3 CHF correlation is used. The WRB-2 correlation is used for the heated rod span above the first mixing vane grid, and the W-3 correlation is used in the heated region below the first mixing vane grid. This is acceptable because the WRB-2 was developed using mixing vane data and is applicable in the mixing vane region. Further, the W-3 correlation, which was based on nonmixing vane data, is applied to the lower fuel span without mixing vane grids.

As the RTDP procedure is used for the thermal-hydraulic analysis, the RTDP design DNBR limits are determined. The RTDP procedure statistically combines the uncertainties of reactor power, reactor coolant flow rate, inlet temperature, pressure, core bypass flow, enthalpy rise nuclear hot channel factor and engineering hot channel factor, the core thermal-hydraulic code, and the system transient code, as well as the uncertainty of the WRB-2M correlation. The RTDP design-limit DNBR value is 1.25 for both the typical cell and the thimble cell for core and axial offset limits; the design-limit DNBR values are 1.22 for the typical cell and 1.21 for the thimble cell for all other RTDP transients. In response to RAI 440.022, Revision 1, the applicant provided, in Tables 440.022 PR1-1 and 440.022 PR1-2, respectively, the derivations of these design-limit DNBRs, including the uncertainty values, and the sensitivity values of the RTDP parameters based on the WRB-2M correlation and the VIPRE-01 code. The results confirm that these RTDP design DNBR limits are acceptable.

The AP1000 Technical Specifications Section 2.1.1, “Reactor Core Safety Limits,” and limiting condition for operation (LCO) 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” in DCD Tier 2, Section 16.1, “Technical Specifications,” identify the limits for power, temperature, pressure, and flow through the Core Operating Limits Report (COLR). The limits specified in the COLR for these parameters, which may vary for each fuel cycle, are a combination of the values assumed in the safety analyses and the associated instrumentation uncertainties for these parameters. Measurement uncertainties for the reactor trip system and the instrumentation setpoints of the engineered safeguards actuation system, as well as other technical specification (TS) limits which can be affected by instrumentation uncertainties, cannot be determined until the plant-specific setpoint calculation is completed by the COL applicants and the actual instrumentation has been selected for the plant. DCD Tier 2, Section 7.1.6, “Combined License Information,” states that COL applicants referencing the AP1000 certified design will provide a calculation of setpoints for protective functions consistent with the methodology presented in WCAP-14605, “Westinghouse Setpoint Methodology for Protection Systems- AP600,” issued in April 1996. WCAP-14605 provides sufficient information on instrument setpoints for the COL applicant to establish setpoints for plant-specific equipment.

In its response to RAI 440.022, the applicant stated that, based on experience, the instrumentation uncertainties are expected to be typical values that bound both the specified and delivered uncertainties for the plant instrumentation. In the unlikely event that the assumed uncertainty values are exceeded when the plant is built, the calculated COLR limits could be adjusted to accommodate any additional uncertainties for the installed instrumentation beyond the originally assumed uncertainty values. In addition, the safety analyses are performed with safety-analysis-limit DNBRs higher than the design-limit DNBR values. The difference between the safety-analysis-limit DNBRs and the design-limit DNBRs is the DNBR margin, which can be used to offset DNB penalties, such as rod bow penalty and unanticipated DNBR penalties. Therefore, the staff believes that even with the revised design-limit DNBR values, the conclusion that the minimum DNBR design limits are not violated during AOOs will remain valid. However, upon installation of the actual instrumentation in the plant, the COL applicant should calculate the design-limit DNBR values using the RTDP with the actual instrumentation uncertainties of the plant's operating parameters. On the basis of this calculation, the COL applicant should confirm that either the design-limit DNBR values, as described in DCD Tier 2, Section 4.4, and the applicant's response to RAI 440.022, Revision 1, remain valid, or the minimum DNBR assumed in the safety analysis bounds the new design-limit DNBR values plus DNBR penalties, such as rod bow penalty. Open Item 4.4-1 identified that DCD Tier 2, Section 4.4.7, did not address this COL action. The applicant revised DCD Tier 2, Section 4.4.7, and added the following paragraph:

> Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in [DCD Tier 2, Section] 7.1.6, combined license applicants will calculate the design limit DNBR values using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in [DCD Tier 2,] Section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design DNBR values plus DNBR penalties, such as rod bow penalty.

This is COL Action Item 4.4-1.

The staff finds the addition of this paragraph to be acceptable. Therefore, Open Item 4.4-1 is resolved.

4.4.2.3 <u>Effects of Fuel Rod Bow on Departure from Nucleate Boiling</u>

The bowing of heated rods reduces the gaps between fuel rods, and produces an adverse effect on the CHF, if the rods are so severely bowed that they produce contact or near contact of two heated rods. This adverse effect on the DNBR is accounted for through the rod bow penalty in the DNBR safety analysis of Condition I and Condition II events for each plant application. The amount of rod bow, and its associated DNBR penalty, is calculated using the NRC-approved methodology described in WCAP-8691, "Fuel Rod Bow Evaluation," Revision 1, issued in July 1979. WCAP-8691 describes the method for determining the amount of rod bow as a function of assembly burnup, and for calculating the DNBR penalty as a result of the bowing. In its letter dated June 18, 1986, the NRC accepted the applicant's request that the maximum rod bow penalty be limited to the value calculated with the assembly average burnup of 24,000 MWD/MTU. At burnup greater than 24,000 MWD/MTU, credit is taken for the effect of $F^{N}_{\Delta H}$ burndown because of the decrease in fissionable isotopes and the buildup of fission product inventory. Therefore, the maximum rod bow penalty will be based on a burnup of 24,000 MWD/MTU.

For the AP1000 design, the amount of rod bow DNBR penalty is calculated to be less than 1.5 percent, based on 24,000 MWD/MTU. The safety analysis for the AP1000 core accommodates this rod bow penalty, and maintains sufficient margin between the safety analysis limit DNBRs and the design DNBR limits, as described in Section 4.4.1.1 of this report.

### 4.4.3 Testing and Verification

DCD Tier 2, Chapter 14, "Initial Test Program," describes the reactor coolant flow test to be performed following fuel loading, but prior to initial criticality. The test verifies that proper coolant flow rates have been used in the core thermal-hydraulic analysis. DCD Tier 2, Chapter 14 also describes core power distribution measurements to be performed at several core power levels during initial power ascension and plant operation. These measurements are used to confirm that the core thermal-hydraulic analysis employed conservative peaking factors. DCD Tier 2, Section 4.2.4, "Testing and Inspection Plan," describes the test and inspection plan for the manufactured fuel. Fabrication measurements critical to the thermal-hydraulic analysis verify that the engineering hot channel factors in the design analysis are met. The staff also identified Open Item 4.4-1 and COL Action Item 4.4-1. Upon installation of the actual instrumentation, the COL applicant should evaluate the instrumentation uncertainties of the operating parameters and confirm the validity of the design-limit DNBR values using the RTDP, as described in DCD Tier 2, Section 4.4, and the response to RAI 440.022, Revision 1.

### 4.4.4 Instrumentation Requirements

GDC 13 requires, in part, that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions, as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. DCD Tier 2, Section 4.4.6, "Instrumentation Requirements," describes the AP1000 instrumentation systems used for monitoring reactor parameters, as discussed below.

#### 4.4.4.1 <u>In-Core Instrumentation</u>

The AP1000 design uses a fixed in-core detector system to measure in-core neutron flux distribution. The AP1000 in-core instrumentation system consists of 42 in-core instrumentation thimble assemblies, which house fixed in-core detectors, core exit thermocouple assemblies contained within an inner and outer sheath assembly, and associated signal processing and data processing equipment. Each in-core instrument thimble assembly is composed of multiple fixed in-core detectors and one thermocouple. The in-core instrument thimble assembly is positioned within the fuel assembly and exits through the top of the reactor vessel to containment. The fixed in-core detector and core exit thermocouple cables are then routed to different data conditioning and processing stations.

The primary function of the in-core instrumentation system is to provide a three-dimensional (3-D) flux map of the reactor core. Flux mapping is used to calibrate neutron detectors (the ex-core nuclear instrumentation input to the overtemperature $\Delta T$ and overpower $\Delta T$ reactor trip setpoints) used by the protection and safety monitoring system (PMS), and to provide information for optimizing core performance. The in-core instrumentation system also provides the PMS with the signal necessary for monitoring the core exit temperature. This is done by grouping the flux mapping detectors with the core exit thermocouples in the same thimble.

During plant operation, the in-core instrumentation system data processor receives the transmitted digitized fixed in-core detector signals from the signal processor, and combines the measured data with analytically derived constants, as well as certain other plant instrumentation sensor signals, to generate a full 3-D indication of nuclear power distribution in the reactor core. The analysis results are available for display in the main control room, and also provide the information needed to activate a visual alarm display to alert the operator about the current existence of, or the potential for, violations of the reactor operating limit.

The flux mapping function is not considered a safety-related function. However, because of its use for calibrating the ex-core nuclear instrumentation input to the overtemperature and overpower ΔT reactor trip setpoints, the quality of the in-core instrumentation system needs to be equivalent to the PMS. The in-core instrumentation system is comprised of seismic Category I, Class 1E equipment, which is qualified for harsh environments. Therefore, the staff finds that the AP1000 in-core instrumentation system satisfies the requirements of GDC 13.

4.4.4.2 Digital Metal Impact Monitoring System

The presence of a loose part in the RCS can indicate degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part in the RCS can contribute to component damage and material wear by frequently impacting with other parts in the system, and can pose a serious threat of partial flow blockage, with attendant DNB. In addition, a loose part increases the potential for control-rod jamming and for accumulation of an increased level of radioactive crud in the RCS. One of the acceptance criteria in SRP Section 4.4 states that the design and proposed procedures of a loose part monitoring system should be consistent with the guidance of RG 1.133, Revision 1, “Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors,” issued in May 1981.

DCD Tier 2, Section 4.4.6.4, “Digital Metal Impact Monitoring System,” describes the AP1000 loose parts monitoring system, which uses the Westinghouse digital metal impact monitoring system (DMIMS).

The DMIMS is a non-safety-related system, and is designed to detect loose parts that weigh between 0.11 and 13.61 kg (0.25 to 30 lbs). The DMIMS can also detect impact with a kinetic energy of 6.78 J (0.5 ft-lbs) on the inside surface of the RCS pressure boundary within 0.91 m (3 ft) of a sensor. The sensors are fastened mechanically to the RCS at potential loose part collection regions, including the upper and lower head region of the reactor vessel, and the inlet region of each steam generator. The DMIMS consists of several active instrumentation channels, each comprising a piezoelectric accelerometer (sensor), signal conditioning, and diagnostic equipment. The DMIMS design incorporates channel checks and functional tests of the database. The DMIMS is calibrated before plant startup. The capability exists for subsequent periodic online channel checks and channel functional tests, and for offline channel calibrations at refueling outages.

The DMIMS conforms to RG 1.133, in terms of sensor location, system sensitivity and alert level, channel separation, data acquisition, capability for sensor channel operability tests, operability for seismic and environmental conditions, and system repair. Therefore, the staff concludes that the AP1000 DMIMS is acceptable.

### 4.4.5 Conclusions and Summary

The staff's review of the thermal-hydraulic design of the AP1000 reactor core included the design-basis and steady-state analysis of the core thermal-hydraulic performance. The acceptance criteria used as the basis for this evaluation are set forth in SRP Section 4.4. The staff has determined that the AP1000 core is designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or AOOs. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance, which was reviewed by the staff, as discussed above, and found to be acceptable. However, the staff also identified Open Item 4.4-1 and COL Action Item 4.4-1. Upon installation of the actual instrumentation in the plant, the COL applicant should calculate the design-limit DNBR values using the RTDP with the actual instrumentation uncertainties of the plant-operating parameters, and confirm that either the design-limit DNBR values, as described in DCD Tier 2, Section 4.4 and the applicant's response to RAI 440.022, Revision 1, remain valid, or the minimum DNBR assumed in the safety analysis bounds the new design-limit DNBR values plus DNBR penalties, such as rod bow penalty. On the basis of the above discussion, the staff concludes that the thermal-hydraulic design of the initial AP1000 core meets the requirements of GDC 10.

## 4.5 <u>Reactor Materials</u>

### 4.5.1 Control Rod Drive System Structural Materials

The staff reviewed DCD Tier 2, Section 4.5.1, "Control Rod Drive System Structural Materials," in accordance with SRP Section 4.5.1, "Control Rod and Drive System Structural Materials". The CRD structural materials are acceptable if the relevant requirements of the following regulations are met:

- GDC 1, "Quality Standards and Records," and 10 CFR 50.55a(a)(1) require, in part, that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These quality standards shall be identified and evaluated to determine their adequacy to ensure a quality product, in keeping with the required safety function.

- GDC 14, "Reactor Coolant Pressure Boundary," requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

- GDC 26 requires, in part, that one of the radioactivity control systems shall use control rods (preferably including a positive means for inserting the rods) and shall be capable of reliably controlling reactivity changes so that specified acceptable fuel design limits are not exceeded, under conditions of normal operation, including AOOs.

The AP1000 CRD system, described in DCD Tier 2, Section 3.9.4.1, "Descriptive Information of CRDS," builds upon a Westinghouse design that has been used in many operating nuclear power plants. As described below, the staff reviewed the structural materials aspects of the CRD, as presented in the DCD, in accordance with the guidelines in SRP Section 4.5.1.

4.5.1.1 Summary of Technical Information

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the control rod drive mechanism (CRDM) and the CRD line. The DCD also provides information about the materials specifications, the fabrication and processing of austenitic stainless steel components, the contamination protection and cleaning of austenitic stainless steel, and items concerned with materials other than austenitic stainless steel.

The parts of the CRDMs and CRD line exposed to reactor coolant are made of metals that resist the corrosive action of the coolant. Three types of metals are used exclusively, stainless steel, Ni-Cr-Fe alloys, and, to a limited extent, cobalt-based alloys. Pressure-retaining materials comply with the ASME Code, Section III, which is incorporated by reference into 10 CFR 50.50a. DCD Tier 2, Table 5.2-1, "Reactor Coolant Pressure Boundary Materials Specifications," includes the materials specifications for portions of the CRDM that are part of the RCPB. These parts are fabricated from austenitic stainless steel (Type 316LN and Type 304LN). Pressure boundary parts and components made of stainless steel do not have specified minimum yield strengths greater than 620.53 MPa (90,000 psi). A Ni-Cr-Fe alloy (specifically, Alloy 690) is used for reactor vessel head penetrations.

Internal latch assembly parts are fabricated of heat-treated martensitic and austenitic stainless steel. Heat treatment prevents the initiation of stress-corrosion cracking (SCC). Components and parts made of stainless steel do not have specified minimum yield strengths greater than 620.53 MPa (90,000 psi). Magnetic pole pieces that are immersed in the reactor coolant are fabricated from Type 410 stainless steel. Nonmagnetic parts, except pins and springs, are fabricated from Type 304 stainless steel. A cobalt alloy or qualified substitute is used to fabricate link pins. Springs are made from a Ni-Cr-Fe alloy (specifically, Alloy 750). Latch arm tips are clad with a suitably hard facing material to provide improved resistance to wear. Hard chrome plate and hard facing are used selectively for bearing and wear surfaces.

The drive rod assembly is immersed in the reactor coolant and uses a Type 410 stainless steel drive rod. The drive rod coupling is machined from Type 403 stainless steel. Springs are fabricated using Ni-Cr-Fe alloy, and the locking button is fabricated of cobalt alloy bar stock, or a qualified substitute. The other parts are fabricated from Type 304 stainless steel.

The coil housing requires a magnetic material and is exposed to the containment atmosphere. Low-carbon cast steel and ductile iron are used. The finished housings are electroless nickel-plated to provide resistance against general corrosion.

#### 4.5.1.2 Evaluation

The staff reviewed and evaluated the information in DCD Tier 2, Section 4.5.1, to ensure that the materials are in accordance with the criteria of SRP Section 4.5.1 and recent guidance related to cracking of reactor pressure vessel head penetration (VHP) nozzles. Recent NRC generic communications, including NRC Bulletins 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles”; 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity”; and 2002-02, “Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs”; have addressed issues related to cracking of the VHP nozzles and degradation of the reactor pressure vessel head in operating PWRs.

##### 4.5.1.2.1 Materials Specifications

The staff reviewed DCD Tier 2, Section 4.5.1.1, “Materials Specifications,” to determine the suitability of the materials for this application. The DCD provides information on the specifications, types, grades, heat treatments, and properties used for the materials of the CRDM components. The CRD components that are part of the RCPB include the latch housing, the rod travel housing, and the CRD VHP nozzles. The housing components are fabricated from austenitic stainless steel (SA-336, Types 316LN and 304LN). These materials comply with the ASME Code, Section II and Section III requirements and are acceptable for use in the AP1000 design. The penetration nozzles are discussed later in this section.

The internal latch assembly components that are not part of the RCPB are fabricated from a variety of materials, including Type 410 stainless steel (magnetic pole pieces), Alloy 750 (springs), a cobalt alloy (link pins), and Type 304 stainless steel. Hard chrome plates and cobalt-based hard facing provide resistance to wear of load-bearing surfaces. The drive rod assembly, which is not part of the RCPB, includes a Type 403 stainless steel drive rod coupling, a Type 410 stainless steel drive rod, Alloy 750 springs, a cobalt-based alloy locking button, and some Type 304 stainless steel parts. The DCD does not provide the applicable materials specifications for the materials of either the latch assembly or the drive rod assembly, with the exception of Type 403 stainless steel and Alloy 750. Because the latch assembly and the drive rod assembly are not part of the RCPB, these component parts do not have to be designed or procured in accordance with the requirements of the ASME Code. The staff considers these non-RCPB materials to be acceptable because the materials selected are appropriate for these applications. In addition, latch and drive rod assemblies made of these materials have provided many years of successful operation in existing nuclear power plants.

##### 4.5.1.2.2 Austenitic Stainless Steel Components

DCD Tier 2, Section 4.5.1.2, “Fabrication and Processing of Austenitic Stainless Steel Components,” refers to DCD Tier 2, Section 5.2.3.4, “Fabrication and Processing of Austenitic Stainless Steel,” for a discussion of the processing, inspections, and tests on austenitic stainless steel components to prevent increased susceptibility to intergranular corrosion caused by sensitization. RG 1.44, “Control of the Use of Sensitized Stainless Steel,” provides the acceptance criteria for testing, alloy compositions, welding, heat treatment, cleaning, and protecting austenitic stainless steels to avoid severe sensitization. The AP1000 design controls, with respect to the use of sensitized stainless steel, imposed on the austenitic stainless steel of the CRDMs conform to the regulatory positions of RG 1.44.

RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," provides the acceptance criteria for delta ferrite in austenitic stainless welds. These acceptance criteria address the recommended range of delta ferrite in stainless steel weld metal to avoid microfissuring in welds. The RG also includes a recommended procedure for ferrite measurement. Welding of austenitic stainless steel components of the CRD system in the AP1000 design conforms to the acceptance criteria of RG 1.31.

DCD Tier 2, Section 5.2.3.4, offers additional discussion on the fabrication and processing of austenitic stainless steel components. Section 5.2.3 of this report documents the staff's review of DCD Tier 2, Section 5.2.3.4.

4.5.1.2.3 Other Materials

The DCD identifies that the springs in the CRDM are made from a Ni-Cr-Fe alloy, Alloy 750. Operating experience with Alloy 750 springs has shown that they have not exhibited SCC in PWR primary water environments. Accordingly, the staff finds their use acceptable.

Cobalt-based alloys have limited use in the AP1000 design. Cobalt-free or low-cobalt, wear-resistant alloys used in the AP1000 design are qualified by wear and corrosion tests, and include those developed and qualified in nuclear industry programs. Based on the qualification testing of these alloys, and the assurance provided by the successful application of these or similar materials in current nuclear power plants, the staff finds the use of these alloys in the CRD system to be acceptable and compatible with the reactor coolant.

4.5.1.2.4 Compatibility of Materials with the Reactor Coolant

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

4.5.1.2.5 Cleaning and Cleanliness Control

The staff's acceptance criteria for cleaning and cleanliness controls conforms with RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The AP1000 design conforms to RG 1.37, with the exception of quality standard American National Standard Institute (ANSI) N.45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," referenced in RG 1.37. Section 17.3 of this report presents the staff's evaluation of quality assurance documents.  The staff finds the provisions for cleaning components and systems acceptable because they conform to the regulatory positions of RG 1.37, with the exception evaluated in Section 17.3 of this report, thus satisfying the quality assurance requirements of 10 CFR Part 50, Appendix B.

4.5.1.2.6  Vessel Head Penetration Nozzles

Recent NRC generic communications, including NRC Bulletins 2001-01 and 2002-02, have addressed issues related to the cracking of VHP nozzles.  In addition, on February 11, 2003, the NRC issued an order, EA-03-009, “Interim Inspection Requirements for Reactor Pressure Vessel Heads at PWRs,” establishing interim inspection requirements for the reactor vessel heads of PWRs.

DCD Tier 2, Section 4.5.1.1, identifies that a Ni-Cr-Fe alloy will be used in fabricating the reactor VHPs.  Table 5.2-1 identifies this material as Alloy 690.  The specification for Alloy 690 is included in the DCD, which also states that the material will be in a thermally treated condition.  The staff finds the selection of Alloy 690 and its equivalent weld metals, Types 52 and 152, as the preferred nickel-based alloy to be acceptable because of its improved corrosion resistance to the reactor coolant environment.

In RAI 252.001, the staff requested information related to the factors that may contribute to the cracking of VHP nozzles.  In its response dated November 26, 2002, the applicant addressed the differences between the current fleet of PWRs and the AP1000 design in terms of the geometry of the VHP nozzle weld joint, fabrication processes used, access for inspection, and operating conditions, including operating temperature and bypass flow.  The staff discussed additional information related to RAI 252.001 during a telephone conference on February 21, 2003.  By letter dated April 7, 2003, the applicant provided specific information in the areas of weld design, residual stresses, operating temperature, and inspections.

The applicant’s responses indicated that the weld geometry of the VHP nozzle is the same as in currently operating Westinghouse PWRs.  However, the process for installing the nozzles in the AP1000 design is an automatic welding process, which may be supplemented with manual welding processes as necessary.  The current Westinghouse nozzle penetrations use manual welding processes.  The automated welding process provides better control of the J-groove weld than the manual processes.  The applicant indicated that a narrow gap for the J-groove weld edge preparation reduces the residual stresses in the weld by reducing the volume of weld metal deposited.

In addition, the use of spray cooling on the inside surface of the head adapter during J-groove welding improves the stress distribution through the adapter wall thickness.  The stresses are balanced by thermal elongation due to the temperature difference between the inner and outer surfaces, and by the large shrinkage of the outer portion of the weld metal.  This acts to improve residual stresses on the inner surface.

The applicant’s response dated November 26, 2002, discusses the accessibility to the AP1000 penetrations for inspection.  The applicant indicated that the access for inspection is the same as that for current PWRs (i.e., from under the head). The thermal sleeves in currently operating Westinghouse PWRs have been eliminated, increasing the inspection accessibility to the inside diameter surface of the CRDM penetrations. This yields an open access tube. Open access tubes allow for easier insertion of inspection probes/end effectors into the penetrations, and permit the use of multiple sensors for improved inspection effectiveness.

In its November 26, 2002 response, the applicant provided information regarding the AP1000 design of the integrated head insulation package, which is permanently attached to the reactor vessel head. This affects access to the top of the vessel head for direct visual inspection, as compared to the current fleet of PWRs. However, the integrated head package has doors just above the vessel head that will allow inspection access. Vessel head insulation configuration and access ports through the insulation allow for the implementation of visual inspection approaches across the vessel head. On the basis of the information in the applicant's November 26, 2002, response, the staff requested drawings/diagrams of the integrated head package to facilitate review of the inspection access of the AP1000 vessel head and VHP nozzles. The staff also requested information addressing the accessability and examination coverage of the design for bare metal visual examination of 360° around each nozzle. The applicant's response dated April 7, 2003, provides information on the access features related to the vessel head inspection. The access features allow for the use of a remote, mobile visual inspection manipulator, which can inspect 360° around each head penetration (CRDM and instrumentation), and look at the vessel head surface in general. DCD Tier 2, Section 5.3.4.7, "Inservice Surveillance," reflects the information in the applicant's RAI response. Because the access features allow for comprehensive visual inspections of each head penetration, the staff finds this acceptable.

The recent experience with VHP nozzle cracking identifies the need for baseline inspection data to determine if an indication for nondestructive examination (NDE) is service-induced cracking or an artifact from fabrication. The staff requested information on what preservice examinations will be performed on the VHP nozzles. In a letter dated April 7, 2003, the applicant responded that preservice examinations for the closure head will include a baseline top-of-the head visual examination; ultrasonic examinations of the inside diameter surface of each VHP; eddy current examination of the surface of the head penetration welds, the outside diameter surface of the vessel penetrations, and the inside diameter surface of the penetrations; and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations. Any indications exceeding the ASME Code Section III requirements would be removed. DCD Tier 2, Section 5.3.4.7 reflects the information in the applicant's RAI response concerning this issue. Open Item 4.5.1-1 identified that the information on preservice examinations also needs to be addressed by a COL applicant.

By letter dated June 23, 2003, Westinghouse indicated that DCD Tier 2, Section 5.2.6.2, "Plant Specific Inspection Program," includes a commitment that the COL applicant will conduct specific preservice examinations of the reactor vessel closure head. This commitment states that the preservice inspection program will include examinations of the reactor vessel closure head equivalent to those outlined in DCD Tier 2, Section 5.3.4.7. This is COL Action Item 4.5.1-1.

The staff finds that the scope of the baseline/preservice examinations of the VHPs described above is comprehensive with respect to inspection methods and coverage with those methods and, therefore, is acceptable. Based on the acceptability of the preservice examinations and the COL commitment included in DCD Tier 2, Section 5.2.6.2, Open Item 4.5.1-1 is resolved.

The discovery of leaks and nozzle cracking at the Davis-Besse Nuclear Power Station and other operating PWR plants highlights the need for more effective inspections of reactor pressure vessel heads and associated penetration nozzles. The current reactor pressure vessel head inspection requirements include visual examination of the insulated surface or surrounding area for signs of leakage. Such inspections have not been sufficient to reliably detect circumferential cracking of reactor pressure vessel head nozzles and corrosion of the reactor pressure vessel head. Circumferential cracking of reactor pressure vessel head nozzles and corrosion of the reactor pressure vessel head pose a safety concern because of the possibility of a nozzle ejection or LOCA if the conditions are not detected and repaired. NRC Order EA-03-009 establishes interim requirements to ensure that current PWR licensees implement and maintain appropriate measures to inspect and, as necessary, repair reactor pressure vessel heads and associated penetration nozzles. This order addresses requirements for both Alloy 600/82/182 materials in the original heads and Alloy 690/52/152 materials in replacement heads, as well as in the AP1000 reactor pressure vessel head design.

Therefore, the staff finds that the COL applicant should perform analyses and inservice inspections and provide reports and notifications equivalent to those contained in Sections IV.A to IV.F of NRC Order EA-03-009. Open Item 4.5.1-2 identified that these activities should include susceptibility calculations and categorization, visual, surface and volumetric examinations, and preparation of reports and notifications.

By letter dated November 7, 2003, Westinghouse provided a revision of DCD Tier 2, Section 5.2.6.2, indicating that the inservice inspection program will address the susceptibility calculations, inspection of the reactor vessel closure head, and associated reports and notifications, as defined in NRC Order EA-03-009. This is COL Action Item 4.5.1-2. Inclusion of these actions in the inservice inspection program is a satisfactory response to Open Item 4.5.1-2. Therefore, based on DCD Tier 2, Section 5.2.6.2, the staff considers Open Item 4.5.1-2 to be resolved because the COL applicant's inservice inspection program will be consistent with NRC Order EA-03-009. If the staff develops new inspection requirements for these components in the future, the staff will consider the need to backfit these requirements to operating reactors and certified designs, including the AP1000.

Further, the staff requested additional information related to the operating conditions of the reactor vessel head (RVH) and VHP nozzles. In its response dated November 11, 2002, the applicant stated that the operating head temperature is approximately 293.3 °C (560 °F). This temperature is in the colder range of current Westinghouse PWR plants. Operation in the colder range of current Westinghouse PWR plants should reduce susceptibility to SCC of the VHP nozzle welds. Bypass flow is used to cool the vessel head. Similar to current Westinghouse PWR plants, the bypass flow is provided through spray nozzles. The staff requested information related to RAI 252.001, regarding the determination of the head operating temperature, during a telephone conference on February 21, 2003. The applicant's response dated April 7, 2003, provided information on how the RVH temperature was determined. The RVH temperature was calculated by a Westinghouse design code. The applicant indicated that the calculated RVH temperature has been verified by comparing it to operating plant data. The staff finds this response to be acceptable. Because the plant is designed to operate in the colder temperature range of current Westinghouse PWR plants, and because the exact temperature value is not crucial for maintaining structural integrity, the staff finds the applicant's approach of verifying the code-calculated RVH temperature by comparing it to operating plant data to be acceptable.

4.5.1.3 Conclusions

The staff finds the materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls to be acceptable because they satisfy regulatory requirements or positions described above (for RCPB materials), or because they have been demonstrated to be acceptable based on appropriate materials selections and acceptable operating experience (for non-RCPB materials).

The changes made to the AP1000 construction materials and fabrication processes of the VHP nozzles, as compared to current operating reactors, provide for improved resistance to SCC and allow for periodic inspections at least equivalent to those identified in NRC orders issued on February 11, 2003.

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

**4.5.2 Reactor Internal and Core Support Materials**

The staff reviewed DCD Tier 2, Section 4.5.2, "Reactor Internals and Core Support Materials," in accordance with SRP Section 4.5.2, "Reactor Internals and Core Support Materials." The design, fabrication, and testing of the materials used in the reactor internals and core support structures are acceptable if they meet codes and standards commensurate with the safety functions to be performed. This will ensure that the relevant requirements of 10 CFR 50.55a, "Codes and Standards," and GDC 1, are met. The following specific acceptance criteria are necessary to meet these relevant requirements:

- Materials Specifications, Selection, and Heat Treatment

  For core support structures and reactor internals, ASME Code, Section III, Division 1, NG-2000, identifies the permitted materials specification. ASME Code, Section II, NG-2000 describe the specifications for these materials. Additional permitted materials and their applications are identified in ASME Code Cases approved for use, as described in RG 1.84, "Design and Fabrication and Materials Code Case Acceptability, ASME Section III."

  All materials used for reactor internals and core support structures must be compatible with the reactor coolant, as described in ASME Code, Section III, Division 1, Subarticles

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 core welding support structures and reactor internals must conform to ASME Code, Section III, Division 1, NG-4000. The welds must be examined and meet the acceptance criteria as specified in the ASME Code, NG-5000.

- Nondestructive Examination

  This examination shall conform to the requirements of ASME Code, Section III, Division 1, NG-2500. The acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Division 1, NG-5300.

- Austenitic Stainless Steels

  SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," Subsections II.2 and II.4.a, b, d, and e, provide the acceptance criteria for these 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 assuring the integrity of welds in stainless steel components.

- Other Considerations

  These structures could be susceptible to irradiation-assisted stress-corrosion cracking (IASCC). IASCC is an aging mechanism that causes cracking in irradiated stainless steel components. COL applicants should determine whether the components will receive sufficient neutron irradiation to be susceptible to IASCC. If the components are susceptible to IASCC, the COL applicant should propose a program to monitor cracking, or commit to incorporate the results of an industry program that will address this issue. EPRI NP-4767, "Evaluation of BWR Top-Guide Integrity," issued in November 1986, provides criteria regarding the susceptibility of stainless steel components to IASCC.

#### 4.5.2.1 <u>Summary of Technical Information</u>

DCD Tier 2, Sections 4.5.2.1–4.5.2.5, respectively, describe the materials specifications, controls on welding, nondestructive examination, austenitic stainless steel components, and contamination protection and cleaning of austenitic stainless steel.

### 4.5.2.2 Staff Evaluation

The staff divided its evaluation of the reactor internals and core support materials into five sections equivalent to those described in the SRP, including materials specifications, selection and heat treatment, controls on welding, nondestructive examination, fabrication and processing of austenitic stainless steel components, and other considerations.

#### 4.5.2.2.1 Materials Specifications, Selection, and Heat Treatment

DCD Tier 2, Section 4.5.2.1, “Materials Specifications,” indicates the materials and Code cases to be used in fabricating reactor internals and core supports. The major core support materials are SA-182, SA-479, or SA-240 Type 304LN stainless steel. For threaded structural fasteners, the material used is strain-hardened Type 316 stainless steel. The remaining internals parts, which are not fabricated from Type 304LN stainless steel, typically include wear surfaces containing cobalt-free hard faces, Type 316 stainless steel dowel pins, Type 403 stainless steel modified holddown springs, and Type 302 irradiation specimen springs. ASME Code, Section III, Article NG, and Section II, as supplemented by Code Case N-60-5 and N-4-11, specify the core support structure and threaded structural transfer material. Code Case N-60 identifies materials to be used for core support structures. Code Case N-4 specifies material properties and heat treatments to be used when modified forgings of Type 403, which is a martensitic stainless steel, are used for core support structures.

RG 1.84 indicates that Code Case N-60-5 is acceptable, with certain limitations (i.e., the welding of age-hardenable alloy SA-453 Grade 660 and SA-637 Grade 688 should be performed when the material is in the solution-treated condition, and the maximum yield strength of strain-hardened austenitic stainless steel should not exceed 620.5 MPa (90,000 psi)). RG 1.84 also indicates that Code Case N-4-11 is acceptable without limitations. DCD Tier 2, Appendix 1A, “Conformance with Regulatory Guides,” states that the AP1000 will meet the limitations in Regulatory Position C.1 of RG 1.84, which would indicate that the materials in the AP1000 reactor internals and core supports will meet the limitations for Code Case N-60-5. Therefore, the staff finds the materials specifications, selection, and heat treatment for the reactor internals and core support materials to be acceptable because they are consistent with the requirements of ASME Code, Section III, Division I, and they rely on Code cases (with the limitations described above) that are approved in RG 1.84.

In addition, this section of the DCD indicates that the reactor internals and core supports will use low-cobalt or cobalt-free wear-resistant alloys. Section 4.5.1 of this report addresses the qualification of these wear-resistant alloys for use in the reactor coolant system.

#### 4.5.2.2.2 Controls on Welding

The discussions provided regarding controls on welding in DCD Tier 2, Section 5.2.3 are applicable to the welding of reactor internals and core support structures. Therefore, Section 5.2.3 of this report discusses the controls on welding.

#### 4.5.2.2.3 Nondestructive Examination

DCD Tier 2, Section 4.5.2.3, “Nondestructive Examination of Tubular Products and Fittings,” indicates that nondestructive examination of wrought seamless tubular products and fittings will be conducted in accordance with ASME Code, Section III, Article NG-2500, and the acceptance standards meet the requirements of ASME Code, Section III, Article NG-5300. Although the DCD only addresses explicitly the examination of wrought seamless tubular products and fittings, the nondestructive examination and acceptance standards of Article NG-2500 and NG-5300, respectively, are applicable to all core support structural materials. DCD Tier 2, Section 5.2.1.1, “Compliance with 10 CFR 50.55a,” indicates that ASME Code, Section III, including Subsection NG, will be met. This subsection is applicable to the ASME Code-designed core support structural materials, and not just wrought tubular products and fittings. Therefore, the nondestructive examination of the core support structural materials is acceptable.

#### 4.5.2.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

DCD Tier 2, Sections 4.5.2.4, “Fabrication and Processing of Austenitic Stainless Steel Components,” and 4.5.2.5, “Contamination Protection and Cleaning of Austenitic Stainless Steel,” discuss fabrication and processing of austenitic stainless steel components and contamination protection and cleaning of austenitic stainless steel, respectively. Section 5.2.3 of this report also discusses the fabrication and processing of stainless steel components and contamination protection and cleaning of austenitic stainless steel. DCD Tier 2, Appendix 1A, indicates (1) the control of ferrite content in stainless steel welds will conform to RG 1.31, and (2) the control and use of sensitized stainless steel will conform to RG 1.44. RG 1.31 specifies materials, weld test samples, ferrite measurement methods, and a ferrite number range that will prevent microfissuring in stainless steel weld metal. RG 1.44 provides guidance on materials, contaminants, heat treatment, test methods, and water chemistry to limit sensitization of stainless steel, as well as design criteria for the safe operation of nuclear power plants with sensitized stainless steels. The specifications in the DCD regarding the fabrication and processing of austenitic stainless steel components conform to the guidance given in RG 1.44 and RG 1.31 and, therefore, are acceptable.

#### 4.5.2.2.5 Other Considerations

As a result of evaluating the integrity of core support structures for operating plant license renewal, the staff identified that these structures could be susceptible to IASCC and void swelling. In response to RAI 251.011, the applicant indicated that the estimated peak neutron fluence for the AP1000 reactor vessel internals is 9E21 n/cm$^2$. At this neutron fluence, neither IASCC nor void swelling is expected. In addition, the ongoing EPRI/MRP reactor internals program addresses these issues. DCD Tier 2, Section 4.5.2.1, indicates that the COL applicant should address the findings from the EPRI/MRP reactor internals program applicable to the AP1000 reactor internals design. Because neither IASCC nor void swelling is expected to occur, and because the findings of the EPRI/MRP program should be addressed by the COL applicant, the staff concludes that issues related to IASCC and void swelling can be adequately addressed. This is COL Action Item 3.9.2.4-1.

4.5.2.3 Conclusions

As set forth above, with the addition of COL Action Item 3.9.4.2-1, the staff concludes that the reactor internals and core support materials are acceptable and satisfy the relevant requirements of 10 CFR 50.55a and GDC 1. This conclusion is based on the AP1000 reactor vessel internals meeting ASME Code, Section III, Division I, using Code cases that are approved in RG 1.84, using RGs 1.31 and 1.44 for processing of austenitic stainless steel, and incorporating the applicable findings from the EPRI/MRP reactor internals program into the AP1000 reactor internals design.

## 4.6 Functional Design of Reactivity Control Systems

The reactivity control systems for the AP1000 were designed to conform with GDC 26, 27, and 28, in accordance with SRP Section 4.6, "Functional Design of Control Rod Drive System."

Sections 3.9.4 and 7.2 of this report, respectively, discuss the mechanical and electrical aspects of the reactivity control system (i.e., the control rod drive system (CRDS)).

The staff's review of the functional design of the AP1000 reactivity control systems confirmed that the design has the following capabilities to satisfy the various reactivity control conditions for all modes of plant operations:

- the capability to operate in the critical, full-power mode throughout plant life
- the capability to vary power level from full power to hot shutdown and have power distributions within acceptable limits at any power level
- the capability to shut down the reactor to mitigate the effects of postulated events, discussed in Chapter 15 of this report

The reactivity control systems for the facility are the CRD, the reactor trip system, and the passive core cooling system. No credit is taken for the boration capabilities of the chemical and volume control system (CVS).

The CRD contains a magnetically operated jack (magjack). When electrical power is removed from the coils of the magjack, the armature springs automatically disengage the holding latches from the magjack drive shaft, allowing insertion of the control rod and the gray rods by gravity. There are 53 full-strength control rods and 16 GRCAs. The regulating CRD may be used to compensate for changes in reactivity associated with changes in power level and power distribution, variations in moderator temperature, or changes in boron concentration. The gray rods, which have lower worth than the full-strength control rods, control reactivity and axial power shape during power operations.

The CVS is a non-safety-related system designed to control slow or long-term reactivity changes, such as those caused by fuel burnup and variations in coolant temperature, and the xenon concentration. The CVS controls reactivity by adjusting the dissolved boron concentration in the RCS. The boron concentration is adjusted to obtain optimum positioning of the control rods. In addition, boron concentration is used to compensate for reactivity changes during startup, power changes, and shutdown. Also, boron concentration is used to provide shutdown margin throughout the cycle for maintenance and refueling operations, or emergencies. The charging and letdown portions of the CVS control the boric acid concentration in the RCS.

The CVS can be used to maintain reactivity within the TS limit by means of the automatic makeup system. This system replaces minor coolant leakage without significantly changing the boron concentration in the RCS system. Dilution of the RCS boron concentration is necessary to compensate for reactivity losses from fuel depletion. Manual operation of the CVS achieves dilution. DCD Tier 2, Section 9.3.6, “Chemical and Volume Control System,” and Section 9.3.6 of this report discuss the CVS.

The CRDS is the primary shutdown mechanism for normal operation, accidents, and transients. Control rods are inserted automatically in accident and transient conditions to shut down the reactor. In addition, concentrated boric acid solution is injected by the passive core cooling system in the event of a LOCA, steamline break, loss of normal feedwater flow, steam generator tube rupture, or control rod ejection, as described in Section 6.3 of this report. Therefore, the AP1000 design complies with GDC 20, which requires automatic protective systems (1) to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded, and (2) to sense accident conditions and actuate safety-related systems and components.

Functional test programs verify the operability of the CRDS. These tests verify that the trip time achieved by the CRDM meets design specifications. The trip time is confirmed for each CRDM prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the TSs. At every refueling shutdown, the CRDS will be stepped over its entire range of movement and the RCCAs are drop-tested to demonstrate their ability to drop in the required time. The CRDS is designed such that a single failure will not result in loss of the protection system, and the removal of a channel or component from service will not result in a loss of redundancy.

Based on the staff’s review of the design of the reactivity control system information provided in DCD Tier 2, Section 4.6, “Functional Design of Reactivity Control System,” the staff has determined that the CRDS and the passive core cooling system provide the necessary reactivity control and redundancy. These systems also provide a reliable means of protecting the control rod assemblies within the reactor core under conditions of normal plant transients or postulated accident conditions.

Accordingly, the staff finds that the analysis performed by the applicant in support of the reactivity control systems satisfies the acceptance criteria of 10 CFR Part 50, Appendix A, (particularly GDC 26, 27, and 28) and, therefore, is acceptable.