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WCAP-17524-NP-A
APP-GW-GLR-156
Revision 1

May 2015

AP1000 Core Reference Report

Westinghouse

WCAP-17524-NP-A
APP-GW-GLR-156
Revision 1

AP1000 Core Reference Report

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Section A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 19, 2015

Mr. Keith J. Drudy, Manager
PWR Core Methods
Westinghouse Electric Company
1000 Westinghouse Dr. Suite 429
Cranberry Township, PA 16066

SUBJECT: WESTINGHOUSE ELECTRIC COMPANY'S FINAL TOPICAL REPORT SAFETY EVALUATION FOR WCAP-17524, REVISION 1, "AP1000 CORE REFERENCE REPORT"

Dear Mr. Drudy:

The U.S. Nuclear Regulatory Commission staff prepared a final Topical Report Safety Evaluation (TRSE) for WCAP-17524, Revision 1, "AP1000 Core Reference Report," in support of the AP1000 post – licensing activities submitted by Westinghouse Electric Company (WEC).

The staff requests that WEC publish the applicable version(s) of the TRSE listed above within one month of receipt of this letter. The accepted version of the topical report shall incorporate this letter and the enclosed TRSE and add an "-A" (designated accepted) following the report identification number.

The staff has found that WCAP-17524, Revision 1, is acceptable for referencing in licensing applications for AP1000 designed pressurized water reactors to the extent specified and under the limitations delineated in the topical report and in the enclosed final TRSE. The final TRSE defines the basis for our acceptance of the topical report.

If the U.S. Nuclear Regulatory Commission's (NRC) criteria or regulations change, such that the acceptability of the TRSE conclusion is invalidated, WEC and/or the applicant referencing the TRSE will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the TRSE without revision of the respective documentation.

Prior to placing the public version of this document in the public document room the staff requests that WEC perform a final review of the TRSE for proprietary or security-related information not previously identified. If you believe that any additional information meets the criteria, please identify such information line by line and define the basis pursuant to the criteria established in Title 10 of the *Code of Federal Regulations* Part 2, Section 390.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosure, this document is "DECONTROLLED."

If after a 10–day period, you do not request that all or portions of the TRSE be withheld from public disclosure, the TRSE will be made available for public inspection through the NRC Public Document Room and the Publicly Available Records component of NRC's Agencywide Documents Access and Management System and placed on the NRC's public web page for this application.

If you have any questions or comments concerning this matter, please contact me at 301-415-6715 or via e-mail address at Bruce.Bavol@nrc.gov.

Sincerely,

R. B.

Bruce M. Bavol, Project Manager
Licensing Branch 4
Division of New Reactor Licensing
Office of New Reactors

Project No. 0793

Enclosures:
1. Safety Evaluation Report – (Non-Proprietary)
2. Safety Evaluation Report – (Proprietary)

cc: See next page (w/o Enclosure 2)

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WCAP-17524-P, REVISION 1, "AP1000 Core Reference Report"

1.0 Introduction

By letter dated March 6, 2012, Westinghouse Electric Company (Westinghouse) submitted licensee topical report (LTR) WCAP-17524-P, "AP1000 Core Reference Report" (Reference 1) to the staff of the U.S. Nuclear Regulatory Commission (NRC staff) for acceptance review (Reference 16) and approval. Westinghouse submitted the following supplemental information:

- LTR-NRC-12-56, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address the Impact of Thermal Conductivity Degradation on Additional Events," (Reference 2)
- LTR-NRC-12-46, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address Thermal Conductivity Degradation," (Reference 10)
- LTR-NRC-13-26, "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor," (Reference 3)
- LTR-NRC-14-20, "Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models," (Reference 41).

By letter dated March 21, 2014, Westinghouse submitted WCAP-17524-P, Revision 1, "AP1000 Core Reference Report," (Reference 37) to capture updates to the topical report as a result of the review process.

This safety evaluation report (SER) is based on the submitted letter, supplemental information letters, and responses to requests for additional information (RAIs). WCAP-17524-P, Revision 1, is designed to be referenced as part of a post combined operating license (COL) licensing amendment request or in an initial COL application as desired. The subject topical report (Reference 37) describes changes to the fuel system design, which relate to the following review areas of the standard review plan (SRP), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Reference 4):

- Chapter 4.2, "Fuel System Design"
- Chapter 4.3, "Nuclear Design"
- Chapter 4.4, "Thermal and Hydraulic Design"
- Chapter 15, "Transient and Accident Analysis"

Enclosure 1

WCAP-17524-P, Revision 1, describes an enhanced AP1000 fuel and core design that relies on newer components currently in operational use within the existing U.S. pressurized-water reactor (PWR) fleet, a new tungsten gray rod control assembly (GRCA) design based on the approved topical report WCAP-16943-P-A (Reference 7), and a new core loading plan. This SER does not cover any plant designs beyond the AP1000 standard plant.

This SER is divided into seven sections. Section 1 is the introduction, Section 2 presents a summary of applicable regulatory criteria and guidance, Section 3 contains a summary of the information presented in the topical report, and Section 4 contains the technical evaluation of the major components of WCAP-17524-P, Revision 1. These components include the following:

(1) fuel system design

(2) nuclear design

(3) thermal hydraulic design

(4) loss-of-coolant accident (LOCA)

(5) non-LOCA

Section 5 presents the conclusions of this review, Section 6 contains the restrictions and limitations on the use of the core reference report (CRR) topical report, and Section 7 outlines the utilized references.

2.0 <u>Regulatory Criteria</u>

The AP1000 CRR includes modifications to the AP1000 standard plant design that cover a variety of technical review areas. The following sections present the relevant requirements and guidance that the staff utilized for its review.

2.1. Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor Design," requires, in part, that control and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including the effects of anticipated operational occurrences. The SAFDLs associated with the AP1000 plant design are defined by the AP1000 standard plant design (approved as design control document (DCD) Revision 19). The fuel design changes provided in WCAP-17524-P, Revision 1, do not affect the previously reviewed and approved SAFDLs.

Regulations in 10 CFR Part 50, Appendix A, GDC 27, "Combined Reactivity Control Systems Capability," require the following:

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of

reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Regulations in 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," require the following:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

2.2. Relevant Guidance

NUREG-0800 (Reference 4) provides detailed review guidance that the staff finds acceptable in meeting the applicable regulatory requirements. In particular, NUREG-0800 sections that contain guidance relevant to this review are:

1. Section 4.2, "Fuel System Design"
2. Section 4.3, "Nuclear Design"
3. Section 4.4, "Thermal and Hydraulic Design"
4. Section 15, "Transient and Accident Analysis"

3.0 Summary of Technical Information

WCAP-17524-P, Revision 1, provides a detailed description and justification for a new fuel design and advanced first core (AFC) loading pattern to make use of improvements in fuel assembly design, core components and analysis methods when compared with the approved revised AP1000 design certification (DCD Revision 19). The purpose of the CRR topical report is to provide the latest updates to be incorporated into the initial COL applications and in COL license amendment requests, as appropriate. The design changes described in WCAP-17524-P, Revision 1, include:

- implementation of the AFC loading pattern
- incorporation of the enhanced GRCA
- minor modifications to the rod cluster control assembly (RCCA)
- incorporation of robust protective grid (RPG)
- modifications to the grid strap heights of the mid grids and intermediate flow mixing (IFM) grids

In addition to the DCD Chapter 4 design changes listed in the CRR, the applicant re-analyzed DCD Chapter 15 accident analyses based on the new fuel assembly design. In some cases (e.g., reactivity initiated accident analysis), the methodology used in the analysis is based on an updated and previously approved methodology.

3.1. Fuel System Design

The application (WCAP-17524-P, Revision 1) presents evaluations and analyses based on the physical changes to the grid designs in the new AP1000 PWR fuel assembly.

Modifications to the mid and IFM grid designs are analyzed for their impacts on the applicable safety analyses. These changes are addressed in the Thermal Hydraulic Design and Accident Analysis Sections.

The RCCA design has been modified to simplify manufacturability and improve reliability. Specific changes include the use of lower carbon SS-304L, integral design of vanes and fingers with the spider body, chrome plating for wear resistance, and physical changes to reduce the chances of pellet-cladding interaction caused by irradiation swelling.

The previously approved GRCA design is replaced with a tungsten-based design, as previously reviewed and approved in WCAP-16943-P-A (Reference 7). However, similar design changes, as seen for the RCCAs, are included in the GRCA design.

The fuel assembly protective grid design has been changed to RPG, which is identical to the design used in some operating plants. The changes associated with the RPG (page 2-2 of WCAP-17524-P, Revision 1), are designed to mitigate vibration and fatigue induced failures, as well as primary water stress corrosion cracking. The applicant analyzes the effects of the protective grid design change to the GSI-191 in-vessel downstream effects analysis and concludes that no adverse impacts will be caused by the change.

3.2. Nuclear Design

Section 3 and Appendix A Section 4.3 of the CRR presents changes and updates to the nuclear design as compared with Revision 19 of the AP1000 DCD. In Reference 1, Westinghouse proposed changes to the following areas related to the AP1000 nuclear design:

(1) the AFC, or first cycle fuel assembly loading pattern

(2) GRCA design

(3) the discrete burnable absorber design (specified to use wet annular burnable absorber (WABA) rather than Pyrex)

The nuclear design parameters affected include the minimum shutdown margin (SDM) and the enthalpy rise hot channel factor ($F_{\Delta H}$).

The revised nuclear design presented in the application (WCAP-17524-P, Revision 1) requires changes to the technical specifications (TS) and bases. This includes the TS limiting condition for operation (LCO) that describes the axial flux difference limits. This information is contained in Section 6.2 of the application (WCAP-17524-P, Revision 1).

3.3. Thermal Hydraulic Design

Section 4 and Appendix A of Section 4.4 of the application (WCAP-17524-P, Revision 1) address proposed changes to the thermal hydraulic design described in Revision 19 of the AP1000 DCD. As described in WCAP-17524-P, Revision 1, Westinghouse proposed changes to the following areas related to the AP1000 thermal hydraulic design:

- replace the DNBR limits,
- add two CHF correlations for use in certain conditions,
- adjust the new CHF correlations,
- apply the enthalpy rise hot channel factor ($F_{\Delta H}$),
- include the additional departure for nucleate boiling (DNB) correlations, and
- include the fuel assembly design enhancements.

3.4. Non-LOCA

Section 5 of the application (WCAP-17524-P, Revision 1) presents the revised non-LOCA accident analyses. The following changes are addressed:

- revision of Cycle 1 loading pattern
- implementation of WABA
- implementation of enhanced GRCA
- implementation of the RPG
- increase in $F_{\Delta H}$ limit

- increase to mid-grid and IFM height
- addition of RCCA drop time input
- incorporation of design changes to the AP1000 Revision 19 DCD as part of the finalization

All of the non-LOCA analyses of Chapter 15 have been revised for the Advanced First Core design changes.

Section 15 in Appendix B of the application (WCAP-17524-P, Revision 1) addresses proposed changes and updates to the AP1000 Revision 19 DCD accident analyses. The non-LOCA portion of the accident analyses is presented in Section 15, excluding Section 15.6.5. The general change items are described in the "Change Road Map" that precedes Section 15.0 in Appendix B of the application (WCAP-17524-P, Revision 1). The editorial changes include further clarification on how a loss of alternating current (ac) power is addressed for each event and the application of the protection system delay between reactor trip and turbine trip.

The system transient analyses are performed using the LOFTRAN computer code. The original and previous system analyses used a modified version of LOFTRAN referred to as LOFTRAN-AP (see Reference 9). LOFTRAN was modified to include the passive safety system of the AP1000 design to create LOFTRAN-AP. The passive plant changes have been combined with the latest version of LOFTRAN to create a single version of the code for use in AP1000 analyses and operating plant analyses. This latest version of LOFTRAN has been used for the Advanced First Core analyses.

The applicant added Section 15.0.11.6 to discuss the advanced nodal code (ANC) that is used in the analysis of the rod ejection event. The nuclear design parameters affected include the minimum SDM and the enthalpy rise hot channel factor ($F_{\Delta H}$).

3.5. LOCA

Section 15.6.5 in Appendix B of WCAP-17524-P, Revision 1, presents the changes and updates to the limiting small break loss-of-coolant accident (SBLOCA) and the limiting large break loss-of-coolant accident (LBLOCA) analysis described in Revision 19 of the AP1000 DCD. Other supporting documentation includes Reference 2 and Reference 10.

In WCAP-17524-P, Revision 1, the subsections describing SBLOCA and LBLOCA analyses remain largely unchanged outside of updates, including minor corrections, additional descriptions, minor update of analysis parameters, and a minor update of analysis results. The analysis conclusions were unaffected. All changes are described in the "Change Road Map," which precedes Section 15.6 in Appendix B of WCAP-17524-P, Revision 1. The only nuclear design parameter that was updated in the LOCA analyses was the enthalpy rise hot channel factor ($F_{\Delta H}$).

Reference 2 provides an additional discussion of the SBLOCA analyses, and Reference 10 discusses major changes to the LBLOCA analyses. These changes address concerns in Reference 11 about the potential impact on calculated peak cladding temperature (PCT) caused

by fuel thermal conductivity degradation (TCD) with burnup in realistic emergency core cooling system (ECCS) evaluation models. Since all major changes are a result of accounting for fuel TCD, with the exception of the SBLOCA analysis update for a change in the limiting single failure, the technical evaluation is focused on the supplemental information contained in Reference 2 and Reference 10.

4.0 <u>Technical Evaluation</u>

4.1. Fuel Mechanical Design

The topical report outlines the physical fuel assembly design changes and evaluates the resultant changes to the applicable safety analyses. The applicant left out of the report analysis topics it identified as obviously unaffected. The staff compared the review topics identified in SRP Section 4.2 to the topical report and notes that the technical areas of fuel system damage and fuel rod failure are not directly analyzed.

The staff reviewed the changes to the AP1000 fuel assembly design identified in WCAP-17524-P, Revision 1. During the review of the documentation, the staff found that:

- The fuel assembly design changes (RPG and increased IFM/mid-grid heights) would not increase the stress, strain, or loading limits on the structural members compared to what has already been reviewed and approved by the staff previously.

- The contact points of the grids use the same design as the previously approved spacer grids and IFM grids.

- No new materials are introduced for any of the design changes in the fuel assemblies.

- The design changes would not lead to increased fatigue cycles or changes in irradiation growth predictions.

Based on the above observations, the staff concludes that the fuel system damage analysis previously reviewed and approved in the AP1000 DCD Revision 19 remains applicable and no new analysis is necessary.

The staff reviewed the fuel assembly design changes following the guidance related to fuel rod failure found in SRP Section 4.2 and found that:

- The hydriding analysis previously approved would not be affected because the materials and processing have not been changed by the new design.

- Since the design and material for the fuel cladding and fuel pellet are not changed by the new fuel design presented in WCAP-17524-P, Revision 1, the respective analyses are not impacted (hydriding, collapse, overheating, and pellet/cladding interaction (PCI) and pellet/cladding mechanical interaction (PCMI)).

Based on the above observations, the staff concludes that the fuel rod failure analysis previously reviewed and approved in the AP1000 DCD Revision 19 remains applicable, and no new analysis is necessary.

4.1.1. GSI-191

The applicant provides an analysis of the protective grid design change on GSI-191 in WCAP-17524-P, Revision 1. The analysis relies on a comparison of the protective grid design as presented in the topical report compared to the GSI-191 testing performed in support of the AP1000 DCD Revision 19 and the PWR Owner's Group (PWROG) submittals. The testing programs covered a variety of operating parameters, break locations, and debris conditions. As a result of these test programs and a qualitative analysis of the changes to the fuel design, the applicant concludes that the majority of the pressure drop (ΔP) occurs across the bottom nozzle and protective grid of the fuel assembly.

In RAI CRR-013, the staff requested a description of all fuel design changes beyond the protective grid change presented in Section 2.6 of WCAP-17524-P, Revision 1. In response (Reference 5), Westinghouse described changes that included the mid grids and IFM grids previously approved in the AP1000 DCD Revision 19. These changes consisted of increasing the height of the grid straps to improve the grid to rod fretting margin and to improve the seismic performance (mid grids only). The overall design of the grids, including the springs, remains the same. In Section 2.6 of WCAP-17524-P, Revision 1, the applicant states that the majority of the debris collects at the bottom nozzle and protective grid of the assembly and, therefore, the other components such as mid-grids and IFMs have a negligible impact on the GSI-191 testing results.

During an audit (Reference 14) of the Westinghouse GSI-191 testing program results for both the AP1000 (Reference 12) and PWROG (Reference 8), the staff reviewed photos of the resultant debris distribution, plots of the ΔP measurements during the test, and the test assembly configurations. The staff observed that the debris accumulated mostly on the bottom nozzle and the protective grid, with relatively minimal debris accumulating on the mid-grids and IFMs. The ΔP measurements recorded for the upper versus lower portions of the assembly confirmed this finding. These results are consistent with the statements provided in WCAP-17524-P, Revision 1.

Based on photos from previous test results provided during an audit, the staff notes that the debris depositing on the mid grids and IFM grids deposits on the leading edge, with no appreciable accumulation evident inside the grid itself. The response to RAI CRR-013 (Reference 5) also includes discussion regarding the pressure drop across the grid region compared with the overall assembly pressure drop, which demonstrates a relative lack of importance. Based on pressure drop comparison and the debris accumulation results, the staff agrees that the change in design to increase the grid heights of the mid grids and IFMs is acceptable and does not require additional testing.

In RAI CRR-014, the staff requested a comparison of the fuel design used in WCAP-17524-P, Revision 1, with the fuel designs used in the AP1000 DCD Revision 19, the AP1000 DCD Revision 19 GSI-191 testing (Reference 12), and the PWROG testing (Reference 8). In response (Reference 5), Westinghouse provided the dimensions for the components that make up the fuel assembly design for comparison. The response to RAI CRR-014, Table 1, shows

that most of the fuel components have been used to support testing for either the AP1000 DCD Revision 19 fuel design or PWROG. The only components with a new design (as compared with operating plant fuel assembly designs or the previous AP1000 fuel assembly design) are the mid grids and IFMs discussed previously. The components that lead to the largest pressure drop in the presence of debris-laden coolant are the bottom nozzle, protective grid, and bottom grid. These components are identical in the fuel design described by WCAP-17524-P, Revision 1, and the fuel design tested as part of the PWROG testing (Reference 8). Figure 1 of RAI response CRR-013 (Reference 5) shows an example pressure drop measured during the PWROG cold leg break testing. The results demonstrate the relative importance between the pressure drop measured across the bottom nozzle and RPG versus the total measured assembly pressure drop.

Based on the information provided, the staff concludes that the testing performed on a similar fuel design as part of the PWROG testing program (Reference 8) contains identical key components, including the bottom nozzle, RPG, and bottom grid. These components are the locations at which debris beds form during GSI-191 testing leading to the largest contribution of the total pressure drop. The testing demonstrates that sufficient coolant would be available for long-term cooling, even in the presence of debris. Therefore, the staff concludes that the fuel design presented in WCAP-17524-P, Revision 1, would not lead to failures during long-term cooling as defined by GSI-191.

4.1.2. Fuel Seismic Response

In the staff's acceptance letter (Reference 16) on the review of Reference 1, the staff noted that additional information would be needed about the fuel seismic response at end of life (EOL) conditions because of reduced assembly stiffness caused by grid spring relaxation during irradiation. During the review process, the staff sent three RAIs related to fuel seismic response (CRR-024, CRR-025, and CRR-027). The responses to these RAIs were included in a supplemental information package submitted on April 30, 2013 (Reference 3).

In the supplemental information package for EOL seismic and LOCA calculations, the applicant determined EOL grid strength by relaxing the grid springs on sample grids and then performing impact tests following the same methodology used for beginning of life (BOL) AP1000 grid impact tests. The amount of relaxation chosen for the EOL condition tests was based on post irradiation experimentation (PIE) data from assemblies with burnups comparable to the AP1000 EOL burnups and with same-sized pins and RFA style grids. The applicant chose a gap size that exceeded the 95-percent confidence level of the mean based on these PIE data. This methodology for determining the amount of spring relaxation for EOL testing generally follows the guidance provided in SRP Section 4.2 Appendix A for BOL grids allowable crush load (P_{crit}). By basing the gapped cell methodology for EOL condition grids on a related approved methodology, the staff finds the approach acceptable.

Using both BOL grids and EOL surrogate grids, Westinghouse performed a total of 60 tests to calculate P_{crit}. Westinghouse identified two different collision mechanisms based on core geometry. These were identified as grid-to-grid and grid-to-core shroud collisions. Westinghouse modeled these collision mechanisms in testing by varying the back plate that the grids rested against in the test rig. For grid-to-grid collisions, a flat back plate was used, and for grid-to-core shroud collisions, a smaller back plate with one rounded edge was used. This modified back plate represented core locations on the periphery where the edge of a core

shroud projected partway past an assembly. During an audit on April 16, 2013 (Reference 15), the staff reviewed core drawings, the back plates used, and the testing setup. Based on the information provided, and confirmed during the audit of supporting materials, the staff finds that the methodology for determining P_{crit} follows the guidance provided in SRP Section 4.2 Appendix A and is acceptable.

In Section 3.0 of Reference 3, Westinghouse states that the modal frequencies for EOL conditions are lower than for BOL conditions. These lower frequencies result in higher grid impact loads. Westinghouse performed vibration tests using the same methodology as previously used in BOL tests, with the exception of using gapped grid cells for the EOL surrogate assembly. During an audit on April 16, 2013 (Reference 15), the staff saw the assembly used to generate the fuel assembly modal frequencies listed in Table 2 of Reference 3 and confirmed that the assembly was based on the fuel design described by the topical report. The staff accepts the modal frequencies listed in Table 2 of Reference 3 as representative of the AP1000 CRR fuel design based on the use of previously approved testing methodology, full-sized assembly based on the new fuel design, and appropriately gapped grid cells to simulate EOL grids for the EOL assembly testing.

Section 5.0 of Reference 3 discusses the methodology used to take credit for damping of the fuel assembly vibration response. The core dynamic analysis for BOL conditions conservatively assumes still water and a damping ratio of 20 percent. Figure 4 of Reference 3 presents the data used to support the damping ratio. This follows the methodology used in previously approved Westinghouse submittals and the staff finds it is acceptable for the new fuel design in WCAP-17524-P, Revision 1.

The core dynamic analysis for EOL conditions takes credit for vibrational damping caused by the flowing water throughout the fuel assembly. Figure 5 of Reference 3 presents fuel assembly damping data in flowing and still water for various flow rates and temperatures. Figure 7 summarizes the statistical evaluation of the damping data, and Figure 8 presents the damping design curve. The damping curve includes adjustments for measurement uncertainty, temperature effects, and reserved margin.

Specific testing of the fuel design presented in the topical report did not generate the damping data presented in Reference 3. Instead, this data is based on other Westinghouse fuel design data and publicly available data for fuel designs from other vendors. The data presented in Table 3 demonstrates that the amount of damping is not strongly design dependent. The staff agrees that this demonstrates that flow induced vibration damping is not strongly dependent upon the grids, which are the primary concern for fuel failures during seismic events.

The damping curve presented in Figure 8 provides a damping coefficient as a function of flow velocity based on the data and uncertainties, as previously discussed. By crediting flow-induced damping in calculating the grid impact forces for EOL conditions, it is necessary to account for conditions in which the flow rate (and also the flow-induced damping credit) could be reduced. Westinghouse identified a pump coastdown event resulting from a loss of offsite power during a seismic event as leading to the most limiting conditions from the standpoint of flow-induced vibration damping. The methodology and analysis for the pump coastdown event is listed in Section 5.0 of Reference 3. By calculating the lowest pump flow rate occurring between the event initiation and RCCA insertion to the dashpot region, and using this flow rate as the basis for the damping coefficients used in the EOL grid analysis, the staff finds that the

event is conservatively modeled and follows the guidelines found in SRP Section 4.2 Appendix A. Therefore, the staff finds this acceptable.

To confirm that the grids are the most limiting fuel assembly component, Westinghouse additionally analyzed the fuel rods to ensure that fuel rod fragmentation does not occur. Section 6.0 of Reference 3 lists the results, which demonstrate significant margin to fuel rod fragmentation.

In summary, the staff finds that the applicant follows regulatory guidance, where applicable, provides conservative methodologies when regulatory guidance is not available, and demonstrates that the AP1000 grid crush strength is not exceeded for BOL or EOL condition.

4.2. Nuclear Design

WCAP-17524-P, Revision 1, Appendix A, Section 4.3, "Nuclear Design," presents the changes to the design bases for the AP1000 nuclear design. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including anticipated operational occurrences (AOOs), and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

4.2.1. Reactivity Control and Shutdown Margin

A significant change highlighted in WCAP-17524-P, Revision 1, is in the GRCA design. This is important to the nuclear design because of the change in the absorber material from Ag-In-Cd to tungsten. A reference to WCAP-16943-P-A (Reference 7) was added to support the use of the enhanced GRCA rodlet design as part of the AFC nuclear design. The NRC approved the use of this GRCA design for the AP1000, as stated in the conclusions of the staff's SER. The tungsten GRCAs are not safety-related components and do not adversely affect SAFDLs, as discussed in the staff's SER.

The staff also observed in Table 4.3-1 of WCAP-17524-P, Revision 1, a first core design change from Pyrex burnable absorber to WABA. The NRC previously approved WABA for use in Revision 19 of the DCD and also for general use in PWR cores in the staff's SER for Reference 17. Therefore, the staff finds this to be acceptable.

In Section 4.3.2.4, "Control Requirements," (WCAP-17524-P, Revision 1, Appendix A) two paragraphs were added explaining that the GRCAs are not credited in the calculation of available shutdown margin, despite being released into the core upon reactor trip. The credit of only the RCCAs allows the continued use of the 7-percent uncertainty allowance for the credited rod worth. It is further stated that the GRCAs may be credited after various conditions are met after the reactor is shut down. Since the GRCAs are not relied upon to safely shut down the reactor and, therefore, not credited in the available shutdown margin calculation, the staff finds this to be conservative and acceptable. Since SDM requirements remain to be met, GDC 27 is satisfied.

In Section 4.3.2.4.5, "Rod Insertion Allowance," (WCAP-17524-P, Revision 1, Appendix A), the paragraph was revised to emphasize how the pre-trip control rod insertion can affect the available trip rod worth and how the GRCA positions also can have a small effect on the worth

of the RCCAs. It explains that the most limiting allowed control rod insertion is implicitly included in the calculated trip rod worth and total power defect values reported in Table 4.3-3 (WCAP-17524-P, Revision 1, Appendix A).

In reviewing the changes to Table 4.3-3 in WCAP-17524-P, Revision 1, Appendix A, the staff observed that, despite there being less of a power defect to overcome, the rods are worth less. This translates into less SDM available, as seen in Table 4.3-3. However, there is still significant margin to the requirement. Therefore, the staff finds this acceptable, as GDC 27 continues to be met. It is also noted again that the GRCAs are not credited in the SDM calculation, which is a conservative assumption.

Section 4.3.2.4.16, "Load Follow Control and Xenon Control," (WCAP-17524-P, Revision 1, Appendix A) was modified to elaborate on the use of soluble boron above 30 percent rated thermal power (RTP) to accommodate large load changes. It is implied that the automatic mechanical shim (MSHIM) typically operates without the need for boron control, but in this case, it is explained that soluble boron may optionally be used during MSHIM operation to maintain optimum GRCA positioning and to minimize use of the more reactive RCCAs. The staff finds the allowance of soluble boron use during load follow control maneuvers acceptable since this is a change that was implemented to allow more operational flexibility.

4.2.2. Reactivity Coefficients

The staff reviewed changes to the reactivity coefficients as presented in Table 4.3-2 of WCAP-17524-P, Revision 1. Table 4.3-2 shows only slight changes to the typical best estimate reactivity coefficients (i.e., they are more negative, which is conservative). The design limits for the reactivity coefficients remain unchanged.

As part of an audit (Reference 14), the staff reviewed various rod ejection analyses that form the conclusions drawn in DCD Section 15.4.8 (see Section 4.6.26 of this report) discussing rod ejection accident analyses in more detail. In all of the analyses reviewed, a multiplier was used to adjust the effect of the Doppler feedback, which directly impacts the transient response predicted by ANC, the three-dimensional (3-D) spatial kinetics code used in all of the rod ejection simulations. Westinghouse explained that this multiplier operates by conservatively reducing the change in the fast fuel absorption cross section due to changes in the effective fuel temperature, which in turn conservatively increases the magnitude of the power spike seen after the simulated rod ejection. Westinghouse explained that it builds a minimum [ ] percent reserve margin into the Doppler feedback for the rod ejection transient consistent with the approved methodology in Reference 27. This can be compared to the Doppler feedback coefficient by reducing the change in the fast fuel absorption cross section due to the change in the effective fuel temperature by [ ] percent or by multiplying by [ ]. The analysis documented in Reference 37 used a value of [ ] which is a modification to the Doppler coefficient which continues to meet the licensed methodology. The staff also concludes that 10 CFR Part 50 GDC 11, "Reactor Inherent Protection," remains satisfied.

4.2.3. Peaking Factors

Table 4.3-2 of WCAP-17524-P, Revision 1, Appendix A, shows no change to the heat flux hot channel factor (F_Q); however, the staff observed an increase to all of the enthalpy rise hot channel factors ($F_{\Delta H}$) in the table. Note (g) states that the hot channel factors presented in

Table 4.3-2 are for fully inserted rods, which is a conservative assumption since this maximizes peaking.

DCD Table 4.4-1 (Sheet 1 of 2) was updated to reflect the reduction of the two minimum departure from nucleate boiling ratio (DNBR) values calculated at nominal conditions; these values remain well above the minimum DNBR limit prescribed for design transients.
As stated in Section 3.5 and Section 5.2 of WCAP-17524-P, Revision 1, the revised Chapter 15 transient analyses were performed assuming the increased nuclear design $F_{\Delta H}$ limit, but this is with respect to the previous nuclear design value and not the value assumed in the Chapter 15 transient analysis for DCD Revision 19, which included additional margin. A review of the revised Chapter 15 subsections given in Appendix B of WCAP-17524-P, Revision 1, found that an "AP1000 Core Reference Report DCD (Revision 19) Change Road Map" before each subsection summarizes all of the changes for the respective subsections. The report notes that the nuclear design $F_{\Delta H}$ limit increase is documented as a change for each transient scenario in which $F_{\Delta H}$ is relevant. The revised Table 15.6.5-4, "Major Plant Parameter Assumptions Used in the Best-Estimate Large-Break LOCA Analysis," documents further verification of the nuclear design $F_{\Delta H}$ limit increase in the Chapter 15 transients. However, it is noted that the previously assumed $F_{\Delta H}$ limit for the LBLOCA analysis was actually higher in the DCD Revision 19 analysis; the update to the limit is now consistent with the design value based on the revised nuclear design and is appropriate for use in the Chapter 15 transient analyses.

4.2.4. Vessel Fluence

WCAP-17524-P, Revision 1, Appendix A, Section 4.3.2.8, "Vessel Irradiation," describes the methods and analyses used to predict reactor pressure vessel (RPV) irradiation, and refers to Section 5.3 of the DCD for a more detailed discussion that includes the irradiation surveillance program. The staff identified a potential impact to the projected EOL RPV fluence calculation with the change to the AFC and subsequent core loading patterns. During an audit (Reference 14), the various AFC relative assembly powers in WCAP-17524-P, Revision 1, (both for the first cycle and for the equilibrium cycle) were compared to the original DCD relative assembly powers for the equilibrium core. The DCD relative assembly powers for the equilibrium core are higher than the AFC low leakage pattern (L3P) design on the periphery, which indicates that the vessel fluence will be lower for the AFC loading pattern since these peripheral assemblies carry the most weight with respect to fluence accumulation within the reactor pressure vessel. This follows with the concept of the L3P, which is used to enhance neutron economy and minimize neutron leakage from the core. Furthermore, the L3P is designed to look like the equilibrium cycle. Therefore, it is expected that the equilibrium cycle for the updated core design loading will produce similar relative assembly powers.

Consequently, Revision 19 of the DCD contains a more conservative estimate for vessel fluence compared to the WCAP-17524-P, Revision 1, changes. Therefore, the design basis nil-ductility transition temperature that factors into the RT_{NDT} limit in Table 5.3-3 of the DCD remains conservative, along with the pressure and temperature limit curves and the low temperature overpressure protection system setpoints referenced in the Chapter 16 technical specifications. Furthermore, the requirements of 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," remain satisfied.

4.2.5. Spent Fuel Pool Criticality

During the Phase 1 audit, the NRC held discussions with Westinghouse related to the impact of the AFC and subsequent core reload designs on the SFP criticality safety analysis performed in Section 9.1.1 of the DCD. The staff's main concern was with the fuel assembly limiting depletion characteristics, including the impact on the limiting axial burnup profile assumption. To capture the information discussed, the staff issued RAI CRR-001 asking Westinghouse to discuss how the change to the AFC and subsequent core designs—in addition to how the designs are managed from cycle to cycle—impact the assumptions and conclusions made in Reference 18. In the response to RAI CRR-001 (Reference 6), Westinghouse indicated that it took into account the AFC fuel design characteristics, burnable absorber loading and control rod management schemes, and corresponding axial burnup profile shapes in the current analysis. As a result, the conclusions of the SFP criticality safety analysis are unchanged, and the criteria in 10 CFR 50.68, "Criticality accident requirements," remain satisfied. Westinghouse also noted that a minor discrepancy was identified and entered into the Westinghouse Corrective Action Process. The discrepancy was with the diameter of the GRCA absorber assumed in the criticality analysis, which is inconsistent with the final design of the GRCAs used in the newer AFC design. The assumed diameter is smaller than the actual absorber diameter, which would have a nonconservative impact on the criticality analysis. In the revised analysis performed by Westinghouse, as indicated by the response to RAI CRR-001 (Reference 6), it is demonstrated that the impact of the revised GRCA rodlet diameter on the spent fuel pool criticality analysis is negligible (Reference 28). The staff agrees that the design changes specified in WCAP-17524-P, Revision 1, continue to meet the regulatory requirements regarding spent fuel pool criticality by following the guidance provided in SRP 9.1.1.

The staff also observed that the reference to APP-GW-GLR-029P, Revision 3, "AP1000 SFP Storage Racks Criticality Analysis," was deleted and replaced by a reference to the CASMO-4 user's manual (new Reference 53 in Section 4.3). It is explained that this was done because only the criticality methods are described in Section 4.3.2.6.1 and not the analysis. Section 9.1.6 describes the analysis results. Since this reference change did not introduce any new methods and is just for clarification, the staff finds this to be acceptable.

4.2.6. Technical Specifications

Section 4.3.2.2.6, "Limiting Power Distributions," of WCAP-17524-P, Revision 1, Appendix A, mentions relaxed axial offset control (RAOC), which defines the envelope for allowable axial offset during reactor operation. Westinghouse identified conforming changes to LCO 3.2.1 and LCO 3.2.3 in the TS and corresponding TS bases that result from a change from RAOC to constant axial offset control (CAOC). CAOC is operationally more restrictive than RAOC and, consequently, results in added conservatism or margin with respect to minimizing xenon oscillations and subsequent effects on power distribution. Therefore, the staff finds the use of CAOC instead of RAOC to be acceptable and 10 CFR Part 50, GDC 12, "Suppression of Reactor Power Oscillations," remains satisfied. Reference 5 in the TS bases for LCO 3.2.1, which was previously reviewed and approved by the NRC, discusses the differences between RAOC and CAOC.

4.2.7. Codes and Methods

The CRR includes various updates to the referenced codes and methodologies used in the nuclear design. Reference 19 was added to allow use of PARAGON instead of PHOENIX-P to generate all of the cross sections used in the nuclear design. In Section 4.0, entitled, "Conditions and Limitations," of the staff's SER for PARAGON, Item 1 states that "the PARAGON code can be used as a replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies." Since the staff previously approved PHOENIX-P for use in nuclear design analyses, the staff finds the use of PARAGON as an alternative to PHOENIX-P to be acceptable per the PARAGON SER conclusions (Reference 20).

The addition of Reference 71 provides additional information describing the cross section data used in the PHOENIX-P code. Since this is supplemental information, the staff finds the addition of the reference is acceptable.

A reference for WCAP-16045-P-A, "Qualification of the NEXUS Nuclear Data Methodology" (Reference 22), was added to allow use of the NEXUS nuclear data methodology, which passes nuclear data to the 3-D ANC. The staff approved this methodology for performing calculations on uranium-fueled PWRs, as documented in the corresponding SER conclusions (Reference 21). Consequently, the staff finds the use of this code is acceptable for applicable nuclear design analyses for the AP1000 design.

WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology" (Reference 23), was added to the references to reflect an updated methodology to be used along with the ANC. The staff's SER states that this methodology is acceptable as long as the nuclear data generated as input to ANC originates from the PARAGON and NEXUS code systems. During an audit (Reference 14), the staff asked if there were any instances in which PHOENIX-P is used instead of PARAGON to generate data for ANC since the SER conclusions for the pin power recovery methodology only give approval for use of the methodology with the PARAGON/NEXUS/ANC system. Westinghouse stated that 1-D parameters, such as leakage factors, are input into the ANC directly from PHOENIX-P. It explained that instead of re-generating these 1-D data using PARAGON, the data from PHOENIX-P was used to avoid additional calculations. Westinghouse stated that it plans to move solely to PARAGON for the nuclear design at a later date. It further explained that the transport methods used to generate this data are different between PHOENIX-P and PARAGON (i.e., discrete ordinates vs. collision probability), so the data generated will be different, but are expected to be consistent. With the approval of the pin power recovery methodology described in Reference 23, the limitations and conditions in the NRC SER apply only to the use of ANC for rodded and unrodded pin power predictions. RAI CRR-002 was issued to confirm that PHOENIX-P is being used appropriately, and in the response (Reference 5), Westinghouse stated that the use of PHOENIX-P data as input to ANC is only used for nonfuel radial reflector regions located outside of the active fuel in which no pin power recovery calculations are performed. Therefore, the staff finds the limited use of PHOENIX-P for the AP1000, as described above, continues to meet the regulations identified in Section 4.3 of SRP/NUREG-0800 and is therefore acceptable.

4.3. Thermal Hydraulic Design

In the CRR, Appendix A, Section 4.4, "Thermal and Hydraulic Design," the applicant proposed changes to the AP1000 thermal hydraulic design as compared with Revision 19 of the DCD. The principal function of the thermal hydraulic design is to provide adequate heat removal capability to ensure that the SAFDLs will not be exceeded during normal operation and transients. The applicant proposed the following five changes related to the thermal hydraulic design:

(1) replace the DNBR limits of 1.22 and 1.21 with a single conservative limit of 1.25

(2) add the ABB-NV and WLOP (ABB-NV modified for low pressure) correlations for certain conditions

(3) adjust F factors used for the WRB-2M, ABB-NV, and WLOP CHF correlations,

(4) change the enthalpy rise hot channel factor ($F_{\Delta H}$) limit from 1.65 to 1.72 and the corresponding factor for the Revised Thermal Design Procedure (RTDP) from 1.590 to 1.654

(5) revise the description to clarify the minimum incore nuclear detector instrumentation system requirements.

In addition, the thermal conductivity of the uranium dioxide fuel described in Section 4.4.2.11.1 is dependent on the fuel temperature without considering the degradation of thermal conductivity from the burnup effect. The staff included a review of the uranium dioxide (UO_2) fuel thermal conductivity degradation TCD impact on heat transfer properties with increasing fuel exposure.

In Section 4.4.1.1.2, the RTDP design limit DNBR values of 1.25 for the typical cell and 1.25 for the thimble cell replace the DNBR limits of 1.22 and 1.21, respectively. The DNBR limits were for those transients that use the VIPRE-01 computer program in Subsection 4.4.4.5.2 and the WRB-2M correlation in Subsection 4.4.2.2. These values are also changed in Table 4.4-1 and may be revised when plant-specific uncertainties are available. The staff finds this change to the higher DNBR limit of 1.25 to be conservative and, therefore, acceptable.

The thermal hydraulic CHF calculation of the AP1000 fuel uses the WRB-2M CHF correlation. The WRB-2M correlation applies to the Robust Fuel Assemblies planned to be used in the AP1000 core. The applicable range of parameters for the WRB-2M correlation is described in Section 4.4.2.2.1 of the CRR. Whenever the condition exists in which the WRB-2M is not applicable, other applicable CHF correlations are used. In the heated region below the first mixing vane grid, the W-3 CHF correlation, which does not take credit for mixing vane grids, was used to calculate DBNR values. In addition, the W-3 correlation was applied in the analysis of accident conditions when the system pressure was below the range of the WRB-2M correlation.

In Section 4.4.2.2.1, the applicant added two new CHF correlations (ABB-NV and WLOP) that augmented the W-3 correlation as a configuration enhancement to the DNBR calculation. Both ABB-NV and WLOP CHF correlations were developed based on rod bundle CHF test data and

provide more accurate prediction of the DNBR margin than the W-3 correlation, which was developed with tube data. The uses of these two correlations have been approved by the NRC, as described in addendums to WCAP-14565-P-A. Addendum 1-A, "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code" (Reference 32), addresses the ABB-NV CHF correlation, and Addendum 2, "Addendum 2 to WCAP-14565-P-A Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications" (Reference 33) addresses the extended application of CHF ABB-NV correlations and the WLOP CHF correlation for PWR low pressure applications. For instance, the ABB-NV correlation is used when conditions, in the fuel region below the first mixing vane grid, cause the DNB to become limiting with a severely bottom-skewed axial power distribution, such as rod withdrawal from a subcritical accident; whereas, the WLOP correlation was added for the analysis of Hot Zero Power Steamline Break event in which low flow and low pressure conditions exist. The replacement of the W-3 correlation with the ABB-NV and WLOP correlations are for consistency with the AP1000 fuel design enhancements and improvement in the DNBR calculation. Since the use of the ABB-NV and WLOP correlations have been approved by the NRC, the staff concludes that the changes are acceptable.

Section 4.4.2.2.2 describes the definition of DNBR as the predicted critical or DNB heat flux divided by the actual local heat flux. Since the W-3 CHF correlation is augmented by the ABB-NV and WLOP correlations, the description related to the W-3 correlation is deleted, which the staff finds acceptable. The predicted DNB heat flux is calculated by the equivalent uniform DNB heat flux predicted by a DNB correlation divided by the flux shape F factor to account for the nonuniform axial heat flux distribution. Section 4.4.2.2.2 states that adjusted F factors are used for the WRB-2M, ABB-NV, and WLOP correlations. An adjustment factor is applied to the F-factor for the calculation of the DNB heat flux with nonuniform axial power shape. The staff had evaluated the adjustment factor used for the WRB-2M correlation. As stated in a letter of February 3, 2006 (Reference 34), the staff concluded that the effect of the adjustment factor is to lower the DNBR, which provides more conservatism in the plant reload evaluations than the current correlation. The DNB margin retained in the safety analysis DNBR limit remains unaffected. Therefore, the staff finds the use of the adjustment factor is acceptable.

The WRB-2M correlation, documented in WCAP-15025-P-A, was developed for the modified low pressure drop (MLPD) mid-grid design and the modified intermediate flow mixer (MIFM) grid design. The AP1000 mid-grid and IFM grid designs have been enhanced relative to the MLPD and MIFM grid designs by increasing the grid strap heights (by about 50 percent) to improve fretting wear resistance of the AP1000 fuel assembly. Since the grid enhancement changes are limited to the increase of the grid strap heights, the applicant contends that the DNB correlations remain applicable to the AP1000 fuel with enhanced grid design. In Section 4.5 of the AP1000 CRR, the applicant states that the AP1000 mid and IFM grids have the same DNB performance as the MLPD and MIFM grids because of the following:

- [ ] (MLPD grid height is 2.25 inches (5.7 centimeters) and MIFM height is 0.66 inches (1.7 centimeters)), [ ].
- The AP1000 grids maintain the same mixing vane design and mixing vane pattern as the MLPD and MIFM grids, which is the dominant DNB parameter.

- The important DNB performance parameters in the Westinghouse Fuel Criteria Evaluation Process focus only on lateral grid parameters (not axial parameters like grid height).

The applicant therefore concludes that the DNB correlations are applicable to the AP1000 fuel having the enhanced mid grid and IFM grid design. In the staff audit of WCAP-17524-P (Reference 37), the applicant provided a figure of the measured-to-predicted CHF ratios versus grid strap heights to demonstrate that grid strap height does not have an effect on DNB performance. The figure was based on the DNB tests for the WNG-1 DNB correlation, which was described in topical report WCAP-16766-P-A (Reference 42) and has been approved by the NRC. The DNB tests were performed with test assemblies with various grid strap heights but same mixing vane design. The figure shows no trend of the measured-to-predicted CHF ratios against the grid strap height, indicating that grid strap height has no significant effect on the DNB performance. The staff concludes that the increased strap height used in the fuel design presented in WCAP-17524-P, Revision 1, has insignificant effect on the DNB performance, as supported by a previously reviewed and approved topical report and, therefore, the WRB-2M correlation is applicable.

Section 4.4.2.11.2 describes a formula for calculating the radial power distribution in the UO_2 fuel rods with a radial power depression factor F, which is determined using the radial power distributions predicted by the neutron transport theory code LASER. A second equation was added to account for annular fuel pellets that required a mathematical correlation applied to the first equation. The basis for adding this equation is that the AP1000 fuel rod design also may include annular fuel pellets in the top and bottom 20 centimeters (8 inches) of the fuel stack, as described in Subsection 4.2.2.1., "Fuel Rods," of the final safety analysis report. The staff finds the proposed change acceptable because the annular cylindrical relationship incorporated in the general heat conduction equation was confirmed by mathematical derivation.

In Sections 4.4.4.3.1 and 4.4.4.3.2, the applicant revised the "enthalpy rise hot channel factor" limit, at RTP condition, from 1.65 to 1.72, as required for the AFC configuration. As a result, the corresponding full-power enthalpy rise hot channel factor for the RTDP increased from 1.590 to 1.654, which included a statistical measurement uncertainty of 4 percent (1.72/1.04 = 1.654). Therefore, the revised AP1000 thermal hydraulic and safety analyses were performed assuming an "enthalpy rise hot channel factor" limit of 1.654. As a result of increasing the enthalpy rise hot channel factor, the hot channel void fractions listed in Table 4.4-2 also are increased. The staff finds that the changes are necessary for the AP1000 AFC configuration and are acceptable. However, the staff notes that analyses must continue to follow the approved design methodologies using the revised enthalpy rise hot channel factor, as required, to change TS parameters defined in the COL report.

In Section 4.4.6.1, the applicant discusses the incore instrumentation system detector operability requirement permitted to perform the primary function of providing a 3-D flux map of the reactor core. The 3-D flux data is used to calibrate the neutron detectors used by the protection and safety monitoring system, as well as to optimize core performance. To clarify the minimum number of detector segments required in each core quadrant, the applicant described the incore instrumentation thimble assembly as consisting of multiple fixed incore detector elements that start at the top of the active fuel and are sequentially placed along the vertical

axis to the bottom of the active fuel of the fuel assembly. In addition, a detector segment was defined as the difference in signal from two operable detectors within the same assembly. With a total of 42 incore instrument thimble assemblies placed throughout the core, the calculation algorithms measures the signal from each operable detector segment. The minimum number of incore assemblies detectors required for operating the system is at least 75 percent operating detector segments during the initial power distribution measurement required in each operating cycle, and at least 40 percent operating detector segments following the cycle initial power distribution measurement. A minimum of 15 operating detector segments in each quadrant with at least six detector segments below the core mid-plane and six detector segments above the core midplane in each quadrant is required both before and following the cycle initial power distribution measurement. The basis for the change is to eliminate misinterpretation of the requirement. For instance, a configuration could exist in which each quadrant has 30 operable detectors but less than 15 operable detector segments, as needed, based on the analysis. Since this change of the incore instrumentation description provides clarification to prevent misinterpretation of the operability requirement, the staff finds it acceptable.

4.4. Fuel Thermal Conductivity Degradation (TCD) Evaluation Of DCD Section 4.4

AP1000 DCD Section 4.4.2.11, "Fuel and Cladding Temperatures," discusses the principal factors employed in the determination of the fuel pellet temperature distribution over the fuel rod lifetime, such as gap size, internal gas pressure, gas composition, pellet density, fuel clad surface condition, and UO_2 thermal conductivity. The fuel pellet thermal conductivity is by far the primary factor that influences the pellet temperature profile. Since the UO_2 fuel pellet thermal conductivity is a function of fuel burnup that leads to TCD, as described in NRC Information Notice 2009-23 (Reference 34), "Nuclear Fuel Thermal Conductivity Degradation," the staff submitted several RAIs related to the calculation method used in the performance analysis and design (PAD) 4.0 code and the impact on the fuel pellet temperature distribution and centerline temperature related to thermal hydraulic design. The applicant developed the "PAD 4.0 TCD" code version as an assessment tool to evaluate the effects of fuel burnup TCD for the operating fleet and is also using it in the evaluation of the AP1000 fuel design presented in the CRR.

It should be noted that the staff is not performing a standalone safety evaluation of the PAD 4.0 TCD code (also known as the "PAD 4.0 TCD assessment tool"), but is evaluating its use for AP1000 and the impact on the fuel pellet heat transfer reduction and center line temperature rise prediction as a function of fuel burnup for the fuel design and core design presented in the CRR.

During the audit and evaluation of the CRR Section 4.4 proposed changes, the staff had concerns about the PAD 4.0 TCD code computation techniques of the TCD and its impact on the final AFC limit designs. The staff concerns focus on the: (1) TCD calculation with increasing bundle exposure, and (2) TCD effects on the thermal hydraulic and safety analysis. The staff submitted two RAIs—CRR-010 and CRR-011—each of which contain a series of questions pertaining to the TCD effects on the thermal hydraulic analysis.

In RAI CRR-010, the staff requested a discussion of the development and validation of the TCD model used in the interim version of PAD 4.0 TCD, including all coefficients of the TCD equation. The applicant provided the fuel thermal conductivity equation as modeled in the

licensed PAD 4.0 with an explanation of the equation's components, including a fuel exposure dependent function, f(Bu), which is presently deactivated pending NRC approval of this function for PWR licensed PAD 4.0. Therefore, the licensed PAD 4.0 fuel performance models described in WCAP-15063-P-A, "Westinghouse Improved Fuel Performance Analysis and Design Model (PAD 4.0)," do not address the impact of fuel thermal conductivity degradation in the thermal hydraulic analysis.

To assess the TCD impact on the fuel, the burnup dependent function, f(Bu), defined by a set of non-zero fuel exposure dependent coefficients, was incorporated into the PAD 4.0 code. This model was based on a TCD with burnup model recommended by the Halden Project Report, "Thermal Performance of High Burnup Fuel—In-Pile Temperature Data and Analysis," (Reference 35), which was previously integrated into the Westinghouse STAV 7.2 fuel performance code described in WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors—Supplement 1" (Reference 36). The STAV 7.2 code is employed in European reactors and approved by the NRC in the United States for boiling-water reactor (BWR) applications. The applicant's initial effort in developing a tool for assessing TCD effects resulted in a small conservative adjustment to the TCD impact on fuel thermal conductivity in the STAV 7.3 code version that yielded coefficients used in the PAD 4.0 TCD code.

In the development of the PAD 4.0 TCD assessment tool, the applicant considered two models: (1) the STAV TCD model based on the Halden TCD model, and (2) the modified Nuclear Fuels Industries (NFI) model, which is currently integrated into the FRAPCON 3.4 NRC fuel performance code described in NUREG/CR-7022 Volume 1, "FRAPCON–3.4: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," and Volume 2, "FRAPCON–3.4: Integral Assessment," (References 29 and 30). The applicant selected the STAV code model for the PAD 4.0 TCD because of the similarity in the modeling forms for fuel thermal conductivity in both the PAD 4.0 and STAV 7.2 codes. Although the STAV 7.2 code was developed for European BWR reactor designs, the fuel thermal conductivity property is not dependent on reactor design but is a function of fuel material, temperature, fuel density, burnup, and thermal expansion. Therefore, the STAV TCD model for f(Bu) is applicable to PAD 4.0 TCD PWR analysis.

The applicant validated the PAD 4.0 TCD assessment tool by comparing the PAD 4.0 TCD-predicted centerline temperatures with the measured fuel centerline temperatures from the Halden test reactor database, with a range of fuel burnups. The fuel centerline temperatures predicted by PAD 4.0 TCD were found to be in good agreement with measured values, with no further adjustment to the TCD model coefficients. The applicant provided four plots of thermal conductivity versus temperature at burnups of 0, 20, 40, and 65 gigawatt-days per metric ton of uranium (GWd/MTU) with or without the burnup coefficients default option enabled. As expected, the staff confirmed the significance of the fuel thermal conductivity as a function of fuel burnup and noted the following observations: (1) as fuel exposure increases, there is a significant downward shift in fuel thermal conductivity, and (2) the maximum shift occurs at low temperature and virtually converges above the temperature of 2,500 °C. Therefore, the staff concludes that the data confirms the fuel thermal conductivity is a function of fuel burnup that needs to be addressed in the safety analysis.

In response to a staff request, the applicant provided plots demonstrating the differences between the approved PAD 4.0 and the "PAD 4.0 TCD assessment tool" thermal conductivity models. For each plot, the difference between the measured and the predicted (M-P) centerline

temperature (°F) was plotted against fuel burnup (MWD/MTU). The staff observed that the plot without TCD compensation showed a linear increase in the difference between the measured and predicted temperature with a burnup between 0 and approximately 75,000 MWD/MTU, with a maximum temperature difference of approximately 204 °C (400 °F). Above 75,000 MWD/MTU, the difference in temperature (M-P) decreases with increasing fuel burnup. Whereas, the plot using PAD 4.0 TCD assessment tool showed almost no change in the difference temperature over the full range of fuel burnup. Therefore, the PAD 4.0 TCD assessment tool is consistently accurate with the measured data at any fuel exposure. The staff concludes that the PAD 4.0 TCD assessment tool provides an accurate temperature profile across the full range of approved fuel exposure with a high degree of reliability.
The applicant provided a description of the PAD 4.0 fuel performance code thermal model calibration and validation process that included three initial NRC-sponsored Halden tests followed by additional experiments conducted to further understand fuel properties as a function of burnup and to reduce the uncertainties in the model.

Each of the three Halden tests (IFA-431, IFA-432, and IFA-513) consisted of six test rodlets fabricated at the PNNL facility. To accommodate the initial pellet-clad gaps used in commercial fuel rod design, the test rod fabrication parameters were designed with a range of initial pellet-clad gaps of 2.2 mils to 15.1 mils. In addition, the IFA-431 and IFA-432 test rods were pre-pressurized with one atmosphere helium. The IFA-513 test rods had a range of pre-pressurization conditions that included 100 percent helium at one or three atmospheres and a mixture of helium and argon at one atmosphere. All test rods were instrumented with thermocouples at the top and bottom of the fuel stack.

The data obtained from the testing were used in PAD 4.0 thermal model calibration and development at low burnup conditions. This included online fuel centerline temperature measurements as a function of burnup, up to 5 GWd/MtU. Also, the data obtained, because of initial gap conditions simulated the range of gap conditions expected in Westinghouse PWR operation, were considered applicable to the full operating fuel burnup range. In addition, the relatively low burnup data provided an opportunity to minimize the uncertainties associated with cladding creep, fission gas release, and thermocouple decalibration.

To obtain data of the irradiation effects on thermocouple response behavior at extended burnup conditions, the IFA-432 test rod was irradiated in excess of 45 GWd/MtU and, from this data, the Halden project developed decalibration correction factors to compensate for the thermocouple response. The PAD 4.0 thermal model was further validated using the higher burnup data for the IFA-432 test rods. Although the measured versus predicted fuel temperature comparisons for the extended burnup data did not show a clear trend with burnup, the NRC approved the PAD 4.0 thermal model without the fuel thermal conductivity degradation function.

Subsequent to these NRC-sponsored Halden tests, additional experiments have been conducted to further understand fuel properties as a function of burnup. To avoid uncertainties associated with thermocouple decalibration, two alternative approaches were considered. First, tests were run with annular pellet-fueled rods with expansion wire thermometers in the annulus. This instrumentation is not subject to decalibration under irradiation, and these tests provided the first definitive results confirming TCD. To improve the expansion wire thermometer results by reducing the potential error caused by pellet fragment interference with the wire, Westinghouse performed additional tests using re-fabricated commercially irradiated fuel rods into test rods instrumented with new thermocouples. These tests provided high quality

temperature measurements at high burnup but introduced an additional uncertainty in the fuel rod conditions because of re-fabrication. However, the composite data results of tests discussed above were used to validate the PAD 4.0 TCD model.

For these reasons, the staff released IN 2011-21, Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation" (Reference 11), which notified the industry of the information related to the impact of irradiation on fuel thermal conductivity and its potential to cause errors in realistic emergency core cooling system evaluation models.

The burnup-dependent TCD equation from STAV 7.3 is used in the interim version of PAD 4.0, called PAD 4.0 TCD. The STAV 7.3 code is approved for BWR fuel design analysis and is designed to handle fuel with gadolinia as a burnable absorber. The AP1000 PWR fuel design may include axial blankets (fuel pellets of a reduced enrichment), annular fuel pellets in the top and bottom 20 centimeters (8 inches) of the fuel stack (fully enriched or partially enriched), and burnable absorber (ZrB_2-coated fuel pellets or fuel pellets containing gadolinia oxide mixed with uranium oxide). In RAI CRR-011, the staff requested that the applicant address the impacts that this fuel design may have on the thermal conductivity correlation, and discuss the applicability of the TCD equation from STAV 7.3 for the integral fuel burnable absorber (IFBA) coated pellets.

The applicant states that the "Halden test rods used for thermal model validation are applicable to the full range of fuel pellet types used in the AP1000 PWR fuel rod design," including the UO_2 material properties for uranium-235 enrichments and natural enriched uranium in the axial blanket pellets. The effects of enrichment on pellet radial power distributions are accounted for as input to the PAD code. The PAD 4.0 TCD model validation is, therefore, applicable to the range of pellet types used in the axial blanket region.

The IFBA pellets have a coating of ZrB_2 on the outside pellet surface. This coating has only a negligible thermal resistance and is conservatively modeled in PAD as a small increase in the fuel diameter. The coating has no impact on the temperature gradient across the fuel pellet, and the Halden test results for uncoated pellets are applicable to the ZrB_2 IFBA design. Gadolinia bearing burnable absorber fuel pellets are also an option in the AP1000 PWR fuel rod design. The inclusion of gadolinia in the fuel matrix acts as an impurity, and the unirradiated thermal conductivity of the Gd_2O_3-UO_2 fuel are appropriately reduced as a function of the gadolinia content. The staff agrees with the applicant's method and conclusion based on the conservative assumptions made in the applicant's models.

The fission product generation in Gd_2O_3-UO_2 fuel is similar to that for UO_2 fuel, and no further changes are required to appropriately assess TCD in gadolinia fuel. In summary, the PAD 4.0 TCD assessment tool is applicable for use with all fuel types considered in the AP1000 PWR core design. Based on the Halden test rods design as being applicable to the fuel pellet types used in the AP1000 PWR fuel rod design, the "effects of enrichment" as input parameter, the conservative modeling of the ZrB_2 coating, and the fission product generation in Gd_2O_3-UO_2, the staff concludes that the PAD 4.0 TCD assessment tool is suitable for use with all fuel types considered in the AP1000 PWR core design.

4.5. <u>LOCA Analysis</u>

4.5.1. SBLOCA

4.5.1.1. Thermal Conductivity Degradation

Reference 2 discusses the impact of TCD on the SBLOCA analyses for the AP1000 design. Westinghouse concludes that the increase in stored energy caused by TCD would not impact the conclusions in the documented analyses since stored energy is not a significant effect in the SBLOCA transients. Westinghouse also mentions that variations in rod internal pressures relative to burnup are already covered to a large extent in the SBLOCA burnup studies that are performed as necessary with the NOTRUMP evaluation model (References 24 and 25). Westinghouse further states that burnup studies are not required specifically for the AP1000 since the SBLOCA analyses for the AP1000 plant design show that no core uncovery occurs, which means no clad heatup calculations are needed. Westinghouse explains that since clad heatup calculations are not performed, it would not be necessary to perform burnup studies to look at the effects of TCD.

Reference 2 describes a quantitative analysis demonstrating that TCD has a negligible impact on the SBLOCA analysis of records performed with the Westinghouse NOTRUMP evaluation methodology for Westinghouse nuclear stream supply system (NSSS) plant designs. It is further stated that there is no need to credit additional margins to continue to meet the acceptance criteria. Westinghouse explains that the conclusions relative to the impact of TCD from the quantitative analyses are applicable to the AP1000.

In RAI CRR-009-SI, the staff requested the applicant to analyze how much fuel rod heatup is calculated and whether this leads to significant changes to the previously calculated rod internal pressures during the SBLOCA when considering TCD effects. In the response to RAI-009-SI (Reference 13), Westinghouse states that during a SBLOCA, the fuel rods are surrounded by a two-phase mixture of steam and liquid water for the majority of the transient and that stored energy in the core because of the initial fuel temperature is not a significant effect in SBLOCA transients because the initial phase of the transient is gradual enough to remove the energy from the system before any core uncovery.

After the initial submittal of the CRR (Reference 1), and subsequent RAI response (Reference 13), Westinghouse identified the single failure of one automatic depressurization system stage 4 (ADS-4) valve on the nonpressurizer side of the reactor coolant system (RCS) to be more limiting than the analysis initially presented, with the failure on the pressurizer side. Westinghouse submitted the revised analysis in Revision 1 of the CRR (Reference 37) for the 2-inch (5 centimeter) cold leg break and inadvertent ADS transient simulations, in which containment backpressure was now used instead of atmospheric pressure during the transient. The revised analysis (Reference 37) shows core uncovery for the Inadvertent ADS and the 2-inch (5 centimeters) cold leg break cases. Since the analysis now shows that there is core uncovery, the analysis uses the SBLOCTA computer code to calculate the PCT during the transient. The SBLOCTA code is based on the LOCTA-IV code documented in WCAP-8301-P (Reference 38) modified as described in WCAP-10054-P-A and WCAP-10081-A (References 25 and 39) and approved with that application. The revised analyses still produces results that demonstrate significant margin to peak cladding temperature regulatory limits. Also, as a consequence of the new analysis showing that core uncovery occurs, the applicant

assessed the effect of TCD on the revised analysis and concluded that the SBLOCA analysis is negligibly impacted (Reference 37).

During an audit held on November 25, 2013 and November 26, 2013 (Reference 40) Westinghouse provided clarifying information on the impact of the revised analysis and containment backpressure methodology that is being used in the SBLOCA analysis. The applicant provided additional information to support that the containment pressure used was a conservatively low minimum containment pressure. Based on the information provided in References 37 and 40, the staff finds the new analysis, based on approved methodologies, demonstrates that the SBLOCA analysis results are negligibly impacted when considering the effects of TCD and that the revised analysis demonstrates compliance with the regulatory requirements.

4.5.2. LBLOCA

4.5.2.1. Thermal Conductivity Degradation

Section 4.0 of Reference 10 discusses the impact of TCD on the LBLOCA analyses. Westinghouse explains how it estimated the effect of incorporating a fuel thermal conductivity model that considers fuel TCD as a function of burnup on ASTRUM (automated statistical treatment of uncertainty method). ASTRUM is documented in Reference 26, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," which does not explicitly account for fuel TCD. To account for fuel TCD, Westinghouse distinguishes between the "margin PCT" and the "integrated PCT." The margin PCT is established to serve as a reference PCT that better captures the true impact of the fuel TCD effects established by the integrated PCT. The margin PCT uses a reduced F_Q peaking factor conservatism only for the limiting two PCT cases from the original ASTRUM analysis where the conservatism was found to be attributed to an unreasonably conservative F_Q rather than some real physical phenomena. The margin PCT is then established by finding the limiting PCT value among this updated original set of 124 WCOBRA/TRAC results. The limiting PCT was reduced by 39 °C (70 °F) in this more realistic estimate (again, not assessing the impact of TCD).

To establish the integrated PCT (or the PCT considering TCD effects in addition to adjustment of the F_Q conservatism in the original two limiting cases), 31 WCOBRA/TRAC cases were re-run, selected to be "among the most limiting cases from the original 124-run ASTRUM analysis." The overall process to quantify the effect of TCD in these 31 cases was to (1) update the previously approved PAD 4.0 fuel performance code to include a burnup-dependent fuel TCD model to initialize the WCOBRA/TRAC runs, (2) update the previously approved HOTSPOT code to use a burnup-dependent fuel TCD model (different from that used in PAD 4.0) throughout the extent of the transient, and (3) use the burnup-dependent peaking factors as given in Table 5-2 of Reference 10. Table 5-3 in Reference 10 shows that the design basis limiting PCT increased by 38 °C (69 °F). Comparing the margin PCT to the integrated PCT, the effect of TCD seen would cause an approximate increase of 77 °C (139 °F) between the updated analysis and the previous analysis.

In RAI CRR-004, the staff requested that Westinghouse explain the process of determining average assembly peaking factors and burnups with ASTRUM to ensure that the initial stored energy of these average fuel assemblies is not being underestimated. The staff also requested

justification for limiting the burnup of the average assemblies to 30 GWD/MTU. In the response to RAI CRR-004 (Reference 6), Westinghouse explains that when applying ASTRUM to the AP1000 design analyses, values between [ ] are produced during sampling average burnup. Sensitivity studies were performed for (1) the limiting ASTRUM case before the TCD analysis, and (2) the limiting ASTRUM case from the TCD analysis. In both sensitivity studies, the average burnup was fixed to approximately [ ] to show the impact (at the much higher average burnup upper limit) of the increased stored energy on PCT for the limiting hot rod location. The results showed that "the increased temperature in the average rods does not substantially impact the global response and had only a small impact on the calculated hot rod PCTs (Figures 3A and 3B of the RAI CRR-004 response (Reference 6))." The staff analyzed the figures presented in the response and confirmed that there was negligible impact on the maximum hot rod PCT. Therefore, the issue of potential underestimation of stored energy in the average assemblies is resolved.

4.5.2.2. *Axial Power Shape Generation for WCOBRA/TRAC Cases*

In RAI CRR-005, the staff asked Westinghouse to describe the process used to adjust the F_Q values in select cases closer to the desired value and to explain why this adjustment is appropriate with respect to how certain parameters are sampled within ASTRUM. The parameters in question are called PBOT and PMID, and are sampled (within a prescribed envelope based on a specific nuclear core design) when performing an ASTRUM analysis. PBOT and PMID, with inputs such as the peak linear rate (PLHR) and FQ for a given WCOBRA/TRAC case, define the axial power shape for a given case. In the response to RAI CRR-005 (Reference 5), Westinghouse explains that the PSHAPE routine [

]. Since PLHR is required to be at the desired value—or greater (Westinghouse points to Table 12-6 in the ASTRUM topical report (Reference 26), which defines the criterion)—this would imply that F_Q is required to be at the desired value or greater. Therefore, reducing the F_Q value to the value corresponding to the desired value for the PLHR would be consistent with the as-approved ASTRUM. Westinghouse states: [

.]

Table 2 of the CRR-005 response (Reference 5) shows that the FQ values used in the final margin case runs exceed the initially sampled FQ values by the PSHAPE routine (only the top two PCT cases were re-run). It also shows that PLHR was also matched, as shown in Table 3 of the response. A more in-depth review of the PSHAPE code used to generate the axial power shapes for the WCOBRA/TRAC calculations revealed that Westinghouse internal procedures allow randomly sampled parameters to be changed or biased in certain circumstances within a certain prescribed tolerance. Specifically, this is the case for the sampled PBOT and PMID parameters where internal guidance allows for randomly sampled PBOT and PMID values to be changed by [ ] to allow physical power shapes for WCOBRA/TRAC cases to be defined while also maintaining certain criteria, such as the PLHR (and hence FQ) and

integrated hot rod power criteria as given in Table 12-6 of the ASTRUM methodology topical report (Reference 26).

The staff notes that the ASTRUM methodology topical report (Reference 26) does not explicitly document the process for sampling the input parameters. However, the staff confirmed that the biasing methodology used with PSHAPE was limited and did not significantly change the physical behavior of a given case. Additionally, the applicant demonstrated that the methodology used satisfies the criteria for acceptable steady-state listed in Table 12-6 of the ASTRUM methodology topical report (Reference 26). Therefore the staff finds the axial power shape generation for the CRR acceptable.

4.5.2.3. Peaking Factor Burndown Credit

As discussed in Reference 10, Westinghouse is taking credit for peaking factor burndown concurrent with accounting for TCD effects. In RAI CRR-006, the staff asked Westinghouse to describe how the burnup-dependent peaking factors were determined and then implemented, in the LBLOCA analysis. In the response to RAI CRR-006 (Reference 6), Westinghouse explains that the process of determining the hot rod burnup, and $F_{\Delta H}/F_Q$ at the corresponding burnup, is [

.] Peaking factor limits were then defined by conservatively bounding the steady-state AP1000 peaking factors as shown in Figures 1 and 2 of the response to RAI CRR-006-S1 (Reference 13). This means that $F_{\Delta H}/F_Q$ is determined [
].

Westinghouse states that it developed the peaking factor curves [

] which can be seen in Figures 1 and 2 of the response to RAI CRR-006-S1 (Reference 13).

Taking credit for peaking factor burndown for LBLOCA, the analyses are consistent with ASTRUM, since the maximum $F_{\Delta H}/F_Q$ (for any given time-in-cycle) is still being used, only with less margin being maintained between the defined limit and a given sampled peaking factor. Westinghouse also states that the reduced peaking factors "will be confirmed to be met [
] during the future reload process, similar to other limits." Tables 1 and 2 of the CRR-006-S1 response (Reference 13) give peaking factor limits for the AP1000, which will be confirmed as part of the cycle design analysis. No changes to the plant Technical Specifications or Core Operating Limits Report will be made. Since Westinghouse has demonstrated that the AP1000 LBLOCA analyses still conform to the as-approved ASTRUM, the staff finds taking credit for peaking factor burndown in the manner described above to be acceptable.

4.5.2.4. Summary of LBLOCA Analysis Changes in References 1 and 37

During an audit of the CRR (Reference 14), the staff reviewed a calculation note that led to questions about the assumptions used in the updated LBLOCA analyses that address the TCD concerns. There was a specific question regarding the type of reactor coolant pump (RCP) used. RAI CRR-007 was issued requesting clarification of the significant differences or changes

to the LBLOCA analyses that address TCD, along with why it was necessary and appropriate for each change. In the response to RAI CRR-007(Reference 5), Westinghouse clarified that the RUV RCP designed by KSB was not used for DCD or CRR best estimate LOCA analyses, and the analyses in the calculation note in question are not applicable to the AP1000. Westinghouse goes on to state the primary differences between the DCD and CRR LBLOCA analyses, including the reasons for the changes, are as follows:

1. changes to the fuel assembly mechanical design features (primarily the grid design) because of the fuel design change

2. changes to the RCP design and homologous curves because of updated parameters resulting from design finalization

3. changes to the upper head structures because of updated parameters resulting from design finalization

4. increase in the time before RCP trip following an LBLOCA when offsite power is available for consistency with updated non-LOCA analyses for the CRR

5. change in PBOT or PMID box because of the core design change

6. change in $F_{\Delta H}$ limit because of the core design change

The response to RAI CRR-007 (Reference 5) also includes a notification of changes to ASTRUM to account for burnup-dependent aspects of the fuel performance changes since TCD and peaking factor burndown were not explicitly considered in the as-approved ASTRUM. These changes effectively include (1) expanding the fuel performance data to account for the effects of TCD, (2) [ ], (3) updating the thermal conductivity model in WCOBRA/TRAC and HOTSPOT, and (4) [ ].

Based on the above discussion, the staff finds that Westinghouse has incorporated all of the important design changes between the DCD and WCAP-17524-P, Revision 1, in the LBLOCA analyses. The changes that were made resulted from changes as detailed in WCAP-17524-P, Revision 1, and described in Reference 10, and were found to be appropriate.

To properly address fuel TCD for the LBLOCA analysis, the staff requested, in RAI CRR-008, that Westinghouse re-analyze all 124 WCOBRA/TRAC cases, as required by the ASTRUM evaluation methodology (Reference 26), rather than the subset of cases that Westinghouse determined to be the most limiting. In the response to RAI CRR-008 (Reference 5), Westinghouse states that a plant-specific adaptation of ASTRUM was applied to consider the burnup-dependent aspects of the fuel performance changes considering TCD. All of the previous DCD cases were re-run using the same inputs, with a few exceptions. The exceptions occurred for the following reasons:

1. to account for TCD effects (this includes use of PAD 4.0 with TCD modeled to initialize the WCOBRA/TRAC and HOTSPOT fuel rod temperatures and a modified fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT)

2. to account for peaking factor burndown (only sampled average burnups greater than 30 GWD/MTU were affected)

3. to allow modification of PBOT/PMID parameters for three limiting PCT cases i.e., F_Q margin was reduced for three limiting PCT cases consistent with the PLHR and integrated hot rod power criteria in the originally approved ASTRUM evaluation methodology).

In the response, Westinghouse also notes that the burndown after 55 GWD/MTU is different than presented in Reference 10 and that the additional burndown towards the end of life better reflects the power and peaking limits expected for higher burnup assemblies. Additionally, [

]."

Since all previous LBLOCA cases were re-run, [

], the staff finds that Westinghouse performed the minimum number of cases as required by ASTRUM, and that the updated analysis is acceptable because the applicable regulations in SRP/NUREG-0800 Section 15.6.5 continue to be met.

The staff also asked about the impact of the corrected fuel TCD models on the peak containment pressure and temperature calculation in RAI CRR-023. In the response to RAI-CRR-023 (Reference 6), Westinghouse states that the current value for initial core stored energy used for input to the LBLOCA mass and energy release remains conservative when the effects of TCD are considered and, therefore, there is no change to the peak calculated containment pressure or temperature resulting from a LBLOCA. More details on the method used are given in the response to RAI CRR-026, and this response was reviewed in Section 4.5 of this document.

In LTR-NRC-14-20 (Reference 41), the applicant notified the staff of errors identified in the evaluation models for the LBLOCA analysis presented in WCAP-17524-P, Revision 1. Reference 41 contains a tally of all errors and modifications to the LBLOCA and SBLOCA evaluation models per the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The cumulative estimated PCT impact on the LBLOCA analysis is 19 °C (34 °F), which is less than the 28 °C (50 °F) threshold specified in 10 CFR 50.46. Per Section 5.1 of WCAP-17524-P, Revision 1, any applicant referencing this topical report must account for any and all errors/modifications in the evaluation model and disposition them in the results presented in connection with the site-specific application of WCAP-17524-P, Revision 1. This would include the 19 °C (34 °F) reported in Reference 41. The staff finds that this meets the requirements of 10 CFR 50.46.

4.6. Non-LOCA Analyses

WCAP-17524-P, Revision 1, Appendix B, Section 15.0, "Accident Analyses," presents the changes to the design bases for the AP1000 accident analyses. Several proposed changes in assumptions and design parameters affect several analyses and are discussed in this section. Staff evaluation of specific analyses is discussed later.

The CRR includes an initial reactor core power of 3,449.15 MWt, which assumes a calorimetric uncertainty of 1 percent, in analyses that previously assumed 3,483.3 MWt. This is consistent with the power uncertainty assumed for the large-break LOCA analysis. The difference of 15.15 MWt between non-LOCA and LOCA analyses is because of the reactor coolant pumps, which are not assumed to be operating during a LOCA because they trip automatically following an "S" signal. The applicant previously revised DCD Section 15.0.15 to include COL Information Item 15.0.15.1, which requires the COL holder to calculate the primary power calorimetric uncertainty before fuel load to confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated value. Therefore, the staff found the use of 1 percent initial core power uncertainty acceptable since it continues to follow the guidance found in SRP/NUREG-0800 Section 15.

In RAI CRR-018, the staff requested Westinghouse to demonstrate that the in-containment equipment program has already taken into account the containment heatup events resulting from new procedures that discharge high temperature and high pressure primary coolant in the in-containment RWST. In its response (Reference 6), the applicant referred to DCD Appendix 3D for the equipment qualification methodology and stated that the aging program includes the effect of temperature for normal plant operation, plus 72 hours of accumulated Group 1 conditions and 24 hours of accumulated Group 2 conditions. The applicant stated that as long as these events do not exceed the temperatures and accumulated durations allowed for Group 1 or Group 2 events, then the affected components are bounded by component pre-accident aging design basis conditions. Based on the staff's review of the applicant's response (Reference 5), the staff found the response acceptable and considers RAI CRR-018 closed.

In RAI CRR-021, the staff requested that the applicant evaluate the difference between the FIGHTH code and the fuel performance code, PAD, which models the thermal conductivity degradation properly. The staff further requested that the applicant use this evaluation to determine the impact on the calculated DNBR, peak linear power density, transient power, power level, and cladding strain. In its response (Reference 5), the applicant stated that the PAD and FIGHTH results tend to agree very well, except for some specific instances. The differences between PAD and FIGHTH occur in the burnup range of 5 to 10 GWD/T and linear heat rates of 16.3 kW/ft or greater. The LHGR rate peaks at approximately 13.4 kW/ft, which is lower than the level at which differences are observed between PAD and FIGHTH. The applicant further stated that the PAD code is used for the safety analyses; therefore, the differences in the FIGHTH and PAD code will not have any impact on the calculation of DNBR, peak fuel temperatures, and enthalpy. Based the staff's review of the additional information provided by the applicant in Reference 5, the staff found the response acceptable and considers RAI CRR-021 closed.

In RAI CRR-026, the staff requested that the applicant describe the core initial stored energy calculation method and explain how zero rated power can be most limiting in regard to initial core stored energy. In its response (Reference 5), the applicant provided a description of the core stored energy calculation, which is normalized with respect to local linear power. Because of the normalization while the absolute core stored energy increases with regard to core power, the normalized core stored energy is limiting at low power (1.5 kW/ft), not zero power. The applicant provided analysis showing that when the effects of thermal conductivity degradation are included, the maximum normalized core stored energy is at a high burnup and is higher than the previously calculated normalized core stored energy. The applicant further stated that when

using a more representative calculation that assumed a less conservative power distribution and a weighted average for burnup, this offset the increase in the normalized core stored energy by including the effects of thermal conductivity degradation. Based on the staff's review of the applicant's response (Reference 5), the staff found the response acceptable and considers RAI CRR-026 closed.

The CRR contains changes to Tables 15.2-1 and 15.5-1 of the DCD. These changes incorporate required operator action to open the reactor vessel head vents (RVHVs) in order to prevent the pressurizer from overfilling with water during several design basis events (15.2.7 loss of normal feedwater flow, 15.5.1 inadvertent operation of the core makeup tanks at power, and 15.5.2 chemical and volume control system malfunction that increases reactor coolant system inventory). In RAI CRR-029, NRC staff inquired about the addition of these required operator actions. The applicant's response (Reference 43) stated that Chapter 15 of the DCD Rev. 19 presented analyses for the events in 15.5.1 and 15.5.2 that resulted in the smallest margin to pressurizer overfill. However, there are scenarios that do require operator action to open the RVHVs in order to prevent pressurizer overfill. Scenarios where operator action is taken to open the RVHVs result in a greater margin to pressurizer overfill than the scenarios presented in the DCD Rev. 19. The CRR presented scenarios for the events in 15.5.1 and 15.5.2 that resulted in the minimum time requirement for operator action. NRC staff acknowledged the presence of these operator actions in Section 15.2.5.1 of the SER for the AP600 DCD (Reference 44) and found them to be acceptable. Based on the discussion above, NRC staff finds the applicant's response to RAI CRR-029 concerning operator action to open the RVHVs to be consistent with Sections 15.5.1.3 and 15.5.2.3 in the DCD for AP600 and AP1000, and is acceptable. The impact on the 15.2.7 loss of normal feedwater flow event is discussed further in Section 4.6.13 of this SER.

4.6.1. (SRP 15.1.1) Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

The staff reviewed Section 15.1.1 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow."

The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased reactor vessel (RV) diameter for the neutron pad addition

- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The new analysis indicates that the decrease in feedwater flow temperature remains bounded by the analysis of excessive increase in secondary steam flow (Section 15.1.3) but is no longer bounded by the analysis for feedwater system malfunctions that result in an increase in feedwater flow (Section 15.1.2). The staff found these changes acceptable and confirms that the analysis of Section 15.1.3 continues to bound the analysis of Section 15.1.1.

4.6.2. (SRP 15.1.2) Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The staff reviewed Section 15.1.2 in Appendix B WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis: increased $F_{\Delta H}$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients. The analysis was also updated and expanded to include flow increases to both steam generators. Because a primary coolant coastdown no longer occurs in the analysis, the FRACTRAN and VIPRE-01, which were previously used for this portion of the analysis, are no longer used.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.3. (SRP 15.1.3) Excessive Increase in Secondary Steam Flow

The staff reviewed Section 15.1.3 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased main steam safety valve (MSSV) inlet piping diameter (increased 3 centimeters (1.2 inches))

- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Because a primary coolant coastdown no longer occurs in the analysis, the FRACTRAN and VIPRE-01 codes, which were previously used for this portion of the analysis, are no longer used.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow" and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.4. (SRP 15.1.4) Inadvertent Opening of a Steam Generator Relief or Safety Valve

The staff reviewed Section 15.1.4 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increase MSSV inlet piping diameter (increased 3 centimeters (1.2 inches))
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.5. (SRP 15.1.5) Steam System Piping Failure

The staff reviewed Section 15.1.5 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)

- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

In addition to the changes to the analysis, editorial changes were also included. In particular, the initial iodine and noble gas primary coolant concentrations are more accurately described as being based on their respective TS (i.e., equilibrium operating limits) because the TS limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The doses were revised based on updated analysis.

The applicant included in WCAP-17524-P, Revision 1, the analyses for steam system piping failures initiated from at power conditions that had been previously considered by the applicant. The results of analyses for steam system piping failures initiated from full power were demonstrated to meet the acceptance criteria.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.6. (SRP 15.1.1 and 15.1.4) Inadvertent Operation of the Passive Residual Heat Removal (PRHR) Heat Exchanger

The staff reviewed Section 15.1.6 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 and 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow."

This section was largely unchanged except for minor editorial changes. The staff accepts the editorial changes and agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.7. (SRP 15.2.1) Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.8. (SRP 15.2.2) Loss of External Electrical Load

Editorial changes were incorporated. The staff accepts the editorial changes and agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.9. (SRP 15.2.3) Turbine Trip

The staff reviewed Section 15.2.3 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.2.1 through 15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.10. (SRP 15.2.4) Inadvertent Closure of Main Steam Isolation Valves

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.11. (SRP 15.2.5) Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.12. (SRP 15.2.6) Loss of Power to the Plant Auxiliaries

The staff reviewed Section 15.2.6 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- applied containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The loss of ac power to the plant auxiliaries case presented in the DCD, where feedwater flow is lost at time zero, and power to the reactor coolant pumps is lost as a result of the turbine trip, was renamed in Section 15.2.7 as "Loss of Normal Feedwater Flow with Loss of Offsite Power." The case presented in Section 15.2.6 now assumes a loss of reactor coolant pumps and loss of feedwater pumps at event initiation.

In RAI CRR-022, the staff requested quantitative evidence to justify the conclusion that the effect of TCD on loss of flow and loss of offsite power (LOOP) events would not exceed the 50 psid (pounds per square inch differential) margin of system pressure. In its response (Reference 5), the applicant provided results from its quantitative analysis showing sufficient margin to the reactor coolant system (RCS) Pressure Acceptance Criterion for both the loss of load and turbine trip event and the locked rotor event. For both events, the applicant showed that the increase in peak RCS pressure was less than 3 psid, which is small compared to the margin available in the analysis. Based on the quantitative analysis, the applicant provided the response (Reference 5). The staff finds the response to RAI CRR-022 acceptable and considers the matter closed.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.6, "Loss of Nonemergency AC Power to the Station Auxiliaries." Based on the information provided in WCAP-17524-P, Revision 1, and the response to RAI CRR-022 (Reference 5), the staff finds the changes and analysis acceptable.

4.6.13. (SRP 15.2.7) Loss of Normal Feedwater Flow

The staff reviewed Section 15.2.7 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- containment backpressure effects on PRHR heat transfer
- operator action to open RVHVs is credited
- addition of the flow skirt, increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve
- nozzle, and piping pressure loss coefficients

The operator action to open the RVHVs is credited in the CRR analysis for the loss of normal feedwater flow design basis event. The inclusion of the effects of containment backpressure on

the performance of the passive residual heat removal heat exchanger in the CRR results in a greater expansion of the reactor coolant system such that operator action is required to mitigate this event. This operator action is consistent with the operator action credited in design basis events 15.5.1 inadvertent operation of the core makeup tanks at power and 15.5.2 chemical and volume control system malfunction that increases reactor coolant system inventory, and is reviewed by NRC staff in Section 4.6 of this SER. An additional case, loss of normal feedwater flow with loss of offsite power, was added to this section.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.7, "Loss of Normal Feedwater Flow," and found the changes acceptable and that the regulatory requirements continue to be met.

4.6.14. (SRP 15.2.8) Feedwater System Pipe Break

The staff reviewed Section 15.2.8 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- applied containment backpressure effects on PRHR heat transfer
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment," and found the changes acceptable and that the regulatory requirements continue to be met.

4.6.15. (SRP 15.3.1) Partial Loss of Forced Reactor Coolant Flow

The staff reviewed Section 15.3.1 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt

- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.1 through 15.3.2, "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions," and found the changes acceptable and that the regulatory requirements continue to be met.

4.6.16. (SRP 15.3.2) Complete Loss of Forced Reactor Coolant Flow

The staff reviewed Section 15.3.2 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.1 through 15.3.2, "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions," and found the changes acceptable and that the regulatory requirements continue to be met.

4.6.17. (SRP 15.3.3) Reactor Coolant Pump Shaft Seizure (Locked Rotor)

The staff reviewed Section 15.3.3 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.3 through 15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.6.18. (SRP 15.3.4) Reactor Coolant Pump Shaft Break

Only editorial changes were incorporated. The staff accepts the editorial changes and agrees that no changes to the analysis are necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.19. (SRP 15.4.1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

The staff reviewed Section 15.4.1 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low

Power Startup Condition," and found the changes acceptable and that the regulations continue to be met.

4.6.20. (SRP 15.4.2) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The staff reviewed Section 15.4.2 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," and found the changes acceptable and that the regulations continue to be met.

4.6.21. (SRP 15.4.3) Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

The staff reviewed Section 15.4.3 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error," and found the changes acceptable and that the regulations continue to be met.

4.6.22. (SRP 15.4.4) Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.23. (SRP 15.4.5) A Malfunction or Failure of the Flow Controller in a Boiling-Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.24. (SRP 15.4.6) Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

The staff reviewed Section 15.4.6 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)" and found the changes acceptable and that the regulations continue to be met.

4.6.25. (SRP 15.4.7) Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

4.6.26. (SRP 15.4.8) Spectrum of Rod Cluster Control Assembly Ejection Accidents

The staff reviewed Section 15.4.8 in Appendix B of Reference 1, which addresses changes and updates to the spectrum of rod ejection accidents described in Revision 19 of the AP1000 DCD.

In RAI CRR-019, the staff noted that the rod ejection accident was analyzed using the previous fuel pellet thermal conductivity model and asked what the quantitative impact of the corrected TCD model with burnup dependence will be on the rod ejection accident analysis. In its response dated January 2, 2013, the applicant stated that the effects of TCD were already taken into account in the nuclear power transient in the calculation of the fuel temperatures used for Doppler feedback effects and for the maximum heat flux and minimum DNBR portion of the hot rod transient response analysis, which, [ ]. For the minimum heat flux and maximum fuel temperature and enthalpy portion of the hot rod transient response, the applicant provided quantitative analysis showing that the primary effect of TCD on the CRR rod ejection peak fuel temperatures and enthalpies will be in the [ ]. Furthermore, the applicant provided quantitative analysis results that show that the change in enthalpy rise is more than offset by the margin calculated for the CRR to the criterion in SRP 4.2 Revision 3 and that the peak fuel centerline temperature also remains well below the fuel melt temperature. Based on the staff's review of the applicant's response dated January 2, 2013, the staff accepts the response and RAI CRR-019 is considered closed.

In RAI CRR-020, the staff noted that a 3-D kinetics code is used to analyze the AP1000 rod ejection accident and asked if the same code will be used for the future reload analysis. In its response dated January 2, 2013, the applicant stated that the same 3-D kinetics methodology used in Reference 1 rod ejection analysis will be used for reload cycles. The applicant further stated that the methodology is consistent with Reference 27 and that benchmark testing discussed in RAI CRR-003 demonstrated that the 3-D kinetics code used to perform the calculations gives equivalent results to SPNOVA for representative rod ejection accidents. Based on the staff's review of the applicant's response dated January 2, 2013, the staff accepts the response and RAI CRR-020 is considered closed.

During the Phase 2 audit (Reference 14), several rod ejection analyses were reviewed, including the hot zero power (HZP) enthalpy analysis [ ], the hot full power (HFP) enthalpy analysis [ ], and the DNBR analysis were reviewed during the audit. All cases were analyzed for both the first cycle and the equilibrium cycle.

Westinghouse identified the HZP analysis as limiting for the PCMI and the zero power high clad temperature failure criteria. The corresponding acceptance criterion is less than170 cal/g (or less than 150 cal/g if rod internal pressure exceeds RCS pressure) to mitigate high clad temperature failure. The PCMI failure criterion compares the peak fuel average enthalpy rise versus the cladding oxide to wall thickness ratio. Westinghouse states that the PCMI failure criterion is converted to a rod average burnup basis for comparison directly to ANC. For the HZP [ ] scenario, peak enthalpy rise values were calculated for the first cycle and the reference 18-month equilibrium cycle, [ ] cal/g and [ ] cal/g, respectively, which are well below the PCMI limit. A figure in the corresponding Westinghouse calculation note comparing the PCMI failure criterion to the calculated enthalpy rise results of the re-analysis as a function of rod average burnup also was reviewed, along with another figure showing the calculated enthalpy rise margin. It was noted that the minimum margin is [ ] percent or [ ] cal/g. The calculated peak enthalpy results of [ ] cal/g and [ ] cal/g for the first cycle and the reference 18-month equilibrium cycle were also found to be well below the high clad temperature failure criterion.

Westinghouse identified the HFP analysis as limiting for the core coolability criterion, which requires the peak enthalpy to be less than 230 cal/g. Westinghouse states that it has traditionally assumed the peak enthalpy limit to be 200 cal/g. The analysis results show a peak enthalpy of [ ] cal/g ([ ] °C or [ ] °F with [ ] percent melt), which is significantly less than the recommended SRP 4.2 limit of 230 cal/g, and also less than the 200 cal/g internal limit imposed by Westinghouse.

The final analysis reviewed by the staff addresses the changes made to the rod ejection DNBR analyses based on the core design changes documented in the CRR. The corresponding calculation note reviewed showed calculation results for the first cycle and reference 18-month equilibrium cycle. The acceptance criterion, as dictated by SRP 4.2, for high cladding temperature failure by DNB, is that the number of failed rods cannot exceed that number assumed in the dose analysis. For the AP1000, this number corresponds to less than or equal to [ ] percent of the rods in the core. The safety analysis limit for DNBR is [ ] for the AP1000. Westinghouse assumes that [ ] with a hot rod DNBR less than [ ] will fail, and are counted toward the [ ] percent rod failure allowance, which is noted to be conservative because [ ]. The results based on the updated analysis show that the first cycle failure rate is [ ] percent and the equilibrium cycle is [ ] percent, both of which meet the [ ] percent limit with significant margin.

The staff conducted its evaluation in accordance with the guidelines provided in SRP Section 15.4.8, Revision 3, "Spectrum of Rod Ejection Accidents (PWR)," SRP Section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff concluded that the calculations reviewed during the audit supported the applicant's analysis presented in WCAP-17524-P, Revision 1. Additionally, the staff finds that the applicant follows the guidance provided in SRP Sections 15.4.8 and 4.2, Appendix B. Therefore, the staff finds the analysis to be acceptable.

4.6.27. (SRP 15.5.1) Inadvertent Operation of the CMT during Power Operation

The staff reviewed Section 15.5.1 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Editorial changes were made to the inadvertent core makeup tank (CMT) analyses to identify an operator action to open the safety-related reactor vessel head vent to prevent filling the reactor coolant system water solid.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.5.1 through 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and found the changes acceptable and that the regulations continue to be met.

4.6.28. (SRP 15.5.2) CVS Malfunction that Increases Reactor Coolant Inventory

The staff reviewed Section 15.5.2 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Editorial changes were made to the inadvertent chemical and volume control analyses to identify an operator action to open the safety-related reactor vessel head vent to prevent filling the reactor coolant system water solid.

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.5.1 through 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and found the changes acceptable and that the regulations continue to be met.

4.6.29. (SRP 15.6.1) Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

The staff reviewed Section 15.6.1 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

As stated in 10 CFR Part 50, GDC 17, "Electric Power Systems," analysis of coincident loss of ac power for an RCS depressurization event is not required based on the turbine and RCP response to this scenario. With a loss of ac power, the OTΔT is the trip signal. The low pressurizer pressure signal is the actuated protection signal. Since the applicant no longer assumes a loss of offsite power and accompanying primary coolant coastdown in the analysis, the FRACTRAN and VIPRE-01 which were previously used for this portion of the analysis, are no longer used. The staff finds this to be acceptable since it conforms to the new analysis methodology and provides greater clarity.

4.6.30. (SRP 15.6.2) Failure of Small Lines Carrying Primary Coolant Outside Containment

The staff reviewed Section 15.6.2 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The applicant updated the description of the initial iodine and noble gas primary coolant concentrations as based on their respective TS (i.e., equilibrium operating limits) because the TS limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses.

The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.6.2, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.6.31. (SRP 15.6.3) Steam Generator Tube Rupture

The staff reviewed Section 15.6.3 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The applicant revised the analysis to incorporate updates to the NSSS model and also incorporate the resolution to the containment backpressure issue. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased MSSV inlet piping diameter (increased 3 centimeters or 1.2 inches)
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.6.32. (SRP 15.7.4) Radioactive Release from a Subsystem or Component

The staff reviewed Section 15.7.4 in Appendix B of WCAP-17524-P, Revision 1, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- used a 1 percent power measurement uncertainty
- revised the core source terms based on the AFC
- increased the radial peaking factor from 1.65 to 1.75

The changes in the analysis reflect the design information provided in WCAP-17524-P, Revision 1. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," and SRP 15.0.3,

"Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.6.33. (SRP 15.8) Anticipated Transients without Scram

There are no proposed changes in this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in WCAP-17524-P, Revision 1.

5.0 Staff Conclusions

The staff has completed its review of the AP1000 CRR—as described in Reference 1 and supplemented by Reference 2, Reference 3, Reference 10, WCAP-17524-P, Revision 1, and Reference 41—and concludes that the applicant has demonstrated that the improvements to the fuel, core components, and core design continue to comply with the regulatory requirements by following the guidance provided in the respective NUREG-0800 SRP sections, with respect to the AP1000 standard plant design. The staff's conclusions for specific technical topics are found within the respective technical evaluation sections of this report. The staff notes that per Section 5.1 of WCAP-17524-P, Revision 1, any applicant referencing this topical report is required to include all impacts of changes to the methods and models to demonstrate acceptable results for the ECCS. This includes all changes to the methods and models since the submittal of revised LOCA analyses, beginning with the impacts noted in Reference 41.

The staff, therefore, approves the use of the AP1000 CRR (Reference 37) to be referenced by AP1000 COL license holders or COL applicants as desired in accordance with applicable license requirements such as 10 CFR Part 52, Appendix D.

6.0 Conditions and Limitations

The staff's approval of this topical report is specific to the AP1000 generic design. Any use in whole or in part for other designs would require additional applicability review by the staff.

7.0 References

1. WCAP-17524-P, Revision 0, "AP1000 Core Reference Report," dated March 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML120750528).

2. LTR-NRC-12-56, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' To Address the Impact of Thermal Conductivity Degradation on Additional Events," dated August 21, 2012, (ADAMS Accession No. ML12242A287).

3. LTR-NRC-13-26, "Supplemental Information on End-of-Life Seismic/LOCA Calculations for the AP1000 Pressurized Water Reactor," dated April 30, 2013, (ADAMS Accession Nos. ML13128A018 and ML13128A017).

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," dated March 2007 (ADAMS Accession No. ML070810350).

5. LTR-NRC-12-86, "Westinghouse Response to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report," dated January 2, 2013 (ADAMS Accession Nos. ML130080344 and ML13008A267).

6. LTR-NRC-13-3, "Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, "AP1000 Core Reference Repot," dated January 10, 2013 (ADAMS Accession Nos. ML13022A106 and ML13022A105).

7. WCAP-16943-P-A, "Enhanced GRCA Rodlet Design," dated September 2012, (ADAMS Accession No. ML12284A086).

8. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, dated October 2011 (ADAMS dated No. ML11292A021).

9. WCAP-154644-P, Revision 2, "AP1000 Code Applicability Report," dated March 2004 (ADAMS Accession Nos. ML040890149 and ML032671338).

10. LTR-NRC-12-46, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' To Address Thermal Conductivity Degradation," dated October 10, 2012 (ADAMS Accession No. ML12171A311).

11. NRC Information Notice 2011-21 "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," dated December 13, 2011 (ADAMS Accession No. ML113430785).

12. WCAP-17028-P, Revision 6, "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies during Loss of Coolant Accidents," dated June 2010 (ADAMS Accession No. ML102030188).

13. LTR-NRC-13-18, "Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report,'" dated March 2013 (ADAMS Accession Nos. ML13095A124 and ML13095A123).

14. "Audit Summary for Review of WCAP-17524, 'AP1000 Core Reference Report' and Supplemental Information," dated May 2013 (ADAMS Accession Nos. ML13057A940 and ML13057A913).

15. "Audit Summary for Review of Supporting Seismic Response Information for Topical Report WCAP-17524-P, 'AP1000 Core Reference Report,'" dated May 2013 (ADAMS Accession Nos. ML13134A315 and ML13134A300).

16. "Acceptance for Review of Westinghouse Topical Report WCAP-17524-P, Revision 0, AP1000 Core Reference Report," dated June 2012 (ADAMS Accession No. ML12144A201).

17. WCAP-10021-P-A, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," dated August 1983 (ADAMS Archives No. 8308190034).

18. APP-GW-GLR-029P, Revision 3, "AP1000 Spent Fuel Storage Racks Criticality Analysis," dated March 2011 (ADAMS Accession No. ML110670297).

19. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," dated September 2007 (ADAMS Accession No. ML072570321).

20. Final SE for Topical Report WCAP-16045-P, Revision 0, "Qualification of the Two-Dimensional Transport Code Paragon," dated March 2004 (ADAMS Accession No. ML040780402).

21. Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-16045-P-A, Addendum 1, "Qualification of the Nexus Nuclear Data Methodology," dated February 2007 (ADAMS Accession No. ML070320398).

22. WCAP-16045-P-A Addendum 1, "Qualification of the NEXUS Nuclear Data Methodology," dated August 2007 (ADAMS Accession No. ML072570321).

23. WCAP-10965-P-A, Addendum 2-A, Revision 0, "Qualification of the New Pin Power Recovery Methodology," dated September 2010 (ADAMS Accession No. ML103370630).

24. WCAP-10079-P-A, "NOTRUMP—A Nodal Transient Small Break and General Network Code," dated April 1985 (ADAMS Accession No. ML083520343).

25. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," dated August 1985 (ADAMS Accession No. ML100050586).

26. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," dated January 2005 (ADAMS Accession No. ML050910162).

27. WCAP-15806-P-A, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," dated November 2003 (ADAMS Accession No. ML033350109).

28. LTR-NRC-13-43-P, "Supplemental Information Related to Closure of the Corrective Action Identified in the Response to CRR-001," dated June 26, 2013 (ADAMS Accession No. ML13221A156).

29. NUREG/CR-7022, Volume 1, "FRAPCON-3.4: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," dated March 2011 (ADAMS Accession No. ML11101A005).

30. NUREG/CR-7022, Volume 2, "FRAPCON-3.4: Integral Assessment," dated March 2011 (ADAMS Accession No. ML11101A006).

31. NUREG/CR-6534, Volume 1, "FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application," dated October 1997 (ADAMS Accession No. ML092950544).

32. "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code", WCAP-14565-P-A, Addendum 1-A, dated August 2004 (ADAMS Accession Nos. ML042610371 and ML042610368).

33. "Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," WCAP-14565-P-A, Addendum 2-P-A, Revision 0, dated April 2008 (ADAMS Accession No. ML081280713).

34. NRC Information Notice 2009-23: Nuclear Fuel Thermal Conductivity Degradation," dated October 8, 2009 (ADAMS Accession No. ML091550527).

35. W. Wiesenack and T. Tverberg, "Thermal Performance of High Burnup Fuel—In-Pile Temperature Data and Analysis," International Topical Meeting on Light Water Reactor Fuel Performance, dated April 10–13, 2000.

36. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors—Supplement 1," dated April 2006 (ADAMS Accession Nos. ML061220485 and ML061220455).

37. WCAP-17524-P, Revision 1, "AP1000 Core Reference Report," dated March 21, 2014 (ADAMS Accession No. ML14111A418).

38. WCAP-8301-P, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," dated June 30, 1974 (ADAMS Accession No. ML080630438).

39. WCAP- 10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," dated August 1985 (ADAMS Accession No. ML070890332).

40. Enclosure 3 of LTR-NRC-13-81, "Submittal of Presentations from the NRC Audit of Calculations Supporting WCAP-17524, "AP1000 Core Reference Report," dated December 20, 2013, (ADAMS Accession No. ML13353A069).

41. LTR-NRC-14-20, "Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models (Non-Proprietary)," dated April 4, 2014 (ADAMS Accession No. ML14099A531).

42. WCAP-16766-P-A, "Westinghouse Next Generation Correlation (WNG-1) for Predicting Critical Heat Flux in Rod Bundles with Split Vane Mixing Grids,"dated February 28, 2010 (ADAMS Accession No. ML100850528).

43. LTR-NRC-14-75, "Westinghouse Response to NRC RAI Letter No. 3 on WCAP-17524, Revision 1, 'AP1000 Core Reference Report,'" dated November 17, 2014 (ADAMS Accession No. ML14330A235).

44. NUREG-1512, "Final Safety Evaluation Report Related to Certification of AP600 Standard Design," dated September 30, 1998 (ADAMS Accession No. ML081160453).

Section B

Westinghouse

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LTR-NRC-14-14

March 21, 2014

Subject: Submittal of WCAP-17524-P, Revision 1 and WCAP-17524-NP, Revision 1, "**AP1000** Core Reference Report," (Proprietary/Non-Proprietary)

Enclosed are the proprietary and non-proprietary versions of WCAP-17524, Revision 1, "**AP1000** Core Reference Report," dated March 2014, submitted for review and approval under the NRC's licensing topical report program for referencing in licensing actions. This licensing topical report provides information on the Advanced First Core for the **AP1000**® Pressurized Water Reactor (PWR).

Also enclosed is:

1. One (1) copy of the Application for Withholding Proprietary Information from Public Disclosure, AW-14-3928 (Non-Proprietary) with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference AW-14-3928 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

BFMann for

James A. Gresham, Manager
Regulatory Compliance

Enclosures

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AW-14-3928

March 21, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17524-P, Revision 1, "**AP1000** Core Reference Report" (Proprietary)

Reference: Letter from James A. Gresham to Document Control Desk, LTR-NRC-14-14, dated March 21, 2014

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-14-3928 accompanies this Application for Withholding Proprietary Information from Public Disclosure, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of this application for withholding or the accompanying Affidavit should reference AW-14-3928 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

BFMaurer

Bradley F. Maurer, Principal Engineer

Plant Licensing

Enclosures

AW-14-3928

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared Bradley F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Bradley F. Maurer, Principal Engineer
Plant Licensing

Sworn to and subscribed before me
this 21st day of March 2014

Renee' Giampole
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Renee Giampole, Notary Public
Penn Twp., Westmoreland County
My Commission Expires Sept. 25, 2017
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Principal Engineer, Plant Licensing, in Engineering, Equipment and Major Projects, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17524-P, Revision 1, "**AP1000** Core Reference Report" (Proprietary), dated March 2014, for submittal to the Commission, being transmitted by Westinghouse letter, LTR-NRC-14-14, and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of WCAP-17524, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Obtain NRC approval of WCAP-17524, "**AP1000** Core Reference Report".

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of assisting customers in obtaining license changes for the **AP1000** PWR.

(ii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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1 INTRODUCTION - OVERVIEW

1.1 INTRODUCTION

Subsequent to the submittal of Revision 0 of this report, an error was identified in the analysis of the Small Break Loss of Coolant Accident (SBLOCA) which impacted the conclusions of the analysis. As a result of this error, Revision 1 of this report has been prepared in order to update the SBLOCA analysis. Also included in Revision 1 are various minor updates that fulfill commitments made during the review to date. In general, these commitments include: 1) incorporation of minor corrections to the report text as identified in various RAI responses, 2) minor updates resulting for the previous commitment to disposition open items in engineering documents supporting the Chapter 15 analysis update, and 3) addressing the Main Control Room (MCR) dose calculation error. Changes which have been included in the appendices as a part of this revision have been identified with revision bars.

During the time between the initial application for Design Certification (DCD Revision 15) and the issuance of the amended Design Certification (DCD Revision 19), there have been accumulated changes to the plant design and improvements to the fuel, core components and core design that represent a significant positive impact to the core and fuel performance. Consistent with Westinghouse's intention to use the best fuel, core components and core design available to support initial plant startup and future plant operations, the AP1000® Core Reference Report (CRR) provides the detailed description and justification for the Advanced First Core (AFC) loading pattern utilizing improvements in the fuel assembly and core components.

Given these changes and enhancements, Westinghouse proposed in Reference 1-1 a process that would be used to update the standard plant core and fuel design as a post-COL license amendment. This update can also be incorporated in the initial COL Application (COLA), as appropriate. This report, known as the AP1000 CRR, represents the embodiment of the process described in Reference 1-1 as applied to update the amended, certified AP1000 Pressurized Water Reactor (PWR) core and fuel design that was previously documented in Reference 1-2 (DCD Revision 19, hereafter referred to as "the DCD"). The CRR provides all of the information necessary for a licensee to implement these improvements in their plant specific licensing basis.

The CRR describes the updated design of the core and fuel features. The design changes are listed below, and the impact to each area is discussed in detail in the corresponding Sections.

- Implementation of the AFC loading pattern
- Incorporation of the enhanced Gray Rod Cluster Assembly (GRCA)
- Minor modifications to the Rod Cluster Control Assembly (RCCA)
- Incorporation of Robust Protective Grid (RPG)
- Modification to the grid strap heights of the mid grids and intermediate flow mixing grids

In addition to the incorporation of the aforementioned design changes, the CRR also presents a complete update to the accident analyses documented in Chapter 15 of the DCD. This update represents a complete reanalysis of the design basis accidents. This reanalysis incorporates the aforementioned design changes along with explicit consideration of changes incorporated in the certified design that have been evaluated, but not explicitly incorporated in the accident analyses reflected in the DCD Revision 19.

The CRR is divided into 7 sections and 6 appendices. The individual Sections 2 through 6 contain an overview of changes associated with Mechanical Design, Core Design, Thermal and Hydraulic Design, Safety Analyses, and Technical Specifications. Section 7 provides additional information necessary to implement the core and fuel design changes documented in this report as a post-COL licensing amendment, or as part of a COL application. Appendices A-D of this report provide a representative example of the proposed FSAR markups based on the NRC approved DCD. Final versions of the markups will be included in the licensees' submittal to the NRC. Specifically the CRR contents include:

Section 1	This section is an introduction and overview of the CRR.
Section 2	This section describes the mechanical design changes to the fuel assembly and core components.
Section 3	This section provides the Nuclear Design bases for the changes described in the CRR.
Section 4	This section evaluates the impacts of the fuel and core design changes described in Sections 2 and 3 on core thermal characteristics.
Section 5	This section provides the revised safety analysis and evaluation for the accumulated changes described in the CRR.
Section 6	This section provides the justification for revising the Technical Specifications.
Section 7	This section describes the regulatory impacts of the CRR.
Appendix A	Provides a markup of the DCD Chapter 4 to describe the fuel design changes. Appendix A also includes a roadmap identifying the changes made to the DCD and an explanation of the change.
Appendix B	Provides a markup of DCD Chapter 15 to describe the safety analysis changes. Appendix B also includes a roadmap identifying the changes made to the DCD and an explanation of the change.
Appendix C	Provides suggested markups of DCD Chapter 16 to describe changes to the Technical Specifications and Bases.
Appendix D	This Appendix describes the impacts the AFC has on the additional chapters of the DCD.
Appendix E	Provides a clean copy of Chapter 4 without markups.

Appendix F Provides a clean copy of Chapter 15 without markups.

1.2 SUMMARY

In summary, the fuel, core components and core design described in the DCD represent the reference typical core design for the AP1000 PWR. The AP1000 CRR updates and supplements the approved referenced design as provided in the DCD, and establishes a generic licensed design topical suitable for reference in COL applications or for future License Amendment Request (LAR) submitted after the receipt of a COL, or as part of a COL application.

1.3 REFERENCES

1-1. WCAP-16652-NP, "AP1000 Core & Fuel Design Technical Report," October 2006.

1-2. Westinghouse AP1000 Design Control Document Rev. 19, ML11171A500, June 2011.

2 FUEL MECHANICAL DESIGN

2.1 INTRODUCTION

Section 4.2 (Fuel System Design) of the DCD provides detailed descriptions of the mechanical design features of the AP1000 fuel assembly and core components. Except for a few changes identified in the markups to section 4.2 of the DCD provided in Appendix A, these descriptions are consistent with the final design of the AP1000 fuel assembly and core components. These changes are described below and relate to the rod cluster control assembly (RCCA), the gray rod cluster assembly (GRCA), the fuel assembly mid and intermediate flow mixer (IFM) grids, and the fuel assembly protective grid.

2.2 ROD CLUSTER CONTROL ASSEMBLY (RCCA)

The rod cluster control assembly (RCCA) consists of a group of individual neutron absorber rods fastened at the top end to a common spider assembly. The RCCA is inserted and withdrawn from the core by the control rod drive mechanism as necessary to control reactivity. A detailed description of the RCCA design is provided in the DCD (subsection 4.2.2.3.1 and Figures 4.2-9 & 4.2-10). Except for some minor modifications to improve manufacturability and reliability, the final AP1000 RCCA design is consistent with the description provided in the DCD. These modifications have no impact on safety and are identified in the DCD markups provided in Appendix A. These modifications include changes to the spider manufacturing process (instead of joining the vanes, fingers, and spider hub together by welding and brazing, the vanes and fingers are integral with the spider body), the use of stainless steel materials with lower levels of carbon (e.g., use of 304L instead of 304) for improved corrosion resistance, an improved rod to spider locking design, use of either chrome plating or ion nitriding as the rod surface treatment for wear resistance, an increase in the length of the lower reduced diameter absorber section and the addition of a small diameter axial thru-hole in the center of the lower absorber. These last two changes minimize the potential for interaction with the cladding due to irradiation swelling.

2.3 GRAY ROD CLUSTER ASSEMBLY (GRCA)

The gray rod cluster assembly (GRCA) consists of a group of individual low worth neutron absorber rods fastened at the top end to a common spider assembly. The GRCA design described in the DCD (subsection 4.2.2.3.2 and Figure 4.2-11) has twelve rods containing Ag-In-Cd (silver-indium-cadmium) absorber material and twelve stainless steel rods. The final GRCA design for AP1000 PWR uses tungsten as the absorber material, and it is used in all 24 of the rods. These rods have an outside stainless steel cladding similar to the original design. Inside the cladding, the tungsten material is further enclosed in an Alloy-718 sleeve. The use of tungsten as the absorber material will enhance the mechanical and nuclear performance of the control rod (i.e., increased lifetime and constant reactivity) relative to the Ag-In-Cd absorber. Additional details and advantages of the final GRCA design with the tungsten absorber material are described in a NRC approved topical report (Reference 2-1).

In addition to the design changes associated with the GRCA absorber, the GRCA design elements common with the RCCA have also been modified consistent with the changes described in Section 2.2 above for the RCCA. This includes the modifications to the spider, the material changes, the improved rod to spider locking design, and the use of either chrome plating or ion nitriding for rod wear resistance.

2.4 MID AND INTERMEDIATE FLOW MIXER GRIDS

The final fuel assembly mid and intermediate flow mixer (IFM) grid designs are consistent with the design descriptions provided in the DCD except for the grid strap heights which have been increased to provide additional fuel rod fretting wear margin. The mid grid strap height has been increased from 1.50 to 2.25 inches and the IFM grid strap height has been increased from 0.475 to 0.660 inches. The impacts of these changes are addressed in Sections 4 (Thermal Hydraulic Design) and 5 (Accident Analysis).

2.5 PROTECTIVE GRID

The fuel assembly protective grid (P-grid) is positioned in the bottom of the fuel rod bundle just above the bottom nozzle and is used to prevent debris from entering the fuel rod region and to provide support at the end of the fuel rods. The design of the protective grid, which is described in the DCD, is identical to the design used in operating plants. Recently, it was discovered that the protective grids were experiencing fractures in operating plants due to both vibration induced fatigue failures and primary water stress corrosion cracking (PWSCC). To resolve these issues, the design of the protective grid has been enhanced for operating plants as well as for AP1000 plants. This new design, referred to as the Robust Protective Grid (RPG), reduces both the static and dynamic (vibration induced) stresses.

The primary differences between the previous P-grid design and the RPG are summarized below.

- A "Saw Tooth" vibration mitigation feature on both the top and bottom ligaments
- An increased dimple window width to reduce localized stresses
- A decreased dimple window length to increase the material and thus the strength
- Increased inner strap height to accommodate "Saw Tooth" design
- Increased number of grid supports from four to eight to reduce stresses
- Modified outer strap dimple heights to reduce cell size outages

Figure 2-1 provides an illustrative comparison of the inner grid straps between the previous P-grid design and the RPG design. Figure 2-2 shows the difference in the number of supports used on the previous P-grid design versus the RPG design.

The final RPG underwent a significant amount of testing which specifically examined the ligament vibration and showed a sufficient amount of margin was gained with the new RPG design. In addition, pressure drop testing was performed which showed that the RPG has the same pressure drop as the previous P-grid design. Finally, debris filtering tests were run which demonstrated that with respect to the debris mitigation capability, the RPG performed essentially the same as the standard P-grid. In conclusion, the RPG has been demonstrated to have the same pressure drop and debris mitigation capabilities as the previous standard P-grid.

It is worth noting that the RPG has been successfully employed in a number of operating plants and was licensed following the approved FCEP process in which the NRC was formally notified of the design change via letter; LTR-NRC-11-26 (Reference 2-2).

2.6 IMPACTS OF PROTECTIVE GRID DESIGN CHANGE ON GSI-191

The design change to the protective grid was evaluated with respect to Generic Safety Issue (GSI) 191(Reference 2-3). Prior to this design change, Westinghouse performed an extensive series of tests over several years to quantify the effect of fibrous and particulate debris and containment chemical reaction products on the head-loss across the fuel assemblies of an AP1000 PWR during a postulated loss of coolant accident (LOCA), as documented in WCAP-17028 (Reference 2-4). This testing program included multiple reviews and meetings with both the NRC and ACRS. The testing program covered a spectrum of different flow rates, debris quantities, debris types, methods of debris addition and other experimental variables, in addition to hot and cold leg breaks, to ensure that the most challenging conditions were tested. [

]^{a,c} These tests also demonstrated that the AP1000 design provides considerable margin in the long term core cooling analysis of the AP1000 plant following a loss of coolant accident.

[

]^{a,c} Therefore, the RPG design

[

]^{a,c} such that the GSI-191 performance was expected to be the same as for the previous P-grid design. To confirm this expectation and thereby verify the acceptability of the RPG, [

]^{a,c} as documented in WCAP-17028 (Reference 2-4). These [ ]^{a,c} demonstrated that, the RPG performed as well if not better than the standard P-grid.

In conclusion, [

]^{a,c} the RPG can be implemented for the AP1000 fuel assembly with no adverse impact on the GSI-191 debris issue.

2.7 REFERENCES

2-1. Conner, M.E., et al., "Enhanced GRCA Rodlet Design," WCAP-16943-P-A (Proprietary) and WCAP-16943-NP-A (Non-Proprietary), September 2012.

2-2. Gresham, J.A., "Fuel Criterion Evaluation Process (FCEP) Notification of the Robust Protective Grid (RPG) Design (Proprietary/Non-Proprietary)," LTR-NRC-11-26, June, 2011.

2-3. US NRC Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance."

2-4. Ruth, K.L., et al., "Evaluation of Debris-Loading Head-Loss Tests for AP1000 Fuel Assemblies during Loss of Coolant Accidents," WCAP-17028-P (Proprietary) and WCAP-17028-NP (Non-Proprietary), June 2010.

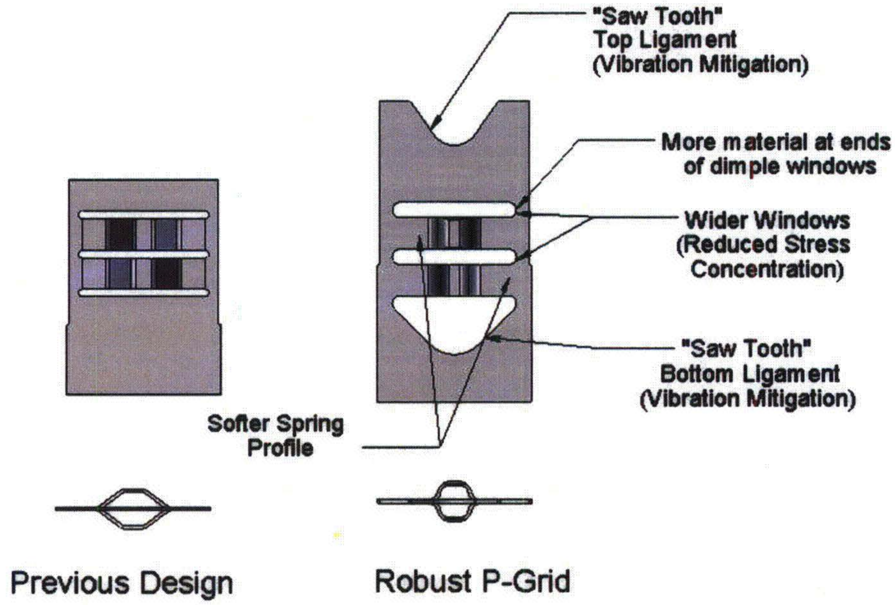

Figure 2-1 Comparison of the Previous Protective Grid and the RPG

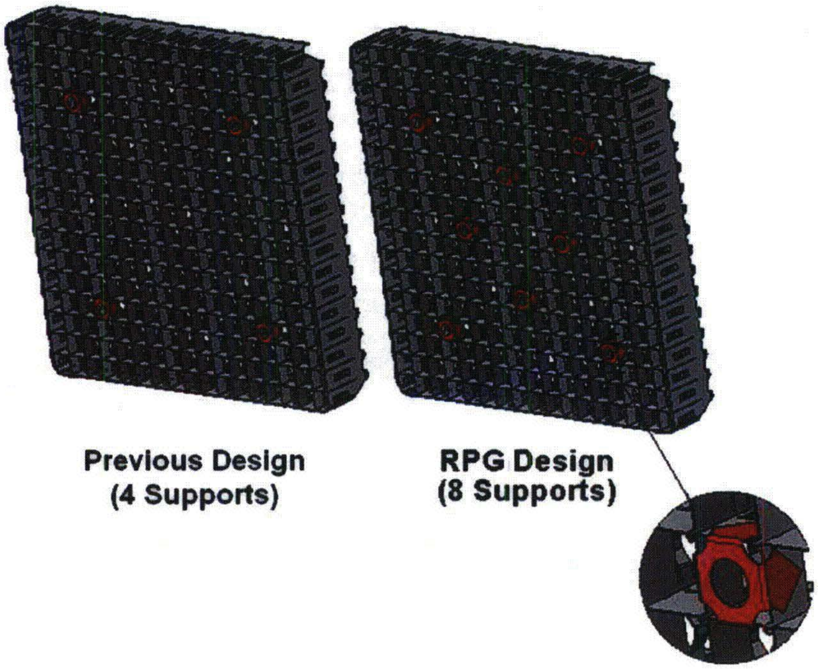

Figure 2-2 Previous Design (4 Supports) and RPG Design (8 Supports)

3 NUCLEAR DESIGN

3.1 INTRODUCTION

Section 4.3 of the DCD describes a loading pattern hereafter referred to as "the DCD core design". The most significant design change being incorporated as part of this report is the change in the first cycle loading pattern. This section provides a summary of the components of the Advanced First Core (AFC) loading pattern design and the implications of the incorporation of this design into the AP1000 plant.

With respect to the nuclear design, discussed in section 4.3 of the DCD, the incorporation of the AFC loading pattern envelops the following changes:

1. the first cycle loading pattern; implementation of the AFC fuel management strategy,
2. the discrete burnable absorber design; utilization of the Wet Annular Burnable Absorber (WABA) rather than the Pyrex glass based absorbers,
3. the GRCA design; utilization of the enhanced, tungsten based GRCA rather than the Ag-In-Cd based design currently described in the DCD,
4. and the $F_{\Delta H}^{N}$ limit; increased from 1.65 to 1.72 [ ]^{a,c}

3.2 CHANGE TO THE FIRST CYCLE LOADING PATTERN

The DCD describes the first cycle loading pattern as a traditional 3-region design. This design is based on decades old fuel management technology and results in the placement of the highest enriched region of fuel on the core periphery. Key disadvantages of this type of fuel management strategy are known to include:

- Non-optimal uranium utilization, especially in fuel discharged after one cycle
- Challenging fuel management for early transition cycles (e.g., Cycle 2)
- High radial neutron leakage

Based on customer requests for improved uranium utilization and the requirements for plant standardization, Westinghouse developed the AFC design, which is illustrated in quarter core geometry in Figure 3-1 (lower-right quadrant). The AFC design was developed to more closely resemble a modern, reload core design using a low leakage loading pattern (L^3P) strategy. This is accomplished by using a 5-region first core design, [ ]^{a,c} Additionally, a combination of Integral Fuel Burnable Absorbers (IFBA), WABA, and intra-assembly radial enrichment zoning are used [ ]^{a,c}

The key advantages associated with incorporating the AFC design into the AP1000 plant are: significantly improved first core fuel cycle economics, first core operational characteristics [ ]^{a,c} that more closely resemble a reload core design, and a simpler and more rapid transition to equilibrium reload fuel management. As summarized in Table 3-1 and demonstrated in more detail in

the revision to Section 4.3 of the DCD included with this report, these advantages are obtained while maintaining the key core characteristics within the envelope of key safety parameters associated with the plant safety analyses. Even though the DCD Chapter 15 Safety Analysis has been revised as described in Section 5 of this report, the revision to the first core design itself did not result in a need for such a reanalysis.

3.3 CHANGE TO BURNABLE ABSORBER DESIGN

The AFC also utilizes a combination of IFBA and discrete burnable absorbers to optimize core reactivity and power distribution characteristics. The discrete burnable absorber design has been changed to implement the WABA design rather than the Pyrex design used in the DCD core design. The option to use WABA rather than Pyrex burnable absorbers was previously described in DCD Section 4.2 and was generally approved for use as documented in WCAP-10021-P-A, Revision 1 (Reference 3-1). Should discrete burnable absorbers be needed, future core designs will only consider the WABA design since this is the more modern technology.

As additional information relative to the use of burnable absorbers used in the AFC design, all fuel rods containing IFBA utilize an axial absorber coating of 152 inches in length and centered within the active fuel stack. The remaining top and bottom 8 inches of the IFBA fuel rod consists of 3.2 w/o annular pellets. The WABA in the AFC design are of three different lengths and are offset within the fuel assemblies to provide fine tuning of the axial power shape versus cycle exposure. [

]^{a,c}

3.4 CHANGE TO GRAY ROD CLUSTER ABSORBER (GRCA) DESIGN

The DCD core design utilizes a GRCA design consisting of 12 rods of Ag-In-Cd and 12 rods of stainless steel alloy 304 (SS-304). As part of the implementation of the advanced first core design, this GRCA design has now been replaced with a tungsten-based, enhanced GRCA design as previously described in Section 2 of this report. As such, all revised nuclear analyses performed for the advanced first core explicitly consider this enhanced GRCA design in the associated core models.

3.5 CHANGE IN ENTHALPY RISE HOT CHANNEL FACTOR LIMIT

As described in Sections 4 and 5 of this report, the revised thermal-hydraulic and safety analyses for the AP1000 plant were performed assuming an increased $F_{\Delta H}^{N}$ limit of 1.72. This is compared to the limit previously assumed in the DCD of 1.65. This change was proposed as part of the revised nuclear design and as such is associated with the revised first cycle core design. [

]^{a,c}

Sections 4 and 5 demonstrate that the increased $F_{\Delta H}^{N}$ limit of 1.72 is supported by the revised thermal-hydraulic design analysis and safety analysis. These analyses continue to follow the approved design methodologies as required to change Technical Specification parameters defined in the COLR.

3.6 CONCLUSION

The first core design for the AP1000 PWR has been modified as described in the revised DCD Chapter 4 provided with this report. The basis for these changes has been summarized above. Key benefits associated with the revised first core design include both improved fuel cycle economics and improved reload core design considerations. It is noteworthy that these benefits have been realized through an advanced core design that does not result in significant changes to the key safety analysis input parameters or changes to the conclusions of the thermal-hydraulic design and safety analysis conclusions previously documented in the DCD.

3.7 REFERENCES

3-1. WCAP-10021-P-A, Revision 1, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," October 1983.

Table 3-1 AP1000 PWR Comparison of Reactivity Parameters and Peaking Factors for the DCD and the Final Advanced First Core Designs, Best Estimate Value

Nuclear Design Parameter	The DCD Design, Typical BE Value	Final AFC Design, Typical BE Value	SAC Limit
Doppler-only Power Coefficient (pcm/% power)	-13.3 to -8.7 (Most-negative) -11.3 to -8.4 (Least-negative)	-14.6 to -9.0 (Most-negative) -12.4 to -8.9 (Least-negative)	-19.4 to -12.6 -10.2 to -6.7
Doppler Temperature Coefficient (pcm/°F)	-2.1 to -1.3	-2.1 to -1.4	-3.5 to -1.0
Moderator Temperature Coefficient (pcm/°F)	0 to -35	same as DCD design	0 to -40
Boron Coefficient (pcm/ppm)	-10.5 to -6.9	-11.3 to -7.2	-13.5 to -5.0
ARO $F_{\Delta H}$ (BOL / EOL)	1.40 / 1.33	1.44 / 1.38	1.72
HFP, ARO, No Xenon Boron Concentration (ppm)	1184	1160	--
HFP, ARO, Equilibrium Xenon Boron Concentration (ppm)	827	844	--

	H	G	F	E	D	C	B	A
8	B -- / --	D 12W / 68I	B -- / --	D 12W / 68I	B -- / --	D 12W / 68I	E-124 -- / 124I	C -- / --
9	D 12W / 68I	B -- / --	D 12W / 68I	B -- / --	D 12W / 68I	B -- / --	E-88 4W / 88I	A -- / --
10	B -- / --	D 12W / 68I	B -- / --	D 12W / 68I	B -- / --	E-124 8W / 124I	C -- / --	
11	D 12W / 68I	B -- / --	D 12W / 68I	B -- / --	E-124 8W / 124I	C -- / --	A -- / --	
12	B -- / --	D 12W / 68I	B -- / --	E-124 8W / 124I	B -- / --	C -- / --		
13	D 12W / 68I	B -- / --	E-124 8W / 124I	C -- / --	C -- / --			
14	E-124 -- / 124I	E-88 4W / 88I	C -- / --	A -- / --				
15	C -- / --	A -- / --						

Assembly Type
 # WABA / # IFBA

Region A (0.740 w/o)

Region B (1.580 w/o)

Region C (3.200 w/o)

Region D (3.776 w/o)

Region E (4.376 w/o)

Enrichments are Central Zone Assembly Average

Figure 3-1 AP1000 PWR Advanced First Core (AFC) Loading Pattern

4 THERMAL HYDRAULIC DESIGN

4.1 INTRODUCTION

Chapter 4 of the DCD has been revised primarily to support the implementation of the Advanced First Core (AFC) design. Concurrent with the implementation of the AFC, however, additional modifications to the plant design and safety analysis have also been incorporated, as appropriate. With respect to the thermal-hydraulic analysis, as originally reflected in section 4.4 of the DCD, the following design changes have now been incorporated and further analyzed:

1. Revised Cycle 1 loading pattern. The implementation of the AFC does not directly impact the thermal-hydraulic analysis as is further described in Section 4.2 below.

2. Implementation of WABA. Use of WABA has been approved by the USNRC as documented in Reference 3-1 earlier and as had been previously described in section 4.2 of the DCD. For the AFC, WABA is now used in place of Pyrex. The thermal-hydraulic evaluation confirms all applicable design criteria including the core bypass flow are met with the use of WABA.

3. Implementation of enhanced (tungsten) GRCA. The change in GRCA absorber material from Ag-In-Cd to tungsten is described in Reference 2-1 earlier. The thermal-hydraulic evaluation confirms all applicable design criteria including the core bypass flow are met with the use of GRCA.

4. Increase in $F_{\Delta H}^{N}$ limit. The $F_{\Delta H}^{N}$ design limit, including measurement uncertainties, was increased from 1.65 to 1.72 and its impact on the thermal-hydraulic analysis is described in Section 4.3 below.

5. Implementation of alternate DNBR correlations. The DNBR correlations used in the revised thermal-hydraulic analysis now include WRB-2M, WRB-2, ABB-NV, and WLOP as described in Section 4.4 below.

6. Implementation of fuel assembly enhancements. As denoted in Section 2 earlier, enhancements were made for the AP1000 fuel design relative to that described in the DCD. Impacts of these fuel design enhancements on thermal-hydraulic design are described in Sections 4.5 and 4.6 below.

4.2 IMPLEMENTATION OF AFC DESIGN

The Advanced First Core design does not directly affect the bounding power distribution inputs used in the DCD thermal-hydraulic analysis. Power distribution analyses specific to the AFC were performed to confirm the bounding nature of the $F_{\Delta H}^{N}$ and axial power distribution assumptions made in the thermal-hydraulic analysis.

4.3 CHANGE IN ENTHALPY RISE HOT CHANNEL FACTOR LIMIT

As first noted in Section 3, the $F_{\Delta H}^{N}$ design limit was increased from 1.65 to 1.72. The corresponding Revised Thermal Design Procedure (RTDP) $F_{\Delta H}^{N}$ limit is thus increased from 1.590 to 1.654, as the 4%

measurement uncertainty is statistically incorporated into the RTDP design DNBR limit. It has been demonstrated that the revised DNBR Core Limits with the increased $F_{\Delta H}^{N}$ limit will continue to provide adequate plant operating flexibility while maintaining the DNBR design basis under various Condition II accident scenarios. As described in Section 5.2 below, the revised $F_{\Delta H}^{N}$ limit was also considered in the re-analyses of the DCD Chapter 15 transients and the DNBR criteria remain satisfied.

4.4 IMPLEMENTATION OF ALTERNATE DNB CORRELATIONS

The DNBR methodology remains unchanged from that discussed in the DCD with the WRB-2M correlation as the primary DNB correlation. However, additional / enhanced DNB correlations were used in the revised analysis for conditions where the WRB-2M correlation is not applicable, as follows:

- The ABB-NV correlation has been added as a replacement for the W-3 correlation. It is used as a supplemental correlation under conditions where the primary correlation (WRB-2M) was not applicable. These conditions occur when the fuel region below the first mixing vane grid may become DNB limiting with a severely bottom skewed axial power distribution, such as the conditions during the Rod Withdrawal from Subcritical accident. The ABB-NV correlation was developed based exclusively on rod bundle data and provides a more accurate prediction of DNBR margin than the W-3 correlation. The approval of the ABB-NV correlation by the USNRC is documented in Reference 4-1.

- The WLOP correlation has been added for use in the analysis of Hot Zero Power (Low Flow) Steamline Break event. The WLOP correlation developed from rod bundle data is applicable specifically in the range of fluid conditions associated with this accident event (i.e., low pressure and low flow). The approval of the WLOP correlation by the USNRC is documented in Reference 4-1.

4.5 APPLICABILITY OF DNB CORRELATIONS

As stated in section 4.4.2.2.1 of the DCD (Rev. 19), the primary DNB correlation used for the analysis of the AP1000 fuel is the WRB-2M correlation. WRB-2M was developed for the modified low pressure drop (MLPD) mid grid design and the modified intermediate flow mixer (MIFM) grid design as documented in WCAP-15025-P-A (Reference 4-2).

The AP1000 mid and IFM grid designs have been enhanced relative to the MLPD and MIFM grid designs to improve fretting wear resistance of the AP1000 fuel assembly. The primary difference between the AP1000 grids and MLPD/MIFM grids is the grid strap height. The AP1000 mid-grid height increases from 1.50 inch to 2.25 inch relative to MLPD, while the AP1000 IFM height increases from 0.475 inch to 0.66 inch relative to MIFM. The fundamental characteristics of the MLPD/MIFM grids – diagonal mid grid spring with closed window, specific mixing vane design, smaller dimples, fewer vanes around the thimble tube, quadrant-symmetric vane pattern (Reference 4-2) – have been maintained for the AP1000 mid and IFM grid designs.

The AP1000 mid and IFM grids have the same DNB performance as the MLPD and MIFM grids for the following reasons:

- [
]a,c

- [
]a,c

- [
]a,c

In the application of the WRB-2M DNB correlation to the AP1000 fuel design, the conditions outlined in the SER "Conclusions" section of the WRB-2M WCAP (Reference 4-2) are maintained.

Based on the discussions above, the WRB-2M DNB correlation is applicable to the AP1000 fuel design. In addition to WRB-2M, there are other DNB correlations used for the AP1000 fuel design. The WRB-2, ABB-NV, and WLOP DNB correlations are used whenever the WRB-2M correlation is not applicable. All three of these DNB correlations are applicable to the AP1000 fuel design. WRB-2 (Reference 4-4) can be conservatively applied in place of WRB-2M (Reference 4-2). ABB-NV and WLOP can be applied to any 17x17 fuel product with a 0.374 inch fuel rod OD (Reference 4-1), which includes the AP1000 fuel design. Therefore, with respect to the AP1000 fuel design, ABB-NV can be applied below the first mixing vane grid and WLOP can be applied under low pressure conditions.

4.6 IMPLEMENTATION OF FUEL DESIGN ENHANCEMENTS

As described in Section 2 above, a modified protective grid design was incorporated into the AP1000 fuel assembly mechanical design. Additionally, the strap heights of the mid and IFM grids were increased as discussed in Section 4.5. The strap height increase results in slightly higher grid hydraulic loss coefficients, while the modified protective grid design change has no adverse effect on the thermal-hydraulic design. As part of the revised thermal-hydraulic design described in the revised DCD Chapters 4 and 15 provided in Appendices A and B, the thermal-hydraulic design models were updated to accurately reflect the characteristics of the final fuel assembly mechanical design. As demonstrated in these revised DCD chapters, the DNB design basis continues to be met and the conclusions of all DNB sensitive safety analysis events are unchanged.

4.7 REFERENCES

4-1. "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," WCAP-14565-P-A, Addendum 2-A, April 2008.

4-2. "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-P-A, April 1999.

4-3. "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994.

4-4. "Reference Core Report – VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985, "VANTAGE 5H Fuel Assembly," WCAP-10444-P-A, Addendum 2-A, February 1989.

Table 4-1	Assessment of AP1000 Grids versus MLPD/MIFM Grids using FCEP Parameters		
#	**Parameter**	**AP1000 PWR Value**	**Comparison of AP1000 grids to MLPD/MIFM grids**
1	Rod Diameter	0.374"	Same
2	Thimble Tube Diameter	0.482"	Same, MLPD/MIFM grids have either 0.474" or 0.482" thimble tube
3	Rod Pitch	0.496"	Same
4	Heated Length	14'	Like MLPD/MIFM, within range of applicability of WRB-2M
5	Gridded to Ungridded Cell Flow Area	0.755 mid grid, 0.758 IFM	Within MLPD/MIFM range of 0.755 to 0.771
6	Grid Spacing (centerline-to-centerline axial distance)	10.15" to 20.3"	Like MLPD/MIFM, within range of applicability of WRB-2M
7	Ratio of vane area to flow area	Same as MLPD/MIFM	Same – AP1000 grids use the same vanes as MLPD/MIFM
8	Vane orientation	Same as MLPD/MIFM	Same – AP1000 grids use the same vane pattern as MLPD/MIFM
9	Azimuthal extension of the vanes around the rod circumference	Same as MLPD/MIFM	Same – AP1000 grids use the same vanes as MLPD/MIFM

5 ACCIDENT ANALYSIS

5.1 INTRODUCTION

The accident analyses in AP1000 Design Control Document (DCD) Revision 19 (Reference 1-2) have been revised to assess the impact of the Advanced First Core (AFC) design changes and to incorporate the impact of the design changes incorporated in the certified design that have been evaluated, but not explicitly incorporated in the accident analyses reflected in DCD Revision 19. The impact of the revised accident analysis on DCD Chapter 15 is provided in Appendix B.

The following sections provide additional details regarding the content provided in the revised accident analyses in this report. Section 5.2 discusses the basis for the revised accident analyses, including identification of significant changes incorporated as a result of the AFC and AP1000 design finalization. A comparison of the event frequency designations and acceptance criteria used in the DCD and maintained in this report versus that identified in NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 15, Revision 3 (Reference 5-1) is provided in Section 5.3. The comparison provides the basis for retention of the existing DCD event frequency designations. In addition, Section 5.4 identifies changes incorporated into Section 15.4.8 of Appendix B of this report necessary to demonstrate compliance with NRC guidance provided in NUREG-1793 Supplement 2, Section 15.2.4.8.2.

It is recognized that changes in the methods and models to demonstrate acceptable results for the emergency core cooling system may occur. Reportability requirements for the impact of these changes on LOCA Peak Clad temperature (PCT) are addressed in 10 CFR 50.46 for holders of an operating license or construction permit. As the analyses presented in Appendices B and F are representative, changes which may occur during the review process of this report will not be explicitly addressed in the appendices (Appendices B and F) to this report. These changes will, however, be dispositioned in the results presented in connection with a site specific application of this report.

5.2 NON-LOCA AND LOCA BASES FOR ANALYSES

The accident analyses reflected in DCD Revision 19 have been revised to support the AFC design and explicitly model the AP1000 plant design with design changes incorporated in the certified design that have been evaluated, but not explicitly incorporated in the accident analyses reflected in DCD Revision 19. The design changes incorporated also include the rod drop time increase. The impact of the revised accident analyses on DCD Chapter 15 is provided in Appendix B. The analyses performed use the same methodology as those presented in the DCD with the exception of the spectrum of rod ejection events (section 15.4.8 of the DCD), and the large break loss of coolant accident analysis (Section 15.6.5.4A of the DCD). The rod ejection methodology is discussed specifically in Section 5.4 of this report.

The following specific changes are addressed in the revised CRR accident analyses:

1. Revised Cycle 1 loading pattern. The AFC design itself does not require a complete reanalysis of the Chapter 15 transients since it does not result in any significant changes to the nuclear design Safety Analysis Checklist (SAC) inputs assumed in the transient analyses. However, since the following additional changes were also being incorporated into a complete set of revised analyses, all SAC

parameters were reassessed against the AFC design as well as potential reload cycle fuel management characteristics in order to ensure the robustness of the revised analyses. Additionally the core and coolant source terms were recalculated based on the revised cycle 1 loading pattern for use in the radiological analyses.

2. Implementation of WABA. Use of WABA has been approved by the USNRC in 1983 as documented in Reference 3-1. The use of WABA has no bearing on the SAC parameters assumed in the accident analyses.

3. Implementation of enhanced (tungsten) GRCA. The change in GRCA absorber material from Ag-In-Cd to tungsten does not have any bearing on the SAC parameters assumed in the accident analyses. In addition, since the geometry of the GRCA has not changed from that assumed in the DCD, and remains consistent with the RCCA geometry, there is no impact on the non-LOCA and LOCA transient analysis models.

4. Implementation of the Robust Protective Grid (RPG) and the improved RCCA design as described in Section 2. Since the RPG has the same pressure drop as the previous protective grid design and the geometry of the RCCA has not changed from that assumed in the DCD, there is no impact on the accident analyses from these changes.

5. Increase in $F_{\Delta H}^{N}$ limit. The normal operation limit on $F_{\Delta H}^{N}$ was increased from 1.65 to 1.72. All Chapter 15 transient analyses have been updated to consider the increased $F_{\Delta H}^{N}$ limit of 1.72

6. Fuel assembly Intermediate Flow Mixing (IFM) and mid-grid height changes. This change increased the IFM inner grid strap height from 0.475 inch to 0.660 inch and the mid-grid strap height from 1.5 inches to 2.25 inches to accommodate a larger dimple contact area to provide fretting wear margin. This resulted in an increased core pressure loss coefficient. The Chapter 15 accident analyses were reanalyzed to explicitly account for the impact of this change.

7. RCCA drop time accident analysis input. This design change increased the accident analysis assumed rod drop time to the dashpot from 2.47 seconds to 2.7 seconds. The rod drop time is an important parameter for the following transient events which were reanalyzed:

   - Complete Loss of Flow
   - Turbine Trip
   - Locked Rotor
   - Rod Ejection

8. During the design finalization of the AP1000 PWR, a number of design changes have been incorporated into the certified design. As part of the design certification, the changes were evaluated individually to quantify the impact on the accident analyses and confirm the conclusions of the analyses remained unchanged, but were not explicitly incorporated into the DCD accident analyses. The Core Reference Report analyses represent an integrated response to the design changes. It is important to note that these design changes have already been approved via issuance of NUREG-1793 Supplement 2 (Reference 5-5).

9. Specific to Revision 1: Revised single failure for Small Break Loss-of-Coolant Accident Analysis. The SBLOCA analysis transients have been reanalyzed to account for the failure of one automatic depressurization system stage 4 (ADS-4) valve on the non-pressurizer side of the reactor coolant system (RCS) rather than a failure of one ADS-4 valve on the pressurizer side of the RCS as previously assumed in the DCD. Additionally, a subset of the SBLOCA transients have incorporated a time-dependent containment pressure response generated from the WGOTHIC containment model simulations. The transient results are presented in subsection 15.6.5. Note that the impact of thermal conductivity degradation (TCD) on the SBLOCA results was originally assessed in the supplemental information provided in Reference 5-9, Reference 5-10, and Reference 5-11. Similarly, the revised SBLOCA analysis results were assessed, and it was concluded that the SBLOCA analysis is negligibly impacted when considering the effects of TCD.

10. Specific to Revision 1: Revised Large Break Loss-of-Coolant Accident (LBLOCA) Analysis. A plant-specific adaptation of the ASTRUM methodology was applied to explicitly consider the burnup-dependent aspects of fuel performance changes considering thermal conductivity degradation and peaking factor burndown. The revised LBLOCA results are presented in Section 15.6.5.4A. The plant-specific adaptation of the methodology is described in Reference 5-10.

The revision of the accident analysis to reflect changes associated with the AFC and AP1000 design finalization has been completed and all analysis results meet the established formal acceptance criteria.

It is noted that Section 15.5.6 of Appendix B does not provide the total LOCA main control room operator dose or the individual contributions for direct radiation from adjacent structures and for sky shine. Similarly, Section 6.4.4 in Appendix D does not provide the main control room operator doses. These doses will be provided in the future as they are dependent upon design changes that are outside of the scope of the core reference report. However, the impact on the MCR operator dose of the adoption of the AFC has been calculated and is reflected in the dose contribution for the airborne activity entering the main control room shown in Section 15.5.6 of Appendix B for a LOCA, which increases negligibly, by only +0.01 rem.

5.3 COMPARISON OF STANDARD REVIEW PLAN (SRP) REVISION ACCIDENT CATEGORIZATIONS

The DCD Chapter 15 accident analyses in Reference 1-2 describe the analyses of various design-basis transients and accidents to demonstrate compliance of the AP1000 design with the acceptance criteria for these events. The acceptance criteria for the various events are based on meeting the relevant regulations and general design criteria (GDC) of GDC 10, GDC 15 and Title 10, Section 50.46 of the Code of Federal Regulations (10 CFR 50.46) and are a function of the anticipated frequency of occurrence of the event and potential radiological consequences to the public. As such, each design-basis event is categorized accordingly based on these considerations. The event frequency grouping used in Reference 1-2 is consistent with the guidelines of Regulatory Guide 1.70 (Reference 5-2) and the criteria of American Nuclear Society (ANS) 18.2 (Reference 5-3). The four event frequency categories used in Reference 1-2 are as follows:

- Condition I:Normal operation and operational transients
- Condition II:Faults of moderate frequency

- Condition III:Infrequent faults
- Condition IV:Limiting faults

Reference 5-1 provides the initiating events categorized as either an Anticipated Operational Occurrence (AOO) or as a Postulated Accident. It indicates that AOOs refer to events that are categorized in Reference 5-2 and Regulatory Guide 1.206 (Reference 5-4) as incidents of moderate frequency and infrequent events which are also known as Condition II and Condition III events, respectively, in Reference 5-3. Postulated Accidents are identified in Reference 5-1 as unanticipated occurrences and are also known as Condition IV events in Reference 5-3.

The Reference 1-2 acceptance criteria for Condition II events are the same as that listed in Reference 5-1 for AOOs. Similarly, the Reference 1-2 acceptance criteria for Condition IV events are the same as that listed in Reference 5-1 for Postulated Accidents. However, the acceptance criteria for Condition III events evaluated in Reference 1-2 are not identical to that specified in Reference 5-1. Therefore each Condition III event will be evaluated to demonstrate compliance with acceptance criteria of Reference 5-1.

The following provides a brief discussion as to why the current acceptance criteria for Reference 1-2 Condition III events meet the acceptance criteria of Reference 5-1.

- Steam system piping failure (minor) (see subsection 15.1.5)
    - This event is categorized as an AOO in Reference 5-1. The Reference 1-2 acceptance criteria for a minor steam line break are consistent with a Condition II event and therefore the current analysis meets the requirements of Reference 5-1.
- Complete loss of forced reactor coolant flow (see subsection 15.3.2)
    - This event is categorized as an AOO in Reference 5-1. The Reference 1-2 acceptance criteria for a loss of forced reactor coolant flow are consistent with a Condition II event and therefore the current analysis meets the requirements of Reference 5-1.
- RCCA misalignment [single RCCA withdrawal at full power] (see subsection 15.4.3)
    - Reference 1-2 Section 15.4.3 states, "No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full-power operation. The operator could withdraw a single RCCA in the control bank because this feature is necessary to retrieve an assembly should one be accidentally dropped. The event analyzed results from multiple wiring failures or multiple significant operator errors and subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage." Since multiple system failures or multiple operator errors are necessary for this event to occur, the event is not considered credible and, therefore, the potential for fuel failure is judged to be remote. Consequently, the analysis presented in Reference 1-2 Section 15.4.3 should be considered bounding.

- Inadvertent loading and operation of a fuel assembly in an improper position (see subsection 15.4.7)
  - Reference 5-1 classifies these events as an AOO. Reference 1-2, Section 15.4.7.3 states, "In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes." Therefore, the acceptance criteria of Reference 5-1 will be met.
- Inadvertent operation of automatic depressurization system (see subsection 15.6.1)
  - While this event is not directly specified in Reference 5-1, the event characteristics (for the initial portion of it) are similar to the inadvertent opening of the pressurizer safety value, which is described in the SRP as an AOO. The Reference 1-2 acceptance criteria for an inadvertent operation of the ADS are consistent with a Condition II event and therefore the current analysis meets the requirements of Reference 5-1.
- Loss-of-coolant accidents (LOCAs) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (small break) (see subsection 15.6.5)
  - The acceptance criteria for this event for both in Reference 1-2 and Reference 5-1 are to meet the 10CFR 50.46 criteria. Therefore the current analysis meets the requirements of Reference 5-1.
- Radiological release events in Chapter 15.7 (not Fuel Handling)
  - Section 15.7.1, 15.7.2 and 15.7.3 all identify that, "The Standard Review Plan no longer includes this event as part of the review. Therefore, no analysis is provided." As such, no change is needed to this section.

As discussed above, the current Chapter 15 acceptance criteria are consistent with that currently specified in Reference 5-1. Therefore, the Accident Analyses presented in Appendix B continue to use the Condition I through IV event frequency designations provided in the DCD.

5.4 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

While neither the implementation of the AFC design nor the incorporation of previously described design changes into the revised accident analyses required a significant change to the analyses associated with the spectrum of rod ejection scenarios, it is noted that NUREG-1793 Supplement 2 (Reference 5-5), Section 15.2.4.8.2 states that:

> *The* [NRC] *staff will evaluate the reactivity-initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a*

change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapter 4, "Reactor," or Chapter 15, "Transient and Accident Analysis."

As a result, the analysis presented in Section 15.4.8 of Appendix B to this report has been updated to address compliance with the acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3 and the criteria are summarized below:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2 Revision 3 Appendix B.

- The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.

- For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g. DNBR).

- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.

- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.

- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

- Peak reactor coolant pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.

Additionally, SRP Section 4.2, Revision 3, Appendix B requires the enthalpy increase be considered in the source term generated for the dose analysis, and presents an equation to use for this purpose. More recent NRC guidance, i.e. Draft Guide 1199 (Reference 5-6) and the subsequent clarification to DG-1199 (Reference 5-7) expand upon the SRP 4.2 Revision 3 requirements, changing the pre-accident gap fractions and the increased gap activity due to a reactivity insertion event. The changes to the gap fraction were incorporated into the rod ejection dose analysis.

The AFC was analyzed in accordance with Reference 5-8 to determine acceptability with respect to these criteria. Reference 5-8 is generally applicable to all Westinghouse reactors, and describes the 3D methods to analyze the rod ejection transient. The complete analysis and summary of conclusions are presented in Section 15.4.8 of Appendix B of this document. Section 15.4.8 demonstrates that all of the above criteria are met for the AP1000 PWR.

5.5 REFERENCES

5-1. NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Revision 3, March 2007.

5-2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

5-3. American National Standards Institute N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.

5-4. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

5-5. NUREG-1793 Supplement 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design", Docket No. 52-006, ADAMS Accession No. ML112061231.

5-6. Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009, ADAMS Accession No. ML090960464.

5-7. "Technical Basis for Revised Regulatory Guide 1.183 (DG-1199) Fission Product Fuel-To-Cladding Gap Inventory," July 26, 2011, ADAMS Accession No. ML111890397.

5-8. Beard, C. L. et. al, "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics", WCAP-15806-P-A (Proprietary) and WCAP-15807-NP-A (Nonproprietary), November, 2003.

5-9. Gresham, J.A., "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' to Address the Impact of Thermal Conductivity Degradation on Additional Events (Proprietary)," LTR-NRC-12-56, August 2012.

5-10.Gresham, J.A., "Westinghouse Response to NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report' (Proprietary/Non-Proprietary)," LTR-NRC-12-86, January 2013.

5-11.Gresham, J.A., "Westinghouse Response to Supplemental NRC RAIs on WCAP-17524, 'AP1000 Core Reference Report' (Proprietary/Non-Proprietary)," LTR-NRC-13-18, March 2013.

6 TECHNICAL SPECIFICATIONS AND BASES

6.1 INTRODUCTION

DCD Revision 19 Chapter 16 contains the Technical Specifications and Bases for the AP1000 PWR as approved by the NRC. This section describes the impact of the changes included in the Advanced First Core (AFC) design and analysis on the Technical Specifications. The Technical Specifications presented in Appendix C of this report are the Technical Specifications approved by the NRC in DCD Revision 19. It should be noted that these Technical Specifications are used as representative Technical Specifications. When a utility prepares a COLA or a License Amendment Request, the plant specific Technical Specifications will have to be reviewed for the impact of these changes.

6.2 CHANGES TO TECHNICAL SPECIFICATIONS AND BASES

The implementation of AFC requires conforming changes to Technical Specifications to maintain the bases of the AFC design and safety analyses. The required Technical Specification and Bases changes are described below. Appendix C shows the applicable pages of AP1000 DCD Chapter 16, Revision 19 marked with suggested changes. NUREG-1431, Revision 3 (Reference 6-1) is used as a guide in developing the suggested changes. Appendix C shows only representative Technical Specifications and Bases. Actual adoption of the AFC will require that specific Technical Specifications be submitted to the NRC for approval in a similar manner as any other license amendment request that includes Technical Specification changes.

There are three changes identified which affect the Technical Specification and Bases as described below:

Control Rod Drop Time

As described in Section 5.2 (item 7), the safety analysis performed for the AFC assumed 2.7 seconds for the rod drop time. Therefore it is necessary to change the rod drop time criterion of Surveillance Requirement 3.1.4.3 from 2.47 seconds to 2.7 seconds.

RAOC to CAOC

A conforming change is required to the plant technical specifications to replace axial power distribution control limits associated with the OPDMS not functional (LCO 3.2.5 not applicable) from a Relaxation of Axial Offset Control (RAOC) limit to a Constant Axial Offset Control (CAOC) limit. In general, it is noted that a CAOC specification is more operationally limiting than a RAOC specification, and as such this specification change inherently increased margins to safety analysis limits in the event of operation with OPDMS not functional. This change is consistent with the core, fuel, and safety analyses performed for the updated core and fuel design described in Section 5. Furthermore, it is noted that, because of the axial power distribution control capabilities provided by the MSHIM operation and control strategy that will be used during AP1000 plant operation, the margins typically afforded by a RAOC technical specification have been generally shown to be unnecessary for this plant.

In order to change the axial power distribution control limits from RAOC to CAOC as described above, LCO 3.2.1, "Heat Flux Hot Channel Factor," LCO 3.2.3, "AXIAL FLUX DIFFERENCE" (AFD), and their associated bases must be updated from specifications based on the Standard

Technical Specifications (Reference 6-1) for RAOC with $F_Q(z)$ surveillance to CAOC with $F_Q(z)$ surveillance.

Acceptance Criteria for Reactivity Initiated Accidents

As discussed in Section 5.4, consistent with NUREG-1793 Supplement 2, Section 15.2.4.8.2 (Reference 5-5), the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, has been followed for the analysis of the RCCA ejection accident. This requires that the Technical Specification Bases specifying a limit on energy deposition to the fuel for an ejected rod accident, be revised to reflect the new limits and reference. Technical Specification Bases sections B 3.1.1, B 3.2.1, B 3.2.2, B 3.2.4, and B 3.2.5 must be revised to reflect the correct reference of the limits applied.

6.3 CONCLUSION

The core design for the first cycle of operation for the AP1000 PWR has been modified. Changes to the Technical Specifications and Bases are necessary with the implementation of the Advanced First Core. Representative examples of the changes to DCD Revision 19 Chapter 16 are provided in Appendix C.

6.4 REFERENCES

6-1. NUREG-1431, Revision 3.0, Standard Technical Specifications Westinghouse Plants, June 2004.

7 REGULATORY IMPACTS

7.1 INTRODUCTION

This section describes the licensing and implementation plans for the core and fuel design changes documented in this report.

Consistent with the process outlined in Reference 7-1 and the advantages of maintaining a standard design approach, the intention is for the core and fuel design described by the AP1000 CRR to be reviewed and approved as a generic topical report. Once approved, the CRR will provide the information necessary for a licensee to implement these improvements into their plant specific licensing basis via their COLA or as a post-COL license amendment request.

This section addresses the framework under which the fuel and core design modifications described in this report were developed and the implementation of these changes:

1) Core and fuel design changes developed as part of the Advanced First Core are identified and evaluated using the established design change process (Section 7.2),

2) Identified fuel and core design changes are included in the CRR as a generic topical report being submitted for review and approval by the USNRC (Section 7.3), and

3) Approved CRR is incorporated into individual plant licensing bases either as part of their COLA or as a license amendment by COL holders (Section 7.4).

Finally, Section 7.5 of this report provides additional information related to incorporation of future core and fuel design changes, specifically for subsequent fuel cycles.

7.2 DEVELOPMENT OF DESIGN CHANGES

Westinghouse has continued to develop and update the core, fuel, and core component designs for the AP1000 PWR in order to ensure the best available technologies are incorporated into first core load for the plant. The design changes were initiated, reviewed, approved, and implemented within Westinghouse in accordance with the established design change processes.

7.3 SUBMITTAL FOR REVIEW AS A GENERIC TOPICAL REPORT

The intention is for this topical report to provide the updated core and fuel design information necessary to supersede and/or supplement the information that is contained in the licensee FSAR. This includes some editorial and additional information that is provided for clarification in order to fully address the comprehensive nature of the changes being submitted. Further, Appendices A-C of this report provide a representative example of the proposed FSAR markups based on the NRC reviewed and approved AP1000 DCD, Rev. 19. It is noted that:

1) There are no changes to the Tier 1 information provided by the DCD.

2) There are no changes to the Tier 2* Principal Design Requirements for the core and fuel design provided in subsection 4.1.1 of the DCD. Furthermore, the methodologies used in design of the core and fuel as well as those used to perform the revised plant safety analyses have been previously reviewed and approved by the USNRC, or have been reviewed by the USNRC as part of this topical report.

This report follows the format of similar fuel update reports previously submitted by Westinghouse, such as Power Uprate and Reload Transition Safety Reports. This is also consistent with the framework outlined in Reference 7-1.

7.4 INCORPORATION OF CORE REFERENCE REPORT AS A LICENSE AMENDMENT

In accordance with the requirements of 10 CFR 50.90, each individual licensee will incorporate the information contained in the CRR via a license amendment request into their individual plant licensing bases. In this submittal final versions of the FSAR markups describing the changes described in the CRR will be included. Alternatively, this information may be incorporated into the COLA, as appropriate.

7.5 CORE AND FUEL DESIGN CHANGES FOR SUBSEQUENT FUEL CYCLES

In general, subsequent core reloads following an initial plant startup will follow the same processes as currently followed for the operating fleet. Specifically, evaluations of the impacts associated with a change in core and/or fuel design will be performed consistent with the methodologies described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" (Reference 7-2) and WCAP-12488-A, "Westinghouse Fuel Acceptance Criteria Evaluation Process" (Reference 4-3). Changes to the plant licensing basis associated with core and/or fuel design changes will be screened / evaluated pursuant to 10 CFR 50.59 / 10 CFR Part 52 Appendix D Section VIII as appropriate.

7.6 REFERENCES

7-1. WCAP-16652-NP, "AP1000 Core & Fuel Design Technical Report," October 2006.

7-2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" July 1985.

APPENDIX A

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 4 Section 4.1	Change Summary Description
[4.1-1]		No change.
[4.1-2]	4.1 paragraph 8 Table 4.1-1	Administrative change only to remove "TM" from ZIRLO™ at each usage in this section and to replace it with the registered trademark ZIRLO® and an accompanying footnote at its first usage.
[4.1-3]	4.1 paragraph 9	Deleted the description of gadolinium oxide/uranium oxide fuel pellets as not used for initial core design.
[4.1-4]	4.1 paragraph 11	Deleted the description of non-integral discrete burnable absorber (BA) as not used for initial core design.
[4.1-5]	4.1 paragraph 12	Deleted use of non-integral discrete burnable absorbers (BAs) while retaining the description of WABA and IFBA burnable absorbers used for initial core design.
[4.1-6]	4.1 paragraph 15	Deleted the description of silver-indium-cadmium (AIC-12) gray rod cluster assembly (GRCA) and added description of tungsten (W-24) GRCA design; also added new Reference 11 for WCAP-16943-P-A.
[4.1-7]	4.1 paragraph 15	Paragraph revised to update the GRCA design description to replace "An Alloy 718 spacer is" with "Stainless steel spacers are".
[4.1-8]	4.1 paragraph 16	Clarified the description of GRCA use in plant operations to include base load operation consistent with descriptions previously provided in DCD Chapter 7.
[4.1-9]	4.1.1 paragraph 1	Editorial change only. Deleted redundant use of the word "rod'.
[4.1-10]	4.1.3 Reference 11	Added new Reference 11 for WCAP-16943-P-A and WCAP-16943-NP-A, "Enhanced GRCA Rodlet Design".
[4.1-11]	Table 4.1-1 (all 4 Sheets)	Extended and reformatted Table 4.1-1 to add Sheet 4 of 4.
[4.1-12]	Table 4.1-1 (Sheet 1 of 4)	Removed ">1.22" and ">1.21" for AP1000 DNBR consistent with the change described in sub-section 4.4.1.1.2.
[4.1-13]	Table 4.1-1 (Sheet 2 of 4)	Changed the number of regions from 3 to 5 consistent with the description of the Advanced First Core design.
[4.1-14]	Table 4.1-1 (Sheet 3 of 4)	Revised Table 4.1-1 to include the description of the tungsten absorber rodlets in GRCAs.
[4.1-15]	Table 4.1-1 (Sheet 3 of 4)	Revised the neutron absorber cladding materials to include "or 304L".
[4.1-16]	Table 4.1-1 (Sheet 3 of 4)	Revised table for clad thickness to replace "Ag In Cd" with "RCCA" in order to separately describe the RCCA and GRCA clad thicknesses.
[4.1-17]	Table 4.1-1 (Sheet 4 of 4)	Revised first cycle fuel enrichment regions to add new regions 4 and 5.
[4.1-18]	Table 4.1-1 Note b.	Revised table footnote b. for replaced W-3 with alternative correlations ABB-NV and WLOP.

[4.1-19]	Table 4.1-1 Note d.	Removed DNBR limits of 1.22 and 1.21 for consistency with change described in sub-section 4.4.1.1.2.
[4.1-20]	Table 4.1-1 Note j.	Added footnote j. "For AP1000, the assembly average enrichments are given for the mid-zone and axial blanket regions."
[4.1-21]	Table 4.1-2 (Sheet 1 of 2)	Updated the listing of nuclear design computer codes to include PARAGON for consistency with the changes to Sections 4.3.3.2 and 4.3.3.3.
[4.1-22]	Table 4.1-2 (Sheet 1 of 2)	Corrected "AMPX system of codes, KENO-Va" to "MCNP4a" for consistency with Section 4.3.2.6.1.
[4.1-23]	Table 4.1-3	Editorial change only. Added "thimble plug" for consistency with the list of core components similarly provided in other sections, e.g. Section 4.2.1.6.
[4.1-24]	Table 4.1-3	Editorial change only. The word "holddown" is added to more accurately describe the spring.
[4.1-25]	Table 4.1-3	Editorial change only. Table number is corrected from "Table 3.9.1-1" to "Table 3.9-1" for consistency with DCD Chapter 3.
[4.1-26]	4.1.3	Updated the reference for WCAP-16943 to reference the approved version of this topical report.

CHAPTER 4

REACTOR

4.1 Summary Description

This chapter describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the nuclear design, and the thermal-hydraulic design.

The reactor contains a matrix of fuel rods assembled into mechanically identical fuel assemblies along with control and structural elements. The assemblies, containing various fuel enrichments, are configured into the core arrangement located and supported by the reactor internals. The reactor internals also direct the flow of the coolant past the fuel rods. The coolant and moderator are light water at a normal operating pressure of 2250 psia. The fuel, internals, and coolant are contained within a heavy walled reactor pressure vessel. An AP1000 fuel assembly consists of 264 fuel rods in a 17x17 square array. The center position in the fuel assembly has a guide thimble that is reserved for in-core instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids.

The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring fingers and dimples that grip and support the fuel rods. The intermediate mixing vane grids also have coolant mixing vanes. In addition, there are four intermediate flow mixing (IFM) grids. The IFM grid straps contain support dimples and coolant mixing vanes only. The top and bottom grids and protective grid do not contain mixing vanes.

The AP1000 fuel assemblies are similar to the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies have an active fuel length of 12 feet and three intermediate flow mixing grids in the top mixing vane grid spans. The 17x17 XL Robust fuel assemblies have an active fuel length of 14 feet (same as AP1000 fuel assemblies) and no intermediate flow mixing grids. The AP1000 fuel assemblies have four intermediate flow mixing grids in the top mixing vane grid spans.

Deleted: with

Deleted: are the same as the 17x17 XL Robust fuel assemblies except that they

There is substantial operating experience with the 17x17 Robust and 17x17 XL Robust fuel assemblies. The 17x17 Robust fuel assemblies are described in References 1, 2 and 3. The 17x17 XL Robust fuel assemblies are described in References 4 and 5.

The XL Robust fuel assembly evolved from the previous VANTAGE+, VANTAGE 5 and VANTAGE 5 HYBRID designs. The XL Robust fuel assembly is based on the substantial

design and operating experience with those designs. The design is described and evaluated in References 2, 3, 6 through 10.

A number of proven design features have been incorporated in the AP1000 fuel assembly design. The AP1000 fuel assembly design includes: low pressure drop intermediate grids, four intermediate flow mixing (IFM) grids, a reconstitutable Westinghouse integral nozzle (WIN), and extended burnup capability. The bottom nozzle is a debris filter bottom nozzle (DFBN) that minimizes the potential for fuel damage due to debris in the reactor coolant. The AP1000 fuel assembly design also includes a protective grid for enhanced debris resistance.

The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in ZIRLO®[1] (Reference 8) tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and to improve fuel utilization.

Comment [A1]: [4.1-02]

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Other types of fuel rods may be used to varying degrees within some fuel assemblies. One type uses an integral fuel burnable absorber (IFBA) containing a thin boride coating on the surface of the fuel pellets. The boride-coated fuel pellets provide a burnable absorber integral to the fuel.

Comment [A2]: [4.1-03]

Deleted: Another type uses fuel pellets containing gadolinium oxide mixed with uranium oxide. The boride-coated fuel pellets and gadolinium oxide/uranium oxide

Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown during operation and thereby prevent clad flattening. The fuel rods in the AP1000 fuel assemblies contain additional gas space below the fuel pellets, compared to the 17x17 Robust, 17x17 XL Robust and other previous fuel assembly designs to allow for increased fission gas production due to high fuel burnups.

Depending on the position of the assembly in the core, the guide thimbles are used for rod cluster control assemblies (RCCAs), gray rod cluster assemblies (GRCAs), neutron source assemblies, wet annular **burnable absorber** (WABA) assemblies, or thimble plugs.

Comment [A3]: [4.1-04]

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For the initial core design, WABAs **and** IFBAs are used. Discrete burnable absorber designs, integral fuel burnable absorber designs, or combinations may be used in subsequent reloads.

Comment [A4]: [4.1-05]

Deleted: discrete burnable absorbers (BAs)

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The bottom nozzle is a box-like structure that serves as the lower structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The size of flow passages through the bottom nozzle limits the size of debris that can enter the fuel assembly.

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The top nozzle assembly serves as the upper structural element of the fuel assembly and provides a partial protective housing for the rod cluster control assembly or other components.

The rod cluster control assemblies consist of 24 absorber rodlets fastened at the top end to a common hub, or spider assembly. Each absorber rod consists of an alloy of silver-indium-cadmium, which is clad in stainless steel. The rod cluster control assemblies are used to control relatively rapid changes in reactivity and to control the axial power distribution.

The gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub, or spider assembly (Reference 11). Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber rods consists of tungsten contained within a nickel-chromium-iron Alloy 718 sleeve. Stainless steel spacers are provided at the bottom of the rodlet between the bottom of the sleeve and the bottom end plug of the stainless steel cladding.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering. The assemblies provide a mechanical shim reactivity mechanism to minimize the need for changes to the concentration of soluble boron.

The reactor core is cooled and moderated by light water at a pressure of 2250 psia. Soluble boron in the moderator/coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup. Burnable absorbers are also employed in the initial cycle to limit the amount of soluble boron required and, thereby maintain the desired negative reactivity coefficients.

The nuclear design analyses establish the core locations for control rods and burnable absorbers. The analyses define design parameters, such as fuel enrichments and boron concentration in the coolant.

The nuclear design establishes that the reactor core and the reactor control system satisfy design criteria, even if the rod cluster control assembly of the highest reactivity worth is in the fully withdrawn position.

The core has inherent stability against diametral and azimuthal power oscillations. Axial power oscillations, which may be induced by load changes, and resultant transient xenon may be suppressed by the use of the rod cluster control assemblies.

The control rod drive mechanisms used to withdraw and insert the rod cluster control assemblies and the gray rod cluster assemblies are described in subsection 3.9.4.

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Deleted: 24 rodlets are fabricated of stainless

Comment [A5]: [4.1-06]

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Deleted: remaining 12 are silver-indium-cadmium (of a reduced diameter as compared to

Deleted: RCCA absorber) with

Comment [A6]: [4.1-07]

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Comment [A7]: [4.1-08]

The thermal-hydraulic design analyses establish that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers induce additional flow mixing between the various flow channels within a fuel assembly, as well as between adjacent assemblies.

The reactor internals direct the flow of coolant to and from the fuel assemblies and are described in subsection 3.9.5.

The performance of the core is monitored by fixed neutron detectors outside the core, fixed neutron detectors within the core, and thermocouples at the outlet of selected fuel assemblies. The ex-core nuclear instrumentation provides input to automatic control functions.

Table 4.1-1 presents a summary of the principal nuclear, thermal-hydraulic, and mechanical design parameters of the AP1000 fuel. A comparison is provided to the fuel design used in AP1000, AP600 and in a licensed Westinghouse-designed plant using XL Robust fuel. For the comparison with a plant containing XL Robust fuel, a 193 fuel assembly plant is used, since no domestic, Westinghouse-designed 157 fuel assembly plants use 17x17 XL Robust fuel.

Table 4.1-2 tabulates the analytical techniques employed in the core design. The design basis must be met using these analytical techniques. Enhancements may be made to these techniques provided that the changes are bounded by NRC-approved methods, models, or criteria. In addition, application of the process described in WCAP-12488-A, (Reference 9) allows the Combined License holder to make fuel mechanical changes. Table 4.1-3 tabulates the mechanical loading conditions considered for the core internals and components. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel assemblies in subsection 4.2.1.5; control rods (RCCAs and GRCAs), burnable absorber rods, and neutron source rods, in subsection 4.2.1.6. The dynamic analyses, input forcing functions, and response loadings for the control rod drive system and reactor vessel internals are presented in subsections 3.9.4 and 3.9.5.

4.1.1 Principal Design Requirements

The fuel and control rod mechanism are designed so the performance and safety criteria described in Chapter 4 and Chapter 15 are met. [*The mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) are designed to achieve these criteria, referred to as Principal Design Requirements:*

Comment [A8]: [4.1-09]

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

- *Fuel damage, defined as penetration of the fuel cladding, is predicted not to occur during normal operation and anticipated operational transients.*

- *Materials used in the fuel assembly and in-core control components are selected to be compatible in a pressurized water reactor environment.*

- *For normal operation and anticipated transient conditions, the minimum DNBR calculated using the WRB-2M correlation is greater than or equal to 1.14.*

- *Fuel melting will not occur at the overpower limit for Condition I or II events.*

- *The maximum fuel rod cladding temperature following a loss-of-coolant accident is calculated to be less than 2200°F.*

- *For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.718 kw/ft for the initial fuel cycle.*

- *For normal operation and anticipated transient conditions, the calculated total heat flux hot channel factor, F_Q, is less than or equal to 2.60 for the initial fuel cycle.*

- *Calculated rod worths provide sufficient reactivity to account for the power defect from full power to zero power and provide the required shutdown margin, with allowance for the worst stuck rod.*

- *Calculations of the accidental withdrawal of two control banks using the maximum reactivity change rate predict that the peak linear heat rate and DNBR limits are met.*

- *The maximum rod control cluster assembly and gray rod speed (or travel rate) is 45 inches per minute.*

- *The control rod drive mechanisms are hydrotested after manufacture at a minimum of 125 percent of system design pressure.*

- *For the initial fuel cycle, the fuel rod temperature coefficient is calculated to be negative for power operating conditions.*

- *For the initial fuel cycle, the moderator temperature coefficient is calculated to be negative for power operating conditions.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.1.2 Combined License Information

This section contains no requirement for additional information to be provided in support of Combined License.

4.1.3 References

Comment [A9]: [4.1-26]

1. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (NRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications," NSD-NRC-97-5189, June 30, 1997.

2. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "Transmittal of Presentation Material for NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996," NSD-NRC-96-4964, April 22, 1996.

3. Letter from Westinghouse to NRC, "Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design," NSD-NRC-98-5796, October 13, 1998.

4. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application," NSD-NRC-98-5618, March 25, 1998.

5. Letter from H. A. Sepp (Westinghouse) to T. E. Collins (NRC), "Fuel Criteria Evaluation Process Notification for the Revised Guide Thimble Dashpot Design for the 17x17 XL Robust Fuel Assembly Design," NSD-NRC-98-5722, June 23, 1998.

6. Davidson, S. L., and Kramer, W. R., (Ed.), "Reference Core Report Vantage 5 Fuel Assembly," WCAP-10444-P-A (Proprietary), September 1985 and WCAP-10445-A (Non-Proprietary), December 1983.

7. Davidson, S. L., (Ed.), "VANTAGE 5H Fuel Assembly," Addendum 2-A, WCAP-10444-P-A (Proprietary) and WCAP-10445-NP-A (Non-Proprietary), February 1989.

8. Davidson, S. L., and Nuhfer, D. L., (Ed.), "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (Proprietary) and WCAP-14342-A (Non-Proprietary), April 1995.

9. [*Davidson, S. L. (Ed.), "Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary) and WCAP-14204-A (Non-Proprietary), October 1994.*]*

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

10. Letter, Peralta, J. D. (NRC) to Maurer, B. F. (Westinghouse), "Approval for Increase in Licensing Burnup Limit to 62,000 MWD/MTU (TAC No. MD1486)," May 25, 2006.

11. Conner, M. E., et al. "Enhanced GRCA Rodlet Design," WCAP-16943-P-A (Proprietary) and WCAP-16943-NP-A (Non-Proprietary), September 2012.

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Comment [A9]: [4.1-10]

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Comment [A10]: [4.1-11]
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Table 4.1-1 (Sheet 1 of 4)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 Btu/hr)	11,601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure, nominal (psia)	2250	2250	2250
System pressure, minimum steady-state (psia)	2190	2200	2204
Minimum departure from nuclear boiling (DNBR) for design transients Typical flow channel Thimble (cold wall) flow channel	 >1.25[(d)] >1.25[(d)]	 >1.23 >1.22	 >1.26 >1.24
Departure from nucleate boiling (DNB) correlation[(b)]	WRB-2M[(b)]	WRB-2	WRB-1[(a)]
Coolant Flow[(c)]			
Total vessel thermal design flow rate (10^6 lbm/hr) Effective flow rate for heat transfer (10^6 lbm/hr) Effective flow area for heat transfer (ft^2) Average velocity along fuel rods (ft/s) Average mass velocity (10^6 lbm/hr-ft^2)	113.5 106.8 41.8 15.8 2.55	72.9 66.3 38.5 10.6 1.72	145.0 132.7 51.1 16.6 2.60
Coolant Temperature[(c)(e)]			
Nominal inlet (°F) Average rise in vessel (°F) Average rise in core (°F) Average in core (°F) Average in vessel (°F)	535.0 77.2 81.4 578.1 573.6	532.8 69.6 75.8 572.6 567.6	561.2 63.6 68.7 597.8 593.0
Heat Transfer			
Active heat transfer surface area (ft^2) Avg. heat flux (BTU/hr-ft^2) Maximum heat flux for normal operation (BTU/hr-ft^2)[(f)] Average linear power (kW/ft)[(g)] Peak linear power for normal operation (kW/ft)[(f)(g)] Peak linear power (kW/ft)[(f)(h)] (Resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	56,700 199,300 518,200 5.72 14.9 ≤22.45	44,884 143,000 372,226 4.11 10.7 22.5	69,700 181,200 489,200 5.20 14.0 ≤22.45

Comment [A11]: [4.1-12]
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Table 4.1-1 (Sheet 2 of 4)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Heat flux hot channel factor (F_Q)	2.60	2.60	2.70
Peak fuel center line temperature (°F) (For prevention of center-line melt)	4700	4700	4700
Fuel assembly design	17x17 XL Robust Fuel	17x17	17x17 XL Robust Fuel/ No IFM
Number of fuel assemblies	157	145	193
Uranium dioxide rods per assembly	264	264	264
Rod pitch (in.)	0.496	0.496	0.496
Overall dimensions (in.)	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
Fuel weight, as uranium dioxide (lb)	211,588	167,360	261,000
Clad weight (lb)	43,105	35,555	63,200
Number of grids per assembly Top and bottom - (Ni-Cr-Fe Alloy 718) Intermediate Intermediate flow mixing Protective Grid - (Ni-Cr-Fe Alloy 718)	 2(i) 8 ZIRLO 4 ZIRLO 1	 2(i) 7 Zircaloy-4 or 7 ZIRLO 4 Zircaloy-4 or 5 ZIRLO 1	 2 8 ZIRLO 0 1
Loading technique, first cycle	5 region nonuniform	3 region nonuniform	3 region nonuniform
Fuel Rods			
Number	41,448	38,280	50,952
Outside diameter (in.)	0.374	0.374	0.374
Diametral gap (non-IFBA) (in.)	0.0065	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225	0.0225
Clad material	ZIRLO	Zircaloy-4 or ZIRLO	Zircaloy-4/ ZIRLO
Fuel Pellets			
Material	UO_2 sintered	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95.5	95	95
Diameter (in.)	0.3225	0.3225	0.3225
Length (in.)	0.387	0.387	0.387

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Comment [A13]: [4.1-02]
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Comment [A14]: [4.1-13]
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Comment [A15]: [4.1-02]
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Table 4.1-1 (Sheet 3 of 4) REACTOR DESIGN COMPARISON TABLE			
Rod Cluster Control Assemblies	**AP1000**	**AP600**	**Typical XL Plant**
Neutron Absorber			
RCCA GRCA	24 Ag-In-Cd rodlets 24 Tungsten rodlets	24 Ag-In-Cd rodlets 20 304 SS rodlets 4 Ag-In-Cd rodlets	24 Hafnium or Ag-In-Cd
Cladding material	Type 304 or 304L SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness (in.) - RCCA - GRCA	0.0185 0.0255	0.0185 0.0185	0.0185
Number of clusters	53 RCCAs 16 GRCAs	45 RCCAs 16 GRCAs	57 RCCAs 0 GRCAs
Core Structure			
Core barrel, ID/OD (in.)	133.75/137.75	133.75/137.75	148.0/152.5
Thermal shield	Neutron Panel	None	Neutron Panel
Baffle thickness (in.)	Core Shroud	Radial reflector	0.875
Structure Characteristics			
Core diameter, equivalent (in.)	119.7	115.0	132.7
Core height, cold, active fuel (in.)	168.0	144.0	168.0

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Comment [A17]: [4.1-14]

Deleted: 12 Ag-In-Cd rodlets

Comment [A18]: [4.1-15

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Comment [A19]: [4.1-16]

Table 4.1-1 (Sheet 4 of 4) REACTOR DESIGN COMPARISON TABLE			
Fuel Enrichment First Cycle (Weight Percent)	**AP1000[(j)]**	**AP600**	**Typical XL Plant**
Region 1	0.74 / --	1.90	Typical
Region 2	1.58 / --	2.80	3.8 to 4.4
Region 3	3.20 / 1.58	3.70	(5.0 Max)
Region 4	3.776 / 3.20		
Region 5	4.376 / 3.20		

Notes:

a. WRB-2M will be used in future reloads.

b. See subsection 4.4.2.2.1 for the use of the ABB-NV, WLOP, WRB-2 and WRB-2M correlations.

c. Flow rates and temperatures are based on 10 percent steam generator tube plugging for the AP600 and AP1000 designs.

d. The Design Limit DNBR is 1.25.

e. Coolant temperatures based on thermal design flow (for AP600 and AP1000).

f. Based on F_Q of 2.60 for AP600 and AP1000.

g. Based on densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft.

h. See subsection 4.3.2.2.6.

i. The top grid will be fabricated of nickel-chromium-iron Alloy 718.

j. For the AP1000 design, the assembly average enrichments are given for the mid-zone and axial blanket regions.

Comment [A20]: [4.1-11]

Comment [A21]: [4.1-17]

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Comment [A22]: [4.1-17]

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Comment [A23]: [4.1-17]

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Comment [A24]: [4.1-17

Comment [A25]: [4.1-17]

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Comment [A26]: [4.1-18]

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Comment [A27]: [4.1-19]

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Comment [A28]: [4.1-20]

Table 4.1-2 (Sheet 1 of 2) **ANALYTICAL TECHNIQUES IN CORE DESIGN**			
Analysis	**Technique**	**Computer Code**	**Subsection Referenced**
Mechanical design of core internals loads, deflections, and stress analysis	Static and dynamic modeling	BLOWDOWN code, FORCE, finite element structural analysis code, and others	3.7.2.1 3.9.2 3.9.3
Fuel rod design Fuel performance characteristics (such as, temperature, internal pressure, and clad stress)	Semi-empirical thermal model of fuel rod with considerations such as fuel density changes, heat transfer, and fission gas release.	Westinghouse fuel rod design model	4.2.1.1 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.11
Nuclear design Cross-sections and group constants	Microscopic data; macroscopic constants for homogenized core regions	Modified ENDF/B library with PHOENIX-P or PARAGON	4.3.3.2 4.3.3.3
X-Y and X-Y-Z power distributions, fuel depletion, critical boron concentrations, X-Y and X-Y-Z xenon distributions, reactivity coefficients	2-group diffusion theory, 2-group nodal theory	ANC (2-D or 3-D)	4.3.3.3
Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	APOLLO	4.3.3.3
Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighing function	REPAD	4.3.3.1
Criticality of reactor and fuel assemblies	3-D, Monte Carlo theory	MCNP4a	4.3.2.6
Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.2

Comment [A29]: [4.1-21]

Comment [A30]: [4.1-22]

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Table 4.1-2 (Sheet 2 of 2) ANALYTICAL TECHNIQUES IN CORE DESIGN			
Analysis	**Technique**	**Computer Code**	**Subsection Referenced**
Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.4

Table 4.1-3 **DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS**
• Fuel assembly weight and core component weights (burnable absorbers, sources, RCCA, GRCA, thimble plug)
• Fuel assembly holddown spring forces and core component spring forces
• Internals weight
• Control rod trip (equivalent static load)
• Differential pressure
• Spring preloads
• Coolant flow forces (static)
• Temperature gradients
• Thermal expansion
• Interference between components
• Vibration (mechanically or hydraulically induced)
• Operational transients listed in Table 3.9-1
• Pump overspeed
• Seismic loads (safe shutdown earthquake)
• Blowdown forces (due to pipe rupture)

Comment [A31]: [4.1-23]

Comment [A32]: [4.1-24]

Comment [A33]: [4.1-25]

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AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 4 Section 4.2	Change Summary Description
[4.2-1]		No change.
[4.2-2]	4.2.1.1.1 paragraph 1 4.2.1.1.4 paragraph 1 4.2.2 paragraph 3 4.2.2.1 paragraph 1 4.2.2.2.3 paragraph 1 4.2.2.2.4 paragraph 4 4.2.2.2.4 paragraph 6 4.2.3.1.3 paragraph 1	Administrative change only to remove "TM" from ZIRLO™ at each usage in this section.
[4.2-3]	4.2.1.5.3 paragraph 1 sub-bullet 3	Editorial change provided to remove incorrectly formatted sentence break.
[4.2-4]	4.2.1.6 paragraph 1	Editorial clarification/correction only. Revised the paragraph to describe the thimble plugs as permanent instead of temporary. Thimble plugs are required for the AP1000.
[4.2-5]	4.2.1.6.1 paragraph 2 bullet 6	Revised the description to separately address RCCA and GRCA absorber melting temperatures; Also added new Reference 25 for WCAP-16943-P-A which includes tungsten material properties for the enhanced GRCAs.
[4.2-6]	4.2.1.6.2 paragraph 2 and 3	Revised and combined paragraphs 2 and 3 to describe the use of wet annular burnable absorbers (WABAs) for the first core design. The previous description of borosilicate glass (Pyrex) burnable absorbers has been removed, including associated Reference 23 to WCAP-7113 which is no longer used.
[4.2-7]	4.2.2.1 paragraph 7	Revised the description of integral fuel burnable absorbers to reflect the use of boride-coated fuel pellets in the final design. The previous description of fuel pellets containing gadolinium oxide has been removed...
[4.2-8]	4.2.2.2.1 paragraph 1	Revised the description to clarify that Grade CF-3 stainless steel materials are used in the fabrication of the bottom nozzle.
[4.2-9]	4.2.2.2.2 paragraph 1	Clarified the description by replacing "304" with "304L and Grade CF-3" consistent with the top nozzle design.
[4.2-10]	4.2.2.2.3 paragraph 4	Editorial correction only. Revised the description to delete "with exception noted for the bottom nickel chromium iron Alloy 718 grid" consistent with the design.
[4.2-11]	4.2.2.2.3 paragraph 5	Editorial correction only. Revised the number of bulges shown for the top grid from "three" (typically) to "two" consistent with the fuel design.
[4.2-12]	4.2.2.2.3 paragraph 9	Revised the description to include "bottom" for clarification that the instrumentation tube is also secured to the bottom grid.
[4.2-13]	4.2.2.3.1 paragraph 3	Editorial change only. Clarified definition of "black" rods.

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[4.2-14]	4.2.2.3.1 paragraph 5	Revised paragraph to reflect the use of RCCA low carbon grade materials.
[4.2-15]	4.2.2.3.1 paragraph 6	Revised the description to reference more recent ASME Code Tables and Sections.
[4.2-16]	4.2.2.3.1 paragraph 6	Revised the description to add "or 308L" to reflect use of RCCA low carbon grade materials.
[4.2-17]	4.2.2.3.1 paragraph 7	Revised the description consistent with the RCCA integral design.
[4.2-18]	4.2.2.3.1 paragraph 8	Revised paragraph to reflect the use of RCCA low carbon grade materials.
[4.2-19]	4.2.2.3.1 paragraph 9	Revised paragraph to reflect the improved RCCA design for locking the absorber rod to the spider assembly.
[4.2-20]	4.2.2.3.2 paragraph 1	Revised paragraph to clarify that the "external" design of the GRCA and RCCA is identical.
[4.2-21]	4.2.2.3.2 paragraph 2	Revised the paragraph to add the description of the tungsten (W-24) GRCA design which replaces the previous description of the silver-indium-cadmium (AIC-12) gray rod cluster assembly (GRCA) design.
[4.2-22]	4.2.2.3.2 paragraph 3	Clarified the description of the gray rods to include base load operation.
[4.2-23]	4.2.2.3.3 paragraph 2	Deleted paragraph 2 which previously described the alternative use of two types of burnable absorber assemblies; Also deleted the description of borosilicate glass (Pyrex) burnable absorbers, along with the associated Reference 23 and Figure 4.2-13.
[4.2-24]	4.2.2.3.3 paragraph 3 (now paragraph 2)	Revised the description to clarify use of the wet annular burnable absorber rods (WABA) in the final design.
[4.2-25]	4.2.2.3.3 paragraph 3 (now paragraph 2)	Revised the description to clarify use of the bottom-end spacer by adding "as necessary".
[4.2-26]	4.2.2.3.4 paragraph 8	Revised paragraph to include low carbon grade materials added during design finalization to minimize possibility of sensitization during welding.
[4.2-27]	4.2.3.1.2 paragraph 1 bullet 2	Revised bullet to add new "(Reference 26)" to WCAP-8963-P-A, Addendum 1-A, Revision 1-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)" June 2006.
[4.2-28]	4.2.3.3.1 paragraph 8	Example description clarified consistent with the initial core design by replacing "region 3" with the generalized phrase "a fuel assembly".
[4.2-29]	4.2.3.3.2 paragraph 1	Revised reference for operational experience from WCAP-8183 (Reference 3) to CENPD-404-P-A (Reference 27).
[4.2-30]	4.2.3.6.1 paragraph 1	Revised the description to include that the tungsten gray rod (GRCA) design provides sufficient void volume to accommodate the internal pressure increase during operation.
[4.2-31]	4.2.3.6.1 paragraph 1	Revised the description to separately address the rod cluster control assembly (RCCA) by describing that no significant amount of gas is release by the Ag-In-Cd absorber material.
[4.2-32]	4.2.3.6.1 paragraph 2	Added "and gray rods" to indicate that adequate void volume is available to limit internal pressure.

[4.2-33]	4.2.3.6.1 paragraph 2	Added reference to Figure 4.2-11 for the gray rod assembly.
[4.2-34]	4.2.3.6.2 paragraph 2	Added description of tungsten GRCA and new Reference 25 to WCAP-16943-P-A.
[4.2-35]	4.2.3.6.2 paragraph 3	Deleted previous paragraph 3 and the corresponding Reference 23 to WCAP-7113. The deleted paragraph was previously used to describe the borosilicate glass burnable absorber which is not used in the advanced first core design.
[4.2-36]	4.2.3.6.2 paragraph 3	Added that the wet annular burnable absorber (WABA) assemblies are used in the first core.
[4.2-37]	4.2.3.6.3 paragraph 1	Added description of irradiation stability for tungsten absorber and new Reference 25 to WCAP-16943-P-A for tungsten GRCA design topical.
[4.2-38]	4.2.3.6.3 paragraph 2	Revised the paragraph to distinguish behavior between RCCA and GRCA absorber materials.
[4.2-39]	4.2.3.6.3 paragraph 3	Added new paragraph to address irradiation effects in tungsten for the enhanced GRCA design added per DCP-2268.
[4.2-40]	4.2.3.6.4 paragraph 5	Added new paragraph to address the effects on the tungsten absorber in the unlikely event of a GRCA rod cladding breach; also added the associated Reference 25 to WCAP-16943-P-A Enhanced (tungsten) GRCA design topical.
[4.2-41]	4.2.4.4 paragraph 2 sub-bullet 2	Added "with brazed and welded vanes and fingers" for additional clarification. (Integral RCCA spiders do not require proof-testing.)
[4.2-42]	4.2.4.4 paragraph 4	Added "rod cluster control" assemblies as clarification to "periodic drop tests of the assemblies," and deleted "on every assembly" since the technical specification surveillance 3.1.4.2 does not actually require a periodic free movement test for gray rods.
[4.2-43]	4.2.4.4 paragraph 5	Added clarification to describe the status of the rod cluster control assembly to be "untrippable."
[4.2-44]	4.2.4.4 paragraph 5	Deleted the discussion on operation with more than one inoperable RCCA being potentially tolerated with additional demands on the plant operator. Tech Spec LCO 3.1.4 does not distinguish between single and multiple inoperable RCCAs and requires that the same actions be followed under either condition (that the plant verify or restore shutdown margin in 1 hour and be taken to Mode 3 within 6 hours).
[4.2-45]	4.2.4.4 paragraph 5	Deleted sentence which said the number of inoperable assemblies has been limited to one. This sentence implied that there were different actions for single and multiple inoperable control rods, and that long term operation with a single inoperable RCCA could be permitted. LCO 3.1.4 does not distinguish between single and multiple inoperable control rods and requires that the same actions be followed under either condition (that the plant verify or restore shutdown margin in 1 hour and be taken to Mode 3 within 6 hours).
[4.2-46]	4.2.4.6 paragraph 1	Replaced referenced topical from WCAP-8183 (Reference 3) to CENPD-404-P-A (Reference 27) for fuel assembly operating experience.

[4.2-47]	4.2.6 Reference 3	Deleted "(revised annually)" from Reference 3. WCAP-8183, "Operational Experience with Westinghouse Core." This topical is no longer updated on an annual basis.
[4.2-48]	4.2.6 Reference 23	Deleted WCAP-7113 and replaced with "Not used" - consistent with the change to 4.2.1.6.2 paragraph 2, which no longer includes the description of borosilicate glass (Pyrex) burnable absorbers and its associated reference to WCAP-7113.
[4.2-49]	4.2.6 Reference 25	Added new reference to WCAP-16943-P-A, "Enhanced GRCA Rodlet Design" topical report - consistent with the revised descriptions for the tungsten GRCAs.
[4.2-50]	4.2.6 Reference 26	Added new reference WCAP-8963-P-A, Addendum 1-A, Revision 1-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)," June 2006 - consistent with the change to 4.2.3.1.2.
[4.2-51]	4.2.6 Reference 27	Added new reference CENPD-404-P-A, Revision 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001 - consistent with the change to 4.2.3.3.2.
[4.2-52]	Figure 4.2-1	Figure has been redrawn with metric units added. The basic content of this figure has not been changed.
[4.2-53]	Figure 4.2-2	Figure has been redrawn and includes the addition of the top nozzle fuel alignment pin hole pitch dimension for clarification. Metric units have also been added. The basic content of this figure has not been changed.
[4.2-54]	Figure 4.2-3	Figure has been redrawn and metric units added. The basic content of this figure has not been changed.
[4.2-55]	Figure 4.2-4	Figure has been redrawn and metric units added. Except for pictorial representation of the number of bulges below the top grid, the basic content of this figure has not been changed.
[4.2-56]	Figure 4.2-5	Figure has been redrawn with grid height dimension added consistent with DCD Revision 19 which stated that the dimension will be provided in the Core Reference Report. Metric units have also been added. The basic content of this figure has not been changed.
[4.2-57]	Figure 4.2-6	Figure has been redrawn with dimensions added consistent with DCD Revision 19 which stated that the dimension will be provided in the Core Reference Report. Metric units have also been added. The basic content of this figure has not been changed.
[4.2-58]	Figure 4.2-7	Figure has been redrawn. With the exception of the number of P-Grid spacers (changed from 4 to 8), the basic content of this figure has not been changed.
[4.2-59]	Figure 4.2-8	Figure has been redrawn to remove any stray marks. No changes have been made to the content of this figure.
[4.2-60]	Figure 4.2-9	Figure has been updated based on enhanced RCCA design options. Metric units have also been added.
[4.2-61]	Figure 4.2-10	Figure has been updated based on enhanced RCCA design options. Metric units have also been added.

Deleted: (This topical is presently undergoing NRC staff review.)

[4.2-62]	Figure 4.2-11	Figure has been updated based on enhanced RCCA design options and tungsten GRCA design. Metric units have also been added.
[4.2-63]	Figure 4.2-12	Revised figure from "Discrete Burnable Absorber Assembly" to "Wet Annular Burnable Absorber Assembly" consistent with the advanced first core design change as described in sub-section 4.2.2.3.3.
[4.2-64]	Figure 4.2-13	Deleted figure of "Burnable Absorber Rod Assembly (Pyrex) Borosilicate Glass" and replaced with "Not used" consistent with the advanced first core design change as described in sub-section 4.2.2.3.3.
[4.2-65]	Figure 4.2-14	Figure has been redrawn and metric units added. The basic content of this figure has not been changed.
[4.2-66]	Figure 4.2-15	Figure has been redrawn and metric units added. In addition, the source length has been corrected from 97 to 88 in. for consistency with Section 4.2.2.3.4, which already correctly specified the length as 88 in.
[4.2-67]	4.2.1.6.1 4.2.3.6.2 4.2.3.6.3 4.2.3.6.4 4.2.6	Updated the reference for WCAP-16943 to reference the approved version of this topical report.

4.2 Fuel System Design

The plant conditions for design are divided into four categories.

- Condition I - normal operation and operational transients
- Condition II - events of moderate frequency
- Condition III - infrequent incidents
- Condition IV - limiting faults

Chapter 15 describes bases and plant operation and events involving each condition.

The reactor is designed so that its components meet the following performance and safety criteria:

- The mechanical design and physical arrangement of the reactor core components, together with corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) provide that:

  – Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II events. A very small amount of fuel damage may occur. This is within the capability of the plant cleanup system and is consistent with the plant design bases.

  – The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged. The fraction of fuel rods damaged must be limited to meet the dose guidelines identified in Chapter 15 although sufficient fuel damage might occur to preclude immediate resumption of operation.

  – The reactor can be brought to a safe state and the core kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

- The fuel assemblies are designed to withstand non-operational loads induced during shipping, handling, and core loading without exceeding the criteria of subsection 4.2.1.5.1.

- The fuel assemblies are designed to accept control rod insertions to provide the required reactivity control for power operations and reactivity shutdown conditions.

- The fuel assemblies have provisions for the insertion of in-core instrumentation.

- The reactor vessel and internals, in conjunction with the fuel assembly structure, directs reactor coolant through the core. Because of the resulting flow distribution and bypass flow, the heat transfer performance requirements are met for the modes of operation.

The following subsection provides the fuel system design bases and design limits. It is consistent with the criteria of the Standard Review Plan, Section 4.2.

Consistent with the growth in technology, Westinghouse modifies fuel system designs. These modifications utilize NRC approved methods. [*A set of design fuel criteria to be satisfied by new fuel designs was issued to the NRC in WCAP-12488-A and its addendums (Reference 1)*]* and also presented below in subsection 4.2.1.

4.2.1 Design Basis

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2 of the Standard Review Plan. [*The design bases and acceptance limits used by Westinghouse are also described in the Westinghouse Fuel Criteria Evaluation Process, WCAP-12488-A and its addendums (Reference 1).*]*

The fuel rods are designed to satisfy the fuel rod design criteria for rod burnup levels up to the design discharge burnup using the extended burnup design methods described in the Extended Burnup Evaluation report, WCAP-10125-P-A (Reference 2).

The AP1000 fuel rod design considers effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is provided by designing to prevent excessive fuel temperatures as discussed in subsection 4.2.1.2.1; excessive internal rod gas pressures due to fission gas releases as discussed in subsections 4.2.1.3.1 and 4.2.1.3.2; and excessive cladding stresses, strains, and strain fatigue, as discussed in subsections 4.2.1.1.2 and 4.2.1.1.3. The fuel rods are designed so that the conservative design bases of the following events envelope the lifetime operating conditions of the fuel. For each design basis, the performance of the limiting fuel rod, with appropriate consideration for uncertainties, does not exceed the limits specified by the design basis. The detailed fuel rod design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre-pressurization level.

Integrity of the fuel assembly structure is provided by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. Three types of loads are considered:

- Non-operational loads, such as those due to shipping and handling
- Normal and abnormal loads, which are defined for Conditions I and II
- Abnormal loads, which are defined for Conditions III and IV

The design bases for the in-core control components are described in subsection 4.2.1.6.

4.2.1.1 Cladding

4.2.1.1.1 Mechanical Properties

The ZIRLO cladding material combines neutron economy (low absorption cross-section); high corrosion resistance to coolant, fuel, and fission products; and high strength and ductility at operating temperatures. ZIRLO is an advanced zirconium based alloy that has the same or similar properties and advantages as Zircaloy-4 and was developed to support extended fuel burnup. WCAP-12610-P-A (Reference 5) provides a discussion of chemical and mechanical properties of the ZIRLO cladding material and a comparison to Zircaloy-4.

Comment [A1]: [4.2-02]

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.1.2 Stress-Strain Limits

Clad Stress

[*The volume average effective stress calculated with the Von Mises equation (considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences) is less than the 0.2 percent offset yield stress with due consideration to temperature and irradiation effects for Condition I and II events, WCAP-12488-A (Reference 1).*]* While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design limit. The allowable stress limits due to Condition III and IV loadings, described in subsection 4.2.1.5.3, are also applied to the fuel rod.

Clad Strain

[*The total plastic tensile creep strain due to uniform clad creep, and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than one percent from the unirradiated condition, WCAP-12488-A (Reference 1).*]* The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than one percent from the pre-transient value. These limits are consistent with proven practice.

4.2.1.1.3 Fatigue and Vibration

Fatigue

[*The usage factor due to cycle fatigue is less than 1.0, WCAP-12488-A (Reference 1).*]* That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure. The fatigue curve is based on a safety factor of two on the stress amplitude or a safety factor of 20 on the number of cycles, whichever is more conservative.

Vibration

Potential fretting wear due to vibration is prevented, giving confidence that the stress-strain limits are not exceeded during design life. Fretting of the clad surface can occur due to flow-induced vibration between the fuel rods and fuel assembly grid springs. Vibration and fretting forces may vary during the fuel life due to clad diameter creep down combined with grid spring relaxation.

4.2.1.1.4 Chemical Properties

Chemical properties of the ZIRLO cladding are discussed in WCAP-12610 (Reference 5).

Comment [A2]: [4.2-02]

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.2 Fuel Material

4.2.1.2.1 Thermal-Physical Properties

The center temperature of the hottest pellet is below the melting temperature of the uranium dioxide. The melting temperature of unirradiated uranium dioxide, 5080°F, decreases by 58°F per 10,000 megawatt days per metric ton of uranium, as discussed in WCAP-9179 (Reference 4). Fuel melting will not occur at the overpower limit for Condition I or II events. This provides sufficient margin for uncertainties as described in subsection 4.4.2.9.

The nominal design density of the fuel is approximately 95.5 percent of the theoretical density. Additional information on fuel properties is provided in WCAP-9179 (Reference 4).

4.2.1.2.2 Fuel Densification and Fission Product Swelling

The design bases and models used for fuel densification and swelling are provided in WCAP-10851-P-A (Reference 7), and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.1.2.3 Chemical Properties

WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5) provide the basis for justifying that no adverse chemical interactions occur between the fuel and its adjacent material.

4.2.1.3 Fuel Rod Performance

4.2.1.3.1 Fuel Rod Models

The basic fuel rod models and the ability to predict fuel rod operating characteristics are given in WCAP-15063-P-A, Revision 1 (Reference 21) and subsection 4.2.3.

4.2.1.3.2 Mechanical Design Limits

Cladding collapse is precluded during the fuel rod design lifetime. Current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps in the pellet stack necessary for clad flattening do not occur and therefore, clad flattening will not occur. Clad flattening methodologies are described in WCAP-13589-A, (Reference 8) and WCAP-8377 (Reference 22).

The rod internal gas pressure remains below the value which causes the fuel/clad diametral gap to increase due to outward cladding creep during steady-state operation. Rod pressure is also limited such that extensive departure from nucleate boiling propagation does not occur as discussed in WCAP-8963-P-A (Reference 9).

4.2.1.4 Spacer Grids

4.2.1.4.1 Mechanical Limits and Materials Properties

The grid component strength criteria are based on experimental tests. The limit is established at the 95-percent confidence level on the true mean crush strength at operating temperature. This limit is sufficient to provide that, under worst-case combined seismic and pipe rupture event, the core will maintain a geometry amenable to cooling. As an integral part of the fuel

assembly structure, the grids satisfy the applicable fuel assembly design bases and limits defined in subsection 4.2.1.5.

The grid material and chemical properties are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5).

4.2.1.4.2 Vibration and Fatigue

The grids provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear within acceptable limits (defined in subsection 4.2.1.1).

4.2.1.5 Fuel Assembly Structural Design

As discussed in subsection 4.2.1, the structural integrity of the fuel assemblies is provided by setting design limits on stresses and deformations due to various non-operational, operational, and accident loads. These limits are applied to the design and evaluation of the top and bottom nozzles, guide thimbles, grids, and thimble joints. [*Design changes to the fuel assembly structure qualify for evaluation in WCAP-12488-A (Reference 1).*]*

The design bases for evaluating the structural integrity of the fuel assemblies are discussed in subsections 4.2.1.5.1 through 4.2.1.5.3.

4.2.1.5.1 Non-Operational

The non-operational load is a loading of 4 g axial (longitudinal) and 6 g lateral (transverse) with dimensional stability.

4.2.1.5.2 Normal Operation and Operational Transients (Condition I) and Events of Moderate Frequency (Condition II)

For the normal operation (Condition I) and upset (Condition II) conditions, the fuel assembly component structural design criteria are established for the two primary material categories, austenitic steels and zirconium alloys. The stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide. The maximum shear theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the largest numerical difference between the various principal stresses in a three-dimensional field. The design stress intensity value, S_m, for austenitic steels and zirconium alloys is given by the lowest of the following:

- One-third of the specified minimum tensile strength or two-thirds of the specified minimum yield strength at room temperature
- One-third of the tensile strength or 90 percent of the yield strength at temperature, but not to exceed two-thirds of the specified minimum yield strength at room temperature

The stress limits for the austenitic steel components are given below. Stress nomenclature follows the ASME Code, Section III.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Stress Intensity Limits

Categories	Limit
General primary membrane stress intensity	S_m
Local primary membrane stress intensity	1.5 S_m
Primary membrane plus bending stress intensity	1.5 S_m
Total primary plus secondary stress intensity	3.0 S_m

The zirconium alloy structural components, which consist of guide thimbles and fuel tubes, are in turn subdivided into two categories because of material difference and functional requirements. The fuel tube design criteria are covered separately in subsection 4.2.1.1. The maximum shear theory is used to evaluate the guide thimble design. For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

4.2.1.5.3 Infrequent Incidents (Condition III) and Limiting Faults (Condition IV)

Typical worse case abnormal loads during Conditions III and IV are represented by seismic and pipe rupture loadings. The design criteria for this category of loadings are as follows:

- Deflections or excessive deformation of components cannot interfere with capability of insertion of the control rods or emergency cooling of the fuel rods.

- The fuel assembly structural components stresses under faulted conditions are evaluated primarily using the methods outlined in Appendix F of the ASME Code, Section III. Since the current analytical methods use linear elastic analysis, the stress allowables are defined as the smaller value of 2.4 S_m or 0.70 S_u for primary membrane and 3.6 S_m or 1.05 S_u for primary membrane plus primary bending. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section for normal operating conditions. For the zirconium alloy components, the stress intensity limits are set at two-thirds of the material yield strength, S_y, at reactor operating temperature. This results in zirconium alloy stress limits being the smaller value of 1.6 S_y or 0.70 S_u for primary membrane and 2.4 S_y or 1.05 S_u for primary membrane plus bending.

- For conservative purposes, the zirconium alloy unirradiated properties are used to define the stress limits.

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Comment [A3]: [4.2-03]

The material and chemical properties of the fuel assembly components are given in WCAP-9179 (Reference 4) and WCAP-12610 (Reference 5). Subsection 4.2.3.4 discusses the spacer grid crush testing.

Thermal-hydraulic design is discussed in Section 4.4.

4.2.1.6 In-core Control Components

The in-core control components are subdivided into permanent and temporary devices. The permanent components are the rod cluster control assemblies, gray rod cluster assemblies, thimble plugs, and secondary neutron source assemblies. The temporary components are the primary neutron source assemblies (which are normally used only in the initial core) and the burnable absorber assemblies. For some reloads, the use of burnable absorbers may be necessary for power distribution control and/or to achieve an acceptable moderator temperature coefficient throughout core life (See Subsection 4.3.1.2.2). [*Design changes to the in-core control components qualify for evaluation using the criteria defined in WCAP-12488-A (Reference 1).*]*

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Comment [A4]: [4.2-04]

Deleted: , and the thimble plugs

Materials are selected for:

- Compatibility in a pressurized water reactor environment
- Adequate mechanical properties at room and operating temperatures
- Resistance to adverse property changes in a radioactive environment

Compatibility with interfacing components

Material properties are given in WCAP-9179 (Reference 4).

The design bases for the in-core control components are given in subsections 4.2.1.6.1 through 4.2.1.6.3.

4.2.1.6.1 Control Rods

Comment [A5]: [4.2-67]

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the RCCA and GRCA structural parts in addition to absorber cladding.

Design conditions considered under the ASME Code, Section III, are as follows:

- External pressure equal to the reactor coolant system operating pressure with appropriate allowance for overpressure transients
- Wear allowance equivalent to 1000 reactor trips
- Bending of the rod due to a misalignment in the guide thimble
- Forces imposed on the rods during rod drop
- Loads imposed by the accelerations of the control rod drive mechanism
- Radiation exposure during maximum core life. The RCCA absorber material temperature does not exceed its melting temperature (1454°F for silver-indium-cadmium [Ag-In-Cd]), (see WCAP-9179, Reference 4). Similarly, the GRCA absorber material temperature does not exceed its melting temperature (6116°F for tungsten), (see WCAP-16943-P-A, Reference 25).

Comment [A6]: [4.2-05]

- Temperature effects at operating conditions

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4.2.1.6.2 Burnable Absorber Rods

For Conditions I and II, the stress categories and strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the burnable absorber cladding. For abnormal loads during Conditions III and IV, code stresses are not considered limiting. Failures of the burnable absorber rods during these conditions must not interfere with reactor shutdown or emergency cooling of the fuel rods. The burnable absorber material is nonstructural. The structural elements of the burnable absorber rod are designed to maintain the absorber geometry even if the absorber material is fractured.

To reduce the dissolved boron requirement for control of excess reactivity, wet annular burnable absorber (WABA) rods have been incorporated in the first core design. The burnable absorber material is boron carbide contained in an alumina matrix. Thermal-physical and gas release properties of alumina-boron carbide are described in WCAP-9179 (Reference 4) and WCAP-10021-P-A (Reference 10). Discrete burnable absorber rods are designed so that the absorber temperature does not exceed 1200°F during normal operation or an overpower transient. The 1200°F maximum temperature helium gas release in a discrete burnable absorber rod will not exceed 30 percent of theoretical. See WCAP-10021-P-A (Reference 10).

Deleted: burnable absorber rods have been incorporated in the core design. In the first core, the burnable absorber rods (Pyrex) consist of borosilicate glass tubes contained within Type 304 stainless steel tubular cladding, which is plugged and seal welded at the ends to encapsulate the glass. The absorber material temperature does not exceed its design limit of 1220°F. Mechanical and thermal design and nuclear evaluation of the burnable absorber rods are described in WCAP-7113 (Reference 23).¶
An alternative discrete burnable absorber is the wet annular burnable absorber (WABA).

Comment [A6]: [4.2-06]

4.2.1.6.3 Neutron Source Rods

The neutron source rods are designed to withstand the following:

- The external pressure equal to reactor coolant system operating pressure with appropriate allowance for overpressure transients
- An internal pressure equal to the pressure generated by released gases over the source rod life

4.2.1.7 Surveillance Program

Subsection 4.2.4.6 discusses the testing and fuel surveillance operation experience program that has been and is being conducted to verify the adequacy of the fuel performance and design bases. Fuel surveillance and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and to confirm that the design bases and safety criteria are satisfied.

4.2.2 Description and Design Drawings

The fuel assembly, fuel rod, and in-core control component design data is given in Table 4.3-1.

Each fuel assembly consists of 264 fuel rods, 24 guide thimbles, and 1 instrumentation tube arranged within a supporting structure. The instrumentation thimble is located in the center position and provides a channel for insertion of an in-core neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, a gray rod cluster assembly, a neutron source assembly, a burnable absorber assembly, or a thimble plug, depending on the position

of the particular fuel assembly in the core. Figure 4.2-1 shows a cross-section of the fuel assembly array, and Figure 4.2-2 shows a fuel assembly full-length view.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. The fuel rods are supported within the fuel assembly structure by fourteen grids (top grid (1), bottom grid (1), intermediate grids (8) and intermediate flow mixer (IFM) grids (4)), plus one protective grid. The top grid is fabricated from nickel-chromium-iron Alloy 718. The bottom grid is fabricated from nickel-chromium-iron Alloy 718. The intermediate grids and the IFM grids are fabricated from ZIRLO (see WCAP-12610-P-A, Reference 5). Top, bottom, and intermediate grids provide axial and lateral support to the fuel rods. In addition, the four IFM grids located near the center of the fuel assembly and between the intermediate grids provide additional fuel rod restraint. The protective grid, in combination with the debris filter bottom nozzle (DFBN), the protective zirconium oxide coated fuel cladding, and the long, solid fuel rod bottom end plug, provide debris failure mitigation.

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Comment [A8]: [4.2-02]

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Fuel assemblies are installed vertically in the reactor vessel and stand upright on the lower core plate, which is fitted with alignment pins to locate and orient each assembly. After the fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears down against the hold-down springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate. The assembly is oriented with respect to the handling tool and the core by means of a pin inserted into this indexing hole. Visual confirmation of proper orientation is also provided by an engraved identification number on the opposite corner clamp.

4.2.2.1 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets 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 selected for its mechanical properties and low neutron absorption cross-section (see WCAP-12610-P-A, Reference 5). Figure 4.2-3 shows a schematic of the fuel rod. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly, to allow greater axial expansion at the pellet centerline and to increase the void volume for fission gas release. The ends of each pellet also have a small chamfer at the outer cylindrical surface which improves manufacturability, and mitigates potential pellet damage due to fuel rod handling.

Comment [A9]: [4.2-02]

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Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. To facilitate the extended burnup capability necessitated by longer operating cycles, the fuel rod is designed with two plenums (upper and lower) to accommodate the additional fission gas release. The upper plenum volume is maintained by a fuel pellet hold-down spring. The lower plenum volume is maintained by a standoff assembly.

Shifting of the fuel within the clad during handling or shipping, prior to core loading, is prevented by a stainless steel helical spring which bears on top of the fuel pellet stack. Assembly consists of plugging and welding the bottom of the cladding, installing the bottom plenum spacer assembly, fuel pellets and top plenum spring, and then plugging and welding the top of the rod. The solid bottom end plug has an internal grip feature and tapered end to facilitate fuel rod loading during fuel assembly fabrication and reconstitution. Additionally, the bottom end plug is designed to be sufficiently long to extend through the protective grid. The bottom section of the fuel rod has a protective zirconium oxide coated surface feature. Use of the protective grid with a longer end plug and the debris filter bottom nozzle, in addition to the coated cladding surface, constitutes a three-level debris protection package, which enhances the fuel reliability performance against trapped debris. This precludes any breach in the fuel rod pressure boundary due to clad fretting wear induced by debris trapped at the bottom section of the fuel assembly.

The fuel rods are internally pressurized with helium during the welding process to minimize compressive clad stresses and prevent clad flattening under reactor coolant operating pressures. The fuel rods are pre-pressurized and designed so that:

- The internal gas pressure mechanical design limit referred to in subsection 4.2.1.3 is not exceeded
- The cladding stress-strain limits (subsection 4.2.1.1) are not exceeded for Condition I and II events
- Clad flattening will not occur during the fuel core life

The AP1000 fuel rod design may also include axial blankets. The axial blankets consist of fuel pellets of a reduced enrichment at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage axially and improve fuel utilization. The axial blankets use chamfered pellets that are longer than the enriched pellets to help prevent accidental mixing during manufacturing. Furthermore, axial blankets have no impact on the source range detector response, since the reduction in power from the axial blanket is limited to the top and bottom 0.67 feet of the core, while the source range detectors are centered typically about three feet from the bottom of the core.

The AP1000 fuel rod design may also include annular fuel pellets in the top and bottom 8 inches of the fuel stack. These pellets can be either fully enriched or partially enriched. The annular fuel pellets provide additional void volume in the fuel rod to accommodate fission gas release.

The AP1000 fuel rods include integral fuel burnable absorbers. The integral fuel burnable absorbers are boride-coated fuel pellets. The boride-coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of integral fuel burnable absorber rods within an assembly may vary depending on specific application. See WCAP-12610-P-A (Reference 5).

Comment [A9]: [4.2-07]

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Deleted: or fuel pellets containing gadolinium oxide mixed with uranium oxide

4.2.2.2 Fuel Assembly Structure

As shown in Figure 4.2-2, the fuel assembly structure consists of a bottom nozzle, top nozzle, fuel rods, guide thimbles, and grids.

4.2.2.2.1 Bottom Nozzle

The bottom nozzle serves as the bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The nozzle is fabricated from Type 304 and Grade CF-3 stainless steel and consists of a perforated plate, and casting which incorporates a skirt and four angle legs with bearing pads. Figure 4.2-2 illustrates this concept. The legs and skirt form a plenum to direct the inlet coolant flow to the fuel assembly. The perforated plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide thimbles by locked thimble screws, which penetrate through the nozzle and engage with a threaded plug in each guide thimble.

Comment [A10]: [4.2-08]

Coolant flows from the plenum in the bottom nozzle, upward through the penetrations in the plate, to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

In addition to serving as the bottom structural element of the fuel assembly, the bottom nozzle also functions as a debris filter. The bottom nozzle perforated plate contains a multiplicity of flow holes which are sized to minimize passage of detrimental debris particles into the active fuel region of the core while maintaining sufficient hydraulic and structural margins. Furthermore, the skirt provides improved bottom nozzle structural stability and increased design margins to reduce damage due to abnormal handling.

Axial loads (from top nozzle hold-down springs) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing pads that mate with locating pins in the lower core plate. Lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

The AP1000 bottom nozzle also has a reconstitution design feature which facilitates the easy removal of the nozzle from the fuel assembly. This design incorporates a thimble screw with a circular locking cup located around the screw head. The locking cup is crimped into a local spherical radius relief on the bottom nozzle. To remove the bottom nozzle, a counterclockwise torque is applied to the thimble screw until the locking cup (detents) is relaxed and the thimble screw is removed. This reconstitutable design permits the remote unlocking, the removal, and the relocking of the thimble screws, as the same or a new bottom nozzle is reattached to the fuel assembly.

4.2.2.2.2 Top Nozzle

The reconstitutable top nozzle functions as the upper structural component of the fuel assembly and, in addition, provides a partial protective housing for the rod cluster control assembly, discrete burnable absorber, or other core components. As shown in Figure 4.2-2, the top nozzle assembly includes four sets of hold-down springs, which are secured to the top nozzle top plate. The springs are made of nickel-chromium-iron Alloy 718. The other top nozzle components are made of Type 304L and Grade CF-3 stainless steel.

Comment [A11]: [4.2-09]

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The adapter plate is provided with round penetrations and slots (with semicircular ends) to permit the flow of coolant upward through the top nozzle. Other round holes are provided in the adapter plate to accept (guide thimble) inserts which are mechanically locked to the adapter plate using a lock tube. The unique design of the insert joint and lock tube are the key design features of the reconstitutable top nozzle.

The ligaments in the adapter plate cover the top of the fuel rods precluding any upward ejection of the fuel rods from the fuel assembly. The enclosure is a box-like structure which establishes the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the rod cluster control assembly, burnable absorber assembly, or other components. Hold-down springs are mounted on the top plate and are retained by retaining pins located at diagonally opposite corners of the top plate.

The top plate also contains integral pads located on the two remaining top nozzle corners. The pads include alignment holes which, when fully engaged with the reactor internals upper core plate guide pins, provide proper alignment to the fuel assembly, reactor internals, and rod control cluster assembly.

As shown in Figure 4.2-4, to remove the top nozzle assembly a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tubes from the inserts. After the lock tubes have been removed, the nozzle assembly is removed by raising it off the upper slotted ends of the nozzle inserts, which deflect inwardly under the axial lift load.

With the top nozzle assembly removed, direct access is provided for fuel rod examination or replacement. Reconstitution is completed by the remounting of the nozzle assembly and the insertion of lock tubes. Details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in WCAP-10444-P-A (Reference 11).

4.2.2.2.3 Guide Thimbles and Instrument Tube

The guide thimbles are structural members that provide channels for the neutron absorber rods, burnable absorber rods, neutron source rods, or other assemblies. Each guide thimble is fabricated from Zircaloy-4 or ZIRLO with constant OD and ID over the entire length. Separate dashpot tubes, which are made from Zircaloy-4 or ZIRLO tubing, are inserted into the bottom portion of the guide thimble tubes. The larger tube diameter at the top section provides a relatively large annular area necessary to permit rapid control rod insertion during a reactor trip, as well as to accommodate the flow of coolant during normal operation. Holes are provided on the guide thimble above the dashpot to reduce the rod drop time. The lower portion of the guide thimble with the dashpot tube results in a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug, which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation.

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As stated previously, the AP1000 fuel assembly includes a reconstitutable top nozzle as a standard feature. To accommodate the reconstitutable feature, the top end of the zirconium alloy guide thimble is fastened to a tubular sleeve, or insert, by a three tier expansion bulge joint. An expansion tool is inserted inside the nozzle insert and guide thimble to the proper

elevation. The four lobes on the expansion tool force the guide thimble and insert outward locally to a predetermined diameter, therefore joining the two components.

Upon installation of the top nozzle assembly, the bulge near the top of the nozzle insert is captured in a corresponding groove in the hole of the top nozzle adapter plate. As shown in Figure 4.2-4, the mechanical connection between the nozzle insert-guide thimble and top nozzle is made by insertion of a lock tube into the insert. The design of the top grid sleeve-guide thimble and top nozzle insert-guide thimble bulge joint connections have been mechanically tested and found to meet applicable design criteria.

The fuel rod support grids are secured to the guide thimbles using a similar bulge joint connection to create an integral structure. Attachment of the intermediate mixing vane and intermediate flow mixer (IFM) zirconium alloy grids to the guide thimbles is performed using the fastening technique depicted in Figures 4.2-5 and 4.2-6.

Comment [A13]: [4.2-10]

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The intermediate mixing vane and intermediate flow mixer grids employ a single tier bulge connection between the grid sleeve and guide thimble as compared to the two tier bulge connection used for the top grid. The design of the single tier bulge joint connection has also been mechanically tested and meets the design requirements.

Comment [A14]: [4.2-11]

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The bottom nickel-chromium-iron Alloy 718 grid is secured to the guide thimble assembly by a double tier bulge connection between the grid sleeve and guide thimble. The design of the double tier bulge joint connection has also been mechanically tested and meets the design requirements.

The lower end of the guide thimble is fitted with a welded end plug. The nickel-chromium-iron Alloy 718 protective grid is secured to the guide thimble assembly by nickel-chromium-iron Alloy 718 spacers that are spot-welded to the grid. As shown in Figure 4.2-7, the spacer is captured between the guide thimble end plug and the bottom nozzle by means of a (thimble) locking screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of zirconium alloy guide thimbles in 1969.

The central instrumentation tube in each fuel assembly is constrained by seating in counterbores located in both top and bottom nozzles. The instrumentation tube has a constant diameter and provides an unrestricted passageway for the in-core neutron detector which enters the fuel assembly from the top nozzle. Furthermore, the instrumentation tube is secured to the top, bottom, and mid-grids with bulge joint connections similar to those previously discussed for securing the grids to the guide thimbles.

Comment [A15]: [4.2-12]

4.2.2.2.4 Grid Assemblies

As shown in Figure 4.2-2, the fuel rods are supported at intervals along their lengths by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps assembled and interlocked into an egg-crate type arrangement with the straps permanently joined at their points of intersection. The straps may contain springs, support dimples, and mixing vanes; or any such combination.

Two types of structural grid assemblies are used on the AP1000 fuel assembly. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The other type, located at the top and bottom of the assembly, does not contain mixing vanes on the internal straps. The outside straps on the grids contain mixing vanes that, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

Because of its corrosion resistance and high strength properties, the bottom grid material chosen for the AP1000 fuel assembly design is nickel-chromium-iron Alloy 718. The top grid is fabricated from nickel-chromium-iron Alloy 718. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies are designed to allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The eight intermediate (mixing vane), or structural grids on the AP1000 fuel assembly are made of ZIRLO. This material was selected to take advantage of the material's inherent low neutron capture cross-section. The zirconium alloy grids have thicker straps than the nickel-chromium-iron alloy grids. The zirconium alloy grid incorporates the same grid cell support configuration as the nickel-chromium-iron alloy grid. The zirconium alloy interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the nickel-chromium-iron alloy grid joints (except the protective grid) are brazed. The interlocking strap joints for the protective grid are also fabricated by laser welding. The mixing vanes incorporated in the zirconium alloy intermediate grids induce additional flow mixing among the various flow channels in a fuel assembly as well as between adjacent fuel assemblies. This additional flow mixing enhances thermal performance.

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As shown in Figure 4.2-2, the intermediate flow mixer grids are located at selected spans between the zirconium alloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hotter fuel assembly spans. Each intermediate flow mixer grid cell contains four dimples that are designed to prevent mid-span channel closure in the spans containing intermediate flow mixers and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the intermediate flow mixer grid can accomplish its flow mixing objective with minimal pressure drop.

The intermediate flow mixer (IFM) grids, like the mixing vane grids, are fabricated from ZIRLO. The intermediate flow mixer grids are manufactured using the same basic techniques as the zirconium alloy structural grid assemblies and are joined to the guide thimbles via sleeves which are welded at the bottom of appropriate grid cells.

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Grid impact testing has been performed on zirconium alloy structural grids and the intermediate flow mixer grids indicative of the AP1000 design. The purpose of the testing was to determine the dynamic buckling, or crush, strength of the grids. The grid impact testing was performed at an elevated temperature of 600°F. This temperature is a conservative value representing the core average temperature at the mid-grid locations.

The intermediate flow mixer grids are not intended to be structural members. The intermediate flow mixer grids do, however, share the loads of the structural grids during

faulted loading and, as such, contribute to enhance the load carrying capability of the AP1000 fuel assembly.

The dynamic crush strength of the AP1000 structural grids and intermediate flow mixer grids envelope the calculated grid impact loading during combined seismic and pipe rupture events. A coolable geometry is, therefore, provided at the intermediate flow mixer grid elevations, as well as at the structural grid elevations.

4.2.2.3 In-core Control Components

Reactivity control is provided by neutron absorbing rods, gray rods, burnable absorber rods, and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as:

- Fuel depletion and fission product buildup
- Cold to hot, zero power reactivity changes
- Reactivity change produced by intermediate-term fission products such as xenon and samarium
- Burnable absorber depletion

The chemical and volume control system, which is used to adjust the level of boron in the coolant, is discussed in Section 9.3.

The rod cluster control assemblies provide reactivity control for:

- Shutdown
- Reactivity changes due to coolant temperature changes in the power range
- Reactivity changes associated with the power coefficient of reactivity
- Reactivity changes due to void formation

A negative power coefficient is maintained at hot, full-power conditions throughout the entire cycle to reduce possible deleterious effects caused by a positive coefficient during pipe rupture or loss-of-flow accidents. The first fuel cycle needs more excess reactivity than subsequent cycles due to the loading of fresh (unburned) fuel. Since soluble boron alone is insufficient to provide a negative moderator coefficient, burnable absorber assemblies are also used. Use of burnable absorber assemblies during reloads is discussed in subsection 4.3.1.2.2.

The most effective reactivity control components are the rod cluster control assemblies and the corresponding drive rod assemblies, which along with the gray rod cluster assemblies, are the only kinetic parts in the reactor. Figure 4.2-8 identifies the rod cluster control and drive rod assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly, guide thimbles, and control rod drive mechanism. The arrangement for the gray rod cluster assemblies is the same.

As shown in Figure 4.2-8, the guidance system for the rod cluster control assembly is provided by the guide thimbles. The guide thimbles provide two regimes of guidance: first,

in the lower section, a continuous guidance system provides support immediately above the core, which protects the rod against excessive deformation and wear caused by hydraulic loading. Second, the region above the continuous section provides support and guidance at uniformly spaced intervals.

As shown in Figure 4.2-9, the envelope of support is determined by the pattern of the control rod cluster. The guide thimbles provide alignment and support of the control rods, spider body, and drive rod while maintaining trip times at or below required limits.

Subsections 4.2.2.3.1 through 4.2.2.3.4 describe each reactivity control component in detail. The control rod drive mechanism assembly is described in subsection 3.9.4. The neutron source assemblies provide a means of monitoring the core during periods of low neutron activity.

4.2.2.3.1 Rod Cluster Control Assemblies

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, that is, power and temperature variations. Two nuclear design criteria have been employed for selection of the control group. First, the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to confirm that the power capability is met. The control and shutdown groups provide adequate shutdown margin.

As illustrated in Figure 4.2-9, a rod cluster control assembly is comprised of a group of individual neutron absorber rods fastened at the top end to a common spider assembly.

The absorber material used in these rods is silver-indium-cadmium alloy, which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase worth. As such, these rods are sometimes referred to as "black" rods. As shown in Figure 4.2-10, the absorber material is in the form of solid bars sealed in cold-worked stainless steel tubes. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions.

The control rods have bottom plugs with bullet-like tips to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles.

The material used in the absorber rod end plugs is Type 308 or 308L stainless steel. The design stresses used for these materials are the same as those defined in the ASME B&PV Code for Type 304 or 304L stainless steel which have essentially the same strength properties as Type 308 and 308L stainless steel, respectively.

The allowable stresses used as a function of temperature are listed in Table 2A of the ASME Code, Section II, Part D. The fatigue strength for the Type 308 or 308L material is based on the S-N curve for austenitic stainless steels in Figure I-9.2 of the ASME Code, Section III.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Internal groove-like profiles to facilitate

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handling tool and drive rod assembly connection are machined into the upper end of the hub. Coil springs inside the spider body absorb the impact energy at the end of a trip insertion. The radial vanes may either be joined to the hub by welding and brazing, and the fingers joined to the vanes by brazing, or the vanes and fingers may be integral with the spider body. A bolt, which holds the springs and retainer, is threaded into the hub within the skirt and welded to prevent loosening while in service.

The components of the spider assembly are made from Types 304, 304L and/or CF-3 (casting equivalent of 304L) stainless steel except for the retainer, which is of Type 630 material, and the springs, which are nickel-chromium-iron Alloy 718.

The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then secured with a locking device. The end plug below the thread is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length of the rod cluster control assembly is such that, when the assembly is withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

4.2.2.3.2 Gray Rod Cluster Assemblies

Externally the mechanical design of the gray rod cluster assembly is identical to the rod cluster control assembly. In addition, the control rod drive mechanism and the interface with the fuel assemblies and guide thimbles are identical to those of the rod cluster control assembly.

As shown in Figure 4.2-11, the gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that the absorber material consists of tungsten encapsulated in a nickel-chromium-iron Alloy 718 sleeve and clad with stainless steel cladding which has the same outer diameter as the rod cluster control assembly cladding. The lower portion of the rodlets consists of a stainless steel spacer.

The gray rod cluster assemblies are used in base load operation and load follow maneuvering and provide a mechanical shim to replace the use of changes in the concentration of soluble boron, that is, a chemical shim, normally used for this purpose. The AP1000 uses 53 rod cluster control assemblies and 16 gray rod cluster assemblies.

4.2.2.3.3 Burnable Absorber Assembly

Each burnable absorber assembly consists of discrete burnable absorber rods attached to a hold-down assembly. Figure 4.2-12 shows the burnable absorber assemblies. When needed for nuclear considerations, burnable absorber assemblies are inserted into selected thimbles within fuel assemblies.

The wet annular burnable absorber rods (WABA) consist of pellets of alumina-boron carbide material contained within zirconium alloy tubes. These zirconium alloy tubes, which form the

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Comment [A22]: [4.2-17]

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Deleted: 24 rodlets are stainless steel while the remaining 12 contain the reduced diameter silver-indium-cadmium

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Comment [A28]: [4.2-23]

Deleted: The burnable absorber rods (Pyrex) consist of borosilicate glass tubes contained within Type 304 stainless steel tubular cladding, which is plugged and seal welded at the ends to encapsulate the glass. The burnable absorber assembly is shown in Figure 4.2-13.

Comment [A29]: [4.2-24]

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outer clad for the burnable absorber rod, are plugged, pressurized with helium, and seal-welded at each end to encapsulate the stack of absorber material. The absorber stack length, shown in Figure 4.2-12, is positioned axially within the burnable absorber rod by the use of a zirconium alloy bottom-end spacer as necessary.

Comment [A30]: [4.2-25]

The burnable absorber rods in each fuel assembly are grouped and attached together at the top end of the rods to a hold-down assembly by a flat, perforated retaining plate, which fits within the fuel assembly top nozzle and rests on the adapter plate.

The retaining plate and the burnable absorber rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. With this arrangement, the burnable absorber rods cannot be ejected from the core by flow forces. Each rod is attached to the baseplate by a nut that is crimped into place.

4.2.2.3.4 Neutron Source Assemblies

The purpose of a neutron source assembly is to provide a base neutron level to give confidence that the detectors are operational and responding to core multiplication neutrons. For the first core, a neutron source is placed in the reactor to provide a positive neutron count of at least two counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily during subcritical modes of core operation.

The source assembly also permits detection of changes in the core multiplication factor during core loading, refueling, and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Changes in the multiplication factor can be detected during addition of fuel assemblies while loading the core, changes in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods are used. The primary source rod, containing a radioactive material, spontaneously emits neutrons during initial core loading, reactor startup, and initial operation of the first core. After the primary source rod decays beyond the desired neutron flux level, neutrons are then supplied by the secondary source rod. The secondary source rod contains a stable material, which is activated during reactor operation. The activation results in the subsequent release of neutrons.

Four source assemblies are typically installed in initial load of the reactor core: two primary source assemblies and two secondary source assemblies. Each primary source assembly contains one primary source rod and a number of burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of secondary source rodlets. Figure 4.2-14 shows the primary source assembly. Figure 4.2-15 shows the secondary source assembly.

Neutron source assemblies are employed at opposite sides of the core. The source assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected locations.

As shown in Figures 4.2-14 and 4.2-15, the source assemblies contain a hold-down assembly identical to that of the burnable absorber assembly.

The primary and secondary source rods both use the same cladding material as the absorber rods. The secondary source rods contain antimony-beryllium pellets stacked to a height of approximately 88 inches. The primary source rods contain capsules of californium (plutonium-beryllium possible alternate) source material and alumina spacers to position the source material within the cladding. The rods in each assembly are fastened at the top end to a hold-down assembly.

The other structural members, except for the springs, are constructed of Type 304, 304L, and 308L stainless steel. The springs exposed to the reactor coolant are nickel-chromium-iron Alloy 718.

Comment [A31]: [4.2-26]

4.2.3 Design Evaluation

[*The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria of*]* Section 4.2 of the Standard Review Plan, the mechanical design bases of subsection 4.2.1 and [*the Fuel Criteria Evaluation Process per WCAP-12488-A (Reference 1)*]*, and other interfacing nuclear and thermal and hydraulic design bases specified in Sections 4.3 and 4.4.

Effects of Conditions II, III, IV or anticipated transients without trip on fuel integrity are presented in Chapter 15.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rods whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria, the limiting rod is the lead burnup rod of a fuel region. In other instances, it may be the maximum power or the minimum burnup rod. For the most part, no single rod is limiting with respect to all the design criteria.

After identifying the limiting rod(s), an analysis is performed to consider the effects of rod operating history, model uncertainties, and dimensional variations. To verify adherence to the design criteria, the evaluation considers the effects of postulated transient power changes during operation consistent with Conditions I and II. These transient power increases can affect both rod average and local power levels. Parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses, these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and assembly models used for the performance evaluations are documented and maintained under an appropriate control system. Material properties used in the design evaluations are given in WCAP-12610 (Reference 5).

4.2.3.1 Cladding

4.2.3.1.1 Vibration and Wear

Fuel rod vibrations are flow induced. The effect of vibration on the fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

The reaction force on the grid supports, due to rod vibration motions, is also small and is much less than the spring preload. Adequate fuel clad spring contact is maintained. No significant wear of the clad or grid supports is predicted during the life of the fuel assembly based on out-of-pile flow tests, performance of similarly designed fuel in operating reactors, and design analyses.

Clad fretting and fuel vibration has been experimentally investigated, as shown in WCAP-8278 (Reference 13).

4.2.3.1.2 Fuel Rod Internal Pressure and Cladding Stresses

A burnup-dependent fission gas release model (WCAP-15063-P-A, Revision 1 [Reference 21]) is used in determining the internal gas pressure as a function of irradiation time. The plenum volume of the fuel rod has been designed to provide that the maximum internal pressure of the fuel rod will not exceed the value which would cause:

- The fuel/clad diametral gap to increase during steady-state operation
- Extensive departure from nucleate boiling propagation to occur (Reference 26)

Comment [A32]: [4.2-27]

The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure. Because of the pre-pressurization with helium, the volume average effective stresses are always less than approximately 14,000 psi at the pressurization level used in the AP1000 fuel rod design. Stresses due to the temperature gradient are not included in this average effective stress because thermal stresses are, in general, negative at the clad inside diameter and positive at the clad outside diameter, and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady-state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at beginning-of-life due to low internal gas pressure and decreases as rod power increases. Thermal stresses are maximum in the maximum power rod due to the larger temperature gradient and decrease as the rod power is decreased.

The internal gas pressure at beginning-of-life ranges from approximately 200 to 750 psi for typical lead burnup fuel rods. The total tangential stress at the clad inside diameter at beginning-of-life is approximately 19,500 psi compressive (approximately 18,500 psi due to ΔP and approximately 1,000 due to ΔT) for a low-power rod operating at four kilowatts/foot. Total tangential stress is approximately 20,500 psi compressive (approximately 18,000 psi due to ΔP and approximately 2,500 psi due to ΔT) for a high-power rod operating at 10 kilowatts/foot. However, the volume average effective stress at beginning-of-life is between approximately 13,500 psi (high-power rod) and approximately 14,000 psi (low-power rod). These stresses are substantially below even the unirradiated clad yield strength (approximately 55,500 psi) at a typical clad mean operating temperature of 700°F.

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Swelling of the fuel pellet can result in small clad strains (less than one percent) for expected discharge burnups, but the associated clad stresses are very low because of clad creep (thermal- and irradiation-induced creep). The one percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow. A detailed discussion of fuel rod performance is given in subsection 4.2.3.3.

4.2.3.1.3 Material and Chemical Evaluation

ZIRLO clad has a high corrosion resistance to the coolant, fuel, and fission products. As shown in WCAP-8183 (Reference 3), there is considerable pressurized water reactor operating experience on the capability of Zircaloy-4 as a clad material. ZIRLO, an advanced zirconium based alloy, has equal or better corrosion resistance than Zircaloy-4 (see WCAP-12610-P-A, [Reference 5]). Controls on fuel fabrication specify maximum moisture levels to preclude clad hydriding.

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Metallographic examination of irradiated commercial fuel rods has shown occurrences of fuel/clad chemical interaction. Reaction layers of less than one mil in thickness have been observed between fuel and clad at limited points around the circumference. Metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out-of-pile tests have shown that in the presence of high clad tensile stresses, large concentrations of iodine can chemically attack the zirconium alloy tubing and may lead to eventual clad cracking. Extensive post-irradiation examination has produced no evidence that this mechanism has been operative in Westinghouse commercial pressurized water reactor fuel.

4.2.3.1.4 Rod Bowing

WCAP-8691 (Reference 14) presents the model used for evaluation of AP1000 fuel rod bowing. This model has been used for bow assessment in 14x14, 15x15, and 17x17 type cores.

4.2.3.1.5 Consequences of Power Coolant Mismatch

Consequences of power coolant mismatch are discussed in Chapter 15.

4.2.3.1.6 Creep Collapse and Creepdown

This subject and the associated irradiation stability of cladding have been evaluated. In WCAP-13589-A (Reference 8), it is shown that current generation Westinghouse fuel is sufficiently stable with respect to fuel densification. Significant axial gaps do not form in the pellet stack, preventing clad collapse from occurring. The design basis of no clad collapse during planned core life is therefore satisfied. Cladding collapse analyses, if required, would be performed using the methods described in WCAP-8377 (Reference 22).

4.2.3.2 Fuel Materials Considerations

Sintered, high-density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration, although limited fuel erosion can occur. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products, including those which are gaseous or highly volatile.

Observations from several early Westinghouse pressurized water reactors as discussed in WCAP-8218-P-A (Reference 6) have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets can result in local and distributed gaps in the fuel rods. The densification process is related to the elimination of very small as-fabricated porosity in the fuel during irradiation. Early fuels were intentionally manufactured to low initial density and were undersintered, which resulted in a large fraction of very small pores. Densification behavior in current fuel is controlled by improved manufacturing process controls and by specifying a nominal 95.5 percent initial fuel density, which results in reduced levels of small, densifying porosity.

The evaluation of fuel densification effects and the treatment of fuel swelling and fission gas release are described in WCAP-13589-A (Reference 8) and WCAP-15063-P-A, Revision 1 (Reference 21).

4.2.3.3 Fuel Rod Performance

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors are considered:

- Clad creep and elastic deflection
- Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup
- Internal pressure as a function of fission gas release, rod geometry, and temperature distribution

These effects are evaluated using fuel rod design models, as discussed in WCAP-15063-P-A, Revision 1 (Reference 21), that include appropriate models for time dependent fuel densification. With these interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and clad temperatures, and clad deflections are calculated. The fuel rod is divided into several axial sections and radially into a number of annular zones. Fuel density changes are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for clad flattening. It is limited, however, by the design criteria for the rod internal pressure, as discussed in subsection 4.2.1.3.

The gap conductance between the pellet surface and the clad inner diameter is calculated as a function of the composition, temperature and pressure of the gas mixture, and the gap size or contact pressure between the clad and pellet. After computing the fuel temperature for each pellet zone, the fractional fission gas release is assessed using an empirical model derived from experimental data, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate, which, in turn, is a function of burnup. Finally, the gas released is summed over the zones, and the pressure is calculated.

The model shows close agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures, and clad deflections, as detailed in WCAP-15063-P-A, Revision 1 (Reference 21). These data include variations in power, time, fuel density, and geometry.

4.2.3.3.1 Fuel/Cladding Mechanical Interaction

One factor in fuel element duty is potential mechanical interaction of the fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these, in turn, reduce clad life. The reduction of fuel/clad interaction is therefore a goal of design. The technology for using pre-pressurized fuel rods in Westinghouse pressurized water reactors has been developed to further this objective.

The gap between the fuel and clad is initially sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in fuel/clad contact. Once fuel/clad contact occurs, changes in power level result in changes in clad stresses and strains. By using pre-pressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod pre-pressurization delays the time at which fuel/clad contact occurs and, hence, significantly reduces the extent of cyclic stresses and strains experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability.

A two-dimensional (r,θ) finite element model has been established to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration herein is defined as the difference between the maximum clad stress in the θ direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium; and, as a result, the stress in the clad is close to zero. In subsequent cases the pellet power is increased in steps and the resultant fuel thermal expansion imposes tensile stress in the clad.

In addition to uniform clad stresses, stress concentrations develop in the clad adjacent to radial cracks in the pellet. These radial cracks have a tendency to open during a power increase, but the frictional forces between fuel and clad oppose the opening of these cracks and result in localized increases in clad stress. As the power is further increased, large tensile stresses exceed the ultimate tensile strength of uranium dioxide and additional cracks in the fuel pellet are created, limiting the magnitude of the stress concentration in the clad.

As part of the standard fuel rod design analysis, the maximum stress concentration evaluated from finite element calculations is added to the volume-averaged effective stress in the clad as determined from one-dimensional stress/strain calculations. The resultant clad stress is then compared to the temperature-dependent cladding yield stress to confirm that the stress/strain criteria are satisfied.

The transient evaluation method is described in the following paragraphs.

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad.

Power increases in commercial reactors can result from fuel shuffling (for example, a fuel assembly positioned near the core center for cycle 2 operation after operating near the periphery during cycle 1), reactor power escalation following extended reduced power operation, and full-length control rod movement. In the mechanical design model, lead rods are depleted using best-estimate power histories as determined by core physics calculations. During burnup, the amount of diametral gap closure is evaluated based upon the pellet expansion cracking model, clad creep model, and fuel swelling model. At various times during the depletion, the power is increased locally in the rod to the burnup-dependent attainable power density as determined by core physics calculations. The radial, tangential, and axial clad stresses resulting from the power increase are combined into a volume average effective clad stress.

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The von Mises criterion is used to determine whether the clad yield stress has been exceeded. This criterion states that an isotropic material in multi-axial stress will begin to yield plastically when the effective stress exceeds the yield stress as determined by an axial tensile test. The yield stress correlation is that for irradiated cladding, since fuel/clad interaction occurs at high burnup. In applying this criterion, the effective stress is increased by an allowance which accounts for stress concentrations in the clad adjacent to radial cracks in the pellet, prior to the comparison with the yield stress. This allowance was evaluated using a two-dimensional (r,θ) finite element model.

Slow transient power increases can result in large clad strains without exceeding the clad yield stress because of clad creep and stress relaxation. Therefore, in addition to the yield stress criterion, a criterion on allowable clad strain is necessary. Based upon high strain rate burst and tensile test data on irradiated tubing, one percent strain was determined to be a conservative lower limit on irradiated clad ductility and that was adopted as a design criterion.

In addition to the mechanical design models and design criteria, the AP1000 fuel rod design relies on performance data accumulated through transient power test programs in experimental and commercial reactors, and through normal operation in commercial reactors.

It is recognized that a possible limitation to the satisfactory behavior of the fuel rods in a reactor subjected to daily load follow is the failure of the cladding by low-cycle strain fatigue. During their normal residence time in the reactor, the fuel rods may be subjected to on the order of 1000 load follow cycles, with typical changes in power level from 50 to 100 percent of their steady-state values.

The assessment of the fatigue life of the fuel rod cladding is subjected to considerable uncertainty because of the difficulty of evaluating the strain range which results from the cyclic interaction of the fuel pellets and cladding. This difficulty arises, for example, from such highly unpredictable phenomena as pellet cracking, fragmentation, and relocation. Westinghouse investigated this particular phenomenon both analytically and experimentally. Strain fatigue tests on irradiated and nonirradiated hydrided Zircaloy-4 cladding were performed. These tests permitted the definition of a conservative fatigue-life limit and recommendation of a methodology to treat the strain fatigue evaluation of the Westinghouse-referenced fuel rod designs. (See WCAP-9500-P-A, Reference 15.)

Successful load follow operation has been performed on several reactors. There was no significant coolant activity increase that could be associated with the load follow mode of operation.

The Westinghouse analytical approach to strain fatigue is based on a comprehensive review of the available strain fatigue models. The review included the Langer-O'Donnell model (Reference 16) the Yao-Munse model, and the Manson-Halford model. Upon completion of this review, and using the results of the Westinghouse experimental programs as documented in WCAP-9500-P-A (Reference 15), it was concluded that the approach defined by Langer-O'Donnell would be retained and the empirical factors of their correlation modified to conservatively bound the results of the Westinghouse testing program.

The design equations followed the concept for the fatigue design criterion according to the ASME Code, Section III:

- The calculated pseudo stress amplitude (S_a) has to be multiplied by a factor of two to obtain the allowable number of cycles (N_f).
- The allowable cycles for a given S_a is five percent of N_f or a safety factor of 20 on cycles.

The lesser of the two allowable numbers of cycles is selected. The cumulative fatigue life fraction is then computed as:

$$\sum_{1}^{k} \frac{n_k}{N_{fk}} \leq 1$$

where:

n_k = number of diurnal cycles of mode k.

N_{fk} = number of allowable cycles.

4.2.3.3.2 Irradiation Experience

Westinghouse fuel operational experience is presented in CENPD-404-P-A (Reference 27). Additional test assembly and test rod experience is given in WCAP-10125-P-A (Reference 2).

Comment [A35]: [4.2-29]

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4.2.3.3.3 Fuel and Cladding Temperature

The methods used for evaluation of fuel rod temperatures are presented in subsection 4.4.2.11.

4.2.3.3.4 Potentially Damaging Temperature Effects During Transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad/pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad/pellet interaction is discussed in subsection 4.2.3.3.1.

Clad flattening has been observed in some operating power reactors. This is no longer a concern because clad flattening is precluded during the fuel residence in the core (subsection 4.2.3.1) by the use of stable fuel.

Potential differential thermal expansion between the fuel rods and the guide thimbles during a transient is considered in the design. Excessive bowing of the fuel rods is precluded because the grid assemblies allow axial movement of the fuel rods relative to the grids. Specifically, thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods.

4.2.3.3.5 Fuel Element Burnout and Potential Energy Release

As discussed in subsection 4.4.2.2, the core is protected from departure from nucleate boiling over the full range of possible operating conditions. In the extremely unlikely event that departure from nucleate boiling should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is a potential for chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following departure from nucleate boiling, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

4.2.3.3.6 Coolant Flow Blockage Effects on Fuel Rods

The coolant flow blockage effects on fuel rods are presented in subsection 4.4.4.7.

4.2.3.4 Spacer Grids

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is maintained through the clamping force of the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in WCAP-8236 (Reference 17) and WCAP-10444-P-A (Reference 11).

4.2.3.5 Fuel Assembly

4.2.3.5.1 Stresses and Deflections

The fuel assembly component stress levels are limited by the design. For example, stresses in the fuel rod due to thermal expansion and zirconium alloy irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that zirconium alloy irradiation growth does not result in rod end interference. Stresses in the fuel assembly caused by tripping of the rod cluster control assembly have little influence on fatigue usage margin because of the small number of events during the life of an assembly. Assembly components and prototype

fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping have been established at 4 g axial and 6 g lateral. Accelerometers are permanently placed in the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Experience indicates that loads that exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components, such as the grid assembly, sleeves, inserts, and structure joints, have been performed to confirm that the shipping design limits do not result in impairment of fuel assembly function. Seismic analysis methodology of the fuel assembly is presented in WCAP-8236 (Reference 17), WCAP-9401-P-A (Reference 18), and WCAP-10444-P-A (Reference 11).

To demonstrate that the fuel assemblies will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event, a plant specific or bounding seismic analysis is performed.

The fuel assembly response resulting from safe shutdown earthquake condition is analyzed using time-history numerical techniques. The vessel motion for this type of event primarily causes lateral loads on the reactor core. Consequently, the methodology and analytical procedures as described in WCAP-8236 (Reference 17) and WCAP-9401-P-A (Reference 18) are used to assess the fuel assembly deflections and impact forces.

The motions of the reactor internals upper and lower core plates and the core barrel at the upper core plate elevation, which are simultaneously applied to simulate the reactor core input motion, are obtained from the time-history analysis of the reactor vessel and internals. The fuel assembly response, namely the displacements and impact forces, is obtained with the reactor core model. Similar dynamic analyses of the core were performed using reactor internals motions indicative of the postulated pipe rupture. Scenarios regarding breaches in the pressure boundary are investigated to determine the most limiting structural loads for the fuel assembly. The application of leak-before-break limits the size of the pipe rupture loads for which the fuel assemblies must be analyzed. The pipe rupture used in the fuel assembly analysis is the largest pipe connected to the reactor coolant system which does not satisfy the leak-before-break criteria. Subsection 3.6.3 discusses mechanistic pipe break.

4.2.3.5.1.1 Grid Analyses

The maximum grid impact force obtained from seismic analyses is less than the allowable grid strength. With respect to the guidelines of Appendix A of the Standard Review Plan, Section 4.2, Westinghouse has demonstrated that a simultaneous safe shutdown earthquake and pipe rupture event is highly unlikely. The fatigue cycles, crack initiation, and crack growth due to normal operating and seismic events will not realistically lead to a pipe rupture. More information is available in WCAP-9283 (Reference 19).

Based on the deterministic fracture mechanics evaluation of small flaws in piping components, Westinghouse has demonstrated that the dynamic affects of a large pipe rupture in the primary coolant piping system for the AP1000 design does not have to be considered.

A design basis for the piping design in the AP1000 is that the reactor coolant loop and surge lines will satisfy the leak-before-break criteria for mechanistic pipe break. In addition, the

piping connected to the reactor coolant system that is six inch nominal diameter or larger is evaluated for leak-before-break. The result of a pipe leakage event consistent with the mechanistic pipe break evaluation would be to impose insignificant asymmetric loadings on the reactor core system. Thus, fuel assembly grid loads due to large pipe ruptures are unrealistic and, as such, are not included in the analysis.

The pressure boundary integrity for numerous branch lines is analyzed to determine the most limiting break of a line not qualified for leak-before-break for the dynamic loading of the reactor core. Grid loads resulting from a combined seismic and pipe rupture event do not cause unacceptable grid deformation as to preclude a core coolable geometry.

4.2.3.5.1.2 Nongrid Analyses

The stresses induced in the various fuel assembly nongrid components are assessed based on the most limiting seismic condition. The fuel assembly axial forces resulting from the hold-down spring load together with its own weight distribution are the primary sources of the stresses in the guide thimbles and fuel assembly nozzles. The fuel rod accident induced stresses, which are generally very small, are caused by bending due to the fuel assembly deflections during a seismic event. The seismic-induced stresses are compared with the allowable stress limits for the fuel assembly major components. The component stresses, which include normal operating stresses, are below the established allowable limits. Consequently, the structural designs of the fuel assembly components are acceptable for the design basis accident conditions for the AP1000.

4.2.3.5.2 Dimensional Stability

Localized yielding and slight deformation in some fuel assembly components are allowed to occur during a Condition III or IV event. The maximum permanent deflection, or deformations, do not result in any violation of the functional requirements of the fuel assembly.

4.2.3.6 Reactivity Control Assemblies and Burnable Absorber Rods

4.2.3.6.1 Internal Pressure and Cladding Stresses during Normal, Transient, and Accident Conditions

The designs of the burnable absorber, source, and gray rods provide a sufficient cold void volume to accommodate the internal pressure increase during operation. This is not a concern for the rod cluster control assembly absorber rodlets because no significant amount of gas is released by the silver-indium-cadmium absorber material.

For the discrete burnable absorber rod, there is sufficient cold void volume to limit the internal pressure to a value, which satisfies the design criteria. For the source rods and gray rods, a void volume is provided within the rod to limit the maximum internal pressure increase at end-of-life. Figures 4.2-14 and 4.2-15 detail the primary and secondary source assemblies and Figure 4.2-11 details the gray rod cluster assembly.

During normal transient and accident conditions, the void volume limits the internal pressures to values that satisfy the criteria in subsection 4.2.1.6. These limits are established not only to prevent the peak stresses from reaching unacceptable values, but also to limit the amplitude

Comment [A36]: [4.2-30]

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Comment [A37]: [4.2-31]

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Comment [A38]: [4.2-32]

Comment [A39]: [4.2-33]

of the oscillatory stress component in consideration of the fatigue characteristics of the materials.

Rod, guide thimble, and dashpot flow analyses indicate that the flow is sufficient to prevent coolant boiling within the guide thimble. Therefore, clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures are maintained.

4.2.3.6.2 Thermal Stability of the Absorber Material, Including Changes and Thermal Expansion

Comment [A41]: [4.2-67]

The radial and axial temperature profiles within the source and absorber rods are determined by considering gap conductance, thermal expansion, neutron or gamma heating of the contained material as well as gamma heating of the clad.

The maximum temperatures of the silver-indium-cadmium RCCA or tungsten GRCA absorber materials are calculated and found to be significantly less than the material melting point and found to occur axially at only the highest flux region. The mechanical and thermal expansion properties of the silver-indium-cadmium absorber material are discussed in WCAP-9179 (Reference 4). The mechanical and thermal expansion properties of the tungsten absorber material are discussed in WCAP-16943-P-A (Reference 25).

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Comment [A42]: [4.2-34]

The wet annular burnable absorber (WABA) assemblies are used in the first core. The maximum temperature of the alumina-boron carbide burnable absorber pellet is expected to be less than 1200°F which takes place following the initial power ascent. As the operating cycle proceeds, the burnable absorber pellet temperature decreases due to a reduction in heat generation due to boron depletion and better gap conduction as the helium produced diffuses into the gap.

Comment [A43]: [4.2-35]

Comment [A44]: [4.2-36]

Deleted: In the first core, theThe wet annular burnable absorber rods (Pyrex) consist of borosilicate glass tubes contained within Type 304 stainless steel tubular cladding, which is plugged and seal welded at the ends to encapsulate the glass. The absorber material temperature does not exceed its design limit of 1220°F. Mechanical and thermal design and nuclear evaluation of the burnable absorber rods are described in WCAP-7113 (Reference 23).¶

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Sufficient diametral and end clearances have been provided in the neutron absorber, burnable absorber, and source rods to accommodate the relative thermal expansions between the enclosed material and the surrounding clad and end plug.

4.2.3.6.3 Irradiation Stability of the Absorber Material, Taking into Consideration Gas Release and Swelling

The irradiation stability of the silver-indium-cadmium absorber material is discussed in WCAP-9179 (Reference 4). Irradiation produces no deleterious effects in the absorber material. The irradiation stability of the tungsten absorber material is discussed in WCAP-16943-P-A (Reference 25).

Comment [A45]: [4.2-67]

Comment [A46]: [4.2-37]

As mentioned in subsection 4.2.3.6.1, gas release is not a concern for the rod cluster control rod material because no gas is produced by the absorber material. Sufficient diametral and end clearances are provided to accommodate any potential expansion and/or swelling of the absorber material for both RCCA and GRCA absorber rods.

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Irradiation produces no deleterious effects in the tungsten absorber material of the gray rodlets. Some minor cracking of the tungsten material may occur, but this cracking does not affect the absorber column geometric stability due to the small clearance between absorber and sleeve (Reference 25).

Comment [A48]: [4.2-39]

The alumina-boron carbide burnable absorber pellets are designed such that gross swelling or crumbling of the pellets is not predicted to occur during reactor operation. Some minor cracking of the pellets may occur, but this cracking should not affect the overall absorber and stack integrity.

4.2.3.6.4 Potential for Chemical Interaction, Including Possible Waterlogging Rupture

Comment [A49]: [4.2-67]

The structural materials selected have good resistance to irradiation damage and are compatible with the reactor environment.

Corrosion of the materials exposed to the coolant is quite low, and proper control of chloride and oxygen in the coolant minimizes potential for the occurrence of stress corrosion. The potential for the interference with rod cluster control assembly movement due to possible corrosion phenomena is very low.

Waterlogging rupture is not a failure mechanism associated with the AP1000 control rods. Furthermore, a breach of the cladding for any postulated reason does not result in serious consequences.

The silver-indium-cadmium absorber material is relatively inert and will remain inert even when subjected to high coolant velocity regions. Rapid loss of reactivity control material will not occur. Test results detailed in WCAP-9179 (Reference 4) concluded that additions of indium and cadmium to silver, in the amounts to form the silver-indium-cadmium absorber material composition, result in small corrosion rates.

In the unlikely event of GRCA rod cladding breach, loss of absorber material will not occur because the inner sleeve encapsulates the tungsten absorber (WCAP-16943-P-A, Reference 25).

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Comment [A50]: [4.2-40]

For the discrete burnable absorber, in the unlikely event that the zirconium alloy clad is breached, the boron carbide in the affected rod(s) could be leached out by the coolant water. If this occurred early, in-core instruments could detect large peaking factor changes, and corrective action would be taken, if warranted. A postulated clad breach after substantial irradiation would have no significant effect on peaking factors since the boron will have been depleted. Breaching of the zirconium alloy clad by internal hydriding is not expected due to moisture controls employed during fabrication. Rods of this design have performed very well with no failures observed.

4.2.4 Testing and Inspection Plan

4.2.4.1 Quality Assurance Program

The Quality Assurance Program Plan of the Westinghouse Commercial Nuclear Fuel Division for the AP1000 is summarized in Chapter 17.

The program provides for control over activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and material specifications identify the inspections to be performed.

4.2.4.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted.

4.2.4.2.1 Fuel System Components and Parts

The characteristics inspected depend on the component parts. The quality control program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

The material used in the AP1000 core is accepted and released by Quality Control.

4.2.4.2.2 Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards.

Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a specified sample basis throughout pellet production.

4.2.4.2.3 Rod Inspection

Fuel rod, rod cluster control rod, discrete burnable absorber rod, and source rod inspections consists of the following nondestructive examination techniques and methods, as applicable:

- Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.
- Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specifications meeting the requirements of ASTM-E-142-86 (Reference 20).
- Rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.
- Fuel rods are inspected by gamma scanning or other approved methods, as discussed in subsection 4.2.4.5, to confirm proper plenum dimensions.
- Fuel rods are inspected by gamma scanning, or other approved methods, as discussed in subsection 4.2.4.5, to confirm that no significant gaps exist between pellets.

- Fuel rods are actively and/or passively gamma scanned to verify enrichment control prior to acceptance for assembly loading.
- Traceability of rods and associated rod components is established by quality control.

4.2.4.2.4 Assemblies

Each fuel rod, control rod, burnable absorber rod and source rod assembly is inspected for compliance with drawing and/or specification requirements. Other in-core control component inspection and specification requirements are given in subsection 4.2.4.4.

4.2.4.2.5 Other Inspections

The following inspections are performed as part of the routine inspection operation:

- Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on serialized tools. Complete records are kept of calibration and conditions of tools.
- Audits are performed of inspection activities and records to confirm that prescribed methods are followed and that records are correct and properly maintained.
- Surveillance inspection, where appropriate, and audits of outside contractors are performed to confirm conformance with specified requirements.

4.2.4.2.6 Process Control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and unique barcode numbers. A quality control identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. The pellet production lines are physically separated from each other, and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by quality control. Physical barriers are used to prevent mixing of pellets of different nominal densities and enrichments in the pellet storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines; only one density and enrichment (with possible exception for top and bottom (axial blanket) zones) are loaded on a line at a time.

A serialized traceability code is placed on each fuel tube, which identifies the contract and enrichment. The end plugs are inserted and then welded (in an inert gas atmosphere) to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

Similar traceability is provided for wet annular burnable absorber, source, and control rods, as required.

4.2.4.3 Letdown Radiation Monitoring

Radiation monitoring of the reactor coolant is made by grab samples and laboratory analysis of the primary coolant. Refer to information presented in subsections 9.3.3 and 9.3.6, and Table 9.3.3-1.

4.2.4.4 In-core Control Component Testing and Inspection

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the rod cluster control assembly, prototype testing has been conducted. Manufacturing test/inspections and functional testing at the plant site are both performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to provide the proper functioning during reactor operation:

- Materials are procured to specifications to attain the desired standard of quality.

- Spider assemblies with brazed and welded vanes and fingers are proof-tested by applying a 5000-pound load to the spider body, so that approximately 310 pounds is applied to each vane. This proof load provides a bending moment at the spider body approximately equivalent to 1.4 times the load caused by the acceleration imposed by the control rod drive mechanism.

- Rods are checked for integrity by the applicable nondestructive methods described in subsection 4.2.4.2.3.

- To confirm proper fit with the fuel assembly, the rod cluster control, discrete burnable absorber, and source assemblies are installed in the fuel assembly and checked for binding in the dry condition.

The rod cluster control assemblies and gray rod cluster assemblies are also functionally tested, following core loading but prior to criticality, to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) one time at full-flow/hot conditions. In addition, any assembly that has a drop time greater than a two sigma limit from the average rod drop time is subjected to additional rod drops to confirm drop time. Thus, each assembly is sufficiently tested to confirm proper functioning and operation.

To demonstrate continuous free movement of the rod cluster control assemblies, and gray rod cluster assemblies, and to provide acceptable core power distributions during operations, partial movement checks are performed as required by the technical specifications. In

Comment [A47]: [4.2-41]

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addition, periodic drop tests of the rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements.

Comment [A52]: [4.2-42]

If a rod cluster control assembly and/or gray rod cluster assembly cannot be moved by its mechanism, and is determined to be untrippable, adjustments in the boron concentration of the coolant provide that adequate shutdown margin will be achieved following a trip. Thus, inability to move one assembly can be tolerated until the reactor can be safely taken to Mode 3.

Comment [A53]: [4.2-43]

Comment [A54]: [4.2-44]

Comment [A55]: [4.2-45]

Deleted: More than one inoperable assembly could be tolerated but would impose additional demands on the plant operator. Therefore, the number of inoperable assemblies has been limited to one.

4.2.4.5 Tests and Inspections by Others

For tests and inspections performed by others, Westinghouse reviews and approves the quality control procedures, and inspection plans to be utilized to confirm that they are equivalent to the description provided in subsections 4.2.4.1 through 4.2.4.4 and are performed properly to meet Westinghouse requirements.

4.2.4.6 Inservice Surveillance

As detailed in CENPD-404-P-A (Reference 27), significant 17x17 fuel assembly operating experience has been obtained. A surveillance program is expected to be established for the AP1000 for inspection of post-irradiated fuel assemblies. This surveillance program will establish the schedule, guidelines, and inspection criteria for conducting visual inspection of post-irradiated fuel assemblies and/or insert components. The surveillance program includes a visual examination of some discharged fuel assemblies from each refueling. This program also includes criteria for additional inspection requirements for post-irradiated fuel assemblies if unusual characteristics are noticed in the visual inspection or if plant instrumentation and subsequent laboratory analysis indicates gross failed fuel. The post-irradiated fuel surveillance program will address disposition of fuel assemblies and/or insert components receiving an unsatisfactory visual inspection. Those post-irradiated fuel assemblies receiving an unsatisfactory visual inspection are not reinserted into the core until a more detailed inspection and/or evaluation can be performed.

Comment [A56]: [4.2-46]

Deleted: WCAP-8183 (Reference 3)

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Deleted: Normally the fuel assemblies are taken to the spent fuel inspection station.

4.2.4.7 Onsite Inspection

Written procedures are used for the post-shipment inspection of the new fuel assemblies in addition to reactivity control and source components. Fuel handling procedures specify the sequence in which handling and inspection take place.

Loaded fuel containers, when received onsite, are externally inspected to confirm that labels and markings are intact and security seals are unbroken. After the containers are opened, the shock indicators attached to the suspended internals are inspected to determine whether movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures, the fuel assembly plastic wrapper is examined for evidence of damage. The polyethylene wrapper is then removed, and a visual inspection of the entire fuel assembly is performed.

Control rod, gray rod, secondary source rod and discrete burnable absorber rod assemblies are usually shipped in fuel assemblies. They are inspected either prior to removal of the fuel assembly from the container or after the fuel assemblies are placed in the new fuel storage racks. The control rod assembly is withdrawn a few inches from the fuel assembly to confirm free and unrestricted movement, and the exposed section is visually inspected for mechanical integrity, replaced in the fuel assembly, and stored with the fuel assembly. Control rod, secondary source or discrete burnable absorber assemblies may be stored separately or within fuel assemblies in the new fuel storage area.

4.2.5 Combined License Information

The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-059 (Reference 24), and the applicable changes have been incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

> Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD.

4.2.6 References

Comment [A57]: [4.2-67]

1. [*Davidson, S. L. (Ed.), "Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary) and WCAP-14204-A (Non-Proprietary), October 1994.*]*

2. Davidson, S. L. (Ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary) and WCAP-10126-NP-A (Non-Proprietary), December 1985.

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4. Beaumont, M. D., et al., "Properties of Fuel and Core Component Materials," WCAP-9179, Revision 1 (Proprietary) and WCAP-9224 (Non-Proprietary), July 1978.

5. Davidson, S. L., and Nuhfer, D. L. (Ed.), "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (Proprietary), June 1990 and WCAP-14342-A (Non-Proprietary), April 1995.

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9. Risher, D., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A (Proprietary), November 1976 and WCAP-8964-A (Non-Proprietary), August 1977.

10. Skarita, J., et al., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A, Revision 1 (Proprietary) and WCAP-10377-NP-A, Revision 2 (Non-Proprietary), October 1983.

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12. ASTM-A-580-90, Specification for Stainless and Heat-resisting Steel Wire.

13. Demario, E. E., "Hydraulic Flow Test of the 17x17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974.

14. Skaritka, J. (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary) and WCAP-8692, Revision 1 (Non-Proprietary), July 1979.

Comment [A58]: [4.2-47]

Deleted: , (revised annually)

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

15. Davidson, S. L. and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-P-A (Proprietary) and WCAP-9500-A (Nonproprietary), May 1982.

16. O'Donnell, W. J., and Langer, B. F., "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering 20, pp 1-12, 1964.

17. Gesinski, L., and Chiang, D., "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary) and WCAP-8288 (Nonproprietary), December 1973.

18. Davidson, S. L., et al., "Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A (Proprietary) and WCAP-9402-A (Nonproprietary), August 1981.

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23. Not used.

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26. Sidener, S. E., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)," WCAP-8963-P-A, Addendum 1-A, Revision 1-A, June 2006.

27. CENPD-404-P-A, Revision 0, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs," November 2001.

Comment [A54]: [4.2-48]

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Comment [A55]: [4.2-49]

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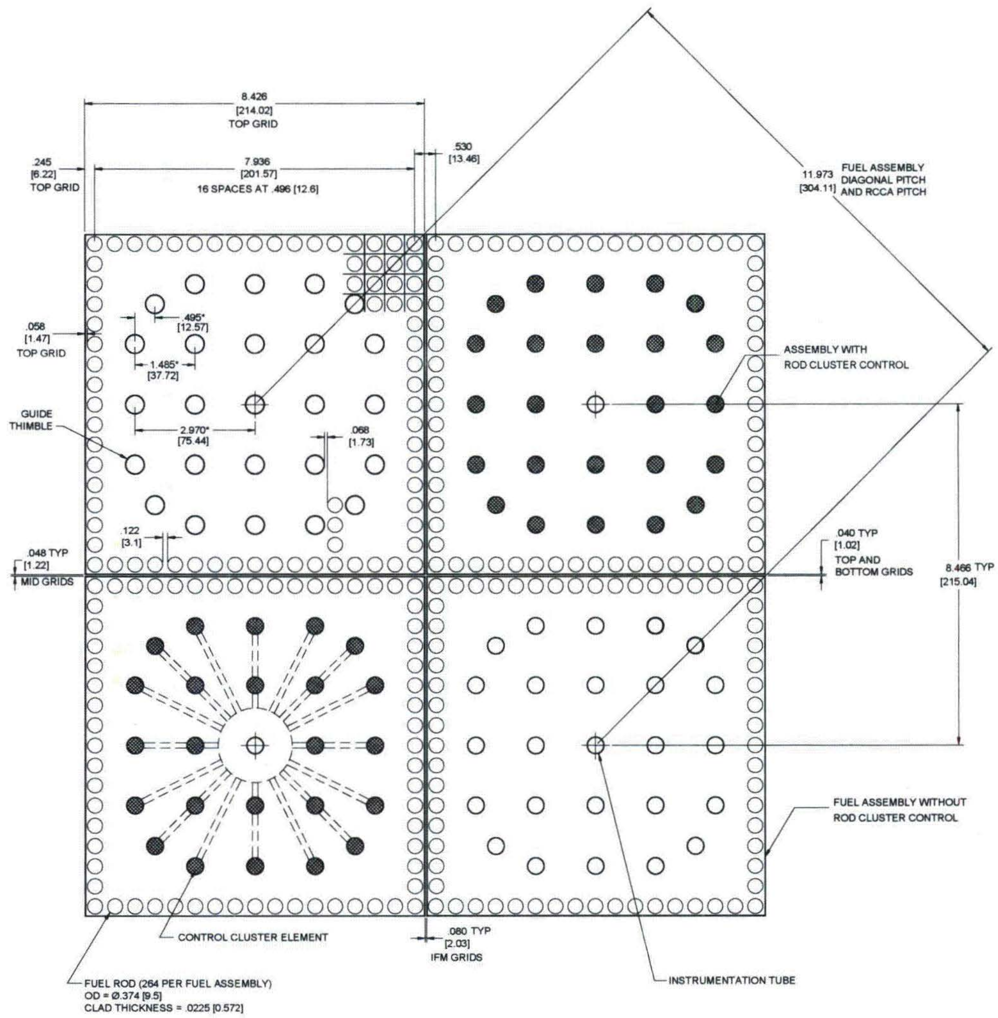
Comment [A56]: [4.2-50]

Comment [A57]: [4.2-51]



PRIMARY DIMENSIONS ARE IN INCHES (NOMINAL)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

* GUIDE THIMBLE LOCATIONS
AT TOP NOZZLE ADAPTER PLATE

Figure 4.2-1

Comment [A58]: [4.2-52]

Fuel Assembly Cross-Section

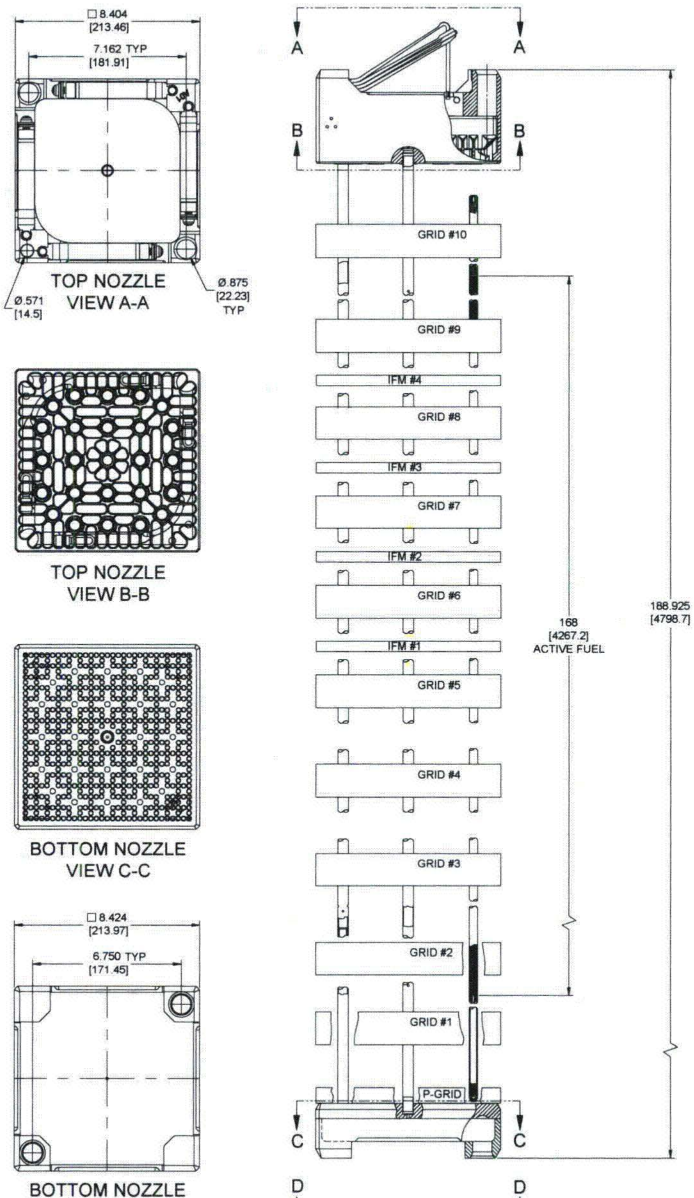


Figure 4.2-2

Comment [A59]: [4.2-53]

Fuel Assembly Outline

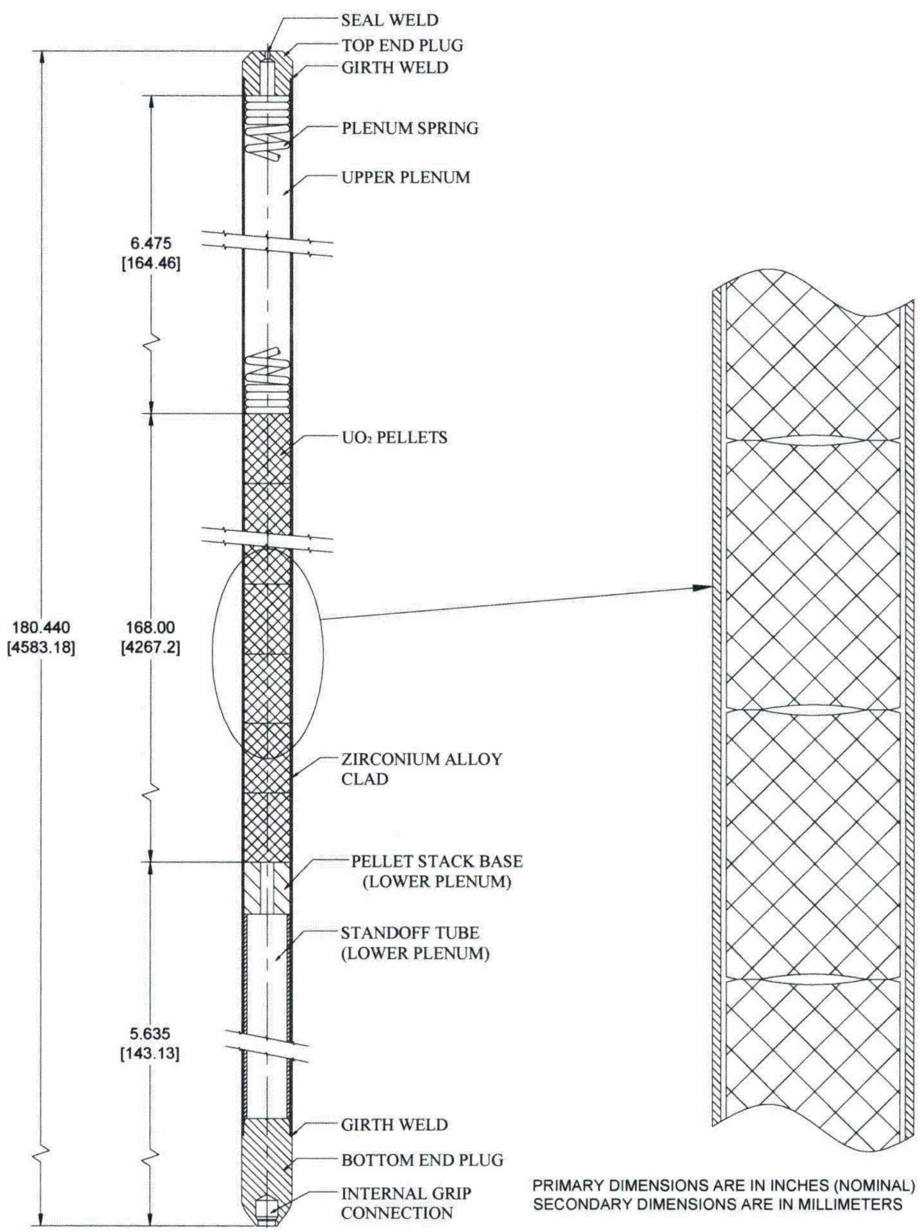


Figure 4.2-3

Comment [A60]: [4.2-54]

Fuel Rod Schematic

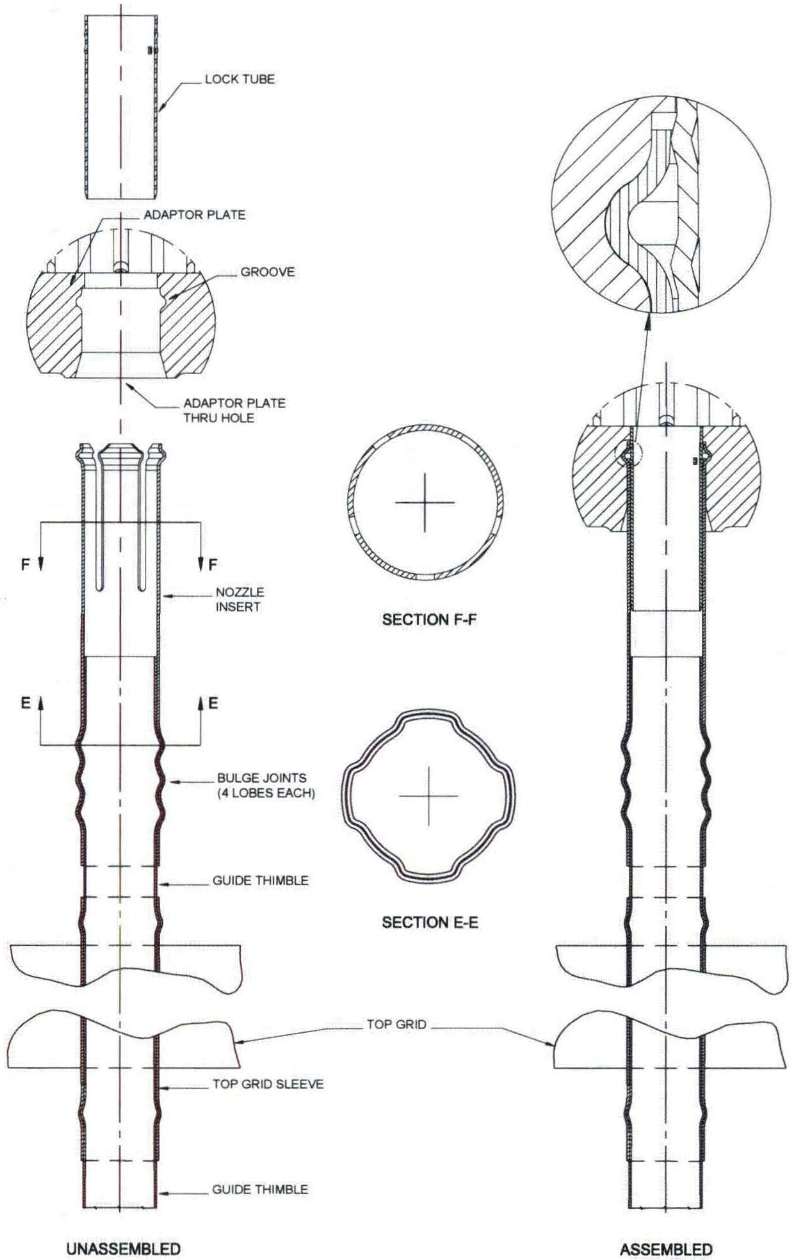


Figure 4.2-4

Top Grid Sleeve Detail

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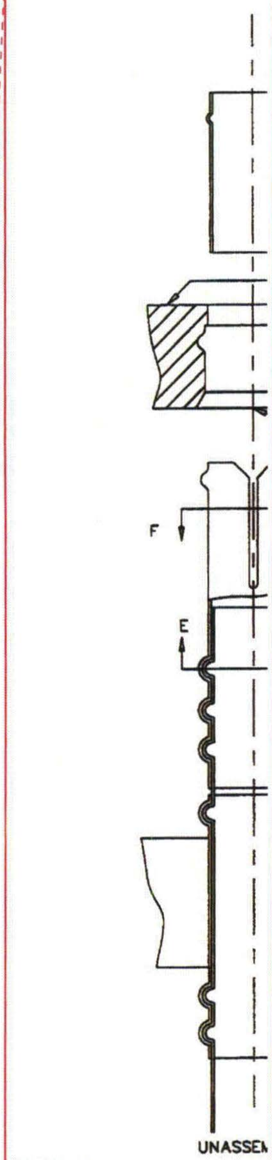

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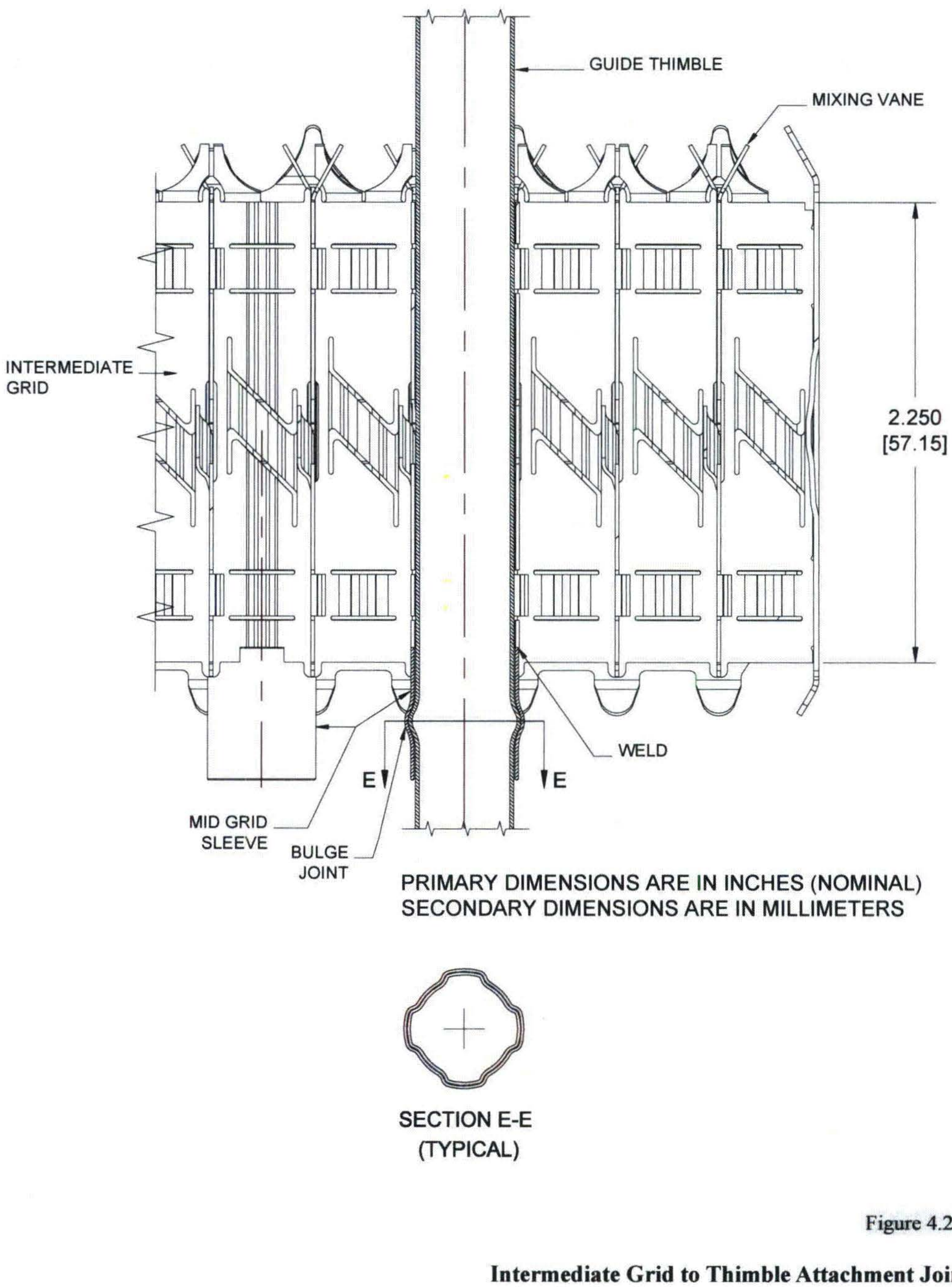


Figure 4.2-5

Comment [A62]: [4.2-56]

Intermediate Grid to Thimble Attachment Joint

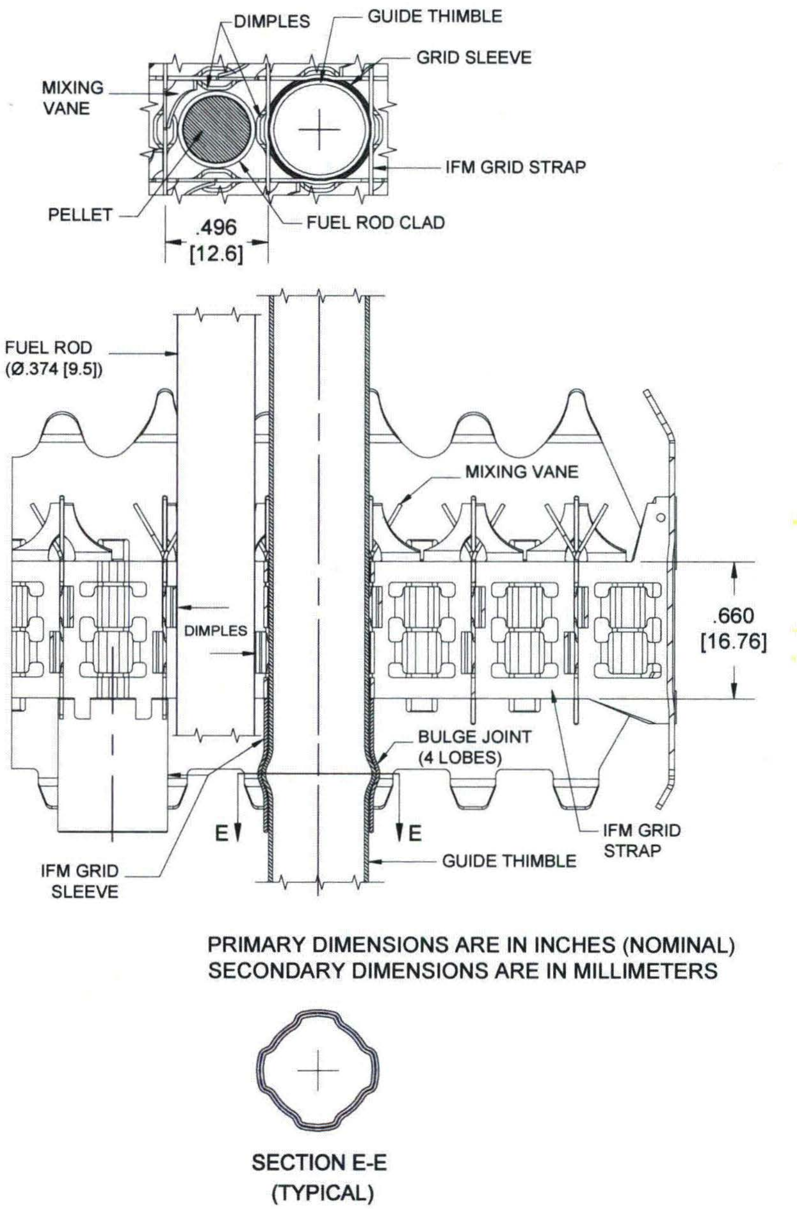
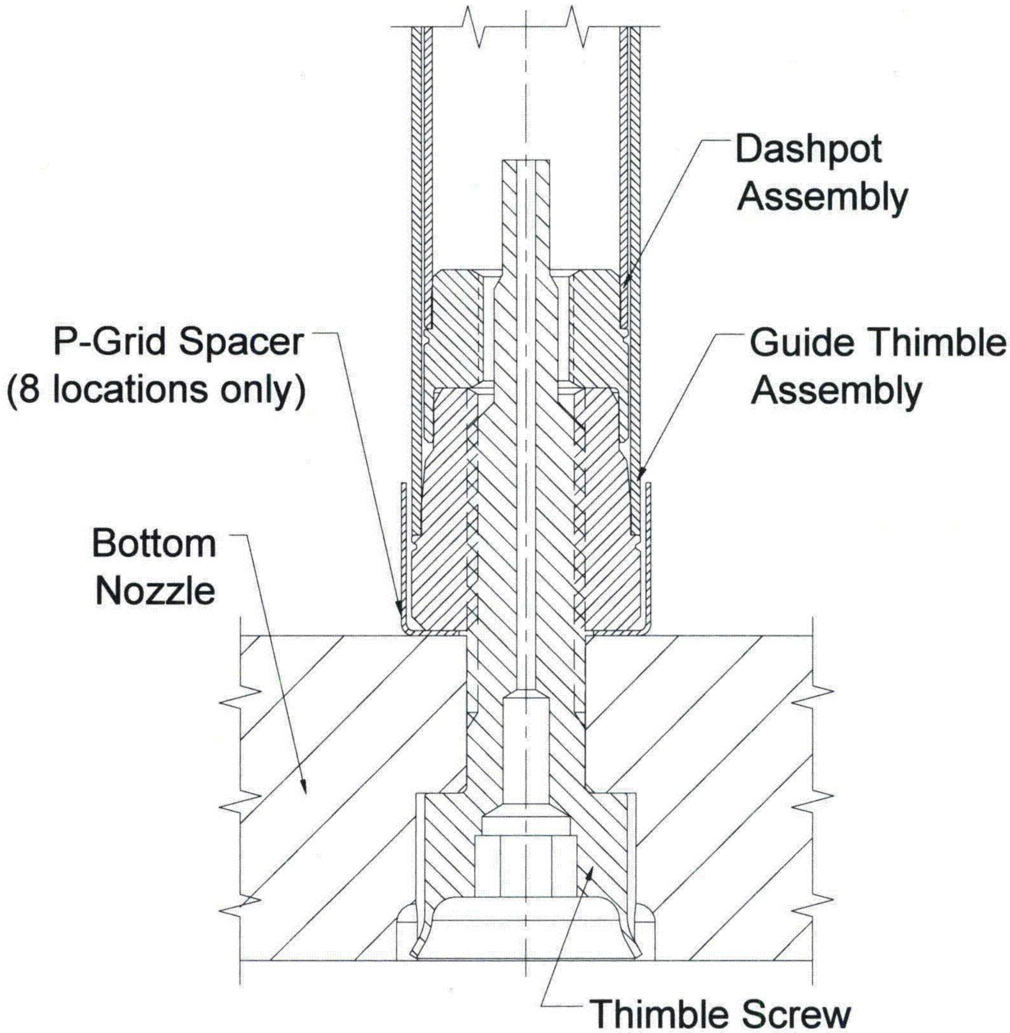


Figure 4.2-6

Comment [A63]: [4.2-57]

Intermediate Flow Mixer
Grid to Thimble Attachment



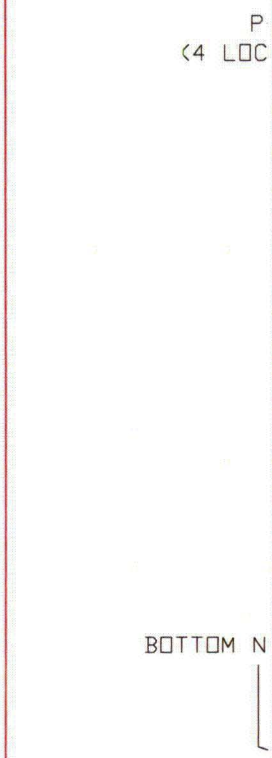


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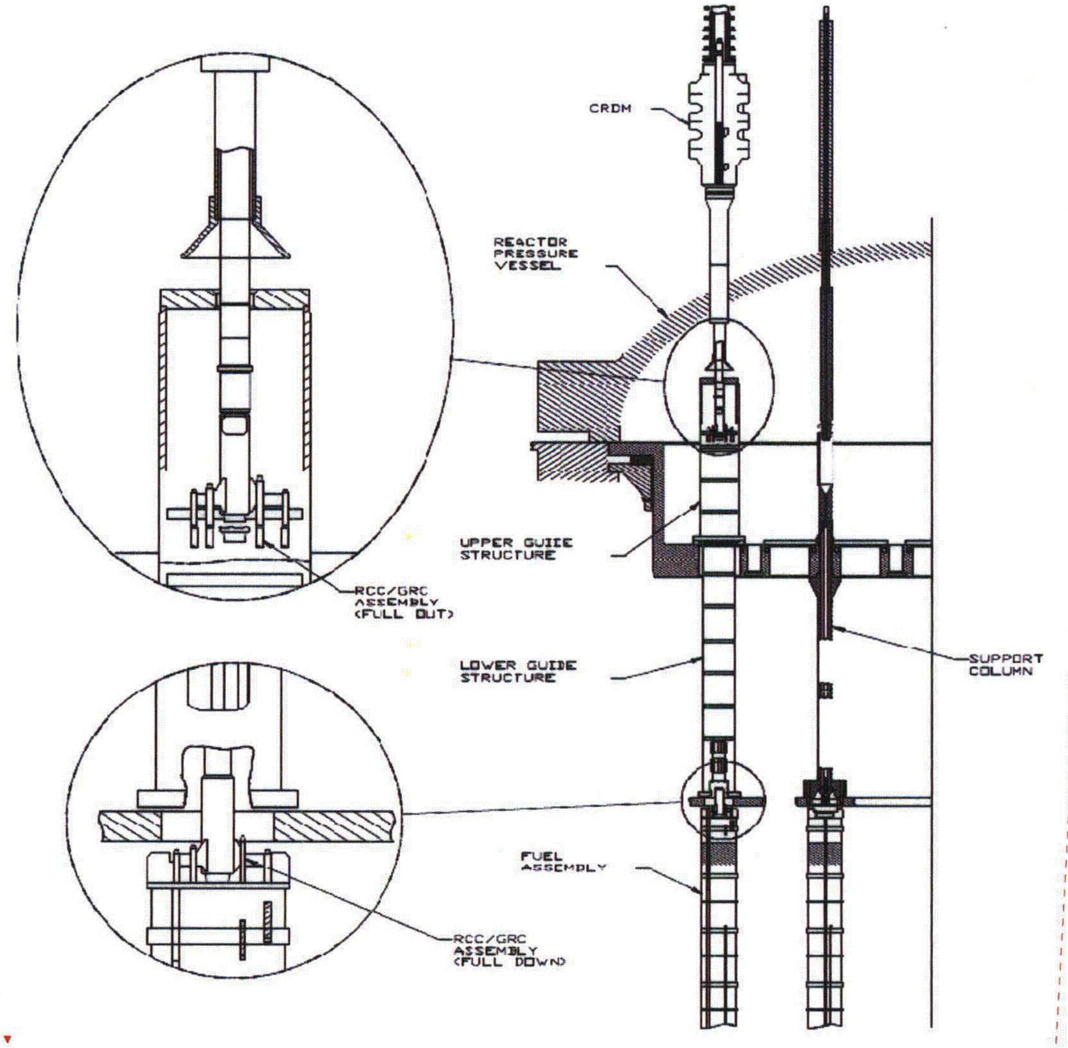
Figure 4.2-7

Comment [A64]: [4.2-58]

Grid Thimble to Bottom Nozzle Joint



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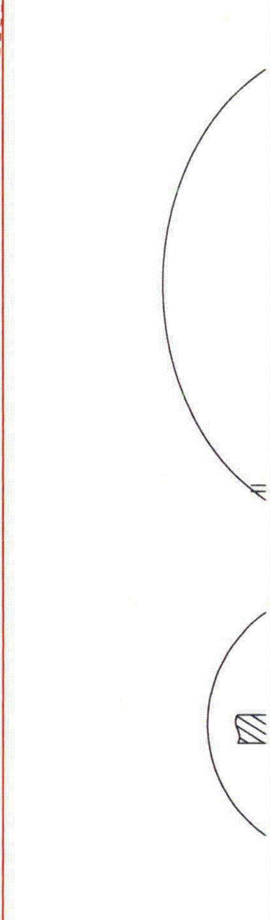
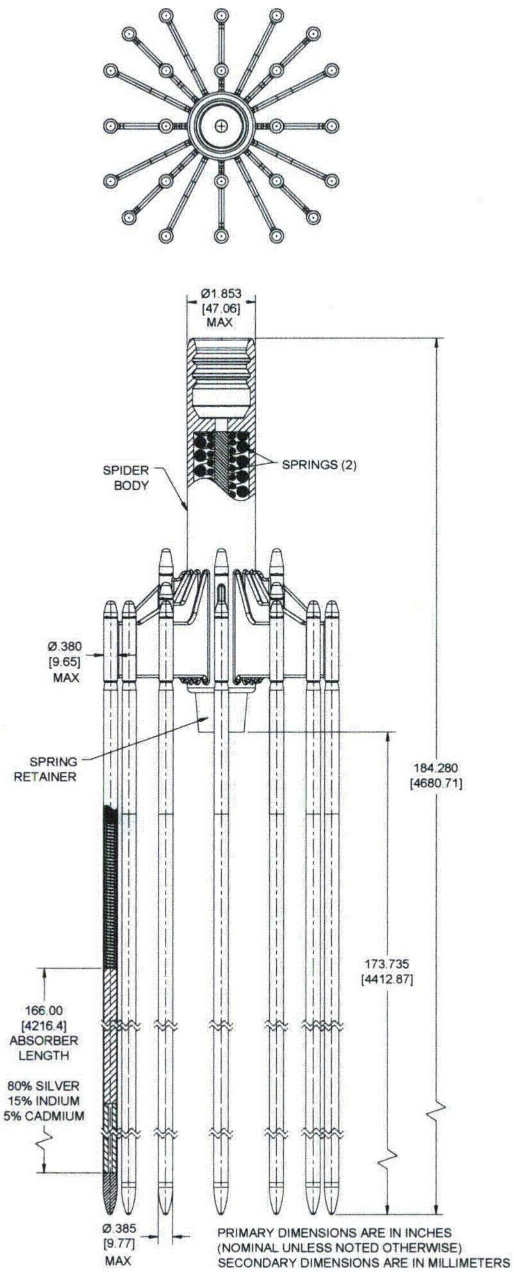


Figure 4.2-8

Rod Cluster Control and Drive Rod Assembly With Interfacing Components

Comment [A65]: [4.2-59]



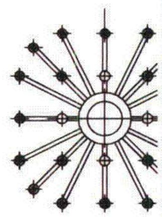


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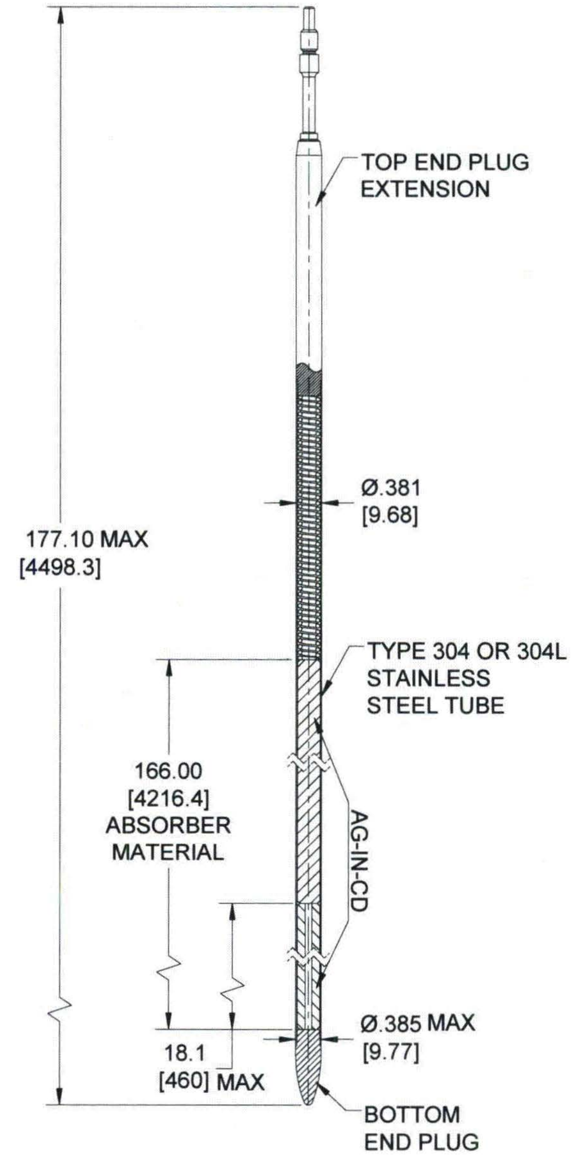
Figure 4.2-9

Comment [A66]: [4.2-60]

Rod Cluster Control Assembly



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(NOMINAL UNLESS NOTED OTHERWISE)
SECONDARY DIMENSIONS ARE IN MILLIMETERS

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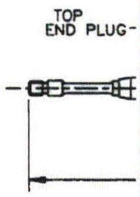

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Figure 4.2-10

Comment [A67]: [4.2-61]

Absorber Rod Detail

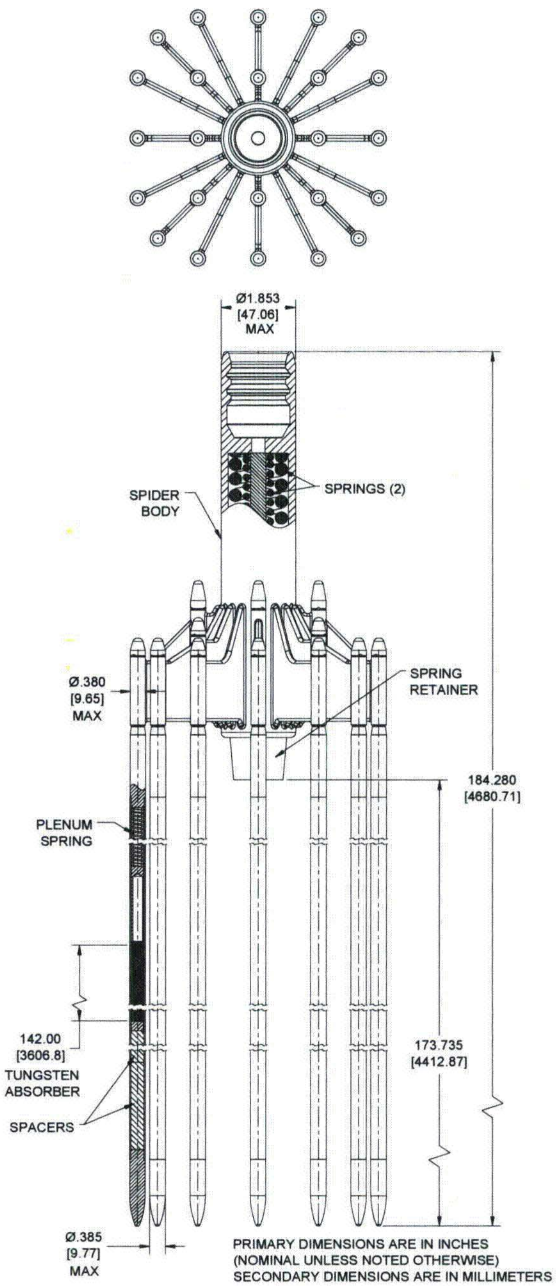
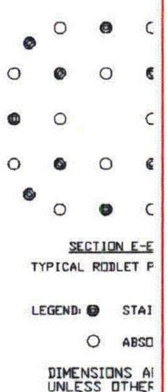


Figure 4.2-11

Gray Rod Cluster Assembly

Comment [A68]: [4.2-62]



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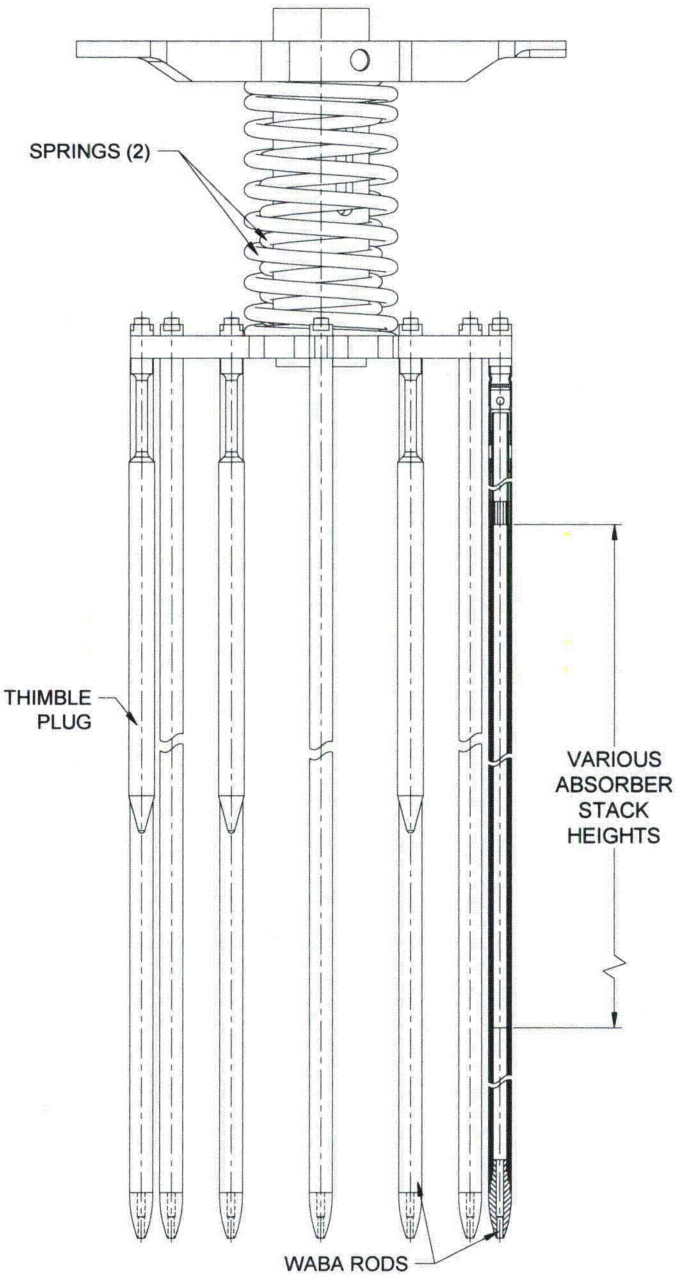

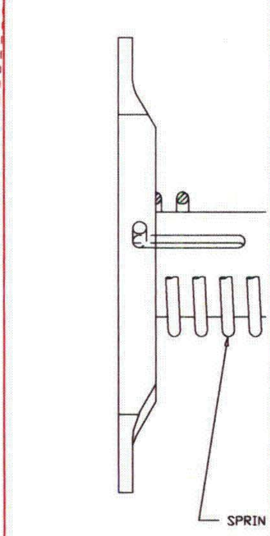

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Figure 4.2-12

Comment [A69]: [4.2-63]

Wet Annular Burnable Absorber Assembly

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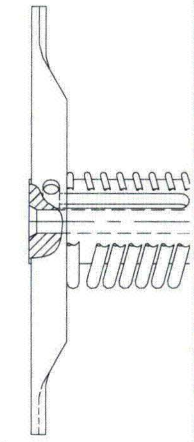

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Comment [A70]: [4.2-64]

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(Pyrex) Borosilicate Glass¶
Page Break

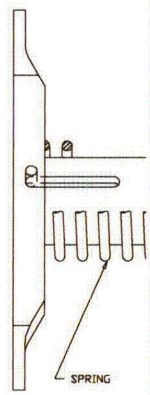


Figure 4.2-13

Not used.

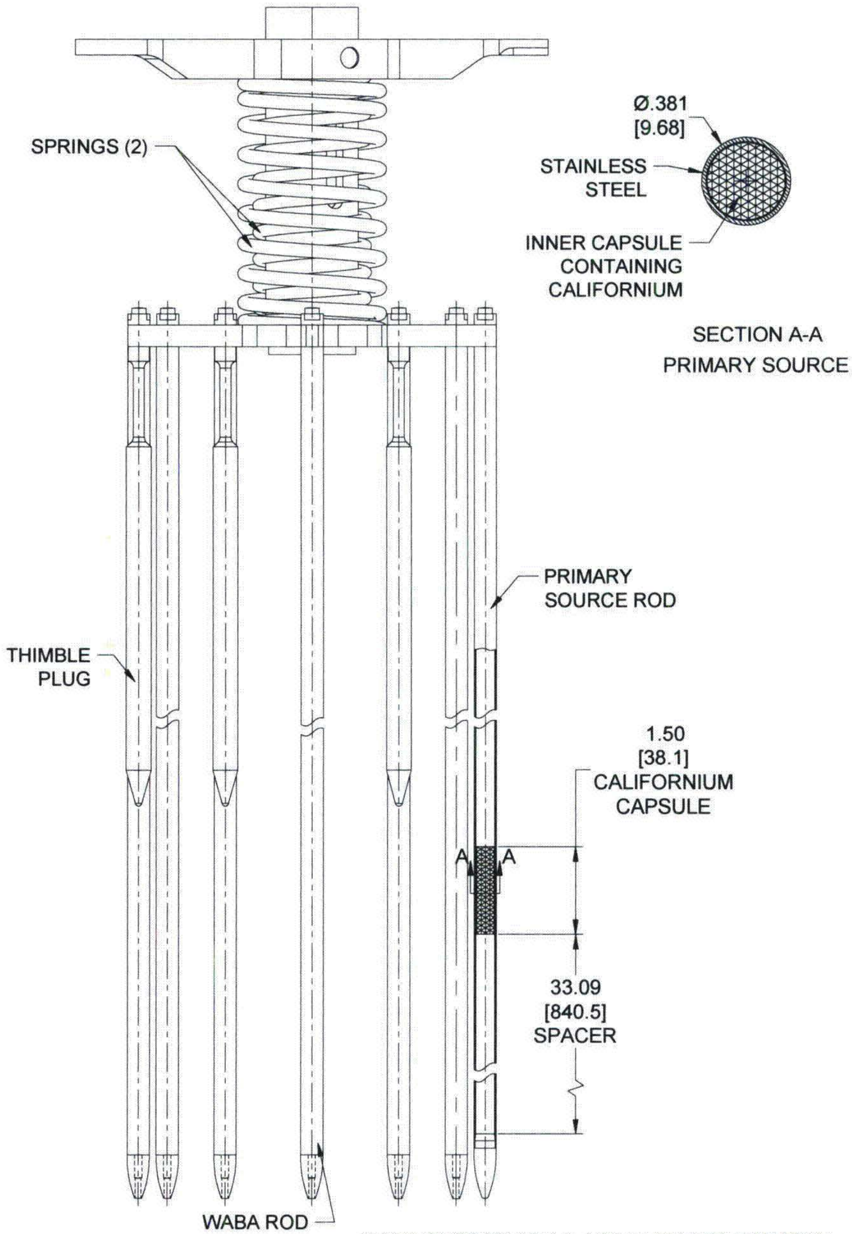
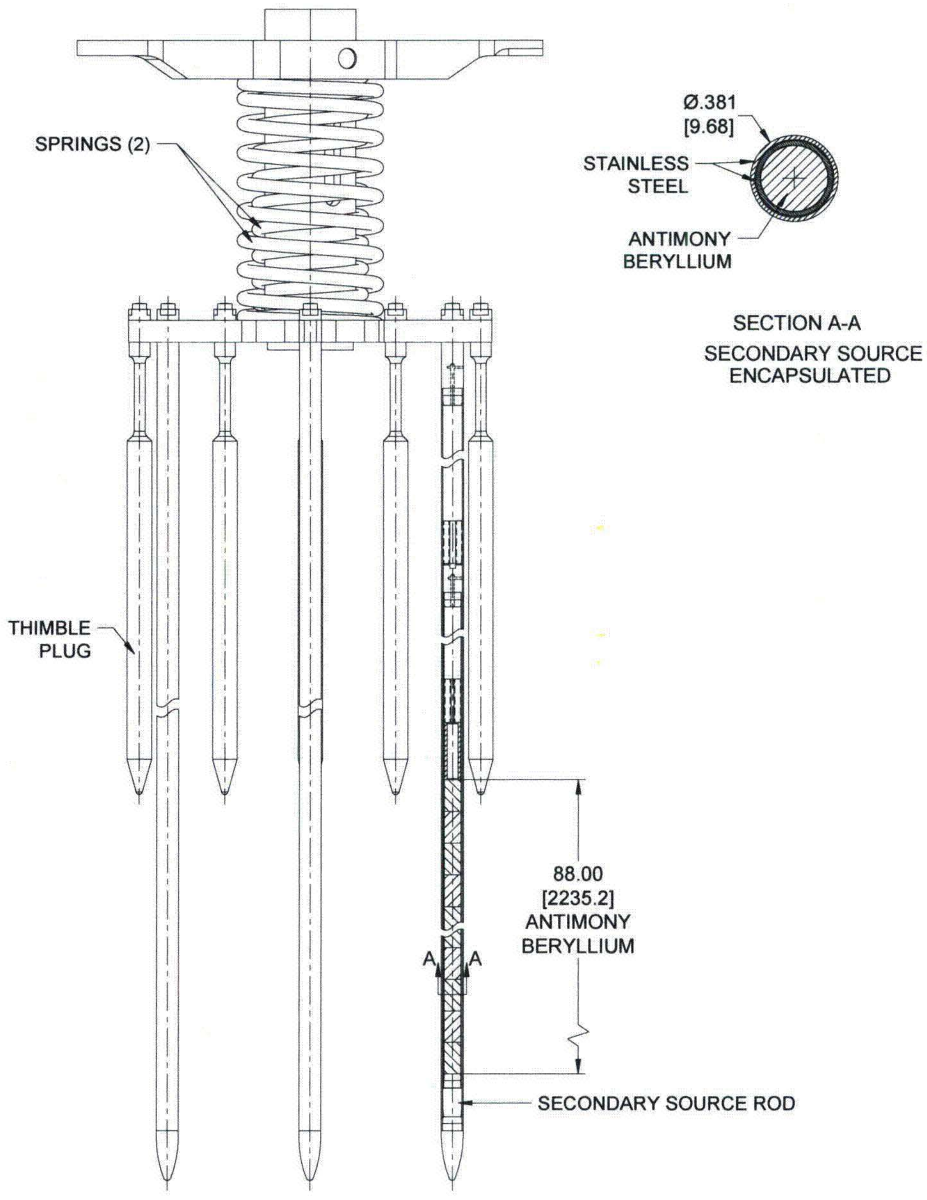


Figure 4.2-14

Comment [A71]: [4.2-65]

Primary Source Assembly



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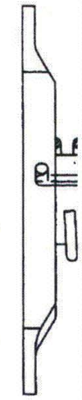

Figure 4.2-15

Secondary Source Assembly

Comment [A72]: [4.2-66]

AP1000 CORE REFERENCE REPORT
DCD (Rev. 19) Change Road Map

Change No.	Chapter 4 Section 4.3	Change Summary Description
[4.3-1]	4.3.1.7 paragraph 1	No change.
[4.3-2]	4.3.2.1 new paragraph 3	Editorial change has moved the current fourth paragraph forward to provide a better transition and introduce the description of the low leakage first core.
[4.3-3]	4.3.2.1 paragraph 4	Revised the description of the initial core loading to reflect the use of up to five average enrichments. Also deleted the earlier paragraph which was moved forward to introduce the initial core design description.
[4.3-4]	4.3.2.1 paragraph 5	Paragraph revised to replace "and/or" with "and" consistent with the final design. The first cycle core loading pattern will utilize both IFBA and WABA absorbers.
[4.3-5]	4.3.2.1 paragraph 5	Editorial change provided for consistency when referring to "the first operating cycle".
[4.3-6]	4.3.2.2.2 paragraph 1	Revised description of core figures from "one eighth" to "one quarter" consistent with revised Figures 4.3-6 through 4.3-11.
[4.3-7]	4.3.2.2.3 paragraph 1	Revised the description to include "representative hot channel" which is more accurate and consistent with the use of the term "hot channel" in the previous paragraph. The word "representative" is included because there are multiple assemblies in the core with this power distribution (due to symmetry).
[4.3-8]	4.3.2.2.4 paragraph 2	Editorial change only to delete extraneous "the" prior to "each ion chamber."
[4.3-9]	4.3.2.3.2.1 paragraph 5	Editorial change only provided to remove the extra comma from "hot, full-power" and replace it with "hot full-power".
[4.3-10]	4.3.2.3.2 (corrected to 4.3.2.3.2.3)	Editorial correction to section number provided to replace "4.3.2.3.2" with "4.3.2.3.2.3" for consistent numbering sequence.
[4.3-11]	4.3.2.3.5 paragraph 1	Editorial change only provided to replace "the" with "typical" for consistency with Table 4.3-2. Table 4.3-2 provides ranges for the reactivity coefficients to reflect the variability of these parameters over the cycle.
[4.3-12]	4.3.2.4 paragraph 2	Revised the description to clarify that Table 4.3-3 represents the result of a bounding calculation only for Cycle 1 and an equilibrium reload cycle.
[4.3-13]	4.3.2.4 new paragraph 3	New paragraph 3 is added to clarify how tungsten GRCAs are treated in bounding SDM analysis calculations.
[4.3-14]	4.3.2.4 new paragraph 4	New paragraph 4 is added to clarify both how online SDM is monitored and how tungsten GRCAs are treated in this monitoring.
[4.3-15]	4.3.2.4.1paragraph 1	The description is revised to clarify that the Doppler effect is included in the total power defect since it is no longer stated for each component in Table 4.3-3.
[4.3-16]	4.3.2.4.2 paragraph 2	The description is revised to clarify that the average moderator temperature is included in the total power defect since it is no longer stated for each component in Table 4.3-3.

[4.3-17]	4.3.2.4.3 paragraph 1	The description is revised to clarify that the three-dimensional total power defects specified in Table 4.3-3 include the effects of an adversely skewed axial xenon distribution in order to conservatively increase the redistribution effect, since it is no longer stated for each component in Table 4.3-3.
[4.3-18]	4.3.2.4.4 paragraph 1	The description is revised to clarify that this is included in the total power defect since we no longer break out each component in Table 4.3-3.
[4.3-19]	4.3.2.4.5 paragraph 1	The discussion on calculating RIA is revised to reflect the addition of tungsten GRCAs and the method used to bound the effects of allowed control rod insertion in the bounding shutdown margin calculation.
[4.3-20]	4.3.2.4.6 paragraph 1	Revised the discussion of excess reactivity to indicate that core excess reactivity is also partially controlled by control rod insertion as well as soluble boron and burnable absorbers
[4.3-21]	4.3.2.4.8 paragraph 1	Editorial change provided to correct the WCAP number from "3896-8" to "3696-8" consistent with Reference 18.
[4.3-22]	4.3.2.4.11 bullet 2	Revised description of transient xenon and samarium reactivity effects following power changes. The use of the boron system to control xenon and samarium reactivity effects is not limited to any specific power level.
[4.3-23]	4.3.2.4.12 paragraph 3	Deleted "and technical specifications"
[4.3-24]	4.3.2.4.13 paragraph 1	Replaced description of silver-indium-cadmium GRCA design with description of tungsten GRCA design.
[4.3-25]	4.3.2.4.13 paragraph 1	Description revised to clarify GRCA use in plant operations. GRCAs are used in baseload as well as load follow operations. The movement of GRCAs reduces the need for boron concentration changes, but does not completely eliminate the need for all boron changes. For example, boron changes are still needed to compensate for long term fuel depletion effects in base load operation, and during extended load follow operations occurring over multiple days.
[4.3-26]	4.3.2.4.14 paragraph 1	Description revised consistent with the use of discrete and integral fuel burnable absorbers in the final design.
[4.3-27]	4.3.2.4.16 paragraph 1	Discussion revised to state that the soluble boron concentration may also be changed during large load change maneuvers or during extended reduced power operation to maintain the control rods in the desired operating range for power distribution control.
[4.3-28]	4.3.2.4.16 paragraphs 2 and 3	Moved discussion of the target axial offset and the required negative bias from third paragraph into second paragraph where the concept is first introduced. Also added some discussion to clarify that the boron system can optionally be used (in place of a more negative target axial offset) to maintain control rods in the more desirable operating range for power distribution control. The benefits of operating with a less negative target axial offset may outweigh the negatives of making a boron change in some load change maneuvers.
[4.3-29]	4.3.2.4.17 paragraph 1	Added "control rod insertion" to list of reactivity control methods.
[4.3-30a]	4.3.2.5 paragraph 1	The shutdown bank labels are revised consistent with Figure 4.3-27and Tech Specs usage.
[4.3-30b]	4.3.2.5 paragraph 7	Discussion revised to clarify that Figures 4.3-29 and 4.3-30 are representative of typical trip reactivity curves.
[4.3-31]	4.3.2.7.4.1 second bullet	Editorial change to clarify that section 4.3.2.7.3 is being referred to.

[4.3-32]	4.3.2.7.5 paragraph 1	Editorial change provided to correct the WCAP number from "WCAP-7084-P-A" to "WCAP-7048-P-A" for consistency with Reference 35.
[4.3-33]	4.3.3 bullet 2	Revised "microscopic few group parameters" to "few group cross sections" for consistency with section 4.3.3.2 (which describes cross section generation in more detail).
[4.3-34]	4.3.3.2 paragraph 1	Description revised to include the use of PARAGON (WCAP-16045-P-A, Reference 69).
[4.3-35]	4.3.3.2 paragraph 5	Description revised to update the reference for ENDF/B-V to Reference. 71 and reflect that the PHOENIX-P code now uses 70 energy microgroups exclusively
[4.3-36]	4.3.3.2 paragraph 6	Revised "control rod" to "RCCA" for component clarification.
[4.3-37]	4.3.3.2 paragraph 10	Added new paragraph and reference information to describe cross section generation for the tungsten GRCAs.
[4.3-38]	4.3.3.2 paragraph 11	Added new paragraph to further describe the use of PARAGON as an approved replacement for PHOENIX-P for generating all cross sections used in nuclear design.
[4.3-39]	4.3.3.3 paragraph 1	Added Reference 73 to the list of approved ANC references. Reference 73 is an approved addendum to WCAP-10965-P-A which describes the revised pin power methodology.
[4.3-40]	4.3.3.3 paragraph 1	Editorial change provided to correct the WCAP number from "WCAP-7084-P-A" to "WCAP-7048-P-A" for consistency with Reference 35.
[4.3-41]	4.3.3.3 paragraph 1	Paragraph revised to clarify that the use of APOLLO is optional instead of required. Axial power distributions can be obtained from full 3-dimensional calculations as well as from APOLLO 1-dimensional calculations. Similarly, the use of a two-dimensional collapse of the 3D ANC model is optional for performing X-Y calculations and radial power distributions can also be obtained from full 3-dimensional calculations.
[4.3-42]	4.3.3.3 paragraph 4	Paragraph revised to state that one-dimensional axial calculations are optional for use in place of the full three-dimensional nodal model calculations for the determination of differential rod worths and to demonstrate load follow capability. This reflects the latest design methods, which typically perform these calculations in with full 3-dimensional models
[4.3-43]	4.3.3.3 paragraph 6	Paragraph revised to reflect the updated qualification data for PARAGON and NEXUS ANC per References 40, 69, and 72.
[4.3-44]	4.3.5 Reference 15	Reference WCAP-8498,"Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors," July 1975 is deleted and changed to "Not Used." This reference was previously cited by section 4.3.2.2.7 but has since been replaced by Reference 4. This corrects a minor error in the DCD which is not related to the advanced first core. Reference 15 should have been deleted when the first paragraph in section 4.3.2.2.7 was revised in DCD Revision 16.
[4.3-45]	4.3.5 Reference 23	Corrects a minor error in the author's name (Hoovler, instead of Hoovier)
[4.3-46]	4.3.5 Reference 49	Corrects an error in the WCAP number. The correct number is WCAP-3385-56 which is consistent with DCD Table 1.6-1 (Sheet 8 of 21).

[4.3-47]	4.3.5 Reference 53	Replaced Reference 53 (DCD R19) to APP-GW-GLR-029P, Revision 3, "AP1000 Spent Fuel Storage Racks Criticality Analysis," Westinghouse Electric Company LLC (Westinghouse Proprietary) with Edenius, M., Ekberg, K., Forssén, B. H., and Knott, D., "CASMO-4 A Fuel Assembly Burnup Program User's Manual," Studsvik/SOA-95/1, Studsvik of America, Inc. and Studsvik Core Analysis AB (Proprietary). This reference change is made because only the criticality methods are described in section 4.3.2.6.1. The results of the criticality analysis (which are documented in APP-GW-GLR-029) are described in section 9.1.6.
[4.3-48]	4.3.5 Reference 69	Reference 69 added: Ouisloumen M., et al., "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (Proprietary), August 2004.
[4.3-49]	4.3.5 Reference 70	Reference 70 added: Conner, M. E., et al. "Enhanced GRCA Rodlet Design," WCAP-16943-P-A (Proprietary) and WCAP-16943-NP-A (Non-Proprietary), September 2012.
[4.3-50]	4.3.5 Reference 71	Reference 71 added: Ford, W. E., et. al., "CSRL-V: Processed ENDF/B-V 227-Neutron-Group and Point-wise Cross Section Libraries for Criticality Safety, Reactor and Shielding Studies," NUREF/CR-2306, ORNL/CSD/TM-160 (1982).
[4.3-51]	4.3.5 Reference 72	Reference 72 added: Zhang, B., et. al., "Qualification of the NEXUS Nuclear Data Methodology," WCAP-16045-P-A Addendum 1-A (Proprietary)/WCAP-16045-NP-A Addendum 1-A (Non-Proprietary), August 2007.
[4.3-52]	4.3.5 Reference 73	Reference 73 added: Zhang, B., et. al., "Qualification of the New Pin Power Recovery Methodology," WCAP-10965-P-A Addendum 2-A (Proprietary)/WCAP-10966-A Addendum 2-A (Non-Proprietary), September 2010
[4.3-53]	Table 4.3-1 (Sheet 1 of 3)	Revised to remove "TM" from ZIRLO™ from each usage in this section and to add the registered trademark ZIRLO® and an accompanying footnote for the first usage only in this section.
[4.3-54]	Table 4.3-1 (Sheet 1 of 3)	Revised table to include the fuel assembly protective grid.
[4.3-55]	Table 4.3-1 (Sheet 1 of 3)	Revised table footnote (a) to add "and bottom" to the description of grid fabrication material.
[4.3-56]	Table 4.3-1 (Sheet 2 of 3)	Revised table to add the first core fuel enrichments for regions 1 through 5 and clarify that the enrichments are average values.
[4.3-57]	Table 4.3-1 (Sheet 2 of 3)	Revised table to update the RCCA cladding material and OD dimension.
[4.3-58]	Table 4.3-1 (Sheet 2 of 3)	Revised table to describe the enhanced GRCA absorber rodlet materials and dimensions.
[4.3-59]	Table 4.3-1 (Sheet 2 of 3)	Revised table to change the number of GRCA absorber rods per cluster from "12 Ag-In-Cd/12 304SS" to "24" tungsten.
[4.3-60]	Table 4.3-1 (Sheet 3 of 3)	Revised table to replace the description of borosilicate glass Pyrex absorbers with alumina boron-carbide WABA absorbers for the discrete burnable absorber rods used for the first core. Also revised the number of discrete burnable absorber rods used in the first core.
[4.3-61]	Table 4.3-1 (Sheet 3 of 3)	Revised the number of integral burnable absorber rods for the first core.

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[4.3-62]	Table 4.3-1 (Sheet 3 of 3)	Revised table to reflect changes to the excess reactivity for the first core.
[4.3-63]	Table 4.3-2 (Sheet 1 of 2)	Revised table to add "≤" in front of the value of F_Q to indicate that the first core will remain below the F_Q limit.
[4.3-64]	Table 4.3-2 (Sheet 1 of 2)	Revised table to reflect change to the nuclear enthalpy rise hot channel factor limit and added "≤" in front of this value to indicate that the first core will remain below the limit.
[4.3-65]	Table 4.3-2 (Sheet 1 of 2)	Revised table to reflect changes to the first cycle nuclear design parameters for reactivity coefficients.
[4.3-66]	Table 4.3-2 (Sheet 1 of 2)	Revised table to reflect changes to the typical bank worths.
[4.3-67]	Table 4.3-2 (Sheet 2 of 2)	Revised table to reflect changes to the typical hot channel factors.
[4.3-68]	Table 4.3-2 (Sheet 2 of 2)	Revised table to add "Typical" in front of "Boron Concentrations".
[4.3-69]	Table 4.3-2 (Sheet 2 of 2)	Revised table to reflect changes to the typical boron concentration values predicted for the first core.
[4.3-70]	Table 4.3-2 (Sheet 2 of 2) Note (b)	Editorial correction provided to replace use of the word "form" with "from".
[4.3-71]	Table 4.3-2 (Sheet 2 of 2) Note (g)	Added table note (g) "Rodded hot channel factors reflect full insertion of each bank at hot full power conditions".
[4.3-72]	Table 4.3-3	Revised format of table to reflect results from a 3-dimensional ANC calculation of the shutdown margin. The total power defect values include all control requirements in a single value for each core condition, and the bounding shutdown margin calculations are updated for the first core and the equilibrium reload cycle.
[4.3-73]	Figure 4.3-1	Figure has been revised to show the first core enrichment for regions 1 through 5.
[4.3-74]	Figure 4.3-3	Figure has been updated and redrawn.
[4.3-75]	Figure 4.3-4a (Sheet 1 of 4)	Figure has been updated and redrawn.
[4.3-76]	Figure 4.3-4a (Sheet 2 of 4)	Figure has been updated and redrawn.
[4.3-77]	Figure 4.3-4a (Sheet 3 of 4)	Figure has been updated and redrawn.
[4.3-78]	Figure 4.3-4a (Sheet 4 of 4)	Figure has been updated and redrawn.
[4.3-79]	Figure 4.3-4b	Figure has been added to show axial composition of fuel assemblies containing burnable absorbers
[4.3-80]	Figure 4.3-5	Figure has been updated and redrawn.
[4.3-81]	Figure 4.3-6	Figure has been updated and redrawn.
[4.3-82]	Figure 4.3-7	Figure has been updated and redrawn.
[4.3-83]	Figure 4.3-8	Figure has been updated and redrawn.
[4.3-84]	Figure 4.3-9	Figure has been updated and redrawn.
[4.3-85]	Figure 4.3-10	Figure has been updated and redrawn.
[4.3-86]	Figure 4.3-11	Figure has been updated and redrawn.

[4.3-87]	Figure 4.3-12	Figure has been updated and redrawn.
[4.3-88]	Figure 4.3-13	Figure has been updated and redrawn.
[4.3-89]	Figure 4.3-18	Figure has been updated and redrawn.
[4.3-90]	Figure 4.3-19	Figure has been updated and redrawn.
[4.3-91]	Figure 4.3-20	Figure has been updated and redrawn.
[4.3-92]	Figure 4.3-24	Figure has been updated and redrawn.
[4.3-93]	Figure 4.3-25	Figure has been updated and redrawn.
[4.3-94]	Figure 4.3-26	Figure has been updated and redrawn.
[4.3-95]	Figure 4.3-27	Figure has been updated and redrawn.
[4.3-96]	4.3.3.2 paragraph 5	Revised text to correctly reference ENDF/B-VI library as described in Westinghouse Response to CRR-002 in LTR-NRC-12-86.
[4.3-97]	4.3.3.2 paragraph 10 4.3.5	Updated the reference for WCAP-16943 to reference the approved version of this topical report.