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AP1000 Core & Fuel Design Technical Report

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1 INTRODUCTION

The purpose of this report is to provide the supplemental technical information required for closure of the Combined License Information Items contained in Chapter 4 of the AP1000 Design Control Document (DCD). Chapter 4, "Reactor," 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. This report addresses the Combined License Information Items and supplements the core and fuel design information previously reviewed and approved in sections 4.1, 4.2, 4.3, and 4.4 of the DCD. Sections 4.5 and 4.6 had no changes, but were included for completeness. The combination of information provided in the approved DCD and supplemented by this report, is intended to comprise the AP1000 Combined License (COL) licensed design and thereby provide a standard reference for the core and fuel design suitable for reference by near-term AP1000 COL applications. Alternatively, the information provided by this report may be incorporated into the DCD through an amended design certification.

This report additionally describes and defines the process that Westinghouse intends to use to manage future changes to the AP1000 COL licensed design in order to accommodate enhanced core and fuel component designs and core designs consistent with the requirements of the DCD and 10 CFR Part 52. A generic Core Reference Report will be issued to address final core and fuel design changes prior to initial core load. The Core Reference Report will provide a standard reference to be used by COL holders in the submittal of License Amendment Requests.

This report specifically addresses the Combined License Information Items contained in Chapter 4 and listed in Table 1.8-2 of the DCD which are provided as follows:

COL Information Item 4.2-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.2.5

> 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.

COL Information Item 4.3-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.3.4

> This section contains no requirement for additional information to be provided in support of the combined license. 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.

COL Information Item 4.4-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.4.7

> 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.

COL Information Item 4.4-2 (FSER Action Item 4.4-1) DCD Subsection 4.4.7

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

It should be noted that while DCD subsection 4.1.2, "Combined License Information," contains no requirement for additional information to be provided in support of a Combined License, changes to section 4.1, "Summary Description," are appropriate and are included in this report for consistency with changes and clarifications provided to sections 4.2, 4.3, and 4.4 of the approved DCD.

An NRC Safety Evaluation Report (SER) is requested to confirm the satisfactory completion of the Combined License Information provided in this report for changes to sections 4.1 through 4.4 of the DCD. Additionally, an NRC Safety Evaluation Report (SER) is requested for approval of the proposed change management process, such that the expectations are documented for implementing changes and enhancements where needed to the initial core and fuel component designs and core designs following the issuance of the expected AP1000 COL(s).

1.1 BACKGROUND

In March 2002, Westinghouse tendered its application with the NRC for the certification of the AP1000 standard plant design under 10 CFR Part 52. The NRC subsequently issued both the Final Safety Evaluation Report (FSER) and the Final Design Approval (FDA) in September 2004. These approvals were followed by the final design certification rule approved in December 2005.

As a result, the work necessary to support the initial application for certification of the AP1000, and the subsequent work to support final rule-making, required Westinghouse to identify and define a reference fuel product and initial core design.

While the AP1000 plant is a standardized, licensed design developed with the philosophy that changes to the Nuclear Steam Supply System (NSSS) should be minimized, Westinghouse and our customers clearly recognize that the fuel, core components, core designs, and design methods are regularly changed in support of reload requirements for an operating nuclear plant, and frequently undergo improvements to enhance reliability and the service lifetime. Consistent with the industry and INPO objectives to enhance fuel performance, Westinghouse continuously strives to improve the Westinghouse core and fuel products and core designs that are used in both currently operating and new reactors.

Since the time between the initial application for Design Certification and the initial startup of an AP1000 plant may be significant, accounting for as much as 10 to 15 years, the accumulated changes and improvements to the fuel, core components and core design could be substantial, and may represent a significant positive impact on the core and fuel performance. Thus, it is Westinghouse's intention to use the best fuel, core components and core design available to support initial plant startup and future plant

operations. Given these anticipated and potential future (yet to be identified) changes and enhancements, Westinghouse wishes to adopt a clearly recognized means to incorporate changes as our products, design, and methods are improved.

Consistent with this objective and with the current regulations, Westinghouse has initially established a baseline typical AP1000 COL licensed design for the fuel, core components and core design which is described by the combination of the approved DCD and this technical report. Thus, the COL "licensed design" is provided to support the COL application process. In addition, Westinghouse is formally establishing a change management process to incorporate anticipated and future fuel, core component, and core design enhancements following the submittal of the COL application. The purpose is to establish a process such that the fuel, core components and core design to be applied in the initial startup of an AP1000, provides for the optimal use of available, proven technology in order to meet the needs of the customer.

1.2 SUMMARY

In summary, the fuel, core components and core design described in the approved DCD represent the reference typical core design for the AP1000. This technical report updates and supplements the approved referenced design as provided in the DCD, and establishes a COL licensed design to be used as a basis for reference in COL applications. It is anticipated that the COL licensed design described in the COL application will continue to be enhanced as industry operating experience and advancements in technology dictate. These enhancements will be implemented in accordance with the current regulations, following the process described in this technical report. As required by the current regulations, some of these changes may be subject to prior NRC approval and thus the subject of a future License Amendment Request (LAR) to be submitted after the submittal of a COL application.

2 INITIAL CORE CHANGE CONTROL PROCESS

The AP1000 Core & Fuel Design Technical Report supplements the DCD to establish a consistent and uniform approach for the introduction of fuel into the AP1000. The NRC review and approval of this report provides for an efficient one time review process, allowing for one review to apply to several COL applications, for addressing fuel changes across the fleet of AP1000 COL holders.

2.1 AP1000 CORE & FUEL DESIGN DEVELOPMENT PROCESS

The AP1000 fuel, core components and core design is being developed in three distinct stages:

- Reference Design: defined by the NRC Certified AP1000 Design Control Document (DCD)
- Licensed Design: defined by the DCD plus the AP1000 Core & Fuel Design Technical Report (WCAP-16652-P) both of which are referenced in a COL Application
- Final Design: defined by the COL plus the Core Reference Report

Reference Design – The fuel reference design provides a reference design upon which to base the AP1000 plant certification. This design establishes the basis for the AP1000 plant requirements. However, the core and fuel design described in the approved DCD represents a reference or typical core design for the AP1000 plant that existed at the time of the initial design certification.

Licensed Design – The fuel licensed design provides the fuel, core component and core design information required to support a COL application. As a part of a COL Application, the Combined License applicants referencing the AP1000 certified design (DCD) must address changes and updates to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design from that presented in the NRC approved DCD.

The AP1000 Core & Fuel Design Technical Report addresses the COL Information items required by Chapter 4 by modifying and supplementing the information contained in the approved referenced design in order to establish the COL licensed design, which is suitable for reference in a COL application. The design described by the DCD and supplemented by the changes provided in this report, represent the core and fuel design at the time of the submittal of the COL application and address the required Combined License Information defined by the approved DCD.

While this technical report describes the differences between the DCD reference design and the COL license design needed to support the COL application, it should be recognized that the time interval between the initial core and fuel design inputs as provided for Design Certification, the COL application submittal and subsequent approval, and the initial core start-up, is potentially substantial. Therefore, additional changes and enhancements to the fuel, core components and core design are anticipated prior to startup.

Final Design – The final fuel design describes the fuel, core component and core design to be used in the startup of the AP1000, if different from the licensed design. The final design will be documented in an NRC reviewed and approved AP1000 Core Reference Report. That report will represent the COL

holder's actual initial core (cycle 1) fuel loading pattern, control rod patterns (both Rod Cluster Control Assemblies (RCCAs) and Gray Rod Control Assemblies (GRCAs)) and the associated core physics parameters at the time of initial start-up.

The objective of any subsequent changes from the proposed licensed core and fuel design described in the COL would be to improve or enhance the fuel, core components and core performance through the use of improved fuel products and fuel design methods, which are anticipated to occur as the result of industry operating experience and advancements in technology. For the first core (approved prior to initial startup) these changes will be consistent with the Principal Design Requirements described in the NRC approved DCD, section 4.1.1, and the conclusions of the core design safety criteria as described in Chapter 15 of the DCD.

AP1000 Core Reference Report – An NRC reviewed and approved AP1000 Core Reference Report, along with the DCD and the COL application information, will serve as the standard licensing basis for the final design of the fuel, core components and core design to be used in the initial startups of the AP1000 plants.

Westinghouse revises the core and fuel designs and associated methodologies over time as new information, materials and analytical techniques become available to enhance fuel performance and safety. Thus, it should be recognized that the information provided in Chapter 4 is subject to change as newer core and fuel designs are implemented to take advantage of these enhancements. The Core Reference Report will capture these improvements to be implemented in the licensed design which are anticipated as a result of information gained through operating experience and through advancements made by technology developments.

The Core Reference Report would be submitted for NRC review and approval as a Westinghouse generic technical report or topical report that would be referenced in each individual license amendment request prior to the initial startup, in order to provide sufficient time for NRC review. The AP1000 Core Reference Report, once reviewed and approved by the NRC, would address any final changes to the fuel assembly design, methods and requirements prior to initial core load.

Use of the AP1000 Core Reference Report would be approved by a License Amendment Request submitted in accordance with 10 CFR 50.90 and incorporated by COL license amendment.

2.1.1 Anticipated Core and Fuel Design Enhancements

Westinghouse continues to develop improvements in the core and fuel design in order to enhance performance. Thus changes to the core and fuel design are anticipated and are being considered that will be addressed in accordance with the proposed change control process.

Changes are anticipated to enhance the performance of the Gray Rod Control Assembly (GRCA) design by increasing the life of the gray rods and increasing the margin to the fuel performance limits. Changes are also anticipated for Enhanced Core Designs and to upgrade burnable absorber rods. Westinghouse strives to continuously develop codes, models and methods enhancements based on its growing operating experience database and advances gained through vigorous technology development. As a result, these methods may be improved and enhanced prior to use for initial startup.

2.2 REGULATORY REQUIREMENTS

Any changes to the initial core and fuel design following the submittal of a COL application will be in accordance with the requirements of 10 CFR Part 52, Appendix D, Section VIII, "Process for Changes and Departures," and 10 CFR 52.63, "Finality of Standard Design Certification." In general, Tier 1 items will be changed through rulemaking or an exemption as described in 10 CFR Part 52.63 (a)(1) and (b)(1).

- Note that no changes to Tier 1 items are included in this technical report or are anticipated.

Changes to Tier 2* items following the submittal of a COL application and prior to COL issuance will be in accordance with the requirements of 10 CFR Part 52, Appendix D, Section VIII, "Process for Changes and Departures," and the normal COL review and approval process. After the COL is issued, changes to Tier 2* items will obtain prior NRC review and approval through the standard License Amendment Request (LAR) as described in 10 CFR 50.92 in order to:

- Correct errors in the certification information
- Enhance the fuel, core components and core design
- Contribute to increased standardization of the certification information

Tier 2 items will be reviewed in accordance with 10 CFR Part 52, Appendix D, Section VIII (B)(5) (a, b, and c) to:

- Correct errors in the certification information
- Enhance the fuel, core components and core design
- Contribute to increased standardization of the certification information

and are included in the Technical Report and Core Reference Report for completeness.

The change control process, as described here for the Initial Core, is consistent with the proposed rule change to 10 CFR 52.63, "Finality of Standard Design Certification" allows changes to the certified design for the following:

(a)(1) Notwithstanding any provision in 10 CFR 50.109, while a standard design certification rule is in effect under §§ 52.55 or 52.61, the Commission may not modify, rescind, or impose new requirements on the certification information, whether on its own motion, or in response to a petition from any person, unless the Commission determines in a rulemaking that the change:

(i) Is necessary either to bring the certification information or the referencing plants into compliance with the Commission's regulations applicable and in effect at the time the certification was issued;

(ii) Is necessary to provide adequate protection of the public health and safety or the common defense and security;

(iii) Reduces unnecessary regulatory burden and maintains protection to public health and safety and the common defense and security;

(iv) Provides the detailed design information to be verified under those inspections, tests, analyses, and acceptance criteria (ITAAC) which are directed at certification information (i.e., design acceptance criteria);

(v) Corrects errors in the certification information (e.g., ITAAC, site parameters, and interface requirements); or

(vi) Contributes to increased standardization of the certification information.

Note that changes to subsequent fuel loads following an initial plant startup will follow the requirements of 10 CFR Part 52, as described and approved in the certified design. AP1000 reloads will be implemented consistent with WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" (also intended to include NRC approved revisions or addendums), and WCAP-12488-P-A (including NRC approved revisions or addendums). Changes can also be implemented in accordance with 10 CFR Part 52, Appendix D, Section VIII, "Process for Changes and Departures," and changes to the Plant License (COL) will be implemented under 10 CFR 50.92, "Issuance of Amendment."

3 RESPONSE TO COL INFORMATION ITEMS & JUSTIFICATION FOR PROPOSED CHANGES

The following information is provided to 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, and to provide the information necessary for closure of Combined License Information Items 4.2-1, 4.3-1, and 4.4-1 contained in Chapter 4 subsections 4.2.5, 4.3.4, and 4.4.7 of the DCD (FSER Action Item 4.2.8-1). This information must be addressed by COL applicants referencing the AP1000 certified design.

The following sections of this report provide the specific changes and their supporting justifications that are needed to provide an update to the reference core and fuel design descriptions contained in sections 4.1, 4.2, 4.3, and 4.4 of the approved DCD. These changes are also included in a tabulation of individual changes and in the DCD Mark-ups provided in Appendices 1 of this report.

The following changes affect only Tier 2 information and do not impact the Tier 1 information presented in the DCD. In a few instances, changes are provided to information identified by italics as Tier 2*. These instances include changes provided for consistency between similar descriptions included in multiple locations in Chapter 4.

Justification for changes which apply to multiple sections of the DCD are described in the section primarily affected.

Regulatory Basis

These changes are being implemented in accordance with proposed 10 CFR 52.63, "Finality of standard design" and 10 CFR 52 Appendix D VIII (B.5) and (B.6), "Process for Changes and Departures."

In accordance with these regulatory requirements, changes to the Tier 2* items are being submitted to the NRC for review and approval via this technical report.

Tier 2 items are included in this report to address COL information items.

3.1 CHANGES TO SECTION 4.1, "SUMMARY DESCRIPTION"

DCD subsection 4.1.2, "Combined License Information, " contains no requirements for additional information to be provided in support of a Combined License. However, an update to section 4.1 is appropriate to clarify ambiguities in the write-up and to be consistent with related changes made to sections 4.2, 4.3, and 4.4 of the DCD.

Changes to the reference design information contained in section 4.1 are as follows:

1. Clarification of the value for core average linear power listed in "Principal Design Requirements"

   Basis for Change: The value listed in DCD subsection 4.1.1, "Principal Design Requirements," "For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.71 kw/ft for the initial fuel cycle"

is provided as "5.71" kw/ft. This value should be "5.718" kw/ft because the last significant digit "8" was inadvertently truncated from the calculated value.

Justification for the Change: The Principal Design Requirements provided in subsection 4.1.1 of the DCD are not altered. However, a correction is provided to clarify a truncation error affecting the value "5.71" kw/ft as listed in "For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.71 kw/ft for the initial fuel cycle." The listed value should be "5.718" kW/ft because the last significant digit of "5.718" was inadvertently truncated from the calculated value. This value is consistent with to the AP1000 certified design for nuclear steam supply power rating which is 3400 megawatts thermal (MWt). The calculated value continues to be based on the certified AP1000 design rating of 3400 MWt.

This change also applies to the value listed for core average linear power provided in Tier 2* Table 4.3-2 which should be similarly corrected from "5.71" to "5.72" kW/ft. The value is rounded (instead of truncated) to be consistent with the calculated value of "5.718" kW/ft which is corrected in subsection 4.1.1. This change is also applied to Note (g) of Table 4.1-1 and Note (f) of Table 4.4-1 which are similarly clarified by adding that "The value for AP1000 is rounded to 5.72 kW/ft." (See Appendix 1, DCD mark-ups identified as 4.1 [7], 4.1 [11], 4.3 [17], and 4.4 [10].)

Additional Changes/Clarifications: The following changes to section 4.1 reflect related changes further described in sections 4.2, 4.3, and 4.4, which are the primary sections affected by these changes. (See Appendix 1 for DCD mark-ups).

2. Adoption of a new Gray Rod Control Assembly (GRCA) design to meet Westinghouse fuel design criteria and provide acceptable margin to the fuel performance limits. (See description and justification of new GRCA design affected changes provided in section 4.2.)

3. The current DCD description of the AP1000 Fuel combined the Bottom Grid with the Protective Grid. For accuracy and clarification, the Protective Grid should be addressed separately. (See changes to clarify the fuel assembly debris mitigation features provided in section 4.2.)

4. The core structure design includes a Neutron Panel. This panel should be explicitly recognized for accuracy and consistency with the reactor internals design. (See Appendix 1, DCD markup 4.1 [10] .)

5. The DCD describes the use of a reconstitutable integral clamp top nozzle (ICTN). (ICTN) should be changed to reflect the use of the Westinghouse Integral Nozzle (WIN). (See description and justification of changes describing the WIN top nozzle design provided in section 4.2.)

6. Changes to section 4.1, Table 4.1-1, are included for consistency with the revised definition of core flow area described in section 4.4. (See full description and justification of changes provided to reflect the revised definition of core flow in section 4.4.)

3.2 CHANGES TO SECTION 4.2, "FUEL SYSTEM DESIGN"

As provided in DCD subsection 4.2.5, "Combined License Information," COL applicants referencing the AP1000 certified design must 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.

The COL Information Item changes and clarifications addressed in this section are summarized below and included in the DCD page mark-ups provided in Appendix 1.

Changes to the reference design contained in section 4.2 are described as follows:

1. Adoption of a new Gray Rod Control Assembly (GRCA) design.

   Basis for Change: A new Gray Rod (GRCA) design is provided to meet Westinghouse fuel design criteria and provide acceptable margin to the fuel performance limits.

   Justification for the Change: The new GRCA design provides a more distributed absorber material within the Gray Rod Control Assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbations. Reductions in these fuel rod power perturbations ensures that Westinghouse fuel design criteria are met and acceptable margin to the fuel performance limits is maintained. While the expected power perturbations on fuel rods adjacent to the absorber rodlets are naturally larger than those currently observed in typically unrodded operations in current designs, this change is expected to significantly improve the magnitude of these power perturbations relative to those observed in the original gray rod designs.

   This change in the distribution of gray rods from 4 to 12 represents only a minor design refinement. Since GRCA reactivity worth is maintained, this change to the GRCA design does not impact the Principal Design Requirements or the conclusions of the Chapter 15 Accident Analysis.

   This change affects Tier 2* information contained in section 4.3, Table 4.3-1 and information also provided in Chapter 4, subsections 4.1, 4.2.1.6.1, 4.2.2.3.2, 4.3.2.4.13, Table 4.1-1, and Figure 4.2-11. (See Appendix 1 for individual DCD change mark-ups identified as No. 4.1 [4], 4.1 [5], 4.1 [6], 4.2 [7], 4.2 [20], 4.2 [38], 4.3 [10], and 4.3 [16].)

   This change also affects the Gray Rod Control Assembly description included in FSER ("Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design (NUREG-1793)," Section 4.2.4.

2. Revised references to reflect current Westinghouse NRC approved Fuel Rod Design Methodology: The following changes are provided to clarify the topical report references to the PAD fuel performance codes described in WCAP-10851-P-A (PAD 3.4) and WCAP-15063-P-A Revision 1 (PAD 4.0) which are included in Chapter 4, subsections 4.2.1.2.2, 4.2.3.1.2, 4.2.3.2, 4.2.3.3, and 4.4.2.11 of the approved DCD.

Basis for Change: These changes reflect the current licensed Westinghouse Fuel Rod Design Methodology which is PAD 4.0.

Justification for the Change: These changes are justified because PAD 4.0 is not used to establish criteria or limits, but is used to substantiate that these criteria are met. Also, any changes to the licensed criteria or licensed plant-specific limits would require prior NRC review and approval.

In addition, these changes are consistent with the information provided in FSER section 4.2.6.1, "Fuel Rod Performance Evaluation," which includes that "The applicant performed most of the analyses using the PAD fuel performance code described in WCAP-15063-P-A, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," Revision 1, issued in July 2000, and WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," issued in August 1988.

(See Appendix 1 for DCD mark-ups identified as No. 4.2 [3], 4.2 [23], 4.2 [25], 4.2 [26], 4.2 [27], 4.2 [28], and 4.4 [5].)

Additional Changes and Clarifications: The following changes to section 4.2 are provided to more accurately reflect the current AP1000 core and fuel design:

3. Section 4.2 of the approved DCD currently recognizes the use of WCAP-12488-A, "Fuel Evaluation Criteria" (see DCD section 4.2.6 Reference 1) as Tier 2*. A change is added to clarify that the citation of WCAP-12488-A in subsections 4.2 and 4.2.1 should include all revisions and updates to this Topical as approved by the NRC. (See Appendix 1, DCD mark-ups No. 4.2 [1].)

4. A change is provided to the value listed for the nominal design density of the fuel to replace "95" with "95.5" for the approximate percent of the theoretical density. This change is consistent with the Tier 2* value provided for Fuel pellets - Density included in Table 4.3-1. This change affects subsection 4.2.1.2.1, "Thermal-Physical Properties," subsection 4.2.3.2, "Fuel Materials Considerations," and Table 4.4-1 Note (j). The change provided to Table 4.4-1 Note (j) also adds "for AP1000, 95 percent for others" because both theoretical densities are used in this table. (See Appendix 1, DCD mark-ups identified as No. 4.2 [2], 4.2 [24], and 4.4 [11].)

5. A change is provided to add a reference to WCAP-12610-P-A, current Reference 5 in DCD subsection 4.2.6 for ZIRLO properties, to subsections 4.2.1.4.1 and 4.2.1.5.3 for consistency with the associated descriptions provided in Section 4.2. (See Appendix 1, DCD mark-up No. 4.2 [4].)

6. A change to subsection 4.2.1.5.2 is provided to delete "room" from "room temperature" for consistency with the design limit definition. (See Appendix 1, DCD mark-up No. 4.2 [5].)

7. Clarification to include "or thimble plugs" is added to subsections 4.1, 4.2.1.6, and 4.2.2 for consistency with the current design. (See Appendix 1, DCD markups identified as No. 4.1 [3], 4.2 [6], and 4.2 [10] .)

8. Information is added to section 4.2 (specifically subsections 4.2.1.6.2, 4.2.2.3.3, 4.2.3.6.2, 4.2.6 Reference 23, and Figure 4.2-13) to clarify the design descriptions provided for the burnable absorber rods that are incorporated in the core design. These changes address the design and the application of the Pyrex and Wet Annular Burnable Absorbers (WABA). In the first core, the burnable absorber rods (Pyrex) consist of borosilicate glass tube contained within Type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. (Also, a reference to WCAP-7113 is added on borosilicate glass burnable absorber rods.) An alternative discrete burnable absorber is WABA in which the burnable absorber material consists of boron carbide pellets contained in an alumina matrix. (See Appendix 1, DCD mark-ups identified as No. 4.2 [8], 4.2 [9], 4.2 [21], 4.2 [22], 4.2 [29], 4.2 [30], 4.2 [33], and 4.2 [39].)

9. Changes to remove "ZIRLO" from top grid material options are provided in Table 4.1-1 Note (i), subsections 4.2.2 and 4.2.2.2.4; and Tier 2* Table 4.3-1 Note (a). (See Appendix 1, DCD mark-ups identified as No. 4.1 [12], 4.2 [11], and 4.3 [15].)

10. Changes are provided to the description of the AP1000 Fuel in subsections 4.1, 4.2.2, 4.2.2.1, 4.2.2.4 and Table 4.1-1 which combine the Bottom Grid with the Protective Grid. For accuracy and clarification, the Protective Grid should be called out separately. Changes to provide clarification of the fuel assembly debris mitigation features are provided in the DCD markups. (See Appendix 1, DCD markups identified as No. 4.1 [1], 4.1 [9], 4.2 [12], 4.2 [13], and 4.2 [18].)

11. Subsection 4.2.2.1 is revised to delete the reference to WCAP-8183 which does not have IFBA data. (See Appendix 1, DCD markups 4.2 [14].)

12. A change is provided to the description of the top nozzle in subsection 4.2.2.2.1 to reflect that there is no welding on the top nozzle. (See Appendix 1, DCD markup 4.2 [15].)

13. The reference DCD describes the use of a reconstitutable Integral Clamp Top Nozzle (ICTN). The ICTN nomenclature has been changed to WIN (Westinghouse Integral Nozzle). The WIN is a proven enhancement to the ICTN and it is currently in use in the Westinghouse fleet. Both the ICTN and WIN designs eliminate the need for the top nozzle spring screws and separate clamps. There are no significant design differences between the ICTN and the WIN. For consistency with regard to the ICTN and WIN designs, discussions regarding the top nozzle spring screws and clamps are removed from the DCD. These changes address clarification of WIN top nozzle design features described in subsections 4.1 and 4.2.2.2.2. (See Appendix 1, DCD markups 4.1 [2], 4.2 [16], and 4.2 [17].)

14. Subsection 4.2.2.3.1 is revised to replace "17-4 PH" with "Type 630" for clarification of the spider retainer material. (See Appendix 1, DCD markup 4.2 [19].)

15. Subsection 4.2.6 Reference 3 of 4.2.6 is revised to clarify that this report is no longer revised annually.(See Appendix 1, DCD mark-up 4.2 [32].)

16. Changes are provide to Figure 4.2-1 and Figure 4.2-6 to replace the values that were carried-over from the AP600 descriptions with values consistent with current AP1000 dimensions. (see Appendix 1, DCD markups 4.2 [34] and 4.2 [37].)

17. Figure 4.2-2 is revised to be consistent with the interface drawings. Changes include the Top Nozzle Alignment Hole Diameter, Top Nozzle S-Hole Diameter, and Bottom Nozzle S-Hole Diameter. (see Appendix 1, DCD markups 4.2 [35].)

18. Figure 4.2-3 is revised to correct the direction of girth weld indicator arrows. (See Appendix 1, DCD markup 4.2 [36].)

19. For clarification, all of the dimensions and values shown in Figures 4.2-1 thru 4.2-15 are typical.

20. Subsection 4.2.5 is revised to add "Completed" and WCAP-15662-P (this report) is added to subsection 4.2.6 as new Reference 23. This information is added to reflect that COL Information Item 4.2-1 is complete. (See Appendix 1, DCD markup 4.2 [31].)

21. Subsection 4.2.4.7 is revised to removed "attached to the suspended internals" as an unnecessary level detail not needed for the design description.

3.3 CHANGES TO SECTION 4.3, "NUCLEAR DESIGN"

As provide in DCD subsection 4.3.4, "Combined License Information," Combined License applicants referencing the AP1000 certified design should 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.

Changes to the reference design contained in section 4.3 are described as follows:

1. Adoption of a new Gray Rod Control Assembly (GRCA) design to meet Westinghouse fuel design criteria and provide increased margin to the fuel performance limits. (See discussion and justification provided in section 4.2.)

   A related change to the descriptions of GRCAs is provided to subsection 4.3.2.2.2 of the DCD which shows the gray bank as MO. This should be correctly identified as the MA+MB banks. (Gray rods are in the MA, MB, MC and MD banks) This change also applies to corrections to titles on Figures 4.3-8 and 4.3-11. (See Appendix 1, DCD mark-up 4.3 [1].)

   Also, the current description of plant conditions does not explicitly address the gray rod operations. For completeness, a change to subsection 4.3.2.4.16 is added to include that "gray rod operations" is a Condition I event which includes the periodic exchange of gray rod banks. (See Appendix 1, DCD mark-up 4.3 [11].)

2. Clarification is provided regarding BEACON for section 4.3.2.2.4, 4.3.2.2.6, and 4.3.2.2.9. Regarding Limiting Power Distributions, there are no limits on the axial flux difference when the on-line monitoring system is in service. In the unlikely event the on-line monitoring system is out of service, limits placed on the axial flux difference are designed so that the heat flux hot channel factor F_Q and DNBR are maintained within acceptable limits. This clarification also impacts the 2nd paragraph on pg 4.3-15 the statement "Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within specified axial flux difference limits." should be deleted. This is only true when the monitoring

system is out of service. (See Appendix 1, DCD markups 4.3 [2], 4.3 [3], 4.3 [4], 4.3 [5], 4.3 [6], and 4.3 [8].)

3. Subsection 4.3.2.2.7 is revised to reflect the correct reference. (See Appendix 1, DCD mark-up 4.3 [7].)

4. Subsection 4.3.2.4.2 is revised to reflect that the allowance for deadband and measurement errors is no longer a fixed value of 4 degrees. (See Appendix 1, DCD markup 4.3 [9].)

5. Table 4.3-1, Fuel assemblies, Composition of guide thimbles, is revised to reflect that guide thimbles will not be Zircaloy-4. (See Appendix 1, DCD markup 4.3 [13].)

6. Table 4.3-1, Fuel assemblies, Diameter of guide thimbles, lower part (in.), is revised to be consistent with the other dimension values given in the table. (See Appendix 1, DCD markup 4.3 [14].)

7. Table 4.3-2, Nuclear Design Parameters (First Cycle), Reactivity coefficients, Doppler-only power coefficients, is revised to correctly refer to Figure 15.0.4-1. (See Appendix 1, DCD markup 4.3 [18].)

8. Table 4.3-2, Nuclear Design Parameters (First Cycle), Boron concentrations (ppm), Design basis refueling boron concentration, is revised consistent with the accumulator boron concentration provided in the DCD Chapter 15 Accident Analysis. (See Appendix 1, DCD markup 4.3 [19].)

9. Subsection 4.3.4 is revised to add "Completed" and WCAP-15662-P (this report) is added to subsection 4.3.5 as new Reference 62. This information is added to reflect that COL Information Item 4.3-1 is complete. (See Appendix 1, DCD markup 4.3 [12].)

3.4 CHANGES TO SECTION 4.4, "THERMAL AND HYDRAULIC DESIGN"

As provided in DCD subsection 4.4.7, "Combined License Information," Combined License applicants referencing the AP1000 certified design are to 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.

The required changes and clarifications to this section are summarized below and reflected in the DCD Mark-ups provided in Appendix 1.

1. In subsection 4.4.1.3, clarification of the Core Flow Design Basis is required to accurately describe the core cavity flow area. Specifically, a typical minimum value of 94.1 percent of the thermal flow rate is assumed to pass through the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and the core shroud, head cooling flow, and leakage to the vessel outlet nozzles are not considered effective for heat removal.

   Basis for Change: The DCD change is provided to accurately describe the "core cavity flow area" due to a change from the core baffle-former design to a welded core shroud design. The

welded core shroud design eliminates the need for bolts in the high fluence regions immediately adjacent to the reactor core which can be susceptible to cracking due to irradiation-induced degradation mechanisms.

Justification for Change: Core cooling evaluations are based on the thermal design flow rate (minimum flow) entering the reactor vessel. A typical maximum value of 5.9 percent of this value is allotted as bypass flow. This includes rod cluster control guide thimble and instrumentation tube cooling flow, leakage between the core barrel and the core shroud, head cooling flow, and leakage to the vessel outlet nozzles. The shroud core cavity flow is now considered as active flow that is effective for fuel rod cooling. The maximum bypass flow fraction of 5.9 percent assumes the use of thimble plugging devices in the rod cluster control guide thimble tubes that do not contain any other core components.

The changes are provided to reflect changes to the definition of core flow area and affect descriptions provided in Chapter 4, subsections 4.4.1.3.1, 4.4.1.3.2, Table 4.1-1, and Table 4.4-1. (See Appendix 1, DCD mark-ups identified as No. 4.1 [8], 4.4 [1], 4.4 [2], and 4.4 [9].)

Additional Changes and Clarifications to section 4.4: Clarification to the referenced design described in section 4.4 that are required to appropriately reflect the current AP1000 fuel design are provided as follow:

2. VIPRE-01 is referenced throughout section 4.4. For accuracy and completeness, VIPRE-W should also be called out as an acceptable alternative.

3. Subsection 4.4.2.2.1, DNB Technology, describes the use of the DNB correlation on AP1000. The primary DNB correlation used for the analysis of the AP1000 fuel is the WRB-2M correlation. (A correlation limit of 1.14 was approved for use with WRB-2M.) Additional correlations (WRB-2 and W-3) can be used where appropriate and within their range of applicability. Additional correlations can be applied where approved by the NRC for the application.

4. A change is provided to subsection 4.4.2.11.6 on Treatment of Peaking Factors, to replace the value of "22.5" is with "less than 22.45" for consistency with Table 4.1-1. (See Appendix 1, DCD mark-up 4.4 [6].)

5. A change is provided to subsection 4.4.2.2.5 on the Effects of Rod Bow DNBR, to replace the value of "1.5" with "about 2" percent for consistency with the basis used for 3D FAC analysis. (See Appendix 1, DCD markup 4.4 [3].)

6. Subsection 4.4.2.6.2 Hydraulics Loads is revised for consistency with the current design procedures. (See Appendix 1, DCD mark-up 4.4 [4].)

7. A change is provided to the value listed for the nominal design density of the fuel from "95" to "95.5" in Table 4.4-1 Note (j) and added clarification that "for AP1000, 95 percent for others" because both theoretical densities are used in this table. (See detailed description and justification provided in section 4.2.)

8. Subsection 4.4.7 is revised to add "Completed" and WCAP-15662-P (this report) is added to subsection 4.4.8 as new Reference 87. This information is added to reflect that COL Information Item 4.4-1 is complete. (See Appendix 1, DCD markup 4.4 [7].)

DCD subsection 4.4.7, "Combined License Information" (COL Information Item 4.4-2)

DCD subsection 4.4.7, "Combined License Information," also requires that following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in subsection 7.1.6, Combined License applicants will calculate the design limit DNBR values using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as the rod bow penalty.

This action is associated with verifying that the DNB analysis is consistent with the as-procured instrumentation uncertainties. The verification is based on design changes during procurement and construction. The timing of the verification of these uncertainties is such that that it can not be provided by an applicant for a COL. The as-built verification will be deferred to be completed prior to fuel load. The confirmation of instrument uncertainties includes activities that require procurement and installation of the instruments including evaluation of changes in sensor design and location. The confirmation of instrument uncertainties will be completed after installation of the instruments. Confirmation of instrument uncertainties after completion of the installation does not alter the methods of evaluations used to establish setpoints in the technical specifications.

Note that the design limit DNBR values were based on the plant specifications for instrumentation uncertainties. This DNBR design limit should remain valid through plant procurement. However, as a final verification this item should remain open until the instrumentation is procured and the uncertainties are confirmed. The confirmation of plant instrument uncertainties will be deferred until the as-built specifications are available.

3.5 CHANGES TO SECTION 4.5, "REACTOR MATERIALS"

Subsection 4.5.3, "Combined License Information" – This section contains no requirement for additional information to be provided in support of the Combined License application.

Future changes to the core and fuel design, as necessary, will be addressed in a separate report.

3.6 CHANGES TO SECTION 4.6, "FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS REACTOR MATERIALS"

Subsection 4.6.6, "Combined License Information" – This section contains no requirement for additional information to be provided in support of the Combined License application.

4 REGULATORY IMPACT

NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design" (FSER) Chapter 4 discusses the core and fuel design for the AP1000. Most of the design changes and clarifications included in this report are at a level of detail such that they do not impact the FSER write-up. However, the following three FSER subsections have been identified as being impacted by the changes provided in this report. These include:

- In FSER subsection 4.2.2, the descriptions of the burnable absorber assemblies are not fully consistent with the design described in this report.
- In FSER subsection 4.2.4, the description of the gray rod assemblies (GRCAs) is not consistent with the design described in this report.
- In FSER subsection 4.4.1.3, the description of the core bypass flow is not consistent with the assumptions described in this report.

These changes have been evaluated and the conclusions of the FSER are not altered as a result of the changes provided in this report.

In addition, the Principal Design Requirements for the core and fuel design identified in subsection 4.1.1 of the DCD are not altered. However, an editorial correction is provided to clarify a truncation error affecting the listed value of "5.71" kw/ft which is included in, "For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 5.71 kw/ft for the initial fuel cycle." The listed value should be "5.718" kw/ft because the last significant digit of "5.718" was inadvertently truncated from the calculated value. The calculated number continues to be based on the certified AP1000 design rating.

These changes do not alter the design of the systems that provide cooling and water flow to the fuel. There are no changes to the design functions of these systems. The instrumentation that monitors core and fuel parameters is not altered by the changes in this report. These changes do not involve a change to a procedure that adversely affects how the design functions of the systems are performed or controlled.

The methods used in the mechanical, thermal, and hydraulic evaluations of the fuel do not represent a departure from the methods described in Chapter 4 of the DCD. The changes in core bypass flow assumptions described in this report are consistent with the assumptions use in the evaluation of operating plants using a core shroud reactor internals design.

The FSER in Chapter 16 discusses the review of the technical specifications. The FSER includes information on what is included in the technical specifications related to the systems in the AP1000. The technical specification included for the systems identified are not altered by confirmation of instrument uncertainties after the COL is issued. The information in FSER Chapter 16 regarding the development of setpoints is not altered by deferring confirmation of instrument uncertainties until after the COL is issued. The conclusions regarding the technical specifications in FSER Chapter 16 are not altered by deferring confirmation of instrument uncertainties until after the COL is issued.

These changes do not alter the design of the systems controlled by the technical specifications. There are no changes to the design functions of these systems. These changes do not involve a change to a procedure that adversely affects how the design functions of the systems are performed or controlled. The methodology for establishing setpoints is not altered. These changes do not involve a test or experiment.

By their inclusion in this report, the changes to the Tier 2* information described in this report are hereby submitted for NRC review and approval in accordance with Sections VIII. B. 6. a and B. 6. b of Appendix D to 10 CFR Part 52.

The changes to the Tier 2 information described in this report do not require a license amendment per the criteria of VIII. B. 5. b. of Appendix D to 10 CFR Part 52 as considered in the following:

1. **Will the change result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the DCD?**

   No. The changes in core and fuel design do not increase the initiation or progression of corrosion in primary pressure boundary materials. The changes in core and fuel design will not increase the frequency of accidents that may result from primary pressure boundary degradation such as pipe or tube ruptures. The changes in core and fuel design do not introduce a new failure mode in components that would result in an accident previously evaluated.

2. **Will the change result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the DCD?**

   No. The changes in core and fuel design do not introduce the possibility of a change in the likelihood of a malfunction because core and fuel design are not an initiator of any malfunctions. The changes in core and fuel design will not adversely alter heat transfer or flow rates in equipment relied on to cool or transfer reactor coolant. The changes in core and fuel design do not introduce a new failure mode in equipment relied upon to prevent or mitigate design basis accidents.

3. **Will the change result in more than a minimal increase in the consequences of an accident previously evaluated in the DCD?**

   No. The changes in core and fuel design do not introduce the possibility of a change in the consequences of an accident described in Chapter 15. The changes in core and fuel design do not affect the core source terms and do not change the response of the reactor coolant system and engineered safeguard systems to postulated accident conditions.

4. **Will the change result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the DCD?**

   No. The changes in core and fuel design do not introduce the possibility of a change in the consequences of a malfunction because the changes in core and fuel design will not cause pumps, valves, and heat exchangers to malfunction and result in a larger release to the environment. The

changes in core and fuel design have no effect on systems and components used to mitigate the consequences of postulated accidents.

5. **Will the change create a possibility for an accident of a different type than any evaluated previously in the DCD?**

   No. The changes in core and fuel design do not introduce the possibility of a new accident because the changes in core and fuel design do not introduce a new failure mode in systems that provide fission product barriers and mitigate postulated accidents. The changes in core and fuel design will not change the manner in which the operator controls the plant or responds to transients or accident conditions. The changes in core and fuel design will not alter the response of the reactor coolant system or engineered safeguards systems to transient conditions.

   The changes in core and fuel design do not introduce the possibility of a new accident with respect to the fuel because the changes in core and fuel design do not introduce a new failure mode in the fuel.

6. **Will the change create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the DCD?**

   No. The changes in core and fuel design do not introduce the possibility for a malfunction of an SSC with a different result because the changes in core and fuel design do not change the operation or function of systems and components and do not introduce a new failure mode in systems and components. Clearances and dimensions in the core are not altered by the changes in core and fuel design.

7. **Will the change result in a design basis limit for a fission product barrier as described in the DCD being exceeded or altered?**

   No. The changes in core and fuel design do not result in a change that would cause a system parameter to change. The changes in core and fuel design will have no adverse effect on the stresses in the fuel. The fuel performance design evaluation models described in Chapter 15 are not changed by the changes in core and fuel design. Therefore, the changes in core and fuel design do not result in a design basis limit for a fission product barrier as described in the DCD being exceeded or altered.

8. **Will the change result in a departure from a method of evaluation described in the DCD used in establishing the design basis or in the safety analyses?**

   No. The methods used to evaluate the changes in core and fuel design do not constitute a departure from a method of evaluation described in the DCD.

The core and fuel design changes and deferral of the activities in the COL information item do not impact design features associated with mitigation of severe accidents and do not require a license amendment based on the criteria of VIII. B. 5. c of Appendix D to 10 CFR Part 52.

The core and fuel design changes and deferral of the activities in the COL information item do not alter barriers or alarms that control access to protected areas of the plant. The core and fuel design changes and deferral of the activities in the COL information item do not alter requirements for security personnel. Therefore, the core and fuel design changes and deferral of the COL information item activities do not have an adverse impact on the security assessment of the AP1000.

5 CONCLUSION

This report contributes to increased standardization of the certification information by providing a standard reference for COL applicants to use in addressing the AP1000 core and fuel design. Upon NRC review and approval of this report, COL Information Item 4.2-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.2.5, COL Information Item 4.3-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.3.4, and COL Information Item 4.4-1 (FSER Action Item 4.2.8-1) DCD Subsection 4.4.7, will be completed for the COL applications.

Also based on the acceptance of this report, resolution of COL Information Item 4.4-2 (FSER Action Item 4.4-1) DCD Subsection 4.4.7 is deferred until plant instrumentation is procured and as-built instrument uncertainties are confirmed by the COL holder. COL Information Item 4.4-2 provides the following actions:

> **Open Item** – Combined License applicants will need to confirm the calculated design limit DNBR values using the RTDP with actual (procured) plant instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty. Final verification of this item should remain open until the instrumentation is procured and the uncertainties are confirmed. The confirmation of plant instrument uncertainties will be deferred until the as-built specifications are available.

6 REFERENCES

1. AP-1000 Design Control Document Revision 15 (APP-GW-GL-700) updated December 8, 2005.

2. WCAP-12488-A Westinghouse Fuel Acceptance Criteria Evaluation Process approved October 1994.

3. Federal Register/Vol. 71, No. 18/ Friday, January 27, 2006 – AP1000 Design Certification.

4. Draft Regulatory Guide 1145 (DG-1145) – "Combined License Applications for Nuclear Power Plants (LWR Edition)."

5. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology"

APPENDIX 1
CHANGE SUMMARY

Changes affecting Tier 2 information (identified by italics in the DCD) are listed in italics in this table.

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.1 [1]	4.1	4.1-1	Add "and protective grid"	For consistency with current design
4.1 [2]	4.1	4.1-1	Replace "ICTN" with "WIN"	Provides clarification of terminology for top nozzle design
4.1 [3]	4.1	4.1-2	Add "or thimble plugs."	For consistency with current design
4.1 [4]	4.1 Table 4.1-1	4.1-2 4.1-9	Change to "12" SS & Ag-In-Cd rodlets	To reflect the corrected GRCA design
4.1 [5]	4.1	4.1-2	Add that Gray Ag-In-Cd rodlets are "of a reduced diameter as compared to the RCCA absorber"	To reflect the corrected GRCA design
4.1 [6]	4.1	4.1-4	Replace "neutron absorber rods, gray rods" with "control rods (RCCAs and GRCAs)"	To make a distinction between "control rods" which are comprised of RCCAs and GRCAs (Also see same change 4.2[7] to subsection 4.2.1.6.1)
4.1 [7]	*4.1.1*	*4.1-4*	*Replace "5.71" with "5.718"*	*To correct truncated value. (Also see related change 4.1[11] to Table 4.1-1, Note (g), change 4.3[17] to Table 4.3-2, and change 4.4[3] to Table 4.4-1, Note (f).)*
4.1 [8]	Table 4.1-1	4.1-7	Coolant Flow for AP1000; Effective flow area for heat transfer: Replace "41.5" with "41.8"; Average velocity along fuel rods: Replace "15.9" with "15.8"; Average mass velocity: Replace "2.41" with "2.40"	For consistency with definition of core flow area as provided by Reactor Internals group. (Also see related change 4.4[1] to subsection 4.4.1.3.1, 4.4[2] to subsection 4.4.1.3.2, and 4.4[8] to Table 4.4-1.
4.1 [9]	Table 4.1-1	4.1-8	Add "Protective grid" in each column	For consistency with description provided in Section 4.1
4.1 [10]	Table 4.1-1	4.1-9	Replace "None" with "Neutron Panel"	For consistency with reactor internals design
4.1 [11]	Table 4.1-1 Note (g)	4.1-9	Add "AP1000 table value rounded to 5.72 kW/ft."	For consistency with value of 5.718 kW/ft as provided in subsection 4.4.1, Principal Design Requirements, line 6 (Also see related change 4.1[7] to section 4.1, change 4.3[17] to Table 4.3-2, and change 4.4[9] to Table 4.4-1 Note (f).)
4.1 [12]	Table 4.1-1 Note (i)	4.1-9	Delete "ZIRLO™" from the top grid material options	To remove 1 of the 2 options for top grid material. The 2nd option is unchanged. (See same change 4.2[11] to subsections 4.2.2 and 4.2.2.4.)

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.2 [1]	*4.2* *4.2.1*	*4.2-2*	*Add "and its addendums" to follow reference to WCAP-12488-A*	*To clarify that citation of WCAP-12488-A includes all revisions and updates to this Topical that are approved by the NRC.*
4.2 [2]	4.2.1.2.1	4.2-4	Replace "95" with "95.5"	For consistency with value in Table 4.3 (Also applies to change 4.4[10] to Table 4.4-1 Note (j).)
4.2 [3]	4.2.1.2.2	4.2-4	Delete "WCAP-8218-P-A (Reference 6)" and replace "WCAP-13589-A (Reference 8)" with "and WCAP-15063-P-A, Revision 1 (Reference 21)."	Reference WCAP-15063-P-A, Revision 1, to PAD 4.0 is added to reflect the current licensed method.
4.2 [4]	4.2.1.4.1 4.2.1.5.3	4.2-5 4.2-7	Added a reference for ZIRLO™ grid (WCAP-12610)	For consistency with description provided in remainder of Section 4.2
4.2 [5]	4.2.1.5.2	4.2-6	Delete "room" from first "room temperature"	For consistency with design limit definition
4.2 [6]	4.2.1.6	4.2-7	Add "and the thimble plugs."	For consistency with current design
4.2 [7]	4.2.1.6.1	4.2-7	Replace "control rod assembly" with "RCCA and GRCA"	To make a distinction between "control rods" which are comprised of RCCAs and GRCAs (Also see same change 4.1[6] to Section 4.1.)
4.2 [8]	4.2.1.6.2	4.2-8	Add a paragraph on borosilicate glass burnable absorber rods (Pyrex)	For clarification of burnable absorber rod and assembly description used for first core
4.2 [9]	4.2.1.6.2	4.2-8	Add clarification of (WABA) burnable absorber rods	For clarification of burnable absorber rod and assembly description
4.2 [10]	4.2.2	4.2-9	Add "or thimble plugs."	For consistency with current design
4.2 [11]	4.2.2 4.2.2.2.4	4.2-9 4.2-14	Delete "ZIRLO™" from the top grid material options	To remove 1 of the 2 options for top grid material. The 2nd option is unchanged. (See same change 4.1[12] to Table 4.1-1, Note (i).)
4.2 [12]	4.2.2	4.2-9	Add "protective zirconium oxide coated fuel clad" to the debris mitigation features	For clarification of the debris mitigation features of the fuel assembly
4.2 [13]	4.2.2.1	4.2-10	Replace "bottom" with "protective" and add "three level debris protection package"	For clarification of the debris mitigation features of the fuel assembly

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.2 [14]	4.2.2.1	4.2-11	Delete "An evaluation and test program for the integral fuel burnable absorber design features for the boride-coated fuel pellets is summarized in Section 2.5 of WCAP-8183 (Reference 3)."	To delete reference to WCAP-8183 which does not have IFBA data
4.2 [15]	4.2.2.2.2	4.2-12	Delete "The basic components of the welded top nozzle include the adapter plate, enclosure, and top plate."	For clarification because there is no welding on the top nozzle
4.2 [16]	4.2.2.2.2	4.2-12	Delete "and associated spring screws and clamps" and delete "The spring screws are made of nickel-chromium-iron Alloy 718."	For clarification of WIN top nozzle design features
4.2 [17]	4.2.2.2.2	4.2-12	Replace "spring screws" with "retaining pins"	For clarification of WIN top nozzle design features
4.2 [18]	4.2.2.2.4	4.2-14	Add a statement on P-grid fabrication	For clarification of protective grid manufacturing process versus other Inconel grids
4.2 [19]	4.2.2.3.1	4.2-17	Replace "17-4 PH" with "Type 630"	For clarification of the spider retainer material
4.2 [20]	4.2.2.3.2	4.2-17	"12" SS & Ag-In-Cd and "smaller" diameter	To reflect the corrected GRCA design (Same as changes 4.1[4] and 4.1[5] described above.)
4.2 [21]	4.2.2.3.3	4.2-18	Add a paragraph on borosilicate glass burnable absorber rods (Pyrex)	For clarification of burnable absorber rod and assembly description used for first core
4.2 [22]	4.2.2.3.3	4.2-18	Add clarification of (WABA) burnable absorber rods	For clarification of burnable absorber rod and assembly description
4.2 [23]	4.2.3.1.2	4.2-20	Add "Revision 1" to WCAP-15063 and delete "and WCAP-10851-P-A (Reference 7)"	WCAP-10851 is the PAD 3.4 model and AP1000 used PAD 4.0 WCAP-15063, Revision 1. Revised to eliminate confusion over what PAD model was used
4.2 [24]	4.2.3.2	4.2-22	Replace "95" with "95.5"	For consistency with Table 4.3. (Also see change 4.2[2] to Subsection 4.2.1.2.1 and change 4.4[10] to Table 4.4-1 Note (j).)
4.2 [25]	4.2.3.2	4.2-22	Replace "WCAP-10851-P-A" with "WCAP-15063-P-A, Revision 1" and reference "7" with "21"	To change reference form PAD 3.4 models to licensed PAD 4.0 models (WCAP-15063-P-A)

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.2 [26]	4.2.3.3	4.2-22	Delete "and WCAP-10851-P-A (Reference 7)"	WCAP-10851 is the PAD 3.4 model and AP1000 used PAD 4.0 WCAP-15063. Revised to eliminate confusion over what PAD model was used
4.2 [27]	4.2.3.3	4.2-23	Delete "and WCAP-10851-P-A (Reference 7)"	WCAP-10851 is the PAD 3.4 model and AP1000 used PAD 4.0 WCAP-15063. Revised to eliminate confusion over what PAD model was used
4.2 [28]	4.2.3.3	4.2-23	Delete "and WCAP-10851-P-A (Reference 7)"	WCAP-10851 is the PAD 3.4 model and AP1000 used PAD 4.0 WCAP-15063. Revised to eliminate confusion over what PAD model was used
4.2 [29]	4.2.3.6.2	4.2-29	Add a paragraph on Pyrex borosilicate glass burnable absorber rods	For clarification of burnable absorber rod and assembly description used for first core
4.2 [30]	4.2.3.6.2	4.2-29	Clarification on WABA burnable absorber rods	For clarification of burnable absorber rod and assembly description
4.2 [31]	4.2.5 4.2.6 Ref. 23	4.2-35 4.2-37	Add "Completed" and reference to WCAP-15662-P	To reflect completion of Combined License Information item to address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design to that presented in the DCD.
4.2 [32]	4.2.6 Ref. 3	4.2-35	Add "Revision 23, January 1996" to WCAP-8183 and delete "(revised annually)"	To clarify that this report is no longer revised
4.2 [33]	4.2.6 Ref. 23	4.2-37	Add reference WCAP-7113 on borosilicate glass burnable absorber rods	To add missing reference on borosilicate glass burnable absorber rods
4.2 [34]	Fig. 4.2-1	4.2-38	Correct values in Figure 4.2-1 that are carry-over values from AP600	For consistency with AP1000 dimensions
4.2 [35]	Fig. 4.2-2	4.2-39	Replace Figure 4.2-2	Figure revised to be consistent with the interface drawings. Changes include the Top Nozzle Alignment Hole Diameter, Top Nozzle S-Hole Diameter, and Bottom Nozzle S-Hole Diameter.
4.2 [36]	Fig. 4.2-3	4.2-41	Replace Figure 4.2-3	To correct girth weld indicator arrows
4.2 [37]	Fig. 4.2-6	4.2-44	Remove incorrect fuel rod diameter on Figure 4.2-6	To remove of AP600 carry-over dimension
4.2 [38]	Fig. 4.2-11	4.2-49	Replace Figure 4.2-11	For clarification of GRCA assembly layout
4.2 [39]	Fig. 4.2-13	4.2-51	Add Figure 4.2-13 to replace "not used"	To add Burnable Absorber Rod Assembly (Borosilicate Glass) for first core

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.2 [40]	4.2.4.7	4.2-35	Delete "attached to the suspended internals"	To remove unnecessary detail
4.3 [1]	4.3.2.2.2 Fig. 4.3-8 Fig. 4.3-11	4.3-10 4.3-65 4.3-68	Replace "M0" with "MA + MB" (2 places) and in Figure titles.	To correct error in bank designation
4.3 [2]	4.3.2.2.4	4.3-11	Replace "on demand" with "continuously"	"continuously" is more technically correct for BEACON
4.3 [3]	4.3.2.2.6	4.3-13	Replace "of" with "for short term"	For clarification that only short term operation is permitted if BEACON is out of service
4.3 [4]	4.3.2.2.6	4.3-13	Add new sentence "Limits are placed on the axial flux difference so that the heat flux hot channel factor F_Q is maintained within acceptable limits."	For clarification that AFD Limits are only applicable when BEACON is out of service
4.3 [5]	4.3.2.2.6	4.3-14	Move "Limits placed on the axial flux difference are designated so that the heat flux hot channel factor F_Q is maintained within acceptable limits." Also, delete "or reportable occurances."	To relocate this description to previous page. "or reportable occurances." is understood and not needed.
4.3 [6]	4.3.2.2.6	4.3-15	Delete "Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within specified axial flux differences limits. These procedures are detailed in the technical specifications for the case of the online monitoring system not being available, and are followed by relying only upon ex-core surveillance supplemented by the normal monthly full core map requirement and by computer-based alarms on deviation from the allowed flux difference band."	To revise the description to delete these sentences as no longer appropriate to the discussion.

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.3 [7]	4.3.2.2.7	4.3-17	Replace "WCAP-8498" with WCAP-12472-P-A" and "Reference 15" with "Reference 4"	To provide correct reference
4.3 [8]	4.3.2.2.9	4.3-18	Delete "in conjunction with the online core monitoring system" and add "when the online monitoring system is out of service."	AFD Limits are only applicable when BEACON is out of service.
4.3 [9]	4.3.2.4.2	4.3-23	Delete "by 4°F"	Allowance for deadband and measurement errors is no longer a fixed value of 4 degrees.
4.3 [10]	4.3.2.4.13	4.3-26	Replace "20" with "12" and delete "four." Add "reduced diameter"	To reflect the corrected GRCA design
4.3 [11]	4.3.2.4.16	4.3-27	Add that GRCA operations is a Condition I event which includes the periodic exchange of gray rod banks.	To address the gray rod operations.
4.3 [12]	4.3.4 4.3.5 Ref. 62	4.3-39 4.3-43	Add "Completed" and reference to WCAP-15662-P	To reflect completion of Combined License Information item to address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design to that presented in the DCD.
4.3 [13]	*Table 4.3-1*	*4.3-44*	*Delete "Zircaloy-4"*	*Guide Thimbles will not be Zircaloy-4*
4.3 [14]	*Table 4.3-1*	*4.3-44*	*Replace "0.439" with "0.482"*	*For consistency with other values in Table 4.3-1*
4.3 [15]	*Table 4.3-1 Note (a)*	*4.3-44*	*Delete "ZIRLO™" from the top grid material options*	*To remove 1 of the 2 options for top grid material. The 2nd option is unchanged.* *(Same as changes 4.1[12] and 4.2[11] described above.)*
4.3 [16]	*Table 4.3-1*	*4.3-45*	*Gray Rod Control Assemblies: Change diameter value and number of absorber rods per cluster*	*To reflect the corrected GRCA design* *(Same as changes 4.1[4], 4.1[5], 4.2[20], 4.2[38], and 4.3[10] described above.)*
4.3 [17]	*Table 4.3-2*	*4.3-47*	*Replace "5.71" with "5.72"*	*For consistency with value of 5.718 kW/ft as provided in Subsection 4.4.1, Principal Design Requirements, line 6 (Also see related change 4.1[7] to Section 4.1, change 4.1[11] to Table 4.1-1 Note (g), and change 4.4[10] to Table 4.4-1 Note (f).)*
4.3 [18]	*Table 4.3-2*	*4.3-47*	*Replace "15.1-5" with "15.0.4-1"*	*To provide correct referral to figure*

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.3 [19]	*Table 4.3-2*	*4.3-48*	*Replace "2500" with "2700"*	*For consistency with the accumulator boron concentration value provided in the DCD Chapter 15 Accident Analysis*
4.4 [1]	4.4.1.3.1	4.4-3	Design Basis, paragraph 1, second sentence: Add "and instrumentation" and replace "from the core barrel-shroud region into the core is" with "between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles are."	To address changes associated with the revised definition of core flow area
4.4 [2]	4.4.1.3.2	4.4-3	Discussion, paragraph 1, second sentence: Add "and instrumentation tubing," add "leakage between the core barrel and the core shroud," and delete "shroud cavity bypass flow." Add new third sentence, "The shroud core cavity flow is now considered as active flow that is effective for fuel rod cooling."	To addresses changes associated with the revised definition of core flow area.
4.4 [3]	4.4.2.2.5	4.4-9	Effects of Rod Bow on DNBR, paragraph 3, first sentence: Replace "1.5" with "about 2" percent.	For consistency with basis used for 3D FAC analysis
4.4 [4]	4.4.2.6.2	4.4-11	Hydraulic Loads, paragraph 2, first and second sentences: Replace "mechanical design" with "best estimate"	For consistency with current design procedures
4.4 [5]	4.4.2.11	4.4-16	Fuel and Cladding Temperatures, paragraph 2, fourth sentence: Add "and WCAP-15063-P-A, Revision 1 (Reference 85)."	To add PAD 4.0 Reference which includes the latest licensed method
4.4 [6]	4.4.2.11.6	4.4-18	Treatment of Peaking Factors, paragraph 2, first sentence: Replace "22.5" with "less than 22.45." Third sentence: delete "greater than."	For consistency with Table 4.1-1

DCD (Rev. 15) Change Summary for Chapter 4 – Sections 4.1, 4.2, 4.3 and 4.4 (Cont'd)				
ID	**Sec**	**Page**	**Update**	**Rationale for DCD Update**
4.4 [7]	4.4.7 4.4.8 Ref. 87	4.4-31 4.4-37	Add "Completed" and reference to WCAP-15662-P	To reflect completion of Combined License Information item to address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design to that presented in the DCD.
4.4 [8]	4.4.7	4.4-31	Add "and prior to fuel load, the Combined License holder will calculate the design limit DNBR values. The calculation will be completed using the ..."	To reflect deferral of the Combined License Information item for plant instrument uncertainties until the instrumentation is procured and the uncertainties are confirmed.
4.4 [9]	Table 4.4-1	4.4-38	Coolant conditions for AP1000; Effective flow area for heat transfer: Replace "41.5" with "41.8"; Average velocity along fuel rods: Replace "15.9" with "15.8"; Average mass velocity: Replace "2.41" with "2.40"	For consistency with the revised definition of core flow area
4.4 [10]	Table 4.4-1 Note (f)	4.4-39	Replace "157 fuel assemblies and hot densified fuel length" with "densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft."	For consistency with value of 5.718 kW/ft as provided in Subsection 4.4.1, Principal Design Requirements, line 6 (Also see related change 4.1[7] to Section 4.1, change 4.1[11] to Table 4.1-1 Note (g), and change 4.3[17] to Table 4.3-2.)
4.4 [11]	Table 4.4-1 Note (j)	4.4-39	Replace "95" with "95.5" and add "for AP1000, 95 percent for others"	Description added because both theoretical densities are used in this table. (Also see change 4.2[2] to Subsection 4.2.1.2.1 and change 4.2[22] to Subsection 4.2.3.2)

APPENDIX 1
DCD MARK-UPS FOR
CHAPTER 4 – SECTION 4.1

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.

Comment: No. 4.1-[1]

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 with no intermediate flow mixing grids. The AP1000 fuel assemblies are the same as the 17x17 XL Robust fuel assemblies except that they have four intermediate flow mixing grids in the top mixing vane grid spans.

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

Comment: No. 4.1-[2]

Deleted: integral clamp top nozzle (ICTN)

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™ (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.

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. Another type uses fuel pellets containing gadolinium oxide mixed with uranium oxide. The boride-coated fuel pellets and gadolinium oxide/uranium oxide fuel pellets provide a burnable absorber integral to the fuel.

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, non-integral discrete burnable absorber (BA) assemblies, or thimble plugs.

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For the initial core design, discrete burnable absorbers (BAs) and integral fuel burnable absorbers are used. Discrete burnable absorber designs, integral fuel burnable absorber designs (including both IFBAs and gadolinium oxide/uranium oxide BAs) or combinations may be used in subsequent reloads.

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. 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 rods 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. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that 12 of the 24 rodlets are fabricated of stainless steel, while the remaining 12 are silver-indium-cadmium (of a reduced diameter as compared to the RCCA absorber) with stainless steel clad.

Comment: No. 4.1 [4] & [5]
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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.

Comment: No. 4.1 [6]
Deleted: neutron absorber rods, gray rods

4.1.1 Principal Design Requirements

The fuel and rod 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:*

- *Fuel damage, defined as penetration of the fuel clad, 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.*

Comment: No. 4.1 [7]
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* NRC Staff approval is required prior to implementing a change in this material; see DCD Introduction Section 3.5.

Table 4.1-1 (Sheet 1 of 3)

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	>1.25[d], >1.22[d]	>1.23	>1.26
Thimble (cold wall) flow channel	>1.25[d], >1.21[d]	>1.22	>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)	113.5	72.9	145.0
Effective flow rate for heat transfer (10^6 lbm/hr)	106.8	66.3	132.7
Effective flow area for heat transfer (ft^2)	41.8	38.5	51.1
Average velocity along fuel rods (ft/s)	15.8	10.6	16.6
Average mass velocity (10^6 lbm/hr-ft^2)	2.40	1.72	2.60
Coolant Temperature[c][e]			
Nominal inlet (°F)	535.0	532.8	561.2
Average rise in vessel (°F)	77.2	69.6	63.6
Average rise in core (°F)	81.4	75.8	68.7
Average in core (°F)	578.1	572.6	597.8
Average in vessel (°F)	573.6	567.6	593.0
Heat Transfer			
Active heat transfer surface area (ft^2)	56,700	44,884	69,700
Avg. heat flux (BTU/hr-ft^2)	199,300	143,000	181,200
Maximum heat flux for normal operation (BTU/hr-ft^2)[f]	518,200	372,226	489,200
Average linear power (kW/ft)[g]	5.72	4.11	5.20
Peak linear power for normal operation (kW/ft)[f][g]	14.9	10.7	14.0
Peak linear power (kW/ft)[f][h] (Resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	≤22.45	22.5	≤22.45

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Table 4.1-1 (Sheet 2 of 3)

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)	2[i]	2[i]	2
Intermediate	8 ZIRLO™	7 Zircaloy-4 or 7 ZIRLO™	8 ZIRLO™
Intermediate flow mixing	4 ZIRLO™	4 Zircaloy-4 or 5 ZIRLO™	0
Protective Grid - (Ni-Cr-Fe Alloy 718)	1	1	1
Loading technique, first cycle	3 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

Comment: No. 4.1-[9]

Table 4.1-1 (Sheet 3 of 3)

REACTOR DESIGN COMPARISON TABLE

Rod Cluster Control Assemblies	AP1000	AP600	Typical XL Plant
Neutron Absorber			
RCCA GRCA	24 Ag-In-Cd rodlets 12 304 SS rodlets 12 Ag-In-Cd 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 SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness, (Ag-In-Cd)	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
Fuel Enrichment First Cycle (Weight Percent)			
Region 1	2.35	1.90	Typical
Region 2	3.40	2.80	3.8 to 4.4
Region 3	4.45	3.70	(5.0 Max)

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Comment: No. 4.1 [10]
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Notes:

a. WRB-2M will be used in future reloads
b. See subsection 4.4.2.2.1 for the use of the W-3, 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. 1.25 applies to core and axial offset limits; 1.22 and 1.21 apply to all other RTDP transients
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.

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APPENDIX 1
DCD MARK-UPS FOR
CHAPTER 4 – 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.

Comment: No. 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).*]*

Comment: No. 4.2.[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

*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).

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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).

Deleted: WCAP-8218-P-A (Reference 6),
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Deleted: WCAP-13589-A (Reference 8)

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). Comment: No. 4.2.[4]

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

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

- 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

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

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.

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.

Comment: No. 4.2-[4]

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, and secondary neutron source assemblies. The temporary components are the primary neutron source assemblies (which are normally used only in the initial core), the burnable absorber assemblies, and the thimble plugs. 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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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

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.

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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 absorber material temperature does not exceed its melting temperature (1454°F for silver-indium-cadmium [Ag-In-Cd]), (see WCAP-9179, Reference 4).
- Temperature effects at operating conditions

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, 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).

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An alternative discrete burnable absorber is (WABA) Wet Annular Burnable Absorber. 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).

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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.

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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 structural 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 clad, and the long, solid fuel rod bottom end plug, provide debris failure mitigation.

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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.

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.

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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 may be boride-coated fuel pellets or fuel pellets containing gadolinium oxide mixed with uranium oxide. 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.

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Deleted: An evaluation and test program for the integral fuel burnable absorber design features for the boride-coated fuel pellets is summarized in Section 2.5 of WCAP-8183 (Reference 3).

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 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.

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 304 stainless steel.

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Deleted: The basic components of the welded top nozzle include the adapter plate, enclosure, and top plate.

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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.

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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

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 and mid-grids with bulge joint connections similar to those previously discussed for securing the grids to the guide thimbles.

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.

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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

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 stainless steel. The design stresses used for the Type 308 material are the same as those defined in the ASME Code, Section III, for Type 304 stainless steel. At room temperature, the yield and ultimate stresses per ASTM 580 (Reference 12) are exactly the same for the two alloys. In view of the similarity of composition of the alloys, the temperature dependence of strength for the two materials is expected to be the same.

The allowable stresses used as a function of temperature are listed in Table I-1.2 of the ASME Code, Section III. The fatigue strength for the Type 308 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 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 are joined to the hub by welding and brazing, and the fingers are joined to the vanes by brazing. 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 and 308 stainless steel except for the retainer, which is of Type 630 material, and the springs, which are nickel-chromium-iron Alloy 718.

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The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness. The pins are then welded in place. The end plug below the pin position 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

The mechanical design of the gray rod cluster assemblies plus the control rod drive mechanism and the interface with the fuel assemblies and guide thimbles are identical to 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 12 of the 24 rodlets are stainless steel while the remaining twelve contain the reduced diameter silver-indium-cadmium absorber material clad with stainless steel as the rod cluster control assemblies.

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The gray rod cluster assemblies are used in 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 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.

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The typical discrete burnable absorber rods (WABA) consist of pellets of alumina-boron carbide material contained within zirconium alloy tubes. These zirconium alloy tubes, which form the 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.

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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

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.

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:

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- The fuel/clad diametral gap to increase during steady-state operation
- Extensive departure from nucleate boiling propagation to occur

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.

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.

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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).

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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

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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.

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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.

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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

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 and source 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 rod or gray rod cluster assembly rodlets because no 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, 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.

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 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

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 control rod absorber material 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).

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). Comment: No. 4.2 [29]

The maximum temperature of the alumina-boron carbide burnable absorber pellet (WABA) 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: No. 4.2 [30]

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 are inspected to determine whether movement during transit exceeded design limitations.

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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 prior to removal of the fuel assembly from the container. 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

Complete. The AP1000 certified design baseline licensed design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design is presented in this Chapter of the DCD and supported by WCAP-16652-P (Reference 24).

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4.2.6 References

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

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Comment: No. 4.2 [32]
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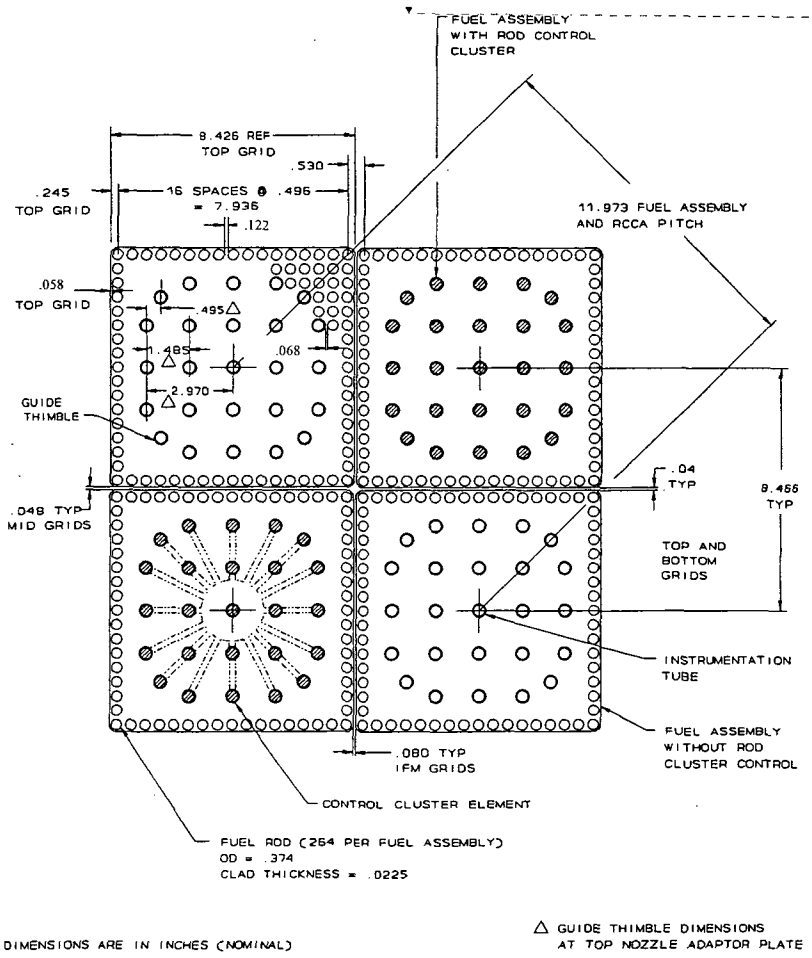

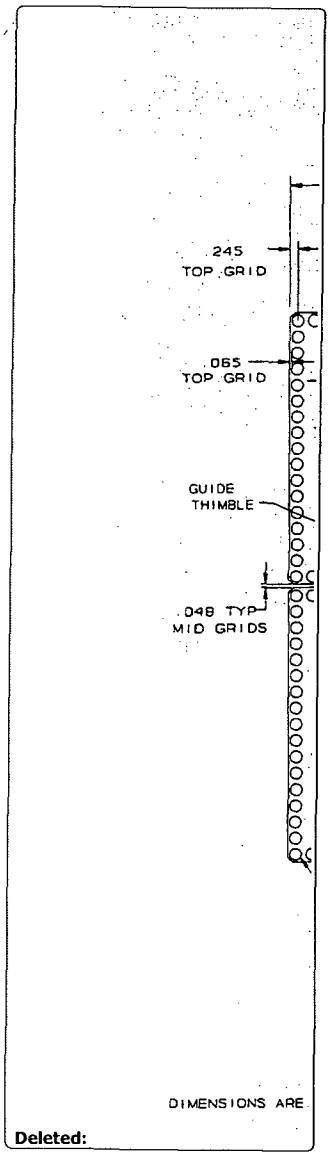

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Figure 4.2-1

Fuel Assembly Cross-Section

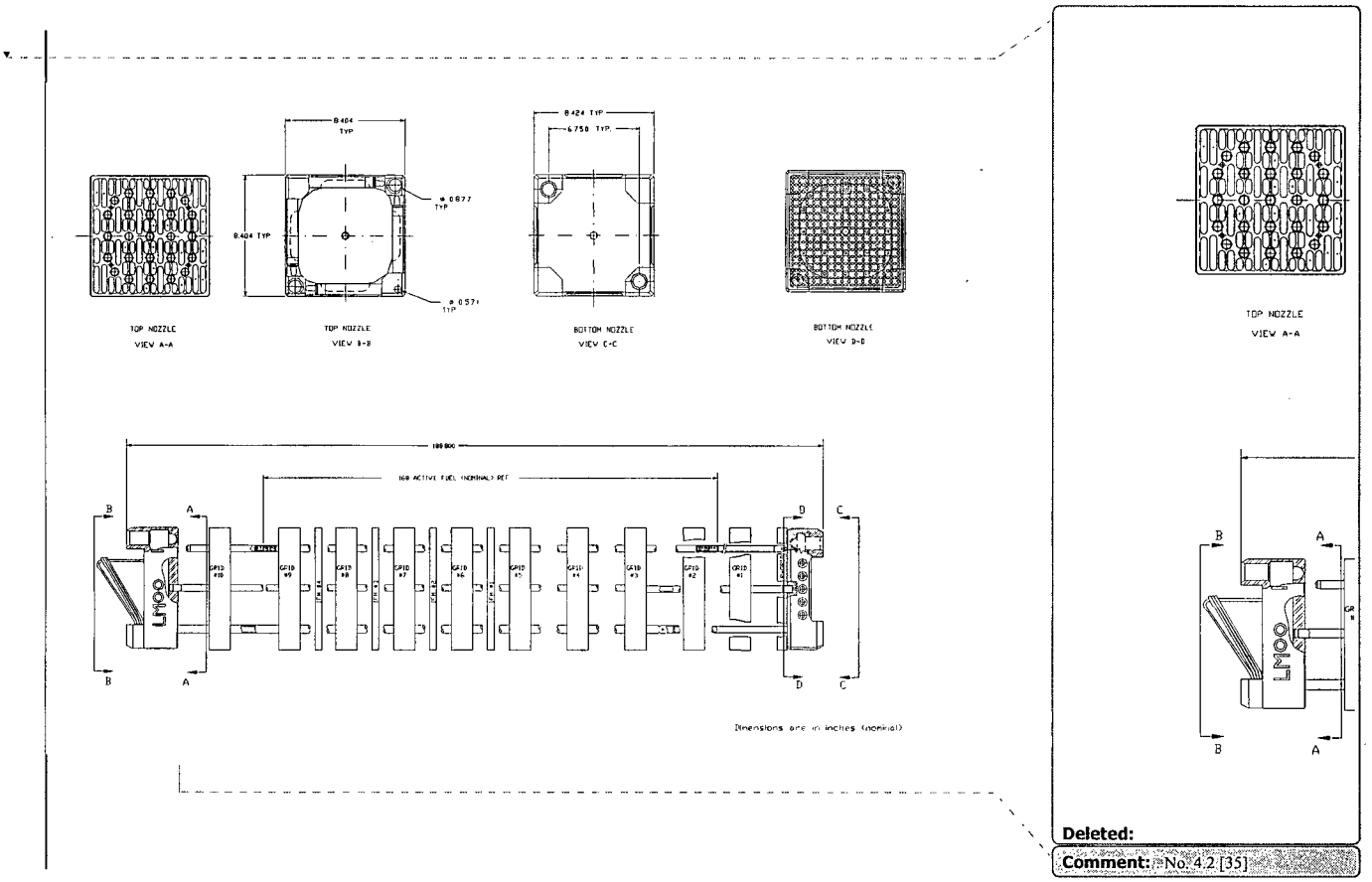

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Figure 4.2-2

Fuel Assembly Outline

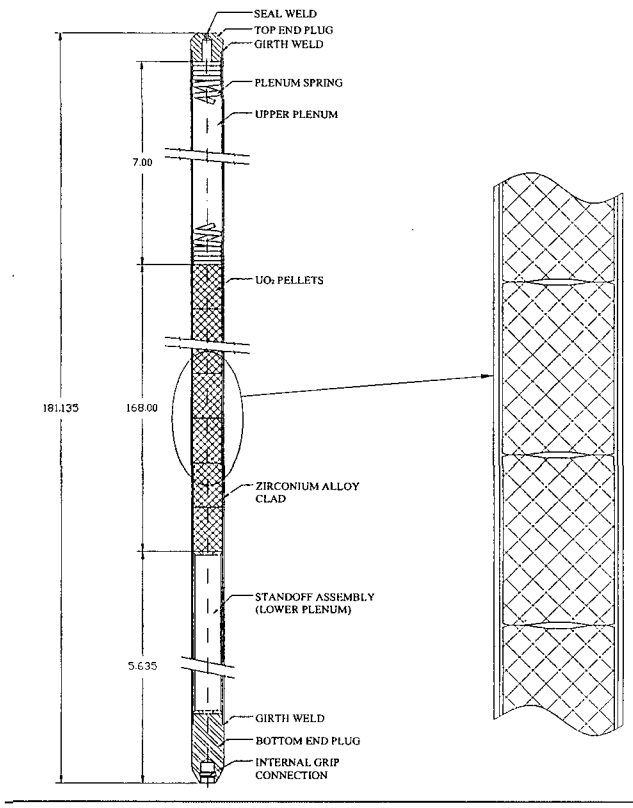
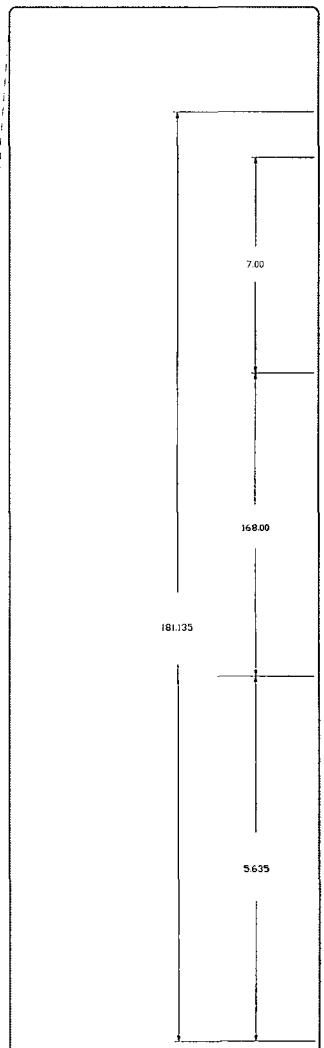


Figure 4.2-3

Fuel Rod Schematic



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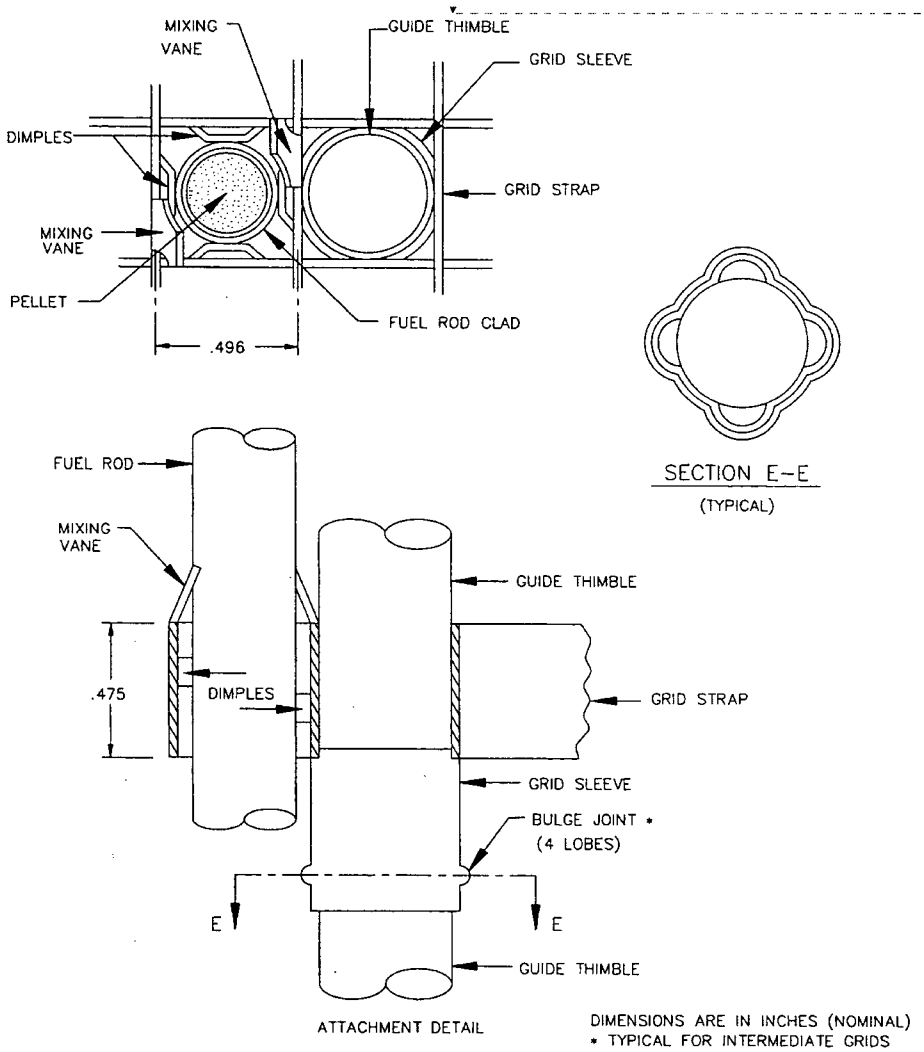

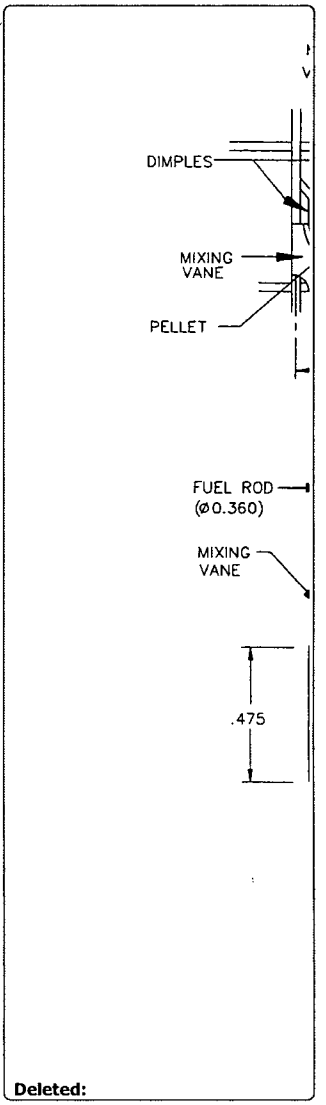

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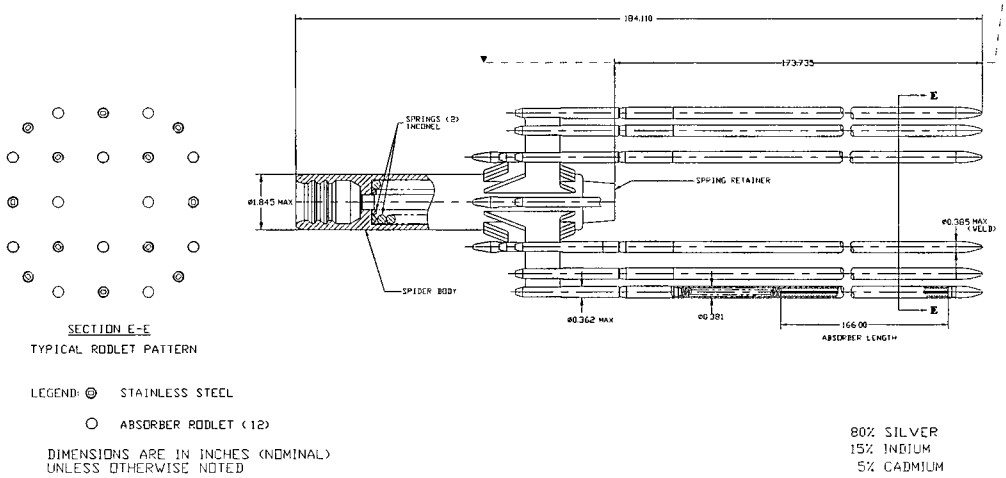
Figure 4.2-6

Intermediate Flow Mixer Grid to Thimble Attachment



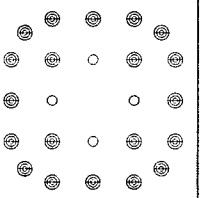


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Figure 4.2-11
Gray Rod Cluster Assembly

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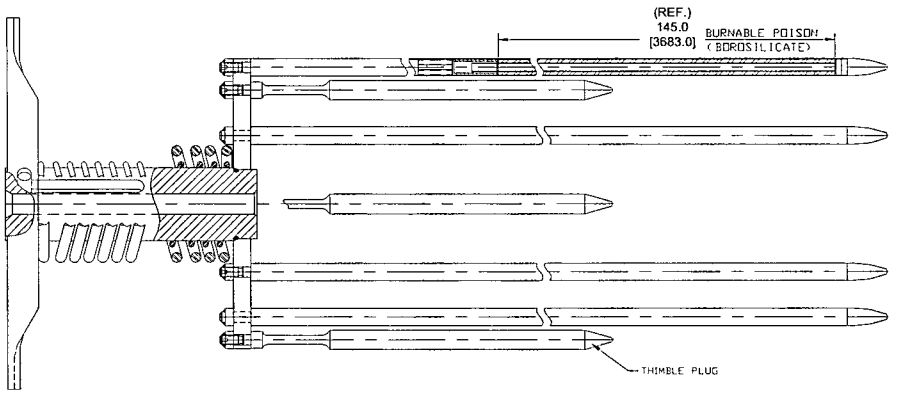

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Figure 4.2-13 Burnable Absorber Rod Assembly (Pyrex) Borosilicate Glass

APPENDIX 1
DCD MARK-UPS FOR
CHAPTER 4 – SECTION 4.3

where F^N_Q and F^E_Q are defined above and:

F^N_U = factor for calculational uncertainty, assumed to be 1.05.

F^N_{XY} = ratio of peak power density to average power density in the horizontal plane of peak local power.

F^N_Z = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then F^N_Z is the core average axial peaking factor.

4.3.2.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel assembly and burnable absorber loading patterns, the control rod pattern, and the fuel burnup distribution. Thus, at any time in the cycle, a horizontal section of the core can be characterized as unrodded or with control rods. These two situations combined with burnup effects determine the radial power shapes which can exist in the core at full power. Typical first cycle values of $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factors from beginning of life (BOL) to end of life (EOL) are given in Table 4.3-2. The effects on radial power shapes of power level, xenon, samarium, and moderator density effects are also considered, but these are quite small. The effect of nonuniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, since the moderator density is directly proportional to enthalpy, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater interest. Figures 4.3-6 through 4.3-11 show typical normalized power density distributions for one-eighth of the core for representative operating conditions. These conditions are as follows:

- Hot full power (HFP) near beginning of life, unrodded, no xenon
- Hot full power near beginning of life, unrodded, equilibrium xenon
- Hot full power near beginning of life, gray bank MA+MB in, equilibrium xenon
- Hot full power near middle of life (MOL), unrodded equilibrium xenon
- Hot full power near end of life, unrodded, equilibrium xenon
- Hot full power near end of life, gray bank MA+MB in, equilibrium xenon

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Since the position of the hot channel varies from time to time, a single-reference radial design power distribution is selected for departure from nucleate boiling calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power. The radial power distribution within a fuel rod and its variation with burnup as utilized in thermal calculations and fuel rod design are discussed in Section 4.4.

4.3.2.2.3 Assembly Power Distributions

For the purpose of illustration, typical rodwise power distributions from the beginning of life and end of life conditions corresponding to Figures 4.3-7 and 4.3-10, respectively, are given for the same assembly in Figures 4.3-12 and 4.3-13, respectively.

Since the detailed power distribution surrounding the hot channel varies from time to time, a conservatively flat radial assembly power distribution is assumed in the departure from nucleate boiling analysis, described in Section 4.4, with the rod of maximum integrated power artificially raised to the design value of $F_{\Delta H}^{N}$. Care is taken in the nuclear design of the fuel cycles and operating conditions to confirm that a flatter assembly power distribution does not occur with limiting values of $F_{\Delta H}^{N}$.

4.3.2.2.4 Axial Power Distributions

The distribution of power in the axial or vertical direction is largely under the control of the operator through either the manual operation of the control rods or the automatic motion of control rods in conjunction with manual operation of the chemical and volume control system. The automated mode of operation is referred to as mechanical shim (MSHIM) and is discussed in subsection 4.3.2.4.16. The rod control system automatically modulates the insertion of the axial offset (AO) control bank controlling the axial power distribution simultaneous with the MSHIM gray and control rod banks to maintain programmed coolant temperature. Operation of the chemical and volume control system is initiated manually by the operator to compensate for fuel burnup and maintain the desired MSHIM bank insertion. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial distribution of xenon, burnup, and axial distribution of fuel enrichment and burnable absorber. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time.

The online core monitoring system provides the operator with detailed power distribution information in both the radial and axial sense continuously using signals from the fixed in-core detectors. Signals are also available to the operator from the ex-core ion chambers, which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from the each ion chamber. The ion chamber signals are processed and calibrated against in-core measurements such that an indication of the power in the top of the core less the power in the bottom of the core is derived. The calibrated difference in power between the core top and bottom halves, called the flux difference (ΔI), is derived for each of the four channels of ex-core detectors and is displayed on the control panel. The principal use of the flux difference is to provide the shape penalty function to the OTΔT DNB protection and the OPΔT overpower protection.

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4.3.2.2.5 Local Power Peaking

Fuel densification occurred early in the evolution of pressurized water reactor fuel manufacture under irradiation in several operating reactors. This caused the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets can result in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary

the operator with the current allowable operating space, detailed current power distribution information, thermal margin assessment and operational recommendations to manage and maintain required thermal margins. As such, the online monitoring system provides the primary means of managing and maintaining required operating thermal margins during normal operation.

In the unlikely event that the online monitoring system is out of service, power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short term reactor operation. Limits are placed on the axial flux difference so that the heat flux hot channel factor F_Q is maintained within acceptable limits. A discussion of precalculated power distribution control in Westinghouse pressurized water reactors (PWRs) is included in WCAP-7811 (Reference 11). Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures, and on the measures taken to preclude exceeding design limits is presented in the Westinghouse topical report on power distribution control and load following procedures WCAP-8385 (Reference 12). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

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The calculations used to establish the upper bound on peaking factors, F_Q and $F^N_{\Delta H}$, include the nuclear effects which influence the radial and axial power distributions throughout core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients.

Power distributions are calculated for the full-power condition. Fuel and moderator temperature feedback effects are included within these calculations in each spatial dimension. The steady-state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the departure from nucleate boiling analysis of accidents. The effect of xenon on radial power distribution is small (compare Figures 4.3-6 and 4.3-7) but is included as part of the normal design process.

The core axial profile can experience significant changes, which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to thermal margin limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plant. Specifically, the nuclear design parameters significant to the axial power distribution analysis are as follows:

- Core power level
- Core height
- Coolant temperature and flow
- Coolant temperature program as a function of reactor power
- Fuel cycle lifetimes
- Rod bank worth
- Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

- Control rods in a single bank move together with no individual rod insertion differing from the bank demand position by more than the number of steps identified in the technical specifications.
- Control banks are sequenced with overlapping banks.
- The control bank insertion limits are not violated.
- Axial power distribution control procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures followed in normal operation with the online monitoring system out of service. In service, the online core monitoring system provides continuous indication of power distribution, shutdown margin, and margin to design limits.

The relaxed axial offset control (RAOC) procedures described in WCAP-10216-P-A (Reference 13) were developed to provide wide control band widths and consequently, more operating flexibility. These wide operating limits, particularly at lower power levels, increase plant availability by allowing quicker plant startup and increased maneuvering flexibility without trip. This procedure has been modified to accommodate AP1000 MSHIM operation. It is applied to analysis of axial power distributions under MSHIM control for the purpose of defining the allowed normal operating space such that Condition I thermal margin limits are maintained and Condition II occurrences are adequately protected by the reactor protection system when the online monitoring system is out of service.

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The purpose of this analysis is to find the widest permissible ΔI versus power operating space by analyzing a wide range of achievable xenon distributions, MSHIM/AO bank insertion, and power level.

The bounding analyses performed off line in anticipation of the online monitoring system being out of service is similar to that based on the relaxed axial offset control analysis, which uses a xenon reconstruction model described in WCAP-10216-P-A (Reference 13). This is a practical method which is used to define the power operating space allowed with AP1000 MSHIM operation. Each resulting power shape is analyzed to determine if loss-of-coolant accident constraints are met or exceeded.

The online monitoring system evaluates the effects of radial xenon distribution changes due to operational parameter changes continuously and therefore eliminates the need for overly conservative bounding evaluations when the online monitoring system is available. A detailed discussion of this effect may be found in WCAP-8385 (Reference 12). The calculated values have been increased by a factor of 1.05 for method uncertainty and a factor of 1.03 for the engineering factor F^E_Q.

The envelope drawn in Figure 4.3-14 represents an upper bound envelope on local power density versus elevation in the core. This envelope is a conservative representation of the bounding values of local power density.

The online monitoring system measures the core condition continuously and evaluates the thermal margin condition directly in terms of peak linear heat rate and margin to departure from nucleate boiling limitations directly.

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Deleted: Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within specified axial flux difference limits. These procedures are detailed in the technical specifications for the case of the online monitoring system not being available, and are followed by relying only upon ex-core surveillance supplemented by the normal monthly full core map requirement and by computer-based alarms on deviation from the allowed flux difference band.

Allowing for fuel densification effects, the average linear power at 3400 MW is 5.72 kW/ft. From Figure 4.3-14, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.60 corresponding to a peak linear heat rate of 15.0 kW/ft at each core elevation at 101 percent power.

To determine reactor protection system setpoints with respect to power distributions, three categories of events are considered: rod control equipment malfunctions and operator errors of commission or omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described above.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence) for both AO and MSHIM banks. Also included are motions of the AO and MSHIM banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences, assuming short-term corrective action; that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations, which include normal xenon transients. It is further assumed in determining the power distributions that total core power level would be limited by reactor trip to below the overpower protection setpoint of nominally 118 percent rated thermal power. Since the study is to determine protection limits with respect to power and axial offset, no credit is taken for OTΔT or OPΔT trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or below 118 percent, is less than that required for fuel centerline melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the AO and/or MSHIM rod banks in violation of the insertion limits and creates short-term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a power distribution limit violation (such as boration/dilution transient) assuming automatic operation of the rod control system which will maintain constant reactor power.

For each of the above categories, the trip setpoints are designed so as not to exceed fuel centerline melt criteria as well as fuel mechanical design criteria.

technology allow a complete functional integration of reaction rate measurement algorithms and the expected reaction rate predictive capability within the same software package. The predictive software integrated within the online monitoring system supplies accurate, detailed information of current reactor conditions. The historical algorithms are described in detail in WCAP-12472-P-A (Reference 4).

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The measured versus calculational comparison is performed continuously by the online monitoring system throughout the core life. The online monitoring system operability requirements are specified in the technical specifications.

In a measurement of the reactor power distribution and the associated thermal margin limiting parameters, with the in-core instrumentation system described in subsections 7.7.1 and 4.4.6, the following uncertainties must be considered:

A. Reproducibility of the measured signal

B. Errors in the calculated relationship between detector current and local power generation within the fuel bundle

C. Errors in the detector current associated with the depletion of the emitter material, manufacturing tolerances and measured detector depletion

D. Errors due to the inference of power generation some distance from the measurement thimble.

The appropriate allowance for category A has been accounted for through the imposition of strict manufacturing tolerances for the individual detectors. This approach is accepted industry practice and has been used in PWRs with fixed in-core instrumentation worldwide. Errors in category B above are quantified by calculation and evaluation of critical experiment data on arrays of rods with simulated guide thimbles, control rods, burnable absorbers, etc. These critical experiments provide the quantification of errors of categories A and D above. Errors in category C have been quantified through direct experimental measurement of the depletion characteristics of the detectors being used including the precision of the in-core instrumentation systems measurement of the current detector depletion. The description of the experimental measurement of detector depletion can be found in EPRI-NP-3814 (Reference 16).

WCAP-7308-L-P-A (Reference 7) describes critical experiments performed at the Westinghouse Reactor Evaluation Center and measurements taken on two Westinghouse plants with movable fission chamber in-core instrumentation systems. The measurement aspects of the movable fission chamber share the previous uncertainty categories less category C which is independent of the other sources of uncertainty. WCAP-7308-L-P-A (Reference 7) concludes that the uncertainty associated with peak linear heat rate (F_Q*P) is less than five percent at the 95 percent confidence level with only five percent of the measurements greater than the inferred value.

In comparing measured power distributions (or detector currents) with calculations for the same operating conditions, it is not possible to isolate the detector reproducibility. Thus, a comparison between measured and predicted power distributions includes some measurement error. Such a comparison is given in Figure 4.3-15 for one of the maps used in WCAP-7308-L-P-A (Reference 7). Since the first publication of WCAP-7308-L-P-A, hundreds of measurements have

been taken on reactors all over the world. These results confirm the adequacy of the five percent uncertainty allowance on the calculated peak linear heat rate (ALHR*F_Q*P).

A similar analysis for the uncertainty in hot rod integrated power $F_{\Delta H}$*P measurements results in an allowance of four percent at the equivalent of a 95 percent confidence level.

A measurement in the fourth cycle of a 157-assembly, 12-foot core is compared with a simplified one-dimensional core average axial calculation in Figure 4.3-16. This calculation does not give explicit representation to the fuel grids.

The accumulated data on power distributions in actual operation are basically of three types:

- Much of the data is obtained in steady-state operation at constant power in the normal operating configuration.
- Data with unusual values of axial offset are obtained as part of the ex-core detector calibration exercise performed monthly.
- Special tests have been performed in load follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in WCAP-7912-P-A (Reference 14). Figure 4.3-17 contains a summary of measured values of F_Q as a function of axial offset for five plants from that report.

4.3.2.2.8 Testing

A series of physics tests are planned to be performed on the first core. These tests and the criteria for satisfactory results are described in Chapter 14. Since not all limiting situations can be created at beginning of life, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the test. Tests performed at the beginning of each reload cycle are limited to verification of the selected safety-related parameters of the reload design.

4.3.2.2.9 Monitoring Instrumentation

The adequacy of instrument numbers, spatial deployment, required correlations between readings and peaking factors, calibration, and errors are described in WCAP-12472-P (Reference 4). The relevant conclusions are summarized in subsection 4.3.2.2.7 and subsection 4.4.6.

Provided the limitations given in subsection 4.3.2.2.6 on rod insertion and flux difference are observed, the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service. Further details of specific limits on the observed rod positions and flux difference are given in the technical specifications, together with a discussion of their bases.

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Limits for alarms and reactor trip are given in the technical specifications. Descriptions of the systems provided are given in Section 7.7.

resulting from power reduction consists of contributions from Doppler effect, moderator temperature, flux redistribution, and reduction in void content as discussed below.

4.3.2.4.1 Doppler Effect

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance cross-sections with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation.

4.3.2.4.2 Variable Average Moderator Temperature

When the core is shut down to the hot zero-power condition, the average moderator temperature changes from the equilibrium full-load value determined by the steam generator and turbine characteristics (such as steam pressure, heat transfer, tube fouling) to the equilibrium no-load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased to account for the control system dead band and measurement errors.

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When the moderator coefficient is negative, there is a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement at EOL.

4.3.2.4.3 Redistribution

During full-power operation, the coolant density decreases with core height. This, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady-state conditions, the relative power distribution will be slightly asymmetric toward the bottom of the core. On the other hand, at hot zero-power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler effect. The result will be a flux distribution which at zero power can be skewed toward the top of the core. Since a three-dimensional calculation is performed in determining total power defect, flux redistribution is implicitly included in this calculation. An additional redistribution allowance for adversely skewed xenon distributions is included in the determination of the total control requirement specified in Table 4.3-3.

4.3.2.4.4 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small positive reactivity contribution.

4.3.2.4.5 Rod Insertion Allowance

At full power, the MSHIM and AO banks are operated within a prescribed band of travel to compensate for small changes in boron concentration, changes in temperature, and very small changes in the xenon concentration not compensated for by a change in boron concentration. When the MSHIM banks reach a predetermined insertion or withdrawal, a change in boron concentration would be required to compensate for additional reactivity changes. Use of soluble boron is limited to fuel depletion and shutdown considerations. Since the insertion limit is set by

4.3.2.4.13 Gray Rod Cluster Assemblies

The rod cluster control assembly control banks include four gray rod banks consisting of gray rod cluster assemblies (GRCAs). Gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, it is the same as a rod cluster control assembly except that 12 of the 24 rodlets are comprised of stainless steel while the remaining 12 rodlets are reduced diameter silver-indium-cadmium clad with stainless steel. The term gray rod refers to the reduced reactivity worth relative to that of a rod cluster control assembly consisting of 24 silver-indium-cadmium rodlets. The gray rod cluster assemblies are used in load follow maneuvering and provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron (that is, chemical shim).

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4.3.2.4.14 Burnable Absorbers

Discrete burnable absorber rods or integral fuel burnable absorber rods or both may be used to provide partial control of the excess reactivity available during the fuel cycle. In doing so, the burnable absorber loading controls peaking factors and prevents the moderator temperature coefficient from being positive at normal operating conditions. The burnable absorbers perform this function by reducing the requirement for soluble boron in the moderator at the beginning of the fuel cycle, as described previously. For purposes of illustration, the initial cycle burnable absorber pattern is shown in Figure 4.3-5. Figures 4.3-4a and 4.3-4b show the burnable absorber distribution within a fuel assembly for several burnable absorber patterns used in the 17 x 17 array. The boron in the rods is depleted with burnup but at a slow rate so that the peaking factor limits are not exceeded and the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains within the limits stated above for power operating conditions.

4.3.2.4.15 Peak Xenon Startup

Compensation for the peak xenon buildup may be accomplished using the boron control system. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution can be made at any time, including during the shutdown period, provided the shutdown margin is maintained.

4.3.2.4.16 Load Follow Control and Xenon Control

During load follow maneuvers, power changes are primarily accomplished using control rod motion alone, as required. Control rod motion is limited by the control rod insertion limits as provided in the technical specifications and discussed in subsections 4.3.2.4.12 and 4.3.2.4.13. The power distribution is maintained within acceptable limits through limitations on control rod insertion. Reactivity changes due to the changing xenon concentration are also controlled by rod motion.

Rapid power increases (five percent/min) from part power during load follow operation are accomplished with rod motion.

The rod control system is designed to automatically provide the power and temperature control described above 30 percent rated power for most of the cycle length without the need to change

boron concentration as a result of the load maneuver. The automated mode of operation is referred to as mechanical shim (MSHIM) because of the usage of mechanical means to control reactivity and power distribution simultaneously. MSHIM operation allows load maneuvering without boron change because of the degree of allowed insertion of the control banks in conjunction with the independent power distribution control of the axial offset (AO) control bank. The worth and overlap of the MA, MB, MC, MD, M1, and M2 control banks are designed such that the AO control bank insertion will always result in a monotonically decreasing axial offset. MSHIM operation uses the MA, MB, MC, MD, M1, and M2 control banks to maintain the programmed coolant average temperature throughout the operating power range. The AO control bank is independently modulated by the rod control system to maintain a nearly constant axial offset throughout the operating power range.

The target axial offset used during MSHIM load follow operation is roughly the base load operation target axial offset less 10 percent. The negative bias is necessary to allow both positive and negative axial offset control effectiveness by the AO control bank. Extended base load operation is performed by controlling axial offset to the equilibrium target with the first moving M bank nearly fully withdrawn (at bite position) and AO bank fully withdrawn. The "bite" position is defined as the minimum control rod bank position required to provide a differential rod worth of at least 2 pcm/step.

Anticipated MSHIM load follow operation operates with two gray banks fully inserted to provide enough reactivity worth to compensate for transient reactivity effects without the need for soluble boron changes. The degree of control rod insertion under MSHIM operation allows rapid return to power without the need to change boron concentration. Gray rod operation is a Condition 1 event which includes the periodic exchange of gray rod banks. Comment: No. 4.3 [11]

4.3.2.4.17 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron and/or burnable absorbers. The boron concentration is limited during operating conditions to maintain the moderator temperature coefficient within its specified limits. A sufficient burnable absorber loading is installed at the beginning of a cycle to give the desired cycle lifetime, without exceeding the boron concentration limit. The end of a fuel cycle is reached when the soluble boron concentration approaches the practical minimum boron concentration in the range of 0 to 10 ppm.

4.3.2.4.18 Rapid Power Reduction System

The reactor power control system is designed with the capability of responding to full load rejection without initiating a reactor trip using the normal rod control system, reactor control system, and the rapid power reduction system. Load rejections requiring greater than a fifty percent reduction of rated thermal power initiate the rapid power reduction system. The rapid power reduction system utilizes preselected control rod groups and/or banks which are intentionally tripped to rapidly reduce reactor power into a range where the rod control and reactor control systems are sufficient to maintain stable plant operation. The consequences of accidental or inappropriate actuation of the rapid power reduction system is included in the cycle specific safety analysis and licensing process.

4.3.4 Combined License Information

Complete. The AP1000 certified design baseline licensed design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design is presented in this Chapter of the DCD and supported by WCAP-16652-P (Reference 62).

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4.3.5 References

1. Bordelon, F. M, et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Proprietary) and WCAP-9273-NP-A (Nonproprietary), July 1985.

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

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

59. Davidson, S. L., et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A (Proprietary) and WCAP-14297-A (Nonproprietary), March 1995.

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61. NTD-NRC-94-4275, Westinghouse's Interpretation of Staff's Position on Extended Burnup, August 29, 1994.

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Comment: No 4.3 [12]

Table 4.3-1 (Sheet 1 of 3)

[REACTOR CORE DESCRIPTION (FIRST CYCLE)]*

Active core	
Equivalent diameter (in.)	*119.7*
Active fuel height first core (in.), cold	*168*
Height-to-diameter ratio	*1.40*
Total cross section area (ft^2)	*78.14*
H_2O/U molecular ratio, cell, cold	*2.40*
Reflector thickness and composition	
Top - water plus steel (in.)	*~10*
Bottom - water plus steel (in.)	*~10*
Side - water plus steel (in.)	*~15*
Fuel assemblies	
Number	*157*
Rod array	*17 x 17*
Rods per assembly	*264*
Rod pitch (in.)	*0.496*
Overall transverse dimensions (in.)	*8.426 x 8.426*
Fuel weight, as UO_2 (lb)	*211,588*
Zircaloy clad weight (lb)	*43,105*
Number of grids per assembly	
Top and bottom - (Ni-Cr-Fe Alloy 718)	*2[(a)]*
Intermediate	*8 ZIRLO™*
Intermediate flow mixing (IFM)	*4 ZIRLO™*
Number of guide thimbles per assembly	*24*
Composition of guide thimbles	*ZIRLO™*
Diameter of guide thimbles, upper part (in.)	*0.442 ID x 0.482 OD*
Diameter of guide thimbles, lower part (in.)	*0.397 ID x 0.482 OD*
Diameter of instrument guide thimbles (in.)	*0.442 ID x 0.482 OD*

Note:

(a) The top grid will be fabricated of nickel-chromium-iron Alloy 718.

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Table 4.3-1 (Sheet 2 of 3)

*[REACTOR CORE DESCRIPTION (FIRST CYCLE)]**

Fuel rods	
Number	*41,448*
Outside diameter (in.)	*0.374*
Diameter gap (in.)	*0.0065*
Clad thickness (in.)	*0.0225*
Clad material	*ZIRLO™*
Fuel pellets	
Material	*UO_2 sintered*
Density (% of theoretical) (nominal)	*95.5*
Fuel enrichments (weight %)	
Region 1	*2.35*
Region 2	*3.40*
Region 3	*4.45*
Diameter (in.)	*0.3225*
Length (in.)	*0.387*
Mass of UO_2 per ft of fuel rod (lb/ft)	*0.366*
Rod Cluster Control Assemblies	
Neutron absorber	*Ag-In-Cd*
Diameter (in.)	*0.341*
Density (lb/in.^{3})	*Ag-In-Cd 0.367*
Cladding material	*Type 304, cold-worked SS*
Clad thickness (in.)	*0.0185*
Number of clusters, full-length	*53*
Number of absorber rods per cluster	*24*
Gray Rod Cluster Assemblies	
Neutron absorber	*Ag-In-Cd/304SS*
Diameter (in.)	*0.160*
Density (lb/in.^{3})	*Ag-In-Cd 0.367 / 304SS 0.285*
Cladding material	*Type 304, cold-worked SS*
Clad thickness (in.)	*0.0185*
Number of clusters, full-length	*16*
Number of absorber rods per cluster	*12 Ag-In-Cd / 12 304SS*

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Table 4.3-2 (Sheet 1 of 2)

*[NUCLEAR DESIGN PARAMETERS (FIRST CYCLE)]**

Core average linear power, including densification effects (kW/ft)		*5.72*
Total heat flux hot channel factor, F_Q		*2.60*
Nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$		*1.65*
Reactivity coefficients [a]	*Design Limits*	*Best Estimate*
Doppler-only power coefficients (see Figure 15.0.4-1) (pcm/% power) [b]		
Upper curve	*-19.4 to -12.6*	*-13.3 to -8.7*
Lower curve	*-10.2 to -6.7*	*-11.3 to -8.4*
Doppler temperature coefficient (pcm/°F) [b]	*-3.5 to -1.0*	*-2.1 to -1.3*
Moderator temperature coefficient (pcm/°F) [b]	*0 to -40*	*0 to -35*
Boron coefficient (pcm/ppm) [b]	*-13.5 to -5.0*	*-10.5 to -6.9*
Rodded moderator density (pcm/g/cm^3) [b]	*$\leq 0.47x10^5$*	*$\leq 0.45x10^5$*
Delayed neutron fraction and lifetime, β_{eff}		*0.0075(0.0044)* [c]
Prompt Neutron Lifetime, ℓ^*, μ s		*19.8*
Control rods		
Rod requirements		*See Table 4.3-3*
Maximum ejected rod worth		*See Chapter 15*
Bank worth HZP no overlap (pcm) [b]	*BOL, Xe Free*	*EOL, Eq. Xe*
MA Bank	*299*	*205*
MB Bank	*131*	*198*
MC Bank	*204*	*270*
MD Bank	*309*	*198*
M1 Bank	*858*	*632*
M2 Bank	*933*	*1405*
AO Bank	*2027*	*1571*

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Table 4.3-2 (Sheet 2 of 2)

*[NUCLEAR DESIGN PARAMETERS (FIRST CYCLE)]**

Typical Hot Channel Factors $F^N_{\Delta H}$	*BOL*	*EOL*
Unrodded	*1.40*	*1.33*
MA bank	*1.46*	*1.38*
MA + MB banks	*1.49*	*1.42*
MA + MB + MC banks	*1.50*	*1.31*
MA + MB + MC + MD banks	*1.50*	*1.37*
MA + MB + MC + MD + M1 banks	*1.52*	*1.45*
AO bank	*1.60*	*1.52*
Boron concentrations (ppm)		
Zero power, k_{eff} = 0.99, cold[(d)] RCCAs out	*1574*	
Zero power, k_{eff} = 0.99, hot[(e)] RCCAs out	*1502*	
Design basis refueling boron concentration	*2700*	
Zero power, $k_{eff} \leq 0.95$, cold[(d)] RCCAs in	*1179*	
Zero power, k_{eff} = 1.00, hot[(e)] RCCAs out	*1382*	
Full power, no xenon, k_{eff} = 1.0, hot RCCAs out	*1184*	
Full power, equilibrium xenon, k = 1.0, hot RCCAs out	*827*	
Reduction with fuel burnup		
First cycle (ppm/(GWD/MTU))[(f)]	*See Figure 4.3-3*	
Reload cycle (ppm/(GWD/MTU))	*~40*	

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Notes:

(a) Uncertainties are given in subsection 4.3.3.3.

(b) 1 pcm = 10^{-5} $\Delta\rho$ where $\Delta\rho$ is calculated form two statepoint values of k_{eff} by ln (k_1/k_2).

(c) Bounding lower value used for safety analysis.

(d) Cold means 68 °F, 1 atm.

(e) Hot means 557 °F, 2250 psia.

(f) 1 GWD = 1000 MWD. During the first cycle, a large complement of burnable absorbers is present which significantly reduce the boron depletion rate compared to reload cycles.

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

1.311				
1.166	1.271			
1.283	1.101	0.971		
1.161	1.257	1.062	1.152	
1.321	1.153	1.221	0.989	0.654
1.232	1.241	1.052	1.022	0.579
1.041	0.970	0.849	0.571	
0.592	0.473			

CALCULATED F-DELTA-H = 1.484

KEY: VALUE REPRESENTS ASSEMBLY RELATIVE POWER

Figure 4.3-8

Normalized Power Density Distribution Near Beginning of Life, Gray Bank MA+MB Inserted, Hot Full Power, Equilibrium Xenon

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0.993				
1.100	0.974			
0.990	1.063	0.770		
1.132	0.998	1.069	0.958	
1.057	1.166	1.023	1.049	0.671
1.206	1.079	1.144	1.129	0.709
1.049	1.184	1.040	0.751	
0.862	0.713			

CALCULATED F-DELTA-H = 1.411

KEY: VALUE REPRESENTS ASSEMBLY RELATIVE POWER

Figure 4.3-11

Normalized Power Density Distribution Near End of Life, Gray Bank MA+MB Inserted, Hot Full Power, Equilibrium Xenon

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APPENDIX 1
DCD MARK-UPS FOR
CHAPTER 4 – SECTION 4.4

have shown that fuel melting will not occur for achievable local burnups up to 75,000 MWD/MTU (Reference 81). The NRC has approved design evaluations up to 60,000 MWD/MTU in Reference 81 and up to 62,000 MWD/MTU in Reference 9.

4.4.1.2.2 Discussion

Fuel rod thermal evaluations are performed at rated power, at maximum overpower, and during transients at various burnups. These analyses confirm that this design basis and the fuel integrity design bases given in Section 4.2 are met. They also provide input for the evaluation of Condition III and IV events given in Chapter 15.

The center-line temperature limit has been applied to reload cores with a lead rod average burnup of up to 60,000 MWD/MTU. For higher burnups, the peak kilowatt-per-foot experienced during Condition I and II events is limited to that maximum value which is sufficient to provide that the fuel center-line temperatures remain below the melting temperature for the fuel rods. Thus, the fuel rod design basis that fuel rod damage not occur due to fuel melting continues to be met.

4.4.1.3 Core Flow Design Basis

4.4.1.3.1 Design Basis

Typical minimum value of 94.1 percent of the thermal flow rate is assumed to pass through the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles are not considered effective for heat removal.

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4.4.1.3.2 Discussion

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A typical maximum value of 5.9 percent of this value is allotted as bypass flow. This includes rod cluster control guide thimble and instrumentation tube cooling flow, leakage between the core barrel and the core shroud, head cooling flow, and leakage to the vessel outlet nozzles. The shroud core cavity flow is now considered as active flow that is effective for fuel rod cooling.

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

4.4.1.4 Hydrodynamic Stability Design Basis

Modes of operation associated with Condition I and II events do not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The design bases described in subsections 4.4.1 through 4.4.1.4 together with the fuel clad and fuel assembly design bases given in subsection 4.2.1 are sufficiently comprehensive that additional limits are not required.

Design Procedure (Reference 2). Uncertainties in these variables are determined from sampling of manufacturing data.

- Inlet flow maldistribution

  The consideration of inlet flow maldistribution in core thermal performances is described in subsection 4.4.4.2.2. A design basis of five-percent reduction in coolant flow to the hot assembly is used in the VIPRE-01 analyses.

- Flow redistribution

  The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the VIPRE-01 analyses for every operating condition evaluated.

- Flow mixing

  The subchannel mixing model incorporated in the VIPRE-01 code and used in reactor design is based on experimental data, as detailed in WCAP-7667-P-A (Reference 18) and discussed in subsections 4.4.2.2.3 and 4.4.4.5.1. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The VIPRE-01 mixing model is discussed in Reference 83.

4.4.2.2.5 Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in WCAP-8691 (Reference 19), is accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatism in the evaluation of DNBR and/or margin obtained form measured plant operating parameters (such as $F^N_{\Delta H}$ or core flow), which are less limiting than those required by the plant safety analysis, can be used to offset the effect of rod bow.

For the safety analysis of the AP1000, sufficient DNBR margin was maintained, as described in subsection 4.4.1.1.2, to accommodate the full and low flow rod bow DNBR penalties identified in Reference 20. The referenced penalties are applicable to the analyses using the WRB-2M or WRB-2 DNB correlations.

The maximum rod bow penalties (less than about 2 percent DNBR) accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required (Reference 21).

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transient. More adverse flow conditions occur during a loss-of-coolant accident. These conditions are presented in subsection 15.6.5.

Hydraulic loads at normal operating conditions are calculated considering the best estimate flow, described in Section 5.1, and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the cold best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create a flow rate 18-percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight.

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Hydraulic verification tests for the fuel assembly are described in Reference 86.

4.4.2.7 Correlation and Physical Data

4.4.2.7.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the Dittus-Boelter correlation (Reference 24), with the properties evaluated at bulk fluid conditions:

$$\frac{hD_c}{K} = 0.023 \frac{D_c G^{0.9}}{\mu} \frac{C_p \mu^{0.4}}{K}$$

where:

h = heat transfer coefficient (btu/h-ft^2-°F)
D_c = equivalent diameter (ft)
K = thermal conductivity (Btu/h-ft-°F)
G = mass velocity (lbm/h-ft^2)
μ = dynamic viscosity (lbm/ft-h)
C_p = heat capacity (Btu/lb-°F)

This correlation has been shown to be conservative (Reference 25) for rod bundle geometries with pitch-to-diameter ratios in the range used by pressurized water reactors.

The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's correlation (Reference 26). After this occurrence, the outer clad wall temperature is determined by:

$$\Delta T_{sat} = [0.072\exp(-P/1260)](q'')^{0.5}$$

where:

ΔT_{sat} = wall superheat, $T_W - T_{sat}$ (°F)
q'' = wall heat flux (Btu/h-ft^2)
P = pressure (psia)
T_w = outer clad wall temperature (°F)
T_{sat} = saturation temperature of coolant at pressure P (°F)

Finally, in the event that unexplained flux tilts do occur, the Technical Specifications provide appropriate corrective actions to provide continued safe operation of the reactor.

4.4.2.11 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases described in subsection 4.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation. A description of fuel clad integrity is presented in subsection 4.2.3.1.

The thermal-hydraulic design provides that the maximum fuel temperature is below the melting point of uranium dioxide, subsection 4.4.1.2. To preclude center melting and to serve as a basis for overpower protection system setpoints, a calculated center-line fuel temperature of 4700°F is selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations, as described in subsection 4.4.2.9.1. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the uranium dioxide thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad gap, and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, have been combined into a semi-empirical thermal model, discussed in subsection 4.2.3.3, that includes a model for time-dependent fuel densification, as given in WCAP-10851-P-A (Reference 49) and WCAP-15063-P-A, Revision 1 (Reference 85). This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to in-pile fuel temperature measurements (References 30 through 36, 50 and 85) and melt radius data (References 51 and 52) with good results.

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Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are performed at several times in the fuel rod lifetime (with consideration of time-dependent densification) to determine the maximum fuel temperatures.

The principal factors employed in the determination of the fuel temperature follow.

4.4.2.11.1 Uranium Dioxide Thermal Conductivity

The thermal conductivity of uranium dioxide was evaluated from data reported in References 37 through 48 and 53. At the higher temperatures, thermal conductivity is best obtained by using the integral conductivity to melt. From an examination of the data, it has been concluded that the best estimate is:

$$\int_{o}^{2800} K dt = 93 \text{ W/cm}$$

This conclusion is based on the integral values reported in References 51 and 53 through 57.

The design curve for the thermal conductivity is shown in Figure 4.4-2. The section of the curve at temperatures between 0° and 1300°C is in agreement with the recommendation of the International Atomic Energy Agency (IAEA) panel (Reference 58). The section of the curve above 1300°C is derived for an integral value of 93 W/cm. (References 51, 53, and 57).

4.4.2.11.5 Fuel Clad Temperatures

The outer surface of the fuel rod at the hotspot operates at a temperature a few degrees above fluid temperature for steady-state operation at rated power throughout core life due to the onset of nucleate boiling. At beginning of life this temperature is the same as the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of clad temperature.

4.4.2.11.6 Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum-to-core-average heat flux. The design value of F_Q, as presented in Table 4.3-2 and described in subsection 4.3.2.2.6, is 2.6 for normal operation.

As described in subsection 4.3.2.2.6, the peak linear power resulting from overpower transients/operator errors (assuming a maximum overpower of 118 percent) is less than 22.45 kW/ft. The centerline fuel temperature must be below the uranium dioxide melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is described in subsection 4.4.1.2 and results in a maximum allowable calculated center-line temperature of 4700°F. The peak linear power for prevention of center-line melt is 22.5 kW/ft. The center-line temperature at the peak linear power resulting from overpower transients/operator errors (assuming a maximum overpower of 118 percent) is below that required to produce melting.

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

4.4.3.1 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided as appropriate in Chapters 5, 6, and 9. Areas of interest are as follows:

- Total coolant flow rates for the reactor coolant system and each loop are provided in Table 5.1-3. Flow rates employed in the evaluation of the core are presented throughout Section 4.4.
- Total reactor coolant system volume including pressurizer and surge line and reactor coolant system liquid volume, including pressurizer water at steady-state power conditions, are given in Table 5.1-2.
- The flow path length through each volume may be calculated from physical data provided in Table 5.1-2.

- Comparing the impact event with the times and type of normally occurring plant operation events received from plant control system such as a control rod stepping, valve motion, pump start-ups, and others.

- Comparing the number of events detected within a given time interval. For example, a impact occurring more than two times in one minute may be considered as valid, but random impact occurring at sporadic intervals longer than one minute may not be considered as a valid alarm.

The sensors of the impact monitoring system are fastened mechanically to the reactor coolant system at potential loose part collection regions including the upper and lower head region of the reactor pressure vessel, and the reactor coolant inlet region of each steam generator. Sensors are mounted in a manner which protects the sensors from mechanical damage, compensates for thermal expansion and provides a constant holding force throughout the operating range, maintains the mounting resonance frequency greater than 17 kHz.

The equipment inside the containment is designed to remain functional through an earthquake of a magnitude equal to 50 percent of the calculated safe shutdown earthquake and normal environments (radiation, vibration, temperature, humidity) anticipated during the operating lifetime. The two instrument channels associated with the redundant sensors at each reactor coolant system location are physically separated from each other starting at the sensor locations to a point in the plant that is always accessible for maintenance during full-power operation.

The digital metal impact monitoring system is calibrated prior to plant startup. Capabilities exist for subsequent periodic online channel checks and channel functional tests and for offline channel calibrations at refueling outages.

4.4.7 Combined License Information

Complete. The AP1000 certified design baseline licensed design of the fuel, burnable absorber rods, rod cluster control assemblies, and initial core design is presented in this Chapter of the DCD and supported by WCAP-16652-P (Reference 87).

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

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4.4.8 References

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2. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Non-Proprietary), April 1989.

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79. Reference deleted

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82. Smith, L. D., et al., "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 (Proprietary) and WCAP-15026-NP (Non-Proprietary), April 1999.

83. Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A and WCAP-15306-NP-A, October 1999.

84. Stewart, C. W., et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Core," Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute.

85. Slagle, W. H. (ed.) et al., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 (Proprietary) and WCAP-15064-NP-A, Revision 1 (Non-Proprietary), July 2000.

86. Kitchen, T. J., "Generic Safety Evaluation for 17x17 Standard Robust Fuel Assembly (17x17 STD RFA)," SECL-98-056, Revision 0, September 30, 1998.

87. WCAP-16652-P, "AP1000 Core & Fuel Design Techncial Report"

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Table 4.4-1 (Sheet 1 of 2)

THERMAL AND HYDRAULIC COMPARISON TABLE (AP1000, AP600 AND A TYPICAL WESTINGHOUSE XL PLANT)

Design Parameters	AP1000[a]	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 BTU/hr)	11601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure, nominal (psia)	2250	2250	2250
System pressure, minimal (psia)	2190	2200	2204
Minimum DNBR at nominal conditions			
Typical flow channel	2.80	3.48	2.20
Thimble (cold wall) flow channel)	2.74	3.33	2.12
Minimum DNBR for design transients			
Typical flow channel	>1.25[b] >1.22[b]	>1.23	>1.26
Thimble (cold wall)flow channel	>1.25[b] >1.21[b]	>1.22	>1.24
DNB correlation[c]	WRB-2M	WRB-2	WRB-1
Coolant conditions[d]			
Vessel minimum measured flow rate (MMF)			
10^6 lbm/hr	115.55	74.4	148.9
gpm	301,670	193,200	403,000
Vessel thermal design flow rate (TDF)			
10^6 lbm/hr	113.5	72.9	145.0
gpm	296,000	189,600	392,000
Effective flow rate for heat transfer[e]			
10^6 lbm/hr	106.8	66.3	132.7
gpm	278,500	172,500	358,700
Effective flow area for heat transfer (ft^2)	41.8	38.5	51.1
Average velocity along fuel rods (ft/s)[c]	15.8	10.6	16.6
Average mass velocity, 10^6 lbm/hr-ft^2[e]	2.40	1.72	2.60
Coolant Temperature[d][e]			
Nominal inlet (°F)	535.0	532.8	561.2
Average rise in vessel (°F)	77.2	69.6	63.6
Average rise in core (°F)	81.4	75.8	68.7
Average in core (°F)	578.1	572.6	597.8
Average in vessel (°F)	573.6	567.6	593.0

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Table 4.4-1 (Sheet 2 of 2)

THERMAL AND HYDRAULIC COMPARISON TABLE (AP1000, AP600 AND A TYPICAL WESTINGHOUSE XL PLANT)

Design Parameters	AP1000[a]	AP600	Typical XL Plant
Heat transfer			
Active heat transfer surface area (ft^2)[f]	56,700	44,884	69,700
Average heat flux (BTU/hr-ft^2)	199,300	143,000	181,200
Maximum heat flux for normal operation (BTU/hr-ft^2)[g]	518,200	372,226	498,200
Average linear power (kW/ft)[f]	5.72	4.11	5.20
Peak linear power for normal operation (kW/ft)[g,h]	14.9	10.7	14.0
Peak linear power resulting from overpower transients/operator errors, assuming a maximum overpower of 118% (kW/ft)[h]	≤22.45	22.5	≤22.45
Peak Linear power for prevention of center-line melt (kW/ft)[i]	22.5	22.5	22.45
Power density (kW/l of core)[j]	109.7	78.82	98.8
Specific power (kW/kg uranium)[j]	40.2	28.89	36.6
Fuel central temperature			
Peak at peak linear power for prevention of centerline melt (°F)	4700	4,700	4700
Pressure drop[k]			
Across core (psi)	39.9 ± 4.0[l]	17.5 ± 1.7	38.8 ± 3.9
Across vessel, including nozzle (psi)	62.3 ± 6.2[l]	45.3 ± 4.5	59.7 ± 6.0

Notes:

(a) Robust Fuel Assembly
(b) 1.25 applies to Core and Axial Offset limits; 1.22 and 1.21 apply to all other RTDP transients
(c) WRB-2M is used for AP1000. WRB-2 or W-3 is used for AP1000 where WRB-2M is not applicable. See subsection 4.4.2.2.1 for use of W-3, WRB-2 and WRB-2M correlations
(d) Based on vessel average temperature equal to 573.6°F. Flow rates and temperatures based on 10 percent steam generator tube plugging
(e) Based on thermal design flow and 5.9 percent bypass flow
(f) Based on densified active fuel length. The value for AP1000 is rounded to 5.72 kW/ft.
(g) Based on 2.60 F_Q peaking factor
(h) See subsection 4.3.2.2.6
(i) See subsection 4.4.2.11.6
(j) Based on cold dimensions and 95.5 percent of theoretical density fuel for AP1000, 95 percent for others
(k) These are typical values based on best-estimate reactor flow rate as discussed in Section 5.1
(l) Inlet temperature = 536.8°F

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