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Your ref: Project Number 740
Our ref: DCP/NRC1900

May 24, 2007

Subject: AP1000 COL Standard Technical Report Submittal of APP-GW-GLR-119, Revision 0

In support of Combined License application pre-application activities, Westinghouse is submitting Revision 0 of AP1000 Standard Combined License Technical Report Number 119. The purpose of Technical Report 119 is to provide a justification for the removal of Tier 2* information from Chapter 4 of the Design Control Document in Revision 16 for the initial core for the AP1000 Combined License. Changes to the Design Control Document identified in Technical Report Number 119 are intended to be incorporated into FSARs referencing the AP1000 design certification or incorporated into an amended design certification. This report is submitted as part of the NuStart Bellefonte COL Project (NRC Project Number 740). The information included in this report is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification.

The purpose for submittal of this report was explained in a March 8, 2006 letter from NuStart to the U.S. Nuclear Regulatory Commission. This revision includes markups to DCD Appendix 3C.

Pursuant to 10 CFR 50.30(b), APP-GW-GLR-119, Revision 0, "AP1000 Design Control Document Chapter 4 Tier 2* Information," Technical Report Number 119, is submitted as Enclosure 1 under the attached Oath of Affirmation.

It is expected that when the NRC review of Technical Report Number 119 is complete, Tier 2* information will be removed from Chapter 4 of the DCD in Revision 16.

Questions or requests for additional information related to the content and preparation of this report should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Westinghouse requests the NRC to provide a schedule for review of this Technical Report within two weeks of its submittal.

D079

Very truly yours,

A. Sterdis, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

/Attachment

1. "Oath of Affirmation," dated May 24, 2007

/Enclosure

1. APP-GW-GLR-0119, Revision 0, "AP1000 Design Control Document Chapter 4 Tier 2* Information," Technical Report Number 119, dated May 2007.

cc:	D. Jaffe	-	U.S. NRC	1E	1A
	E. McKenna	-	U.S. NRC	1E	1A
	G. Curtis	-	TVA	1E	1A
	P. Grendys	-	Westinghouse	1E	1A
	P. Hastings	-	Duke Power	1E	1A
	C. Ionescu	-	Progress Energy	1E	1A
	D. Lindgren	-	Westinghouse	1E	1A
	A. Monroe	-	SCANA	1E	1A
	M. Moran	-	Florida Power & Light	1E	1A
	C. Pierce	-	Southern Company	1E	1A
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	G. Zinke	-	NuStart/Entergy	1E	1A
	M. Asztalos		Westinghouse	1E	1A

ATTACHMENT 1

"Oath of Affirmation"

ATTACHMENT 1

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	)
NuStart Bellefonte COL Project	)
NRC Project Number 740	)

APPLICATION FOR REVIEW OF
"AP1000 GENERAL COMBINED LICENSE INFORMATION"
FOR COL APPLICATION PRE-APPLICATION REVIEW

W. E. Cummins, being duly sworn, states that he is Vice President, Regulatory Affairs & Standardization, for Westinghouse Electric Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this document; that all statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

W E Cummins

W. E. Cummins
Vice President
Regulatory Affairs & Standardization

Subscribed and sworn to
before me this 24th day
of May 2007.

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Debra McCarthy, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Aug. 31, 2009
Member, Pennsylvania Association of Notaries

Debra McCarthy
Notary Public

ENCLOSURE 1

APP-GW-GLR-119, Revision 0

AP1000 Design Control Document Chapter 4 Tier 2* Information

Technical Report Number 119

F-AP-3.2-1 Rev. 0

AP1000 DOCUMENT COVER SHEET

TDC: ________ Permanent File: ________ APY: ________

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APP-GW-GLR-119	0	Page 1 of ~~30~~ 29	W-McGinnis

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TITLE: **AP1000 Design Control Document Chapter 4 Tier 2* Information**

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LEGAL REVIEW Joseph C. Spadacene	SIGNATURE/DATE 5/23/07
PATENT REVIEW M. M. Corletti	SIGNATURE/DATE M. M. Corletti 5/23/07

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* Approval of the responsible manager signifies that document is complete, all required reviews are complete, electronic file is attached and document is released for use.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

APP-GW-GLR-119
Revision 0

May 2007

AP1000 Standard Combined License Technical Report

AP1000 Design Control Document Chapter 4 Tier 2* Information

Westinghouse Electric Company LLC
Nuclear Power Plants
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INTRODUCTION

The purpose of Technical Report 119 is to provide a justification for the removal of Tier 2* information from Chapter 4 of the Design Control Document in Revision 16 for the initial core for the AP1000 Combined License. This information was asterisked because changes to the Tier 2* information must be reviewed by the NRC for the first reactor core. In WCAP-16652-NP (Technical Report 18), Westinghouse committed to provide a generic Core Reference Report 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 license holders in the submittal of License Amendment Requests. The core reference report will contain 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. Technical Report 119 requests permission to change DCD Chapter 4 Tier 2* information to Tier 2 based on the commitment to submit a generic Core Reference Report for the initial core.

Westinghouse proposes to change Chapter 4 Tier 2* information in Revision 16 of the DCD. The original Tier 2* information will be revised to normal text with no brackets, italics, or asterisks. Table 1 of Technical Report 119 commits that changes to the DCD Chapter 4 previous Tier 2* information in this table for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to fuel and core parameters made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

The remaining portions of Technical Report 119 contain full subsections from Revision 15 of Chapter 4 of the AP1000 Design Control Document that contained the Tier 2* information and the proposed changes.

CONCLUSION

Tier 2* information in Revision 15 of the Design Control Document and in Technical Report 18 (WCAP-16652-NP) is marked up in Appendix 1 to reflect removal of italics and bracketed, asterisked, information based on a commitment to submit changes to previous Tier 2* information for the initial core in a Core Reference Report.

REFERENCES

1. APP-GW-GLR-059, "AP1000 Core & Fuel Design Technical Report (18)" (WCAP-16652-NP, Revision 0)

2. APP-GW-GL-700, "AP1000 Design Control Document", Revision 15

APPENDIX 1

TABLE 1
REQUESTED CHANGE SUMMARY

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]	4.1.1 4.1.3	4.1.4 - 4.1.6	Remove Tier 2* information (remove italics, brackets, and asterisk) from the specific principle design requirements in Section 4.1.1. Remove Tier 2* from Reference 9 in Section 4.1.1. The change for the average first cycle linear power from 5.71 to 5.718 was previously made in APP-GW-GLR-059.	Changes to the principle design requirements and for the applicability of Reference 9 for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the principle design requirements and for the applicability of Reference 9 made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [1]	4.2	4.2-2	Remove Tier 2* information for the use of WCAP-12488 as a basis for a set a fuel design criteria.	Changes to the applicability of WCAP-12488 as a basis for a set a fuel design criteria for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the applicability of WCAP-12488 as a basis for a set a fuel design criteria made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [2]	4.2.1	4.2-2	Remove Tier 2* information for fuel design bases and acceptance limits	Changes to the fuel design bases and acceptance limits for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to fuel design bases and acceptance limits made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

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**
4.2 [3]	4.2.1.1.2	4.2-3	Remove Tier 2* information for clad stress and strain results	Changes to the clad stress and strain results for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the clad stress and strain results made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [4]	4.2.1.1.3	4.2-3	Remove Tier 2* information for the usage factor due to cyclic fatigue	Changes to the results of the determination of the usage factor due to cyclic fatigue for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the results of the determination of the usage factor due to cyclic fatigue made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [5]	4.2.1.5	4.2-5	Remove Tier 2* information for the use of WCAP-12488 methodology to evaluate fuel assembly design changes.	Changes to the applicability of WCAP-12488 as a valid methodology to evaluate fuel assembly design changes for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the applicability of WCAP-12488 as a valid methodology to evaluate fuel assembly design changes made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

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**
4.2 [6]	4.2.1.6	4.2-7	Remove Tier 2* information for the use of WCAP-12488 methodology to evaluate in-core control components.	Changes to the applicability of WCAP-12488 as a valid methodology to evaluate in-core control components for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the applicability of WCAP-12488 as a valid methodology to evaluate in-core control components made in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [7]	4.2.3	4.2-19	Remove Tier 2* information for the confirmation that fuel assemblies, fuel rods, and in-core components satisfy SRP Section 4.2 performance and safety criteria.	Changes to the confirmation that fuel assemblies, fuel rods, and in-core components satisfy SRP Section 4.2 performance and safety criteria for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the extent that SRP Section 4.2 performance and safety criteria are met in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.2 [8]	4.2.3, 4.2.6	4.2-19, 4.2.35	Remove Tier 2* information for the use of WCAP-12488 methodology for the fuel criteria evaluation process. Note, References 23 and 24 were previously added to Section 4.2 by APP-GW-GLR-059.	Changes to the methodology for the fuel criteria evaluation process for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the methodology for the fuel criteria evaluation process in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

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**
4.3 [1]	4.3.1, 4.3.5	4.3-1, 4.3-39	Remove Tier 2* information for the use of enhanced analytical techniques outside of those approved by NRC in WCAP-12488-P-A.	Changes to the use of enhanced analytical techniques outside of those approved by NRC in WCAP-12488-P-A for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the use of enhanced analytical techniques outside of those approved by NRC in WCAP-12488-P-A in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.3 [2]	4.3.1.1.1	4.3-2	Remove Tier 2* information for maximum average burnup and maximum extended burnup.	Changes to the maximum average burnup and maximum extended burnup for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the maximum average burnup and maximum extended burnup in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.3 [3]	Table 4.3-1		Remove Tier 2* status for the description of the first reactor core. Note that other changes to Table 4.3-1 are described in APP-GW-GLR-059.	Changes to the description of the first reactor core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the description of the core in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.3 [4]	Table 4.3-2		Remove Tier 2* status for the nuclear design parameters for the first reactor core. Note that other changes to Table 4.3-2 are described in APP-GW-GLR-059.	Changes to the nuclear design parameters for the first reactor core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the nuclear design parameters for the first reactor core in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

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**
4.3 [5]	Table 4.3-3		Remove Tier 2* status for the reactivity requirements for rod cluster control assembly for the first reactor core.	Changes to the reactivity requirements for the rod cluster control assemblies for the first reactor core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the reactivity requirements for the rod cluster control assemblies for the first reactor core in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.
4.4[1]	4.4.8	4.4-32	Remove Tier 2* status the use of WCAP-12488 methodology as the basis for maximum average burnup and maximum extended burnup.	Changes to the use of WCAP-12488 methodology for maximum average burnup and maximum extended burnup for the first core will be submitted as a part of a Core Reference Report to the NRC and submittal of a License Amendment Request (LAR) as described in 10 CFR 50.92 before initial startup. Changes to the use of WCAP-12488 methodology for maximum average burnup and maximum extended burnup in subsequent cycles will be made consistent with Westinghouse's NRC approved reload methodology.

APPENDIX 1
DCD TIER 2* Mark-ups for
Chapter 4 – Section 4.1

(remove brackets, italics, asterisks)

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:~~

~~[~~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,~~ $\cancel{F_Q}$~~, 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.~~

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.**

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

- ~~The control rod drive mechanisms are hydrotested after manufacture at a minimum of 150 percent of system design pressure.~~

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

- ~~For the initial fuel cycle, the moderator temperature coefficient is calculated to be negative for power operating conditions.]*~~ **The maximum rod control cluster assembly and gray rod speed (or travel rate) is 45 inches per minute.**

- **The control rod drive mechanisms are hydrotested after manufacture at a minimum of 150 percent of system design pressure.**

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

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

4.1.3 References

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

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

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

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

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

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

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

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

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

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

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

APPENDIX 1
DCD TIER 2* MARK-UPS FOR
CHAPTER 4 – SECTION 4.2

(remove brackets, italics, asterisks)

4.2 FUEL SYSTEM DESIGN

The plant conditions for design are divided into four categories.

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

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

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

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

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

  - The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged. The fraction of fuel rods damaged must be limited to meet the dose guidelines of 10 CFR 100 although sufficient fuel damage might occur to preclude immediate resumption of operation.

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

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

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

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

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

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

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

4.2.1 Design Basis

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

{The design bases and acceptance limits used by Westinghouse are also described in the Westinghouse Fuel Criteria Evaluation Process, WCAP-12488-A (Reference 1).}*

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

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

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

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

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

4.2.1.1 Cladding

4.2.1.1.1 Mechanical Properties

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

4.2.1.1.2 Stress-Strain Limits

Clad Stress

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

Clad Strain

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

4.2.1.1.3 Fatigue and Vibration

Fatigue

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

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).]*~~

.~~[~~**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.

4.2.1.6 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 control 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).]*~~

[Design changes to the in-core control components qualify for evaluation using the criteria defined in WCAP-12488-A (Reference 1).]*

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).

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

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

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

4.2.3 Design Evaluation

~~[The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria of]*~~**[The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria of]*** -Section 4.2 of the Standard Review Plan, the mechanical design bases of subsection 4.2.1 and ~~[the Fuel Criteria Evaluation Process per WCAP-12488-A (Reference 1)]*~~,**[the Fuel Criteria Evaluation Process per WCAP-12488-A (Reference 1)]***, and other interfacing nuclear and thermal and hydraulic design bases specified in Sections 4.3 and 4.4.

Effects of Conditions II, III, IV or anticipated transients without trip on fuel integrity are presented in Chapter 15.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rods whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria, the limiting rod is the lead burnup rod of a fuel region. In other instances, it may be the maximum power or the minimum burnup rod. For the most part, no single rod is limiting with respect to all the design criteria.

After identifying the limiting rod(s), an analysis is performed to consider the effects of rod operating history, model uncertainties, and dimensional variations. To verify adherence to the design criteria, the evaluation considers the effects of postulated transient power changes during operation consistent with Conditions I and II. These transient power increases can affect both rod average and local power levels. Parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses, these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and assembly models used for the performance evaluations are documented and maintained under an appropriate control system. Material properties used in the design evaluations are given in WCAP-12610 (Reference 5).

4.2.6 References

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

[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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9. Risher, D., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A (Proprietary), November 1976 and WCAP-8964-A (Non-Proprietary), August 1977.

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12. ASTM-A-580-90, Specification for Stainless and Heat-resisting Steel Wire.

13. Demario, E. E., "Hydraulic Flow Test of the 17x17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February 1974.

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15. Davidson, S. L. and Iorii, J. A., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-P-A (Proprietary) and WCAP-9500-A (Nonproprietary), May 1982.

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APPENDIX 1
DCD TIER 2* MARK-UPS FOR CHAPTER 4 – SECTION 4.3

(REMOVE BRACKETS, ITALICS, ASTERISKS)

4.3 NUCLEAR DESIGN

4.3.1 Design Basis

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC). The design bases are the fundamental criteria that must be met using approved analytical techniques. ~~[Enhancements to these techniques may be made provided that the changes are founded by NRC approved methodologies as discussed in]*~~**[Enhancements to these techniques may be made provided that the changes are founded by NRC approved methodologies as discussed in]~~*~~** -WCAP-9272-P-A (Reference 1) ~~and [WCAP-12488-P-A (Reference 2).]*~~

and [WCAP-12488-P-A (Reference 2).]~~*~~

The plant conditions for design are divided into four categories:

- Condition I - Normal operation and operational transients
- Condition II - Events of moderate frequency
- Condition III - Infrequent incidents
- Condition IV - Limiting faults

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

- In general, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.
- Condition II occurrences are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.
- Fuel damage, that is, breach of fuel rod clad pressure boundary, is not expected during Condition I and Condition II occurrences. A very small amount of fuel damage may occur. This is within the capability of the chemical and volume control system (CVS) and is consistent with the plant design basis.
- Condition III occurrences do not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation.

- The release of radioactive material due to Condition III occurrences is not sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary.
- A Condition III occurrence does not by itself generate a Condition IV occurrence or result in a consequential loss of function of the reactor coolant or reactor containment barriers.
- Condition IV faults do not cause a release of radioactive material that results in exceeding the limits of 10 CFR 100. Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which are included in the design.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and are maintained by the action of the control system.

The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters.

The control and protection systems are described in Chapter 7.

The consequences of Condition II, III, and IV occurrences are described in Chapter 15.

4.3.1.1 Fuel Burnup

4.3.1.1.1 Basis

A limitation on initial installed excess reactivity or average discharge burnup is not required other than as is quantified in terms of other design bases, such as overall negative power reactivity feedback discussed below~~. [The NRC has approved, in WCAP-12488-P-A (Reference 2), maximum fuel rod average burnup of 60,000 MWD/MTU. Extended burnup to 62,000 MWD/MTU has been established in Reference 61.]*~~. **{The NRC has approved, in WCAP-12488-P-A (Reference 2), maximum fuel rod average burnup of 60,000 MWD/MTU. Extended burnup to 62,000 MWD/MTU has been established in Reference 61.}***

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.}*

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

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

(other section 4.3 references are not Tier 2*)

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

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 x0.482 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

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

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]	$\leq 0.47\times10^5$	$\leq 0.45\times10^5$
Delayed neutron fraction and lifetime, β_{eff}		0.0075(0.0044)[c]
Prompt Neutron Lifetime, ℓ^*, μs		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

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

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.

~~APPENDIX 1~~

Table 4.3-3

[REACTIVITY REQUIREMENTS FOR ROD CLUSTER CONTROL ASSEMBLIES]*

Reactivity Effects (Percent)	BOL (First Cycle)	EOL (First Cycle)	EOL Representative (Equilibrium Cycle)
1. Control requirements			
Total power defect (%Δρ)(a)	1.89	2.54	3.02
Redistribution (adverse xenon only) (%Δρ)	0.27	0.40	0.32
Rod insertion allowance (%Δρ)	2.00	2.00	2.00
2. Total control (%Δρ)	4.16	4.94	5.34
3. Estimated RCCA worth (69 rods)			
a. All full-length assemblies inserted (%Δρ)	12.69	10.89	10.64
b. All assemblies but one (highest worth) inserted (%Δρ)	10.49	9.27	9.35
4. Estimated RCCA credit with 7 percent adjustment to accommodate uncertainties, item 3b minus 7 percent (%Δρ)	9.76	8.62	8.70
5. Shutdown margin available, item 4 minus item 2 (%Δρ)(b)	5.60	3.68	3.36

Notes:
(a) Includes void effects
(b) The design basis minimum shutdown is 1.60 percent

APPENDIX 1
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CHAPTER 4 – SECTION 4.4

(remove brackets, italics, asterisks)

4.4.8 References

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