AP1000DCDFileNPEm Resource

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Sent: Monday, July 12, 2010 5:08 PM
To: Buckberg, Perry
Cc: Loza, Paul G.; Melton, Michael A
Subject: OI-911-08
Attachments: OI-SRP9.1.1-SRSB-08 R2B-saa.pdf; OI-SRP9.1.1-SRSB-08 R2B Annotated DCD Pages.PDF

Perry,

Here is the rough draft of a revised OI question that I believe you and I agreed should be clarified and answer based on our visits, telecoms, review of other RAI details, criticality references, and some other DCD clean-up items. Please confirm the question and answers cover the type and level of information NRC needs to close this issue as we had discussed. I trust that the proposed DCD clarifications are clear and concise.

Pending your feedback or discussion, I'll modify the rough draft OI and resend to NRC and NuStart for draft review.
Please confirm.

Thank you.

--SCOTT ALTMAYER--
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Hearing Identifier: AP1000_DCD_Review
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AP1000 TECHNICAL REPORT REVIEW

Response to Open Item (OI)

RAI Response Number: OI-SRP9.1.1-SRSB-08
Revision: 2B

Question: Spent Fuel Criticality Analysis

(Revision 0 and 1)

Based on the review of the TR65 (Revision 1) criticality analysis methodology and its application, the NRC staff questioned the applicant's burnup credit assumption that a 5% reactivity uncertainty penalty included the effects of missing nuclide data on the computational biases and uncertainties. This issue was raised in RAI-SRP9.1.1-SRSB-08.

In response to this 5% burnup uncertainty concern by the NRC, the applicant's September 16, 2009 letter described a loading pattern restriction on the Region 2 racks and the applicant's plan to submit a simplified analysis that does not require burnup credit (or which will remove or preclude the need for using burnup credit). This plan will not require any changes to the physical rack design as presented in TR-65. Evaluation of this restricted loading pattern, corresponding analyses, and Technical Specification changes related to this restricted loading pattern are tracked by **OI-SRP9.1.1-SRSB-08**.

New Question: (Revision 2B)

Additional review of the updated Chapter 9 SER proprietary clarifications from Westinghouse, review of various RAI and DCD changes noted in RAI-SRP9.1.1-SRSB-05 to support the criticality methodology for spent fuel storage, redesign of the spent fuel pool racks, and completion of the spent fuel pool and new fuel pit seismic/structural audit action items has occurred. NRC believes additional clarification and changes are needed to the DCD to clarify, correct, and complete statements regarding new fuel rack and spent fuel rack criticality.

Westinghouse Response

(Revision 0 and 1):

This SER Open Item (OI) response completes the three tasks noted below:

1) It clarifies Westinghouse's intent to retain and license a full-capacity spent fuel pool. This clarification is based on a criticality analysis that meets the requirements of both 10 CFR 50.68, and the current guidelines established by the NRC, regarding how to account for burnup uncertainties. The Westinghouse response, technical approach, and revised criticality analysis (i.e. APP-GW-GLR-029 (TR65), Rev. 2) to satisfactorily address the NRC concerns regarding the 5% burnup credit assumption noted above, are stated in the response to RAI-SRP9.1.1-SRSB-08 (Reference 1). The DCD changes describing the full-capacity spent fuel pool (SFP) were submitted in response to RAI-SRP9.1.1-SRSB-05 (Reference 4). These

two RAIs and the supporting criticality analysis are requested to be approved by the NRC for use in the pending AP1000DCD amendment.

2) It retracts the Westinghouse proposal noted in the September 16, 2009 letter (Reference 2) that suggested a restricted loading pattern (i.e. checkerboard pattern) would be pursued as the primary criticality safety basis for the SFP. That letter proposed to submit an alternate conservative SFP criticality analysis that takes no credit for fuel depletion or burnup. This checkerboard pattern would have imposed an adverse loading restriction on the Region 2 racks by reducing the spent fuel storage capacity by over 300 cells. Westinghouse chose not to exercise this proposal and did not describe or submit details of the checkerboard pattern or analysis to NRC.

3) It explicitly requests NRC to evaluate, review, and approve the backup checkerboard pattern and criticality analysis in calendar year 2010 to support the potential future use of this restricted loading pattern for the AP1000 SFP. This backup approach uses a new methodology that meets 10 CFR 50.68, does not use or reference WCAP 14416-NP-A, and does not credit soluble boron or fuel assembly burnup for subcriticality control in the Region 2 spent fuel racks. This backup checkerboard approach and criticality analysis is described in Westinghouse calculation APP-FS02-N1C-003, Rev. 0 (Reference 3). A proprietary version of this calculation is being submitted by separate letter to the NRC for review and audit, as applicable.

   Westinghouse considers this checkerboard approach to be a proactive contingency for potential future use and operational flexibility of the AP1000 SFP if more restrictive decisions or guidelines regarding burnup credit uncertainties are imposed by the NRC in the future during the AP1000 rulemaking process. The intent of submitting this checkerboard pattern for approval is to provide a contingency for use by Westinghouse and the NRC that confirms the AP1000 SFP design remains safe and licensable relative to nuclear criticality.

New Response: (Revision 2B)

During telecoms and finalization efforts, Westinghouse and NRC have identified additional DCD changes that will clarify, correct, and/or complete statements regarding new fuel rack and spent fuel rack criticality, references, and dimensioning. These additional changes are conforming in nature and are supported by existing calculations, RAI responses, design changes, and/or errata to correct grammatical anomalies. They will assure consistency and completeness of the DCD.

The summary of proposed DCD changes and bases are shown in the table below; and are additions to those DCD changes already completed and processed. The table precedes the annotated DCD pages attached to the “DCD changes” section of this RAI.

Reference(s):

1) Westinghouse Letter DCP/NRC_002735, 1/8/10, re: response to RAI-SRP9.1.1-SRSB-08

2) Westinghouse Letter DCP/NRC_002619, 9/16/09, re: alternate restricted loading pattern for spent fuel pool to show 10 CFR 50.68 compliance

3) APP-FS02-N1C-003, Rev. 0, January 2009, “AP1000 Spent Fuel Pool Criticality Analysis without Credit for Soluble Boron or Assembly Burnup” (Proprietary)

4) Westinghouse Letter DCP/NRC_002511, 5/29/09, re: response to RAI-SRP9.1.1-SRSB-05 (contains DCD markups that support criticality analysis for full-capacity SFP loading)

Design Control Document (DCD) Revision:

DCD Changes: (Revision 0, 1)

The DCD changes describing the full-capacity SFP are identified in the response to RAI-SRP9.1.1-SRSB-05 (Reference 4). These DCD changes remain applicable to this response.

If the backup restricted-loading checkerboard pattern and criticality analysis are needed for use in the future, appropriate DCD changes, ISG-11 notifications, and applicable processes will be completed and submitted to the NRC at that time.

DCD Changes: (Revision 2B)

See attached summary table and annotated DCD mark-up pages.

PRA Revision: None.

Technical Report (TR) Revision: None.

Summary of Proposed DCD Changes Regarding Nuclear Criticality			
TOPIC (DCD Section)	**Location (Sect., Page, Para., Line)**	**Change Description**	**Change Basis**
Tier 1, ITAAC			
2.1.1, Fuel Handling and Refueling System	2.1.1, pg. 2.1.1-1 Item 7	Clarify criticality evaluation acceptance criteria by adding Code 10 CFR 50.68 reference and note. (ref. RAI-SRP9.1.1-SRSB-05)	Explain
	Table 2.1.1-1, pg. 2.1.1-3 Item 7, Acceptance Criteria	Clarify criticality evaluation acceptance criteria by adding Code 10 CFR 50.68 reference and note. (ref. RAI-SRP9.1.1-SRSB-05)	Explain
4.3.2.6.1, Criticality Design Method Outside the Reactor	4.3.2.6.1, pg. 4.3-29	Add last sentence to state where new fuel rack and spent fuel rack criticality methods are addressed. **(See Insert #1)**	Simplify
	Table 4.3-4, pg. 4.3-51	Delete table. New benchmark critical experiments exist for spent fuel racks criticality that supercede those listed in the table. (ref. RAI-SRP9.1.1-SRSB-05)	Correction
New Fuel Racks			
9.1.1.1, Design Bases	9.1.1.1, pg. 9.1-1, Para. 2, Line 6-7	Remove duplicate statement on assembly insertion to a full location. Carried from AP600.	Simplify
9.1.1.3, Safety Evaluation	9.1.1.3, pg. 9.1-5, Para. 3, Line 1	Change last sentence to include new fuel rack and directly state Reference 17	Simplify
Spent Fuel Racks			
9.1.2.1, Design Bases	9.1.2.1, Pg. 9.1-5, Para. 2, Line 2	Remove duplicate statement on assembly insertion to a full location. Carried from AP600.	Simplify

9.1.2.3, Safety Evaluation	9.1.2.3, pg. 9.1-11, Para. 5, Line 2	Change last sentence to include spent fuel rack and directly state Reference 20	Simplify
9.1.7, References	9.1.7, pg. 9.1-50	Show Reference 20 as "(Proprietary)" (ref. RAI-SRP9.1.1-SRSB-05)	Correction
Tech Specs			
Tech Spec Design Features, 4.3, Fuel Storage	4.3.1.1.c, pg. 4.0 - 2, Lines 1 and 2	Change "10.90" to "10.93" and "9.028" to 9.04" (ref. RAI-SRP9.1.2-SRSB-06)	Correction
Tech Spec BASES 3.7.11, Fuel Storage Pool Boron Concentration	pg. B 3.7.11-3, Reference 3	Change "critically" to "criticality"	Correction
Tech Spec BASES 3.7.12, Spent Fuel Pool Storage	pg. B 3.7.12-2, **Background (cont'd),** Line 4-5	Typo from DCD Rev. 16. Correct the partial sentences to match the misplaced fuel accident described in B 3.7.11, "Background" (pg. B 3.7.11-1). **(See Insert #2)**	Correction
	pg. B 3.7.12-3, **Actions**, Para. 1, Line 3, Para. 2, Line 3	Change "Region 2 or 3" to "Region 2" (ref. RAI-SRP9.1.1-SRSB-05)	Correction
	pg. B 3.7.12-4, **Surveillance Requirements,** Para. 1, Line 7	Add "…or the Defective Fuel Cells" commensurate with flexibility in Action A.1 and LCO 3.7.12 that this is an acceptable storage location. Consistent with other DCD changes. (ref. RAI-SRP9.1.1-SRSB-05)	Correction

Insert #1

Insert #1 (to 4.3.2.6.1, Criticality Design Method Outside the Reactor):
Add sentence below after first paragraph to state where new fuel rack and spent fuel rack criticality methods are addressed.

"The details of the methodology used for the new fuel rack and spent fuel rack criticality analysis are included in the Chapter 9.1 references."

Insert #2

Insert #2 (pg. B 3.7.12-2, Background (cont'd), *copied from B 3.7.11, Background:*
Clarify accident described in Line 4-5 by replacing with sentences below.

"For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison."

Annotated DCD Changes

OI-SRP9.1.1-SRSB-08 (Revision 2B)

(Cover plus 15 pages)

Annotated DCD Changes

OI-SRP9.1.1-SRSB-08 (Revision 2B)

(Cover plus 15 pages)

2.1.1 Fuel Handling and Refueling System

Design Description

The fuel handling and refueling system (FHS) transfers fuel assemblies and core components during fueling operations and stores new and spent fuel assemblies in the new and spent fuel storage racks. The refueling machine (RM) and the fuel transfer tube are operated during refueling mode. The fuel handling machine (FHM) is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the FHS are as shown in Table 2.1.1-2.

1. The functional arrangement of the FHS is as described in the Design Description of this Section 2.1.1.

2. The FHS has the RM, the FHM, and the new and spent fuel storage racks.

3. The FHS preserves containment integrity by isolation of the fuel transfer tube penetrating containment.

4. The RM and FHM/spent fuel handling tool (SFHT) gripper assemblies are designed to prevent opening while the weight of the fuel assembly is suspended from the grippers.

5. The lift height of the RM mast and FHM hoist(s) is limited such that the minimum required depth of water shielding is maintained.

6. The RM and FHM are designed to maintain their load carrying and structural integrity functions during a safe shutdown earthquake.

7. The new and spent fuel storage racks maintain the effective neutron multiplication factor less than the required limits during normal operation, design basis seismic events, and design basis dropped fuel assembly accidents.

Add: 10 CFR 50.68

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the FHS.

Table 2.1.1-1 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7. The new and spent fuel storage racks maintain the effective neutron multiplication factor less than the required limits during normal operation, design basis seismic events, and design basis dropped fuel assembly accidents.	i) Analyses will be performed to calculate the effective neutron multiplication factor in the new and spent fuel storage racks during normal conditions.	i) The ~~calculated~~ effective neutron multiplication factor for the ~~new~~ and spent fuel storage racks ~~is less than 0.95~~meets the requirements of 10CFR50.68 under normal conditions[1].
	ii) Inspection will be performed to verify that the new and spent fuel storage racks are located on the nuclear island.	ii) The new and spent fuel storage racks are located on the nuclear island.
	iii) Seismic analysis of the new and spent fuel storage racks will be performed.	iii) A report exists and concludes that the new and spent fuel racks can withstand ~~seismic~~ design basis ~~dynamic~~ loads and maintain ~~the~~ calculated effective neutron multiplication factor ~~less than 0.95~~required by 10CFR50.68[1].
	iv) Analysis of the new and spent fuel storage racks under design basis dropped fuel assembly loads will be performed.	~~iv) A report~~ exists and concludes that the new and spent fuel racks can withstand design basis dropped fuel assembly loads and ~~maintain the~~ calculated effective neutron multiplication factor ~~less than 0.95~~required by 10CFR50.68[1].

Note:

1) The requirements of 10CFR50.68 are summarized as follows:

For New Fuel Storage Racks:

a) The effective neutron multiplication factor (k-effective) is less than 0.95 when flooded with unborated water; and,

b) K-effective is less that 0.98 with optimum moderator conditions.

For Spent Fuel Storage Racks:

If methodology does not take credit for soluble boron:

a) K-effective is less than 0.95 when flooded with unborated water.

Or, if methodology takes credit for soluble boron:

a) K-effective is less than 0.95 when flooded with borated water; and,

b) K-effective is less than 1.0 if flooded with unborated water.

The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin, allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling

The basis for maintaining the reactor subcritical during refueling is presented in subsection 4.3.1.5, and a discussion of how control requirements are met is given in subsections 4.3.2.4 and 4.3.2.5.

4.3.2.6.1 Criticality Design Method Outside the Reactor

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping, and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing ~~the minimum separation~~ between assemblies and/or inserting neutron poisons between assemblies. Insert #1

The design criteria are consistent with General Design Criterion (GDC) 62, Reference 19, and NRC guidance given in Reference 20. The applicable 10 CFR Part 50.68 requirements are as follows:

1. The maximum K-effective value, including all biases and uncertainties, must be less than 0.95 with soluble boron credit and less than 1.0 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 4 of Paragraph b. Note that the specific terminology is:

   "If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

2. The maximum enrichment of fresh fuel assemblies must be less than or equal to 5.0 weight-percent U-235. Note this design criterion is provided in 10 CFR Part 50.68, Item 7 of Paragraph b. Note that the specific terminology is:

   "The maximum nominal U–235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight."

The following conditions are assumed in meeting this design bases:

- The fuel assembly contains the highest enrichment authorized without any control rods or non-integral burnable absorber(s) and is at its most reactive point in life.

Insert #1

Insert #1 (to 4.3.2.6.1, Criticality Design Method Outside the Reactor):
Add sentence below after first paragraph to state where new fuel rack and spent fuel rack criticality methods are addressed.

"The details of the methodology used for the new fuel rack and spent fuel rack criticality analysis are included in the Chapter 9.1 references."

Delete

Table 4.3-4

BENCHMARK CRITICAL EXPERIMENTS[a]

Critical Number	General Description	Enrichment ^{235}U w/o	Reflector	Separating Material	Soluble Boron (ppm)	Measured K_{eff}	KENO K_{eff}	KENO K_{eff} One Sigma
1	UO2 Rod Lattice	2.46	water	water	0	1.0002	0.9966	0.0024
2	UO2 Rod Lattice	2.46	water	water	1037	1.0001	0.9914	0.0019
3	UO2 Rod Lattice	2.46	water	water	764	1.0000	0.9943	0.0019
4	UO2 Rod Lattice	2.46	water	B4C pins	0	0.9999	0.9871	0.0022
5	UO2 Rod Lattice	2.46	water	B4C pins	0	1.0000	0.9902	0.0022
6	UO2 Rod Lattice	2.46	water	B4C pins	0	1.0097	0.9948	0.0021
7	UO2 Rod Lattice	2.46	water	B4C pins	0	0.9998	0.9886	0.0021
8	UO2 Rod Lattice	2.46	water	B4C pins	0	1.0083	0.9973	0.0021
9	UO2 Rod Lattice	2.46	water	water	0	1.0030	0.9966	0.0021
10	UO2 Rod Lattice	2.46	water	water	143	1.0001	0.9973	0.0021
11	UO2 Rod Lattice	2.46	water	stainless steel	514	1.0000	0.9992	0.0020
12	UO2 Rod Lattice	2.46	water	stainless steel	217	1.0000	1.0031	0.0021
13	UO2 Rod Lattice	2.46	water	borated aluminum	15	1.0000	0.9939	0.0022
14	UO2 Rod Lattice	2.46	water	borated aluminum	92	1.0001	0.9882	0.0022
15	UO2 Rod Lattice	2.46	water	borated aluminum	395	0.9998	0.9854	0.0021
16	UO2 Rod Lattice	2.46	water	borated aluminum	121	1.0001	0.9848	0.0022
17	UO2 Rod Lattice	2.46	water	borated aluminum	487	1.0000	0.9892	0.0021
18	UO2 Rod Lattice	2.46	water	borated aluminum	197	1.0002	0.9944	0.0022
19	UO2 Rod Lattice	2.46	water	borated aluminum	634	1.0002	0.9956	0.0020
20	UO2 Rod Lattice	2.46	water	borated aluminum	320	1.0003	0.9893	0.0022
21	UO2 Rod Lattice	2.46	water	borated aluminum	72	0.9997	0.9900	0.0022
22	UO2 Rod Lattice	2.35	water	borated aluminum	0	1.0000	0.9980	0.0024
23	UO2 Rod Lattice	2.35	water	stainless steel	0	1.0000	0.9933	0.0022
24	UO2 Rod Lattice	2.35	water	water	0	1.0000	0.9920	0.0024
25	UO2 Rod Lattice	2.35	water	stainless steel	0	1.0000	0.9877	0.0022
26	UO2 Rod Lattice	2.35	water	borated aluminum	0	1.0000	0.9912	0.0022
27	UO2 Rod Lattice	2.35	water	B4C	0	1.0000	0.9921	0.0021
28	UO2 Rod Lattice	4.31	water	stainless steel	0	1.0000	0.9968	0.0023
29	UO2 Rod Lattice	4.31	water	water	0	1.0000	0.9963	0.0025
30	UO2 Rod Lattice	4.31	water	stainless steel	0	1.0000	0.9950	0.0026
31	UO2 Rod Lattice	4.31	water	borated aluminum	0	1.0000	0.9952	0.0025
32	UO2 Rod Lattice	4.31	water	borated aluminum	0	1.0000	1.0006	0.0024
33	U-metal Cylinders	93.2	bare	air	0	1.0000	0.9968	0.0023
34	U-metal Cylinders	93.2	bare	air	0	1.0000	1.0082	0.0025
35	U-metal Cylinders	93.2	bare	air	0	1.0000	0.99[illegible]5	0.0024
36	U-metal Cylinders	93.2	bare	air	0	1.0000	0.9982	0.0028
37	U-metal Cylinders	93.2	bare	air	0	1.0000	0.9916	0.0025
38	U-metal Cylinders	93.2	bare	air	0	1.0000	0.9922	0.0025
39	U-metal Cylinders	93.2	bare	plexiglass	0	1.0000	0.9972	0.0025
40	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	0.9973	0.00[illegible]9
41	U-metal Cylinders	93.2	bare	plexiglass	0	1.0000	1.0019	0.0027
42	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	1.0103	0.0025
43	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	1.0021	0.0026
44	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	1.0022	0.0029

Note:
(a) See References 24, 25, 26, 27, and 28

CHAPTER 9

AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

New fuel is stored in a high density rack which includes integral neutron absorbing material to maintain the required degree of subcriticality. The rack is designed to store fuel of the maximum design basis enrichment. The rack in the new fuel pit consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. This rack module is not anchored to the pit floor.

The new fuel rack includes storage locations for 72 fuel assemblies. The rack layout and array center-to-center spacing is shown in Figure 9.1-1. This spacing provides a minimum separation between adjacent fuel assemblies which is sufficient to maintain a subcritical array even in the event the building is flooded with unborated water or fire extinguishant aerosols or during any design basis event. The design of the rack is such that a fuel assembly cannot be inserted into a location other than a location designed to receive an assembly. ~~An assembly cannot be inserted into a full location.~~ Surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel.

The requirements of ANS 57.1 are addressed in subsection 9.1.4. The rack is designed to withstand nominal operating loads and safe shutdown earthquake seismic loads defined in Table 9.1-1. The new fuel storage rack is designed to meet seismic Category I requirements of Regulatory Guide 1.29. Refer to subsection 1.9.1 for compliance with Regulatory Guides. The rack is also designed to withstand the maximum uplift force of the fuel handling machine.

AP1000 equipment, seismic and ASME Code classifications are discussed in Section 3.2. The requirements of ASME Code Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analysis. The materials are procured in accordance with ASME Code Section III, Division I, Article NF2000. Criticality analyses are performed in accordance with the requirements of ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 1); and analysis codes are validated against the requirements of ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety (Reference 2).

The stress analysis of the new fuel rack satisfies all of the applicable provisions in NRC Regulatory Guide 1.124, Revision 1 for components design by the linear elastic method (Reference 22).

9.1.1.2 Facilities Description

The new fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The facility is protected from the effects of natural phenomena such as earthquakes, pit environment. Neutron absorbing "poison" material used in the rack design has been qualified for the storage environment. Venting of the neutron absorbing material is considered in the detailed design of the storage rack.

The new fuel assemblies are stored dry. The rack structure is designed to maintain a safe geometric array for normal and postulated accident conditions. The fixed neutron absorbing "poison" material maintains the required degree of subcriticality for normal and postulated accident conditions such as flooding with pure water and low density optimum moderator "misting."

A discussion of the methodology used in the new fuel rack criticality analysis is provided in APP-GW-GLR-030 (Reference 17). ~~subsection 4.3.2.6.~~

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. Each rack in the spent fuel pool consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. These rack modules are free-standing, neither anchored to the pool floor nor braced to the pool wall. The spent fuel storage racks include storage locations for 884 fuel assemblies and five defective fuel assemblies. The Region 1 spent fuel rack layout is shown in Figure 9.1-2. The Region 2 spent fuel rack layout is shown in Figure 9.1-3. The overall spent fuel pool rack layout is presented in Figure 9.1-4. All spent fuel racks will be in place whenever fuel is stored in the spent fuel racks. See DCD subsection 3.7.5.2, for discussion of site-specific procedures for activities following an earthquake. An activity will be to address measurement of the post-seismic event gaps between spent fuel racks and to take appropriate corrective actions.

The design of the racks is such that a fuel assembly cannot be inserted into a location other than a location designed to receive an assembly. ~~An assembly cannot be inserted into a full location.~~

AP1000 equipment, seismic and ASME Code classifications are discussed in Section 3.2. The requirements of ASME Section III, Division I, Article NF3000 are used as the criteria for evaluation of stress analyses. The materials are procured in accordance with ASME Section III, Division I, Article NF2000. Criticality analyses are performed in accordance with the requirements of ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Reference 1); analysis codes are validated against the requirements of ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety (Reference 2); and overall requirements for fuel storage are in accordance with ANSI N210-76, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations (Reference 3).

The stress analysis of the spent fuel racks satisfies all of the applicable provisions in NRC Regulatory Guide 1.124, Revision 1 for components designed by the linear elastic analysis method (Reference 22).

The spent fuel pool is designed to preclude inadvertent draining of the water from the pool.

E. Failure of the Fuel Handling Machine

The fuel handling machine is a seismic Category II component. The fuel handling machine is evaluated to show that it does not collapse into the spent fuel pool as a result of a seismic event.

F. Internally Generated Missiles

The spent fuel handling area does not contain any credible sources of internally generated missiles.

Stress analyses are performed by the vendor using loads developed by the dynamic analysis. Stresses are calculated at critical sections of the rack and compared to acceptance criteria referenced in ASME Section III, Division I, Article NF3000.

9.1.2.3 Safety Evaluation

The design and safety evaluation of the spent fuel racks is in accordance with Reference 5. The racks, being Equipment Class D and seismic Category I structures, are designed to withstand normal and postulated dead loads, live loads, loads resulting from thermal effects, and loads caused by the safe shutdown earthquake event.

The design of the racks is such that K_{eff} remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Inadvertent insertion of a fuel assembly between the rack periphery and the pool wall or placement of a fuel assembly across the top of a fuel rack is considered a postulated accident, and as such, realistic initial conditions such as boron in the pool water are assumed. These accident conditions have an acceptable K_{eff} of less than 0.95. The criticality evaluation, which meets the requirements of 10 CFR 50.68, Paragraph b (Reference 21), considers the inherent neutron absorbing effect of the materials of construction, including fixed neutron absorbing "poison" material. Soluble boron in the spent fuel pool, plutonium decay time, integral fuel burnable absorber, and assembly burnup are used as reactivity credits.

The racks are also designed with adequate energy absorption capabilities to withstand the impact of a dropped fuel assembly from the maximum lift height of the fuel handling machine. Handling equipment (cask handling crane) capable of carrying loads heavier than fuel components is prevented by design from carrying loads over the fuel storage area. The fuel storage racks can withstand an uplift force greater than or equal to the uplift capability of the fuel handling machine (5000 pounds).

Materials used in rack construction are compatible with the storage pool environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pool environment. Neutron absorbing "poison" material used in the rack design has been qualified for the storage environment. Venting of the neutron absorbing material is considered in the detailed design of the storage racks.

Design of the spent fuel storage facility is in accordance with Regulatory Guide 1.13. A discussion of the methodology used in the criticality analysis is provided in ~~subsection 4.3.2.6.~~
spent fuel pool APP-GW-GLR-029 (Reference 20)

6. ANS 57.1-1992, Design Requirements for Light Water Reactor Fuel Handling Systems.

7. Specifications for Electric Overhead Travelling Cranes CMAA, Specification 70 - 2000.

8. USNRC, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980.

9. "Overhead and Gantry Cranes," ANSI/ASME B30.2-1990.

10. Not used.

11. USNRC, "Single-Failure-Proof Cranes for Nuclear Power Plants," NUREG-0554, May 1979.

12. "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," ASME NOG-1-1998.

13. Not used.

14. "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More," ANSI N14.6-1993.

15. "Slings," ASME/ANSI B30.9-1996.

16. APP-GW-GLR-026, "New Fuel Storage Rack Structural/Seismic Analysis," Westinghouse Electric Company LLC.

17. APP-GW-GLR-030, "New Fuel Storage Rack Criticality Analysis," Westinghouse Electric Company LLC.

18. APP-GW-GLR-033, "Spent Fuel Storage Rack Structural/Seismic Analysis," Westinghouse Electric Company LLC.

19. APP-GW-GLR-045, "Evaluation of Critical Structures," Westinghouse Electric Company LLC.

20. APP-GW-GLR-029, "Spent Fuel Storage Racks Criticality Analysis," Westinghouse Electric Company LLC (Westinghouse Proprietary)

21. USNRC, 10 CFR 50.68, "Criticality Accident Requirements," January 2003.

22. USNRC, Regulatory Guide 1.124, Revision 1, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports," January 1978.

23. USNRC, Regulatory Issue Summary 2005-25, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," May 2007.

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."

c. A nominal 10.93 ~~10.90~~ inch center-to-center distance between fuel assemblies placed in Region 1, a nominal 9.04 ~~9.028~~ inch center-to-center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks, and a nominal 11.62 inch center-to-center distance between fuel assemblies placed in the Defective Fuel Cells.

d. New or partially spent fuel assemblies with any discharge burnup may be allowed unrestricted storage in Region 1 and the Defective Fuel Cells of Figure 4.3-1;

e. Partially spent fuel assemblies meeting the initial enrichment, burnup, and decay time requirements of LCO 3.7.12, "Spent Fuel Pool Storage," may be stored in Region 2 of Figure 4.3-1, and

f. New and spent fuel assemblies meeting the Figure 4.3-2 location-specific initial enrichment, burnup, and decay time requirements of LCO 3.7.12, "Spent Fuel Pool Storage," may be stored in specified Region 2 locations.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam which includes an allowance for uncertainties as described in Section 9.1, "Fuel Storage and Handling."

d. A nominal 10.90 inch center-to-center distance between fuel assemblies placed in the new fuel storage racks.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.
REFERENCES	1. AP1000 Design Control Document, Rev. 15, Sections 9.1.2, "Spent Fuel Storage" and 15.7.4, "Fuel Handling Accident." 2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 3. APP-GW-GLR-029, "AP1000 Spent Fuel Storage Racks ~~Critically~~ Criticality Analysis," June 2006.

BASES

BACKGROUND (continued)

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. ~~For example, the only accident scenario that has an inadvertent misplacement of a new fuel assembly.~~ Insert #2 This accident has the potential for more than negligible positive reactivity effect is a potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment, burnup and decay time of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.12.1.

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.15, "Fuel Storage Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.995, assuming the pool to be flooded with unborated water and < 0.95, with a boron concentration of greater than 758 ppm.

"All Cell" Storage Configuration

The "All Cell" storage configuration permits storage in all Region 2 locations of spent fuel which meets the combination of initial enrichment, burnup and decay time requirements shown in LCO Figure 3.7.12-1, Fuel Assembly Burnup Requirements for the Region 2, "All Cell" Storage

<u>Insert #2</u>

<u>Insert #2</u> <u>(pg. B 3.7.12-2, Background (cont'd), *copied from B 3.7.11, Background:*</u>
Clarify accident described in Line 4-5 by replacing with sentences below.

"For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison."

BASES

LCO (continued)

Configuration. Figure 3.7.12-1 permits new (no burnup) 1.627 weight percent U-235 fuel to be stored in the All Cell configuration. "1-out-of-4 5.0 weight-percent fresh" Storage Configuration Fuel stored in accordance with the "1-out-of-4 5.0 weight-percent fresh" storage configuration shall be stored in the relative locations shown in Figure 4.3-2. The "1-out-of-4 5.0 weight-percent fresh" storage configuration permits storage of 5.0 weight percent U-235 new (no burnup) fuel or any spent fuel in one specified location, provided fuel stored in the three remaining locations meets the enrichment, burnup and decay time requirements shown in LCO Figure 3.7.12-2, Fuel Assembly Burnup Requirements for the Region 2 "1-out-of-4 5.0 w/o Fresh" Storage Configuration. Figure 3.7.12-2 permits new (no burnup) 1.361 weight percent U-235 fuel to be stored in the three remaining locations. The 4-location configuration may be repeated throughout Region 2.

Interface Requirements

Fuel may be stored in both the "All Cell" and "1-out-of-4 5.0 weight-percent fresh" configurations at the same time, provided fuel stored in the interface locations around the "1-out-of-4 5.0 weight-percent fresh" configuration group(s) meets the LCO Figure 3.7.12-1 requirements. Fuel assemblies not meeting the criteria of Figures 3.7.12-1 and 3.7.12-2 shall be stored in accordance with Specification 4.3.1.1.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of this fuel storage pool.

ACTIONS

LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool storage requirements apply in all MODES when fuel is stored in Region 2 ~~[illegible]~~, the ACTIONS have been modified by a Note stating the LCO 3.0.3 is not applicable. Spent fuel pool storage requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool storage requirements apply in all MODES when fuel is stored in Region 2 ~~[illegible]~~, the ACTIONS have been modified by a Note stating the LCO 3.0.8 is not applicable. Spent fuel pool storage requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

BASES

ACTIONS (continued)

A.1

The LCO is not met if spent fuel assemblies stored in Region 2 "All Cell," "1-out-of-4 5.0 weight-percent fresh" or interface spent fuel assembly storage locations do not meet the applicable initial enrichment, burnup and decay time limits in accordance with Figure 3.7.12-1 or 3.7.12-2.

Additionally, LCO is not met if fuel, required to be stored in the New Fuel location of the "1-out-of-4 5.0 weight-percent fresh" storage configuration, is misplaced. When the LCO is not met, action must be initiated immediately to make the necessary fuel assembly movement(s) in Region 2 to bring the storage configuration into compliance with Figures 3.7.12-1 and 3.7.12-2 or to move fuel to Region 1 or the defective fuel cells.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies by administrative means that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.12-1 or 3.7.12-2 as applicable for "All Cell," "1-out-of-4 5.0 weight-percent fresh" and interface spent fuel assembly storage locations. Fuel stored in Region 2 that does not meet the Figure 3.7.12-1 or 3.7.12-2 limits shall be stored in Figure 4.3-1 "1-out-of-4 5.0 weight-percent fresh" New Fuel location or the Defective Fuel Cells.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. APP-GW-GLR-029, "AP1000 Spent Fuel Storage Racks Criticality Analysis," June 2006.
3. AP1000 Design Control Document, Rev. 15, Sections 9.12, "Spent Fuel Storage" and 15.7.4, "Fuel Handling Accident."