Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555

Direct tel: 412-374-6306
Direct fax: 412-374-5456
e-mail: sterdia@westinghouse.com

Your ref: Project Number 740
Our ref: DCP/NRC1739

May 29, 2006

Subject: AP1000 COL Standard Design Change Submittal

In support of Combined License application pre-application activities, Westinghouse has been preparing and reviewing standard changes to the Design Control Document. These changes are intended to be incorporated into FSARs referencing the AP1000 design certification or incorporated into the design certification using supplemental rulemaking when 10 CFR Part 52 is revised to permit such changes.

Criteria in 10 CFR Part 52 Appendix D Section VIII B. 5. a, b, and c provide for changes in Tier 2 of the Design Control Document that do not require prior NRC approval. The changes included in this submitted report satisfy these criteria. The changes are generic and are expected to apply to all projects referencing the AP1000 Design Certification. This information is submitted as part of the NuStart Bellefonte COL Project (NRC Project Number 740).

Pursuant to 10 CFR 50.30(b), APP-GW-GLR-030, Rev. 0, "Criticality Analysis for New Fuel Rack," (Technical Report Number 67), is submitted as Enclosure 1 under the attached Oath of Affirmation.

The reviews of these changes were included in a table of COL technical reports in a March 8, 2006 letter from NuStart to the NRC.

Questions or requests for additional information related to the content and preparation of these reports 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.

Very truly yours,

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

D079

3775alf.doc

A BNFL Group company

/Attachment

1. "Oath of Affirmation," dated May 30, 2006

/Enclosure

1. APP-GW-GLR-030, Rev. 0, "Criticality Analysis for New Fuel Rack," (Technical Report Number 67)

cc:	S. Bloom	-	U.S. NRC	1E
	G. Curtis	-	TVA	1E
	L. Dudes	-	U.S. NRC	1E
	P. Grendys	-	Westinghouse	1E
	P. Hastings	-	Duke Power	1E
	C. Ionescu	-	Progress Energy	1E
	D. Lindgren	-	Westinghouse	1E
	A. Monroe	-	SCANA	1E
	C. Pierce	-	Southern Company	1E
	E. Schmiech	-	Westinghouse	1E
	G. Zinke	-	NuStart/Entergy	1E

ENCLOSURE 1

APP-GW-GLR-030, Rev. 0

"Criticality Analysis for New Fuel Rack"

Technical Report Number 67

ATTACHMENT 1

Oath of Affirmation

DCP/NRC1739
May 29, 2006

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

Daniel S. Lipman, being duly sworn, states that he is Senior Vice President 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.

Daniel S. Lipman
Senior Vice President

Subscribed and sworn to
before me this 30th day
of May 2006.

Lorraine M. Piplica
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2007
Member, Pennsylvania Association Of Notaries

AP1000 DOCUMENT COVER SHEET

TDC: ________ Permanent File: ________ APY: ________

RFS#: ________ RFS ITEM #: ________

AP1000 DOCUMENT NO.	REVISION NO.		ASSIGNED TO
APP-GW-GLR-030	0	Page 1 of 1	W-A. Sterdis

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: Westinghouse Electric Company

TITLE: **New Fuel Storage Rack Criticality Analysis**

Reviewed [illegible]

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE: APP-FS01-N1C-001, Rev. 0	APP-GW-GEE-073, Rev. 0

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AP1000 RESPONSIBLE MANAGER	SIGNATURE*	APPROVAL DATE
K. Quinn	D.A. Wiseman for K. Quinn	5/25/06

* 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-030
Revision 0

May 2006

AP1000 Standard Combined License Technical Report

New Fuel Storage Rack Criticality Analysis

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

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1 INTRODUCTION

This report summarizes the criticality analysis for the AP1000 New Fuel Storage Rack. The AP1000 New Fuel Storage Rack is used to temporarily store fresh fuel assemblies until they are loaded into the reactor core. The requirements for this analysis are identified in the AP1000 Design Control Document (DCD), subsections 4.3.2.6 and 9.1.1.3 (Reference 1). The completion of this analysis is identified as Combined Operating License (COL) Information Item 9.1-2 (Final Safety Evaluation Report [Reference 2] Action Item 9.1.6-2) in DCD subsection 9.1.6 to be completed by the Combined License applicant:

> COL Information Item 9.1-2: "Perform a confirmatory criticality analysis for the new fuel rack, as described in AP1000 DCD Subsection 9.1.1.3. This analysis should address the degradation of integral neutron absorbing material; in the new fuel pool storage racks as identified in GL-96-04, and assess the integral neutron absorbing material capability to maintain a 5-percent subcriticality margin."

U. S. Nuclear Regulatory Commission (NRC) Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Storage Racks," is not relevant to the Westinghouse AP1000 New Fuel Storage Rack for several reasons. First, Metamic is used as a neutron absorber not Boraflex. The AP1000 New Fuel Storage Rack is in a mild environment – a dry concrete storage vault with no exposure to hot borated water as in the spent fuel pool. The AP1000 New Fuel Storage Rack is not subject to gamma irradiation because it is a temporary storage location for fresh fuel before it is placed in the reactor core. In summary, the mechanisms described in Generic Letter 96-04 that cause degradation of the neutron absorber material are not present in the AP1000 New Fuel Storage Rack.

Associated with the closure of this COL information item, the description of the criticality analysis as discussed in DCD Section 4.3 is updated and the references are also updated.

This COLA technical report closes this COL information item. The calculation "AP1000 New Fuel Storage Rack Criticality Analysis" (Reference 3) is available for NRC audit. The structural/seismic analysis of the AP1000 New Fuel Storage Rack is presented in AP1000 COLA Technical Report, APP-GW-GLR-026, "New Fuel Storage Rack Structural/Seismic Analysis" (Reference 4).

2 TECHNICAL BACKGROUND

This analysis demonstrates that the maximum K-effective (K_{eff}) value, including all biases and uncertainties, for the AP1000 New Fuel Storage Rack (filled with 5.0 weight-percent U-235 Westinghouse AP1000 17x17 Fuel Assemblies) is less than 0.95 assuming full density unborated water. In addition, this analysis demonstrates that the maximum K_{eff} value, including all biases and uncertainties, for optimum moderation conditions is less than 0.98.

2.1 DESIGN CRITERIA

The criticality design criteria for the AP1000 New Fuel Storage Rack are as follows:

1. The maximum K_{eff} value, including all biases and uncertainties, shall be less than 0.95 with full density unborated water. Note this design criterion is provided in 10 CFR Part 50.68, Item 2 of Paragraph B (Reference 5). Note the specific terminology is:

   > "The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used."

2. The maximum K_{eff} value, including all biases and uncertainties, shall be less than 0.98 with optimum moderation and full reflection conditions. Note this design criterion is provided in 10 CFR Part 50.68, Item 3 of Paragraph B (Reference 5). The specific terminology is:

   > "If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used."

3. 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 10CFR Part 50.68, Item 7 of Paragraph B (Reference 5). The specific terminology is:

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

2.2 METHODOLOGY AND ASSUMPTIONS

2.2.1 Methodology

This section describes the methodology used to ensure the criticality safety of the AP1000 New Fuel Storage Rack and to define limits placed on fresh fuel assembly storage configurations. The analysis methodology employs SCALE-PC, a personal computer version of the SCALE-4.4a code system, as documented in Reference 6, with the updated SCALE-4.4a version of the 44 group Evaluated Nuclear Data File, Version 5 (ENDF/B-V) neutron cross-section library.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a.

The SCALE system was developed for the NRC to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on personal computers.

Validation of SCALE-PC for fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs: the Babcock & Wilcox (B&W) experiments in support of Close Proximity Storage of Power Reactor Fuel (Reference 7) and the Pacific Northwest Laboratory Program in support of the design of Fuel Shipping and Storage Configurations. References 8 and 9, as well as several of the relevant thermal experiment evaluations in Reference 10, were useful in updating pertinent experimental data for the Pacific Northwest Laboratory experiments. The validation of SCALE-PC was limited to the 44 group library provided with the SCALE-PC version 4.4a package (Reference 11).

The approach used for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 12 and Reference 13. For a given KENO-calculated value of K_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed. By this definition, there is a 95-percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

2.2.2 Assumptions

The assumptions for the AP1000 New Fuel Storage Rack are as follows:

- The Westinghouse AP1000 17x17 Fuel Assembly was modeled as the design basis fuel assembly with an enrichment equal to 5.0 weight-percent U-235.

- Fresh fuel assemblies were conservatively modeled with a UO_2 density of 10.686 g/cm^3 (97.5% of theoretical density). This translates into a pellet density equal 98.6% of theoretical density with a 1.1% dishing (void) fraction.

- All fresh fuel assemblies were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs that incorporate lower enrichment blanket or annular pellets.

- All fuel assemblies were conservatively modeled as containing no burnable absorber material.

- Storage cell material and Metamic poison material were modeled with a length of 168 inches. The storage cell material and Metamic poison above and below the active fuel length was conservatively omitted.

2.3 DESIGN INPUT

This section provides a brief description of the AP1000 New Fuel Storage Rack and Vault with the objective of establishing a basis for the analytical models used in the criticality analyses described in Section 2.4.

2.3.1 17x17 Fuel Assembly Description

The fuel assembly modeled in this analysis is the Westinghouse AP1000 17x17 Fuel Assembly present in Figure 2-1 in plan view. The bottom elevation of the Metamic poison panel (172 inches long) conservatively covers the active fuel length (168 inches) with a 4-inch overlap (2 inches overlap on each end of the active fuel).

2.3.2 Design Input from Holtec

Design data related to the AP1000 New Fuel Storage Rack were obtained from Holtec. In addition, Holtec supplied Westinghouse a "PWR Rack Data Sheet," which provides detailed New Fuel Storage Rack design information (Reference 14).

2.3.3 AP1000 New Fuel Storage Rack and Vault Description

Westinghouse General Arrangement and Concrete Outline Drawings were used to determine the New Fuel Storage Vault geometry (References 15 and 16). A plan view of the AP1000 New Fuel Storage Rack is shown in Figure 2-2.

The AP1000 New Fuel Storage Rack is located inside a concrete room (vault) in the AP1000 Auxiliary Building. The AP1000 New Fuel Storage Rack is centered inside the vault and is an 8x9 array of storage cells, which provides 72 total storage locations. A hatch lid is provided for the vault for security, and for Foreign Material Exclusion (FME).

2.3.4 Individual Storage Cell Description

The data items related to the individual storage cell description are summarized in Table 2-1.

The individual storage cells of the AP1000 New Fuel Storage Rack are centered on a nominal pitch of 10.9 inches. Each storage cell consists of an inner stainless steel box, which has a nominal inside dimension of 8.8 inches and is 0.075 inches thick. Metamic panels are attached to the outside surfaces of all storage cells except for the outside cell walls directly facing the North and South walls of the vault. No poison is required on these outside cell faces since there is just a small amount of space between the rack and storage vault concrete. However, poison is required on the outside cell faces in the East and West directions (see Figure 2-2) to mitigate the effects of an inadvertent placement of a fuel assembly outside of the rack, but within the vault on these two sides. Each Metamic poison panel is held in place and is centered on the surface of the stainless steel box by an outer stainless steel sheathing panel. There is a small void space between the sheathing and the Metamic panel. The dimensions of the Metamic poison panel are 7.5 inches in width by 0.106 inches in thickness. The sheathing panels are 0.035 inches in thickness.

Each storage cell is nominally 193.25 inches long, and it rests on top of a base plate whose top is 5 inches above the concrete floor. Note that each Metamic poison panel is 172 inches long overlapping the 168-inch active fuel length. The Metamic poison material is a mixture of B_4C nominally 31.0 weight-percent and aluminum 69.0 weight-percent.

2.4 ANALYSIS

2.4.1 KENO Model and Assumptions

The KENO V.a model is a three-dimensional representation of the AP1000 New Fuel Storage Rack and Vault. The 17x17 fuel assemblies are explicitly modeled as shown in Figure 2-1. Note that the entire length of the active fuel (168 inches) is fully enriched with 5 weight-percent U-235.

The 8x9 array of storage cells is modeled with the active fuel of a 17x17 fuel assembly in each location. The 8x9 array of storage cells is centered with the four walls of the concrete vault. The top of the vault is conservatively modeled without a lid. This is a conservative omission because a metal lid would absorb neutrons. Also included in the criticality model is the reflection provided by a postulated 12 inches of full density water.

The fuel rod, guide tube, and instrumentation tube claddings are modeled with Zircaloy in this analysis. This is conservative with respect to the Westinghouse ZIRLO™ product, which is a Zirconium alloy containing additional elements including Niobium. Niobium has a small absorption cross section, which causes more neutron capture in the cladding regions resulting in a lower reactivity. Therefore, this criticality analysis is conservative with respect to fuel assemblies containing ZIRLO cladding in fuel rods, guide tubes, and the instrumentation tube.

There are no burnable absorbers modeled in any of the fuel assemblies. The Zirconium grid straps are conservatively omitted. The 8x9 array of storage cells and vault are modeled at room temperature conditions (20°C), and the system reactivity is evaluated at 11 moderator densities ranging from 1.0 g/cc down to 0.001 g/cc. A total of 1.2 million neutron histories are modeled in 1003 generations with 1200 neutrons per generation. Note that all KENO V.a results for the first three neutron generations are skipped to eliminate preliminary estimates of the system reactivity. The methodology bias and uncertainty are discussed and evaluated in Reference 12. The results of these KENO calculations showing final K_{eff} values (including bias and uncertainties) versus water density are given in Table 2-2.

2.4.2 Hypothetical Fuel Assembly Drop and Impact on Criticality Analysis

It is possible to drop a fresh fuel assembly into or on top of a storage cell in the AP1000 New Fuel Storage Rack as described in subsection 9.1.1.2.1 C of the DCD (Reference 1). In the event that the dropped fuel assembly hits the top of a storage cell, the analyses in subsection 2.8.5 of Reference 4 indicate that both the Metamic and active fuel are not adversely impacted. Therefore, there is no degradation of the criticality safety margin as a result of dropping a fuel assembly on top of a storage cell.

The resulting deformation on the base plate following a drop of fuel assembly straight through an empty cell impacting the rack baseplate is discussed in subsection 2.8.5 of Reference 4. To conservatively bound the deformation results for the base plate, the bottom elevations of 25 fuel assemblies were lowered by 5 inches. (Note that the base plate is 3/4 inches thick and is normally 4.25 inches above the floor.) This is a five-by-five array of fuel assemblies centered on the empty cell impacted by the dropped fuel assembly (refer to Figure 2-10 of Reference 4). Even with the bottom elevation of the active fuel in 25 fuel assemblies lowered by 5 inches, the criticality design limits given in Section 2.1 are still met. This conclusion is based upon the observation that the AP1000 New Fuel Storage Rack is normally dry and not flooded with water. Note that for this hypothetical dropped fuel assembly accident that the AP1000 New Fuel Storage Rack would need to contain at least several feet of water in the bottom of the AP1000 New Fuel Storage Rack before the criticality design basis limits would be exceeded. Requiring two independent accident conditions is beyond the requirements of the double contingency principle. Therefore, the hypothetical dropped fuel assembly through an empty cell scenario does meet the criticality design basis limits.

2.5 RESULTS

Figure 2-2 displays the final K_{eff} values versus water density for the AP1000 New Fuel Storage Rack. The K_{eff} values in Figure 2-2 include all biases and uncertainties.

The maximum fresh fuel enrichment limit for the AP1000 New Fuel Storage Racks is determined to be 5.0 weight-percent U-235 since the final K_{eff} values at this enrichment are less than 0.98 at optimum moderation conditions and less than 0.95 at fully flooded conditions, assuming no soluble boron. Therefore, the AP1000 New Fuel Storage Rack fully loaded with Westinghouse AP1000 17x17 Fuel Assemblies with an enrichment less than or equal to 5.0 weight-percent U-235 satisfies both criticality safety criteria specified in 10 CFR Part 50.68.

Table 2-1 AP1000 New Fuel Storage Rack Storage Cell Description

Parameter	Nominal Dimension (in.) or Material
Cell Pitch	10.9
Cell ID	8.8
Cell Length	193.25
Cell Wall Thickness	0.075
Cell Wall Material	SS-304
Metamic Width	7.5
Metamic Thickness	0.106
Metamic Composition	B_4C/Al
Sheathing Thickness	0.035
Sheathing Material	SS-304

Table 2-2 Final K_{eff} Versus Moderator Density for AP1000 New Fuel Storage Rack (with Bias and Uncertainties)

Water Density [g/cm^3]	K_{eff}
1.00	0.92050
0.90	0.88621
0.80	0.85444
0.70	0.81435
0.60	0.77478
0.50	0.73891
0.40	0.69399
0.30	0.65108
0.20	0.60728
0.10	0.56951
0.001	0.53243

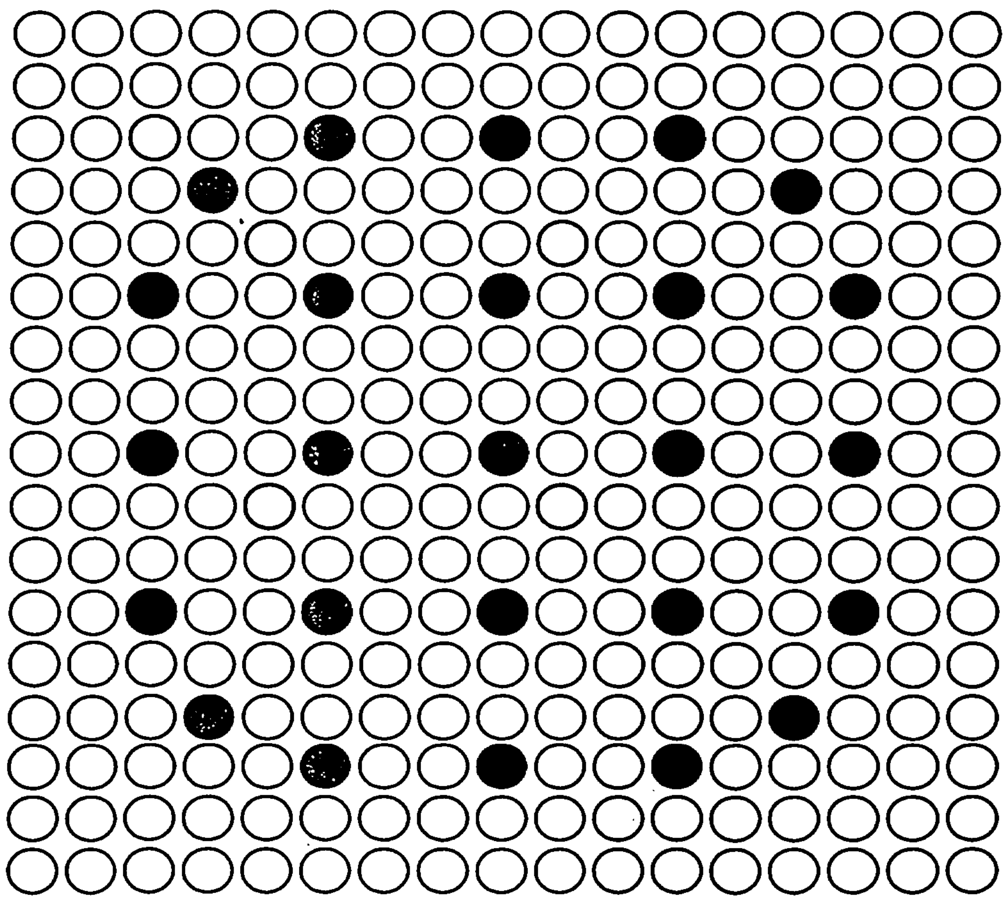


Figure 2-1 Westinghouse AP1000 17x17 Fuel Assembly (Black Circles Indicate Guide Tube Locations)

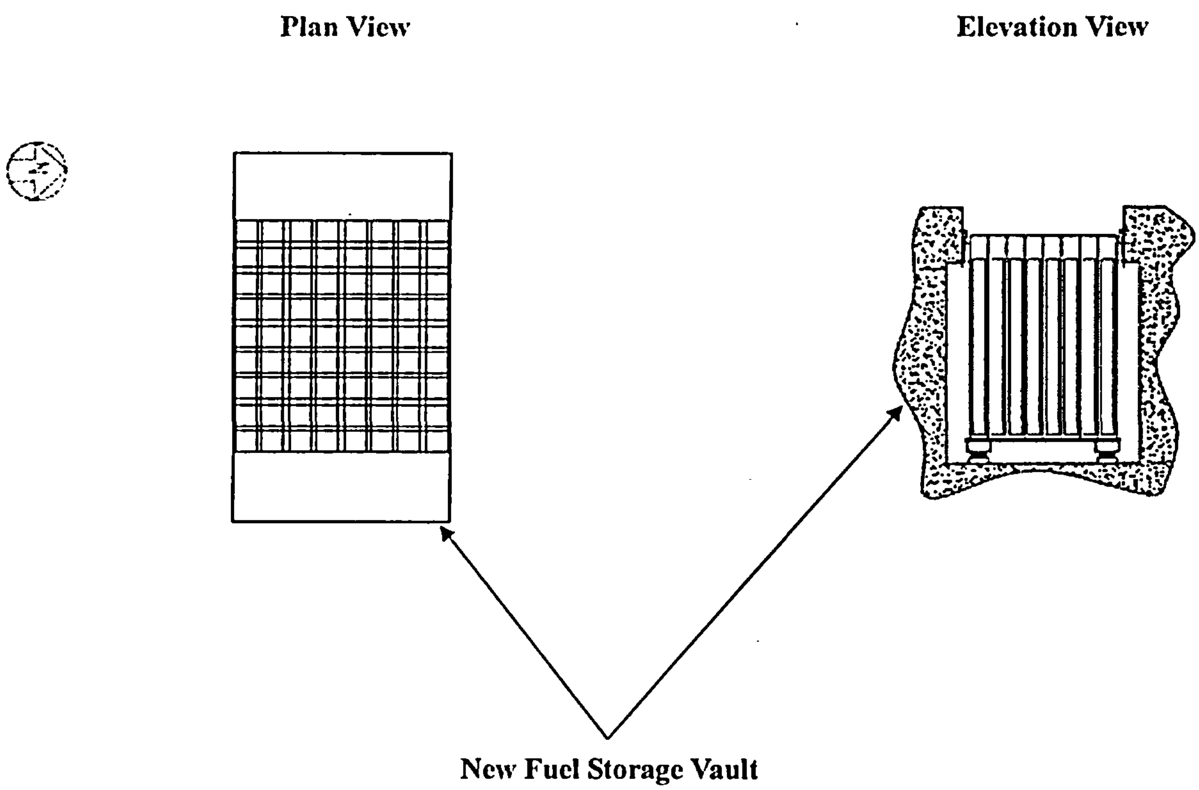


Figure 2-2 Plan and Elevation View AP1000 New Fuel Storage Rack and Vault (Vault Lid Details not Shown)

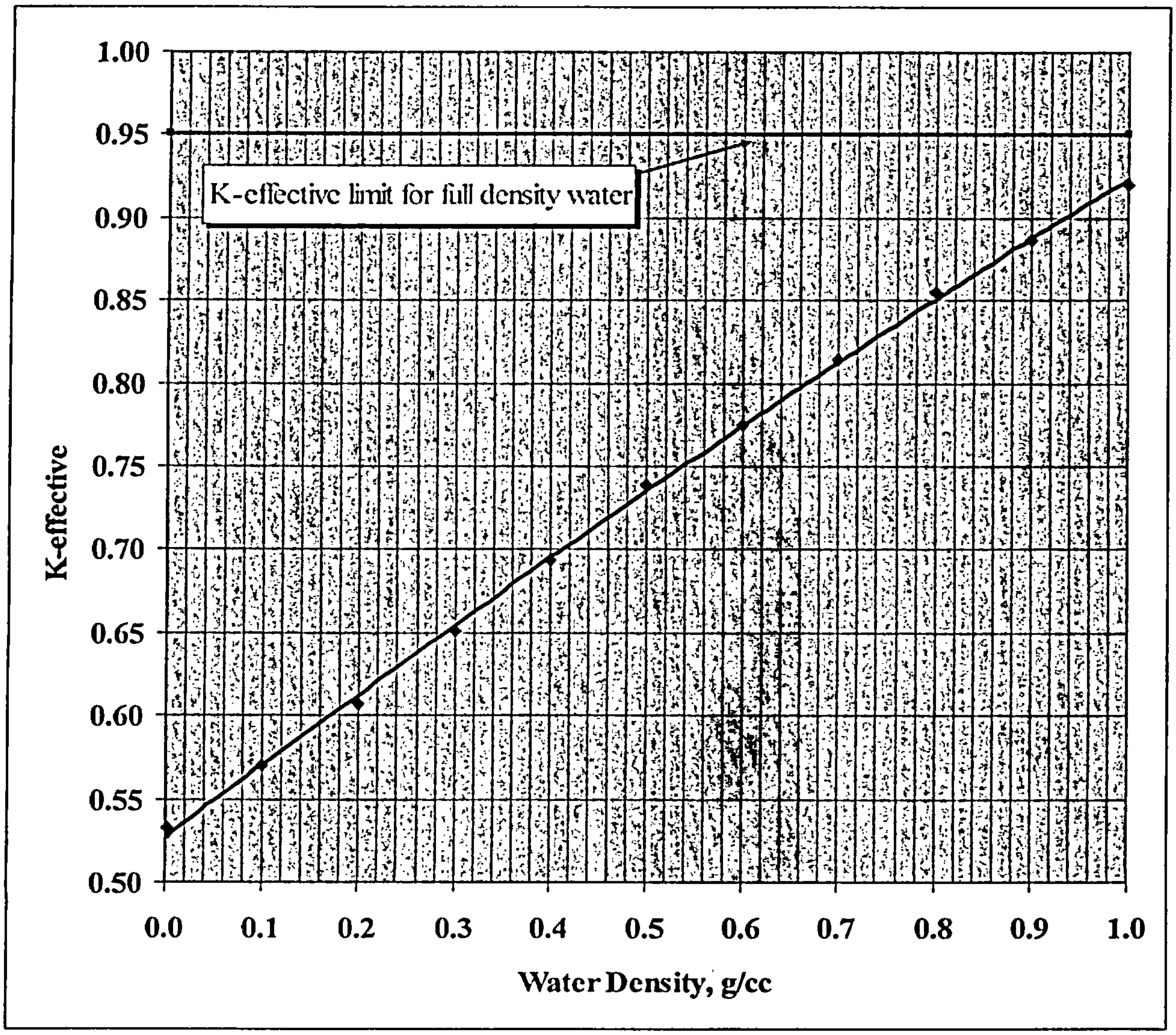


Figure 2-3 K_{eff} as a Function of Water Density in AP1000 New Fuel Storage Rack (All Biases and Uncertainties Included)

3 REGULATORY IMPACT

The design of the AP1000 New Fuel Storage Rack is addressed in subsection 9.1.1, "New Fuel Storage," of the NRC Final Safety Analysis Report (Final Safety Evaluation Report, Reference 2). The completion of the criticality analysis for the AP1000 New Fuel Storage Rack is identified in the Final Safety Evaluation Report as COL Action Item 9.1.6-2.

The changes to the DCD presented in this report do not represent an adverse change to the design functions of the AP1000 New Fuel Storage Rack, or to how design functions are performed or controlled. The criticality analysis of the new fuel rack is consistent with the description of the analysis in subsection 4.3.2.6.1, "Criticality Design Methods Outside the Reactor," and the description of the design presented in subsection 9.1.1, "New Fuel Storage," of the DCD. Therefore, the changes to the DCD do not involve revising or replacing a DCD-described evaluation methodology. The changes to the DCD do not involve a test or experiment not described in the DCD. The updating of the discussion of the criticality analysis in Subsection 4.3.2.6.1 and associated reference changes primarily is due to providing additional information and refinement and does not represent a departure from an evaluation methodology in the DCD used in establishing design bases. The DCD change does not require a license amendment per the criteria of VIII. B. 5.b. of Appendix D to 10 CFR Part 52.

The changes described do not involve design features used to mitigate severe accidents. Therefore, a license amendment based on the criteria of VIII. B. 5.c of Appendix D to 10 CFR Part 52 is not required.

The closure of the COL Information Item will not alter barriers or alarms that control access to protected areas of the plant. The closure of the COL Information Item will not alter requirements for security personnel. Therefore, the closure of the COL Information Item does not have an adverse impact on the security assessment of the AP1000.

4 REFERENCES

1. APP-GW-GL-700, AP1000 Design Control Document, Rev. 15.

2. Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, U. S. Nuclear Regulatory Commission, September 2004.

3. Westinghouse Calculation, APP-FS01-N1C-001 Rev. 0, "AP1000 New Fuel Storage Rack Criticality Analysis," April 2006. (Westinghouse Proprietary)

4. AP1000 Standard Combined License Technical Report, APP-GW-GLR-026, Rev. 0, "New Fuel Storage Rack Structural/Seismic Analysis," May 2006.

5. Code of Federal Regulations, Title 10, Part 50.68, "Criticality Accident Requirements."

6. "SCALE 4.4a- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC CODE PACKAGE CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000.

7. M. N. Baldwin et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report," BAW-1484-7, July 1979.

8. S. R. Bierman and E. D. Clayton, "Critical Experiments with Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6," NUREG/CR-1547, PNL-3314, July 1980.

9. S. R. Bierman and E. D. Clayton, "Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 Wt% U-Enriched UO_2 Rods in Water with Steel Reflecting Walls," Nuclear Technology, Vol. 54, pg. 131, August 1981.

10. International Handbook of Evaluated Criticality Safety Benchmark Experiments, Nuclear Energy Agency and Organization for Economic Cooperation and Development.

11. Westinghouse Calculation CN-CRIT-206, "Unix SCALE 4.4 Criticality Validation and Benchmarking Analysis," M. G. Anness, P. F. O'Donnell, and M. A. Cunningham, 2004. (Westinghouse Proprietary)

12. W. Marshall et al., "Criticality Safety Criteria," TANS Vol. 35, pg. 278, 1980.

13. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," SCR-607, Sandia Corporation Monograph, March 1963.

14. Holtec International Letter Number 1540001.doc, "PWR Rack Data Sheet," Rev. 4, April 27, 2006. (Holtec Proprietary)

15. Drawing APP-1200-CC-959, Rev. A, "Auxiliary Building Section K Areas 5&6." (Westinghouse Proprietary)

16. Drawing APP-1240-CC-564, Rev. A, "Auxiliary Building Concrete Floor EL. 117'-6" Areas 5&6." (Westinghouse Proprietary)

5 DCD MARKUP

The following DCD markup identifies how COL application Final Safety Analysis Reports should be prepared to incorporate the subject change.

Revise subsection 4.3.2.6.1 starting with the third paragraph as follows:

> The design method which determines the criticality safety of fuel assemblies outside the reactor uses the SCALE system~~,~~ **(Reference 21),** ~~Rev. 4,~~which includes the BONAMI and NITAWL-II codes for cross sections generation and the KENO-V.a code for reactivity determination.

The 238~~18~~ groups library obtained from ENDF/B-~~I~~V data is the origin of the 44~~27~~ groups library used in these analyses and in the modeling of the critical experiments which are the basis for the qualification of the SCALE/KENO-V.a (Reference 26~~1~~) calculation system.

A set of 30~~41~~ critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials and enrichments, all of them adequate for qualifying methods to analyze light water reactor lattices (References 22 to 25, 276).

The analysis of the 30~~41~~ critical experiments results in an average K_{eff} of 0.9969~~38~~. Comparison with the measured values results in a method bias of 0.00316~~2~~. The standard deviation of the set of reactivities is 0.00285~~396~~. The 95/95 tolerance factor is 2.22~~118~~.

The total uncertainty (TU) to be added to the criticality calculations:

$$TU = M_{95/95}\left[(ks)^2_{method} + (ks)^2_{KENO} + \sum_i (ks)^2_{mech}\right]^{1/2}$$

where:

$(ks)_{method}$ =method uncertainty as discussed above.

$(ks)_{KENO}$ = the statistical uncertainty associated with the particular KENO calculation being used.

$(ks)_{mech}$ = a series of statistical uncertainties associated with mechanical tolerances, such as thicknesses and spacings. If worst-case assumptions are used for tolerances, this term will be zero.

$M_{95/95}$ = 95/95 tolerance factor, 2.22

The criticality design criteria are met when the calculated effective multiplication factor plus the total uncertainty and biases are ~~is~~ less than 0.95 or, in the special case defined above, 0.98.

The analytical methods employed herein conform with ANSI N18.2 (Reference 3), Section 5.7, Fuel Handling System; ANSI N16.9 (Reference 29), ANSI 57.2 (Reference 19), subsection 6.4.2, ANSI 57.3 (Reference 20), Section 6.2.4; NRC Standard Review Plan, subsection 9.1.2, the NRC guidance, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 30).

Revise References 21 and 26 of subsection 4.3.5 as follows:

4.3.5 References

21. **"SCALE 4.4a- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC CODE PACKAGE CCC-545, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2000**~~NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Oak Ridge National Laboratory; Rev. 4, January 1990~~.

26. **M. G. Anness, P. F. O'Donnell, and M. A. Cunningham, "Unix SCALE 4.4 Criticality Validation and Benchmarking Analysis," CN-CRIT-206, 2004.**~~NEDO-32028 "MCNP Light Water Reactor Critical Benchmarks," S. Sitaraman, March 1992.~~

Revise the third paragraph of subsection 9.1.1.3 as follows:

9.1.1.3 Safety Evaluation

The new fuel storage racks are purchased equipment. The purchase specification for the new fuel storage racks ~~will~~ requires ~~the vendor to perform~~ a criticality analysis of the new fuel storage racks. The criticality evaluation will consider the inherent neutron absorbing effect of the materials of construction, including fixed neutron absorbing "poison" material.

Revise the second paragraph of subsection 9.1.6 as follows:

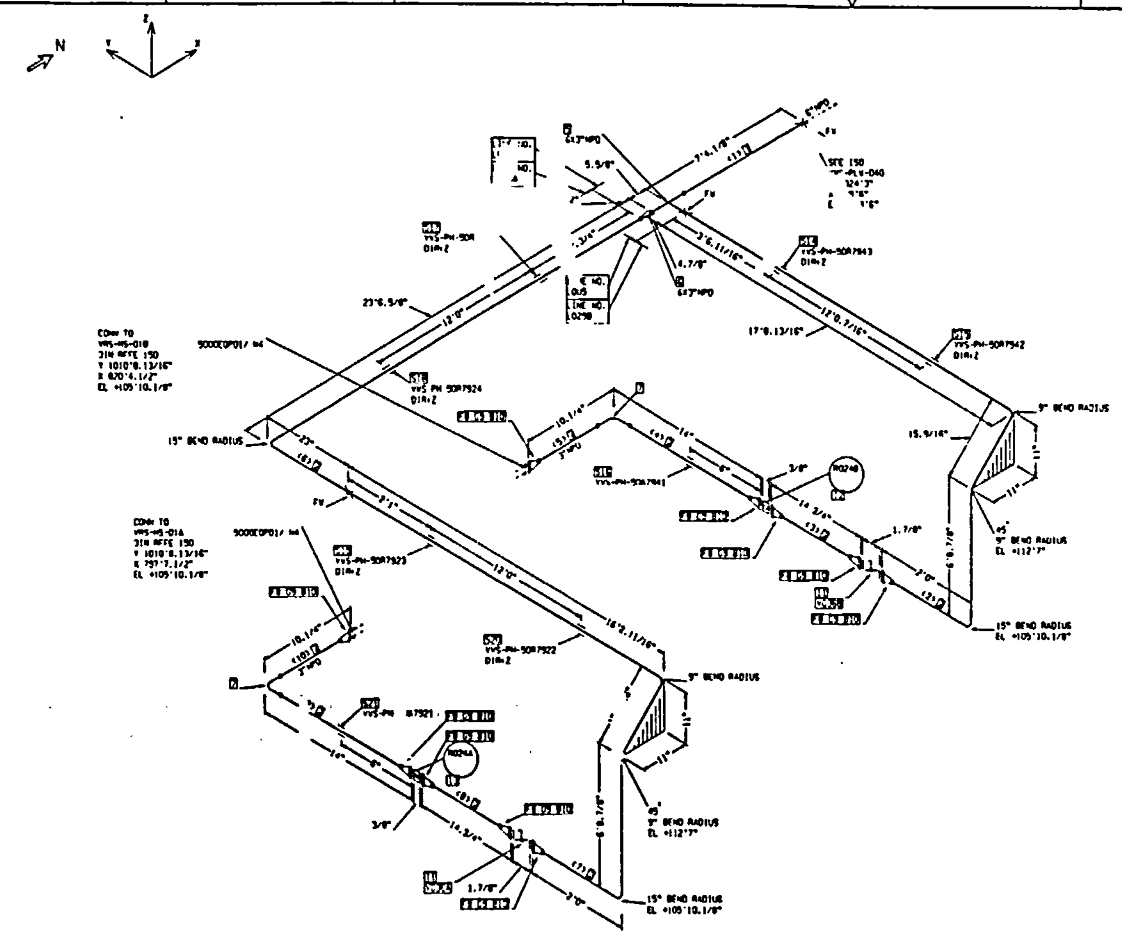
9.1.6 Combined License Information for Fuel Storage and Handling

Complete. ~~The Combined License applicant is responsible for a~~ A confirmatory criticality analysis for the new fuel rack, as described in subsection 9.1.1.3 ~~,~~ **is provided in APP-GW-GLR-030 Rev. 0, AP1000 Standard Combined License Technical Report (Reference 16).** This ~~analysis should~~ **report** addresses the degradation of integral neutron absorbing material in the new fuel pool storage racks as identified in GL-96-04, and assesses the integral neutron absorbing material capability to maintain a 5-percent subcriticality margin.

Revise subsection 9.1.7 as follows:

9.1.7 References

16. **APP-GW-GLR-030 Rev. 0 "AP1000 Standard Combined License Technical Report New Fuel Storage Rack Criticality Analysis", Westinghouse Electric Company LLC, May 2006.**



- SHOP MATERIALS -

PT NO	DESCRIPTION	(IN)	CMDTY CODE	QTY
1	PIPE, S-40, BE, ASTM-A106-B	6	PAAZZAIAAE	6.9'
2	PIPE, S-40, BE, ASTM-A106-B	3	PAAZZAIAAE	70.5'
3	PIPE BEND, S-40, BE, 3-NPD RAD. ASTM-A106-B	3	PAAZZAVAAE	3.6'
4	PIPE BEND, S-40, BE, 5-NPD RAD. ASTM-A106-B	3	PAIZZAVAAE	5.9'
	REDUCING BRANCH TEE, S-40 X S-40, BE, ASTM-A234-WPB, ANSI-B16.9	6X3	VTBAAAVAAA	1
	CONCENTRIC REDUCER, S-40 X S-40 BORE, BE, ASTM-A234-WPB, ANSI-B16.9	6X3	VRAAAAVAAA	1
	90 DEG LR ELBOW, S-40, BE, ASTM-A234-WPB, ANSI-B16.9	3	VAAAAAVAAA	2
	FLANGE, CL150, RFFE/BE, ASTM-A105, ANSI-B16.5, WN, S-40 BORE	:	FAAABAVAAA	10
	FLANGE GASKET, ANSI-B16.5 CL150, BORE TO MATCH, FLEXITALIC STYLE CG, OR APPROVED EQUAL		KDAABAVZZC	10
	ASTM-A193-B7 STUDS W/ A194-2H HEX NUTS, 3.75" BOLT LENGTH	5/8	YZZZHZZFFI	40
	BUTTERFLY VALVE, CL150, RAISED FACE FLANGED END, GEAR OPERATED W/SIDE MOUNTED HANDWHEEL, MATL. PER SPEC.	3	VFABA3OFGP	2

- OTHER THAN SHOP MATERIALS -

PT NO	DESCRIPTION	(IN)	CMDTY CODE	QTY
	INSTRUMENTS			
12	ORIFICE PLATE RO24B	3		1
13	ORIFICE PLATE RO24A	3		1
	PIPE SUPPORTS			
14	RIGID SUPPORT	3	VWS-PH-50R7943	1
15	RIGID SUPPORT	3	VWS-PH-50R7942	1
16	ANCHOR SUPPORT	3	VWS-PH-50A7941	1
17	RIGID SUPPORT	3	VWS-PH-50R7940	1
18	RIGID SUPPORT	3	VWS-PH-50R7924	1
19	RIGID SUPPORT	3	VWS-PH-50R7923	1
20	RIGID SUPPORT	3	VWS-PH-50R7922	1
21	ANCHOR SUPPORT	3	VWS-PH-50A7921	1

- CUT PIPE LENGTH -

	LENGTH (FT-INS)	(IN)	REMARKS		LENGTH (FT-INS)	(IN)	REMARKS
<1>	8'10.1/2"	6	FF WELD(1)	<2>	26' 3/16"	3	FLD BEND(3) FF WELD(1)
<3>	9.1/4"	3		<4>	6"	3	
<5>	."	3		<6>	24' .6"	3	FLD BEND(1) FF WELD(1)
<7>	1.11/16"	3	FLD BEND(3)	<8>	; •	3	
<9>	5.3/4"	3		<10>		3	

LIMITED RIGHTS LEGEND

REFERENCE

PIPING	5000PVWS01
EQUIPMENT	5000EQP01
VALVES	
P&ID	VWS M6 005, REV.0
PENETRATION	

LINE NO	SIZE	CLASS	INSULATION
VWS-PL-L005	6"	JCE	1.00
VWS-PL-L029A	3"	JCE	1.00
VWS-PL-L029B	3"	JCE	1.00

Westinghouse Electric Corporation

AP600

ADVANCED PASSIVE LIGHT WATER REACTOR

BURNS AND ROE COMPANY

5460

CENTRAL CHILLED WATER SYSTEM
PIPING ISO, RADWASTE BUILDING
RETURN PIPING - ELEC/MECH EQUIP RM

D VWS-PLW-060 0

524910

BRC/POK0883