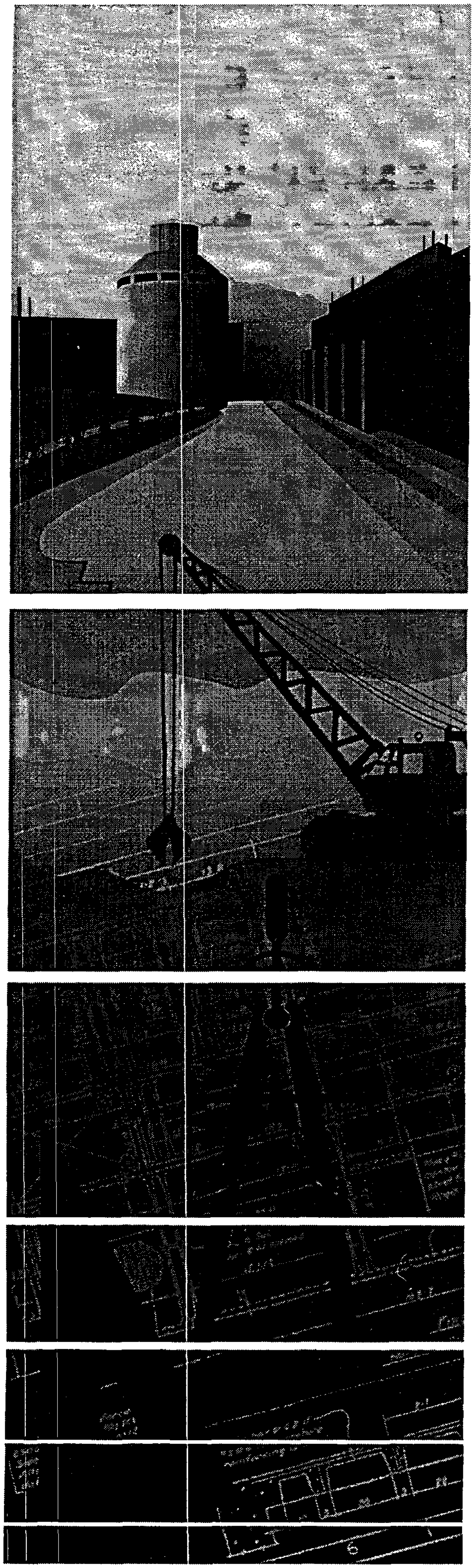

Final Safety Evaluation Report

Related to Certification of the
AP1000 Standard Design

Supplement 1

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

December 2005

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design
Docket No. 52-006

Supplement 1

Division of New Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report supplements the final safety evaluation report (FSER) for the AP1000 standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1793 in September 2004 to document the NRC staff's technical review of the AP1000 design. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westinghouse) in accordance with Subpart B, "Standard Design Certifications," of Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52), and Appendix O, "Standardization of Design: Staff Review of Standard Designs." This supplement documents the NRC staff's review of Westinghouse's changes to the AP1000 design documentation in the design control document (DCD) since the issuance of the FSER. On the basis of the evaluation described in the AP1000 FSER (NUREG-1793) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 15 to the AP1000 DCD) are acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

TABLE OF CONTENTS

Table

1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This report supplements the final safety evaluation report (FSER) for the AP1000 standard plant design. The U.S. Nuclear Regulatory Commission (NRC) staff issued the FSER as NUREG-1793 in September 2004 to document the NRC staff's review of the AP1000. This supplement documents the NRC staff's review of the changes to the AP1000 design documentation since the issuance of the FSER. Each section of this supplement is numbered and titled the same as the section of the FSER that is being updated. The discussions are supplementary to, but not in lieu of, the discussions in the FSER, unless otherwise noted.

Westinghouse Electric Company, LLC (Westinghouse or the applicant) submitted the AP1000 design documentation under Subpart B of "Standard Design Certifications," of Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52). The AP1000 design documentation includes the AP1000 design control document (DCD) and probabilistic risk assessment. Changes to the AP1000 DCD (Docket No. 52-006) were submitted after the FSER was issued. The staff's review of these DCD changes is discussed in Section 1.5 of this report.

This supplement is issued by the Division of New Reactor Licensing in the Office of Nuclear Reactor Regulation, NRC. The NRC's project managers for this part of the AP1000 design certification review are Lauren M. Quinones-Navarro and Jerry N. Wilson, PE. They may be reached by calling 301-415-2007 or 301-415-3145, or by writing to them at the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. The AP1000 design documentation and all revisions are available for public inspection at the NRC's Public Document Room and the NRC's Public Electronic Reading Room (ADAMS).[1] The NRC's Public Electronic Reading Room is at http://www.nrc.gov/reading-rm/adams/web-based.html. Through this Web site, the public can gain access to ADAMS, which provides text and image files of NRC's public documents. The AP1000 FSER (NUREG-1793) and this supplement are also available for public inspection at the NRC's Public Document Room and Electronic Reading Room.

[1]ADAMS (Agencywide Documents Access and Management System) is the NRC's information system that provides access to all image and text documents that the NRC has made public since November 1, 1999, as well as bibliographic records (some with abstracts and full text) that the NRC made public before November 1999. Documents available to the public may be accessed via the Internet at http://www.nrc.gov/reading-rm/adams/web-based.html. Documents may also be viewed by visiting the NRC's Public Document Room at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. Telephone assistance for using web-based ADAMS is available at (800) 397-4209 between 8:30 a.m. and 4:15 p.m., eastern standard time, Monday through Friday, except Federal holidays.

1.5 Summary of Principal Review Matters

By letter dated September 7, 2005 (DCP/NRC 1722), Westinghouse submitted proposed Tier 1 AP1000 DCD changes. The NRC staff reviewed the proposed changes and provided comments. Westinghouse submitted revised changes to the AP1000 DCD by letters dated November 1, 2005 (DCP/NRC 1723) and November 3, 2005 (DCP/NRC 1724), addressing the staff's comments. Westinghouse incorporated the changes in Revision 15 of the DCD and the errata to Revision 15 of the DCD submitted on November 14, 2005 (DCP/NRC 1725) and December 8, 2005 (DCP/NRC 1727), respectively. The changes to the AP1000 DCD include editorial and minor technical changes and clarifications to the inspection, tests, analyses, and acceptance criteria (ITAAC) in the Tier 1 information. Westinghouse identified these changes as a result of the Nuclear Energy Institute (NEI) and NuStart efforts to prepare for future combined license (COL) applications. These changes are listed in Table 1.5-1.

The NRC staff reviewed the changes, which were made to resolve inconsistencies between Tier 1 and Tier 2 information in the AP1000 DCD, to correct administrative errors, and to make editorial clarifications. The staff determined that most of these changes do not affect the staff's findings in the AP1000 FSER and are acceptable. For some of the changes, a supplemental evaluation is provided. Supplemental evaluations are set forth in Sections 5.4.1.2, 14.3, and 15.2.3.1 of this supplement.

Table 1.5-1 Summary of Changes to the AP1000 DCD

TIER 1 SECTION	DESCRIPTION
Section 1.1	Add the definition of "Tag Number" to the list of definitions.
Section 1.2	Clarify the definition of "A report exists and concludes that...".
Section 1.4	Delete the unnecessary line under "Safe Shutdown" in the "Safe-Shutdown Earthquake" definition on page 1.4-5 in the Tier 1 "List of Acronyms and Abbreviations."
Table 2.1.1-1	Provide a precise reference for an ITAAC item.
Subsection 2.1.2 Tables 2.1.2-1 through 2.1.2-4 and Figures 2.1.2-1 and 2.1.2-2	Correct several items in Tier 1, Tables 2.1.2-1 through 2.1.2-3. In Tier 1, Table 2.1.2-4, make an editorial correction in item 3.a, provide a precise reference for ITAAC item 7.c, modify ITAAC item 8.b to verify the pump coastdown curve rather than the calculated pump inertia, and correct the pressurizer heater backup rated capacity in item 9.b.
Table 2.1.2-1, and Tier 2 Table 3.2-3, Table 3.11-1, Subsections 5.4.3.2.1, 5.4.3.2.3, 7.2.1.1.3, Table 7.2-2, Subsections 9A.3.1.1.7, 9A.3.1.1.8, Table 9A-2, Subsections 14.2.9.1.1, 14.2.10.1.17, 14.2.10.4.11, Table 15.0-4a, Subsections 15.3.1.1, 15.3.2.1, Table 15.3-1, Figures 15.3.1-2 through 15.3.1-6, Figures 15.3.3-3 through 15.3.3-7 and Technical Specification (Section 16.1) Table 3.3.1, Subsection 3.4.1 and Bases Subsections 3.3.1 and 3.4.1, Tier 2 Figure 5.1-3, Figure 5.1-5 (Sheet 1 of 3), and Figure 7.2-1 (Sheet 5 of 20)	This change deletes the reactor coolant system (RCS) flow velocity probe and replaces it with a flow signal derived from the hot-leg elbow. References in Tier 2 text and tables to the cold-leg flow signals are also changed to hot-leg signals to be consistent with Tier 1 changes. Chapter 15 figures related to analysis of partial loss of forced reactor coolant flow and the locked rotor event are also revised. The RCS loop layout, the RCS piping and instrument diagram (P&ID), and the protection system functional diagram of the flow instruments are revised.

Subsection 2.1.3; Tables 2.1.3-1, 2.1.3-2, 2.1.3-3, 2.1.3-4; and Figure 2.1.3-3, and Tier 2 Tables 1.3-1 and 5.3-5; Subsections 5.3.1.2 and 5.3.4.1; and Figures 5.3-1 and 5.3-6	Correct various tag numbers in the tables and make minor editorial corrections. Provide a precise reference for an ITAAC item in Tier 1, Table 2.1.3-2.
Tables 2.2.1-1, 2.2.1-2, and 2.2.1-3; and Figure 2.2.1-1	Change the tag numbers and the closure time for the containment purge isolation valves. Also provide a precise reference for ITAAC item 6.c in Tier 1, Table 2.2.1-3.
Subsection 2.2.2; Tables 2.2.2-1, 2.2.2-2, and 2.2.2-3; and Figure 2.2.2-1	Provide clarifications and corrections to Tier 1, Subsection 2.2.2.
Tables 2.2.3-1 through 2.2.3-4 and Table 2.2.3-6; and Figure 2.2.3-1	Provide precise references for ITAAC items in the tables. Correct Tier 1, Figure 2.2.3-1 for the passive core cooling system.
Tables 2.2.4-1, 2.2.4-2, 2.2.4-4, and 2.2.4-5; and Figure 2.2.4-1	Revise certain tables in Tier 1, Subsection 2.2.4 to indicate that all main steam line steam generator pressure sensors are qualified for a harsh environment and to make a few other editorial corrections. Provide precise references for ITAAC items 7.c and 8.c in Tier 1, Table 2.2.4-4.
Subsection 2.2.5, Table 2.2.5-5, and Figure 2.2.5-1	Correct certain tag numbers for equipment in the main control room emergency habitability system and make a few additional editorial changes. Provide a precise reference for ITAAC item 6.b in Table 2.2.5-5.
Table 2.3.1-2	Provide precise references for ITAAC Item 2 in Tier 1, Table 2.3.1-2.

Subsection 2.3.2, and Tables 2.3.2-1, 2.3.2-2, and 2.3.2-4	Correct a typographical error in the numbering of item 2 in Tier 1, Subsection 2.3.2, "Design Description." In Table 2.3.2-1, valve CVS-PL-V092 is erroneously shown as being in a harsh environment. This valve is outside containment and is not in a harsh environment. Revise this valve entry in the table accordingly. In Table 2.3.2-2, delete the pipeline L047 entry for the chemical and volume control system (CVS) letdown containment penetration line. This was erroneously included in place of L051. Add an entry for the piping to the pressurizer auxiliary spray connection. This was erroneously omitted from Table 2.3.2-2. Expand Table 2.3.2-2 to include seismic analysis of all CVS piping inside containment greater than 1-inch nominal diameter and normally exposed to RCS pressure. Add Note 1 to the table to indicate the extent to which seismic analysis of lines with normally closed valves is required. Also, in Table 2.3.2-2, correct several minor typographical errors. In Table 2.3.2-4, provide precise references for ITAAC items. Correct a typographical error in item 12.b.
Table 2.3.3-2	Correct the value for the volume of the ancillary diesel generator fuel tank.
Table 2.3.4-2 and 2.3.4-4; and Tier 2 Figure 9.5.1-1	Provide precise references for ITAAC item 3 in Tier 1, Table 2.3.4-2. Revise the list of fire protection system piping that must remain functional following a safe-shutdown earthquake. Revise Tier 2, Figure 9.5.1-1 to show line numbers.
Subsection 2.3.6; Tables 2.3.6-1, 2.3.6-2, and 2.3.6-4; and Figure 2.3.6-1 and Tier 2 Table 5.4-14	Update and/or correct normal residual heat removal system (RNS) information in Tier 1, Subsection 2.3.6.
Subsection 2.3.7, and Tables 2.3.7-1, 2.3.7-2, 2.3.7-3, and 2.3.7-4	Correct and clarify Tier 1 information for the spent fuel pool cooling system. Also provide precise references for ITAAC items.
Figure 2.3.8-1	Correct an editorial error in Tier 1, Figure 2.3.8-1.
Tables 2.3.10-2 and 2.3.10-4	Provide precise references for ITAAC item 6.a) in Tier 1, Table 2.3.10-4. Revise Table 2.3.10-4 item 7.a) and Table 2.3.10-2 to correct omissions.

Subsection 2.3.13 and Table 2.3.13-3	Make an editorial correction in Tier 1, Table 2.3.13-3 to indicate that there is only one check valve in the primary sampling system. Provide precise references for ITAAC items 6.c) and 7 in Tier 1, Table 2.3.13-3.
Table 2.3.14-2	Provide precise references for ITAAC item 2 in Tier 1, Table 2.3.14-2.
Table 2.3.15-2	Provide precise references for ITAAC item 2 in Tier 1, Table 2.3.15-2.
Table 2.5.2-8	Provide a precise reference for an ITAAC item.
Subsection 2.5.3	Revise the descriptions of the plant control system to clarify that the AP1000 plant control system uses software signal selection rather than the hardware signal selection used by the AP600.
Subsection 2.5.4 and Table 2.5.4-2	Delete the following function of the data display and processing system (DDS): "The DDS provides non-safety-related displays of parameters originating in other systems."
Table 2.5.5-2	Provide a precise reference for an ITAAC item.
Table 2.6.1-4	Provide precise references for ITAAC items.
Table 2.6.3-3	Provide a precise reference for an ITAAC item.
Table 2.6.5-1	Provide a precise reference for an ITAAC item.
Table 2.7.1-4 and Table 2.7.1-5	Provide a precise reference for an ITAAC item in Tier 1, Table 2.7.1-4. In addition, correct certain "Component Name" entries and a "Tag No." entry in Tier 1, Table 2.7.1-5.
Table 2.7.2-2	Provide precise references for ITAAC item 2 in Tier 1, Table 2.7.2-2.
Figure 2.7.2-1	Correct certain tag numbers in Tier 1 Figure 2.7.2-1, sheets 1 and 2 of 2.
Figure 2.7.4-1	Correct certain tag numbers in Tier 1 Figure 2.7.4-1, sheets 1 and 2 of 2.
Table 2.7.5-3	Revise the "Component Location" for certain components in Tier 1, Table 2.7.5-3.
Table 2.7.6-2	Provide precise references for ITAAC item 2 in Tier 1, Table 2.3.1-2. Correct the units of a parameter from "cfm" to "scfm."

Table 2.7.7-1	Revise the "Equipment Name" entries, a "Tag No." entry, and the "Display" entries in Tier 1, Table 2.7.7-1.
Table 3.2-1	Change the acceptance criteria to identify individual criteria that could be completed in phases to facilitate completion and review of the parts of the human factors engineering program.
Subsection 3.3 and Tables 3.3-1 and 3.3-6	Revise Tier 1 Table 3.3-1 to correct and update the dimensions of the concrete walls. Dimensions of the concrete walls for the radwaste building were relocated to Table 3.3-6.
Tables 3.5-1, 3.5-7, and 3.6-1	For Tier 1, Table 3.5-1, correct certain "Tag No." entries and "Safety-Related Display" entries. For Tier 1, Table 3.5-7, correct one "Component Location" entry. Delete references to the specific technology used for containment atmosphere radioactivity from the ITAAC for reactor coolant pressure boundary leak detection. Also provide precise references for the ITAAC electrical separation item.
Table 5.0-1	Correct the description of the "Tornado Missile Spectra" in Table 5.0-1 of Tier 1.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.4.1.2 Coast-Down Capability

In the last paragraph of Section 5.4.1.2 of the final safety evaluation report (FSER), the U.S. Nuclear Regulatory Commission (NRC) staff stated that the acceptance criterion of the reactor coolant pump (RCP) rotating inertia for flow coast-down capability specified in item 8.b of Tier 1, Table 2.1.2-4, is no less than 695.3 kg-m^2 (16,500 lb-ft^2). Westinghouse proposed to change the acceptance criterion to a specific pump coast-down curve. This is done by (1) adding Figure 2.1.2-2, "Flow Transient for Four Cold Legs in Operation, Four Pump Coastdown," to Tier 1, Section 2.1.2; (2) changing the item 8.b entry of the "Inspections, Tests, Analyses" column to read: "A test will be performed to determine the pump flow coast-down curve"; and (3) changing the item 8.b entry of the "Acceptance Criteria" column to read: "The pump flow coast-down will provide RCS flows greater than or equal to the flow shown in Figure 2.1.2-2, 'Flow Transient for Four Cold Legs in Operation, Four Pump Coastdown.' "

Westinghouse proposed this change because the specified pump rotating inertia does not guarantee that flows produced during pump coastdown will satisfy the analyses in the absence of information on pump resistance. The proposed Figure 2.1.2-2 is a four-pump coast-down curve identical to DCD Tier 2, Figure 15.3.2-1, in the safety analysis of the design-basis transient of a complete loss of forced reactor coolant flow described in DCD Tier 2, Subsection 15.3.2. The use of Tier 1, Figure 2.1.2-2, as the inspection, tests, analyses and acceptance criteria (ITAAC) for the RCP flow coast-down capability ensures consistency with the safety analysis and, therefore, the NRC staff finds such use acceptable.

Because of Westinghouse's change to the Tier 1 ITAAC acceptance criteria of the RCP pump coastdown capability, the NRC is revising the last paragraph of Section 5.4.1.2 on page 5-57 of NUREG-1793 to read as follows:

> In DCD Tier 1, Table 2.1.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," Item 8b specifies a test will be performed to determine the pump coastdown curve that demonstrates the pump flow coastdown will provide RCS flows greater than or equal to the flow shown in Tier 1 Figure 2.1.2-2, "Flow Transient for Four Cold Legs in Operation, Four Pump Coastdown." Since the four-pump coast-down curve in Tier 1, Figure 2.1.2-2 is identical to the four pump coastdown curve in DCD Tier 2 Figure 15.3.2-1, which is used in the safety analysis of the design basis transient of a complete loss of forced reactor coolant flow described in DCD Tier 2, Subsection 15.3.2, and based on the above evaluation, the staff finds the RCP coastdown capability curve acceptable as an acceptance criterion for RCP capability.

14. VERIFICATION PROGRAMS

14.3 Tier 1 Information

In Section 14.3 of the final safety evaluation report, the NRC staff provided its evaluation of the AP1000 inspections, tests, analyses, and acceptance criteria (ITAAC) in Tier 1 of the AP1000 design control document. In its submittals, dated November 1 and 3, 2005, Westinghouse proposed many corrections and clarifications to the AP1000 ITAAC, including some revisions to the Tier 1 Introduction. The staff reviewed the proposed changes, performed an audit of the AP1000 structural dimension changes as described below, and compared the revised structural dimensions with those in the Tier 2 information. With the exception of the changes to wall dimensions described below, the corrections and clarifications were not significant, and did not affect the staff conclusions set forth in NUREG-1793.

In one of the Tier 1 changes, Westinghouse proposed a number of revisions to wall dimensions (wall heights, wall lengths, and wall thicknesses) in Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings and Annex Buildings," of the Tier 1 information. According to Westinghouse, when Table 3.3-1 was initially developed from Tier 2 drawings, some wall dimensions were not directly transferred. The purpose of this revision was to make the wall dimensions in Table 3.3-1 consistent with those in the Tier 2 drawings.

The staff's review of Westinghouse's submittal raised the concern that if the revised Tier 1 wall dimensions are not consistent with those of Tier 2 information, which were used for developing the seismic model of the nuclear island buildings, the seismic responses calculated for the design may be significantly affected. During conference calls, Westinghouse stated that the seismic model was developed based on Tier 2 drawings and the revised Table 3.3-1 is now consistent with those Tier 2 drawings. To verify Westinghouse's explanation, the staff conducted an audit at Westinghouse's Rockville office on October 28, 2005, and confirmed that the wall dimensions of the revised Table 3.3-1 are consistent with those in the Tier 2 design drawings. On this basis, the staff concludes that the revised wall dimensions will not have any effect on the seismic responses (design loads), and therefore, the changes to Tier 1 Table 3.3-1 are acceptable.

Based on the NRC staff's review of the AP1000 Tier 1 information in accordance with the guidance in SRP Section 14.3 (as described in Section 14.3 of the FSER) and its recent review of Westinghouse's proposed corrections and clarifications to its Tier 1 information, described above, the staff concludes that the top-level design features and performance characteristics of the AP1000 design are appropriately described in Tier 1 and the Tier 1 information is acceptable. Further, the Tier 1 design descriptions can be adequately verified by ITAAC. Therefore, the staff concludes that the ITAAC are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility referencing the certified design can be constructed and operated in conformity with the design certification and the applicable regulations.

15. TRANSIENT AND ACCIDENT ANALYSES

15.2.3.1 Partial Loss of Forced Reactor Coolant Flow (DCD Tier 2, Section 15.3.1)

The first paragraph of Section 15.2.3.1 of the FSER states:

> A mechanical or electrical failure in an RCP [reactor coolant pump], or a fault in the power supply to the pumps supplied by an RCP bus, may cause partial loss of RCS [reactor coolant system] flow, a moderate-frequency event. The low primary coolant flow reactor trip signal in any reactor coolant loop provides protection against this event.

Westinghouse proposed to change the RCS flow sensors from cold-leg velocity probes to hot-leg elbow taps. The change document provided by Westinghouse states that experience with cold-leg velocity probes has shown that there may be noise problems and signal-to-noise-ratio problems with this type of flow sensor.

Since the AP1000 design has two hot loops and four cold loops, the NRC staff is replacing the first paragraph of Section 15.2.3.1 on page 15-24 of NUREG-1793 with the following to remove any ambiguity in the term "loop:"

> A mechanical or electrical failure in an RCP, or a fault in the power supply to the pumps supplied by an RCP bus, may cause partial loss of RCS flow, a moderate-frequency event. The low primary coolant flow reactor trip signal provides protection against this event.

The deletion of the phrase "in any reactor coolant loop" is to reflect Westinghouse's design change. The change in the location of the sensor will not impair its operation. Therefore, this change does not affect our conclusions in Section 15.2.3.1 of the FSER.

24. CONCLUSION

The Nuclear Regulatory Commission (NRC) staff has reviewed Westinghouse's changes to the AP1000 design documentation (see Section 1.5 of this report). On the basis of the evaluation described in the AP1000 FSER (NUREG-1793) and this report, the NRC staff concludes that the AP1000 design documentation (up to and including Revision 15 to the AP1000 design control document) is acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B, "Standard Design Certifications," of Part 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 52) that are applicable and technically relevant to the AP1000 standard plant design.

NRC FORM 335
(9-2004)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1793, Supplement 1

2. TITLE AND SUBTITLE

Final Safety Evaluation Related to Certification of the AP1000 Standard Design, Supplement 1

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	2005

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Lauren Quinones-Navarro, Jerry N. Wilson, Wesley Held

6. TYPE OF REPORT

Safety Evaluation Report

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of New Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Docket No. 52-006, Project No. 711

11. ABSTRACT *(200 words or less)*

This report supplements the final safety evaluation report (FSER) for the AP1000 standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1793 in September 2004 to document the NRC staff's technical review of the AP1000 design. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westinghouse) in accordance with Subpart B, Standard Design Certifications," of Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part52), and Appendix O, "Standardization of Design: Staff Review of Standard Designs." This supplement documents the NRC staff's review of Westinghouse's changes to the AP1000 design documentation in the design control document (DCD) since the issuance of the FSER. On the basis of the evaluation described in the AP1000 FSER (NUREG-1793) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 15 to the AP1000 DCD) are acceptable and that Westinghouse's application for design certification meets the requirements of Subpart B to 10CFR Part 52 that are applicable and technically relevant to the AP1000 standard plant design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Advanced Reactor
Combined License
Design Certification
Final Design Approval
Inspections, Tests, Analyses, and Acceptance Criteria
Passive Plant Design

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS