Westinghouse Non-Proprietary Class 3

DCP/NRC1532
Docket No. 52-006

November 15, 2002

Attachment 3

"AP1000 Design Certification Review –
Response to Request for Additional Information"

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.005

Question:

RG 1.68, Revision 2, Appendix A, Section 3, "Initial Criticality," first paragraph, 3rd and 4th sentences state:

> A neutron count rate at least ºƒ½ count per second should register on the startup channels before startup begins, and the signal to noise ratio should be known to be greater than two. All systems required for startup or protection of the plant, including the reactor protection system and emergency shutdown system should be operable and in a state of readiness.

RG 1.68, Revision 2, Appendix A, Section 3, first paragraph, 6th sentence, states:

> For reactors that will achieve initial criticality by boron dilution, control rods should be withdrawn before dilution begins. The control rods insertion limits defined in technical specifications should be observed and complied with.

RG 1.68, Revision 2, Appendix A, Section 3, 2nd paragraph, states:

> Criticality predictions for boron concentration (PWR) and control rod positions should be provided and criteria for actions to be taken should be established if actual plant conditions deviate from predicted values. The reactivity addition sequence should be prescribed, and the procedure should require a caution approach in achieving criticality to prevent passing through criticality on a reactor period shorter than approximately 30 seconds (<1 decade per minute).

RG 1.68, Revision 2, Appendix C, "Preparation of Procedures," Section 3, "Initial Criticality Procedures," contains much of the same information noted above. In addition, 1st paragraph, last sentence, and 2nd paragraph, last sentence state:

> Technical Specification requirements must be met.
> High flux scram trips should be set to their lowest value (approximately 5% - 20%).

AP1000 DCD (Tier 2) Section 14.2.10.2, "Initial Criticality Tests," includes four initial criticality tests listed below:

14.2.10.2.1, Initial Criticality and Low-Power Test Sequence
14.2.10.2.2, Initial Criticality
14.2.10.2.3, Nuclear Instrumentation System Verification During Criticality
14.2.10.2.4, Post Critical Reactivity Computer Checkout

a) The NRC staff found that none of the test abstracts contained the information noted above in RG 1.68, Appendix A, Section 3, and Appendix C, "Preparation of Procedures," Section 3. Westinghouse should add this information to DCD (Tier 2) section 14.2.10.2.

b) The NRC staff also found that test abstract 14.2.10.2.1 title should be revised to delete the words "Low-Power Test Sequence" since this is performed in test abstract Section 14.2.10.3, "Low Power Tests." It also appears that test abstract 14.2.10.2.1 should be combined with 14.2.10.2.2.

c) Test abstract 14.2.10.2.2 states:

> As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by rod withdrawal.

RG 1.68, Appendix A, Section 3, states:

> For reactors that will achieve initial criticality by boron dilution, control rods should withdrawn before dilution begins.

The statement in Section 14.2.10.2.2 is contrary to RG1.68. Therefore, the words "or by rod withdrawal" should be deleted.

d) The title to test abstract 14.2.10.2.3 should be revised to delete "During Criticality" since nuclear instrumentation system verification is performed prior to and during initial criticality.

Westinghouse Response:

The following item responses correspond to the RAI question items:

a) The test abstracts in the AP1000 DCD are intend to provide an overview of the tests to be performed on the plant. Detailed test specifications or procedures are beyond the scope of the DCD but will submitted to the NRC for review by the Combined License applicant as identified in DCD subsection 14.4.2. The detailed information requested by the NRC will be provided as part of the Combined License applicant submittal.

b) Westinghouse will revise the title of DCD subsection 14.2.10.2.1 to be "Initial Criticality Test Sequence." The revised title is intended to clarify that 14.2.10.2.1 is an overview of the subsequent subsections of 14.2.10.2. Also with this revised title, subsections 14.2.10.2.1 and 14.2.10.2.2 do not have identical titles and should not be combined.

c) The first bullet in the DCD subsection 14.2.10.2.2, under "Test Method" states:

- "Accomplish initial criticality by the controlled withdrawal of the rods using the same rod withdrawal sequence used for normal plant startup, followed by the dilution of the reactor coolant system boron concentration."

Thus, the rods are withdrawn consistent with RG 1.68.

As identified by the NRC, however, the third bullet under "Test Method" states:

- "As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by rod withdrawal."

Westinghouse notes that although the rods are withdrawn, they may be slightly inserted for control purposes and therefore the third bullet under "Test Method" will be revised as follows:

- "As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by ~~rod~~ withdrawal **of rods that have been slightly inserted for control.**"

d) Westinghouse will revise the title of DCD subsection 14.2.10.2.3 as requested and as shown in the DCD Revision section below.

Design Control Document (DCD) Revision:

The following DCD revisions correspond to the RAI question items:

a) No change to the DCD

b) Revise DCD subsection title 14.2.10.2.1 as follows:

"14.2.10.2.1 Initial Criticality ~~and Low-Power~~ Test Sequence"

c) Revise the third bullet in DCD subsection 14.2.10.2.2, under "Test Method" as follows:

- "As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by ~~rod~~ withdrawal of rods that have been slightly inserted for control."

d) Revise DCD subsection title 14.2.10.2.3 as follows:

"14.2.10.2.3 Nuclear Instrumentation System Verification ~~During Criticality~~"

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.006

Question:

RG 1.68, Revision 2, Appendix A, Section 4, "Low Power Tests," lists 21 low power tests, Items 4a through 4u, that should be conducted if they have not been previously completed during pre-operational hot functional tests. One test, item 4m, can be eliminated from consideration since it is only performed on BWRs. For the AP1000 design, DCD (Tier 2) Section 14.2.10.3, "Low Power Tests," lists only 7 tests.

a) Westinghouse should provide additional information on where the 20 applicable PWR low power tests listed in RG 1.68 are performed and under which part of the initial test program these tests are credited. If any low power tests listed in RG 1.68 are missing and needed during the low power test phase, Westinghouse should either add these tests to DCD (Tier 2) Section 14.2.10.3 or provide appropriate exceptions for not performing these tests at low power.

b) In DCD (Tier 2) Section 14.2.10.4.3, "Nuclear Instrumentation System," Westinghouse includes tests, calibrations, and demonstrates overlap of indication between the source range and intermediate range, during the power ascension test program. However, the overlap is 4×10^3 to 10^6 counts per second on the source range which equals 10^{-11} to 10^{-8} amps on the intermediate range which equals 10^{-6} to 10^{-3} percent power on the power range. The source and intermediate range overlap occurs well below the low power test limit of 5% power. Westinghouse should either justify why these tests for overlap of the source and intermediate range detectors are done during the power ascension test phase instead of the low power test phase or place these tests in DCD (Tier 2) Section 14.2.10.3, "Lower Power Tests."

Westinghouse Response:

a) The twenty applicable PWR "Low Power Tests" identified in RG 1.68 are addressed in the DCD subsections identified in the following table. The item letters in the table correspond to the item letters of RG 1.68, Revision 2, Appendix A, Section 4, "Low Power Tests."

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
4a.	14.2.10.3.4	Low Power Tests [Please see revision to DCD subsection 10.2.10.3.4 in the "Design Control Document (DCD) Revision" section below]
4b.	14.2.10.3.5	Low Power Tests

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
4c.	Please refer to RAI 261.009 for a discussion of this test.	
4d.	14.2.10.2.3	Initial Criticality Test
4e.	14.2.10.4.2	Power Ascension Tests - RG item states "measurements may be taken at higher power level depending on sensitivity of incore flux instrumentation."
4f.	14.2.9.2.18, 14.2.10.4.16	Preoperational Tests and Power Ascension Tests
4g.	14.2.9.2.18, 14.2.10.4.14	Preoperational Tests and Power Ascension Tests
4h.	14.2.9.2.3, 14.2.9.2.20	Preoperational Tests
4i.	14.2.10.1.11, 14.2.10.1.12, 14.2.10.1.13	Initial Fuel Loading and Precritical Tests
4j.	14.2.9.4.11, 14.2.10.4.15	Preoperational Tests and Power Ascension Tests
4k.	NA	AP1000 has no steam-driven engineered safety features or plant auxiliaries
4l.	14.2.9.2.1	Preoperational Tests
4m.	NA	AP1000 is not a BWR
4n.	14.2.9.2.13	Preoperational Tests
4o.	14.2.10.1.14	Initial Fuel Loading and Precritical Tests
4p.	14.2.9.2.1	Preoperational Tests
4q.	14.2.9.2.4, 14.2.9.1.3, 14.2.9.2.1	Preoperational Tests
4r.	14.2.9.2.3	Preoperational Tests
4s.	14.2.9.1.9	Preoperational Tests
4t.	14.2.10.3.6	Low Power Tests (First Plant Only)
4u.	14.2.9.2.12, 14.2.9.2.13	Preoperational Tests

b) Westinghouse performs the tests for overlap of the source and intermediate range detectors in DCD subsection 14.2.10.2.3 during initial criticality tests. DCD 14.2.10.4.3 will be revised as identified in the "Design Control Document (DCD) Revision" section below.

Design Control Document (DCD) Revision:

Revise DCD 14.2.10.3.4 as follows:

14.2.10.3.4 Isothermal Temperature Coefficient Measurement

Objectives

- ~~Measure~~ **Determine** the isothermal temperature coefficient
- **Calculate the moderator temperature coefficient**

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- Instrumentation and equipment used to measure and compute reactivity is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level
- The controlling rod bank is positioned near fully withdrawn ~~or near fully inserted~~

Test Method

- Vary reactor coolant system temperature (heatup/cooldown) while maintaining rods and boron concentration constant
- Monitor reactivity results and determine the isothermal temperature coefficient
- **Calculate the moderator temperature coefficient using the isothermal temperature coefficient and design values**

Performance Criterion

- The measured value for the ~~isothermal~~ **moderator** temperature coefficient is more negative than the Technical Specification limit

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Revise DCD 14.2.10.4.3 as follows:

14.2.10.4.3 Nuclear Instrumentation System

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on ~~source range,~~ intermediate range~~,~~ and power range instrumentation from zero power to at or near full rated thermal power.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms
- Set operational modes of the ~~source range,~~ intermediate range~~,~~ and power range channels for their proper functions, in accordance with the test instructions

Performance Criteria

- The nuclear instrumentation system operates in accordance with the design basis functional requirements as discussed in subsection 4.4.6.
- The nuclear instrumentation system demonstrates an overlap of indication between ~~the source and intermediate ranges and~~ the intermediate and power range instrumentation

PRA Revision:

None.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.007

Question:

RG 1.68, Revision 2, Appendix A, Section 5, "Power Ascension Tests," lists 41 power ascension tests (i.e., tests a through o.o) that should be performed. One test can be eliminated from consideration because it is only performed on BWRs, Item 5c. The AP1000 DCD (Tier 2) Section 14.2.10.4, Power Ascension Tests, lists 28 tests.

a) Westinghouse should provide additional information on where the 40 applicable pressurized water reactor (PWR) power ascension tests listed in RG 1.68 are performed and under which part of the initial test program they are credited. If any power ascension tests are missing and needed under the power ascension test program, Westinghouse should either add these tests to DCD (Tier 2) section 14.2.10.4 or provide appropriate exceptions for not performing these tests at power.

b) Westinghouse should provide additional information on the following RG 1.68, Section 5, "Power Ascension Tests," listed below.

1. Determine the power reactivity coefficients (PWR) are in accordance with design values (25%, 50%, 75%, 100%). This test is designated as "a" in RG 1.68, Appendix A, Section 5.

The isothermal temperature coefficient measurement in DCD (Tier 2) Section 14.2.10.3.4, "Isothermal Temperature Coefficient Measurement," is only tested in the low power test phase. Westinghouse should provide additional information for testing power reactivity coefficients.

2. Pseudo rod ejection test to validate the rod ejection accident analysis (greater than 10% power with control rod banks at the full power rod insertion limit) (PWR). This test need not be repeated for facilities using calculation models and design identical to prototype facilities. This test is designated as "e" in RG 1.68, Appendix A, Section 5.

The NRC staff could not find a test abstract for pseudo rod or Rod Cluster Control Assembly (RCCA) ejection test in DCD (Tier 2) Section 14.2.10.4, "Power Ascension Tests." Please discuss.

3. Demonstrate capability and/or sensitivity, as appropriate for the facility design of incore and excore neutron flux instrumentation, to detect a control rod misalignment equal to or less then the technical specification limits (50%, 100%) (PWR). This test is designated as "i" in RG 1.68, Appendix A, Section 5.

This test is performed in DCD (Tier 2) Section 14.2.10.4.6, Rod Cluster Control Assembly Out of Band Measurements. However, the test is performed at 30% and 50% power. This test is not performed at 100% power. Westinghouse should provide additional information to either add this test at 100% power or provide technical justification for not performing this test at 100% power.

4. Verify proper operation of failed fuel detection systems (25%, 100%). See RAI 261.002b for additional information on CVCS radiation monitors. This test is designated as "q" in RG 1.68, Appendix A, Section 5.

5. m.m. Demonstrate that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves. For PWRs, justification for conducting the test at a lower power level, while still demonstrating proper plant response to this transient, may be submitted for NRC staff review (100%). This test is designated as "m.m" in RG 1.68, Appendix A, Section 5.

Pre-operational test abstract 14.2.9.1.2, "Steam Generator System Testing," does test automatic closure of the main steam line isolation valves during hot functional tests. However, this test doesn't address the dynamic response of the valves at 100% power. Westinghouse should provide additional information that pre-operational hot functional tests are adequate to demonstrate adequate automatic closure of main steam line isolation valves at 100% power.

Westinghouse Response:

a) The forty applicable PWR "Power Ascension Tests" identified in RG 1.68 are addressed in the DCD subsections identified in the following table. The item letters in the table correspond to the item letters of RG 1.68, Revision 2, Appendix A, Section 5, "Power Ascension Tests."

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
5a.	14.2.10.3.3, 14.2.10.4.23, 14.2.7.2	Note 1.
5b.	14.2.10.4.2	Also please see RAI 261.010
5c.	NA	AP1000 is not a BWR
5d.	14.2.10.4.7	
5e.	14.2.10.4.6	(First Plant Only) Please see response to item b)2. of this RAI.
5f.	14.2.10.4.6	(First Plant Only) Please see response to item b)2. of this RAI.
5g.	14.2.10.1.11	Initial Fuel Loading and Precritical Tests - RG item states, "if not previously demonstrated."

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
5h.	14.2.10.1.14	Rod drop times are measured during preoperational testing. Scrams during power ascension testing are not planned, therefore no additional measuring of drop times is performed during power ascension testing.
5i.	14.2.10.4.2, 14.2.10.4.3, 14.2.10.4.6	Note 1
5j.	14.2.10.4.19, 14.2.10.4.21, 14.2.10.4.27	
5k.	NA	AP1000 has no active high pressure safety injection system. The passive core cooling system is tested in DCD 14.2.9.1.3.
5l.	14.2.9.2, 14.2.9.4, 14.2.10.3.6, 14.2.10.3.7, 14.2.10.4.18	Testing to demonstrate residual heat removal capability is performed either during power ascension testing or prior to power ascension testing.
m.	14.2.10.4.5, 14.2.10.4.11, 14.2.9.1.9,	Preoperational Testing and Power Ascension Testing - RG item states, "if not previously demonstrated."
5n.	14.2.9.4.16	Preoperational Testing - RG item states, "if not previously done."
5o.	14.2.9.1.1, 14.2.9.2.13, 14.2.9.2.18, 14.2.9.3.1, 14.2.9.4.11	Preoperational Testing - RG item states, "if not previously demonstrated."
5p.	14.2.9.1.9	Preoperational Testing - RG item states, "if this testing has not been previously completed."
5q.	14.2.9.2.20	Preoperational Testing - AP1000 detects failed fuel via the primary sampling system, which is tested prior to Power Ascension Tests.
5r.	14.2.10.1.8, 14.2.10.1.9, 14.2.10.1.10, 14.2.10.4.2, 14.2.10.4.3, 14.2.10.4.4, 14.2.10.4.5, 14.2.10.4.7, 14.2.10.4.9, 14.2.10.4.10, 14.2.10.4.11, 14.2.10.4.12, 14.2.10.4.13, 14.2.10.4.17	
5s.	14.2.10.4.4, 14.2.10.4.5, 14.2.10.4.9, 14.2.10.4.10,14.2.10.4.12, 14.2.10.4.13, 14.2.10.4.19, 14.2.10.4.20, 14.2.10.4.24, 14.2.9.2.3	

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
5t.	14.2.9.1.1, 14.2.9.2.1, 14.2.10.4.12, 14.2.10.4.20	Preoperational Testing and Power Ascension Testing - RG item states, "if not previously accomplished."
5u.	14.2.9.1.2	Preoperational Testing - This test is performed during preoperational testing to avoid an at-power transient. This is consistent with the valve in-service test requirements (DCD Table 3.9-16). Please see note 20 of Table 3.9-16.
5v.	14.2.9.2.1, 14.2.9.2.2, 14.2.10.4.13, 14.2.10.4.20	
5w.	14.2.9.2.5, 14.2.9.2.6, 14.2.9.2.7, 14.2.10.4.15, 14.2.7.3, 14.2.10.4.16	Adequacy of the radiation shielding is verified as discussed in DCD subsection 14.2.7.3, third bullet and in subsection 14.2.10.4.16. Testing to demonstrate cooling system capability is performed either during power ascension testing or prior to power ascension testing as per the identified DCD subsections.
5x.	14.2.9.1.6	AP1000 has no active engineered safety features and no associated auxiliary systems. The environment for the safety-related I&C is maintained by the main control room habitability system (VES), which is tested in subsection DCD 14.2.9.1.6.
5y.	14.2.10.4.2, 14.2.10.4.3, 14.2.10.4.5, 14.2.10.4.7, 14.2.10.4.9, 14.2.10.4.10, 14.2.10.4.11, 14.2.10.4.17	
5z.	14.2.10.4.14	
5a.a.	14.2.10.4.8	
5b.b.	14.2.10.4.16	
5c.c.	14.2.9.3.1, 14.2.9.3.2, 14.2.9.3.3, 14.2.9.3.4	Also, please see RAI 261.010.
5d.d.	14.2.10.4.28	
5e.e.	14.2.9.4.12	Preoperational Testing - RG item states, "if not previously demonstrated."
5f.f.	14.2.10.4.15	

RG 1.68 App. A	AP1000 DCD Subsection that Addresses RG Item	Comment
5g.g.	14.2.9.2.14	Preoperational Testing - RG item states, "if not previously done."
5h.h.	14.2.10.4.20	
5i.i.	14.2.10.1.18	Test data are gathered during precritical tests. The data is then used to verify the loss of flow analyses in DCD subsections 15.3.1 and 15.3.2.
5j.j.	14.2.10.4.26	
5k.k.	14.2.10.4.27	
5l.l.	14.2.10.4.24	
5m.m.	14.2.10.4.24	
5n.n.	14.2.10.4.24	
o.o	14.2.10.4.18	

Note 1:
The accuracy of analyses of various coefficients and parameters was not very good in 1978 when Regulatory Guide (RG) 1.68, Revision 2, was approved. As a result, requirements in RG 1.68 exist that are written "Determine ..." or "Demonstrate..." instead of "Confirm ...". Today's methodology is to confirm the consistency of the design.

b)

1. Westinghouse performs boron end point tests at both full power and at no load. The results of these tests are used to confirm the necessary power coefficient and power defect parameters. Please also see the response to RAI 261.006 and the response to this RAI item a) "Note 1", above.

2. Westinghouse performs this test as part of the rod cluster control assembly out of bank measurements in DCD subsection 14.2.10.4.6. Westinghouse notes that this test is only performed on the first plant to validate the analysis. This is being reflected in the DCD revision below.

3. The rod cluster control assembly out of bank measurements test is not performed at full power as it would cause the plant to exceed peak power limits. Please also see response to this RAI item a) "Note 1", above.

4. AP1000 detects failed fuel via the primary sampling system, which is tested prior to power ascension tests. While proper operation of the primary sampling system is dependent on system temperature and pressure, it is not dependent on plant power.

5. The dynamic response of the plant to close all main steam isolation valves (MSIVs) is bounded by a plant trip from 100% power, which is performed in test abstract 14.2.10.4.24. MSIV testing is performed during preoperational testing to avoid an at-power transient. This is consistent with the valve in-service test requirements (DCD Table 3.9-16). Please see note 20 of DCD Table 3.9-16.

Design Control Document (DCD) Revision:

Revise the DCD as follows:

14.2.10.4.6 Rod Cluster Control Assembly Out of Bank Measurements (First Plant Only)

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.009

Question:

In addition to the low power test abstracts listed in RAI 261.006; RG 1.68, Revision 2, Appendix A, "Initial Test Program," Section 4, "Low Power Tests," lists test abstract 4(c), "pseudo-rod-ejection test to verify calculation models and accident analysis assumptions." The NRC staff could not find a test abstract in DCD, Tier 2, Section 14.2.10.3, "Low Power Tests," for the pseudo-rod-ejection test. Westinghouse should provide additional information on the pseudo-rod-ejection test for the low power test phase. (Please see RAI 261.007b).

Westinghouse Response:

Westinghouse has obtained sufficient test data from previous plant startups, which allows new plants to require only confirmation of the calculational models. This is consistent with other plants, for example, please see the Watts Bar or Shearon Harris FSAR.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 261.010

Question:

In addition to the power ascension test abstracts listed in RAI 261.007b, Westinghouse should provide additional information on the following power ascension test abstracts:

1. Determine that steady-state core performance is in accordance with design. Sufficient measurements and evaluations should be conducted to establish that flux distributions, local surface heat flux, linear heat rate, departure from nucleate boiling ratio (DNBR), radial and axial power peaking factors, and other important parameters are in accordance with design values throughout the permissible range of power to flow conditions. This test is designated as "b" in RG 1.68, Appendix A, Section 5.

   Power ascension test abstract Section 14.2.10.4.2, Incore Instrumentation Systems, does contain test methods to generate data from incore maps to verify core power peaking and axial distributions that are consistent with design predictions and the limits imposed by plant Technical Specifications. The test objective also states "Obtain data for incore thermocouple and flux maps at various power levels during power ascension to full power to determine flux distributions;" however, there is no statement for collection of test data for local surface heat flux, linear heat rate, departure from nucleate boiling and radial power peaking factors. This information should be added to this test abstract or another applicable test abstract in Section 14.2.10.4, Power Ascension Tests.

2. Demonstrate that gaseous and liquid radioactive waste processing, storage, and release systems operate in accordance with design. This test is designated as "c.c" in RG 1.68, Appendix A, Section 5.

   Low power test abstracts Section 14.2.9.3.1, "Liquid Radwaste System Testing," and Section 14.1.9.3.2, "Gaseous Radwaste System Testing," may cover these tests. However, Westinghouse should justify why these tests are performed at low power instead of the power ascension test phase since a much larger quantity of liquid and gaseous radioisotopes are being produced at power then during the low power test phase.

Westinghouse Response:

1. Westinghouse will revise DCD subsection 14.2.10.4.2 as identified in the **"Design Control Document (DCD) Revision:"** portion of this RAI response to address the concern. As stated in DCD subsection 14.4.2, the Combined License applicant is responsible for

providing test specifications and test procedures for the preoperational and startup tests. The procedures will meet the appropriate regulatory guidance.

2. Testing of the gaseous and liquid radioactive waste processing, storage, and release systems is performed at low power so that the negative impact of any system not performing as designed is minimal. The low power testing confirms that the systems performed as designed and therefore additional testing at high plant power is not necessary. This is consistent with other plants, for example, please see the Watts Bar or Vogtle FSAR.

Design Control Document (DCD) Revision:

Change the second bullet under Test Method, of subsection 14.2.10.4.2 Incore Instrumentation System to read as follows:

Use data from the incore maps to verify that core power ~~peaking factors and axial~~ distribution ~~are~~ is consistent with design predictions and the limits imposed by the plant Technical Specifications, and to calibrate other plant instrumentation.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 280.003

Question:

Section 9.5.1.2.1.1 states that the control of combustible materials is in accordance with National Fire Protection Association Standard 803 (NFPA 803), "Light Water Nuclear Power Plants." NFPA 803 was withdrawn by the NFPA Standards Council in January 2001, subsequent to the issuance of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"; therefore, NFPA 803 is not an appropriate reference. The correct reference relating to the control of combustible materials for the AP1000 is Section 3.3 of NFPA 804, "Fire Protection for Advanced Light Water Reactor Electric Generating Plants," 2001 Edition. NFPA 803 as well as superceded editions of other NFPA standards are cited in several other sections of the AP1000 DCD and WCAP-15871, "AP1000 Assessment Against NFPA 804." Several of the editions cited in the DCD are different than the editions cited in the WCAP. SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advance Light Water Reactor (ALWR) Designs," states that passive plant designs are to be reviewed using the latest industry standards endorsed by the Nuclear Regulatory Commission (NRC). Regulatory Guide (RG) 1.189, "Fire Protection for Operating Nuclear Power Plants," provides a listing of the NFPA Codes and Standards endorsed by the NRC. Consistent with the criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34 (g), please revise the DCD and WCAP accordingly to reflect the applicable NFPA codes and standards that were in effect six months prior to the date of the AP1000 design certification application (March 28, 2002). This should include revisions to all citations to NFPA 803 and other superceded NFPA codes and standards.

Westinghouse Response:

Westinghouse agrees to revise the DCD and WCAP-15871 as shown below. With these changes the DCD code references meet the six-months-before-application requirement stated in the RAI.

Note that there are some editions listed in the proposed changes to the DCD that are different than the editions cited in WCAP-15871. These differences are due to the six-months-before-application requirement for the DCD while the editions cited in WCAP-15871 are those cited in NFPA-804 (2001 edition).

In DCD Appendix 9A there are references to both the 16th and the 18th editions of the Fire Protection Handbook. This is because the fire severity and duration calculations performed for the AP1000 follow the 16th edition of the handbook. These sections have been removed from the 18th edition. Reference to the 16th edition is required to properly describe the calculation methodology.

Design Control Document (DCD) Revision:

8.3.4 References

19. NFPA 780, "Standard for the Installation of Lightning Protection Systems," ~~1997~~2000.

9.4.13 References

27. "Standard for Installation of Air Conditioning and Ventilation Systems," NFPA 90A, 1999.

33. NFPA 92A-~~1993~~2000, "Recommended Practice for Smoke Control Systems."

9.5.1.2.1.1 Plant Fire Prevention and Control Features

Control of Combustible Materials

The plant is constructed of noncombustible materials to the extent practicable. The selection of construction materials and the control of combustible materials are in accordance with BTP CMEB 9.5-1 and Section 3.3 of NFPA ~~803~~-804 (Reference 2) as specified in WCAP-15871 (Reference 20).

The storage and use of hydrogen are according to NFPA 50A and NFPA 50B (Reference 2). Hydrogen lines in safety-related areas are designed to seismic Category I requirements.

Ventilation systems are designed to maintain the hydrogen concentration in the battery rooms well below 2 percent by volume, as described in subsections 9.4.1 and 9.4.2.

The turbine lubrication oil system, located in the turbine building, is separated from areas containing safety-related equipment by 3-hour rated fire barriers.

Outdoor oil-filled transformers are separated from plant buildings according to ~~NFPA 803 and~~ NFPA 804 (Reference 2).

9.5.1.2.1.2 Fire Detection and Alarm Systems

Fire detection and alarm systems are provided where required by the fire protection analysis, in accordance with BTP CMEB 9.5-1~~, NFPA 803, NFPA 804,~~ and NFPA 72 (Reference 2).

Fire detection and alarm systems are generally in accordance with NFPA 804 (Reference 2). See WCAP-15871 (Reference 20) for details.

9.5.1.2.1.4 Automatic Fire Suppression Systems

Automatic fire suppression systems are in accordance with BTP CMEB 9.5-1 and the applicable NFPA standards, with consideration of the unique aspects of each application, including building characteristics, materials of construction, environmental conditions, fire area contents, and adjacent structures.

Fixed automatic fire suppression systems are provided based on the results of the fire protection analysis.

The selection of automatic suppression systems for each plant area is based on the guidance of ~~NFPA 803 and~~ NFPA 804 (Reference 2) as stated in WCAP-15871 (Reference 20). Water systems are preferred, but the use of automatic water suppression systems for firefighting in radiation areas is minimized because of the possible spread of contamination. Halon and carbon dioxide fixed flooding systems are not used.

The fire protection analysis describes the fire suppression systems provided for each fire area.

9.5.5 References

2. National Fire Protection Association Codes and Standards:

   NFPA 10, ~~1990~~1998: Standard for Portable Fire Extinguishers; NFPA 13, ~~1994~~1999: Standard for the Installation of Sprinkler Systems; NFPA 14, ~~1993~~2000: Standard for the Installation of Standpipe, Private Hydrants, and Hose Systems; NFPA 15, ~~1990~~2001: Standard for Water Spray Fixed Systems for Fire Protection; NFPA 20, ~~1993~~1999: Standard for the Installation of ~~Centrifugal Fire~~Stationary Pumps for Fire Protection; NFPA 22, ~~1993~~1998: Standard for Water Tanks for Private Fire Protection; NFPA 24, ~~1992~~1995: Standard for Installation of Private Fire Service Mains and Their Appurtenances; NFPA 30, ~~1993~~2000: Flammable and Combustible Liquids Code; NFPA 50A, ~~1994~~1999: Standard for Gaseous Hydrogen Systems at Consumer Sites; NFPA 50B, 1999: Standard for Liquefied Hydrogen Systems at Consumer Sites; NFPA 72, ~~1993~~1999: National Fire Alarm Code; NFPA 780, ~~1992~~2000: Standard for the Installation of Lightning Protection Systems ~~Lighting Protection Code~~; ~~NFPA 803, 1993: Light Water Nuclear Power Plants;~~ NFPA 804, ~~1995~~2001: Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants.

20. WCAP-15871, "AP1000 Assessment Against NFPA 804," April 2002.

Table 9.5.1-3 (Sheet 1 of 2)

EXCEPTIONS TO NFPA STANDARD REQUIREMENTS

Requirement	AP1000 Exception or Clarification
NFPA 13 Sections ~~4-7.2~~5-14.1.1.2 and 5-15.2 requires fire department connections to individual sprinkler system headers, with no intervening shutoff valves.	Individual connections are not provided. Sprinkler systems are supplied from the proprietary fire water supply system, which can be accessed by the fire department at any hydrant along the yard main. Valves between these connection points and the sprinkler systems are electrically supervised or locked open.
NFPA 14 Section 2-~~6~~5 requires that listed valves be used to control connections to standpipes.	Containment isolation valves controlling the water supply to standpipes inside containment are nuclear safety-related and meet or exceed the requirements for listed valves.
NFPA 14 Section 3-5 prohibits use of dry standpipes for Class II or Class III systems, and in areas not subject to freezing.	The standpipe system inside containment is classified as a dry standpipe system because it is normally isolated by the outboard containment isolation valve as described in subsection 9.5.1.2.1.5.
NFPA 14 Section ~~3.6.1~~3-6.1 requires listed dial spring pressure gauges at specific locations.	Pressure instruments with remote readout at fire protection system panels are provided. These instruments meet or exceed the requirements for listed gauges.
NFPA 14 Section 4-2.2 requires an isolation valve for each standpipe.	One valve is used to isolate two or more short standpipes that supply a small number of hose stations.
NFPA 14 Sections 4-3 and 5-12 require fire department connections for each standpipe system, with no intervening shutoff valves.	Individual connections are not provided. Standpipe systems are supplied from the proprietary fire water supply system, which can be accessed by the fire department at any hydrant along the yard main. Valves between these connection points and the standpipe systems are electrically supervised or locked open, except as described in subsection 9.5.1.2.1.5.
NFPA 14 Section 5-3.2 requires Class I hose connections at each intermediate landing of exit stairways, on each side of horizontal exit openings, in each exit passageway, and on the roof or at the highest landing of stairways.	Class I hose connections are provided in exit stairways at one intermediate landing between most floors, and at other protected exit locations accessible to firefighters entering the buildings from outside. Flow testing of Class I hose connections is accomplished without providing additional connections on the roofs of buildings or at the highest stairway landings.
NFPA 14 Section 5-5 requires standpipes to be interconnected at the bottom, and when supplied by elevated tanks, also at the top.	Standpipes interconnections are constrained by layout considerations and do not always meet these requirements. Each standpipe receives an adequate water supply at an adequate pressure.
NFPA 14 Section 5-11.2 requires a separate drain connection for each	For standpipes located outside radiologically controlled areas and supplied at an elevation above the lowest hose connection, the hose

standpipe.	connection is used to provide a means of draining the standpipe.
NFPA 22 contains requirements for water tanks and supply lines for private fire protection.	The seismic standpipe system is normally supplied from the passive containment cooling system (PCS) water storage tank as described in subsection 9.5.1.2.1.5. The passive containment cooling system tank and supply line are not designed to NFPA 22 but meet or exceed the applicable requirements of that standard.
~~NFPA 803 and~~ NFPA 804 contains requirements specific to light water reactors.	Compliance with portions of ~~these~~ this standards is as identified within Section 9.5.1 and WCAP-15871.

9A.2.3 Fire Severity Categorization

For purposes of evaluating fire barrier adequacy, the expected fire severity for each area/zone is categorized from A (slight) to E (severe) in accordance with Table 7-9E of the NFPA Fire Protection Handbook, 16th edition, (Reference 2) based on the type of materials present.

9A.2.4 Combustible Loading and Equivalent Fire Duration Calculations

Equivalent Fire Duration

The duration of a fire in a given fire area or zone is influenced by many factors, including:

- The properties of the material (ease of ignition and rate of heat release)
- The surface area of the combustible material
- The presence of fire retardant coatings
- Ventilation parameters and availability of oxygen
- The degree of separation or the presence of barriers between groups of combustible materials

Fire duration is estimated based on the fire severity category and the equivalent combustible loading. Equivalent combustible loading is defined as the weight per square foot of ordinary combustibles (wood or paper) having a heat of combustion of 8,000 Btu/lb, that releases the same total heat as the combustibles in the fire area/zone. The equivalent combustible loading is calculated by dividing the maximum heat release per square foot by 8,000 Btu/lb. The fire endurance lines of Figure 7-9B of the NFPA Fire Protection Handbook, 16th edition, (Reference 2) are used to estimate the fire duration in minutes.

Fire barriers are tested by exposure to a fire whose severity follows a time varying temperature curve known as the standard time-temperature curve (NFPA Fire Protection Handbook, 18th edition, (Reference 5) Figure ~~7-9A~~7-5A.) The estimated fire duration for each fire area is normalized based on the standard time-temperature curve to obtain an equivalent fire duration. This value is compared with the fire resistance of the fire area boundaries. This comparison is used in conjunction with other factors, including those listed above, in making a determination of the adequacy of the fire area boundaries.

9A.4 References

4. NFPA 92A-~~1993~~2000, "Recommended Practice for Smoke Control Systems."

5. **Fire Protection Handbook, National Fire Protection Association, 18th edition.**

PRA Revision:

None

WCAP-15871 Revision:

NFPA 804 PARAGRAPH	AP1000 COMPLIANCE STATEMENT
Chapter 3 Fire Protection and Administrative Controls	N/A - Heading
3.3 Control of Combustible Materials.	N/A - Heading
3.3.1.5 The storage and use of hydrogen shall be in accordance with NFPA 50A, *Standard for Gaseous Hydrogen Systems at Consumer Sites*, and NFPA 50B, *Standard for Liquefied Hydrogen Systems at Consumer Sites*.	**Comply** ~~The AP1000 fire protection design criteria document applies NFPA 50A by reference to NFPA 803, it does not apply NFPA 50B.~~
Chapter 9 Fire Protection for the Construction Site	**N/A - Heading** ~~Chapter 9: The AP1000 fire protection criteria document requires conformance to NFPA 803 requirements for the construction period. NFPA 804 Chapter 9 requirements are more prescriptive that those in NFPA 803. A detailed comparison of the NFPA 804 Chapter 9 requirements with the AP1000 construction plan has not been made~~

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 410.007

Question:

(DCD, Tier 2, Section 6.4, 9.4. through 9.4.3 and 9.4.6 through 9.4.11) The required aspects of a control room for nuclear power reactors are documented in 10 CFR Part 50, Appendix A,"General Design Criteria for Nuclear Power Plants." GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

Section 6.4.5.4 states that "[t]esting for main control room in-leakage during VES [main control room emergency habitability system] operation will be conducted once every 10 years. This testing will be conducted in accordance with ASTM [American Society for Testing and Materials] E741, 'Standard Test Method for Determining Leakage Rate by Tracer Dilution'." The staff is currently working with the industry to address control room habitability issues including air in-leakage testing. It is anticipated that the testing frequency will be on the order of 5 to 6 years. The staff expects that testing requirements for the AP1000 design will be consistent with the resolution of the control room habitability issues currently pursued by the industry and the staff. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing in accordance with the anticipated outcome of the joint effort between the NRC staff and industry. Please provide such a commitment and revise Section 6.4.8 to add the ASTM E741 standard.

In addition, consistent with the SRP, Westinghouse should commit to complying with the guidance contained in the latest versions of RG 1.52, "Design, Testing, and Maintenance for Post-Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Westinghouse Response:

Westinghouse recognizes that the NRC staff and the industry are working on in-leakage testing, however it is not reasonable to commit to a standard that does not currently exist. Westinghouse therefore is not providing a commitment to have the Main Control Room Emergency Habitability System (VES) meet the anticipated requirements currently being pursued. The VES design addresses in-leakage and meets the codes and standards that were in effect six months prior to the date of the AP1000 design certification application (March 28, 2002).

Westinghouse is revising the DCD, subsection 6.4 to include ASTM E741.

Westinghouse is including Regulatory Guide (RG) 1.140 (Rev. 2 06/2001) in the DCD in subsections 1A, 3.2 and 9.4. Please see the corresponding DCD revisions below.

RG 1.52, is not applicable to the AP1000 as the AP1000 has no safety-related air filtration systems.

Design Control Document (DCD) Revision:

- ***Changes to DCD 6.4:***

6.4.5.1 Preoperational Inspection and Testing

Preoperational testing of the main control room emergency habitability system is performed to verify that the air flow rate of 65 ±5 scfm is sufficient to maintain pressurization of the main control room envelope of at least 1/8-inch water gauge with respect to the adjacent areas. The positive pressure within the main control room is confirmed via the differential pressure transmitters within the control room. The installed flow meters are utilized to verify the system flow rates. The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits. Air quality within the MCR environment is confirmed to be within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1 by analyzing air samples taken during the pressurization test.

The storage capacity of the compressed air storage tanks is verified to be in excess of 314,132 scf of compressed air at a minimum pressure of 3400 psig. This amount of compressed air will assure 72 hours of air supply to the main control room.

An inspection will verify that the heat loads within the rooms identified in Table 6.4-3 are less than the specified values.

Preoperational testing of the main control room isolation valves in the nuclear island nonradioactive ventilation system is performed to verify the leaktightness of the valves.

Preoperational testing for main control room inleakage during VES operation will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Air Leakage Rate by Tracer Dilution."~~(Reference 4).

Testing and inspection of the radiation monitors is discussed in Section 11.5. The other tests noted above are discussed in Chapter 14.

6.4.5.4 Air Inleakage Testing

Testing for main control room inleakage during VES operation will be conducted once every ten years. This testing will be conducted in accordance with ASTM E741, ~~"Standard Test Method for Determining Leakage Rate by Tracer Dilution."~~(Reference 4).

6.4.8 References

1. "Ventilation for Acceptable Indoor Air Quality," ASHRAE Standard 62 - 1989.
2. "Human Engineering Design Guidelines," MIL-HDBK-759C, 31 July 1995.
3. "Human Engineering," MIL-STD-1472E, 31 October 1996.
4. **"Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," ASTM E741, 2000**

- ***Changes to DCD compliance table for RG 1.140 in DCD Appendix 1A. The following replaces the existing compliance:***

APPENDIX 1A

CONFORMANCE WITH REGULATORY GUIDES

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
DIVISION 1 – Power Reactors			
Reg. Guide 1.140, Rev. 2, 06/01 - Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants			
C.1		Conforms	Regulatory Guide 1.140 endorses ASME Standard N509-1989 (Reference 39), ASME Standard N510-1989 (Reference 40) and ASME AG-1-1997 (Reference 38). The AP1000 uses the latest version of the industry standards (as of 3/2002).
C.2.1-2.4		Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.3.1-3.2		Conforms	
C.3.3	ERDA 76-21, Section 5.6; ASME N509-1989 Section 4.9	Conforms	
C.3.4	Regulatory Guide 8.8	Conforms	
C.3.5		Conforms	
C.3.6	ASME AG-1-1997 Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
	ASME AG-1-1997, Section TA	Conforms	
C.4.1	ASME AG-1-1997, Section FB	Conforms	
C.4.2	ASME AG-1-1997, Section CA	Conforms	
C.4.3	ASME AG-1-1997, Section FC, and Section TA	Conforms	
C.4.4	ASME AG-1-1997, Section FG	Conforms	
C.4.5	ERDA 76-21, Section 4.4; ASME AG-1a-2000, Section HA	Conforms	

Criteria Section Exceptions	Referenced Criteria	AP1000 Position	Clarification/Summary Description of
C.4.6	ASME N509-1989, Section 5.6; ASME AG-1a-2000, Section HA	Conforms	
C.4.7	ASME AG-1-1997, Section CA	Conforms	
C.4.8	ASME AG-1-1997, Section FD or FE	Conforms	
C.4.9	ASME AG-1-1997, Section FD and FE or, Section FF	Conforms	
C.4.10	ASME AG-1-1997 Section SA	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are designed to meet at least SMACNA design standards.
C.4.11		Conforms	
C.4.12	ASME AG-1-1997 Section DA	Conforms	
C.4.13	ASME AG-1-1997, Section BA and SA	Conforms	
C.5.1	ERDA 76-21, Section 2.3.8; ASME AG-1a-2000, Section HA	Conforms	
C.5.2		Conforms	
C.6	ASME N510-1989	Conforms	
C.7	ANSI N509-1989	Conforms	

- ***Add new reference to DCD Appendix 1A:***

1A.1 References

38. ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment" 1997

- ***Changes to DCD 3.2:***

1. Changes to Subsection 3.2.6 References

18. ASME/ANSI ~~N509-89~~**AG-1-1997, "Code on Nuclear Air and Gas Treatment**~~Nuclear Power Plant Air Cleaning Units and Components~~."

2. Changes to Table 3.2-3 sheet 54

Table 3.2-3 (Sheet 54 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code
Nuclear Island Nonradioactive Ventilation System (VBS) (Continued)				
n/a	MCR/TSC Supplemental Air Filtration Units	Note 2	NS	ASME ~~N509~~**AG-1, Note 4**

3. *Changes to DCD Table 3.2-3 sheet 60*

Table 3.2-3 (Sheet 60 of 67)

AP1000 CLASSIFICATION OF MECHANICAL AND FLUID SYSTEMS, COMPONENTS, AND EQUIPMENT

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code
Containment Air Filtration System (Continued)				
n/a	Air Exhaust Filtration Units	R	NS	ASME ~~N509~~AG-1, Note 4
n/a	Fans, Ductwork	L or R	NS	SMACNA or ASME ~~N509~~AG-1, Note 4

4. *Changes to Notes as the end of DCD Table 3.2-3*

Notes:

1. Component performs a safety-related function equivalent to AP1000 equipment Class C. The component is constructed using the standards for Class R and a quality assurance program in conformance with 10 CFR Part 50 Appendix B.
2. Component performs an AP1000 equipment Class D function and is constructed using the standards for Class L or Class R.
3. Fire dampers are constructed to the requirements of UL-555 or UL-555S if they are fire and smoke dampers and are located in Class D, Class L, and Class R ducts.
4. **Construction is non-seismic and meets applicable portions of ASME AG-1 consistent with RG 1.140.**

- ***Changes to Section 9.4***

9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations

- Isolates the HVAC penetrations in the main control room boundary on high-high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

Those portions of the nuclear island nonradioactive ventilation system which penetrate the main control room envelope are safety-related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. Other functions of the system are nonsafety-related. HVAC equipment and ductwork whose failure could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portion of the system is nonsafety-related and nonseismic. The equipment is procured to meet the environmental qualifications used in standard building practice.

The nuclear island nonradioactive ventilation system is designed to control the radiological habitability in the main control room within the guidelines presented in Standard Review Plan (SRP) 6.4 and NUREG 0696 (Reference 1), if the system is operable and ac power is available.

Portions of the system that provide the defense-in-depth function of filtration of main control room/technical support center air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with Generic Issue B-36, as described in Section 1.9 and Regulatory Guide 1.140 (Reference 30), as described in Appendix 1A, and the applicable portions of ASME AG-1 (Reference 36), ASME N509 (Reference 2) and ASME N510 (Reference 3).

9.4.1.2.2 Component Description

The nuclear island nonradioactive ventilation system is comprised of the following major components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Tables 9.4.1-1, 9.4.1-2 and 9.4.1-3 provide design parameters for major components in each subsystem.

Supply Air Handling Units

Each air handling unit consists of a mixing box section, a low efficiency filter bank, high efficiency filter bank, an electric heating coil, a chilled water cooling coil bank, and supply and return/exhaust air fans.

Supply and Return/Exhaust Air Fans

The supply and return/exhaust air fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Ancillary Fans

The ancillary fans are centrifugal type with non-overloading horsepower characteristics. Each can provide a minimum of 1,530 cfm. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5), and ANSI/AMCA 300 (Reference 6).

Supplemental Air Filtration Units

Each supplemental air filtration unit includes a high efficiency filter bank, an electric heating coil, a charcoal adsorber with upstream HEPA filter bank, a downstream postfilter bank and a fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans and controls, and the location of the fans on the filtered side of units are designed, constructed, and tested to meet the applicable performance requirements of ASME AG-1, ASME N509 and ASME N510 (References 36, 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52 and 126 (References 7 and 35). Filter minimum average dust spot efficiency is shown in Table 9.4.1-1 and 9.4.1-2. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509-~~AG-1 (Reference ~~2~~36), Section ~~5.3~~FB. Postfilters downstream of the charcoal filters have a minimum DOP efficiency of 95 percent. The filters meet UL 900 (Reference 8) Class I construction criteria.

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME ~~N509-~~AG-1 (Reference ~~2~~36), Section ~~5.1~~FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol in accordance with ASME AG-1 (Reference 36), Section TA.

Charcoal Adsorbers

Each charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME ~~N509-~~AG-1 (Reference 36), Section ~~5.2~~FE~~; ASME 510, Sections 11, 12, and 16~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with IE Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are multi-stage fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils for the supplemental air filtration subsystem are constructed, qualified, and tested in accordance with ASME ~~N509~~AG-1 (Reference ~~2~~36), Section ~~5.5~~CA.

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL-1996 (Reference 26) and the National Electrical Code NFPA 70 (Reference 28).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifiers

The humidifiers are packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifiers are designed and rated in accordance with ARI 620 (Reference 13).

Isolation Dampers and Valves

Nonsafety-related isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power. The isolation dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~AG-1 (Reference ~~2~~36), Section ~~5.9~~DA.

The main control room pressure boundary penetrations include isolation valves, interconnecting piping, and vent and test connection with manual test valves. The isolation valves are classified as Safety Class C (see subsection 3.2.2.5 and Table 3.2-3) and seismic Category I. Their boundary isolation function will be tested in accordance with ASME N510 (Reference 3).

The main control room pressure boundary isolation valves have electro-hydraulic operators. The valves are designed to fail closed in the event of loss of electrical power. The valves are qualified to shut tight against control room pressure.

Tornado Protection Dampers

The tornado protection dampers are split-wing type and designed to close automatically. The tornado protection dampers are designed against the effect of 300 mph wind.

Shutoff, Balancing and Backdraft Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Backdraft dampers are of the counterbalanced type and are provided to delay smoke migration through ductwork in case of fire. The backdraft dampers meet the Leakage Class II requirements of ASME N509 (Reference 2). Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements in accordance with ANSI/AMCA 500 (Reference 14). The supplemental air filtration subsystem dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 or ASME ~~N509~~AG-1 (Reference ~~2~~36), Section ~~5.9~~DA.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers meet the design, leakage testing, and installation requirements of UL-555S (Reference 25).

Ductwork and Accessories

Ductwork, duct supports, and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. Ductwork, supports, and accessories meet the design and construction requirements of SMACNA Industrial Rectangular and Round Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17). The supplemental air filtration and main control room/technical support center HVAC subsystem's ductwork, including the air filtration units and the portion of the ductwork located outside of the main control room envelope, that maintains integrity of the main control room/technical support center pressure boundary during conditions of abnormal airborne radioactivity are designed in accordance with ASME ~~N509~~AG-1 (Reference ~~2~~36), ~~Section 5.10~~Article SA-4500 to provide low leakage components necessary to maintain main control room/technical support center habitability.

9.4.7.2.2 Component Description

The containment air filtration system is comprised of the following components. These components are located in buildings on the Seismic Category I Nuclear Island and the Seismic Category II portion of the annex building. The seismic design classification, safety classification and principal construction code for Class A, B, C, or D components are listed in Section 3.2. Table 9.4.7-1 provides design parameters for the major components of the system.

Supply Air Handling Units

Each supply air handling unit consists of a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil bank, a chilled water cooling coil bank and a supply fan.

Exhaust Air Filtration Units

Each exhaust air filtration unit consists of an electric heater, an upstream high efficiency filter bank, a HEPA filter bank, a charcoal adsorber with a downstream postfilter bank, and an exhaust fan. The filtration unit configurations, including housing, internal components, ductwork, dampers, fans, and controls, are designed, constructed, and tested to meet the applicable performance requirements of ASME AG-1, N509 and ASME N510 (References 36, 2 and 3) to satisfy the guidelines of Regulatory Guide 1.140 (Reference 30) except as noted in Appendix 1A. The filtration unit housings maximum leakage rates do not exceed one percent of the design flow in accordance with ASME ~~N509~~AG-1. Refer to Table 9.4-1 for a summary of the containment air filtration system filtration efficiencies and Appendix 1A for a comparison of the containment air filtration system exhaust air filtration units with Regulatory Guide 1.140 (Reference 30).

Isolation Dampers

Isolation dampers are bubble tight, single-blade or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or instrument air. The design and construction of the isolation dampers is in accordance with ANSI/AMCA 500 or ASME ~~N509~~-AG-1 (References 14 and ~~2~~36).

Pressure Differential Control Dampers

Pressure differential control dampers utilize opposed-blade type construction and meet the performance requirements of ANSI/AMCA 500 (Reference 14) or ASME ~~N509~~-AG-1 (Reference ~~2~~36), Section ~~5.9~~DA. The dampers maintain a slight negative pressure within the fuel handling building area, with respect to the environment and adjacent non-radiologically controlled plant areas.

Supply and Exhaust Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. Fan performance is rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Containment Penetrations

The containment penetrations include containment isolation valves, interconnecting piping, and vent and test connections with manual test valves. The containment isolation components that maintain the integrity of the containment pressure boundary after a LOCA are classified as Safety Class B and seismic Category I. Seismic Category I debris screens are mounted on Safety Class C, seismic Category I pipe to prevent entrainment of debris through the supply and exhaust openings that may prevent tight valve shutoff. The screens are designed to withstand post-LOCA pressures.

The containment isolation valves inside and outside the containment have air operators. The valves are designed to fail closed in the event of loss of electrical power or air pressure. The valves are controlled by the protection and plant safety monitoring system as discussed in subsection 7.1.1. The valves shut tight against the containment pressure following a design basis accident.

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. The system air ductwork inside containment meets seismic Category II criteria so that it will not fall and damage any safety-related equipment following a safe shutdown earthquake. Ductwork, supports and accessories meet the design and construction requirements of SMACNA Rectangular and Round Industrial Duct Construction Standards (References 16 and 34) and SMACNA HVAC Duct Construction Standard - Metal and Flexible (Reference 17). The exhaust air ductwork and supports meet the design and construction requirements of ASME ~~N509~~-AG-1 (Reference ~~2~~36), ~~Section 5.10~~Article SA-4500.

Shutoff and Balancing Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements of ANSI/AMCA 500 (Reference 14). The containment exhaust air dampers meet the design and construction criteria of ASME ~~N509~~-AG-1 (Reference ~~2~~36), Section ~~5.9~~DA.

Fire Dampers

Fire dampers are provided where the ductwork penetrates a fire barrier to maintain the fire resistance rating of the fire barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15).

Low Efficiency Filters, High Efficiency Filters, and Postfilters

Low and high efficiency filters are rated in accordance with ASHRAE Standard 52 and 126 (References 7 and 35). The minimum average dust spot efficiencies of the filters are shown in Table 9.4.7-1. High efficiency filter performance upstream of HEPA filter banks meet the design requirements of ASME ~~N509~~-AG-1 (Reference ~~2~~36), Section ~~5.3~~FB. Postfilters located downstream of the charcoal adsorbers have a minimum DOP efficiency of 95 percent. The filters meet UL 900 Class I construction criteria (Reference 8).

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with ASME ~~N509~~AG-1 (Reference ~~2~~36), Section ~~5.1~~FC. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol **in accordance with ASME AG-1, Section TA.**

Charcoal Adsorbers

Each charcoal adsorber is designed constructed, qualified, and tested in accordance with ASME ~~N509~~**AG-1 (Reference 36)**, Section ~~5.2~~**FE** ~~(Reference 2); ASME 510, Sections 11, 12, and 16 (Reference 3)~~; and Regulatory Guide 1.40. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with 1E Bulletin 80-03 (Reference 29).

Electric Heating Coils

The electric heating coils are fin tubular type. The electric heating coils meet the requirements of UL-1995 (Reference 10). The coils are constructed, qualified and tested in accordance with ASME ~~N509~~ AG-1 (Reference ~~2~~36), Section ~~5.5~~CA.

Heating Coils

The heating coils are hot water, finned tubular type. The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

9.4.12 Combined License Information

The Combined License applicants referencing the AP1000 certified design will implement a program to maintain compliance with **ASME AG-1 (Reference 36)**, ASME N509 (Reference 2), ASME N510 (Reference 3) and Regulatory Guide 1.140 (Reference 30) for portions of the nuclear island nonradioactive ventilation system and the containment air filtration system identified in subsection 9.4.1 and 9.4.7. The Combined License applicant will also provide a description of the MCR/TSC HVAC subsystem's recirculation mode during toxic emergencies, and how the subsystem equipment isolates and operates, as applicable, consistent with the toxic issues to be addressed by the Combined License applicant as discussed in DCD subsection 6.4.7.

- ***Add new reference 36 to DCD 9.4.13:***

9.4.13 References

36. "Code on Nuclear Air and Gas Treatment," ASME/ANSI AG-1-1997

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.009

Question:

(DCD, Tier 1, Sections 2.2.5 and 2.7.1; Tier 2: Sections 6.4 and 9.4, and Chapter 16, TS 3.7.6 and TS 3.9.5): The NRC staff expects the AP1000 design to commit to compliance with the latest revisions of the applicable Codes and Standards for the following HVAC systems:

- radiologically controlled area ventilation system (VAS),
- non-radioactive ventilation system (VBS),
- containment recirculating cooling system (VCS),
- main control room emergency habitability system (VES),
- containment air filtration system (VFS),
- health physics and hot machine shop HVAC system (VHS),
- radwaste building HVAC system (VRS),
- turbine building ventilation system (VTS),
- annex/auxiliary buildings non-radioactive HVAC system (VXS), and
- diesel generator building heating and ventilation system (VZS)

Please review the applicable portions of the DCD descriptions and TSs to ensure proper references to the latest revisions to the applicable Codes and Standards and revise the DCD as necessary.

Westinghouse Response:

The AP1000 HVAC systems described in the DCD Tier 1 and Tier 2 meets the codes and standards that were in effect six months prior to the date of the AP1000 design certification application. (The AP1000 application date was March 28, 2002). No change to the AP1000 DCD is required.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.014

Question:

(DCD, Tier 2, Section 9.4) The staff issued AP600 RAIs 410.96 ,410.97, 410.99, 410.100, 410.101, 410.104, 410.241 and 410.243 addressing various AP600 ventilation systems. In response to these RAI questions, you provided specific clarifications as follows:

A. AP600 RAI 410.96: Westinghouse indicated that safety-related MCR isolation dampers in VBS conform to the Positions C.1 and C.2 of RG 1.29 as discussed in SSAR Appendix 1A,

B. AP600 RAI 410.97: Westinghouse indicated that safety-related MCR isolation dampers and the VBS design conform to GDC 4, "Environmental and Missile Design Basis,"

C. AP600 RAI 410.99: Westinghouse indicated that the fuel handling area HVAC subsystem of VAS conforms to Positions C.1 through C.4 of RG 1.29 and VFS conforms to Positions C.1.a through C.1.d, and C.2.a through C.2.f, of RG 1.140 and Position C.4 of RG 1.13,

D. AP600 RAI 410.100: Westinghouse indicated that the VFS containment penetrations that provide containment isolation conform to single failure criteria based on SRP Section 9.4.5 guidelines,

E. AP600 RAI 410.101: Westinghouse indicated that VRS conforms to Positions C.1 and C.2 of RG 1.140,

F. AP600 RAI 410.104: Westinghouse indicated that VHS conforms to Positions C.1 and C.2 of RG 1.140,

G. AP600 RAI 410.241: Westinghouse provided tabulated flow data for VAS, VBS, VHS, and VRS to demonstrate that these systems maintain their served areas at certain positive or negative pressures with respect to surrounding spaces and/or outdoor atmosphere, and

H. AP600 RAI 410.243: Westinghouse provided the rationale for the installation of filtration air supply fans upstream of filtration units for VBS in order to meet the requirements of Section 4.7.2, "Habitability Systems," of ASME N509-1989.

It is not clear to the NRC staff that the above clarifications are appropriately reflected in the AP1000 DCD. Please clarify each issue with regard to the applicability to the AP1000, DCD Tier 2 Section 9.4 design, and revise the AP1000 DCD, Tier 2 Section 9.4 accordingly.

AP600 RAIs 410.96, 410.97, 410.99, 410.100, 410.101, and 410.104 were issued by the NRC staff on January 26, 1993 (NUDOCS Accession No. 9303050146). AP600 RAIs 410.241 and 410.243 were issued by the NRC staff on May 23, 1994 (NUDOCS Accession No. 9406230033). Westinghouse provided its response to RAIs 410.96 and 410.97 on March 30, 1993 (NUDOCS Accession No. 9304050085); to RAI 410.100 on April 29, 1993 (NUDOCS Accession No. 9305050162); to RAI 410.99, 410.101, and 410.104 on May 14, 1993 (NUDOCS Accession No. 9305190237); to 410.241 on July 15, 1994 (NUDOCS Accession No. 9407250275); and to 410.243 on July 29, 1994 (NUDOCS Accession No. 9408040228).

Westinghouse Response:

The following HVAC systems are discussed in various portions of this response: radiologically controlled area ventilation system (VAS), nuclear island nonradiological ventilation system (VBS), containment air filtration system (VFS), health physics and hot machine shop HVAC system (VHS), and radwaste building HVAC system (VRS).

A. Re: VBS and Regulatory Guide (RG) 1.29
The applicable portions of the response to AP600 RAI 410.96 are repeated below in italics and the updated Westinghouse responses follow in normal font.

"Compliance with Position C.1 of RG 1.29 does not apply, except for the MCR isolation dampers, because VBS does not perform any other safety-related functions. The MCR isolation dampers are designed to meet seismic Category I requirements."

The portion of the response to AP600 RAI 410.96, above, remains valid for the AP600 VBS and is applicable to the AP1000 VBS except that the isolation "dampers" are "valves" as described in subsection 9.4.1.2.1.1.

"Compliance with Position C.2 of RG 1.29 is satisfied because the nonsafety-related portions of the VBS inside the MCR are designed to meet seismic Category II requirements so that the failure of the VBS components during an SSE will not reduce the functioning of any safety-related plant features."

The portion of the response to AP600 RAI 410.96, above, remains valid for the AP600 VBS and is applicable to the AP1000 VBS as described in subsection 9.4.1.1.1.

B. Re: VBS and General Design Criteria (GDC) 4
The applicable portion of the response to AP600 RAI 410.97 is repeated below in italics and the updated Westinghouse response follows in normal font.

"The main control room (MCR) is located at elevation 117'-6" of the auxiliary building, which is a missile-protected seismic Category I structure. Conformance to GDC 4 tornado missile requirements is discussed in SSAR Section 3.3. Conformance to GDC 4 internally generated missile requirements is discussed in SSAR Section 3.5.

With the exception of the MCR isolation dampers, the nuclear island nonradioactive ventilation system (VBS) is not a nuclear safety-related ventilation system. The MCR isolation dampers are located in the MCR envelope and are protected from missiles. The VBS has no other safety-related design function."

The portion of the response to AP600 RAI 410.97, above, remains valid for the AP600 VBS and is applicable to the AP1000 VBS as described in subsections 3.3, 3.5 and 9.4.1, except that the isolation "dampers" are "valves" as described in subsection 9.4.1.2.1.1.

C. Re: VAS Fuel Handling Area Subsystem and RG 1.29, RG 1.140, RG 1.13
Portions of the response to AP600 RAI 410.99 are repeated below in italics and the Westinghouse has provided updated responses in normal font. Please be advised that some of the subsections referenced in the original A600 RAI responses have changed since the AP600 RAI response was written. In those instances, Westinghouse has included current subsections numbers in brackets, { }, to identify the present subsection number.

Re: RG 1.29
"Fuel Handling Area Subsystem:

Compliance with Position C.1 of Regulatory Guide 1.29 is not applicable to the fuel handling area HVAC subsystem because this HVAC subsystem does not perform any safety-related functions. The calculated radiological releases discussed in SSAR Subsection 15.7.4.2 do not take credit for HVAC isolation or filtration after a design basis fuel handling accident. Because this HVAC subsystem does not perform any safety-related functions, it is not required to remain functional after a safe shutdown earthquake (SSE.)

Compliance with Position C.2 of Regulatory Guide 1.29 is satisfied because the fuel handling area subsystem HVAC subsystem is evaluated for interaction with seismic Category I systems as described in SSAR Subsection 3.7.3.13 so that the fuel handling area HVAC subsystem cannot reduce the functioning of any safety-related plant features.

Compliance with Position C.3 of Regulatory Guide 1.29 is satisfied because the VAS fuel handling area HVAC subsystem does not interface with any seismic Category I components, and the connection of VAS nonseismic Category I equipment and duct supports to Category I structures will not reduce functioning of seismic Category I structures. The containment air filtration system (VFS) provides filtered exhaust from the fuel handling area when high airborne radioactivity is detected (SSAR Subsection 9.4.7.1.2). The nonsafety-

related VFS includes seismic Category I containment penetrations that maintain containment integrity during containment isolation (See SSAR Subsection 9.4.7.1.1). The containment isolation system is designed and fabricated in accordance with the seismic categories assigned by Regulatory Guide 1.29 (SSAR Subsection 6.2.3.1.1) and complies with Position C3 of Regulatory Guide 1.29 as noted in SSAR Appendix 1A.

Compliance with Position C.4 of Regulatory Guide 1.29 is satisfied because the pertinent Quality Assurance requirements of 10 CFR 50, Appendix B are applied..."

The portions of the response to AP600 RAI 410.99, above, remain valid for the AP600 VAS and are applicable to the AP1000 VAS as described in the subsections noted and in Chapter 17, Quality Assurance.

Re: RG 1.140
RG 1.140 has been revised since the issuance of the AP600 Design Certification. Please refer to the response to AP1000 RAI 410.007, which addresses the compliance of the AP1000 normal filtration systems used in AP1000 to the latest revision of RG 1.140.

Re: RG 1.13
"Compliance with Position C.4 of Regulatory Guide 1.13 to provide a controlled leakage building is satisfied because the spent fuel pool is completely enclosed by the radiologically controlled areas of the auxiliary building as shown in SSAR Figures 1.2-9 and 1.2-15 {AP1000 DCD Figure 1.2-9, 1.2-10 and 1.2-14}*... The auxiliary building is a seismic Category I structure (SSAR Table 3.2-2). The fuel handling area HVAC subsystem maintains the normal ambient air pressure slightly negative with respect to adjacent clean plant areas to provide controlled building leakage and monitoring of exhaust air discharge to the plant vent (SSAR Subsection 9.4.3.2.3).*

The ventilation and filtration systems comply with Position C.4 of Regulatory Guide 1.13 to limit the potential release of radioactive iodine and other radioactive materials by monitoring the airborne radioactivity in the normal exhaust duct from the fuel handling area. The normal (unfiltered) exhaust duct is automatically isolated when high exhaust airborne radioactivity is detected (SSAR 9.4.3.2.4){AP1000 DCD 9.4.3.1.2}*. The VFS provides filtered exhaust from the fuel handling area to maintain this area at a slightly negative air pressure during conditions of high airborne radioactivity (SSAR 9.4.7.1.2). As noted in our response to Q460.4D, the VFS exhaust filters include HEPA filters and charcoal adsorbers to filter radioactive particulates and iodine and are designed to comply with the guidelines of Regulatory Guide 1.140."*

The portions of the response to AP600 RAI 410.99, above, remain valid for the AP600 VAS and are applicable to the AP1000 VAS except that the referenced subsections have been updated to reflect current AP1000 subsections as applicable and as identified with brackets, { }.

Position C4 of RG 1.13 references RG 1.25. However, the AP1000 is designed to satisfy the requirements of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, " July 2000, which is an alternative to Regulatory Guide 1.25. Please refer to DCD 15.7.4 for a discussion of how the AP1000 complies with RG 1.183.

D. Re: VFS Containment Penetrations and Single Failure Criteria
The applicable portion of the response to AP600 RAI 410.100 is repeated below in italics and the Westinghouse has provided updated responses in normal font.

"System single failure criteria, based on SRP 9.4.5 guidelines, does not apply to the VFS except for the VFS containment penetrations that provide containment isolation. The containment isolation does meet single active failure criteria based on SSAR Subsection 6.2.3.1.1."

The portion of the response to AP600 RAI 410.100, above, remains valid for the AP600 VFS and is applicable to the AP1000 VFS as described in the noted subsection.

E. Re: VRS and RG 1.140
AP600 and AP1000 no longer incorporate a normal filtration system in the VRS design, please refer to the response to AP1000 RAI 410.011, which discusses this in further detail.

F. Re: VHS and RG 1.140
AP600 and AP1000 no longer incorporate a normal filtration system in the VHS design, please refer to the response to AP1000 RAI 410.011, which discusses this in further detail.

G. Re: VAS, VBS, VHS, and VRS and Positive or Negative Pressures
Please refer to the response to AP1000 RAI 410.021, which addresses how VAS, VBS, VHS, and VRS demonstrate that they maintain their served areas at certain positive or negative pressures with respect to surrounding spaces and/or outdoor atmosphere in the current designs of AP600 and AP1000.

H. Re: VBS Air Filtration Fan Location
The applicable portion of the response to AP600 RAI 410.243 is repeated below in italics and the updated Westinghouse response follows.

"The VBS supplemental air filtration unit supply air fan is located upstream of the air filtration unit in order to meet the requirements of Section 4.7.2, "Habitability Systems" of ASME N509-1989. Subsection 4.7.2.(c) states that "The makeup air fan shall be located upstream of the air-cleaning unit if the air-cleaning unit is in a contaminated space". Subsection 4.7.2.(e) states that "Recirculating system housing should be kept at a positive pressure if located outside the habitable boundary in a contaminated space or interspace". The interspace refers to all other space - contaminated or clean - where the nuclear air treatment system or its parts may be located. VBS supplemental air filtration units are located in the equipment room at elevation 135'-6' of the auxiliary building which could be a contaminated space after a radioactivity release event. Therefore, installing the supply air filter fan upstream of the supplemental air filtration unit and designing the unit housing for positive pressure are consistent with the design philosophy specified by ASME N509-1989. The positive pressure unit housing design enhance constructability, reduces cost, minimizes the unit housing and filter bypass leakage concern compared to a negative pressure unit housing design; therefore, VBS supplemental air filtration unit operability and maintainability is improved."

The portion of the AP600 RAI 410.243 response, above, remains valid for the AP600 VFS and is applicable to the AP1000 VFS. Note: The design is equivalent to recommendations in AG-1 Nonmandatory Appendix SA-B, Figure B-1410-2, scheme 19.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.002

Question:

420.2 (DCD 7.1)

One of the characteristics that is identified in BTP 14 is "security." It is applicable as a management characteristic for the software management, maintenance, and operations plans; a functional characteristic for the software requirement specifications, software architecture, software design specifications, codes, system build documents, and installation tables. Please describe how the security aspects are addressed in AP1000 planning and implementing plant-specific application software in generic digital platforms.

The specific regulations and guidance are as follows:

(1) IEEE 603 Section 5.9 requirements addressing "Control of Access."
(2) SRP Chapter 7, Section 7.1-C states that controls should address access via network connections or via maintenance equipment.
(3) SRP Chapter 7, Section 7.9, "Data Communication (DCS)," states that the DCS does not present an electronic path by which unauthorized personnel can change plant software or display erroneous plant status information to the operators.

Westinghouse Response:

In addition to the overall control of access to areas and rooms of the AP1000 plant, the control of physical access to the AP1000 PMS is assured by the following means,

- All PMS cabinets are locked. Access to keys will be under administrative control to ensure that only appropriate personnel have access to the PMS equipment

- All PMS cabinets have door open alarms that alert the plant operators whenever a cabinet door is open

Software control for the alternative AP1000 PMS platforms is provided as follows,

For the Westinghouse Eagle product,

- Software is loaded onto EEPROM memory using hardware and software that requires an identification number and password to access.

- When new software is loaded into a PMS cabinet, the system processor board must be removed to install the new EEPROM memory. If this is attempted while the system is running, hardware and software failure alarms will be generated.

- Each EEPROM memory has a verification identifier called a checksum loaded onto it when its software is loaded. This checksum is verified constantly against a checksum calculated by the PMS application software as it executes. Any failure to verify the checksum during system operation will cause system failure alarms to be generated.

For the Westinghouse Common Q product,

- Software is downloaded from either the PMS maintenance and test panel or a portable workstation using a specialized configuration tool for the Common Q platform. A limited number of copies of the tool would be available at the plant and access to the maintenance and test panel or a workstation with the tool would require an identification number and password.

- To download software, the maintenance and test panel or portable workstation must be connected by a cable to a serial port on the front of the processor module. The software download process will cause the application program executing on the processor to halt. If this is attempted while the system is running, system failure alarms will be generated.

- After the software is downloaded to a processor, a verification identifier called a cyclical redundancy check (CRC) is generated that is unique to that particular download. The CRC is loaded into the maintenance and test panel as a configuration constant that is verified constantly against a CRC calculated by the PMS application software as it executes. Any failure to verify the CRC during system operation will cause system failure alarms to be generated.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.025

Question:

420.25 (DCD 7.1.7, item8)

Describe the program language to be used for the Common Q system (If information has been provided in the Reference 8, please identify specific section). Justify why that language was chosen for the safety system application.

Westinghouse Response:

There are two building blocks in the Common Q System that require Westinghouse software development: The Advant AC160 Controller (AC160) and the Flat Panel Display System (FPDS).

AC160

The programming tool for the AC160 is called Function Chart Builder (FCB). The software engineer develops a Function Chart in an MS Windows PC environment. An FCB program is comprised of a set of Process Control (PC) and Database (DB) elements. PC elements are control blocks that can be chained together to represent an algorithm. DB elements represent database entities such as I/O points and communications variables. DB elements can be used connected to input and output terminals of the PC elements. As a minimum a FCB program has at least one control module (CONTRM) and one or more PC elements subordinate to the CONTRM. The CONTRM is a periodic executable unit.

Once a programmer develops an FCB program comprised of PC and DB elements, the program is compiled to a downloadable image for the AC160. From the FCB program, one can connect to the AC160 to diagnose the controller and to download the FCB image to RAM and store the image in non-volatile memory for a cold start.

The advantages to using an FCB language for safety-related applications are as follows:

1. The FCB performs data type checking when making connections to PC elements. This helps prevent the programmer from inadvertently using the wrong data as input to a PC element.

2. There is no dynamic reference to memory. This helps to prevent incorrect addressing of memory that can cause system or program crashes.

3. The FCB eliminates runtime errors associated with incorrect interfaces to the operating system. Due to the first two advantages listed above, programmers will not have to debug runtime errors caused by incorrectly calling operating system interfaces or incorrectly addressing memory. This can save significant debug time.

4. Compilation errors are rare as well since the FCB checks for correct data types, the proper PC element interconnections, enforces the use of predefined execution modules, and anything manually entered is validated by FCB so there are no syntactical errors.

The PC Element library provided by the manufacturer is designed for digital control logic. For safety-related systems where intensive calculations are required, the PC Element library can be enhanced by creating Custom PC elements.

Custom PC elements can be used in the FCB just like any other type of element after they are created. However, their creation requires a different software development environment.

Custom PC elements are written in the C language and compiled into Motorola S Records using the Microtec compiler and linker. Each C language program is encapsulated into each Custom PC element. The inputs and outputs for these C programs have to be clearly defined in order to work in the FCB environment. This forces an extra level of discipline in the software development process that is advantageous for safety-related systems. Westinghouse follows its Common Q Software Program Manual process for the development of these Custom PC Elements.

See CENPD-396-P, Rev. 01, "Common Qualified Platform," May 2000 (DCD 7.1.7, Reference 8), Sections 5.1.1 and 6.2.1.2, for additional details about the AC160 software.

<u>FPDS</u>

The software operating environment for the FPDS is the QNX4 operating system. Westinghouse specifically chose this operating system for its maturity in the commercial market place and its operating experience in safety critical applications. As of November 1999, the QNX4 operating system had 6,000 operating years of experience.

The display engine for the FPDS is a product from the same vendor as QNX4. It is called Photon. Westinghouse chose this display product to ensure its capability with the operating system. It too is a mature product with over 1,000 operating years of experience.

Westinghouse develops application software for the QNX4 environment in the C language using the development tools provided by the QNX4 vendor. All software development follows the Common Q Software Program Manual processes.

See CENPD-396-P, Rev. 01, "Common Qualified Platform," May 2000 (DCD 7.1.7, Reference 8), Sections 5.3.1 and 6.2.2.2, for additional details about the FPDS software.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.039

Question:

420.39 (DCD 2.5.2, item 9.c)

DCD 2.5.2 Design Description 9. C) states that the PMS does not allow simultaneous bypass of two redundant channels. Describe the design provision to implement this requirement.

Westinghouse Response:

Both of the alternative AP1000 I&C systems platforms (Westinghouse Eagle product and Westinghouse Common Q product) include features that prevent any channel from being placed into bypass if doing so would prevent a required protection function from being accomplished while meeting the single failure criterion. In the case of the AP1000 design, in which most automatic functions are 2/4 logic, this means that only one channel may be bypassed in a function at a time. When a channel is bypassed the logic reverts to 2/3 of the remaining channels.

The Westinghouse Eagle product prevents a second bypass from being applied in two ways. For a reactor trip function, bypasses are applied through the trip logic subsystems (Global Trip, Trip Enable and Dynamic Trip Bus) in the Plant Protection Subsystems (Integrated Protection Cabinets). These subsystems use a combination of hardware and software to generate a reactor trip signal to be transmitted to the reactor trip switchgear. The bypass status of every channel of every reactor trip function is shared among all of the divisions. For engineered safety features (ESF) actuation functions, the system level actuation logic is performed completely in the software of the Engineered Safety Features Actuation Cabinet. The bypass status of every channel of every function is shared among all of the safety divisions. Any attempt to bypass a second channel of an ESF actuation function would not be instituted by the actuation logic and plant operators would be so informed.

The Westinghouse Common Q product performs both reactor trip and ESF actuation functions completely in the system software. The bypass status of every channel of every function is shared among all of the safety divisions. Any attempt to bypass a second channel of a reactor trip or ESF actuation function would not be instituted by the actuation logic and plant operators would be so informed.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number· 440.074

Question:

As stated on page 15.2-11 of Chapter 15, conservative PRHR heat exchanger heat transfer coefficients are used in the analysis for the loss of ac power to the plant auxiliaries event.

A. Discuss the determination of PRHR heat transfer coefficients associated with the low flow rate caused by the RCP trip, and justify the conservatism of the calculated heat transfer coefficients by comparing with the test data applicable to the temperature, pressure and flow conditions during the loss of ac power event.

B. Perform sensitivity study and quantify the effects of measurement uncertainty for heat transfer coefficients on the calculated peak pressure and decay heat removal capability during the transient.

Westinghouse Response:

The PRHR heat transfer coefficients were developed based on full pressure, full temperature heat transfer tests as documented in WCAP-12980 Rev. 3 "PRHR Test Report" (Reference 1). This report was reviewed and approved by the NRC as part of the AP600 design certification, as documented in NUREG-1512 (Reference 2). The heat transfer coefficients developed from this test program were incorporated into LOFTRAN, as documented in WCAP-14234 Rev. 1 (Reference 3) that was also reviewed and approved as part of the AP600 Design Certification, as documented in NUREG-1512.

The AP1000 PIRT and Scaling Assessment Report (WCAP-15613, Reference 4) and the AP1000 Code Applicability Report (WCAP-15644, Reference 5) were reviewed and approved by the NRC as part of the AP1000 Pre-Application Review. In these reports, Westinghouse demonstrates the applicability of the tests and analysis codes that were developed for the AP600 to the AP1000. These reports include our assessment of the applicability of the PRHR heat transfer tests, and the applicability of the LOFTRAN code to the AP1000. The reports conclude that the PRHR tests documented in WCAP-12980 Rev. 3 bound the conditions that are experienced in the AP1000 PRHR application, and the heat transfer coefficients applied in LOFTRAN are applicable to AP1000.

On pages 15.2-8 to 15.2-12, Section 15.2.6 of the DCD presents the analysis results of the loss of ac power to plant auxiliaries event. The loss of ac power causes the RCPs to coast down and reactor coolant flow decreases to low flow rate consistent with natural circulation flow. Additionally, the loss of ac power causes a complete loss of normal feedwater. The determination of the PRHR heat transfer coefficients is discussed in Chapter 9 of WCAP-12980.

The magnitude of the primary side flow rate impacts the PRHR primary side heat transfer coefficient. In WCAP-12980 the Dittus-Boelter correlation was found to provide reasonable agreement with the PRHR test data. Section 9.3 of WCAP-12980 discusses the selection the Dittus-Boelter correlation for calculating PRHR heat transfer and compares the correlation to PRHR test data.

As documented in WCAP-12980, the operating conditions for the PRHR tests were:

Single heat exchanger tube flow	-	0.3 gpm to 10.0 gpm
Primary water temperature	-	250 □F to 650 □F
Primary water pressure	-	50 psig to 2300 psig

Figures 440.074-1 and 440.074-2 show the PRHR tube flow and the primary temperature during the loss of ac power transient. The transient temperatures and flows are within the range of the PRHR test parameters. The normal operating pressure of the AP1000 is 2250 psia which is within the pressure range of the PRHR tests. During the loss of ac power transient, the primary pressure increases from the normal value (~ 2250 psia) up to a value slightly above the test range to the point at which the pressurizer safety valves open (~ 2500 psia). Because the primary side fluid remains single phase during non-LOCA transients, the impact of pressure on the primary heat transfer coefficient is much less significant than that of the temperature.

The loss of ac power causes a loss of feedwater and causes the reactor coolant pumps to coast down. The PRHR is started on low steam generator level. Later in the transient as the PRHR cools the plant, the core makeup tanks are actuated when the low cold leg temperature setpoint is exceeded. The acceptance criterion for this event is to show no pressurizer overfill. To meet the acceptance criteria, the PRHR must remove the sufficient energy to cool and shrink the reactor coolant fluid even while the CMTs are adding additional fluid to the RCS.

A sensitivity study was performed to evaluate the impact of the primary side heat transfer coefficient on the results of the loss of ac power event. In the sensitivity case, the primary side heat transfer coefficient calculated using the Dittus-Boelter correlation was reduced by 25%. The results from this sensitivity analysis (identified as Case 2) are compared with the loss of ac power results from the AP1000 DCD Section 15.2.6 (identified as Case 1). Figures 440.074-3 through 440.074-5 compare the results of both cases and Table 440.074-1 compares the sequence of events for both cases.

Figure 440.074-3 shows the overall PRHR heat transfer rate. Reducing the primary side heat transfer coefficient by 25% results in a small reduction in the overall PRHR heat transfer rate. The reduction in heat transfer delays the timing of peak values observed and there is no significant impact on the magnitude of the peak RCS pressure or the peak pressurizer water volume obtained.

The sequence of events presented in Table 440.074-1 are identical up to the point at which the PRHR is actuated. Following PRHR actuation the plant is cooled to the point at which the core makeup tanks are actuated on a low cold leg temperature signal. In Case 1, the core makeup tanks are actuated at 5329.4 seconds. In Case 2 with the primary side heat transfer coefficient reduced 25%, the core makeup tanks are actuated later at 7477.7 seconds.

Figure 440.074-4 compares the pressurizer water volume for Cases 1 and 2. In Case 1, the peak pressurizer water volume calculated was 1885 ft3 and in Case 2 with the heat transfer coefficient reduced by 25% the peak pressurizer water volume was 1882. The peak water volume is essentially the same in both cases and well below the point at which the pressurizer would fill.

The PRHR primary side heat transfer coefficient during the low flow conditions of the loss of ac power event is within the flow range of the PRHR tests used to select the correlation. The calculated decay heat removal capability, the peak pressure and peak pressurizer water volume are not strongly affected by the magnitude of the PRHR primary side heat transfer coefficient during the low flow conditions of the loss of ac power transients.

References

1) WCAP-12980, Rev. 3, "AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report," April 1997.

2) NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design Docket No.52-003," September 1998

3) WCAP-14234, Rev. 1, "LOFTRAN & LOFTTR2 AP600 Code Applicability Document," August 1997.

4) WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001

5) WCAP-15644, "AP1000 Code Applicability Report," May 2001

Table 440.074-1
Impact of PRHR Primary Side Heat Transfer Correlation (Dittus-Boelter) on the Sequence of Events for Loss of ac Power to Plant Auxiliaries

Event	Time (seconds)	
	Case 1 Dittus-Boelter Correlation Used (from DCD Section 15.2.6, see sheet 5 of Table 15.2-1)	Case 2 Dittus-Boelter Correlation Reduced 25%
Feedwater flow is lost	10.0	10.0
Low steam generator water level reactor trip setpoint is reached	70.4	70.4
Rods begin to drop, ac power is lost, reactor coolant pumps coast down	72.4	72.4
PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low startup feedwater flow rate)	132.4	132.4
Core makeup tank actuation on low T_{cold} "S" signal	5329.4	7477.7
Steam line isolation on low T_{cold} "S" signal	5341.4	7489.7
Pressurizer water volume peak is reached	22340 (1885. ft^3)	25664 (1882. ft^3)

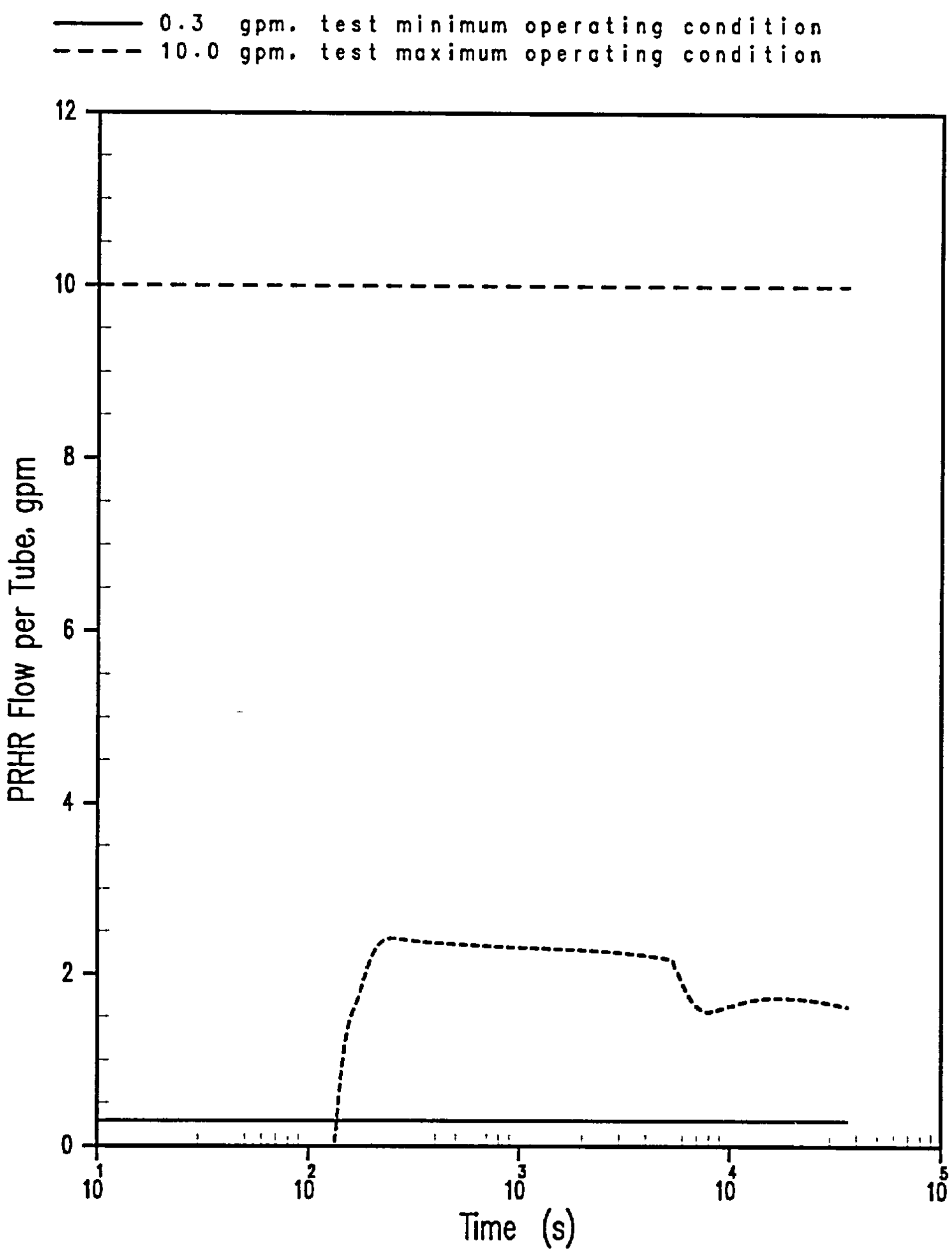


Figure 440.074-1 PRHR Tube Flow During a Loss of ac Power to Plant Auxiliaries

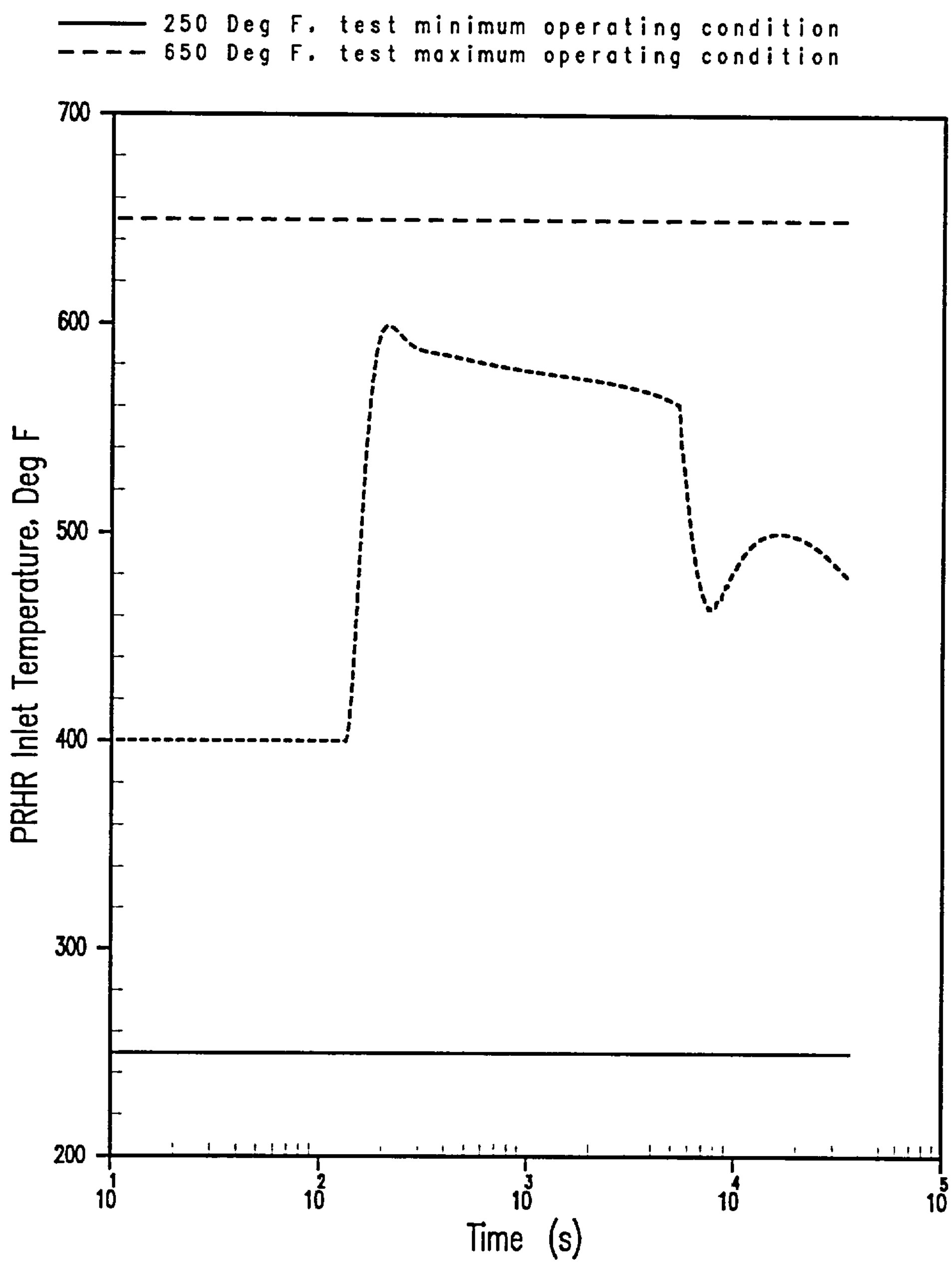


Figure 440.074-2 PRHR Inlet Temperature
During a Loss of ac Power to Plant Auxiliaries

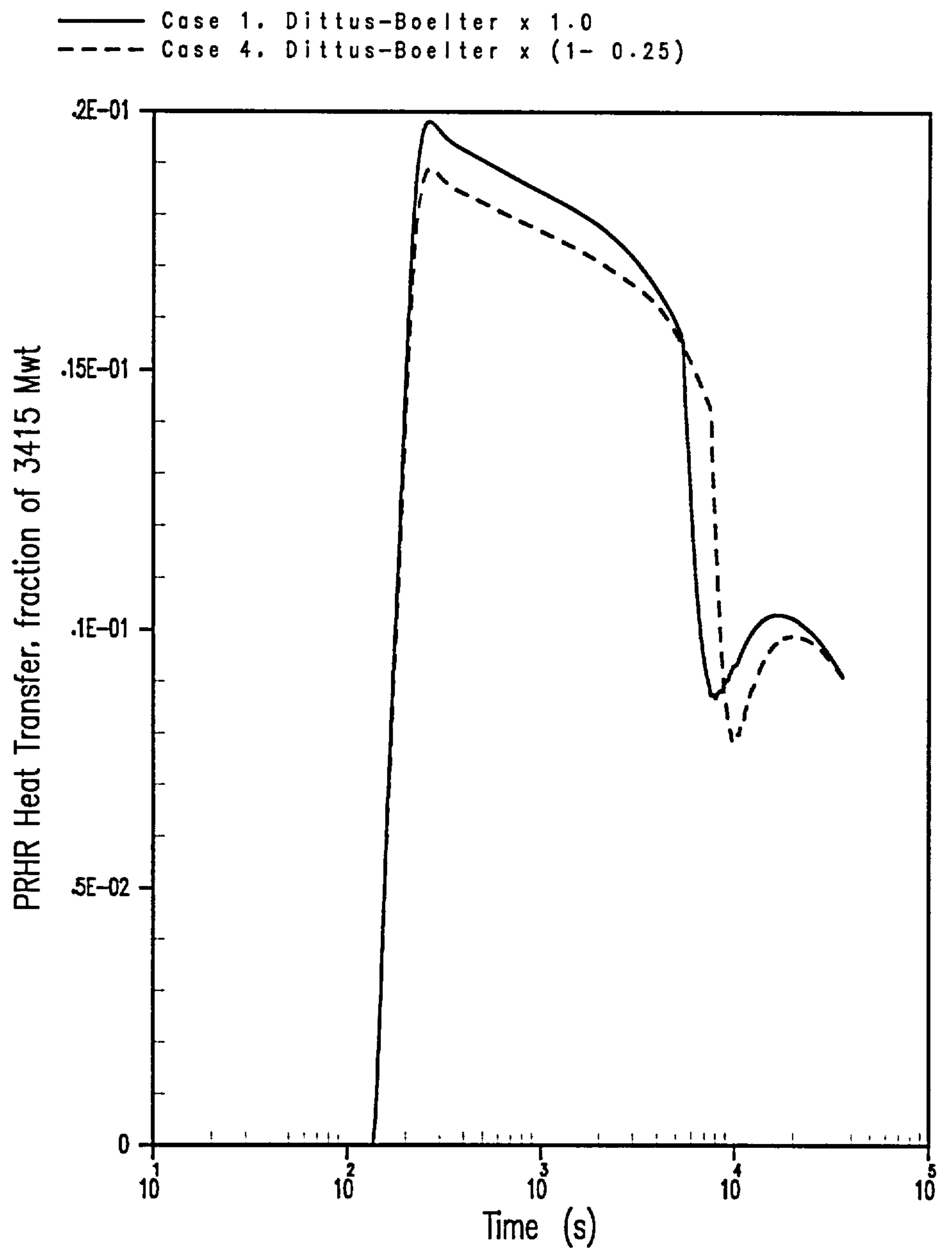


Figure 440.074-3 PRHR Heat Transfer During a Loss of ac Power to Plant Auxiliaries

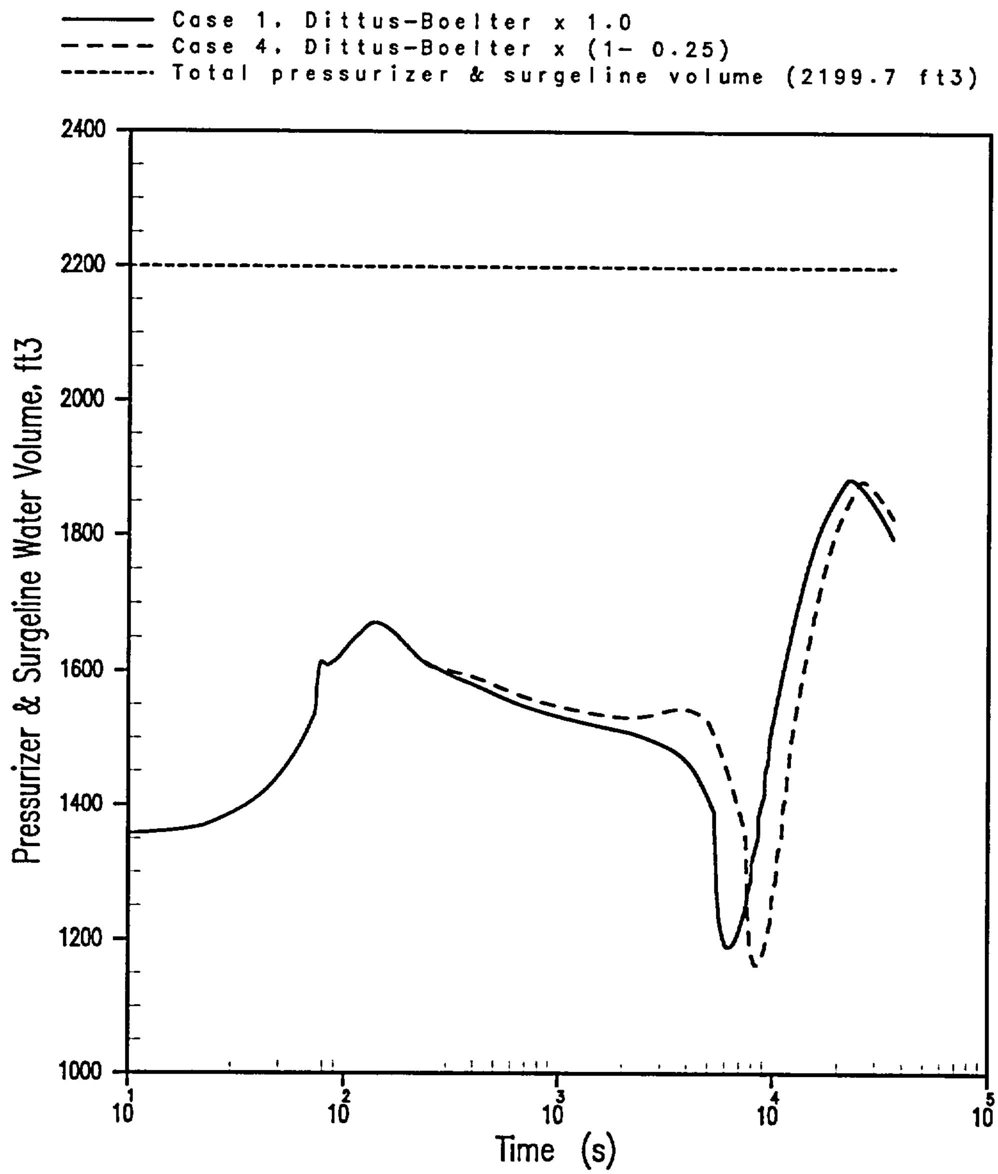


Figure 440.074-4 Pressurizer & Surgeline Water Volume During a Loss of ac Power to Plant Auxiliaries

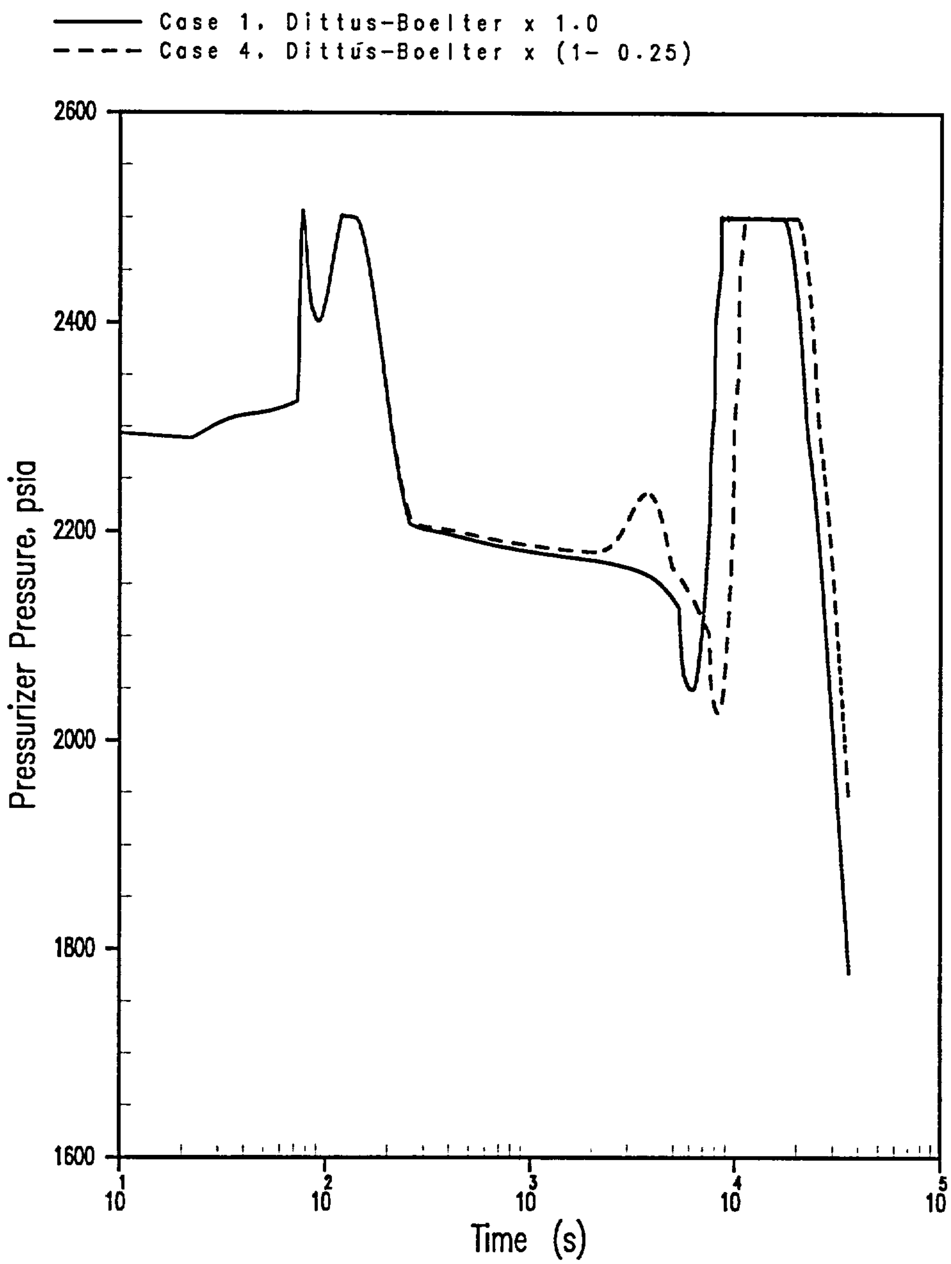


Figure 440.074-5 Pressurizer Pressure
During a Loss of ac Power to Plant Auxiliaries

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.109

Question:

Section 18.9 states that WCAP-14690, "Designer's Input To Procedure Development for the AP600," Revision 1, June 1997, provide input to the Combined License applicant for the development of plant operating procedures, including information on development and design of the AP600 emergency response guidelines (ERGs) and emergency operating procedures. It applies directly to AP1000 since AP1000 is operated in the same manner as AP600. Sections 19E.1.2 and 19E.3.3 reference the AP600 ERGs to address the shutdown operations issues for the AP1000 design, and states that the AP600 ERGs are applicable to AP1000 for the purpose of developing Emergency Operating Procedures.

A. Provide a discussion on the effects of the significant design changes from AP600 to AP1000 on the applicability of the AP600 ERGs to the AP1000 design. Provide technical bases to justify that the AP600 ERGs are applicable to AP1000 emergency response procedures, including shutdown operation issues.

B. Discuss the applicability of the transient data presented in background sections of AP600 ERGs to AP1000 design.

C. Describe the verification and validation (V & V) program for AP1000 ERGs.

Westinghouse Response:

The Emergency Response Guidelines (ERG) provide technical guidance and recovery strategies for terminating accidents and transients that affect plant safety. The ERG contains the technical basis for constructing the Emergency Operating Procedures (EOP). The EOP will be developed using the functional guidelines from the ERG in a presentation philosophy that directs the operating staff to provide timely implementation. The development of the EOP for the AP1000 is a COL action item as specified in DCD Section 13.5. The format and features of the EOP will be determined by the Human Factors Engineering (HFE) design process as discussed in DCD Chapter 18. This process includes the development of EOP; development of functional requirements and design basis for the computerized procedure system and the detailed design of hardware and software and the final HFE verification and validation. The ERG contain the technical basis for constructing the EOP. Other licensing submittals, such as the AP1000 Design Control Document and Probabilistic Risk Assessment will also be used to support construction of the EOP. Design details related to the implementation of the functional guidance specified in the ERG are accommodated in the detailed development of the EOP.

Background

The ERG developed for the AP600 are suitable for the AP1000 for the purposes for which they are intended – namely to provide the starting point for the development of the EOP as part of the HFE process. The ERG provide symptom-based, as opposed to event-based guidance to the operator. For that reason, the ERG do not immediately instruct the operator to attempt to diagnose an event. The ERG guide the operator to assess the plant parameters and operability of the available systems, and provide the most straightforward direction to the operator.

The AP600 and AP1000 employ the same passive safety-related systems that significantly reduces the burden on the operator in an accident scenario when compared with currently operating nuclear power plants. The passive safety systems automatically protect the plant in the event of an accident, without the need for immediate operator actions. The AP600 and AP1000 also employ the same nonsafety-related defense-in-depth systems that, if available, can automatically protect the plant for the more probable postulated transients and accidents. If these systems are available and operate correctly, they will generally prevent the need for the passive safety-related systems. The AP600 ERG integrate the use of the nonsafety-related defense-in-depth systems and the passive safety-related systems to maximize the protection of the plant for design basis and beyond design basis accidents. In general, the ERG address more probable events, and also contain contingencies for beyond design basis accidents. The design of the AP600 and AP1000 are functionally the same with respect to the role of the passive safety systems and active systems provided for defense-in-depth. The symptom-based approach contained in the AP600 ERG allow for them to be used as the starting point to develop the detailed EOP as part of the HFE design process for the AP1000.

The use of the AP600 ERG for the AP1000 is similar to the implementation of the Standard ERG for Westinghouse Operating Plants. The AP600 ERGS are in a format consistent with the Standard Westinghouse ERG. Westinghouse developed two standard sets of ERG for traditional Westinghouse operating plants. The standard Westinghouse operating plants were categorized as either HP or LP plants, depending on whether the design pressure of the high-head safety injection pumps were higher than the operating pressure of the NSSS. Because the ERG are symptom-based, the functional guidance they provide is applicable to a range of plant designs that functionally perform in a similar manner. (For example, the LP ERG can apply to 2-loop, 3-loop or 4-loop plants that contain a range of NSSS and BOP system design features. Therefore, it is reasonable to expect that the AP600 ERG can be used as the starting point for the development of the AP1000 EOP.

The following provides specific responses to the items raised in this RAI.

A. The design philosophy of the AP1000 is to make only those changes necessary to accomplish the higher power output. This resulted in larger NSSS components such as the reactor vessel, steam generators and reactor coolant pumps, as well as a larger containment and turbine island. The capacity of the passive safety systems and active nonsafety systems are increased as necessary to maintain the required safety and operating margins. The configuration of the passive safety systems is the same for both the AP600 and AP1000 plants. Therefore the role of the passive safety systems in mitigating the consequences of an accident are the same.

As stated above, the design details related to the implementation of the functional guidance specified in the ERG are accommodated in the detailed development of the EOP. As part of the EOP implementation, specific design details of the AP1000 will be incorporated into the EOP. Two ERG/EOP related differences have been identified in the AP1000 PRA. These are discussed below as examples of how these types of differences will be addressed in the AP1000 EOP development:

- A functional change has been incorporated in the design of the normal residual heat removal system (RNS) to avoid an adverse system interaction that was identified in the AP600 design. Specifically, the design of the RNS has been changed regarding its operation post-accident. The AP1000 RNS will not be used to inject water from the IRWST following a postulated LOCA. Instead, the RNS is aligned to take suction from the Cask Washdown Pit, and if available, will perform a defense-in-depth function of low pressure RCS makeup in response to an accident. This configuration eliminates the potential adverse system interaction of draining the IRWST following a DVI line break, such that containment recirculation would occur earlier than necessary. The AP1000 RNS configuration eliminates this possibility, and if the RNS is available, it delays the time that containment recirculation would be required. This configuration detail is not included in the AP600 ERG. The ERG contain steps to turn on the RNS for low pressure RCS makeup. In AE-0, Step 30 specifies:

  e. Align RNS to inject into RCS

  f. Verify proper valve alignment
  [Include additional AP600 details on EOPs]

  As seen in this example, the design details of this step is not affected by the AP1000 design change identified.

- The AP1000 Level 2 PRA recommends that Function Restoration Guideline (AFR.C1) be modified to stipulate that the cavity flooding be performed by the operator immediately upon entering this guideline. (See the Westinghouse response to RAI 720.044 for a related discussion). This recommendation is based upon results of the phenomenological analyses performed by Westinghouse as part of the AP1000 PRA. As discussed above, other licensing submittals such as the AP1000 PRA will be used with the ERG in the development of the detailed HFE process as part of the COL Action Item to develop EOP.

  These two examples are provided to demonstrate how details of the AP1000 design and more specifically, how differences from AP600 will be implemented in the AP1000 EOP development. These small differences between AP600 and AP1000 are small in comparison to differences that exist among the HP and LP category plants using the standard Westinghouse ERG.

B. The analysis provided in the ERG Background documentation is suitable for the purpose for which it is intended. Specifically, it provides an example of the role of the operator in performing actions outlined in the ERG. Note that these analyses are specific to the AP600. However, as discussed above, the AP600 and AP1000 designs are similar, and functions performed by the passive and active systems are the same. Because the ERG are symptom-based (as opposed to event-based), the actions outlined in the AP600 ERG are the same as for the AP1000. The timing of the specific accidents analyzed in the Background documents may be slightly different for the two plants, however the response of the operator to any particular plant symptom (i.e. core criticality, RCS system pressure, temperature, water levels, containment pressures and temperatures) or system operating status (i.e. RCP status, CMT status, CVS makeup pump status) is the same.

C. Verification and Validation of the AP1000 ERG is not a required activity of the HFE design process. Verification and Validation of the EOP will be performed as part of the HFE design process.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.154

Question:

Section A.4.4 presents results of a sensitivity study using the Top Offtake Orientation model. In general, the onset for entrainment for a top offtake configuration is:

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C2} \tag{1}$$

where d is the diameter of the branch line.

In the sensitivity study, the values of C_1 and C_2 were varied from their reference values of C_1=0.355 and C_2=2.5. The results were intended to show that the AP1000 performance for the Inadvertent ADS case has little sensitivity to the Top Offtake Orientation correlation.

(a) Since hot leg entrainment is sensitive to steam velocities in the hot leg, show why the Inadvertent ADS case is sufficient to characterize or bound AP1000 performance for other accident scenarios such as a small cold leg break or a direct vessel injection (DVI) line break.

(b) When either the coefficient C_1 is increased or the exponent C_2 is decreased in the sensitivity study, the minimum in-vessel inventory increases and the start of in-containment refueling water storage tank (IRWST) injection is delayed compared to the Reference case. However, when C_1 is decreased, the vessel inventory minimum again increases and the IRWST injection is delayed. Please explain why both increasing and decreasing the onset of entrainment results in this sensitivity.

(c) Provide justification that the variation of the Top Offtake Orientation model in Section A.4.4 is sufficient to account for inaccuracies in the model when compared to experimental data. Figure 2 shows the variations considered in the sensitivity study. When data from ATLATS tests are compared to the correlation and its variations in the sensitivity study, it can be seen that the branch line quality is grossly over-estimated for the two cases in which the offtake quality was reduced compared to the Reference. For example, at Froude numbers less than one the data suggests very low branch line quality while the correlation and its variations predict very high quality (x >0.75).

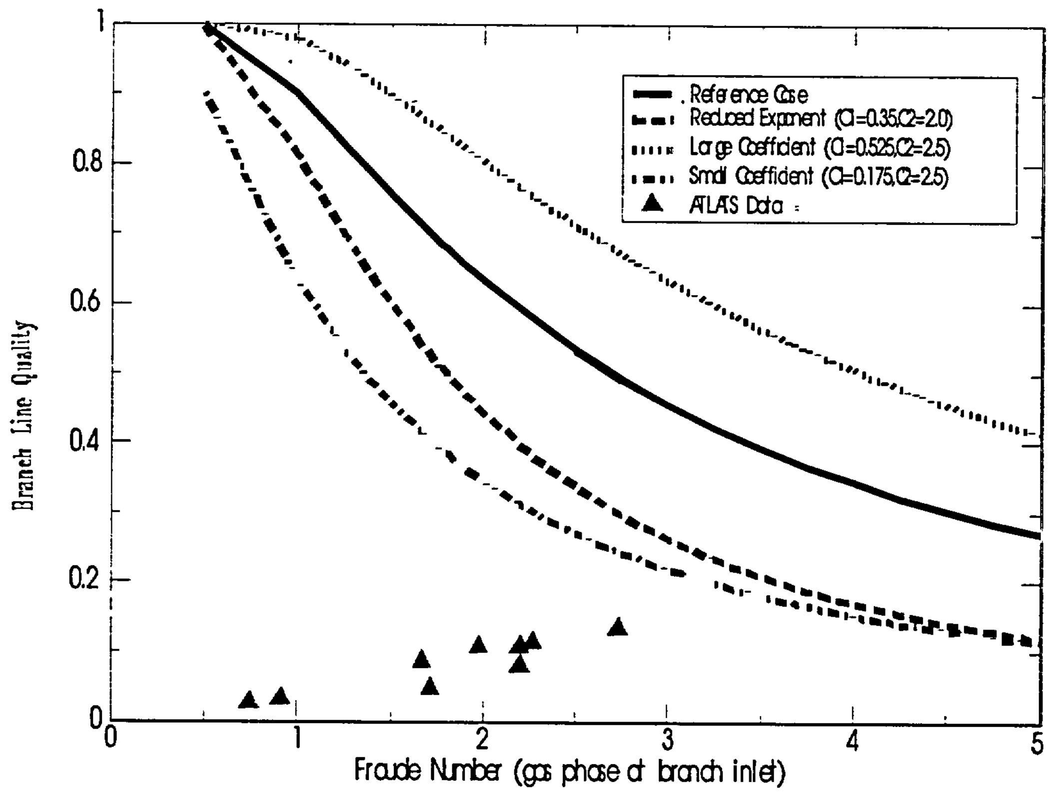


Figure 2. Variation in Branch Line Quality of Entrainment Onset Correlations in Westinghouse Sensitivity Study

Westinghouse Response:

a) A comprehensive set of figures is provided in the response to RAI 440.163 for both the DEDVI break case and the Inadvertent ADS scenario. The flow regime maps for both hot legs are shown in RAI 440.163 for both of the cases. The hot leg behavior of the DEDVI case is such that during the interval from the time at which all the ADS-4 flow paths have opened to the beginning of IRWST injection, the flow regime is almost always horizontal stratified flow. In contrast, there are instances of countercurrent flow in the Inadvertent ADS case during this interval, due at least in part to lower hot leg velocities. Therefore, there are greater flow regime variations predicted for the Inadvertent ADS scenario during the most important portion of the depressurization transient. In order to study the impacts of perturbations over a wider hot leg flow regime range, the judgment is that the Inadvertent ADS scenario is an appropriate case to characterize the sensitivity of AP1000 performance to parameter variations in WCOBRA/TRAC. Small RCS loop break cases would likely be little different from the Inadvertent ADS case in behavior.

b) The attached Figure 440.154-1 compares the downcomer pressure during the ADS-4 IRWST initiation phase for the increased entrainment onset correlation coefficient (solid line) and the decreased entrainment onset correlation coefficient (dashed line). The time scale is the same as in Appendix A.4 of WCAP-15833. Prior to 65 seconds, only one ADS-4 flow path is open. At 65 seconds, the pressure in the small coefficient case is about 3 psi higher than in the large coefficient case. This result is consistent with Figure 154-2 in that the higher branch line quality associated with the large coefficient might be expected to produce a more effective depressurization of the AP1000 than that observed in the small coefficient case, with its lower quality. However, once the additional two ADS-4 flow paths become available at 65 seconds, the pressures converge. This indicates that the AP1000 design has adequate depressurization capability that, even assuming a single failure, the pressure result during ADS-4 operation is not sensitive to the variation in the entrainment onset correlation coefficient.

Figure 440.154-2 compares the hot leg levels in the loop that contains at least one open ADS-4 flow path throughout the ADS-4 IRWST initiation phase for the increased entrainment onset correlation coefficient (solid line) and the decreased entrainment onset correlation coefficient (dashed line) cases. Prior to 65 seconds a difference is established with the reduced coefficient case exhibiting a lower level, which is equivalent to a larger distance "h" between the top of the hot leg and the liquid surface. However, once the additional two ADS-4 flow paths are available at 65 seconds, the collapsed levels become very similar, again indicating a lack of sensitivity to the entrainment onset correlation. The minimum mass inventory results of these cases and the base case are all within 1000 lbm of one another out of a total minimum inventory of 71000 lbm.

c) As noted in the response to RAI440.151, it is reasonable to expect the form of the correlation in WCOBRA/TRAC to remain valid for the AP1000 ADS-4 configuration, with only variation in the coefficient (C1) or exponent (C2). To investigate behavior, sensitivity calculations of entrainment from the hot leg to the ADS-4 offtake in AP1000 were performed with the WCOBRA/TRAC-AP models where the coefficient (C1) and exponent (C2) associated with the entrainment inception correlation were varied as described in Appendix A.4 of WCAP-15833. These calculations indicated that the horizontal flow regime for AP1000 during the ADS-IRWST transition phase of a SBLOCA is primarily stratified in the hot legs upstream of the ADS-4 off-takes. Hence, the correlations are applicable. Also, there is little sensitivity to variations in the coefficient (C1) or in the exponent (C2) associated with the entrainment inception correlation. At the AP1000 Froude number of $\geq$2, a variation of $\pm$ 50% in the branch line quality value is indicated in the figure above for the sensitivity study investigation performed with WCOBRA/TRAC-AP. This is judged to be an adequate range for the studies performed varying parameters in the hot leg entrainment correlations of WCOBRA/TRAC-AP.

With respect to addressing the data associated with the ATLATS separate effects test facility, Westinghouse does not have detailed information regarding these tests (i.e. facility layout, boundary conditions, test procedures, scaling analysis, etc.) and therefore these tests have not been analyzed by Westinghouse. A possible explanation of the different behavior displayed in ATLATS relative to the stratified flow type entrainment behavior expected in AP1000 and seen in other test facilities is that the ATLATS test facility is producing a different flow regime that may be attributed to its non-prototypic and/or incomplete simulation of the actual AP1000 configuration.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.161

Question:

Please provide Figure 2-15 with a key, or clearly indicate what each of the three curves represent. In addition, clarify the as-coded expression for the condensation heat transfer coefficient. (On page 2-15, it is claimed that for condensation heat transfer the "dependence on x is neglected" in the WCOBRA/TRAC-AP model, but on 2-21 it is stated that condensation heat transfer "rises with distance proportional to $x^{0.1}$.")

Westinghouse Response:

The statement on page 2-15 of WCAP-15833 is correct, and Equation (2-31) is the heat transfer coefficient expression used in the WCOBRA/TRAC computer code. The page 2-21 statement refers to the form of the Jensen interfacial heat transfer correlation, and [ ]^{a,c} Figure 2-15 is to be deleted from WCAP-15833,Section 2 as discussed in the response to RAI 440.176.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP-15833 Revision:

Figure 2-15 deleted. See the response to RAI 440.176.

RAI Number. 440.165

Question:

Both the DEDVI and Inadvertent ADS Actuation calculations in Sections 3.2.1 and 3.2.2 show flow qualities of roughly 0.20 (plus/minus 0.1) in the ADS-4 lines for the brief period simulated. Provide justification that Test SB18, which was used to demonstrate that WCOBRA/TRAC can adequately predict ADS-4 flows, has approximately this ADS-4 flow quality.

Westinghouse Response:

The ADS-4 flow quality is not available from the Test SB18 data for the ADS-4 IRWST initiation phase because the steam [

][a,c] The best approximation available for the ADS-4 flow quality is based on the core exit quality of Figure 5.2.2-47 in Reference 440.165-1. This figure shows the core exit quality during the ADS-4 IRWST initiation phase time interval [

][a,c]

A difference exists between the core exit quality and the ADS-4 exit quality in Test SB18 due to the ADS-4 flow contribution from the draining of the pressurizer. During the 937 – 1225 second time interval, the pressurizer mass loss is [

][a,c]. A more detailed look at the AP1000 DEDVI and Inadvertent ADS cases referenced in the RAI indicates the average ADS-4 flow qualities are 0.16 and 0.18 for the time interval from ADS-4 actuation onward to the start of IRWST injection. This is in reasonably close agreement with the Test SB18 value.

Reference:

1. WCAP-14292, Revision 1, *"AP600 Low-Pressure Integral Systems Test at Oregon State University Test Analysis Report"*, September, 1995.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.166

Question:

The comparisons between WCOBRA/TRAC and NOTRUMP in Section 3 show very large differences in results between the two codes. Why should it be expected that the starting conditions for the WCOBRA/TRAC simulations be the same as those from NOTRUMP?

The large deviations suggest that if WCOBRA/TRAC were used to predict the initial parts of each transient, the initial conditions for the WCOBRA/TRAC simulation would be significantly different.

If WCOBRA/TRAC were used to simulate the entire transient, why should it not be expected that there be considerably less vessel mass at the start of ADS-4/IRWST transition given the much higher ADS-4 flow rates predicted by WCOBRA/TRAC?

Westinghouse Response:

Should an AP1000 small-break model be developed using WCOBRA/TRAC to simulate the entire transient, good agreement with the test data would be expected at the ADS-4 actuation time in the test validation cases, and thus similar results to the NOTRUMP predictions would be expected. The WCOBRA/TRAC model utilized and documented in Reference 1, was developed to focus on the ADS-4 to IRWST initiation transition period. This is the time period where the NOTRUMP model is known to contain deficiencies due to the lack of a detailed momentum flux model in the ADS-4 flow paths and the lack of an entrainment model. Prior to the ADS-4/IRWST transition period, no significant deficiencies were identified in the NOTRUMP code, as documented in the NOTRUMP code validation report for the AP600 (Reference 2). Comparisons of the NOTRUMP simulations with results of the SPES-2 and OSU tests in Reference 2 have shown reasonable or good agreement with the test data as a whole. Sections 7.4.4 and 8.4.3 of Reference 2 indicate that NOTRUMP is conservative with respect to mass inventory following ADS-4 for a majority of cases performed. Therefore, validation of a complete WCOBRA/TRAC small-break model against test data would be expected to yield comparable results to that obtained with the NOTRUMP code at the ADS-4 actuation time, and pursuing such a WCOBRA/TRAC model is not justified by any need to increase the confidence in the NOTRUMP results.

The deviation in results between the WCOBRA/TRAC and NOTRUMP results presented in Reference 1 commence during the transition from critical flow to subcritical flow. The NOTRUMP model utilizes the Henry-Fauske/HEM break flow model during critical flow conditions and a break flow blending model during the transition to post-critical flow conditions for the ADS flow paths. The blending model has been demonstrated to under-predict flow during the transition to post-critical conditions. The WCOBRA/TRAC model utilizes the "Small Break LOCA break model" as discussed in Reference 3 for both the ADS-4 and DVI venturi flow rates. The break model differences and the lack of a liquid entrainment model in the NOTRUMP code cause the deviations to be pronounced following ADS-4 actuation. These deficiencies in the NOTRUMP code result in the delayed prediction of IRWST injection. The minimum vessel inventories observed between the two simulations are seen to be comparable, although the WCOBRA/TRAC minimum inventory occurs sooner due to the more rapid depressurization predicted post ADS-4.

References:

1. WCAP-15833, "WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," W.L. Brown and R.M. Kemper, May 2002.
2. WCAP-14807, Revision 5, "NOTRUMP Final Validation Report for AP600", Volume 2, R.L. Fittante et. al., August 1998.
3. WCAP-14776, Revision 4, WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report," March 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.174

Question:

On page 2-4 WCOBRA/TRAC is stated to use the Taitel and Dunkler flow regime map for horizontal flow. Five flow regimes are considered: Slug and plug, stratified smooth, stratified wavy, dispersed bubble, and annular/annular dispersed liquid flow. For the case of the DEDVI line break case in Appendix A to WCAP-15833, Revision 1, please provide the liquid and steam velocities and identify the flow regime for the hot legs between the reactor vessel and the ADS4 takeoff tees. Provide this information as a function of time between ADS4 activation and IRWST injection.

Westinghouse Response:

The information requested is included in the response to RAI 440.163, as Figures 26 and 27.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.175

Question:

On page 3-2 of WCAP-12945, Code Qualification Document for Best Estimate LOCA Analysis," Volume 1, the WCOBRA/TRAC flow regimes for vertical flow are described as small bubble, small to large bubble, churn-turbulent and film/drop. For the case of the double ended DVI line break case in Appendix A to WCAP-15833, Revision 1, please provide the liquid and steam velocities and identify the flow regime for the core exit, upper plenum to the hot leg elevation and in the ADS4 piping. Provide this information as a function of time between ADS4 activation and IRWST injection.

Westinghouse Response:

The requested upper plenum liquid and steam velocities for the AP1000 DEDVI break case in Appendix A of WCAP-15833, Revision 1 are provided in Figures 440.175-1 through 9. The upper plenum model contains two fluid channels, 15 and 47, and the (exit) flow rates of each are provided. Channel 15 represents the upper plenum volume directly above the holes in the upper core plate, and Channel 47 the free volume above the solid metal plate. The steam velocity at the Channel 47 exit into Channel 20 is relatively low, and a circulation flow of liquid is predicted in the upper plenum. The flow regime in upper plenum Channels 15 and 20 is the WCOBRA/TRAC film/drop regime throughout the time interval; in Channel 47, the flow regime alternates between the large bubble and film/drop regimes prior to 560 seconds, and is in the film/drop regime thereafter.

The ADS-4 flow path liquid and steam velocities are provided in Figures 440.175-10 through 13 for the piping in which both valves open and the piping in which the single failure occurs. The velocities shown are in the offtake pipe from the hot leg at a location immediately upstream of the hot leg connection. The flow regime predicted in the ADS-4 pipes is the TRAC annular mist flow regime virtually all the time interval in both ADS-4 paths

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 440.175-1 Core Exit Continuous Liquid Velocity, DEDVI Break Case

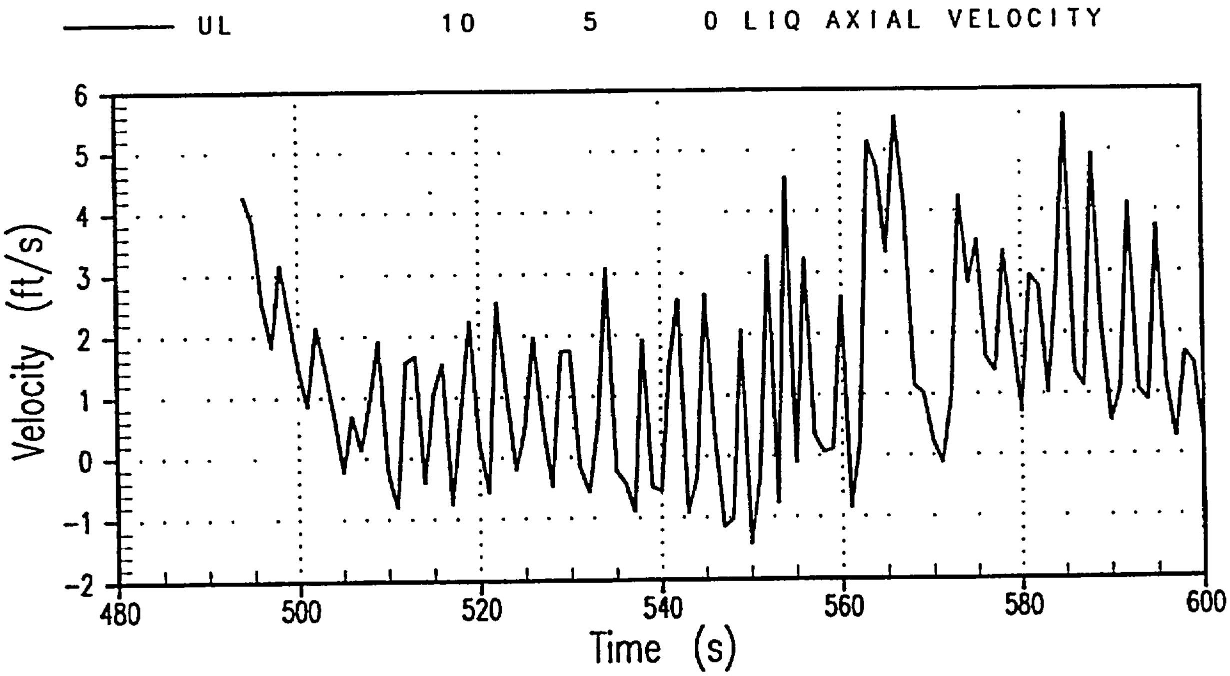


Figure 440.175-2 Core Exit Entrained Liquid Velocity, DEDVI Break Case

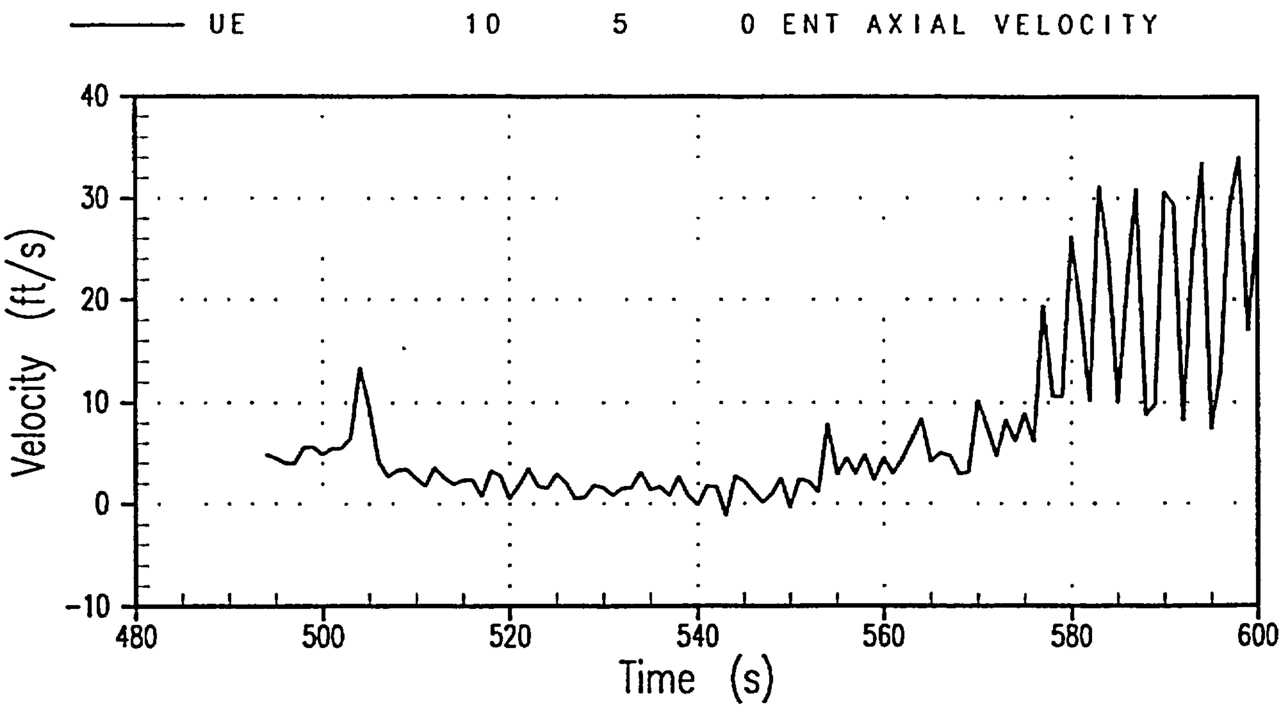


Figure 440.175-3 Core Exit Vapor Velocity
DEDVI Break Case

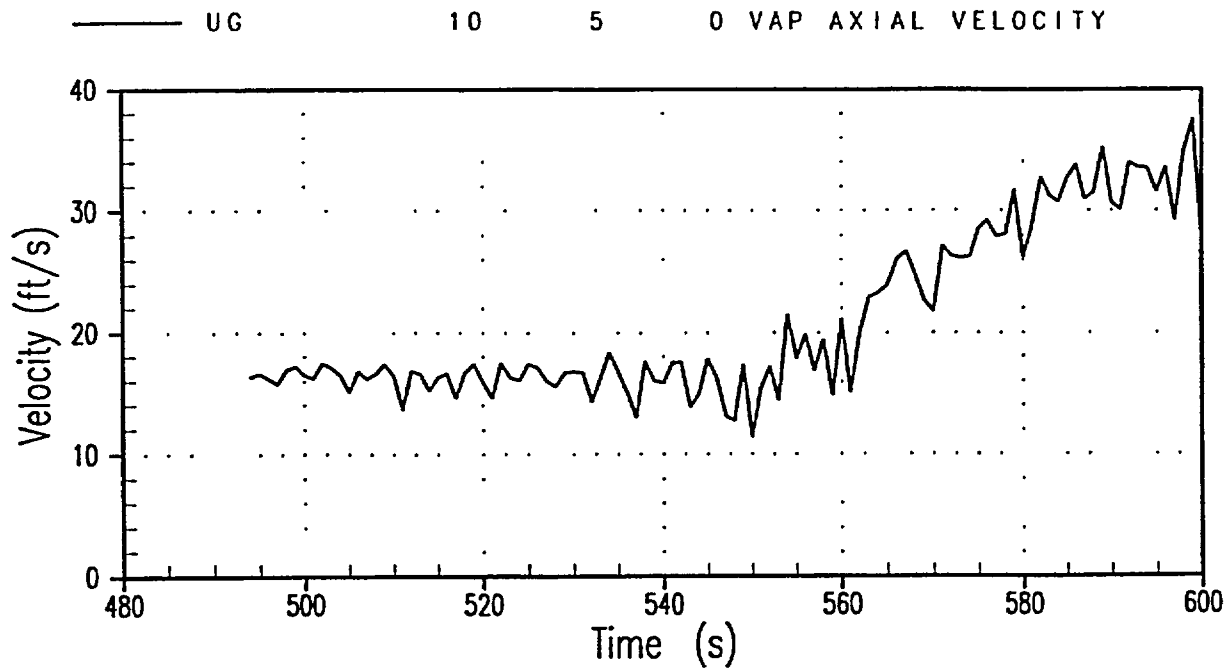


Figure 440.175-4 UP Node 15 Exit Continuous Liquid Velocity, DEDVI Break Case

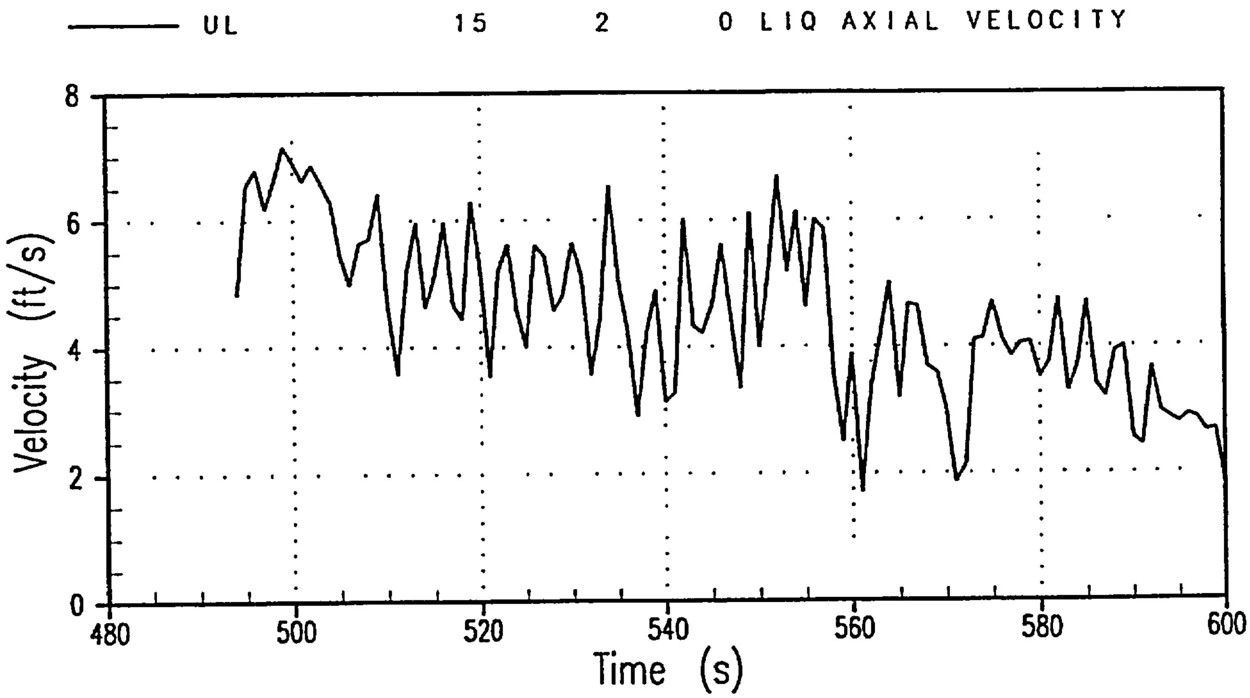


Figure 440.175-5 UP Node 15 Exit Entrained Liquid Velocity, DEDVI Break Case

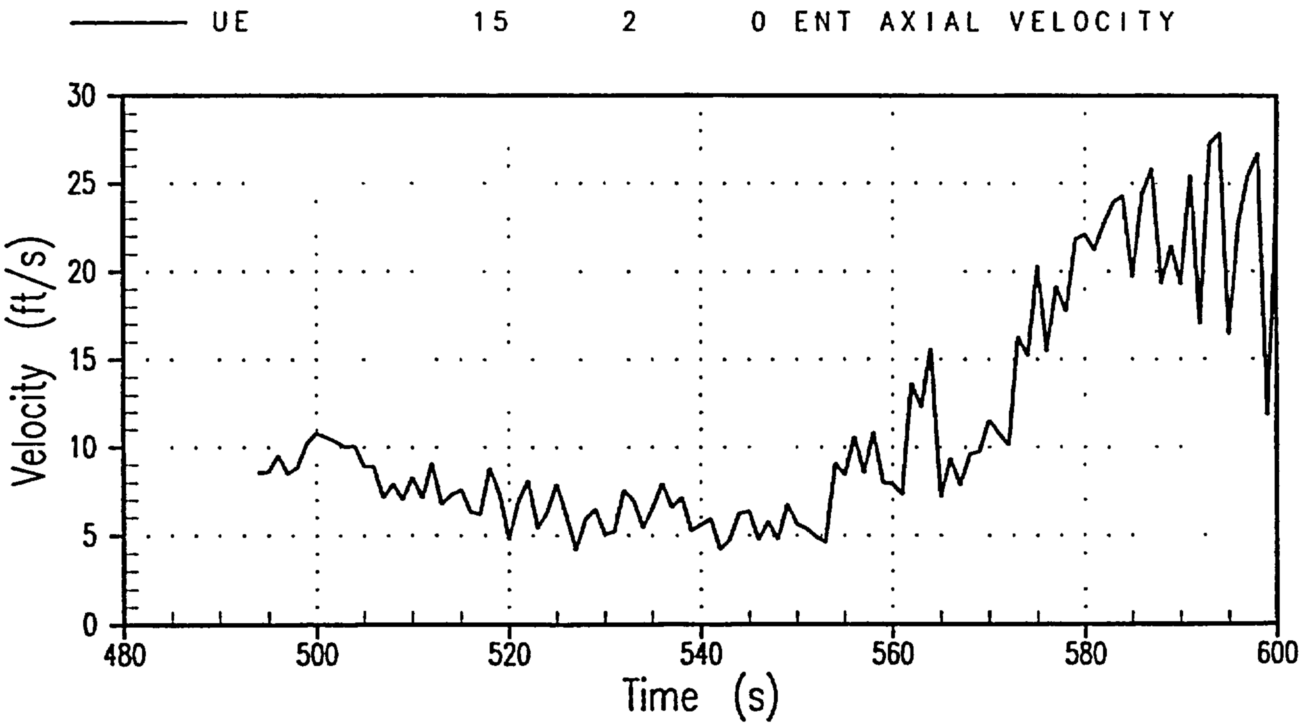


Figure 440.175-6 UP Node 15 Exit Vapor Velocity
DEDVI Break Case

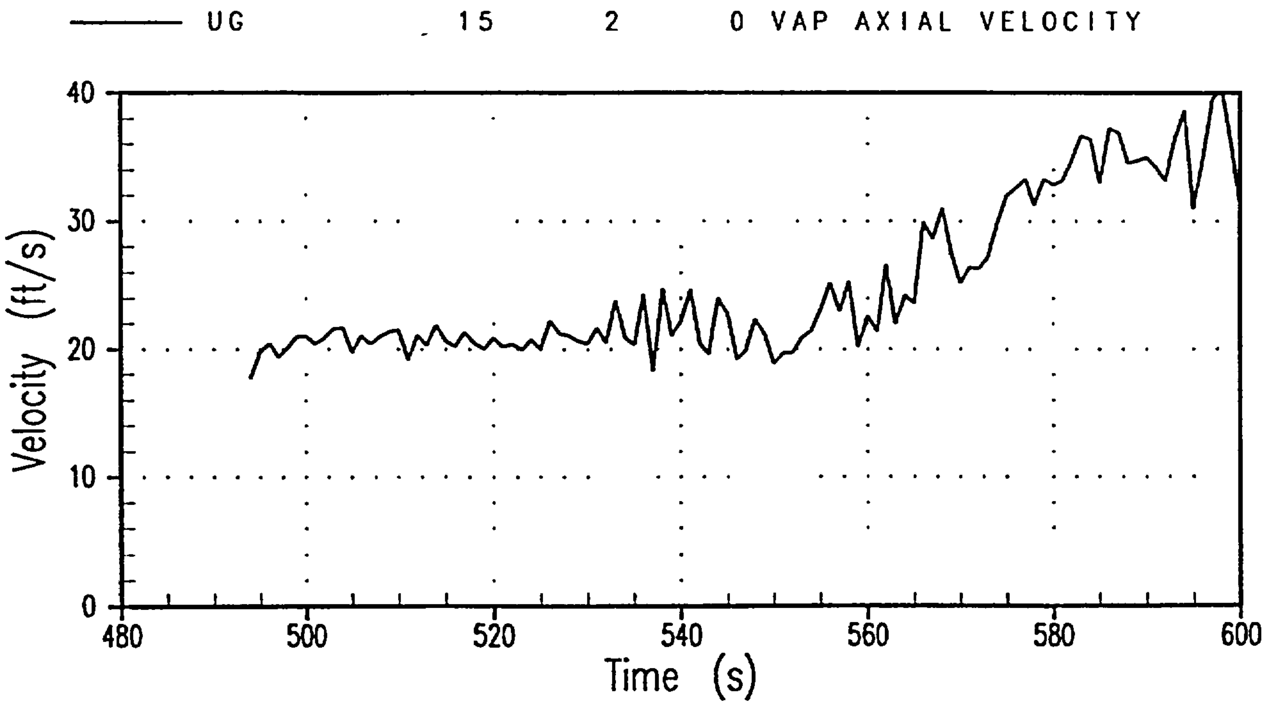


Figure 440.175-7 UP Node 47 Exit Continuous Liquid Velocity, DEDVI Break Case

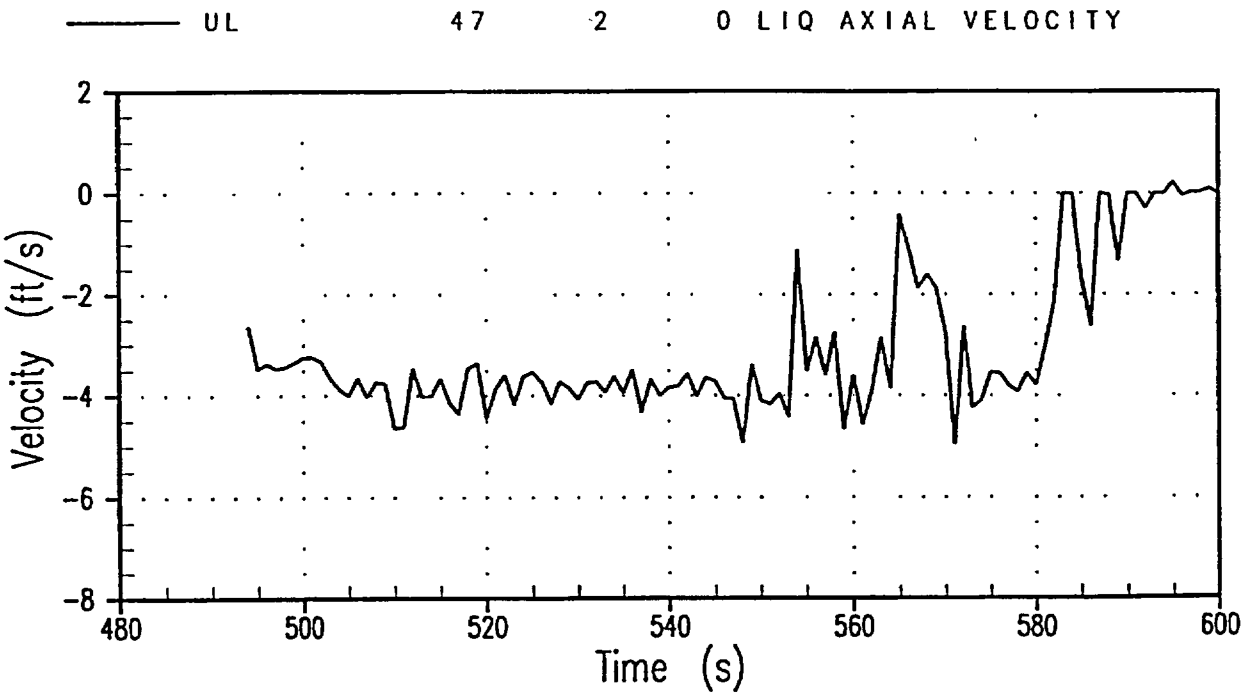


Figure 440.175–8 UP Node 47 Exit Entrained Liquid Velocity, DEDVI Break Case

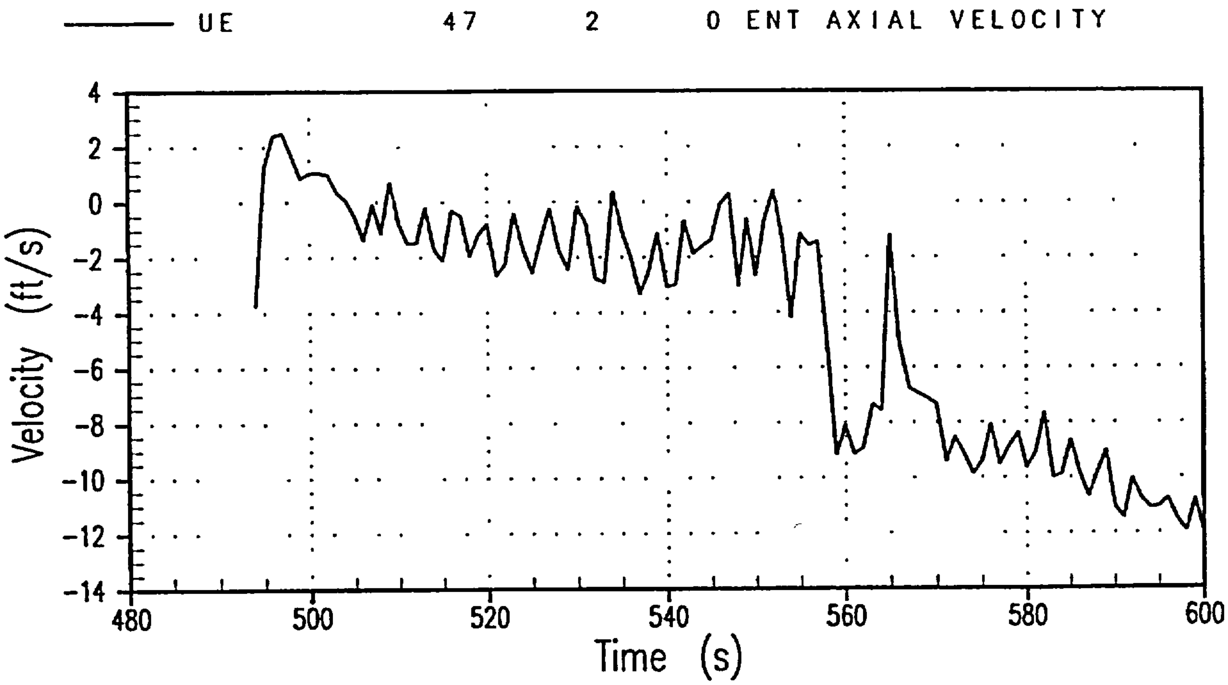


Figure 440.175-9 UP Node 47 Exit Vapor Velocity
DEDVI Break Case

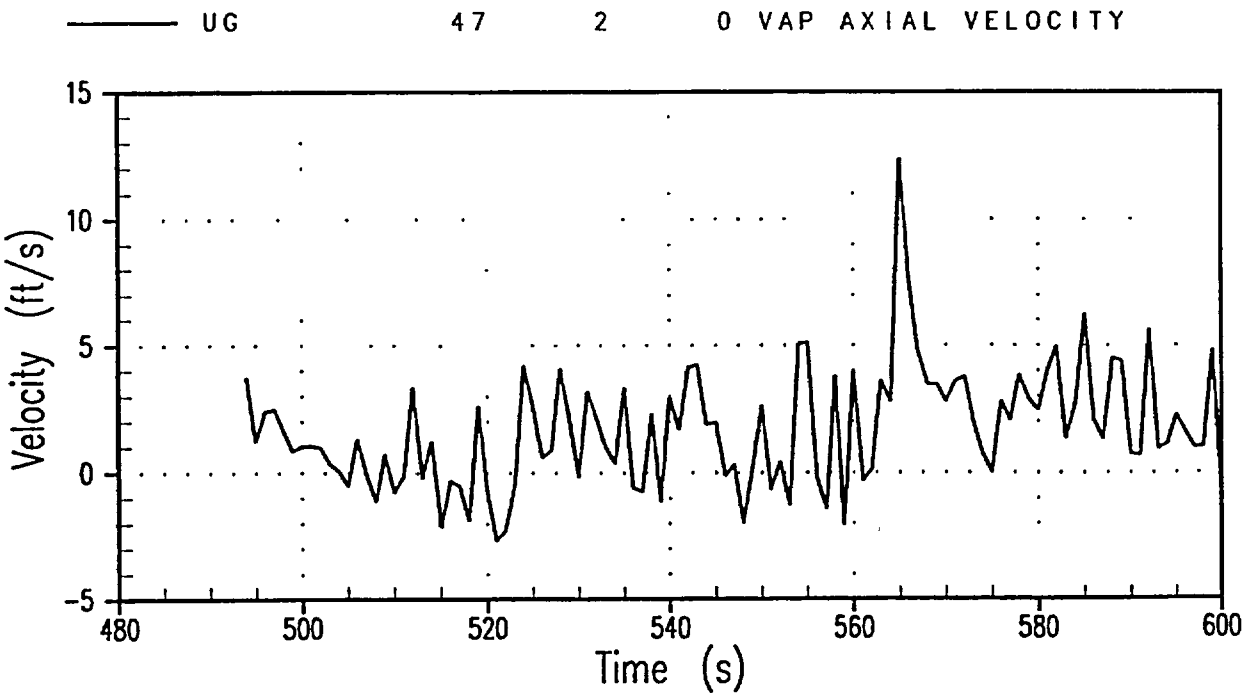


Figure 440.175-10 Double ADS-4 Flow Path Liquid Velocity, DEDVI Break Case

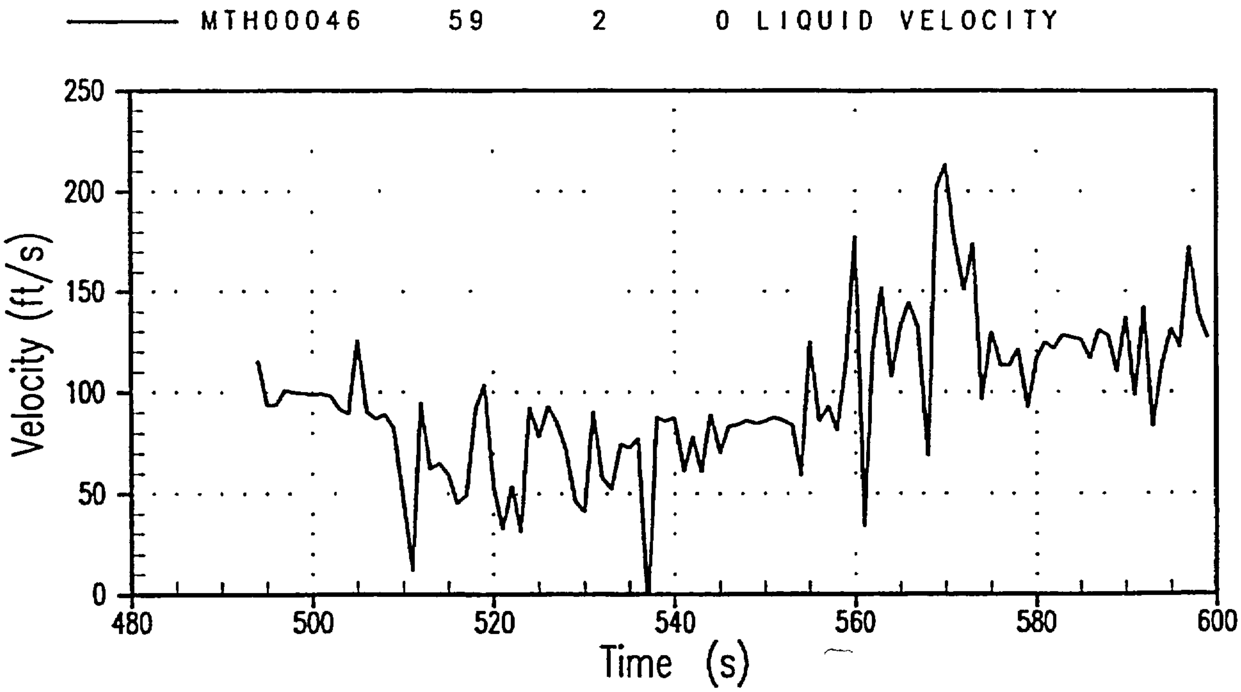


Figure 440.175-11 Double ADS-4 Flow Path Vapor Velocity, DEDVI Break Case

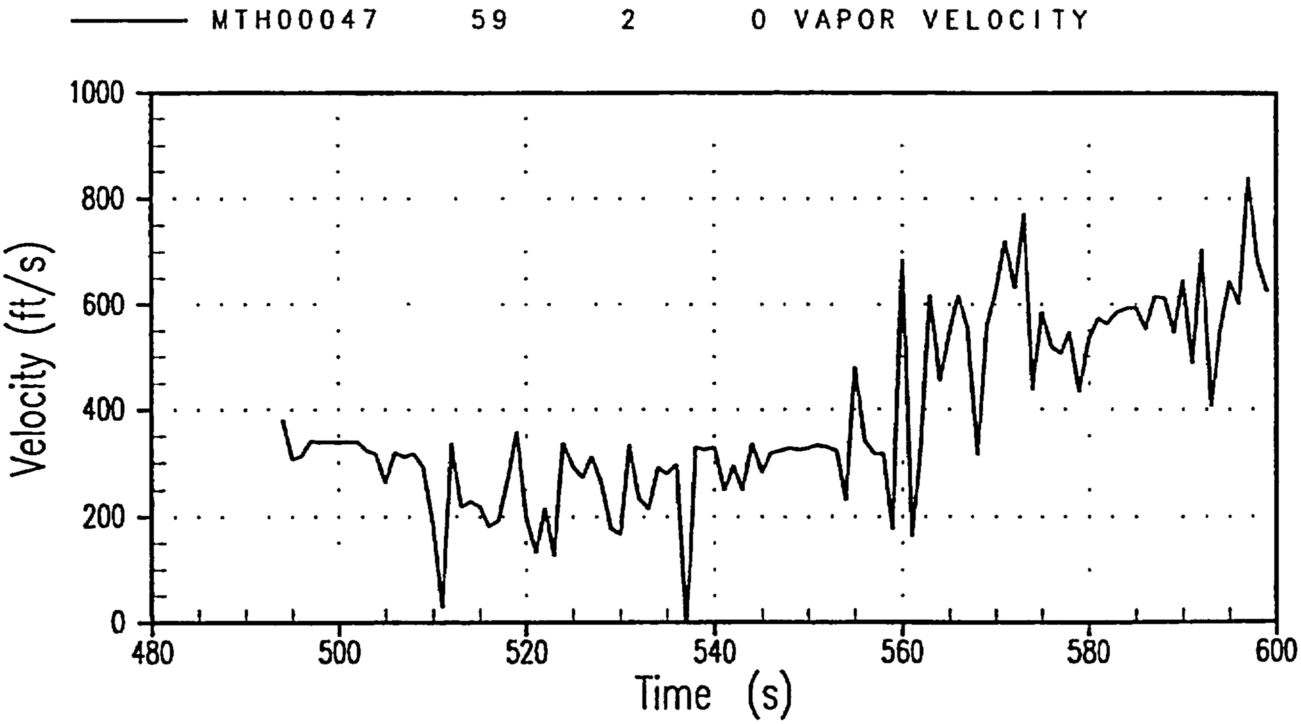


Figure 440.175-12 Single ADS-4 Flow Path Liquid Velocity, DEDVI Break Case

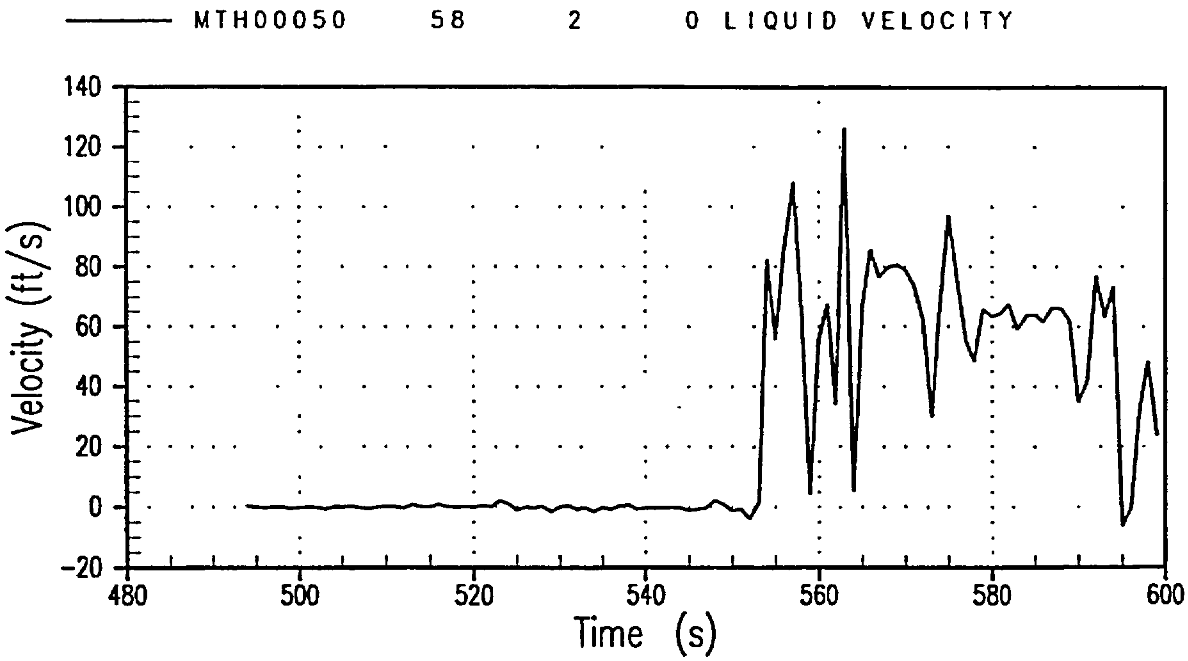
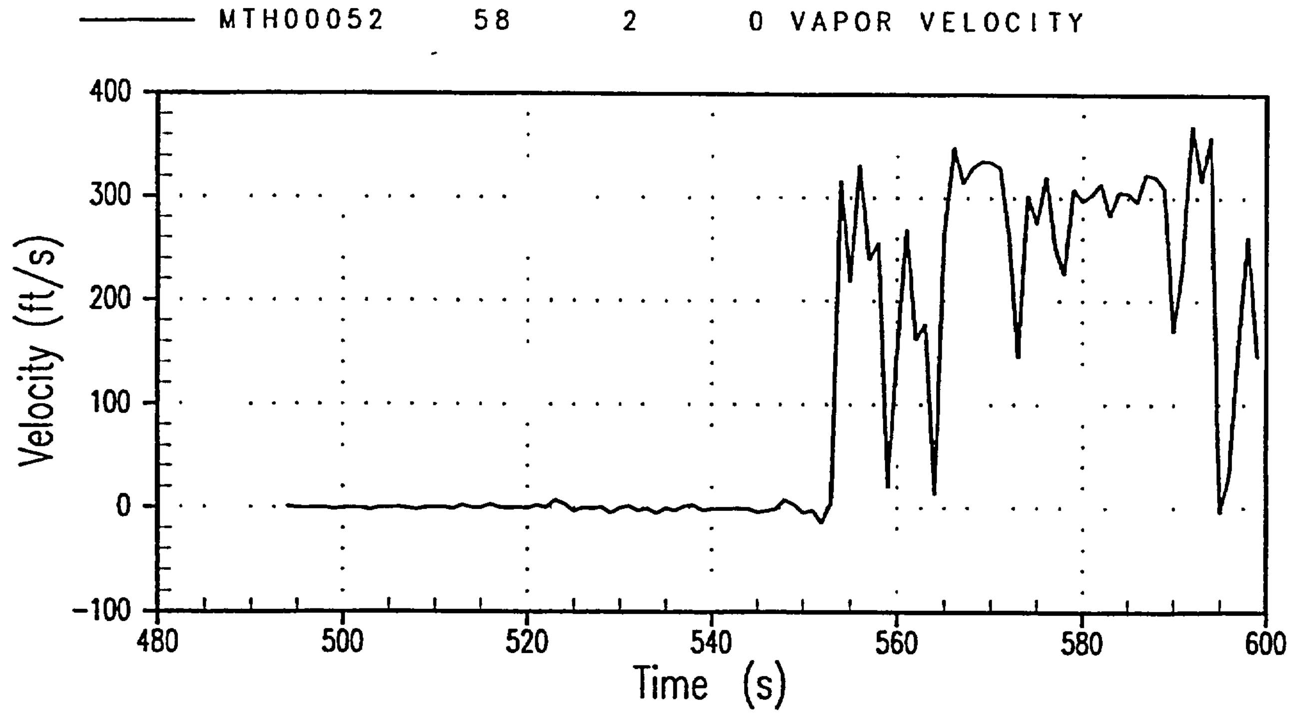


Figure 440.175-13 Single ADS-4 Flow Path Vapor Velocity, DEDVI Break Case



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.176

Question:

Figures 2-14 and 2-15 of WCAP-15833, Revision 1 indicate that condensation heat transfer was initially under predicted by WCOBRA/TRAC for the Lim test section and then over predicted. For the lengths and geometries of the AP1000 hot legs, provide a discussion of the conservatism of the WCOBRA/TRAC model for hot leg condensation for AP1000 based on the correlation of the Lim data.

Westinghouse Response:

The Lim tests provide data for condensation of steam in cocurrent, horizontal flow as described in Section 2.2.2 of WCAP-15833. Condensation heat transfer in the AP1000 hot leg is a minor effect during the ADS-4 to IRWST transition phase as saturated or near-saturated conditions exist in the hot leg during this period. With respect to Figure 2-14 of WCAP-15833, the condensation model used in WCOBRA/TRAC produces conservative results (i.e. the WCOBRA/TRAC calculated steam flow values are larger relative to the test data from Run 275 indicating less condensation heat transfer) for entire length of the test channel. This is further supported by Figure 2-17 that shows that the WCOBRA/TRAC calculated steam flow values are conservative relative to nearly all of the Lim test data. Figure 2-15 provides a comparison of the WCOBRA/TRAC calculation relative to condensation correlations suggested by Lim et al. in NUREG/CR-2289 for horizontal stratified flow with smooth gas-liquid interface and rough gas-liquid interface. However, the correlations suggested by Lim et al. were misapplied in Figure 2-15 and therefore it will be deleted from WCAP-15833.

As condensation heat transfer in the hot legs is a minor effect and the WCOBRA/TRAC model provides reasonable results with respect to the test data, the WCOBRA/TRAC condensation model for horizontal stratified flow is acceptable for application to the hot legs in AP1000.

Design Control Document (DCD) Revision: None

PRA Revision: None

WCAP Revision:

Figure 2-15, "Comparison of Condensation Heat Transfer Correlations" will be deleted along with text that relates to Figure 2-15 on pages 2-20 and 2-21 of Section 2.2.2 as follows:

DELETE

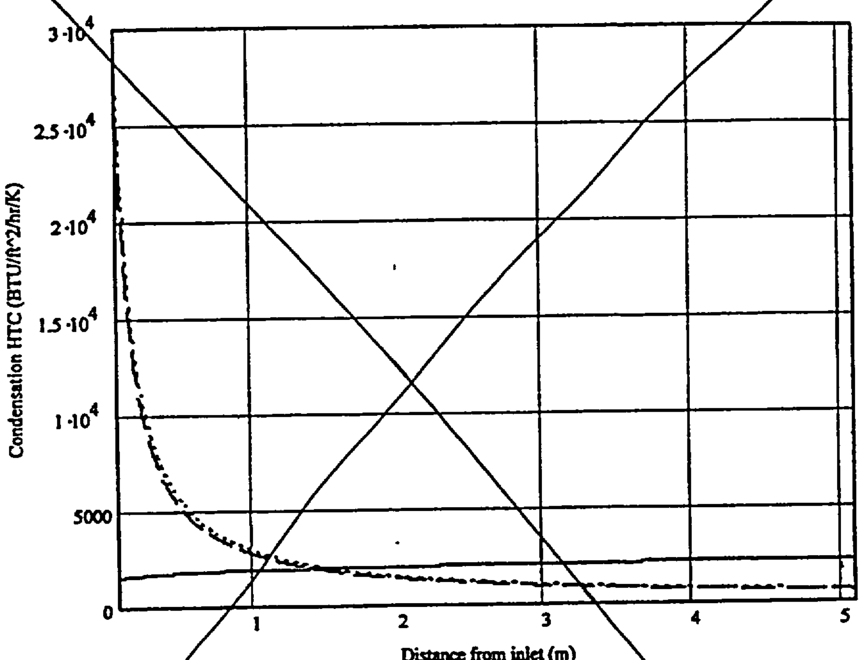


Figure 2-15 Comparison of Condensation Heat Transfer Correlations

The principal difference between the correlations is that the Nu value in WCOBRA/TRAC-AP is [proportional to $x^{1.1}$ (where x is the distance from inlet), while Nu is independent of x in the alternative correlation. As a result, for the alternative correlation, the condensation heat transfer coefficient is undefined (approaches infinity) at a channel inlet and then sharply falls. For the correlation currently used in WCOBRA/TRAC-AP, the condensation heat transfer coefficient rises with distance proportional to $x^{0.1}$ (Figure 2-15). Thus, the difference in the condensation heat transfer coefficient correlations explains the observed discrepancy between measured and calculated steam flowrates.]a,c

Delete

The cumulative results of all tests simulated are shown in Figures 2-16 through 2-19, which show scatter plots of predicted versus measured quantities of the liquid level, steam mass flowrate, liquid temperature at the channel exit, and the pressure drop in the channel, respectively. For most of the cases, liquid level predictions are within ±0.2 inches of the measurements. The steam flowrate is overestimated almost everywhere in the test section, particularly near the channel exit. As a result, the liquid temperature at the channel exit is underpredicted by 20° to 40°F. The large majority (approximately 80 percent) of the pressure drop predictions is within ±33 percent of the experimental data, as shown in Figure 2-19.

Conclusions

WCOBRA/TRAC-AP predictions of two-phase flow in a horizontal channel were verified against data for a rectangular channel with cocurrent water flow at atmospheric pressure. A model of the experimental channel, consisting of [22 vertical computational channels and 3 horizontal layers, was developed.]a,c The pertinent cases among the 35 test cases reported in Lim (Reference 4) were simulated. For most of the cases, liquid level predictions are within ±0.2 inches of the measurements. Depending on the axial position, steam flowrate can be overestimated by a factor of 2 or more (near the channel exit). As a result, the liquid temperature at the channel exit is underpredicted by 20° to 40°F. To address this, values of the condensation heat transfer coefficient calculated by the code were compared with those given by the correlation used in WCOBRA/TRAC-AP and one derived from the experimental data. The difference in the condensation heat transfer coefficient is determined to be due to the correlation used in the code.

Delete

As Condensation heat transfer in AP1000 hot leg horizontal stratified flow is a minor effect during the ADS-4 IRWST initiation phase as saturated or near-saturated conditions exist during this phase of the transient, the WCOBRA/TRAC condensation model for horizontal stratified flow is acceptable for AP1000.

Most of the pressure drop predictions are within ±33 percent of the experimental data, and the number of points for which the pressure drop is underpredicted is approximately the same as the number for which it is overpredicted. Inasmuch as hot leg steam velocities are low when horizontal stratified flow conditions exist in the AP1000 hot legs during the ADS-4 IRWST initiation phase of a small break LOCA event, the hot leg pressure drop prediction is not of major importance in predicting ADS-4 performance.

2.2.3 References

1. Taitel, Y., and Dukler, A. E., 1976, "A Model for Predicting Flow Regime Transitions in Horizontal and Near Horizontal Gas-Flow," AIChE Journal, Vol. 22, No. 1, pp. 47-55.

2. Ishii, M. and Grolmes, M., 1975, "Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow," AIChE Journal, Vol. 21, No. 2, pp. 308-318.

Calculational Results

A total of 35 tests are reported in Lim (Reference 4) as shown in Table 2-1. Those tests in which the horizontal two-phase flow is fully within the wavy or stratified flow regimes (32 in number) were simulated. The experimental results and test conditions for the tests simulated with WCOBRA/TRAC-AP are shown in Table 2-1. Steam density and steam and water velocities were input as boundary conditions in the model's steam and liquid fill components, respectively.

In Table 2-1, steam flowrate and water layer thickness data at locations 1, 2, 3, 4, and 5 correspond to 6.18, 12.05, 23.08, 34.18, and 48.14 inches from the experimental channel inlet. Static pressure difference measurements at 4.88, 10.75, 21.77, 32.87, and 47.24 inches are listed as being at locations 1 through 5. Nomenclature is provided on the table.

Steam density input is calculated using NIST/ASME steam properties for given values of the steam inlet temperature and constant pressure of 16 psi. Due to small variations in the liquid temperature and density among the tests and along the experimental channel, a constant liquid density corresponding to the average liquid temperature of 148.6°F is assumed. Steam and water inlet velocities in the model fill components (Figure 2-6) are calculated using a constant flow area of 0.2083 ft^2.

The WCOBRA/TRAC-AP predictions for a typical case (Run 275) are presented in Figures 2-9, 2-11, and 2-13. Predicted values of liquid level, steam pressure, and steam flowrate are shown for the duration of the test at a number of axial locations. In Figures 2-10, 2-12, and 2-14, the average calculated values of these parameters are compared with the experimental data. There is a reasonably good agreement between the measured and predicted average values of liquid level and pressure drop[1] in the channel as seen in Figures 2-10 and 2-12. While the liquid level at 47.27 inches is significantly underpredicted, the observed trend of the liquid level to recover toward the channel outlet is well reproduced by WCOBRA/TRAC-AP (Figure 2-10). WCOBRA/TRAC-AP overpredicted the steam flowrate axially as seen in Figure 2-14; underpredicting the steam condensation rate is the cause. ~~This matter was investigated further; the condensation heat transfer correlation used in WCOBRA/TRAC-AP (Reference 11), and one derived from the experimental data, were compared to each other for typical flow conditions in the channel. This comparison is presented in Figure 2-15: the solid line is the WCOBRA/TRAC correlation result, and the dashed line(s) the correlation from the experiment.~~

~~The alternative correlation for a smooth interface based on this test data (Lim, et al., 1981) is given by:~~

$$Nu_{x\_As} := 0.631 \cdot (Re_g)^{0.58} \cdot (Re_l)^{0.09} \cdot (Pr_l)^{0.3} \qquad (2\text{-}33)$$

~~where:~~

Nu_{x_As} = is the Nusselt number (Nu)

DELETE

[1] Note that the pressure actually increases as the steam flow proceeds through the channel

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.177

Question:

Section A.4.2 indicates that WCOBRA/TRAC results are relatively insensitive to assumptions made for entrainment rate in the upper plenum. The sensitivity to entrainment rate in the core is not addressed. Provide analyses similar to those of Section A.4.2 in which the core entrainment rate is varied consistently with that of the upper plenum. What is the uncertainty in the WCOBRA/TRAC predictions for core entrainment rate?

Westinghouse Response:

The WCOBRA/TRAC code version used to analyze the AP1000 ADS-4 IRWST initiation phase performance includes the parameter EMULT to facilitate the ranging of the entrainment rate prediction. EMULT is a multiplier on the entrainment rate, which is computed according to the methods documented in Reference 440.177-1, Section 4-6. Section 25-7 of that reference states that [

]^{a,c}

The code behavior in predicting core entrainment during the AP1000 ADS-4 IRWST initiation phase is estimated from the SCTF Run 619 simulation documented in Reference 440.177-1, Volume 3, Appendix A. The SCTF was an experimental facility designed by JAERI to study the system response to LOCA transients in a typical 4-loop PWR with an area scaling ratio of 1/21. Even though SCTF Run 619 is a large break LOCA forced reflood test, the uncertainty in the core entrainment rate for the ADS-4 IRWST initiation phase prediction can be estimated by considering the time period after the core has quenched. Figures C3 to C15 in Appendix A show all elevations are quenched before 370 seconds. Since the power shutoff occurred at 500 seconds, there is a pertinent time interval of about 130 seconds in the subject test during which the core is quenched with the power remaining on.

Figures C31 and C33 in Appendix A show that the core mass is underpredicted throughout the simulation, and the upper plenum mass is overpredicted by the code for the first 450 seconds or so. These figures indicate that a bias to overpredict core entrainment appears to exist when the core is quenched as well as unquenched. The requested sensitivity study is to investigate a range of core (and upper plenum) entrainment rates. Taking into account the WCOBRA/TRAC core entrainment overprediction bias, the upper range value of EMULT is selected to be 1.30, and the lower range value is selected to be 0.5 for this study.

The attached figures provide the results of sensitivity cases in which the core and upper plenum entrainment rates are multiplied by 1.3 and 0.5. The following figures may be compared with the sensitivity case(s) presented in Section A4.2:

Figure 440.177-1: Downcomer Pressure
Figure 440.177-2: Vessel Mass Inventory
Figure 440.177-3: Upper Plenum Collapsed Liquid Level
Figure 440.177-4: Core Collapsed Liquid Level
Figure 440.177-5: Entrained Flow into the Pressurizer Loop Hot Leg
Figure 440.177-6: Entrained Flow into the 2*ADS-4 Loop Hot Leg
Figure 440.177-7: IRWST Injection Flow Rate

The WCOBRA/TRAC prediction exhibits a modest sensitivity to these variations in entrainment consistent with the identified range. Reducing the entrainment prediction by a factor of 2.0 results in an increased minimum vessel mass inventory, as more mass remains in the reactor vessel than in the base case while the steam flows to the ADS-4 flow paths. The increased entrainment case minimum mass inventory exhibits little change from the base case value.

Reference:

1) WCAP-12945-P-A, Revision 1, *Code Qualification Document for Best Estimate LOCA Analysis,* Westinghouse Electric Company, March 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 440.177-1: Downcomer Pressure

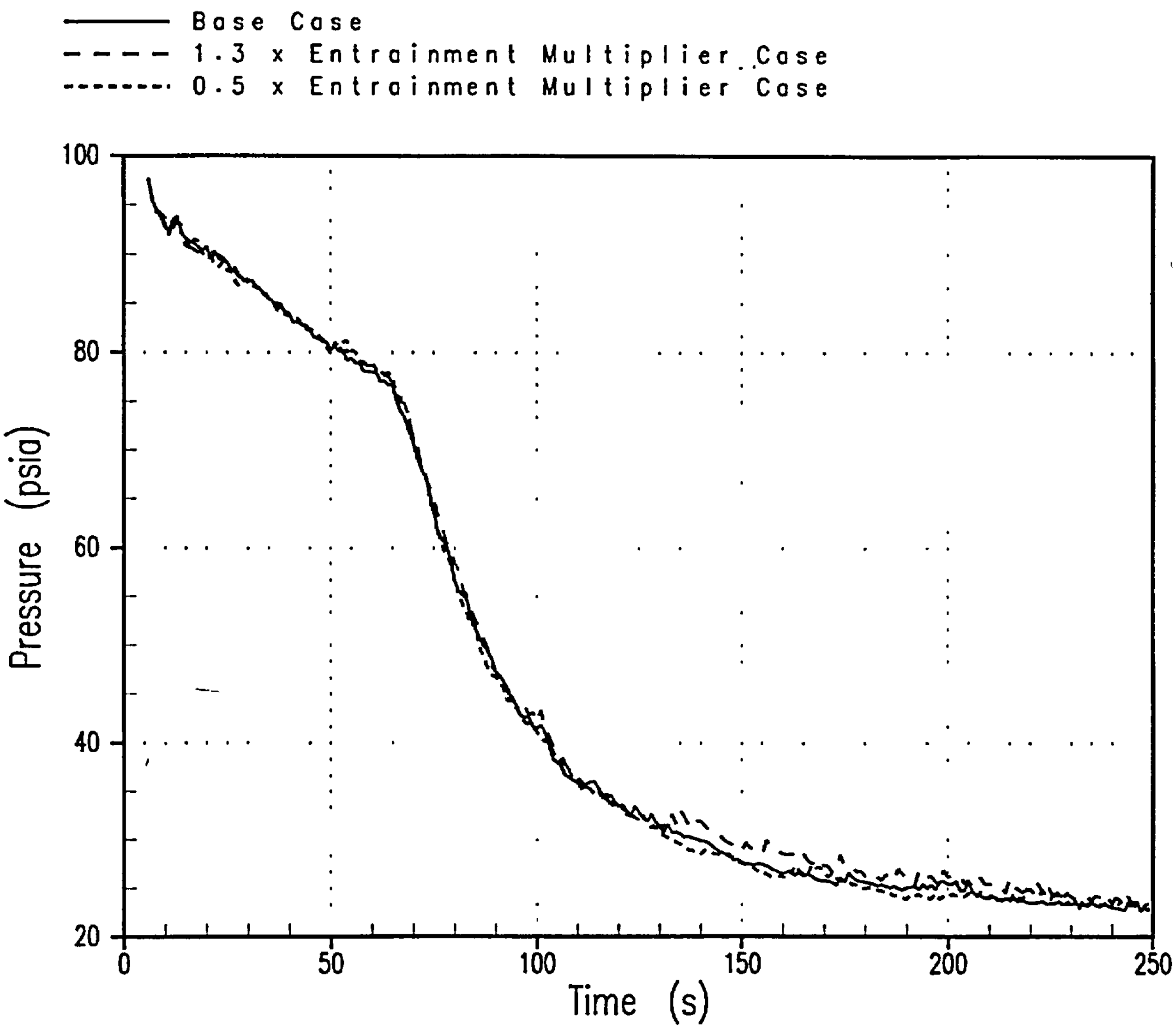


Figure 440.177-2: Vessel Mass Inventory

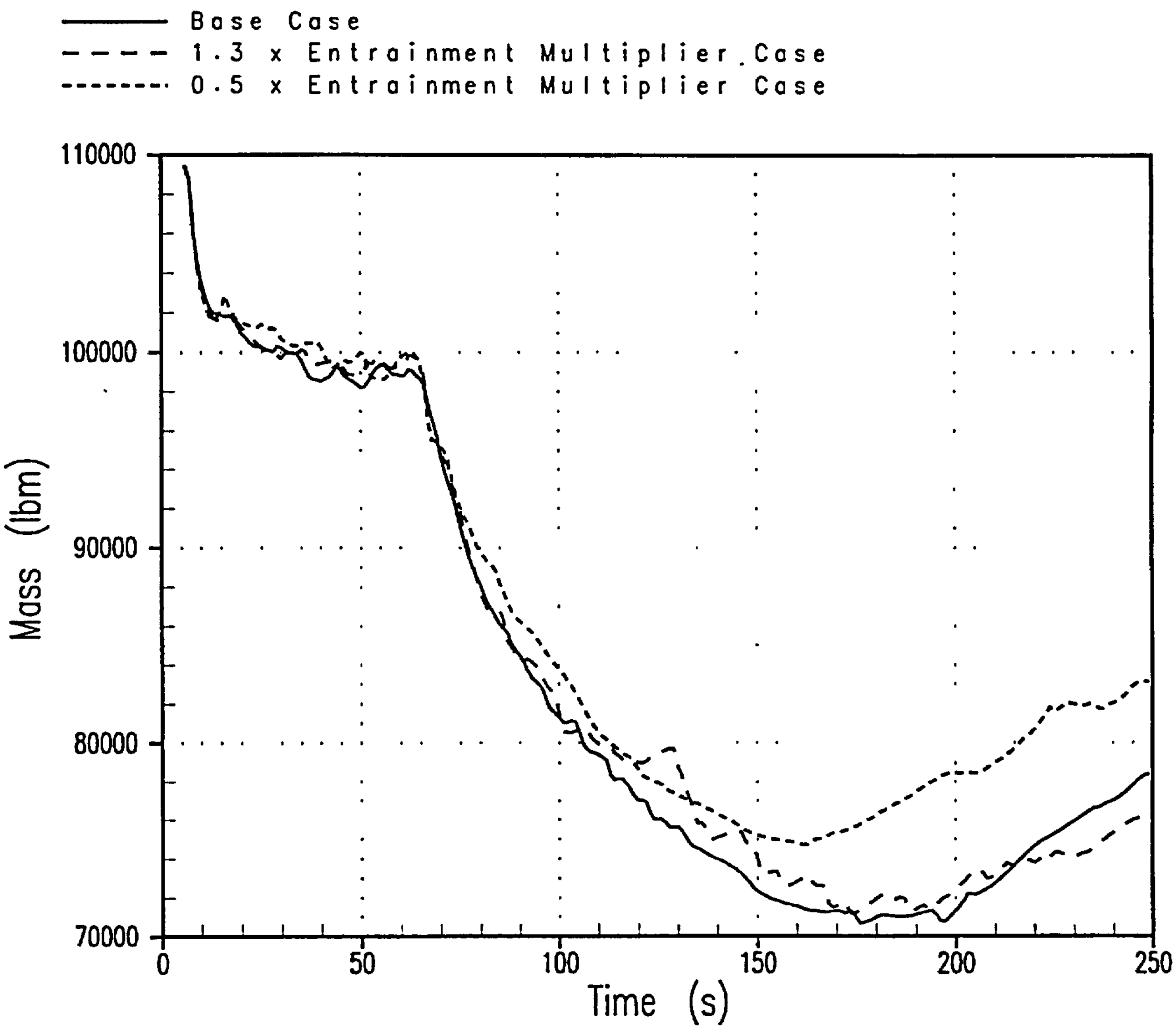


Figure 440.177-3: Upper Plenum Collapsed Liquid Level

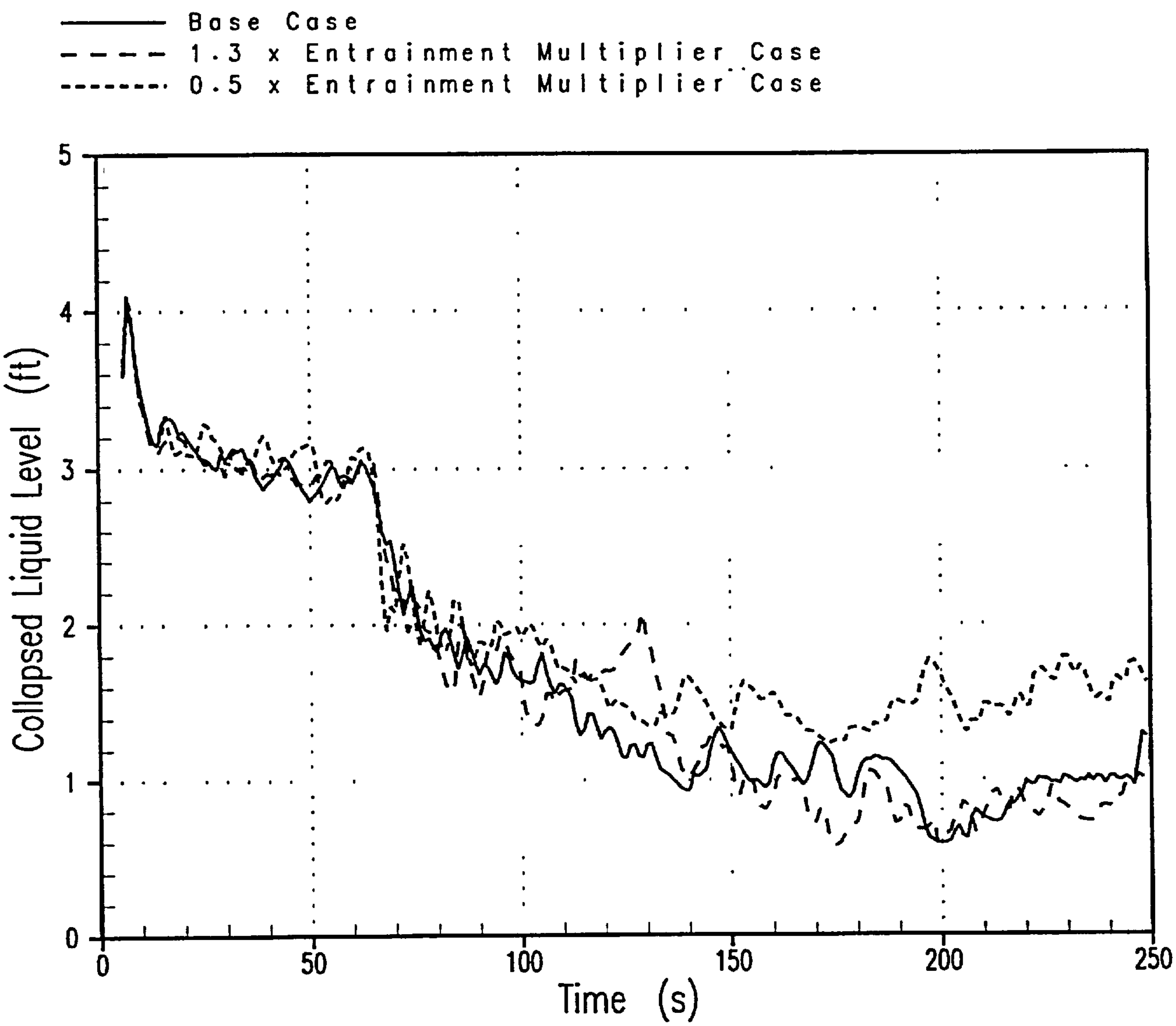


Figure 440.177-4: Core Collapsed Liquid Level

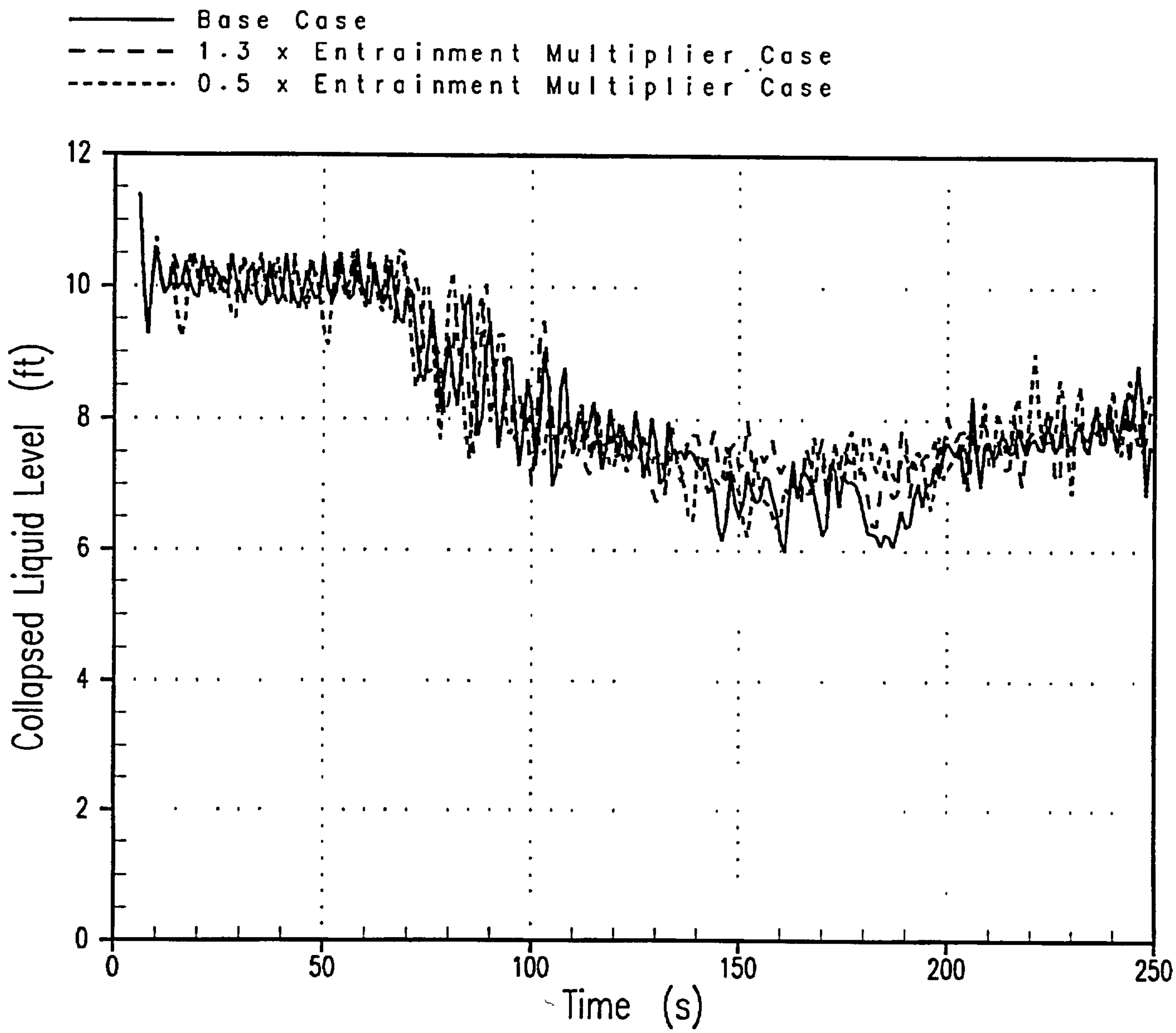
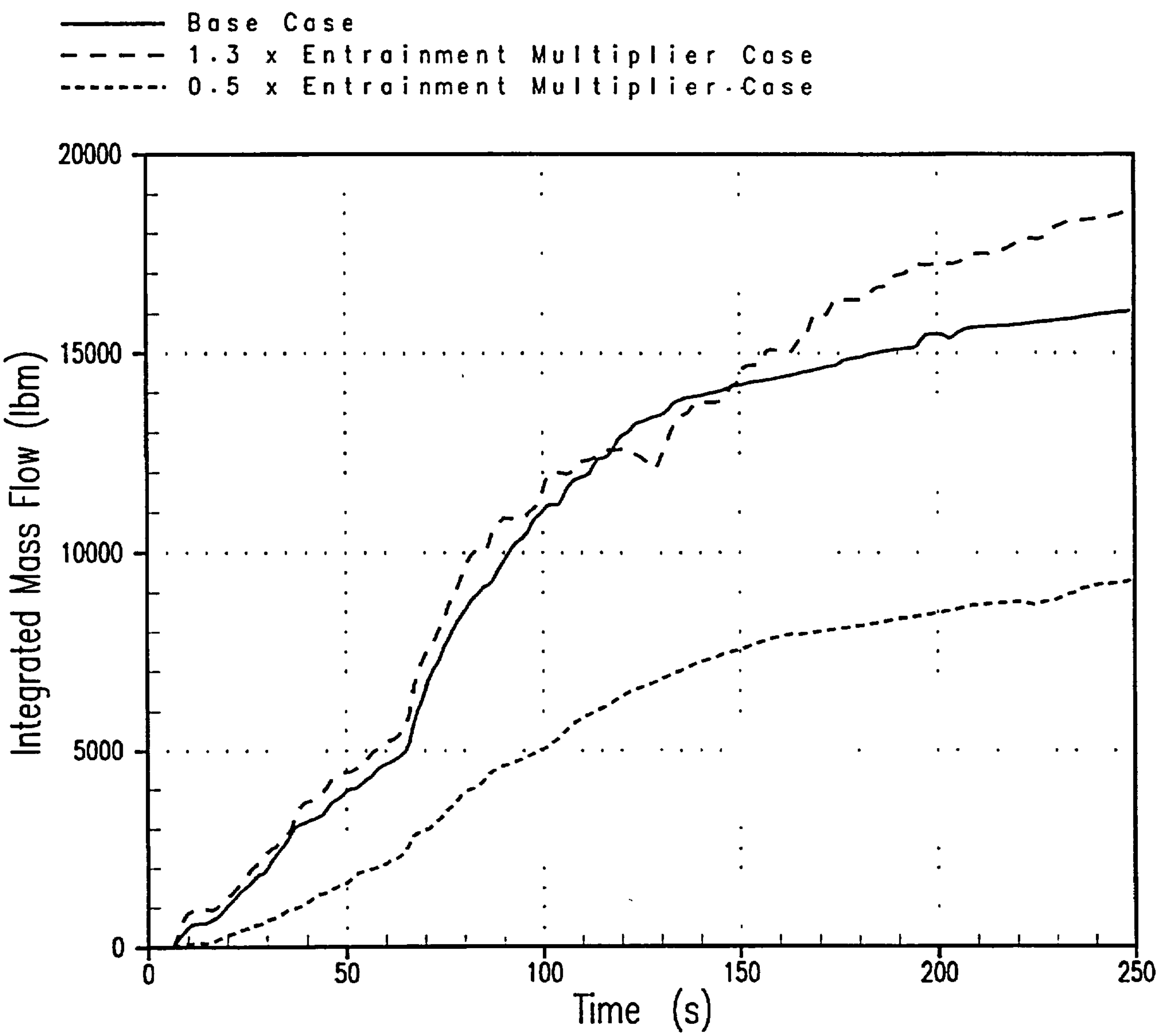


Figure 440.177-5: Entrained Liquid Flow Entering the Pressurizer Loop Hot Leg



'igure 440.177-6: Entrained Liquid Flow Entering the 2*ADS-4 Loop Hot Leg

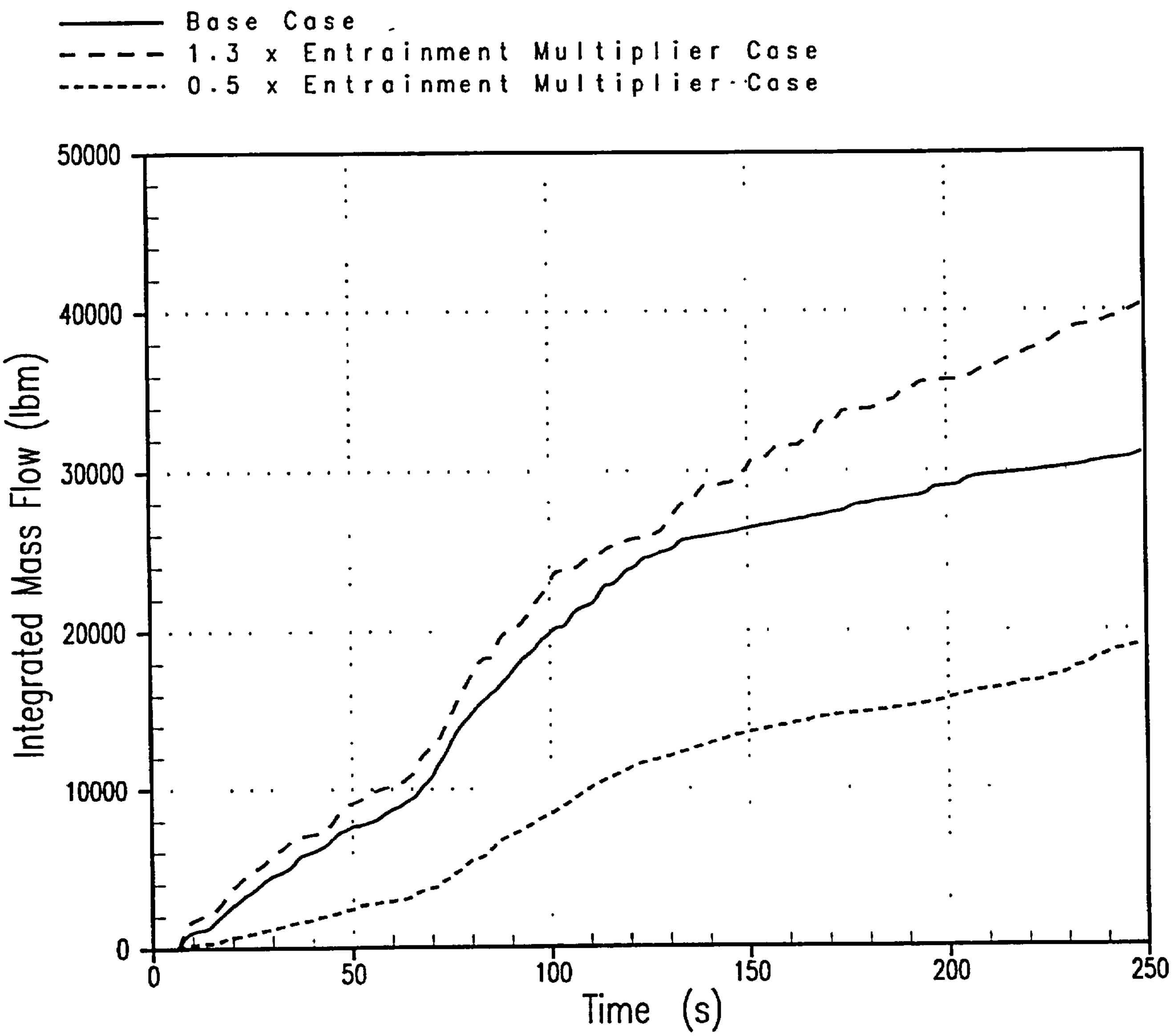
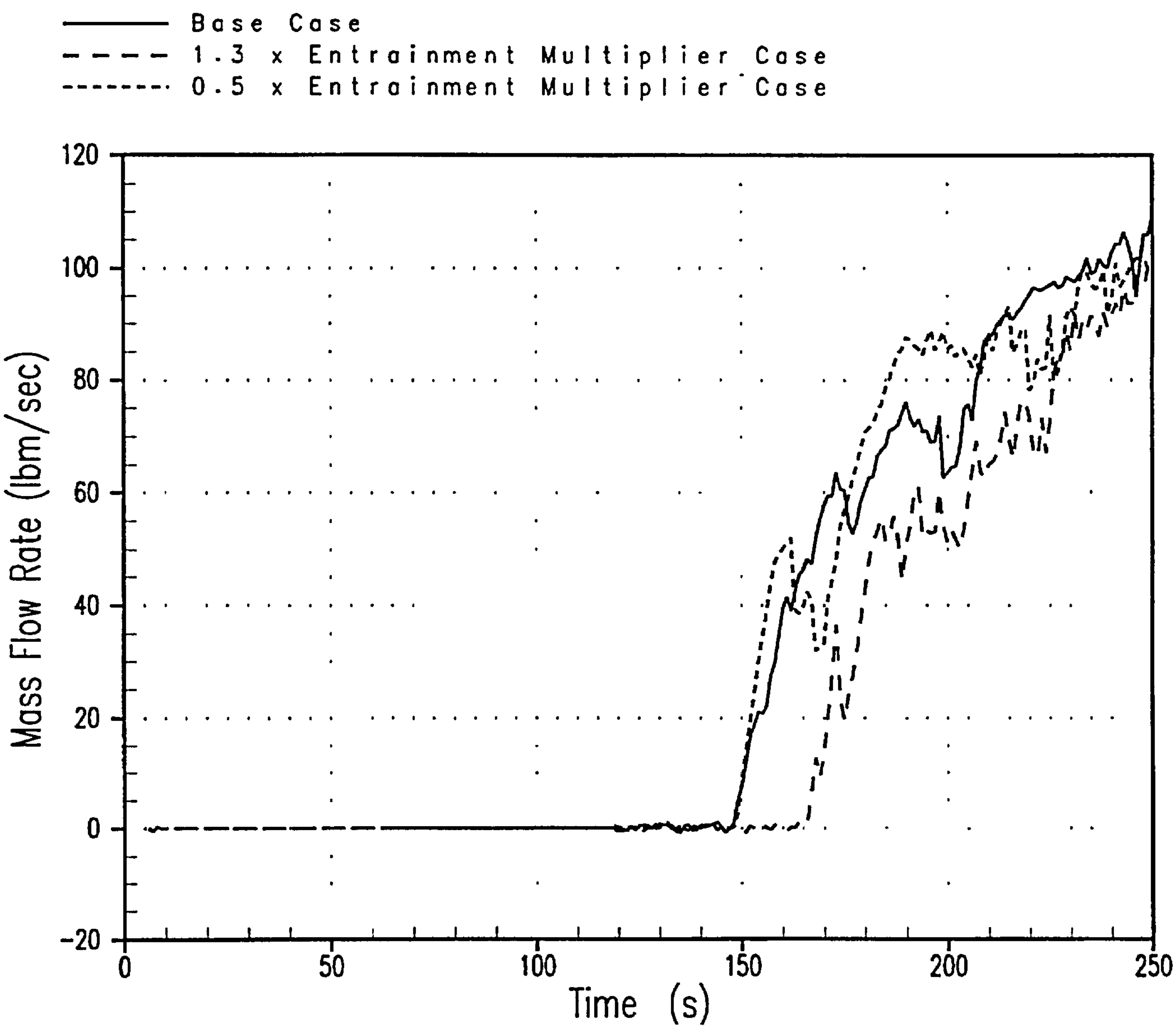


Figure 440.177-7: IRWST Injection Flow Rate



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.178

Question:

Section A.4.3 indicates that WCOBRA/TRAC results are relatively insensitive to assumptions made for interfacial drag in the upper plenum. The sensitivity to interfacial drag in the core is not addressed. Provide analyses similar to those of Section A.4.3 in which the core interfacial drag is varied consistently with that of the upper plenum. What is the uncertainty in the WCOBRA/TRAC predictions for core interfacial drag?

Westinghouse Response:

The WCOBRA/TRAC code version used to analyze the AP1000 ADS-4 IRWST initiation phase performance includes the parameter YDRAG to facilitate the ranging of interfacial drag. YDRAG is a multiplier on the interfacial drag value that is computed according to the vertical flow regime map. The code uncertainty in predicting core interfacial drag for small break LOCA events has been established by simulating rod bundle thermal-hydraulics tests. The ORNL-THTF test bundle was full height and contained 64 electrically heated rods with internal dimensions typical of a 17X17 PWR fuel bundle (Reference 440.178-1). The G-1 series of core uncovery tests was conducted in the Westinghouse ECCS High Pressure Test Facility in a test bundle of [ ]^{a,c}. Additional information on the test facility and the data for the G-1 core uncovery tests are in Reference 440.178-2.

Both the ORNL-THTF and G-1 test facilities were analyzed using a variety of YDRAG values in the simulated core region. The value of YDRAG that enables the WCOBRA/TRAC prediction to match the measured mixture level swell was identified for each test, where mixture level swell is defined as:

{[2-phase mixture level – collapsed liquid level] / collapsed liquid level}.

Table 440.178-1 identifies the YDRAG values at which WCOBRA/TRAC predicts the ORNL-THTF data. Table 440.178-2 identifies the YDRAG values at which WCOBRA/TRAC predicts the G-1 level swell data. The YDRAG values identify the uncertainty range of the WCOBRA/TRAC core interfacial drag prediction to be 0.353 to 1.17.

The attached figures provide the results of runs in which the core and upper plenum interfacial drag calculated rates are multiplied by YDRAG values of 1.3 and 0.4, per the range of values needed to match the test mixture level swell data in the WCOBRA/TRAC test simulations. The following figures provide comparisons of the sensitivity case(s) with the base interfacial drag case:

Figure 440.178-1: Downcomer Pressure
Figure 440.178-2: Vessel Mass Inventory
Figure 440.178-3: Upper Plenum Collapsed Liquid Level
Figure 440.178-4: Core Collapsed Liquid Level
Figure 440.178-5: Entrained Flow into the Pressurizer Loop Hot Leg
Figure 440.178-6: Entrained Flow into the 2*ADS-4 Loop Hot Leg
Figure 440.178-7: IRWST Injection Flow Rate

The WCOBRA/TRAC prediction exhibits a modest sensitivity to variations in interfacial drag. Reducing the interfacial drag prediction by a factor of 2.5 results in a somewhat increased minimum vessel mass inventory relative to the base case, as more mass remains in the reactor vessel while steam generated in the core flows to the ADS-4 flow paths. The increased interfacial drag case minimum vessel mass inventory exhibits little change from the base case value.

References:

1. Anklam, T. M. et al., 1982, *Experimental Investigations of Uncovered Bundle Heat Transfer and Two-Phase Mixture Level Swell Under High Pressure Low Heat Flux Conditions*, NUREG/CR-2456.
2. WCAP-9764, 1980, *Documentation of the Westinghouse Core Uncovery Tests and the Small Break Evaluation Model Core Mixture Level Model*, Proprietary.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Table 440.178-1: YDRAG Values to Match ORNL-THTF Data

Test Number	YDRAG
3.09.10AA	0.827
3.09.10BB	0.908
3.09.10CC	0.698
3.09.10DD	0.881
3.09.10EE	0.752
3.09.10FF	0.635
3.09.10I	0.779
3.09.10J	0.840
3.09.10K	0.871
3.09.10L	0.503
3.09.10M	1.169
3.09.10N	0.61

Table 440.178-2: YDRAG Values to Match G-1 Level Swell Data

Test Number	YDRAG
54	0.926
54	0.414
55	0.686
55	0.577
56	0.353
56	0.354
57	0.644
57	0.775
62	0.90
62	0.50
63	1.121
63	0.80
64	0.55
65	0.89
65	0.897

Figure 440.178-1: Downcomer Pressure

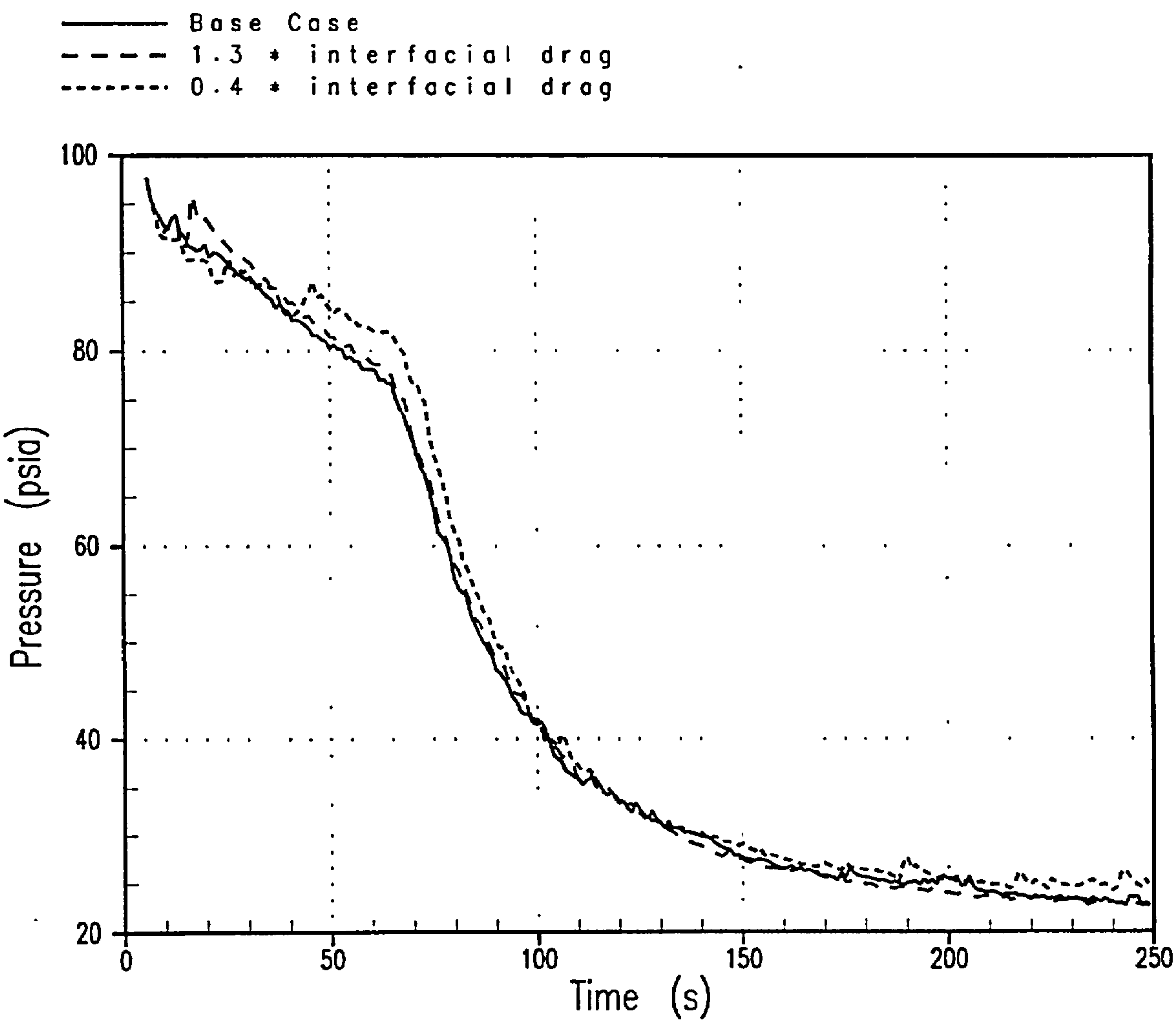


Figure 440.178-2: Vessel Mass Inventory

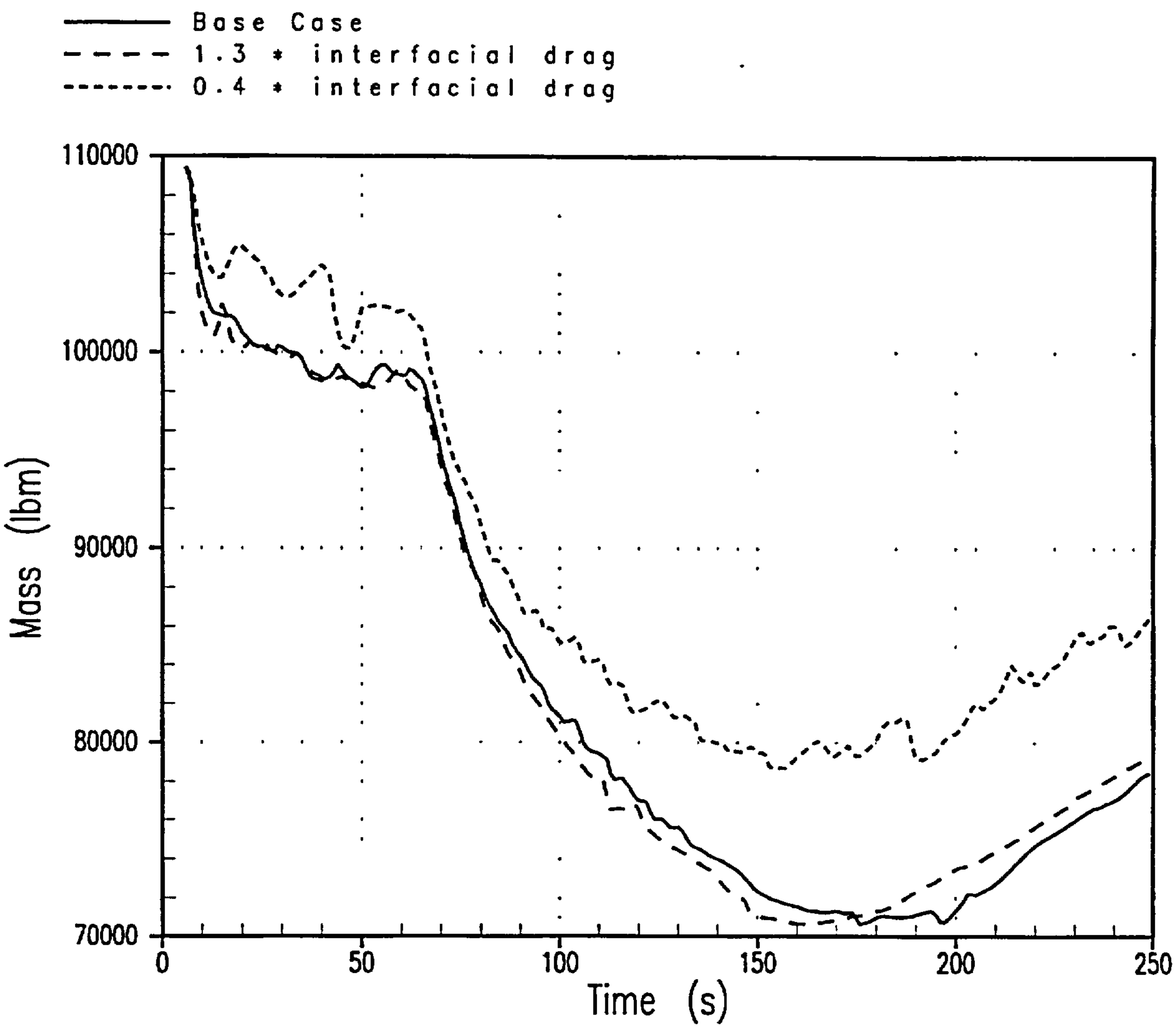


Figure 440.178-3: Upper Plenum Collapsed Liquid Level

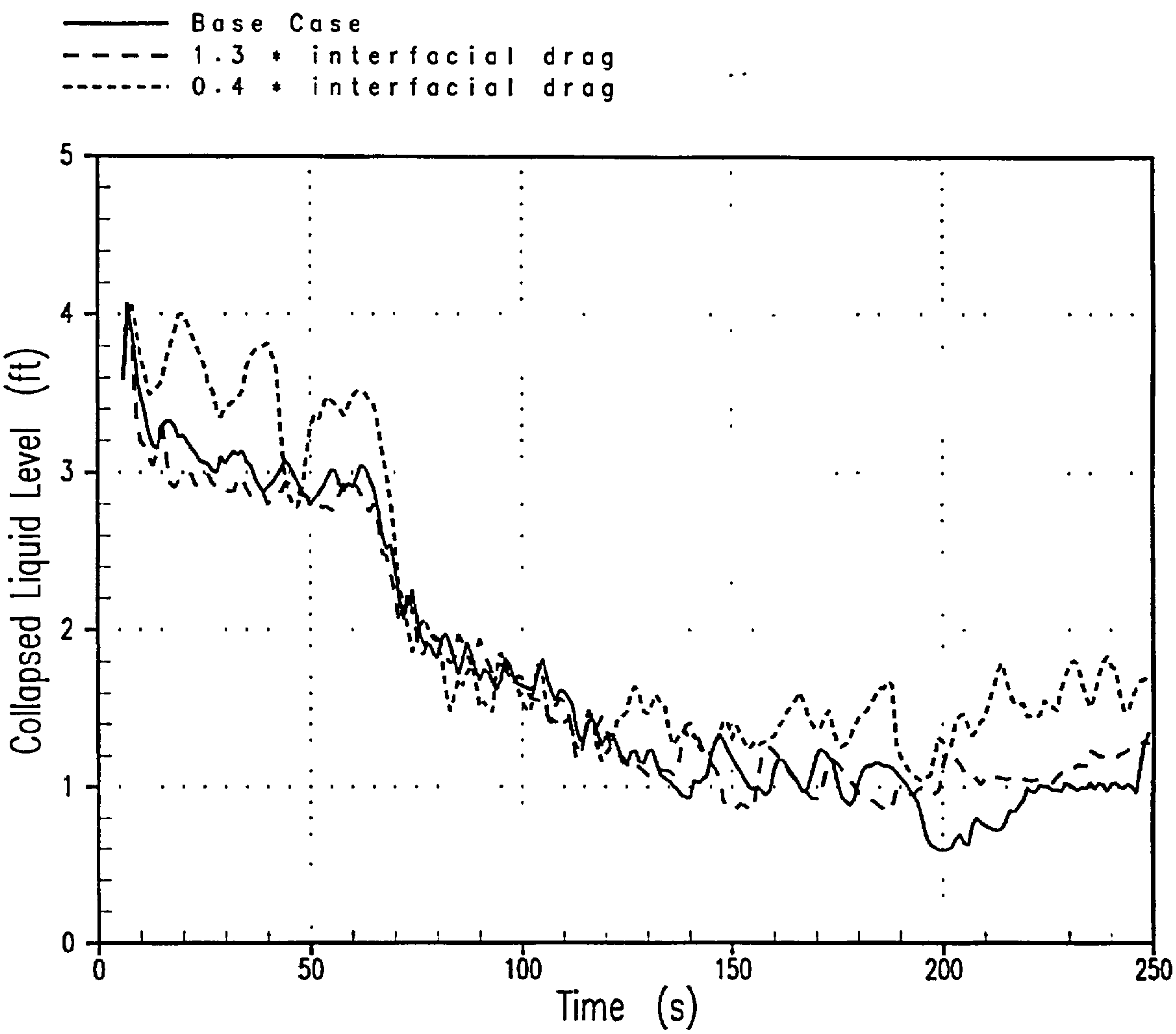


Figure 440.178-4: Core Collapsed Liquid Level

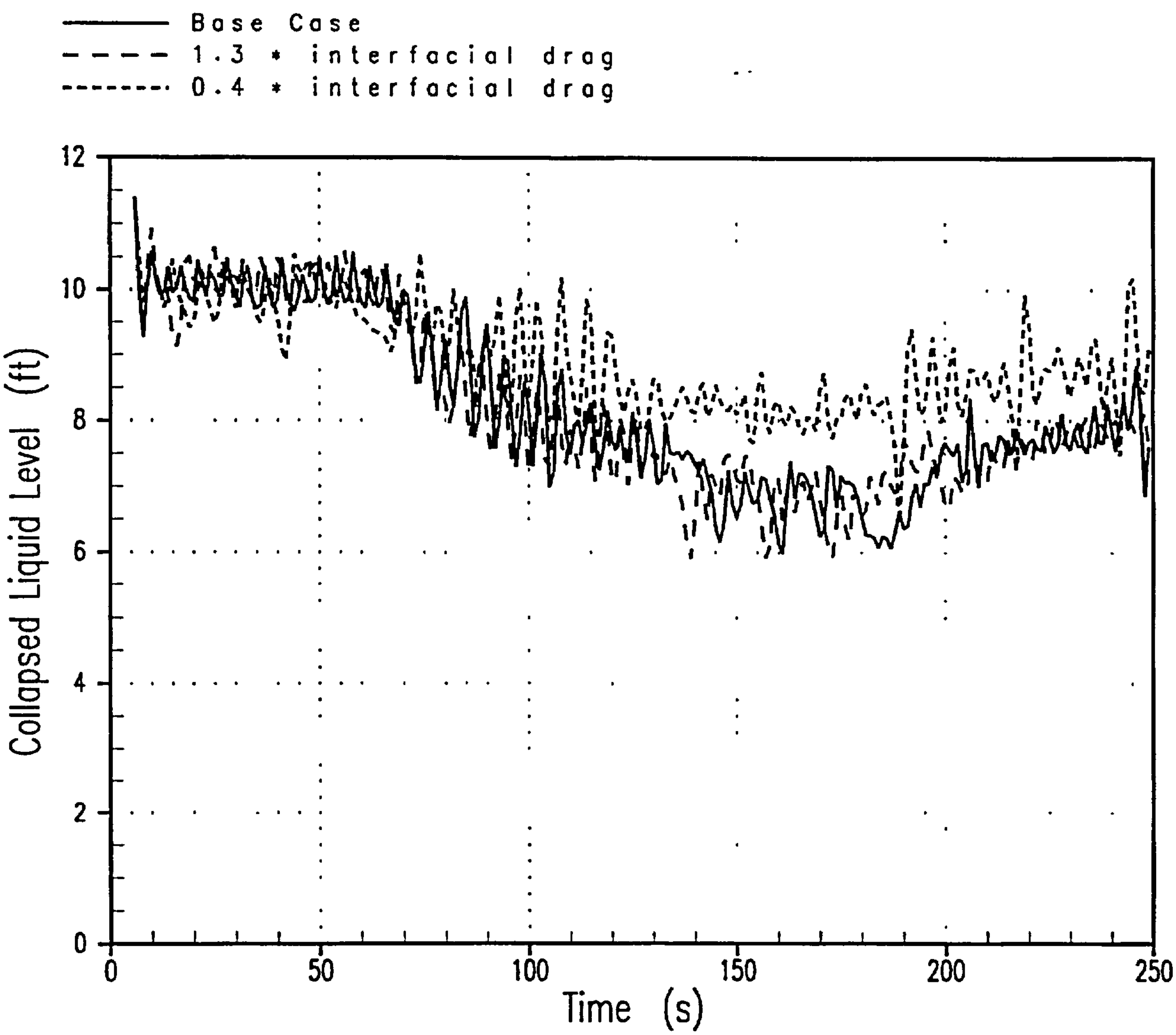


Figure 440.178-5: Entrained Liquid Flow Entering the Pressurizer Loop Hot Leg

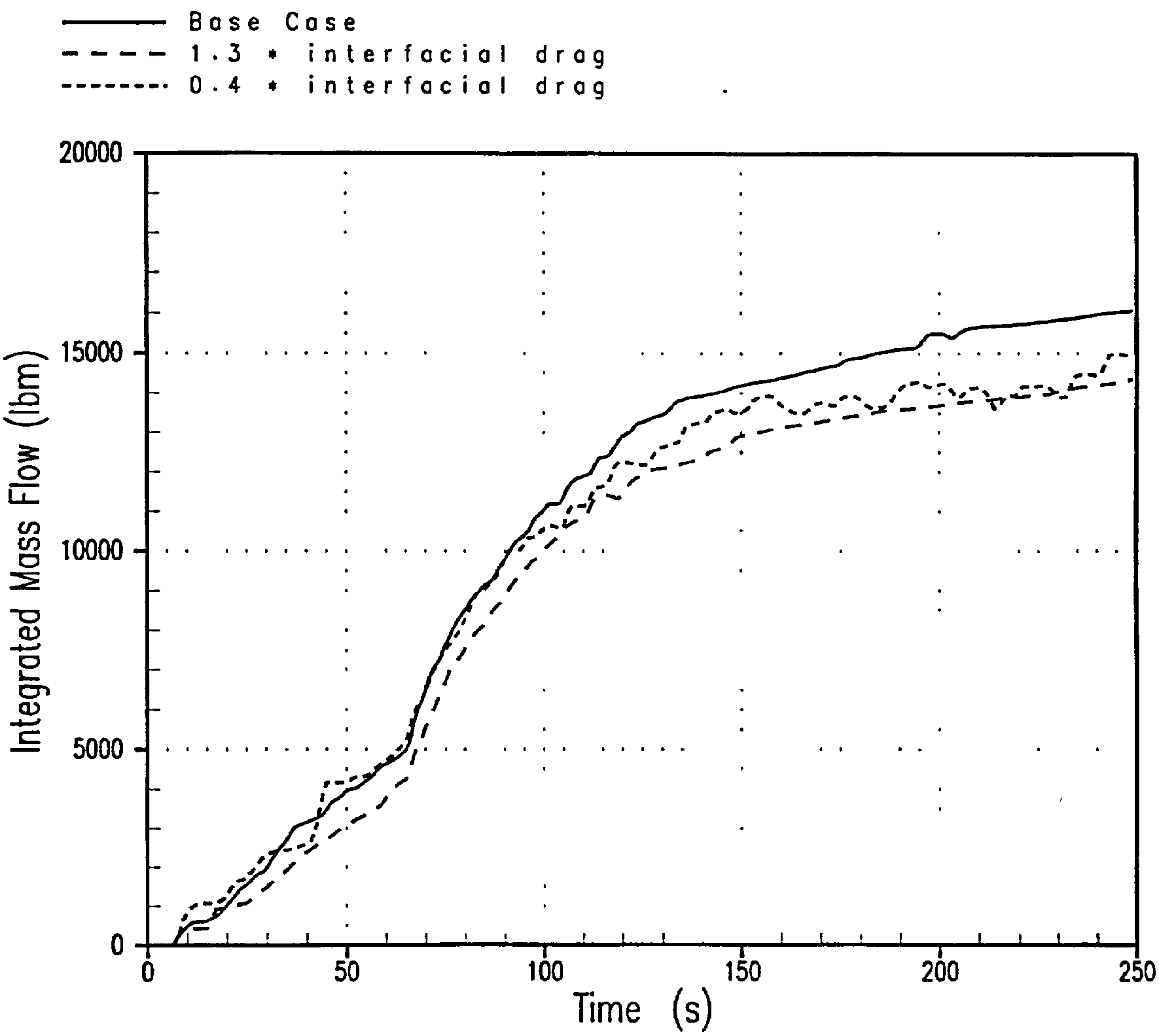


Figure 440.178-6: Entrained Liquid Flow Entering the 2*ADS-4 Loop Hot Leg

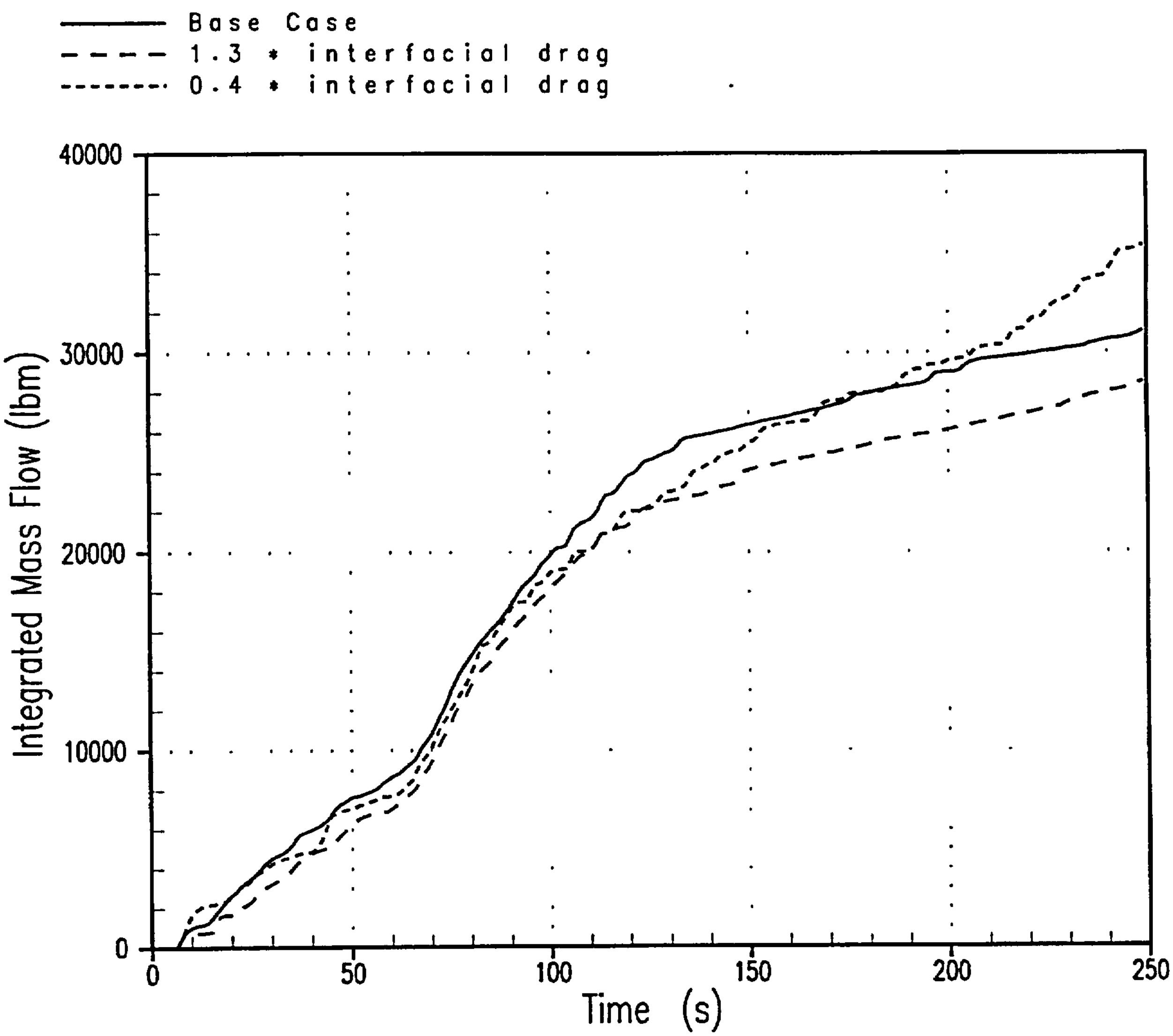
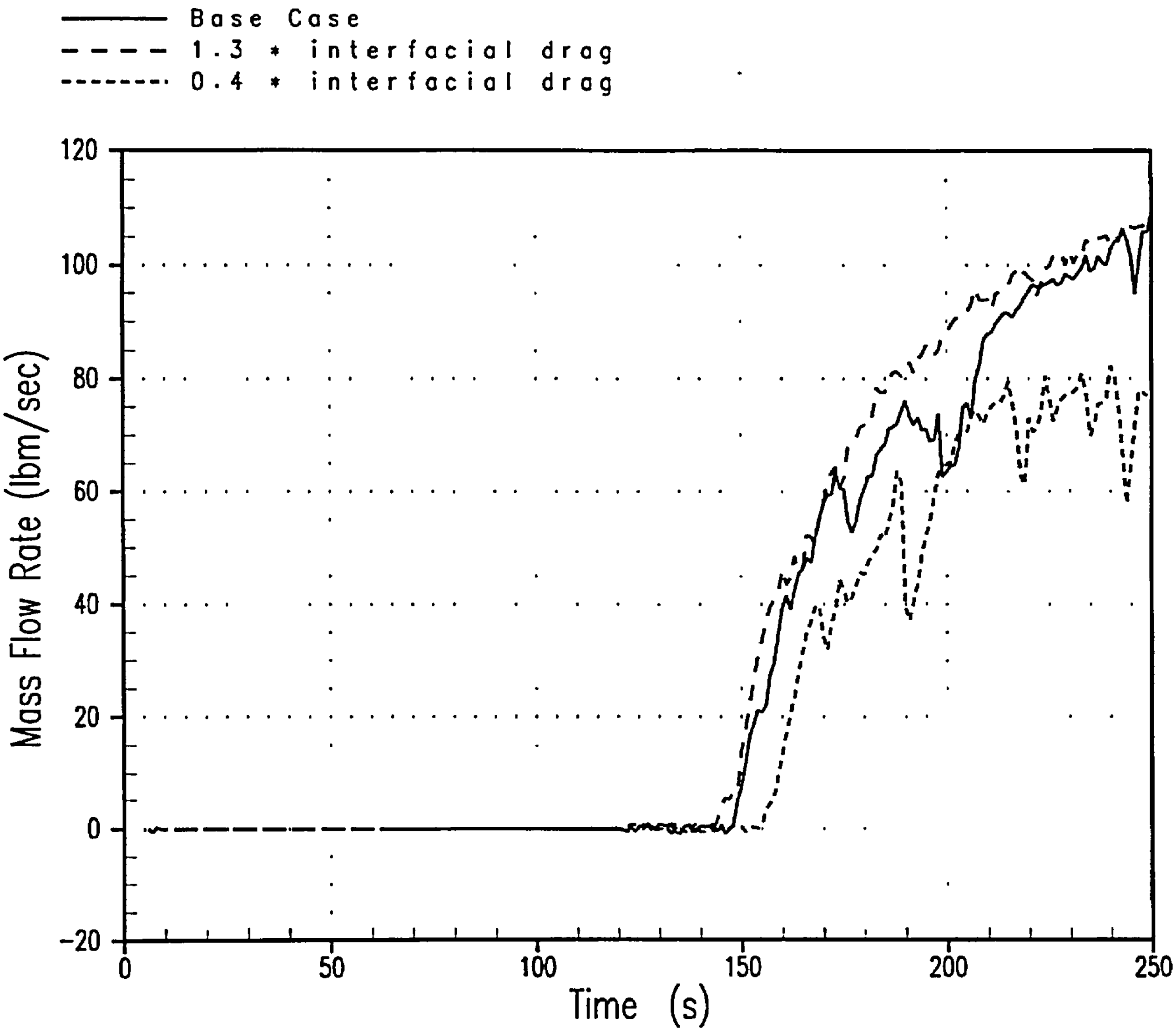


Figure 440.178-7: IRWST Injection Flow Rate



RAI Number: 480.008

Question:

In DCD Section 6.2.1.3.2.1, "Mass and Energy," and in DCD Section 6.2.1.4.1.1, "Plant Power Level," you state that the power level used for the design-basis accident (DBA) analyses is taken to be 101 percent (of full power to account for a 1 percent calorimetric error). This is not consistent with the guidance provided in SRP Sections 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," which indicate that the power level should be taken as 102 percent of full power. This is also inconsistent with the approved methods used by Westinghouse. For example in DCD Reference 3, Section II, it is stated that the core stored energy is based on a conservative value of 102 percent of the engineering safeguards design rated power level. This is also inconsistent with the analyses described in DCD Section 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System," which references DCD Section 15.6.5 wherein the 2 percent calorimetric error is also used (DCD Table 15.6.5-4). The staff is aware that the July 31, 2000, revision to Appendix K to 10 CFR Part 50 allows for the justification of a power level lower than 102 percent for demonstrating compliance with 10 CFR 50.46. The staff is also aware that there is a potential inconsistency with the revised rule and other regulatory guidance. Please provide a new analyses for the limiting pipe breaks to demonstrate that the calculated containment peak pressure and temperature and, if applicable sub-compartment loads, remain acceptable at 102 percent of full power. As an alternative you may provide a justification for review by the staff for the use of the 1 percent calorimetric error for the containment DBA calculations as well as justification for departing from the current SRP guidance and for departing from your approved mass and energy release methodology.

Westinghouse Response:

Section 6.2.1.3 of the SRP says the initial mass of water in the RCS should be based on the reactor coolant system volume calculated for the temperature and pressure conditions existing at 102% of full power. To account for thermal expansion, Westinghouse increases the nominal RCS volume by 1.6%. The 1.016 volume multiplication factor is applied for all LOCA mass and energy release analyses independent of the initial temperature, pressure, or power level.

The Westinghouse mass and energy release methodology documented in WCAP-10325 lists a number of conservatisms that were chosen to maximize the available energy in the system. These conservatisms account for uncertainties in the measured core power level, RCS fluid volume, system temperatures, and SG level/mass.

To account for calorimetric uncertainty, analyses that were submitted for NRC review over the past few years have used 102% of the licensed core power level. Recently, the calorimetric uncertainty has been reduced to less than 0.6% by incorporating an improved flow measurement device. Since the AP1000 will incorporate this flow measurement device, a conservative calorimetric uncertainty of 1% was applied to the NSSS power in both the LOCA and SLB mass and energy release analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.001

Question:

(Sections 11.1, 11.2, and 11.3) The assumption of 0.25 percent fuel defect described in Section 11.1.1.1 for the AP1000 design-basis source term deviates from the fuel defect assumption of 1.0 percent described in SRP Sections 11.2 and 11.3 for the liquid waste and gaseous waste management systems. Please address the following:

A. Justify this deviation, and

B. In the AP1000 DCD, the compliance of liquid and gaseous effluent concentration in unrestricted area with 10 CFR Part 20 limits is not based on a 1 percent fuel failure on the annual average. Table 11.2-9 and 11.3-4 of the DCD should be re-evaluated based on a 1 percent fuel failure and annual average effluent concentrations of radionuclides in unrestricted areas.

Westinghouse Response:

The approach used in AP1000 is identical to that used in AP600.

A. The assumption of 0.25 percent fuel defects for the AP1000 design basis is supported by experience. Westinghouse fuel failure levels have been far below 0.25 percent, with more than 75% of plants operating with zero leakage, and the number of leaking fuel pins less than two per 100,000 pins. Furthermore, AP1000 Technical Specification B 3.4.11 limits for iodine and noble gases are based on 0.25 percent fuel defects, so it is ensured that the plant will not operate with failed fuel exceeding that level.

B. The maximum release concentrations shown in Tables 11.2-9 and 11.3-4 are based on the following:
   - For iodines and noble gases, which are limited by Technical Specification, 0.25% failed fuel.
   - Corrosion product levels are considered to be independent of fuel failures, as discussed in the response to RAI 471.002
   - For all others, 1% failed fuel.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 460.003

Question:

(Sections 11.2, 11.3, and 11.4) Please provide additional information regarding RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Specifically, address the following:

A. According to DCD Appendix 1A relating to RG 1.143, the radwaste building of the AP1000 is designed to Uniform Building Code (UBC)-1991. This is a deviation from Criterion C6.2.1 of RG 1.143, Revision 2, November 2001, which refers to UBC-1997, ASCE (American Society of Civil Engineers) 7-95. However, Section 3.7.2 of the AP1000 DCD refers the building code to UBC-1997. Please clarify the inconsistency and update the DCD accordingly.

B. Section 5 of RG 1.143, Revision 2 defines the classification (RW-IIa, RW II-b, and RW II-c) of radwaste systems for design purposes. In Appendix 1A of AP1000 DCD relating to the conformance with RG 1.143, Revision 2, Westinghouse indicated that systems containing enough activity to possibly be classified as RW-IIa are located in the auxiliary building (which is Seismic Category I). AP1000 conforms with Positions C5.2 and C5.4 with respect to Class RW-IIb. Please identify Class RW-IIb systems, components, and structures in the AP1000 design.

Westinghouse Response:

A. DCD Appendix 1A will be revised as shown on the attached page.

B. All systems and components normally containing quantities of radioactivity which exceed A_1 in 10 CFR Part 71 Appendix A (and therefore would require them to be classified as category RW-IIb) are located in the Auxiliary Building, which is Seismic Category I.

   Solid waste packaging operations for resins and filters will be performed in the Auxiliary Building. Operations in the Radwaste Building are expected to involve minimal radioactivity (garments, respirators, etc). Nevertheless, the Radwaste Building is designed to category RW-IIb.

Design Control Document (DCD) Revision:

Appendix 1A: Criteria Section 6.2.1 of RG 1.143 will be revised to show the AP1000 position as "Conforms." It is noted that AP1000 is designed to ASCE 7-98, while RG 1.143 references ASCE 7-95.

PRA Revision:

None

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
C.4.5		Conforms	In-service testing of the containment penetrations and isolation valves is performed as described in Design Control Document subsection 3.9.6. Other tests, on nonsafety equipment, are performed on an item-by-item basis as judged necessary to confirm proper operation of the systems.
C.5	10 CFR Part 20		
C.5.1		Conforms	Systems containing enough activity to be possibly classified as RW-IIa are located in the Auxiliary Building. The Auxiliary Building is Seismic Category I.
C.5.2		Conforms	
C.5.3	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building.
C.5.4	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building.
C6.1.1	Table 2	Conforms	
C6.1.2	Table 3	Conforms	
C6.1.3	Table 4	Conforms	
C6.1.4	Table 1 & 4	Conforms	
C6.2.1	UBC 1997, ASCE 7-95	~~Exception~~Conforms	The Radwaste Building is designed to UBC-199~~1~~7 and ASCE 7-98.
C6.2.2		Conforms	Shield structures, if used, will comply with Regulatory Guide 1.143, position C.6.2.
C.7	ANSI/ANS55.6-1993	Conforms	The quality assurance program for design, fabrication, procurement, and installation of radwaste systems is in accordance with the overall quality assurance program described in Chapter 17, which meets the requirements of Regulatory Guide 1.143, position C.6.

Reg. Guide 1.144 - Withdrawn

Reg. Guide 1.145, Rev. 1, 11/82 - Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

General		N/A	The atmospheric dispersion factors for use in determining potential accident consequences are selected to be representative of existing nuclear power plant sites and to

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.005

Question:

(Section 11.3) Please provide additional information addressing the following for the gaseous radwaste management systems:

A. demonstration of compliance with BTP ETSB 11.5, with a short-term (0-2 hours) release duration,

B. a discussion of compliance with GDC 3, "Fire Protection," as it relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen. The discussion should include the provisions incorporated in the AP1000 design to control releases due to hydrogen explosions in the gaseous waste management team. Additionally, it should include the type, number and locations of gas analyzers provided in the design of the gaseous waste management system,

C. a discussion of compliance with GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to control of releases of radioactive materials to the environment. The discussion should refer to RG 1.140 and should be consistent with Acceptance Criterion II.6.a of Section 11.3 of the SRP, and

D. a discussion of compliance with GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to radioactivity control in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas.

Westinghouse Response:

A. Branch Technical Position ETSB 11.5 states "radiological consequences of a single failure of an active component in a waste gas system with charcoal delay units assumes that the charcoal unit is bypassed with a 1-hour release to the environs."

In AP600 or AP1000, a failure in the Gaseous Radwaste System (WGS) could be isolated very quickly through several different means, the most direct being isolation of letdown. Therefore, although ETSB 11.5 mentions a 2 hour time interval associated with bringing an alternate charcoal unit on line or reducing reactor power, the AP1000 would be capable of stopping the bypass much more quickly, as noted below.

We have conducted an evaluation consistent with our understanding of the requirements of Branch Technical Position ETSB 11.5, which are:

- A pre-existing, beyond-design-basis condition of operation with 1% fuel defects; (that is, noble gas concentrations in the reactor coolant are assumed to be four times higher than the AP1000 design basis as shown in DCD Table 11.1-2),
- A 1-hour bypass of the WGS charcoal beds,
- 30 minutes decay prior to release to the environs, and
- The site boundary X/Q of 6.0E-4 sec/m^3, consistent with the assumptions documented in DCD Chapter 5 of the Tier 1 documentation. (Note that this is different from AP600, which assumed a X/Q of 1.0E-3 sec/m^3.)

With these assumptions we calculate a site boundary whole body dose of 0.1 rem.

B. Provisions of the AP1000 WGS to prevent an explosive mixture of oxygen and hydrogen are discussed in DCD section 11.3.1.2.3.1. Expanding upon what is stated in that section:

- The WGS operates at a pressure slightly above atmospheric, which prevents the ingress of oxygen which could occur in a sub-atmospheric system.

- A continuous purge flow of nitrogen is provided at the outlet of the WGS in order to prevent back-leakage of air through the discharge check valves.

- Redundant oxygen analyzers are provided for continuous sampling in a side stream taken off the process flow path. These analyzers create alarm messages both locally and in the main control room upon high oxygen level. Setpoints are specified to allow adequate time for operator action. The setpoint for alarm is adjustable, but is generally at about 1.25% concentration. This is approximately one fourth of the lower flammability limit for mixtures of hydrogen and oxygen. The side stream is returned to the process flow after sampling.

- The hydrogen analyzer is provided for direct measurement of hydrogen concentration in the sampling side stream. No specific alarm setpoint is assigned to this analyzer because of the broad range of expected hydrogen concentrations. By using the hydrogen analyzer reading and a flammability chart, the operator can assess the flammability potential of the gases during a hypothetical upset situation that injects oxygen into the system.

- The WGS is of welded construction. The piping and components are metallic conductors. The entire system is electrically at the same potential, which eliminates the buildup of static electricity and sparking.

- The WGS throttling and isolation valves are packless metal diaphragm type which eliminate leakage in or out of the system through the stem seals.

Since there is no normal source of oxygen into the system, automatic isolation of the oxygen source and automatic injection of nitrogen for dilution is not required.

These provisions satisfy GDC 3.

C. A brief discussion of compliance with GDC 60 is provided in DCD Section 3.1. Relative to the WGS, GDC 60 specifies that means should be included to suitably control the release of radioactive materials in gaseous effluents and that sufficient holdup capacity shall be provided, particularly where unfavorable site environmental conditions can be expected to impose limitations on the release of such effluents to the environment.

Suitable control of releases from the WGS is provided by the radiation monitoring system (discussed in DCD Section 11.5), which automatically terminates releases from the WGS when a high-activity setpoint is exceeded in the system discharge line.

Sufficient holdup time is provided by the WGS since the system can be isolated at any time. The system is not normally in operation, and is operated as necessary during changes in reactor coolant system boron concentration and when reductions in the reactor coolant system noble gas inventory are made. It is not expected that any alteration in the system operation will be necessary because of adverse meteorological conditions since anticipated operation in the system provides 61 days holdup of xenon isotopes and over 2 days holdup of krypton isotopes.

In addition to the WGS, the exhaust air from several ventilation systems also may contain airborne radioactivity. These systems are discussed in our response to AP1000 RAI 410.011, and below.

1. The radiologically controlled area ventilation system (VAS) serves the fuel handling area (which encloses the spent fuel pool) and radiologically controlled areas of the auxiliary and annex buildings. These areas are normally maintained at a slightly negative ambient air pressure with respect to adjacent clean plant areas and the environment to provide controlled release and monitoring of airborne effluents at the plant vent.

   Compliance with 10 CFR 20 effluent concentration limits has been evaluated using the PWR-GALE code (NUREG-0017) with the resulted reported in DCD Table 11.3-3, conservatively assuming that the exhaust air discharged by the VAS to the plant vent is unfiltered. This analysis indicates that no HEPA or charcoal filtration is required in order to keep normal plant releases within the specified limits. Therefore, the criteria set forth in RG 1.140 do not specifically apply to the VAS exhaust air system.

The exhaust air upstream of the plant vent is monitored for airborne radioactivity (discussed in DCD Section 11.5). In the event that abnormal airborne radioactivity is detected, the exhaust and supply air is automatically isolated to terminate unfiltered releases to the plant vent (discussed in DCD Section 9.4.3), and the containment air filtration system mitigates exfiltration of unfiltered airborne radioactivity (discussed in DCD Section 9.4.7).

2. The containment air filtration system provides HEPA and charcoal filtration of air exhausted from the containment during normal plant operation. Compliance with 10 CFR 20 effluent concentration limits has been evaluated using the PWR-GALE code (NUREG-0017) with the results reported in DCD Table 11.3-3, assuming that the exhaust filters remove 99 percent of the particulate contaminants and 90 percent of the radioiodides. This filtration system is discussed in DCD Table 9.4-1. The PWR-GALE analysis indicates that the specified filtration is adequate to keep normal plant releases within the specified limits.

   In the event of abnormal containment airborne radioactivity, the containment high-range radiation monitors will automatically isolate the VFS supply and exhaust air containment isolation valves. Additionally, dedicated radiation monitors in the VFS exhaust lines will also provide an alarm signal if abnormal releases to the environment are detected in the exhaust lines.

3. The solid radwaste building ventilation system (VRS) maintains the solid radwaste building at a slightly negative ambient air pressure with respect to the surrounding clean plant areas to provide controlled release and monitoring of airborne effluents at the plant vent.

4. The health physics and hot machine shop HVAC (VHS) maintains the areas that it serves at a slightly negative ambient air pressure with respect to the surrounding clean plant areas to provide controlled release and monitoring of airborne effluents at the plant vent.

See DCD Appendix 1A for a discussion of conformance with Regulatory Guides, and see the response to AP1000 RAI 410.007 for a more detailed discussion of Reg. Guide 1.140.

D. A brief discussion of compliance with GDC 61 is provided in DCD Subsection 3.1.6. The fuel storage and handling areas for the AP1000 include the fuel handling area of the auxiliary building, which encloses the spent fuel pool, and the containment building that encloses the reactor cavity. Based on the calculated radiological releases resulting from a design basis fuel handling accident (DCD Subsection 15.7.4) in either area, there is no need to provide safety-related isolation or filtration systems to maintain plant safety.

As discussed in Item C above, the ventilation systems serving these plant areas incorporate specific design features to mitigate the potential release of abnormal (non-DBA) airborne radioactivity from these areas. In addition to automatic isolation of the fuel handling area or containment purge valves on a high radiation signal, the isolation dampers or valves can be manually controlled from the main control room. The fuel handling area isolation dampers and containment isolation valves are provided with remote position indication to verify proper damper blade or valve disk position during isolation. During abnormal airborne radiological conditions, the containment purge valves can be manually opened to override a high radiation signal, through administrative procedures, to allow cleanup of the containment atmosphere.

The supply and exhaust lines that penetrate the fuel handling and containment areas include the ability to isolate subsystems during a loss of offsite power. The VFS can be manually connected to the onsite diesel generators in order to maintain the fuel handling area at a slightly negative air pressure during such events. The system is redundant. Isolation dampers and isolation valves are controlled by pneumatic operators that fail in a closed position on loss of electric power or air pressure.

DCD Subsection 9.4.7.4 states that the VFS exhaust air filtration units are designed and tested in accordance with ASME N509 and N510, which discuss instrumentation necessary for the periodic inspection and verification of system airflow rates, air temperatures, and filter pressure drops.

These provisions satisfy GDC 61.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 460.007

Question:

(Section 11.4) Staff guidance for the radioactive waste storage capacity that is sufficient to allow time for short-lived radionuclides to decay prior to shipping is discussed in SRP Section 11.4, Paragraphs II.6 and III.4, and BTP ETSB II-3, Position B.III. Please demonstrate that the AP1000 design has such sufficient storage capacity. Clarify which generation rate (expected or maximum) is used for this demonstration. Identify the in-plant storage space (spent resin tanks, spent filter tubes,...etc.) associated with different kind of wastes (wet wastes, dry wastes,...etc.), shipment capacity, and duration for in-plant storage.

Westinghouse Response:

BTP ETSB II-3 Position B.III has the following requirements:

III. WASTE STORAGE

1. *Tanks accumulating spent resins from reactor water purification systems should be capable of accommodating at least 60 days waste generation at normal generation rates. Tanks accumulating spent resins from other sources and tanks accumulating filter sludges should be capable of accommodating at least 30 days waste generation at normal generation rates.*
2. *Storage areas for solidified wastes should be capable of accommodating at least 30 days waste generation at normal generation rates. These storage areas should be located indoors.*
3. *Storage areas for dry wastes and packaged containment equipment should be capable of accommodating at least one full offsite waste shipment.*

As reported in Table 11.4-1 of the AP1000 DCD, the expected spent primary resin generation rate (using ANSI 18.1 source terms) is 400 cubic feet per year. This value is conservatively estimated, and includes the following:

	Vessel Volume ft^3	Change Freq., Months	Generation ft^3/yr
WLS Deep Bed Charcoal	10	3	40
WLS Deep Bed Resin	40	6	80
WLS 1 Resin	30	9	40
WLS 2 Resin	30	9	40
WLS 3 Resin	30	9	40
WGS Guard Bed Charcoal	8	36	3
WGS Delay Bed Charcoal	60	120	6
CVS Mixed Bed	50	18	33
CVS Cation Bed	50	36	17
SFS Resin	75	10	90

No radioactive filter sludges are generated.

Since as described in Table 11.4-10 the AP1000 incorporates two spent resin storage tanks, each with a nominal volume of 300 ft^3, the first point of BTP ETSB II-3 Position B.III is met. The on-site storage capacity for resin would typically be augmented by the use of a high integrity container, located in room 12374, which is the spent resin waste container fill station at the west end of the rail car bay of the auxiliary building.

As also described in DCD Table 11.4-1, the total expected solid radioactive waste generation rate is conservatively calculated to be 5759 ft^3 per year (5359 ft^3 per year excluding the resins discussed above), with a shipped volume of 1964 ft^3 per year (1454 ft^3 per year excluding resins). Handling of these wastes is described in DCD Section 11.4.2.1 and 11.4.2.3.

Low activity material, such as HVAC filters, ground sheets, boot covers, etc., will be packaged and stored before shipment in the radwaste building rooms 50351 and 50352. These rooms have a combined floor area of more than 2000 ft^2 and a combined volume of more than 25,000 ft^3, adequate for several years of waste storage.

Higher activity material, such as spent filter cartridges, are accumulated in the auxiliary building. Space is available within room 12374 (shared with the high integrity container for resin) and in room 12371, at the end of the rail bay.

These storage areas are adequate to meet items 2 and 3 of BTP ETSB II-3 Position B.III.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.008

Question:

(Section 11.4) In Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," the staff provided guidance to licensees on the addition of on-site storage facilities for low-level radioactive wastes generated on-site. The staff recognizes that the need for additional on-site storage facilities is a site-specific issue. However, this is not identified in Section 11.4.6 of the AP1000 DCD as COL action items. Please include it in Section 11.4.6 as a COL action item.

Westinghouse Response:

The DCD will be revised as shown in the attached pages.

Design Control Document (DCD) Revision:

Include a reference to Generic Letter 81-38 in Section 11.4.6, and add it as Reference 10 in Section 11.4.7.

PRA Revision:

None

The waste accumulation room (pre-processing) is divided as needed, using partitions and portable shielding to adjust the storage areas for different waste categories as needed to complement the radioactivity levels and volumes of generated wastes. The accumulation room has lockable doors to minimize unauthorized entry and inadvertent exposure.

The packaged waste storage room may be separated into high- and low-activity areas, using portable shielding to minimize exposure while providing operational flexibility. A lockable door is provided to minimize unauthorized entry and radiation exposure.

The heating and ventilating system for the radwaste building is described in subsection 9.4.8.

11.4.3 System Safety Evaluation

The solid waste management system has no safety-related function and therefore requires no nuclear safety evaluation.

11.4.4 Tests and Inspections

Preoperational tests are conducted as described in subsection 14.2.8. Tests are performed to demonstrate the capability to transfer ion exchange resins and deep bed filtration media from the ion exchangers and filters to the spent resin tanks or directly to a waste disposal container. Preoperational tests of the solid waste management system components are performed to prepare the system for operation.

After plant operations begin, the operability and functional performance of the solid waste management system is periodically evaluated according to Regulatory Guide 1.143 by monitoring for abnormal or deteriorating performance during routine operations. Instruments and setpoints are also calibrated on a scheduled basis. The preventive maintenance program includes periodic inspection and maintenance of active components.

11.4.5 Quality Assurance

The quality assurance program for design, installation, procurement, and fabrication issues of the solid waste management system is in accordance with the overall quality assurance program described in Chapter 17.

11.4.6 Combined License Information for Solid Waste Management System Process Control Program

The Combined License applicant will develop a process control program in compliance with 10 CFR Sections 61.55 and 61.56 for wet solid wastes and 10 CFR Part 71 and DOT regulations for both wet and dry solid wastes. Process control programs will also be provided by vendors providing mobile or portable processing or storage systems. It will be the plant operators responsibility to assure that the vendors have appropriate process control programs for the scope of work being contracted at any particular time. The process control program will identify the operating procedures for storing or processing wet solid wastes. The mobile systems process control program will include a discussion of conformance to Regulatory Guide 1.143 (Reference 7), Generic Letter GL-80-009

(Reference 8), and Generic Letter GL-81-039 (Reference 9) and, information of equipment containing wet solid wastes in the nonseismic Radwaste Building. In the event additional on-site storage facilities are a part of Combined License plans, this program will include a discussion of conformance to Generic Letter GL-81-038 (Reference 10)

11.4.7 References

1. "Shippers-General Requirements for Shipments and Packagings," 49 CFR 173.
2. "Packaging and Transportation of Radioactive Material," 10 CFR 71.
3. "Domestic Licensing of Production and Utilization Facilities," 10 CFR 50.
4. "Standards for Protection Against Radiation," 10 CFR 20.
5. "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR 61.
6. "USNRC Technical Position on Waste Form," Rev. 1, January 1991.
7. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
8. USNRC Generic Letter GL-80-009, "Low Level Radioactive Waste Disposal," dated January 29, 1980.
9. USNRC Generic Letter GL-81-039, "NRC Volume Reduction Policy (Generic Letter No. 81-39)," dated November 30, 1981.
10. USNRC Generic Letter GL-81-038, " Storage of Low-Level Radioactive Wastes at Power Reactor Sites," dated November 10, 1981.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.009

Question:

Section 11.4 of the AP1000 DCD states that solid waste management system does not handle large, radioactive waste materials such as core components or radioactive process wastes from the plant's secondary cycle. Where are these large, radioactive waste materials handled?

Westinghouse Response:

Large or highly radioactive core or primary components will be handled on a specialized basis. In general, these components can be held in the radwaste accumulation area or the spent fuel pool for decay, or decontaminated in place and in the hot machine shop and shipped to off-site facilities. Bays in the Radwaste Building can also be used for temporary staging and processing areas to decontaminate and package such components.

Secondary wastes will normally be non-radioactive, or (in the case of operation with steam generator tube leakage) activity levels will be extremely low. If condensate polishing resins become radioactive they will be transferred from the condensate polishing vessel directly to a temporary processing unit, or to the spent resin tank and then to a temporary processing unit. This processing unit would be located outside the turbine building, and will serve to dewater the resins and process them as required for offsite disposal. After packaging, the resins may be stored in the radwaste building until ready for shipment.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number. 460.010

Question:

SRP Section 11.5, Paragraph II.2.c of the acceptance criteria states that provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system. Please demonstrate that the AP1000 design meets this criterion.

Westinghouse Response:

AP1000 sample points are listed in DCD Tables 9.3.3-1 (sample points in the Primary Sampling System [PSS]) and 9.3.3-2 (sample points in other systems). Those tables identify the samples as either "Continuous" or "Grab."

For the purposes of those tables, "continuous" samples indicate on-line monitoring where the sample is not physically removed from its system of origin. Therefore, these sample streams meet the requirement of SRP Section 11.5 Paragraph II.2.c.

"Grab" samples are non-continuous, manual operations. These are discussed in DCD Sections 9.3.3.2.1 (liquid) and 9.3.3.2.2 (gas). As noted in those sections, purge capability is incorporated, with the purge flow routed to the effluent holdup tank in the Liquid Radwaste System (WLS).

The samples themselves are carried to on-site Radioactive Chemistry Laboratory for analysis. This laboratory has provisions for draining radioactive, reactor coolant-grade samples to the WLS effluent holdup tank (as shown on DCD Figures 9.3.3-1 and 11.2-2). Other radioactive samples are drained from the laboratory to the WLS chemical waste tank, as shown on DCD Figure 11.2-2.

A special provision is made in the PSS for containment atmospheric samples, which are purged and pumped to the containment sump, as illustrated on DCD Figure 9.3.3-1. As a contingency, this return connection could also be used for other samples from inside containment.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 460.011

Question:

SRP Section 11.5, Paragraph II.3 of the acceptance criteria states that provisions should be made for administrative and procedural control, for necessary auxiliary or ancillary equipment, for special features for the instrumented radiological monitoring sampling, and for analysis of process and effluent streams. Please demonstrate that the AP1000 design meets this criterion.

Westinghouse Response:

SRP Section 11.5 Paragraph II.3 cites Regulatory Guides 1.21 and 4.15.

As noted in DCD Tier 2 Chapter 1 Appendix A, conformance with Regulatory Guide 1.21 Sections C.1 and C.3 through C.14 is the responsibility of the Combined License applicant. Item C.2 gives requirements for locations of radiation monitors. AP1000 complies with this requirement, as demonstrated by the DCD Sections 9.3.3 and 11.5, particularly tables 9.3.3-1 and 9.3.3-2 (which list plant sample points) and tables 11.5-1 and 11.5-2 (which list radiation monitoring points). All major and potentially significant paths for release of radioactivity are monitored.

Regulatory Guide 4.15 provides guidance for quality assurance for radiation monitoring programs. As noted in DCD section 11.5.7, conformance with this Guide is the responsibility of the Combined License applicant.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.023

Question:

Section 6.4 states "The entries in Table 6-1 for 'Basis' refer to specific subsections of Section 6.3." The cross references to these subsections appear to be omitted from Table 6-1. Please provide them.

Westinghouse Response:

Table 6-1 has been updated to provide the reference to the analysis or justification that is the basis for declaring the event to be a success for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

The revised section 6.4 and table 6-1 will be included in the next revision of the AP1000 PRA (see attachment).

6.4 Event Tree Top Events Success Criteria

Three tables are provided in this section. They provide:

- Internal event tree top event success criteria and basis (6-1)
- Summary of success criteria for the mitigating systems (6-2)
- Summary of success criteria for operator actions and mission times (6-3)

In this section, the success criteria are described for each event tree top event case, and the bases for the success criteria are presented. For any given system modeled in the event trees, there are typically several cases. The different cases account for differences in availability of other systems or actuation signals, the availability of time for operator action, and so forth.

The results of this section are summarized in Table 6-1, which provides information on success criteria and bases, credited operator actions, system actuation, and affected event trees for each top event case, in alphabetical order by case name. Table 6-3 provides a summary of time windows for each operator action included in the cases, and also indicates in which cases each action is used.

~~The entries in Table 6-1 for "Basis" refer to specific subsections of Section 6.3. These subsections provide some additional description of the events in the sequence, other plant features operating, and the appropriate references for or descriptions of the bases for defining success~~ **The detail of events is provided in PRA Chapter 4. The entries in Table 6-1 for "Basis" refer to specific Chapter of DCD and PRA.**

The containment event tree success criteria and basis are not included in Table 6-1. These are discussed in PRA Chapter 40.

The shutdown event tree success criteria and basis are not included in Table 6-1 and 6-2. These are discussed in PRA Chapter 54.

Table 6-1 (Sheet 1 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis*
AC1A Accumulators	ACC	SI-LB	2 out of 2 check valves open in 1 out of 1 Accumulator lines for injection into the RCS through the intact injection line	None	None	Demand	A Chap10 A A3.3 D 6.3
AC2AB Accumulators	ACC	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW, LCOND, LCAS, LOSP, SLB-D, SLB-U, SLB-V, SPADS, MLOCA, CMTLB, SLOCA, RCSLK, PRSTR, SGTRC	2 out of 2 check valves open in 1 out of 2 Accumulator lines for injection into the RCS	None	None	Demand	A Chap10 A A3.3 D 6.3

*Key to BASIS column entries: ~~C = Calculated value~~D = DCD A = PRA Report~~specific analysis~~ ~~E = Engineering judgment~~
~~D = Design Basis, per DCD or other document~~ O = Other specific justification ~~T = Other transient analysis~~
S = not used in base AP1000 PRA, provided for sensitivity studies
Numbers in BASIS column indicate chapter or subsection number in which ~~additional~~ details are provided.

Table 6-1 (Sheet 2 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ACBOTH	ACC	LLOCA	2 out of 2 check valves open in both accumulator lines for injection into the RCS	None	None	Demand	D 15.6.5 A Chap10 A A3.4
ADA Full Depressurization PRHR Fails CMT Successful	ADS-F	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW, LCOND, LCAS, SLB-D, SLB-U, SLB-V, SLOCA, RCSLK, PRSTR, SGTRC	3 out of 4 ADS Stage 4 valves open **AND EITHER** 1 out of 4 ADS Stages 2,3 valves and isolation valves open to depressurize to Stage 4 automatic actuation setpoint **OR** manual PMS or DAS actuation	Credit is given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	A Chap11 A A3.2 D 6.3
ADAB Full Depressurization PRHR Fails, CMT Success, Power not recovered after LOOP	ADS-F	LOSP	3 out of 4 ADS Stage 4 valves open **AND EITHER** 1 out of 4 ADS Stages 2,3 valves and isolation valves open to depressurize to Stage 4 automatic actuation setpoint **OR** manual PMS or DAS actuation	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	~~A, O~~ A Chap11 A A3.2 D 6.3

Table 6-1 (Sheet 3 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ADAL Full Depressurization PRHR Fails, CMT Success, Power Restored Following LOSP with Diesel Generator Operation	ADS-F	LOSP	3 out of 4 ADS Stage 4 valves open **AND EITHER** 1 out of 4 ADS Stages 2, 3 valves and isolation valves open to depressurize to Stage 4 automatic actuation setpoint **OR** manual PMS or DAS actuation	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, ADN-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	A Chap11 A A3.2 D6.3
ADB Full Depressurization PRHR Fails, CMT Fails, Power not recovered after LOSP	ADS-F	LOSP	3 out of 4 ADS Stage 4 valves open	Credit given for manual actuation only. [Operator Actions: LPM-MAN02, ADN-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3
ADL Full Depressurization PRHR fails, CMT Fails, Power Restored Following LOSP	ADS-F	LOSP	3 out of 4 ADS Stage 4 valves open	Credit given for manual actuation only. [Operator Actions: LPM-MAN02, ADN-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3

Table 6-1 (Sheet 4 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ADM Full Depressurization CMT Successful	ADS-F	MLOCA, CMTLB, SI-LB	3 out of 4 ADS stage 4 valves open	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	A Chap11 A A3.2 D 6.3
ADMA Full Depressurization CMT Successful	ADS-F	LLOCA	3 out of 4 ADS stage 4 valves open	Credit given for automatic actuation only.	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2, 3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	A Chap11 A A3.4 D 6.3 D 15.6.5
ADQ Full Depressurization CMT Fails	ADS-F	MLOCA, CMTLB, SI-LB	3 out of 4 ADS stage 4 valves open	Credit given for manual actuation only. [Operator Actions: LPM-MAN02, AND-MAN01]	None	24 hours	M A Chap11 A A3.3 D 6.3
ADS Full Depressurization PRHR Successful, CMT Successful	ADS-F	SLOCA, RCSLK, PRSTR, SGTRC	3 out of 4 ADS stage 4 valves open	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	M A Chap11 A A3.2 D 6.3

Table 6-1 (Sheet 5 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ADT Full Depressurization CMT Fails	ADS-F	TRANS, LRCS, LMFW1, LMFW, POWEX, LCOND, LCCW, LCAS, SLB-D, SLB-U, SLB-V, SLOCA, PRSTR, RCSLK, SGTRC	3 out of 4 ADS Stage 4 valves open	Credit is given for manual actuation only. [Operator Actions: LPM-MAN01, AND-MAN01]	None	24 hours	M A Chap11 A A3.3 D 6.3
ADW Full Depressurization for ATWS	ADS-F	ATWS, ATW-S, ATW-T	3 out of 4 ADS Stage 4 valves open	Credit is given for manual actuation only. [Operator Actions: LPM-MAN01, AND-MAN01]	None	24 hours	E A Chap11 D 6.3 A A3.1

Table 6-1 (Sheet 6 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
AD1 Partial Depressurization PRHR Fails CMT Fails	ADS-P	TRANS, LRCS, LMFW1, LMFW, POWEX, LCOND, LCCW, LCAS, SLB-D, SLB-U, SLB-V, SLOCA, PRSTR, RCSLK, SGTRC	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit is given for manual actuation only. [Operator Actions: LPM-MAN01, AND-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3
AD1A Partial Depressurization PRHR Fails CMT Successful	ADS-P	TRANS, LRCS, LMFW1, LMFW, POWEX, LCOND, LCCW, LCAS, SLB-D, SLB-U, SLB-V, SLOCA, RCSLK, PRSTR, SGTRC	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit is given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay)	24 hours	A Chap11 A A3.2 D 6.3

Table 6-1 (Sheet 7 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA (Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ADF 1 stage Depressurization to equalize pressures following SGTR	PRDEP	SGTR	1 out of 2 ADS stage 1 valves and isolation valves open	Credit given for manual actuation only. [Operator Actions: ADF-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3
ADR Partial Depressurization PRHR Fails, CMT Fails, Power Restored Following LOOP	ADS-P	LOSP	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit given for manual actuation only. [Operator Actions: LPM-MAN02, AND-MAN01]	None	24 hours	A Chap11 A A3.3 D6.3
ADRA Partial Depressurization PRHR Fails, CMT Success, Power Restored Following LOOP	ADS-P	LOSP	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay)	24 hours	A Chap11 A A3.2 D 6.3
ADU Partial Depressurization CMT Successful	ADS-P	MLOCA	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay)	24 hours	P A Chap11 A A3.2 D 6.3

Table 6-1 (Sheet 8 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
ADUM Partial Depressurization PRHR Fails	ADS-P	MLOCA CMTLB	2 out of 4 ADS stages 2, 3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit given for manual actuation only. [Operator Actions: LPM-MAN02, AND-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3
ADV Partial Depressurization PRHR successful, CMT successful	ADS-P	SLOCA, RCSLK, PRSTR, SGTRC	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, AND-MAN01]	Automatic actuation via PMS (stage 1: low CMT water level plus CMT actuation signal; stages 2,3: stage 1 + delay; stage 4: low-2 CMT level + CMT stage 3 signal + delay + pressure interlock).	24 hours	A Chap11 A A3.2 D 6.3
ADZ Partial Depressurization PRHR successful, CMT Fails	ADS-P	SLOCA, RCSLK, PRSTR, SGTRC, SLB-U, SLB-D, SLB-V	2 out of 4 ADS stages 2,3 valves and isolation valves open **OR** 1 out of 4 ADS stage 4 valves open	Credit is given for manual actuation only. [Operator Actions: LPM-MAN01, AND-MAN01]	None	24 hours	A Chap11 A A3.3 D 6.3
BL Steamline break occurs inside containment	BL	SLB-U	Main steamline break occurs upstream of Main Steam Isolation Valves in portion of the line inside containment.	None	None	Not Applicable	C A Chap31

Table 6-1 (Sheet 9 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
CDS Condensate system provides makeup to deaerator storage tank following reactor trip *(This forms a part of the FWF, FWT, SFWT cases and is not used separately; also see COND)*	MFW, SFW	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW	1 out of 3 condensate pumps provides flow to the deaerator storage tank AND Condensate regulating valves provide automatic flow control	None.	Automatic PLS control	24 hours	D 10.4.7 A Chap14 O (1)
CIA Steam Line Isolation following steamline break	MSISO	SLB-D, SLB-U	Main steam isolation valve in each main steam line closed.	Credit given for manual closure of isolation valves if automatic actuation fails. [Operator Actions: CIA-MAN01]	Automatic actuation via PMS (low steam line pressure, low-2 SG NR level).	24 hours	~~D-1~~ D 15.1.5
CIB Manual SG Isolation following SGTR	SGISO	SGTR	One isolation valve in each line connected with the affected steam generator closed AND Reclosure of any faulted steam generator PORV or safety valves that initially open when steam lines are isolated (See SLSOV3)	Credit given only for manual closure of isolation valves. [Operator Actions: CIB-MAN00, CIB-MAN01]	None	24 hours	D 15.6.3

O (1) AP600 PRA Report Section 6.4.17

Table 6-1 (Sheet 10 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
CIC Containment Isolation prior to core damage	CIS	LLOCA	One isolation valve closed in each containment penetration line (subject to PRA screening criteria) in order to allow fewer containment recirculation paths (see RECIRC, RECIRC1)	Credit for automatic actuation, with manual actuation possible for smaller breaks. [Operator Actions: CIC-MAN01, where applicable]	Automatic Actuation via PMS (SI signal or high containment pressure)	Demand	~~D, A, O~~ A Chap24 A A3.5
CM1A Core Makeup Tanks	CMT	CMTLB, SI-LB	1 out of 1 Core Makeup Tank injects (1 out of 2 CMT isolation AOVs open per CMT).	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, CMN MAN01]	PMS actuation (low prz pressure or high ctmt pressure S signals), PMS and DAS actuation (low prz water level)	Demand	A Chap9 A A3.2 D 6.3
CM2AB Core Makeup Tanks	CMT	TRANS, LRCS, LMFW1, LMFW, POWEX, LCOND, LCCW, LCAS, ATWS, ATW-S, ATW-T	1 out of 2 Core Makeup Tanks inject (1 out of 2 CMT isolation AOVs open per CMT) AND Reactor coolant pumps trip (see RCT)	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, CMN MAN01]	PMS actuation (low wide range SG water level coincident with high hot leg temp), PMS and DAS actuation (low prz. level)	24 hours	A Chap9 A A3.2 D 6.3

Table 6-1 (Sheet 11 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
CM2LA Core Makeup Tanks	CMT	LLOCA SPADS	1 out of 2 Core Makeup Tanks inject (1 out of 2 CMT isolation AOVs open per CMT) in order to actuate IRWST gravity injection line squib valves on low-low CMT level	None credited.	PMS actuation (low prz pressure or high ctmt pressure S signals), PMS and DAS actuation (low prz water level)	Demand	A, O A Chap9 A A3.4 D 6.3
CM2NL Core Makeup Tanks	CMT	MLOCA	1 out of 2 Core Makeup Tanks inject (1 out of 2 CMT isolation AOVs open per CMT) **AND** Reactor coolant pumps trip (see RCN).	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, CMN-MAN01]	PMS actuation (low prz pressure or high ctmt pressure S signals), PMS and DAS actuation (low prz water level)	Demand	A Chap9 A A3.2 D 6.3
CM2P Core Makeup Tank	CMT	LOSP	1 out of 2 Core Makeup Tanks inject (1 out of 2 CMT isolation AOVs open per CMT).	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN02, CMN-MAN01]	PMS actuation (low wide range SG water level coincident with high hot leg temp), PMS and DAS actuation (low prz. level)	24 hours	A Chap9 A A3.2 D 6.3
CM2SL Core Makeup Tank	CMT	SLB-D, SLB-U, SLB-V, SLOCA, RCSLK, PRSTR, SGTRC	1 out of 2 Core Makeup Tanks inject (1 out of 2 CMT isolation AOVs open per CMT) **AND** Reactor coolant pumps trip (see RCL)	Credit given for manual actuation if automatic actuation fails. [Operator Actions: LPM-MAN01, CMN-MAN01]	PMS actuation (low prz pressure S signal), PMS and DAS actuation (low prz water level)	24 hours	A Chap9 A A3.2 D 6.3

Table 6-1 (Sheet 12 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
COND Main Condenser and full steam dump following reactor trip *(This forms a part of the FWF, FWT, SFWT cases and is not used separately; also see CDS)*	MFW, SFW	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW	3 out of 6 steam dump valves open to accommodate post-reactor-trip steam load to condenser and provide source of water for the deaerator storage tank **AND** Condenser available, including operation of 1 out of 2 Circulating Water trains to maintain condenser vacuum.	None	Automatic PLS control of steam dump, feedwater, and circulating water trains	24 hours	~~D, F~~ A Chap14 D 10.4.4 D 10.4.7
COND1 Main condenser and secondary heat removal via steam dump for SGTR depressurization	SGDEP	SGTR	1 out of 6 steam dump valves open for secondary heat removal **AND** Condenser available, including operation of 1 out of 2 Circulating Water trains to maintain condenser vacuum.	Credit given for manual steam dump valves control. [Operation Actions: CIB-MAN00, DUMP-MAN01]	Automatic PLS control of steam dump, feedwater, and circulating water trains	24 hours	~~D, F~~ A Chap14 D 10.4.4 D 10.4.7
CSAX CVS Auxiliary Prz Spray	PRDEP	SGTR	1 out of 2 CVS makeup pumps delivers flow to the pressurizer spray header	Credit given only for manual actuation, and for action to align suction to the spent fuel pool when BAT empties. [Operator Actions: CIB-MAN00, CVN-MAN00]	None.	6 hours	~~C, F~~ A Chap15 D 6.3 D 15.6.3

Table 6-1 (Sheet 13 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
CSBOR1 CVS RCS boration following ATWS	BOMAN	ATWS, ATW-T	1 out of 2 CVS makeup pumps delivers contents of boric acid tank to the reactor coolant system.	Credit given only for manual actuation. [Operator Actions: ATW-MAN11, CVN MAN02,]	None	6 hours	~~C, O~~ A Chap15 D 6.3 O (2)
CVS1 CVS RCS makeup	CVCS	PRSTR, SGTR RCSLK	1 out of 2 CVS makeup pumps delivers flow to the Reactor Coolant system	Credit given for operator action to start standby charging pump if aligned pump fails. [Operator Actions: CVN-MAN03]	Automatic PLS actuation (low prz water level relative to programmed level).	Demand	~~C, M, O, F~~ A Chap15 D 6.3 O (3)
CSP CVS or Manual control of PRHR	CVCS	SLB-D, SLB-U, SLB-V	1 out of 2 CVS makeup pumps delivers flow to the Reactor Coolant system **OR** PRHR manually regulated (stopped initially to limit cooldown, restarted as necessary to remove core heat)	Credit given for manual control of the PRHR system and for operator action to start standby charging pump if aligned pump fails. [Operator Actions: CVN-MAN03, PRN-MAN03]	Automatic CVS actuation via PLS (low prz water level relative to programmed level)or on CMT actuation signal.	6 hours (CVS), 24 hours (PRHR)	~~D, I, O~~ A Chap15 D 6.3 D 15.1.5
DAS DAS - MG-Set Trip & Turbine Trip, ATWS with loss of main feed	DAS	ATWS, ATW-S	Successful generation of DAS signal to trip control rod motor-generator (MG) sets	Credit for manual actuation of DAS if automatic actuation fails. [Operator Actions: ATW-MAN04]	Automatic DAS actuation on low WR SG level	Demand	~~D, O~~ A Chap27 O (4)

O (2) AP600 PRA Report Section 6.4.12
O (3) AP600 PRA Report Section 6.4.13
O (4) AP600 PRA Report Section 6.4.15

Table 6-1 (Sheet 14 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
DAS1 DAS - MG-Set Trip & Turbine Trip, ATWS with main feed available	DAS	ATW-T	Successful generation of DAS signal to trip control rod motor-generator (MG) sets	Credit for manual actuation of DAS if automatic actuation fails. [Operator Actions: ATW-MAN06]	Automatic DAS actuation on low WR SG level	Demand	~~D, O~~ A Chap27 O (4)
DGEN Emergency Diesel Generators operate following LOOP	DGEN	LOSP	Emergency Diesel Generators start to supply power to the associated AC power trains within 30 minutes of LOOP and continue until offsite power is restored **AND** Successful operation of any load shed and sequencing necessary for diesel generator operation	Credit given for manual diesel generator actuation (from the main control room) if automatic actuation fails [Operator Actions: ZON-MAN01]	Automatic actuation of each diesel generator on actuation of the associated bus undervoltage relay; Manual actuation via PLS	2.5 hours	~~O, F~~ A Chap22
FWF Main Feedwater (Loss of feed to 1 SG)	MFW	LMFW1	1 out of 2 main feedwater trains (condensate pump, FW booster pump, FW pump, FW control valve) delivers flow from deaerator storage tank to the 1 available steam generator **AND** Condenser and Turbine Bypass available (see COND) **AND** Feed from condenser to deaerator storage tank (see CDS)	None	Automatic control via PLS	24 hours	~~D~~ A Chap14 D 10.4.7 D 15.2.7

O (4) AP600 PRA Report Section 6.4.15

Table 6-1 (Sheet 15 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
FWT Main Feedwater available after reactor trip	MFW	TRANS, LRCS, POWEX, LCCW, ATW-T	1 out of 2 main feedwater trains (condensate pump, FW booster pump, FW pump, FW control valve) deliver flow from deaerator storage tank to 1 out of 2 steam generators **AND** Condenser and Turbine Bypass available (see COND) **AND** Feed from condenser to deaerator storage tank (see CDS)	None	Automatic control via PLS	24 hours	~~D~~ A Chap14 D 10.4.7 D 15.2.7
IW1A Gravity Injection	IRWST	SI-LB	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 1 intact injection line.	Credit given for manual activation of squib valves if automatic actuation fails. [Operator Actions: ADN-MAN01]	None	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3
IW1AM Gravity Injection	IRWST	SI-LB	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 1 intact injection line.	Credit given for manual actuation only [Operation Actions: ADN-MAN01]	None	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3

Table 6-1 (Sheet 16 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
IW2AB Gravity Injection	IRWST	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW, LCOND, LCAS, SLB-D, SLB-U, SLB-V, MLOCA, CMTLB, SLOCA, RCSLK, PRSTR, SGTRC, ATWS, ATW-S, ATW-T	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out 2 injection lines.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: ADN-MAN01]	Automatic actuation via PMS (low-low CMT level in 1 out of 2 CMTs.)	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3
IW2ABA Gravity Injection	IRWST	LLOCA SPADS	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	None	Automatic actuation via PMS (low-low CMT level in 1 out of 2 CMTs)	24 hours	~~D, A~~ A Chap12 A A3.4 D 6.3 D 15.6.5

Table 6-1 (Sheet 17 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
IW2ABM Gravity Injection	IRWST	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW, LCOND, LCAS, SLB-D, SLB-U, SLB-V	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	Credit given for manual actuation only. [Operator Actions: ADN-MAN01]	None	24 hours	~~D, A~~ A Chap12 A A3.3 D 6.3
IW2ABP Gravity Injection	IRWST	LOSP	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: ADN-MAN01]	Automatic actuation via PMS (low-low CMT level in 1 out of 2 CMTs)	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3
IW2ABPM Gravity Injection	IRWST	LOSP	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	Credit given for manual actuation only. [Operator Actions: ADN-MAN01]	None	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3
IW2ABB Gravity Injection	IRWST	LOSP	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: ADN-MAN01]	Automatic actuation via PMS (low-low CMT level in 1 out of 2 CMTs)	24 hours	~~D, A~~ A Chap12 A A3.2 D 6.3

Table 6-1 (Sheet 18 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
IW2ABBM Gravity Injection	IRWST	LOSP	IRWST injection into the RCS through 1 out of 2 check valve/squib valve paths in 1 out of 2 injection lines.	Credit given for manual actuation only. [Operator Actions: ADN-MAN01]	None	24 hours	D; A A Chap12 A A3.3 D 6.3
MGSET M-G sets trip	MGSET	ATWS, ATW-S, ATW-T	1 out of 2 MGSET exciter coil breakers open on 2 out of 2 MG sets following DAS signal.	None	Actuation via DAS	Demand	D O (5)
PCB Adequate passive containment cooling water drain, after SBO	CHR	LOSP	1 out of 3 PCS water drain valve paths open	None	None	24 hours	D A Chap13 D 6.2 D 6.3
PCP Adequate passive containment cooling water drain	CHR	LOSP	1 out of 3 PCS water drain valve paths open	None	None	24 hours	D A Chap13 D 6.2 D 6.3

O (5) AP600 PRA Report Section 6.4.19

Table 6-1 (Sheet 19 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
PCT Adequate passive containment cooling water drain	CHR	LLOCA, SPADS, MLOCA, CMTLB, SI-LB, SLOCA, PRSTR, RCSLK, SGTR, TRANS, LRCS, MFW1, POWEX, LCCW, LMFW, LCOND, LCAS, SLB-D, SLB-U, SLB-V, ATWS, ATW-S, ATW-T	1 out of 3 PCS water drain valve paths open	None	None	24 hours	D A Chap13 D 6.2 D 6.3
PRES Adequate Pressure Relief during ATWS	PRES	ATWS, ATW-S	2 out of 2 Pressurizer Safety Valves open	None	None	Demand	~~O, C~~ O (6)

O (6) AP600 PRA Report Section 6.4.20

Table 6-1 (Sheet 20 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
PRESU Adequate Pressure Relief during ATWS	PRES	ATWS, ATW-S	2 out of 2 Pressurizer Safety Valves in open position **AND** adequate reactivity feedback to maintain RCS pressure below ASME service level C limit	None	None	Demand	S . O (6)
PRI PRHR Isolation	PRISO	PRSTR	1 out of 1 MOVs on the PRHR inlet line close **AND** 2 out of 2 AOVs on PRHR outlet line reclose.	Credit given only for manual actuation. [Operator Actions: PRI-MAN01]	None	24 hours	~~G~~, F A Chap8 A A3.3 D 6.3
PRL PRHR	PRHR	PRSTR, RCSLK, SLOCA, SGTRC	1 out of 2 AOVs in the PRHR heat exchanger outlet line opens to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	Credit given for manual actuation if automatic actuation fails. [Operator Actions: PRN-MAN02, HPM-MAN01]	Automatic actuation via PMS (CMT actuation signal) or DAS (low wide-range SG level OR high RCS hot leg temperature, depending on the initiating event)	24 hours	~~D~~ A Chap8 A A3.2 D 6.3 D 15.6.3
PRP PRHR following LOOP	PRHR	LOSP	1 out of 2 AOVs on the PRHR heat exchanger outlet line opens to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	Credit given for manual actuation if automatic actuation fails. [Operator Actions: PRN-MAN02, HPM-MAN01]	Automatic actuation via PMS (low NR SG water level coincident with low SFW flow), PMS and DAS (low WR SG water level), or DAS (high hot leg temperature	24 hours	~~D~~ A Chap8 A A3.2 D 6.3 D Chap15

O (6) AP600 PRA Report Section 6.4.20

Table 6-1 (Sheet 21 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
PRS PRHR	PRHR	SGTR	1 out of 2 AOVs on the PRHR heat exchanger outlet line opens to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	Credit given for manual actuation if automatic actuation fail. [Operator Actions: PRN-MAN02, HPM-MAN01]	Automatic actuation via PMS (CMT actuation signal following low prz pressure S signal)	24 hours	~~D~~ A Chap8 A A3.2 D Chap15
PRT PRHR	PRHR	TRANS, LRCS, LMFW1, LMFW, POWEX, LCCW, LCOND, SLB-U, SLB-D, SLB-V	1 out of 2 AOVs on the PRHR heat exchangers outlet line open to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	Credit given for manual actuation if automatic actuation fails. [Operator Actions: PRN-MAN01, HPM-MAN01]	Automatic actuation via PMS (low NR SG water level coincident with low SFW flow), PMS and DAS (low WR SG water level), or DAS (high hot leg temperature)	24 hours	~~D~~ A Chap8 A A3.2 D 6.3 D Chap15
PRTA PRHR	PRHR	ATWS, ATW-S	1 out of 2 AOVs on the PRHR heat exchanger outlet line open to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	None	Automatic actuation via PMS (low NR SG water level coincident with low SFW flow), PMS and DAS (low WR SG water level), or DAS (high hot leg temperature	24 hours	~~D, O, F~~ A A4.2

Table 6-1 (Sheet 22 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
PRW PRHR	PRHR	LCAS, LOSP	1 out of 2 AOVs on the PRHR heat exchangers common outlet line open to allow flow through the heat exchanger **AND** 1 out of 2 AOVs in the IRWST gutter drain line close to divert water into the IRWST	None	Valves will open on loss of support (air or dc power) resulting from the initiating event. Automatic actuation via PMS or DAS (low WR SG water level) would also be expected	24 hours	~~D~~ A Chap8 A A3.2 D 6.3
PRSOV Pressurizer Safety Valves reclose	PRSOV	TRANS, LRCS, LMFW1, POWEX, LMFW, LCCW, LCAS, LOSP, ATWS, LCOND	2 out of 2 pressurizer safety valves reclose after opening	None	None	Demand	~~O, E~~ O (7)
R05 Grid recovery	R05	LOSP	Offsite power recovered within 0.5 hour.	None	None	Demand	~~I, O~~ O (8)
RCL RCP Trip	CMT	SLOCA, SGTR, PRSTR, RCSLK, SLB-D, SLB-U, SLB-V	4 out of 4 Reactor Coolant Pumps tripped.	Credit for manual tripping of RCPs if automatic trip fails. [Operator Actions: RCN-MAN01]	Automatic actuation via PMS (CMT actuation), or DAS (low prz level).	Demand	~~D, E~~ D Chap15 O (9)

O (7) AP600 PRA Report Section 6.4.23
O (8) AP600 PRA Report Section 6.4.24
O (9) AP600 PRA Report Section 6.4.25

Table 6-1 (Sheet 23 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RCN RCP Trip	CMT	MLOCA, CMTLB, SI-LB	4 out of 4 Reactor Coolant Pumps tripped.	Credit for manual tripping of RCPs if automatic trip fails. [Operator Actions: RCN-MAN01]	Automatic actuation via PMS (CMT actuation), or DAS (low prz level).	Demand	~~D, F~~ ~~6-2~~ D Chap15
RCT RCP Trip	CMT	TRANS, LMFW1, POWEX, LMFW, LCOND, LCCW, LCAS, ATWS, ATW-S, ATW-T	4 out of 4 Reactor Coolant Pumps tripped.	Credit for manual tripping of RCPs if automatic trip fails. [Operator Actions: RCN-MAN01]	Automatic actuation via PMS (CMT actuation).	Demand	~~D, F~~ ~~6-4~~ D Chap15
RECIRCB Water Recirculation to RCS, station blackout	RECIR	LOSP	1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A, O~~ A Chap12 A A3.5

Table 6-1 (Sheet 24 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RECIRC1B Water Recirculation to RCS, after station blackout, with failure of containment isolation	RECIR	LOSP	2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A, O~~ A Chap12 A A3.5
RECIRC Water Recirculation to RCS	RECIR	TRANS, LRCS, LMFW1, POWEX, LMFW, LCOND, LCCW, LCAS, SLB-U, SLB-D, SLB-V, LLOCA, SPADS, MLOCA, CMTLB, SI-LB, SLOCA PRSTR, RCSLK, SGTR, ATWS, ATW-S, ATW-T	1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A, O~~ A Chap12 A A3.5

Table 6-1 (Sheet 25 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA (Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RECIRC1 Water Recirculation to RCS, with failure of containment isolation	RECIR	TRANS, LRCS, LMFW1, POWEX, LMFW, LCOND, LCCW, LCAS, SLB-U, SLB-D, SLB-V, LLOCA, SPADS, MLOCA, CMTLB, SI-LB, SLOCA PRSTR, RCSLK, SGTR, ATWS, ATW-S, ATW-T	2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A, O~~ A Chap12 A A3.5

Table 6-1 (Sheet 26 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RECIRCP Water Recirculation to RCS, LOOP with power recovery	RECIR	LOSP	1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A, O~~ A Chap12 A A3.5
RECIRC1P Water Recirculation to RCS, LOOP with power recovery, with failure of containment isolation	RECIR	LOSP	2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. **AND** 1 IRWST injection path remains open; each IRWST path has a check valve in series with a squib valve	Credit given for manual opening of squib recirculation valves if automatic actuation fails. [Operator Actions: REN-MAN02, REN-MAN04, RHN-MAN06]	Automatic actuation via PMS (Low-3 IRWST water level and ADS signal)	24 hours	~~A,~~ O A Chap12 A A3.5

Table 6-1 (Sheet 27 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RNH RNS Injection/Cooling to RCS	NRHR	TRANS, LRCS, LMFW1, POWEX, LMFW, LCOND, LCAS, SLB-U, SLB-D, SLB-V, SLOCA, MLOCA, SI-LB, CMTLB, PRSTR, RCSLK, SGTR, ATWS, ATW-S, ATW-T	1 out of 2 RNS pumps inject from the SFS cask washdown pit into the RCS (i.e., 2 out of 2 containment isolation MOVs open **AND** 1 out of 2 NRHR pumps operate) **AND** CCW cooling is provided to an operable RNS HX **AND EITHER** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if only one RNS pump is running) **OR** 2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. (if both pumps are running) **OR** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running AND operators throttle RNS flowrate to limit pump suction requirements).	Credit given for manual actuation of ~~NRHR~~RNS injection. Credit is given for manual backup of automatic opening of the recirculation line squib valves. [Operator Actions: RHN-MAN01, REN-MAN02, REN-MAN04, RHN-MAN06]	None	24 hours	~~D, O~~ A Chap17 A A3.2 A A3.3 D 6.3

Table 6-1 (Sheet 28 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RNHP RNS Injection/Cooling to RCS, LOOP with power recovery	NRHR	LOSP	1 out of 2 RNS pumps inject from the SFS cask washdown pit into the RCS (i.e., 2 out of 2 containment isolation MOVs open **AND** 1 out of 2 NRHR pumps operate) **AND** CCW cooling is provided to an operable RNS HX **AND EITHER** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if only one RNS pump is running) **OR** 2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve. (if both pumps are running) **OR** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running AND operators throttle RNS flowrate to limit pump suction requirements).	Credit given for manual actuation of ~~NRHR~~ RNS injection. Credit is given for manual backup of automatic opening of the recirculation line squib valves. [Operator Actions: RHN-MAN01, REN-MAN02, REN-MAN04, RHN-MAN06]	None	24 hours	~~D, O~~ A Chap17 A A3.2 A A3.3 D 6.3

Table 6-1 (Sheet 29 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RNP RNS LOOP with power recovery	NRHR	LOSP	1 out of 2 RNS pumps inject from the SFS cask washdown pit into the RCS (i.e., 2 out of 2 containment isolation MOVs open **AND** 1 out of 2 NRHR pumps operate) **AND EITHER** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if only one RNS pump is running) **OR** 2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running) **OR** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running AND operators throttle RNS flowrate to limit pump suction requirements).	Credit given for manual actuation of NRHRRNS injection. Credit is given for manual backup of automatic opening of the recirculation line squib valves. [Operator Actions: RHN-MAN01, REN-MAN02, REN-MAN04, RHN-MAN06]	None	24 hours	D.9 A Chap17 A A3.2 A A3.3 D 6.3

Table 6-1 (Sheet 30 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RNR RNS	NRHR	TRANS, LRCS, LMFW1, POWEX, LMFW, LCOND, LCAS, SLB-U, SLB-D, SLB-V, SLOCA, MLOCA, CMTLB, PRSTR, RCSLK, SGTR, ATWS, ATW-S, ATW-T	1 out of 2 RNS pumps inject from the cask washdown pit into the RCS (2 out of 2 containment isolation MOVs open AND 1 out of 2 NRHR pumps/heat exchangers operate) **AND EITHER** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if only one RNS pump is running) **OR** 2 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running) **OR** 1 out of 4 containment recirc paths open; 2 paths have a check valve in series with a squib valve and the other 2 paths have a squib valve (if both pumps are running AND operators throttle RNS flowrate to limit pump suction requirements).	Credit given for manual actuation of NRHRRNS injection. Credit is given for manual backup of automatic opening of the recirculation line squib valves. [Operator Actions: RHN-MAN01, REN-MAN02, REN-MAN04, RHN-MAN06]	None	24 hours	D, O A Chap17 A A3.2 A A3.3 D 6.3

Table 6-1 (Sheet 31 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RNN RNS Cooling of IRWST	NRHR	TRANS, LRCS, LMFW1, POWEX, LMFW, LCOND, LCAS	1 out of 2 RNS pumps circulate IRWST water with CCW cooling (i.e., 2 out of 2 containment isolation MOVs open **AND** 1 out of 2 NRHR pumps operate) **AND** CCW cooling is provided to an operable RNS HX	Credit given for manual actuation of ~~NRHR~~RNS actuation. [Operator Actions: RNH-MAN01, RNA-MAN09]	None	24 hours	~~D, O~~ A Chap17 D 5.4.7.1
RNNP RNS Cooling of IRWST, LOOP with power recovery	NRHR	LOSP	1 out of 2 RNS pumps circulate IRWST water with CCW cooling (i.e., 2 out of 2 containment isolation MOVs open **AND** 1 out of 2 NRHR pumps operate) **AND** CCW cooling is provided to an operable RNS HX	Credit given for manual actuation of ~~NRHR~~RNS actuation. [Operator Actions: RNH-MAN01, RNH-MAN09]	None	24 hours	~~D, O~~ A Chap17 D 5.4.7.1
RTPMS Reactor Trip via PMS	RTPMS	ATWS, ATW-S	Trip signal provided to the reactor trip breakers in both protection system cabinets **AND** 1 out of 2 reactor trip breakers open on each path of 1 out of 2 protection system cabinets	Credit is given for manual actuation as a backup to automatic actuation [Operator Actions: ATW-MAN03]	Automatic actuation via PMS (several signals anticipated for each initiating event)	Demand	~~D, O~~ A A4.2 O (10)

O (10) AP600 PRA Report Section 6.4.28

Table 6-1 (Sheet 32 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
RTPMS1 Reactor Trip via PMS	RTPMS	ATW-T	Trip signal provided to the reactor trip breakers in both protection system cabinets **AND** 1 out of 2 reactor trip breakers open on each path of 1 out of 2 protection system cabinets	Credit is given for manual actuation as a backup to automatic actuation [Operator Actions: ATW-MAN05]	Automatic actuation via PMS (several signals anticipated for each initiating event)	Demand	~~D, O, E~~ A A4.2 O (10)
RTSTP Manual Actuation of Rod Control System for RCCA Insertion	RTSTP	ATWS, ATW-S	Operator manually initiates control rod insertion via rod control system **AND** at least one bank of control rods insertion for at least 1 minute	Credit is given only for manual actuation of the rod control system [Operator Actions: ATW-MAN01]	Manual actuation via PLS for rod control system.	Demand	S O (11)
SDMAN Manual controlled shutdown	SDMAN	RCSLK	Operators perform a controlled shutdown of the reactor after recognizing that an RCS leak is in progress	Credit is given only for manual action [Operator Actions: RTN-MAN01]	Control systems necessary to establish stable shutdown	Demand	~~E, O~~ D 9.3.6.4 A 4.8.1
SFW Startup Feedwater (condensate system unavailable)	SFW	LCOND	1 out of 2 Startup Feedwater pumps deliver flow from the condensate storage tank to 1 out of 2 SGs.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: FWN-MAN02, HPM-MAN01, REG-MAN00]	Automatic actuation via PLS (low NR SG water level or low FW flow)	24 hours	~~D, O~~ A Chap14 D 10.4.7 D 15.2.7
SFW1 Startup Feedwater (SGTR)	SFW	SGTR	1 out of 2 Startup Feedwater pumps delivers flow from the condensate storage tank to the intact steam generator.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: FWN-MAN02, HPM-MAN01, REG-MAN00]	Automatic actuation via PLS (low NR SG water level or low FW flow)	24 hours	~~D, O~~ A Chap14 D 10.4.7 D 15.6.3

O (10) AP600 PRA Report Section 6.4.28
O (11) AP600 PRA Report Section 6.4.29 – Note that for AP1000 this event case has no meaning since the UET is 0.

Table 6-1 (Sheet 33 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
SFWA Startup Feedwater (ATWS)	PRHR2	ATWS, ATW-S	1 out of 2 startup feedwater pumps deliver flow from the condensate storage tank to 2 out of 2 steam generators	None	Automatic actuation via PLS (low NR SG water level or low FW flow)	24 hours	~~O, F~~ S (1)
SFWM Startup Feedwater (Instrument Air failure)	SFW	LCAS	1 out of 2 Startup Feedwater trains deliver flow from the condensate storage tank to 1 out of 2 SGs **AND** Operators locally operate feed reg valves	Credit for manual actuation if automatic actuation fails, plus required manual action to regulate SFW flow following loss of instrument air initiating event. [Operator Actions: FWN-MAN02, HPM-MAN01, REG-MAN00]	Automatic actuation via PLS (low NR SG water level or low FW flow)	24 hours	~~D, O~~ A Chap14 D 10.4.7 D 15.2.7
SFWP Startup Feedwater (LOOP)	SFW	LOSP	1 out of 2 Startup Feedwater pumps deliver flow from the condensate storage tank to 1 out of 2 SGs, following power recovery or successful diesel generator start/load.	Credit given for manual actuation if automatic actuation fails. [Operator Actions: FWN-MAN03, HPM MAN01, REG-MAN00]	Automatic actuation via PLS (low NR SG water level or low FW flow)or diesel generator sequence	24 hours	~~D, O~~ A Chap14 D 10.4.7 D 15.2.7
SFWT Startup Feedwater	SFW	TRANS, LRCS, LMFW1, POWEX, LCCW, LMFW	1 out of 2 Startup Feedwater pumps deliver flow to 1 out of 2 steam generators from the condensate storage tank **OR** 1 out of 2 Startup Feedwater pumps deliver flow to 1 out of 2 steam generators from the deaerator storage tank AND Condenser and Turbine Bypass available (see COND) **AND** Feed from condenser to deaerator storage tank (see CDS)	Credit given for manual actuation if automatic actuation fails. [Operator Actions: FWN-MAN02, HPM MAN01, REG-MAN00]	Automatic actuation and control via PLS (low NR SG water level or low FW flow)	24 hours	~~D, O~~ A Chap14 D 10.4.7 D 15.2.7

S (1) Sensitivity study has been performed on AP1000 that shows this case to be successful

Table 6-1 (Sheet 34 of 34)

SUMMARY OF EVENT TREE TOP EVENTS SUCCESS CRITERIA
(Internal Initiating Events at Power Cases)

Event Case Name	Top Event Name	Event Trees	Success Criteria	Manual Actions	Dependencies and Modeled Actuations	Mission Time	Basis
SGHL SG overfill protection for SGTR	SGISO	SGTR	Chemical and Volume Control system isolated. **AND** Startup FW flow to faulted SG isolated.	Credit given for manual actuation and automatic actuation. [Operator Actions: SGHL-MAN01]	Automatic actuation via PMS (High-2 steam generator water level).	Demand	~~C, D~~ D 15.6.3
SGTR Tube rupture given MSLB or consequential SLB with 2 SG Blowdown	NSGTR	SLB-D, SLB-U, SLB-V	No consequential steam generator tube rupture in either blowing down steam generator	None	None.	Demand	~~O, C~~ A Chap31 O (12)
SGTR1 Tube Rupture given MSLB with 1 SG Blowdown	NSGTR	SLB-U	No consequential steam generator tube rupture in the blowing down steam generator	None	None	Demand	~~O, C~~ A Chap31 O (12)
SLSOV Reclosure of Main Steamline Valves	SLSOV	TRANS, POWEX, LCCW, LRCS	Reclosure of the power-operated relief valve (PORV) or associated PORV block valve, **AND** reclosure of the one open safety valve in each main steamline, given unavailability of condenser steam dump	None	Steam generator PORV actuation via PLS	Demand	~~D, E, C~~ O (13)
SLSOV1 Reclosure of Main Steamline Valves	SLSOV	LMFW1, LMFW, LCCW, LCOND, LOSP	Reclosure of the power-operated relief valve (PORV) or associated PORV block valve, **AND** reclosure of the two open safety valves in each main steamline, given unavailability of condenser steam dump	None	Steam generator PORV actuation via PLS	Demand	~~D, E, C~~ O (13)
SLSOV2 Reclosure of Main Steamline Valves	SLSOV	LCAS	Reclosure of one PORV and one safety valve on the main steamline from the ruptured generator	None	Steam generator PORV actuation via PLS	Demand	~~D, E, C~~ O (13)
SLSOV3 Reclosure of Main Steamline Valves	SGISO	SGTR	Reclosure of one PORV and one safety valve on the main steamline from the ruptured steam generator	None	Steam generator PORV actuation via PLS	Demand	~~D, E, C~~ O (13)

O (12) AP600 PRA Report Section 6.4.33 O (13) AP600 PRA Report Section 6.4.34

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.024

Question:

Table 6-1 provides a "Summary of Event Tree Top Events Success Criteria." For each event case a basis for success is provided and referred to as "Calculated value," "Design Basis," "Provided for sensitivity studies," "PRA specific analysis," "Other specific justification," "Engineering judgment," or "Other transient analysis."

For each event case in Table 6-1, please provide or give specific reference to the analysis or justification that is the basis for declaring the event case to be a success for AP1000. If the justification involves analyses for AP600, justify that the analysis is applicable to AP1000.

Westinghouse Response:

Table 6-1 has been updated to provide the reference to the analysis or justification that is the basis for declaring the event to be a success for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

The revised section 6.4 and table 6-1 will be included in the next revision of the AP1000 PRA (see RAI 720.023).

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.038

Question:

An important objective of the AP600 design certification PRA was to identify important PRA insights and assumptions and make sure that they have been addressed in the design certification through design certification requirements, such as requirements for inspection, tests, analyses and acceptance criteria (ITAAC), the requirement for a design reliability assurance program (D-RAP), and combined operating license (COL) action items. These requirements were incorporated in the Design Control Document (DCD), Table 19.59-29 "PRA-based insights," to ensure that any future plant which references the design will be built and operated in a manner that is consistent with important assumptions made in the design certification PRA. Please provide similar information for AP1000. Since the major part of this information is expected to be the same as for AP600, please start with DCD Table 19.59-29 and highlight the differences in "insights" between the two designs.

Westinghouse Response:

Requested information will be incorporated in the AP1000 DCD Section 19-59 and PRA Chapter 59.

The AP600 DCD table 19.59-29 with AP1000 markup identifying the differences in "insights" between AP600 and AP1000 is provided here after.

Table 19.59-29 (Sheet 1 of 26) ~~AP600~~ AP1000 PRA-BASED INSIGHTS		
	INSIGHT	**DISPOSITION**
1.	The passive core cooling system (PXS) is composed of the following: - Accumulator subsystem - Core makeup tank (CMT) subsystem - In-containment refueling water storage tank (IRWST) subsystem - Passive residual heat removal (PRHR) subsystem. The automatic depressurization system (ADS), which is part of the reactor coolant system (RCS), also supports passive core cooling functions.	
1a.	The accumulators provide a safety-related means of safety injection of borated water to the RCS. The following are some important aspects of the accumulator subsystem as represented in the PRA:	6.3.2
	- There are two accumulators, each with an injection line to the reactor vessel/direct vessel injection (DVI) nozzle. Each injection line has two check valves in series.	Tier 1 Information
	- The reliability of the accumulator subsystem is important. The accumulator subsystem is included in the ~~RAPD~~-RAP.	17.4
	- Diversity between the accumulator check valves and the CMT check valves minimizes the potential for common cause failures.	6.3.2

Table 19.59-29 (Sheet 2 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
1b. ADS provides a safety-related means of depressurizing the RCS.	Tier 1 Information
The following are some important aspects of ADS as represented in the PRA:	
ADS has four stages. Each stage is arranged into two separate groups of valves and lines. - Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST - Stage 4 discharges from the hot leg to the RCS loop compartment.	Tier 1 Information
Each stage 1, 2, and 3 line contains two motor-operated valves (MOVs).	Tier 1 Information
Each stage 4 line contains an MOV valve and a squib valve.	Tier 1 Information
The valve arrangement and positioning for each stage is designed to reduce spurious actuation of ADS. - Stage 1, 2, and 3 MOVs are normally closed and have separate controls. - Each stage 4 squib valve ~~has redundant, series controllers~~actuation requires signals from two separate PMS cabinets. - Stage 4 is blocked from opening at high RCS pressures.	6.3.2 & 7.3
The ADS valves are automatically and manually actuated via the protection and safety monitoring system (PMS), and manually actuated via the diverse actuation system (DAS).	Tier 1 Information
The ADS valves are powered from Class 1E ~~dc~~-power.	Tier 1 Information
The ADS valve positions are indicated and alarmed in the control room.	6.3.7
Stage 1, 2, and 3 valves are stroke-tested every cold shutdown. Stage 4 squib valve actuators are tested every 2 years for 20% of the valves.	3.9.6
Because of the potential for counter-current flow limitation in the surgeline, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation following an extended loss of RNS when the RCS is open during shutdown operations.	6.3.3.4.3
ADS 4th stage squib valves receive a signal to open during shutdown conditions using PMS low hot leg level logic.	6.3.3.4.3
The reliability of the ADS is important. The ADS is included in the ~~RAPD~~-RAP.	17.4

Table 19.59-29 (Sheet 3 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
1.b (cont.) ADS is required by the Technical Specifications to be available in Modes 1 through 6 without the cavity flooded.	16.1
Stages 1, 2, and 3, connected to the top of the pressurizer, provide a vent path to preclude pressurization of the RCS during shutdown conditions if decay heat removal is lost.	16.1
Depressurization of the RCS through ADS minimizes the potential for high-pressure melt ejection events. - Procedures will be provided for use of the ADS for depressurization of the RCS after core uncovery.	Emergency Response Guidelines
The ADS mitigates high pressure core damage events which can produce challenges to containment integrity due to the following severe accident phenomena: - High pressure melt ejection - Direct containment heating - Induced steam generator tube rupture - Induced RCS piping rupture and rapid hydrogen release to containment	19.36
1c. The CMTs provide safety-related means of high-pressure safety injection of borated water to the RCS.	6.3.1
The following are some important aspects of CMT subsystem as represented in the PRA:	
There are two CMTs, each with an injection line to the reactor vessel/DVI nozzle. - Each CMT has a normally open pressure balance line from an RCS cold leg. - Each injection line is isolated with a parallel set of air-operated valves (AOVs). - These AOVs open on loss of Class 1E dc power, loss of air, or loss of the signal from the PMS. - The injection line for each CMT also has two normally open check valves in series.	6.3.2

Table 19.59-29 (Sheet 4 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
1c. (cont.)	
The CMT AOVs are automatically and manually actuated from PMS and DAS.	Tier 1 Information
CMT level instrumentation provides an actuation signal to initiate automatic ADS and provides the actuation signal for the IRWST squib valves to open.	6.3.1 & 7.3.1
The CMT AOV positions are indicated and alarmed in the control room.	6.3.7
CMT AOVs are stroke-tested quarterly.	3.9.6
The CMTs are risk-important for power conditions because the level indicators in the CMTs provide an open signal to ADS and to the IRWST squib valves as the CMTs empty. - The CMT subsystem is included in the ~~RAPD~~ RAP.	17.4
CMT is required by the Technical Specifications to be available in Modes 1 through ~~6~~ 5 with RCS pressure boundary intact.	16.1
1d. IRWST subsystem provides a safety-related means of performing the following functions: - Low-pressure safety injection following ADS actuation - Long-term core cooling via containment recirculation - Reactor vessel cooling through the flooding of the reactor cavity by draining the IRWST into the containment.	6.3
The following are some important aspects of the IRWST subsystem as represented in the PRA:	
IRWST subsystem has the following flowpaths: - Two (redundant) injection lines from IRWST to reactor vessel/DVI nozzle. Each line is isolated with a parallel set of valves; each set with a check valve in series with a squib valve. - Two (redundant) recirculation lines from the containment to the reactor vessel/DVI injection line. Each recirculation line has two paths: one path contains a squib valve and a MOV, the other path contains a squib valve and a check valve. - The two MOV/squib valve lines also provide the capability to flood the reactor cavity.	Tier 1 Information
There are screens for each IRWST injection line and recirculation line.	Tier 1 Information

Table 19.59-29 (Sheet 5 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
1d. (cont.)	
Squib valves provide the pressure boundary and prevent the check valves from normally seeing a high delta-P.	6.3.3
Squib valves and MOVs are powered by Class 1E ~~dc~~ power.	Tier 1 Information
The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and manually actuated via DAS.	Tier 1 Information
The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS and DAS from the control room.	Tier 1 Information
~~Diversity of the squib valves in the injection lines and recirculation lines minimizes the potential for common cause failure between injection and recirculation/reactor cavity flooding~~ **The injection squib valves and the recirculation squib valves in series with check valves are diverse from the other recirculation squib valves in order to minimize the potential for common cause failure between injection and recirculation / reactor cavity flooding.**	6.3.2
Automatic IRWST injection at shutdown conditions is provided using PMS low hot leg level logic.	~~7.3.1~~6.3.3.4.3 & 7.3.1
The positions of the squib valves and MOVs are indicated and alarmed in the control room.	6.3.7
IRWST injection and recirculation check valves are exercised at each refueling. IRWST injection and recirculation squib valve actuators are tested every 2 years for 20% of the valves (This does not require valve actuation). IRWST recirculation MOVs are stroke-tested quarterly.	3.9.6
The reliability of the IRWST subsystem is important. The IRWST subsystem is included in the ~~RAPD~~ **RAP**.	17.4
IRWST injection and recirculation are required by Technical Specifications to be available in Modes 1 through 6 without the cavity flooded.	16.1
The operator action to flood the reactor cavity is determined in Emergency Response Guideline AFR-C.1, which instructs the operator to flood the reactor cavity ~~if injection to the RCS cannot be recovered or containment radiation reaches a level that indicates fission product releases as determined by a core damage assessment guideline~~ **when the core-exit thermocouples reach 1200F.**	Emergency Response Guidelines
PXS recirculation valves are automatically actuated by a low IRWST level signal or manually from the control room, if automatic actuation fails.	6.3

Table 19.59-29 (Sheet 6 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

	INSIGHT	DISPOSITION
1e.	Passive residual heat removal (PRHR) provides a safety-related means of performing the following functions: - Removes core decay heat during accidents - Allows automatic termination of RCS leak during a steam generator tube rupture (SGTR) without ADS - Allows plant to ride out an ATWS event without rod insertion.	6.3.1 & 6.3.3 PRA App. A4
	The following are some important aspects of the PRHR subsystem as represented in the PRA:	
	PRHR is actuated by opening redundant parallel air-operated valves. These air-operated valves open on loss of Class 1E power, loss of air, or loss of the signal from PMS.	6.3.2
	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	Tier 1 Information
	Diversity of the PRHR air-operated valves from the CMT air-operated valves minimizes the probability for common cause failure of both PRHR and CMT air-operated valves.	6.3.2
	Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. ~~If the steam condensation does not return to the IRWST, the IRWST volume is sufficient for at least 72 hours of PRHR operation~~. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
	Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
	The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
	PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation ~~evert~~ every 10 years.	3.9.6
	PRHR is required by Technical Specifications to be available from Modes 1 through ~~6~~ 5 with RCS pressure boundary intact.	16.1

Table 19.59-29 (Sheet 7 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
1e. (cont.) The PRHR HX, in conjunction with the PCS, can provide core cooling for an indefinite period of time. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST: - IRWST gutter and its drain isolation valves are safety-related - These isolation ~~V~~valves ~~that redirect the flow~~ are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	6.3.2.1.1 & 6.3.7.6
- These ~~I~~isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during ~~safe/cold~~ shutdown conditions with the RCS intact.	16.1
2. The protection and safety monitoring system (PMS) provides a safety-related means of performing the following functions: - Initiates automatic and manual reactor trip - Automatic and manual actuation of engineered safety features (ESF).	Tier 1 Information
PMS monitors the safety-related functions during and following an accident as required by Regulatory Guide 1.97	7.1.1
~~PMS has four divisions of reactor trip and ESF actuation. PMS automatically produces a safety-related reactor trip or ESF initiation upon an attempt to bypass more than two channels of a function that uses 2-out-of-4 logic~~ **PMS initiates an automatic reactor trip and an automatic actuation of ESF. PMS provides manual initaition of reactor trip. PMS 2-out-of-4 initiation logic reverts to a 2-out-of-3 coincidence logic if one of the 4 channels is bypassed. PMS does not allow simultaneous bypass of 2 redundant channels.**	Tier 1 Information
PMS has redundant divisions of safety-related post-accident parameter display.	~~7.1.2.6 & Figure 7.1-8~~ 7.5.2.2.1 & 7.5.4
Each PMS division is powered from its respective Class 1E dc and UPS division.	Tier 1 Information
PMS provides fixed position controls in the control room.	Tier 1 Information

Table 19.59-29 (Sheet 8 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
2. (cont.) Reliability of the PMS is provided by the following: - The reactor trip functions are divided into two ~~functionally diverse~~ subsystems. - The ESF functions are processed by two microprocessor-based subsystems that are functionally identical in both hardware and software.	7.1.2.~~2~~1.1 7.1.2.2~~.6 &~~ ~~7.1.2.3.1~~
Four sensors normally monitor variables used for an ESF actuation. These sensors may monitor the same variable for a reactor trip function.	7.3.1
Continuous automatic PMS system monitoring and failure detection/alarm is provided.	7.1.2
PMS equipment is designed to accommodate a loss of the normal heating, ventilation, and air conditioning (HVAC). PMS equipment is protected by the passive heat sinks upon failure or degradation of the active HVAC.	~~7.1.4.1.6~~ 3.11 & 6.4
The reliability of the PMS is important. The PMS is included in the ~~RAPD~~ RAP.	17.4
The PMS software is designed, tested, and maintained to be reliable under a controlled verification and validation program written in accordance with IEEE 7-4.3.2 (1993) that has been endorsed by Regulatory Guide 1.152. Elements that contribute to a reliable software design include: - A formalized development, modification, and acceptance process in accordance with an approved software QA plan (paraphrased from IEEE standard, section 5.3, "Quality") - A verification and validation program prepared to confirm the design implemented will function as required (IEEE standard, section 5.3.4, "Verification and Validation") - Equipment qualification testing performed to demonstrate that the system will function as required in the environment it is intended to be installed in (IEEE standard, section 5.4, "Equipment Qualification") - Design for system integrity (performing its intended safety function) when subjected to all conditions, external or internal, that have significant potential for defeating the safety function (abnormal conditions and events) (IEEE standard, section 5.5, "System Integrity") - Software configuration management process (IEEE standard, section 5.3.5, "Software Configuration Management").	App 1A (Compliance with Reg. Guide 1.152)

Table 19.59-29 (Sheet 9 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
3. The diverse actuation system (DAS) provides a nonsafety-related means of performing the following functions: - Initiates automatic and manual reactor trip - Automatic and manual actuation of selected engineered safety features.	Tier 1 Information
Diversity is assumed in the PRA that eliminates the potential for common cause failures between PMS and DAS. - The DAS automatic actuation signals are generated in a ~~functionally~~ diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architecture, different hardware implementations, and different software.	Tier 1 Information
DAS provides control room displays and fixed position controls to allow the operators to take manual actions.	7.7.1
DAS actuates using 2-out-of-2 logic. Actuation signals are output to the loads in the form of normally de-energized, energize-to-actuate signals. The normally de-energized output state, along with the dual 2-out-of-2 redundancy, reduces the probability of inadvertent actuation.	7.7.1.11
The actuation devices of DAS and PMS are capable of independent operation that is not affected by the operation of the other. The DAS is designed to actuate components only in a manner that initiates the safety function.	7.7.1.11
The DAS reactor trip function is to trip the control rods by deenergizing the motor-generator set.	7.7.1.11
In the PRA it is assumed the following eliminates the potential for common cause failures between automatic and manual DAS functions. - DAS manual initiation functions are implemented in a manner that bypasses the signal processing equipment of the DAS automatic logic.	Tier 1 Information
The DAS, including the M-G set field breakers, is included in the ~~RAPD~~ RAP.	17.4
4. The plant control system (PLS) provides a nonsafety-related means of controlling nonsafety-related equipment. - Automatic and manual control of nonsafety-related functions, including "defense-in-depth" ~~systems~~ functions. - Provides control room indication for monitoring overall plant and nonsafety-related system performance.	7.1.3 & 7.~~1~~7.1
PLS has appropriate redundancy to minimize plant transients.	7.1.3 & 7.7.1.12
PLS provides capability for both automatic control and manual control.	7.1.3

Table 19.59-29 (Sheet 10 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
4. (cont.) ~~Redundant signal selectors provide PLS with the ability to obtain inputs from the integrated protection cabinets in the PMS. The signal selector function maintains the independence of the PLS and PMS.~~ Signal selector algorithmes provide the PLS with the ability to obtain inputs form the PMS. The signal selectors algorithms select those protection system signals that represent the actual status of the plant and reject erroneous signals.	7.1.3.2
PLS control functions are distributed across multiple distributed controllers so that single failures within a controller do not degrade the performance of control functions performed by other controllers.	7.1.3.1
5. The onsite power system consists of the main ac power system and the dc power system. The main ac power system is a non-Class 1E system. The dc power system consists of two independent systems: the Class 1E dc system and the non-Class 1E dc system.	Tier 1 Information
5a. The onsite main ac power system is a non-Class 1E system comprised of a normal, preferred, and standby power ~~systems~~supplies.	8.3.1.1
The main ac power system distributes power to the reactor, turbine, and balance of plant auxiliary electrical loads for startup, normal operation, and normal/emergency shutdown.	8.3.1.1.~~3~~1
The arrangement of the buses permits feeding functionally redundant pumps or groups of loads from separate buses and enhances the plant operational reliability.	8.3.1.1.1
During power generation mode, the turbine generator normally supplies electric power to the plant auxiliary loads through the unit auxiliary transformers. During plant startup, shutdown, and maintenance, the main ac power is provided from the high-voltage switchyard. The onsite standby power system powered by the two onsite standby diesel generators supplies power to selected loads in the event of loss of normal and preferred ac power supplies.	8.3.1.1.1
Two onsite standby diesel generator units, each furnished with its own support subsystems, provide power to the selected plant nonsafety-related ac loads.	8.3.1.1.2.1
On loss of power to a ~~4160~~ 6900 V diesel-backed bus, the associated diesel generator automatically starts and produces ac power. The normal source circuit breaker and bus load circuit breakers are opened, and the generator is connected to the bus. Each generator has an automatic load sequencer to enable controlled loading on the associated buses.	Tier 1 Information

Table 19.59-29 (Sheet 11 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
5b. The Class 1E dc and uninterruptible power supply (UPS) system (IDS) provides reliable power for the safety-related equipment required for the plant instrumentation, control, monitoring, and other vital functions needed for shutdown of the plant.	8.3.2.1
There are four independent, Class 1E 125 Vdc divisions. Divisions A and D each consists of one battery bank, one switchboard, and one battery charger. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions is designated as the 24-hour battery bank. The second battery bank in Divisions B and C is designated as the 72-hour battery bank.	Tier 1 Information
The 24-hour battery banks provide power to the loads required for the first 24 hours following an event of loss of all ac power sources concurrent with a design basis accident. The 72-hour battery banks provide power to those loads requiring power for 72 hours following the same event.	Tier 1 Information
Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from non-Class 1E 480 Vac diesel-generator-backed motor control centers.	8.3.2.1.1.1
The 24-hour and the 72-hour battery banks are housed in ventilated rooms apart from chargers and distribution equipment.	8.3.2.1.3
Each of the four divisions of dc systems are electrically isolated and physically separated to prevent an event from causing the loss of more than one division.	8.3.2.1.3
The Class 1E batteries are included in the ~~RAPD~~-RAP.	17.4
5c. The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provide dc and uninterruptible ac power to nonsafety-related loads.	Tier 1 Information
The non-Class 1E dc and UPS system consists of two subsystems representing two separate power supply trains.	8.3.2.1.2
EDS load groups 1, 2, and 3 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system.	Tier 1 Information
The onsite standby diesel-generator-backed 480 Vac distribution system provides the normal ac power to the battery chargers	Tier 1 Information
The batteries are sized to supply the system loads for a period of at least two hours after loss of all ac power sources	8.3.2.1.2

Table 19.59-29 (Sheet 12 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
6. The normal residual heat removal system (RNS) provides a safety-related means of performing the following functions: - Containment isolation for the RNS lines that penetrate the containment - Isolation of the reactor coolant system at the RNS suction and discharge lines - Pathway for ~~L~~long-term, post-accident makeup of containment inventory.	Tier 1 Information
RNS provides a nonsafety-related means of core cooling through: - RCS recirculation cooling during ~~at~~ shutdown conditions - Low pressure pumped ~~injection from the IRWST~~makeup flow from the SFS cask loading pit and long-term ~~pumped~~ recirculation from the IRWST and the containment. - Heat removal from IRWST during PRHR operation	5.4.7
The RNS has redundant pumps and heat exchangers. The pumps are powered by non-Class 1E power with backup connections from the diesel generators.	5.4.7 & 8.3
RNS is manually aligned from the control room to perform its core cooling functions. The performance of the RNS is indicated in the control room.	5.4.7
The RNS containment isolation and pressure boundary valves are safety-related. The motor-operated valves are powered by Class 1E dc power.	Tier 1 Information
The RNS containment isolation MOVs are automatically and manually actuated via PMS.	7.3.1.2.20
Interfacing system loss-of-coolant accident (LOCA) between the RNS and the RCS is prevented by: - Each RNS line is isolated by at least three valves. - The RNS equipment outside containment is capable of withstanding the operating pressure of the RCS. - The RCS isolation valves are interlocked to prevent their opening at RCS pressures above its design pressure.	5.4.7.2.2
CCS provides cooling to the RNS heat exchanger.	Tier 1 Information
Planned maintenance affecting the RNS cooling function and its support systems CCS and SWS should be performed in modes 1, 2, and 3, when the RNS is not normally operating.	16.3

Table 19.59-29 (Sheet 13 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
7. The component cooling water system (CCS) is a nonsafety-related system that removes heat from various components and transfers the heat to the service water system.	Tier 1 Information
The CCS has redundant pumps and heat exchanger.	Tier 1 Information
During normal operation, one CCS pump is operating. The standby pump is aligned to automatically start in case of a failure of the operating CCS pump.	9.2.2.4.2
The CCS pumps are automatically loaded on the standby diesel generator in the event of a loss of normal ac power. The CCS, therefore, continues to provide cooling of required components if normal ac power is lost.	9.2.2.4.5.4
8. The service water system (SWS) is a nonsafety-related system that transfers heat from the component cooling water heat exchangers to the atmosphere.	Tier 1 Information
The SWS has redundant pumps, strainers, and cooling tower cells.	9.2.1.2.1
During normal operation, one SWS train of equipment is operating. The standby train is aligned to automatically start in case of a failure of the operating SWS pump.	9.2.1.2.3.3
The SWS pumps and cooling tower fans are automatically loaded onto their associated diesel bus in the event of a loss of normal ac power. Both pumps and cooling tower fans automatically start after power from the diesel generator is available.	9.2.1.2.3.6
9. The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution and to preserve containment integrity by isolation of the CVS lines penetrating the containment.	Tier 1 Information
The CVS provides a nonsafety-related means to perform the following functions: - Makeup water to the RCS during normal plant operation - Boration following a failure of reactor trip - ~~Coolant~~ Makeup water to the pressurizer auxiliary spray line.	Tier 1 Information
Two makeup pumps are provided. Each pump provides capability for normal makeup.	9.3.6.3.1
Two safety-related air-operated valves provide isolation of normal CVS letdown during shutdown operation on low hot leg level.	9.3.6.7

Table 19.59-29 (Sheet 14 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
10. The operation of RNS and its support systems (CCS, SWS, main ac power and onsite power) is RTNSS-important for shutdown decay heat removal during reduced RCS inventory operations.	16.3
- These systems are included in the ~~RAPD~~ RAP.	17.4
Short-term availability controls for the RNS during at-power conditions reduce PRA uncertainties.	16.3
11. The information used by the COL regarding critical human actions (if any) and risk-important tasks from the PRA, as presented in Chapter 18 of the ~~SSAR~~ DCD on human factors engineering, is important in developing and implementing procedures, training, and other human reliability related programs.	18
12. Sufficient instrumentation and control is provided at the remote shutdown workstation to bring the plant to safe shutdown conditions in case the control room must be evacuated.	7.4.3
There are no differences between the main control room and remote shutdown workstation controls and monitoring that would be expected to affect safety system redundancy and reliability.	7.4.3.1.1
13. Separation or protection of the equipment and cabling among the divisions of safety-related equipment and separation of safety-related from nonsafety-related equipment minimizes the probability that a fire or flood would affect more than one safety-related system or train, except in some areas inside containment where equipment will be capable of achieving safe shutdown prior to damage.	3.4.1.1.2, 9.5.1.1.1, 9.5.1.2.1.1 & 9A
Although the containment is a single fire area, adequate design features exist for separation (structural or space), suppression, lack of combustibles, or operator action to ensure the plant can achieve safe shutdown.	9A
To prevent flooding in a radiologically controlled area (RCA) in the Auxiliary Building from propagating to non-radiologically controlled areas, the non-RCAs are separated from the RCAs by 2 and 3-foot walls and floor slabs. In addition, electrical penetrations between RCAs and non-RCAs in the Auxiliary Building are located above the maximum flood level.	3.4.1.2.2.2

Table 19.59-29 (Sheet 15 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
14. The following minimizes the probability for fire and flood propagation from one area to another and helps limit risk from internal fires and floods:	
- Fire barriers are sealed, to the extent possible (i.e., doors).	9.5.1.2.1.1
- Structural barriers which function as flood barriers are watertight below the maximum flood level.	3.4.1.1.2
- Establishing administrative controls to maintain the performance of the fire protection system is the responsibility of the COL applicant.	Table 9.5.1-1, Item 29
15. Fire detection and suppression capability is provided in the design. Flooding control features and sump level indication are provided in the design.	3.4.1, 9.5.1.2.1.2, & 9.5.1.8
Establishing administrative controls to maintain the performance of the fire protection system is the responsibility of the COL applicant.	Table 9.5.1-1, Item 29
16. ~~AP600~~ AP1000 main control room fire ignition frequency is limited as a result of the use of low-voltage, low-current equipment and fiber optic cables.	7.1.2 & 7.1.3
There is no cable spreading room in the ~~AP600~~ AP1000 design.	Table 9.5.1-1
17. Redundancy in control room operations is provided within the control room itself for fires in which control room evacuation is not required.	9.5.1.2.1.1
18. The remote shutdown workstation provides redundancy of control and monitoring for safe shutdown functions in the event that main control room evacuation is required.	7.4.3 & 9.5
The remote shutdown workstation is in a fire and flood area separate from the main control room.	3.4.1.2.2.2, 7.1.2, 7.4.3.1.1. & 9A.3.1.2.5
19. Although a main control room fire may defeat manual actuation of equipment from the main control room, it will not affect the automatic functioning of safe shutdown equipment via PMS or manual operation from the remote shutdown workstation. This is because the ~~ESF and protection logic~~ PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the main control room.	7.1.2.7 & 9A.3

Table 19.59-29 (Sheet 16 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
20. The main control room has its own ventilation system, and is pressurized. This prevents smoke, hot gases, or fire suppressants originating in areas outside the control room from entering the control room via the ventilation system.	9.4.1
There are separate ventilation systems for safety-related equipment divisions (A & C and B & D). This prevents smoke, hot gases, or fire suppressants originating from one fire area to another to the extent that they could adversely affect safe shutdown capabilities.	9.4.1 9.5.1.1.1
The ventilation system for the remote shutdown workstation is independent of the ventilation system for the main control room.	9.4.1
21. ~~AP600~~ AP1000 does not rely on ac power sources for safe shutdown capability since the safety-related passive systems do not require ac power sources for operation. Individual fires resulting in loss of offsite power or affecting onsite standby diesel generator operability do not affect safe shutdown capability.	8.1.4.2
22. Containment isolation functions are not compromised by internal fire or flood. Redundant containment isolation valves in a given line are located in separate fire and flood areas or zones and, if powered, are served by different control and electrical divisions.	6.2.3
One isolation component in a given line is located inside containment, while the other is located outside containment, and the containment wall is a fire/flood barrier.	6.2.3, ~~&~~ 9.5 & 9A
23. The ~~AP600~~ AP1000 design minimizes potential flooding sources in safety-related equipment areas, to the extent possible. The design also minimizes the number of penetrations through enclosure or barrier walls below the probable maximum flood level. Walls, floors, and penetrations are designed to withstand the maximum anticipated hydrodynamic loads.	3.4.1
24. The Combined License applicant will confirm the ~~AP600~~ AP1000 certified design will review differences between the as-built plant and the basis for the ~~AP600~~ AP1000 seismic margin analysis.	19.59.10.65
25. The depressurization of the reactor coolant system below 150 psi facilitates in-vessel retention of molten core debris.	19.36

Table 19.59-29 (Sheet 17 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
26. The reflective reactor vessel insulation provides an engineered flow path to allow the ingression of water and venting of steam for externally cooling the vessel in the event of a severe accident involving core relocation to the lower plenum. The reflective insulation panels and support members can withstand pressure differential loading due to the IVR boiling phenomena. Water inlets and steam vents are provided at the entrance and exit of the insulation boundary.	19.39, 5.3.5 & Tier 1 Information
The reactor vessel insulation is included in the ~~RAPD~~ RAP.	17.4
~~No coatings are applied to the outside surface of the reactor vessel which will inhibit the wettability of the surface.~~	
27. The reactor cavity design provides a reasonable balance between the regulatory requirements for sufficient ex-vessel debris spreading area and the need to quickly submerge the reactor vessel for the in-vessel retention of core debris.	19.39 & Appendix 19B
28. The design can withstand a best-estimate ex-vessel steam explosion without failing the containment integrity.	Appendix 19B
29. The containment design incorporates defense-in-depth for mitigating direct containment heating by providing no significant direct flow path for the transport of particulated molten debris from the reactor cavity to the upper containment regions.	Appendix 19B
30. The hydrogen control system is comprised of passive autocatalytic recombiners (PARs) and hydrogen igniters to limit the concentration of hydrogen in the containment during accidents and beyond design basis accidents, respectively.	Tier 1 Information
Operability of the hydrogen igniters is addressed by short-term availability controls during modes 1, 2, 5 (with RCS pressure boundary open), and 6 (with upper internals in place ~~and~~ or cavity levels less than full).	16.3
The operator action to activate the igniters is the first step in ERG AFR.C-1 to ensure that the igniter activation occurs prior to rapid cladding oxidation.	Emergency Response Guidelines

Table 19.59-29 (Sheet 18 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
31. Mitigation of the effects of a diffusion flames on the containment shell are addressed by the following containment layout features:	1.2, General Arrangement Drawings
- **Vents from the PXS and CVS compartments (where hydrogen releases can be postulated) to the CMT room are located well away from the containment shell and containment penetrations. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to adress severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107'-2".**. ~~are not adjacent to the containment wall and penetrations or are hatched and locked closed~~.	**3.4.1.2.2.1 & 19.41.7**
- IRWST vents ~~near~~ **are designed so that those located away from the containment wall open to vent hydrogen releases. In this situation IRWST vents located close to the containment wall would not open because flow of hydrogen through the other vents would not result in a IRWST pressure sufficient to open them.**~~the containment wall are turned to direct releases away from the containment shell~~	**6.2.4.5.1**
32. The containment structure can withstand the pressurization from a LOCA and the global combustion of hydrogen released in-vessel (10 CFR 50.34(f)).	19.41
33. The steam generator should not be depressurized to cool down the RCS if water is not available to the secondary side. This action protects the tubes from large pressure differential and minimizes the potential for creep rupture. The COL will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914.	19.59.10
34. Depressurizing the RCS and maintaining a water level covering the SG tubes on the secondary side can mitigate fission product releases from a steam generator tube rupture accident. The COL will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914.	19.59.10
35. Loss of ac power does not contribute significantly to the core damage frequency. - Nonsafety-related containment spray does not need to be ac independent.	19.59
36. ~~AP600~~ AP1000 has a nonsafety-related containment spray system.	6.5.2
Containment spray is not credited in the PRA. Failure of the nonsafety-related containment spray does not prevent the plant achieving the safety goals.	19.59
The COL will develop and implement severe accident management guidance for operation of the nonsafety-related containment spray system using the suggested framework provided in WCAP-13914.	19.59.10

37. Passive containment can withstand severe accidents without PCS water cooling the containment shell. Air cooling alone is sufficient to maintain containment pressure below failure pressure **with high probability**. ~~Flooding of the PCS annulus due to failure of the annulus drains is a PRA-postulated mechanism for the failure of PCS cooling.~~	19.40

Table 19.59-29 (Sheet 19 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
38. Operation of ADS stage 4 provides a vent path for the severe accident hydrogen to the steam generator compartments, bypassing the IRWST, and mitigating the conditions required to produce a diffusion flame near the containment wall. ~~The openings from the PXS valve/accumulator rooms and CVS compartments that can vent hydrogen to the maintenance floor are either located away from the containment wall and electrical penetration junction boxes or are covered by a secure hatch. This mitigates the effect of postulated diffusion flames.~~	19.41
39. Containment isolation valves controlled by DAS are important in limiting offsite releases following core melt accidents. The containment isolation valves are included in the ~~RAPD~~ RAP. Operability of DAS for selected containment isolation actuations is addressed by short-term availability controls.	17.4 16.3
40. Reflooding the reactor pressure vessel through the break can have a significant effect on a severe accident by quenching core debris, achieving a controlled stable state, and producing hydrogen.	19.38 & 19.41
41. The type of concrete used in the basemat is not important. The reactor cavity design incorporates features that extend the time to basemat melt-through in the event of RPV failure. The cavity design includes: - A minimum floor area of 48 m^2 available for spreading of the molten core debris - A minimum thickness of concrete above the embedded containment liner of 0.85 m - There is no piping buried in the concrete beneath the reactor cavity; sump drain lines are not enclosed in either of the reactor cavity floor or reactor cavity sump concrete. Thus, there is no direct pathway from the reactor cavity to outside the containment in the event of core-concrete interactions. - The openings between the reactor cavity and cavity sump are small diameter openings in which core debris in the cavity will solidify. Thus, there is no direct pathway for core debris to enter the sump, except in the case where it might spill over the sump curbing.	Appendix 19B Appendix 19B
42. No safety-related equipment is located outside the Nuclear Island.	1.2 & 3.4.1

Table 19.59-29 (Sheet 20 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
43. Capability exists to vent the containment ~~via the RNS suction lines to the spent fuel pool, with the RCS depressurized and open to the containment atmosphere via either the ADS or the vessel failure~~. The COL will develop and implement severe accident management guidance for venting containment using the suggested framework provided in WCAP-13914.	Appendix 19D 19.59.10
44. A list of risk-important systems, structures, and components (SSCs) has been provided in the D-RAP. The risk-significant SSCs are included in the ~~RAP~~ D-RAP.	17.4 17.4
45. The Combined License applicant referencing the ~~AP600~~ AP1000 certified design will review differences between the as-built plant and the design used as the basis for the ~~AP600~~ AP1000 PRA and Table 15.59-29. If the effects of the differences are shown, by a screening analysis, to potentially result in a significant increase in core damage frequency or large release frequency, the PRA will be updated to reflect these differences.	19.59.10
46. There are no watertight doors used for flood protection in the ~~AP600~~ AP1000 design. Plugging of the drain headers is minimized by designing them large enough to accommodate more than the design flow and by making the flow path as straight as possible.	3.4.1.1.2 9.3.5.1.2
47. The maintenance guidelines as described in the Shutdown Evaluation Report (WCAP-14837) should be considered when developing the plant specific operations procedures.	13.5.1
48. Transient combustibles should be controlled.	Table 9.5.1-1, Items 77-83
49. There are two compartments inside containment (PXS-A and PXS-B) containing safe shutdown equipment ~~other than containment isolation valves~~ **that normally do not flood** ~~are floodable (i.e.,~~**although they are below the maximum flood height**~~)~~. Each of these two compartments contains redundant and essentially identical equipment (one accumulator with associated isolation valves as well as isolation valves for one CMT, one IRWST injection line, and one containment recirculation line). **A pipe break in one of these compartments can cause that room to flood.** These two compartments are physically separated to ensure that a flood in one compartment does not propagate to the other. Drain lines from the PXS-A and PXS-B compartments to the reactor vessel cavity and steam generator compartment are protected from backflow by redundant backflow preventers.	3.4.1.2.2.1

Table 19.59-29 (Sheet 21 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
50. There are ~~four~~ seven automatically actuated containment isolation valves inside containment subject to flooding. These ~~four~~ seven normally closed containment isolation valves would not fail open as a result of the compartment flooding. Also, there is a redundant, normally closed, containment isolation valve located outside containment in series with each of these valves.	3.4.1.2.2.1
51. The passive containment cooling system (PCS) cooling water not evaporated from the vessel wall flows down to the bottom of the ~~inner~~ containment annulus. Two 100-percent drain openings, located in the side wall of the Shield Building, are always open with screens provided to prevent entry of small animals into the drains.	19.40
52. The major rooms housing divisional cabling and equipment (the battery rooms, dc equipment rooms, I&C rooms, and penetration rooms) are separated by 3-hour fire rated walls. Separate ventilation subsystems are provided **for A and C and for B and D division rooms.** In order for a fire to propagate from one divisional room to another, it must move past a 3-hour barrier (e.g., a door) into a common corridor and enter the other room through another 3-hour barrier (e.g., another door).	9.5.1 & 9A.3
53. An access bay in the turbine building is provided to protect the north end of the Auxiliary Building, from potential debris produced by a postulated seismic damage of the adjacent Turbine Building.	~~19.55.5~~1.2
54. There are no normally open connections to sources of "unlimited" quantity of water in the electrical and I&C portions of the Auxiliary Building such as that it could affect safe shutdown capabilities.	Figure 9.5.1-1
55. To prevent flooding in a radiologically controlled area (RCA) in the Auxiliary Building from propagating to non-RCAs, the non-RCAs are separated from the RCAs by 2- and 3-foot walls and floor slabs. In addition, electrical penetrations between RCAs and non-RCAs in the Auxiliary Building are located above the maximum flood level.	3.4.1.2.2.2
56. The two 72-hour rated Class 1E division B and C batteries are located above the maximum flood height in the Auxiliary Building considering all possible flooding sources.	~~9A~~**3.4.1.2.2.2**

Table 19.59-29 (Sheet 22 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
57. Flood water ~~propagated from~~in the Turbine Building **drains to the yard and does not affect** ~~to~~ the Auxiliary Building ~~valve/piping penetration room at grade level is directed to drains~~. The presence of watertight walls and floor of the **Auxiliary Building** valve/penetration room prevents flooding from propagating beyond this area.	3.4.1.2.2.2
58. The mechanical equipment and electrical equipment in the Auxiliary Building are separated to prevent propagation of leaks from the piping and mechanical equipment areas to the Class 1E equipment and Class 1E I&C equipment rooms.	3.4.1.2.2.2
59. Connections to sources of "large" quantity of water are located in the Turbine Building. They are the service water system, which interfaces with the component cooling water system; and the circulating water system, which interfaces with the Turbine Building closed cooling system and the condenser. Features that minimize the flood propagation to other buildings are: - Flow from any postulated ruptures above grade level (elevation 100') in the Turbine Building flows down to grade level via floor grating and stairwells. This grating in the floors also prevents any significant propagation of water to the Auxiliary Building via flow under the doors. - A relief panel in the Turbine Building west wall at grade level directs the water outside the building to the yard and limits the maximum flood level in the Turbine Building to less than 6 inches. Flooding propagation to areas of the adjacent Auxiliary Building, via flow under doors or backflow through the drains, is possible but is bounded by a postulated break in those areas.	3.4.1.2.2.3
60. Flood water in the Annex Building grade level is directed by the sloped floor to drains and to the yard area through the door of the Annex Building. Flow from postulated ruptures above grade level in the Annex Building is directed by floor drains to the Annex Building sump, which discharges to the Turbine Building drain tank. Alternate paths include flow to the Turbine Building via flow under access doors and down to grade level via stairwells and elevator shaft. The floors of the Annex Building are sloped away from the access doors to the Auxiliary Building in the vicinity of the access doors to prevent migration of flood water to the non-RCAs of the Nuclear Island where all safety-related equipment~~, except for some containment isolation valves,~~ is located.	3.4.1.2.2.3

Table 19.59-29 (Sheet 23 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
61. There are no connections to sources of "unlimited" quantity of water, except for fire protection, in the Annex Building.	Figure 9.5.1-1
62. To prevent overdraining, the RCS hot and cold legs are vertically offset, which permits draining of the steam generators for nozzle dam insertion with a hot leg level much higher than traditional designs.	~~5.4.6~~7.2.1
To lower the RCS hot leg level at which a vortex occurs in the RNS suction line, a step nozzle connection between the RCS hot leg and the RNS suction line is used.	5.4.7.2.1 & Figure 5.1-5
Should vortexing occur, air entrainment into the RNS pump suction is limited.	5.4.7.2.1
There are two safety-related RCS hot leg level channels, one located in each hot leg. These level instruments are independent and do not share instrument lines. These level indicators are provided primarily to monitor RCS level during midloop operations. One level tap is at the bottom of the hot leg, and the other tap is on the top of the hot leg ~~as~~ close to the steam generator.	**Tier 1 Information** Figure 5.1-5 **19E 2.1.1**
Wide range pressurizer level indication (cold calibrated) is provided that can measure RCS level to the bottom of the hot legs. This nonsafety-related pressurizer level indication can be used as an alternative way of monitoring level and can be used to identify inconsistencies in the safety-related hot leg level instrumentation.	**Tier 1 Information** Figure 5.1-5 **19E 2.1.1**
The RNS pump suction line is sloped continuously upward from the pump to the reactor coolant system hot leg with no local high points. This design eliminates potential problems in refilling the pump suction line if an RNS pump is stopped when cavitating due to excessive air entrainment. This self-venting suction line allows the RNS pumps to be immediately restarted once an adequate level in the hot leg is re-established.	5.4.7.2.1
It is important to maximize the availability of the nonsafety-related wide range pressurizer level indication during RCS draining operations during cold shutdown. The Combined License applicant is responsible for developing procedures and training that encompass this item.	13.5
63. Solid-state switching devices and electro-mechanical relays resistant to relay chatter will be used in the ~~AP600~~ AP1000 safety-related I&C system.	19.55.2.3
64. The annulus drains will have the same or higher HCLPF value as the Shield Building so that the drain system will not fail at lower acceleration levels causing water blocking of the PCS air baffle.	19.59.10

Table 19.59-29 (Sheet 24 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
65. The ability to close containment hatches and penetrations during Modes 5 & 6 prior to steaming to containment is important. The COL is responsible for developing procedures and training that encompass this item.	13.5 & 16.1
66. Spurious actuation of squib valves is prevented by the use of a squib valve controller circuit which requires multiple hot shorts for actuation, physical separation of potential hot short locations (e.g., routing of ADS cables ~~and~~ in low voltage cable trays, and, in the case of PMS, the use of ~~redundant series controllers located in~~ **arm and fire signals from separate PMS cabinets**), and provisions for operator action to remove power from the fire zone.	9A.2.7.1
67. For long-term recirculation operation, the RNS pumps can take suction from one of the two sump recirculation lines. Unrestricted flow through both parallel paths is required for success of the sump recirculation function when both RNS pumps are running. If one of the two parallel paths fails to open, operator action is required to manually throttle the RNS discharge ~~MOV (V011)~~ valve to prevent pump cavitation.	Emergency Response Guidelines
The containment isolation valves in the RNS piping automatically close via PMS with a high radiation signal. The actuation setpoint was established consistent with a DBA non-mechanistic source term associated with a large LOCA. The containment radiation level for other accidents is expected to be below the point that would cause the RNS MOVs to automatically close.	~~7.3.1~~ **6.2.3 & 7.3.1.2.20**
With the RNS pumps aligned either to the IRWST or the containment sump, the pumps' net positive suction head is adequate to prevent pump cavitation and failure even when the IRWST or sump inventory is saturated.	5.4.7
Emergency response guidelines are provided for aligning the RNS from the control room for RCS injection and recirculation.	Emergency Response Guidelines
The following are additional ~~AP600~~ **AP1000** features which contribute to the low likelihood of interfacing system LOCAs between the RNS and the RCS: - A relief valve located in the common RNS discharge line outside containment provides protection against excess pressure. - Two remotely operated MOVs connecting the suction and discharge headers to the IRWST are interlocked with the isolation valves connecting the RNS pumps to the hot leg. This prevents inadvertent opening of these two MOVs when the RNS is aligned for shutdown cooling and potential diversion and draining of reactor coolant system. - Power to the four isolation MOVs connecting the RNS pumps to the RCS hot leg is administratively blocked at their motor control centers during normal power operation.	5.4.7.2

Table 19.59-29 (Sheet 25 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
67. (cont.) Per the Shutdown Evaluation ~~Report (WCAP-14837)~~, operability of the RNS is tested, via connections to the IRWST, before its alignment to the RCS hot leg for shutdown cooling.	~~Shutdown Evaluation Report~~ 19E
Inadvertent opening of RNS valve V024 results in a draindown of RCS inventory to the IRWST and requires gravity injection from the IRWST. The COL applicant is responsible for developing administrative controls to ensure that inadvertent opening of this valve is unlikely.	13.5
The reliability of the IRWST suction isolation valve (V023) to open on demand is important. The IRWST suction isolation valve is included in the ~~RAPD~~ RAP.	17.4
68. The startup feedwater system pumps provide feedwater to the steam generator. This capability provides an alternate core cooling mechanism to the PRHR heat exchangers for non-LOCA or steam generator tube ruptures. The startup feedwater pumps are included in the ~~RAPD~~ RAP.	17.4
69. Capability is provided for on-line testing and calibration of the DAS channels, including sensors.	7.7.1.11
Short-term availability controls of the DAS during at-power conditions reduce PRA uncertainties.	16.3
70. One CVS pump is configured to operate on demand while the other CVS pump is in standby. The operation of these pumps will alternate periodically.	9.3.6.3.1 & 19.15
The safety-related PMS boron dilution signal automatically re-aligns CVS pump suction to the boric acid tank. This signal also closes the two safety-related CVS demineralized water supply valves. This signal actuates on reactor trip signal (interlock P-4), source range flux doubling signal, or low input voltage to the Class 1E dc power system battery chargers.	7.3.1.2.14
71. The COL applicant will maintain procedures to respond to low hot leg level alarms.	Emergency Response Guidelines
~~72. A COL applicant cleanliness program controls foreign debris from being introduced into the IRWST tank and into the containment sump during maintenance and inspection operations.~~	~~6.3.2.2.7.2, 6.3.2.2.7.3, & 6.3.8.1~~
~~73. For floor drains, from the reactor cavity PXS-A and PXS-b rooms, appropriate precautions such as check valves, back-flow preventors, and siphon breaks are assumed to prevent back-flow from a flooded space to a nonflooded space.~~	~~3.4.1.2.2~~

Table 19.59-29 (Sheet 26 of 26)

~~AP600~~ AP1000 PRA-BASED INSIGHTS

INSIGHT	DISPOSITION
72 **The containment recirculation screens are configured such that the chance of clogging is minimized during operation following accidents at power and at shutdown. The configuration features that reduce the chance of clogging include:** - **Redundant screens are provided and located in separate locations** - **Bottom of screens are located well above the lowest containment level as well as the floors around them** - **Top of screens are located well below the containment floodup level** - **Screens have protective plates that are located close to the top of the screens and extend out in front and to the side of the screens** - **Screens have conservative flow areas to account for plugging. Adequate PXS performance can be supported by one screen with at least 90% of its surface area completely blocked** - **During recirculation operation, the velocities approaching the screens are very low which limits the transport of debris.**	**6.3.2**
73. A COL applicant cleanliness program controls foreign debris from being introduced into the IRWST tank and into the containment during maintenance and inspection operations.	**6.3.2.2.7.2, 6.3.2.2.7.3, & 6.3.8.1**
74. For floor drains, from the reactor cavity PXS-A and PXS-B rooms, appropriate precautions such as check valves, back flow preventers, and siphon breaks are assumed to prevent back flow from a flooded space to a nonflooded space.	**3.4.1.2.2**
~~74~~75. Plant ventilation systems include features to prevent smoke originating from one fire area to another to the extent that they could adversely affect safe shutdown capabilities.	9.4.2.2
~~75~~76. An alternative gravity injection path is provided through RNS V-023 during cold shutdown and refueling conditions with the RCS open.	Emergency Response Guidelines
The COL applicant is responsible for developing administrative controls to maximize the likelihood that RNS valve V-023 will be able to open if needed during Mode 5 when the RCS is open, and PRHR cannot be used for core cooling.	13.5
~~76~~77. The IRWST suction isolation valve (V023) and the RCS pressure boundary isolation valves (V001A/B, V002A/B) are environmentally qualified to perform their safety functions.	Tier 1 Information
~~77~~78. Following an extended loss of RNS during safe/cold shutdown with the RCS intact and PRHR unavailable, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation.	19.59.5