20 GENERIC ISSUES

In this chapter, the staff discusses its evaluation of (1) the compliance of the Westinghouse AP600 design with 10 CFR 52.47(a)(1)(iv) and 52.47(a)(1)(ii), and (2) the incorporation of operating experience into the AP600 design. The applicant for a standard design certification is required by 10 CFR 52.47(a)(1)(iv) to propose resolutions of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) defined in NUREG-0933, "A Prioritization of Generic Safety Issues," that are (1) technically relevant to the design and (2) identified in the applicable supplement to NUREG-0933. In addition, the applicant is required under 10 CFR 52.47(a)(1)(ii) to propose resolutions to the technically relevant portions of Three Mile Island (TMI) Action Plan items addressed in 10 CFR 50.34(f).

Because a large number of issues are relevant to the AP600 design, the staff grouped its evaluations into the following sections, according to the issue type in Appendix B of NUREG-0933:

- Section 20.2 contains the task action plan items.
- Section 20.3 contains the new generic issues.
- Section 20.4 contains the TMI Action Plan items.
- Section 20.5 contains the human factors issues.
- Section 20.6 lists the 50.34(f) TMI Action Plan items relevant to the AP600 design.
- Section 20.7 discusses the incorporation of operating experience into the AP600 design through generic communications.

20.1 Overview of Staff Conclusion

20.1.1 Compliance With 10 CFR 52.47(a)(1)(iv)

As stated above, an application for design certification must include proposed resolutions of those USIs and medium- and high-priority GSIs identified in the NUREG-0933 supplement that was current six months prior to the application, and which are technically relevant to the design.

By letter dated June 26, 1992, Westinghouse made its application for standard design certification (SDC) of the AP600 standardized plant design in accordance with the provisions of 10 CFR 52.45. The design is described in the AP600 Standard Safety Analysis Report (SSAR) from Westinghouse. Therefore, the applicable date for the appropriate supplement of NUREG-0933 for paragraph 52.47(a)(1)(iv) is six months prior to the June 1992 date

(i.e., December 1991) and the applicable supplement of NUREG-0933 is Supplement 14, dated December 1991.

The staff reviewed Supplement 14 to NUREG-0933 to identify the list of issues contained in Appendix B of NUREG-0933, "Applicability of NUREG-0933 Issues to Operating and Future Plants," that should be addressed to conform to Section 52.47(a)(1)(iv). In addition, the staff added nine other issues (A-17, A-29, B-5, 14, 22, 29, 43, 82, and II.K.3(5)) that were resolved without the issuance of new requirements, but for which the staff had recommended the development of specific guidance for future plants.

The issues that need to be resolved to comply with Section 52.47(a)(1)(iv) are evaluated in Sections 20.2 to 20.5 of this report. Additional issues that Westinghouse considers applicable to the AP600 design are included in Section 1.9.4 of the SSAR and were evaluated by the staff.

Westinghouse evaluated the issues in Supplement 14 to NUREG-0933 to determine which issues were technically relevant to the AP600 design. Their review updated the status of the items to the status in Supplement 17; however, items added between Supplements 14 and 17 were not reviewed.

The staff concludes that Westinghouse has adequately demonstrated compliance of the AP600 design with 10 CFR 52.47(a)(1)(iv) in that it has addressed the issues in the relevant supplement of NUREG-0933.

20.1.2 Compliance with 10 CFR 52.47(a)(1)(ii)

As stated above, 10 CFR 52.47(a)(1)(ii) requires a design certification applicant to demonstrate compliance with any technically relevant portions of the TMI Action Plan requirements in 10 CFR 50.34(f). Westinghouse addressed these requirements in Section 1.9.3 of the SSAR and these requirements are discussed in Section 20.6 of this report. Because of the overlap between these TMI Action Plan items and those from NUREG-0933 (discussed in Section 20.4 of this report), all the relevant 50.34(f) TMI Action Plan items are listed in Section 20.6 in tabular form. This provides the issue designation and a reference to the appropriate issue in Section 20.4 of this report which contains the evaluation of the 50.34(f) TMI Action Plan item.

The staff concludes that Westinghouse has adequately demonstrated compliance of the AP600 design with 10 CFR 52.47(a)(1)(ii) in that it has addressed the relevant TMI Action Plan items in 10 CFR 50.34(f).

20.1.3 Incorporation of Operating Experience

In a staff requirements memorandum (SRM) from the Commission, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission directed the staff to ensure that the design certification process preserves operating experience insights in the certified design. Westinghouse submitted its evaluation for the AP600 design in the topical report WCAP-13559, "Operational Assessment for AP600," Revision 2, dated March 1998. As discussed in Section 20.7 of this report, the staff concludes that Westinghouse has adequately considered operating experience in that it has addressed generic letters and bulletins issued by the Commission between January 1, 1980, and December 31, 1997, in the AP600 design.

20.1.4 Resolution of Issues Relevant to the AP600 Design

In Table 1.9-2 in the draft Section 1.9.4 of the SSAR in the letter dated May 28, 1993, Westinghouse listed the issues in Supplement 14 of NUREG-0933 that it considered relevant to the AP600 design. The section also provides Westinghouse's justification for considering an issue not relevant to the design. The resolution of the issues that Westinghouse and the staff considered relevant to the design are discussed in Sections 20.2 through 20.6 of this report. This SSAR section was considered a draft because it had not been incorporated into the SSAR. The incorporation of this section into the SSAR was Open Item 20.1-1. Westinghouse subsequently incorporated Table 1.9-2 into the SSAR; therefore, Open Item 20.1-1 is closed.

In Table 20.1-1, the staff lists the USIs and GSIs relevant to the AP600 design, the sections in which these issues appear in this chapter, and the basis for the relevancy of each issue to the design. The relevancy of the issues fall into one of the following categories:

- The issue is required by 10 CFR 52.47(a)(1)(ii) or (iv) (i.e., 52.47).
- The issue was selected by Westinghouse as being relevant in Section 1.9.4 of the SSAR (i.e., W).
- The staff decided to discuss the issue (i.e., staff).

In the last case, as discussed above, the staff added nine other issues (A-17, A-29, B-5, 14, 23, 29, 43, 82, and II.K.3(5)) that were resolved without the issuance of new requirements, but for which the staff had recommended the development of specific guidance for future plants. These issues and the staff evaluations are arranged in Table 20.1-1 in the order in which they appear in Sections 20.2 through 20.5 of this report. Westinghouse did not consider Issues A-17, A-29, B-5, and 82 relevant to the AP600 design and did not provide an evaluation in the draft Section 1.9.4 of the SSAR. The staff's request that Westinghouse address these issues for the AP600 design was Open Item 20.1-2.

Westinghouse subsequently provided acceptable resolutions for these issues in the SSAR. The staff's evaluation of Westinghouse's resolutions appear in Sections 20.2 and 20.3 of this report. Therefore, Open Item 20.1-2 is closed.

Westinghouse provided its justifications for considering an issue not relevant to the AP600 design in Table 1.9-2 of the draft Section 1.9.4 of the SSAR. The staff reviewed these justifications for those issues which the staff considered relevant to the design to meet 10 CFR 52.47(a)(1)(iv). The justifications for the following issues not being relevant to the AP600 design were not considered adequate, and the staff asked Westinghouse to address certain design issues, including issues 24, 67.3.3, 73, 75, 120, 143, 153, I.G.2, II.E.1.3, II.E.6.1, II.J.4.1, II.K.1(5), II.K.1(10), II.K.1(13), II.K.1(17), III.A.3.3, and HF4.4. This was Open Item 20.1-3.

Westinghouse subsequently provided acceptable resolutions for these issues in the SSAR. The staff's evaluation of Westinghouse's resolutions appear in Sections 20.3, 20.4, and 20.5 of this report. Therefore, Open Item 20.1-3 is closed.

For some TMI Action Plan issues, Westinghouse has indicated in Table 1.9-2 that the issue is not applicable to the AP600 design, but has included a review of the TMI issue for the design in Section 1.9.3 of the SSAR (e.g., Issue II.K.2(10)).

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 14) relevant to the AP600 Design

Issue	Title of Issue and Section of this report	Relevancy
	Section 20.2, Task Action Plan Items	
A-1	Water Hammer	52.47/W
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	52.47/W
A-3	Westinghouse Steam Generator Tube Integrity	52.47/W
A-9	Anticipated Transient Without Scram (ATWS)	52.47/W
A-11	Reactor Vessel Materials Toughness	W
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Supports	52.47/W
A-13	Snubber Operability Assurance	52.47/W
A-17	Systems Interactions in Nuclear Power Plants	Staff
A-24	Qualification of Class 1E Safety-Related Equipment	52.47/W
A-25	Non-Safety Loads on Class 1E Safety-Related Equipment	52.47/W
A-26	Reactor Vessel Pressure Transient Protection	52.47/W
A-28	Increase in Spent Fuel Storage Capacity	W
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Sabotage	Staff
A-31	RHR Shutdown Requirements	52.47/W
A-33	NEPA Review of Accident Risks	W
A-35	Adequacy of Offsite Power Systems	52.47/W
A-36	Control of Heavy Loads Near Spent Fuel	52.47/W
A-40	Seismic Design Criteria Short-term Program	52.47/W
A-43	Containment Emergency Sump Performance	52.47/W
A-44	Station Blackout	52.47/W
A-46	Seismic Qualification of Equipment in Operating Plants	W
A-47	Safety Implications of Control Systems	52.47/W
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	52.47/W
A-49	Pressurized Thermal Shock	52.47/W
B-5	Ductibility of Two-Way Slabs and Shells, and Buckling Behavior of Steel Containments	Staff
B-17	Criteria for Safety-Related Operator Actions	52.47/W
B-22	LWR Fuel	W
B-29	Effectiveness of Ultimate Heat Sinks	W
B-32	Ice Effects on Safety-Related Water supplies	W
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for ESF Systems and Normal Ventilation Systems	52.47/W
B-53	Load Break Switch	W
B-56	Diesel Reliability	52.47/W
B-61	Allowable ECCS Equipment Outage Periods	52.47/W
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	52.47/W
B-66	Control Room Infiltration Measurements	52.47/W
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	52.47/W
C-4	Statistical Methods for ECCS Analysis	W
C-5	Decay Heat Update	W
C-6	LOCA Heat Sources	W
C-10	Effective Operation of Containment Sprays in a LOCA	52.47/W
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	52.47/W

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 14) relevant to the AP600 Design (Continued)

Issue	Title of Issue and Section of Chapter	Relevancy
	Section 20.3, New Generic Issues	
14	PWR Pipe Cracks	Staff
15	Radiation Effects on Reactor Vessel Supports	52.47/W
22	Inadvertent Boron Dilution Events	Staff
23	Reactor Coolant Pump Seal Failures	52.47/W
24	Automatic ECCS Switchover to Recirculation	52.47
29	Bolting Degradation or Failure in Nuclear Power Plants	52.47/W
43	Reliability of Air Systems	Staff
45	Inoperability of Instrumentation Due to Extreme Cold Weather	52.47/W
51	Improving the Reliability of Open-Cycle Service Water Systems	52.47/W
57	Effects of Fire Protection Systems Actuation on Safety-Related Equipment	52.47/W
67.3.3	Improved Accident Monitoring	52.47
70	PORV and Block Valve Reliability	52.47/W
73	Detached Thermal Sleeves	52.47
75	Generic Implications of ATWS Events at Salem Nuclear Plant	52.47
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	W
82	Beyond-Design-Basis Accidents in Spent Fuel Pools	Staff
83	Control Room Habitability	52.47/W
87	Failure of HPCI Steamline Without Isolation	52.47/W
93	Steam Binding of Auxiliary Feedwater Pumps	52.47/W
94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors	52.47/W
103	Design for Probable Maximum Precipitation	52.47/W
105	Interfacing System LOCA at LWRs	W
106	Piping and Use of Combustible Gases in Vital Areas	52.47/W
113	Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers	52.47/W
120	On-Line Testability of Protection Systems	52.47
121	Hydrogen Control for Large, Dry PWR Containments	52.47/W
122.2	Initiating Feed and Bleed	Staff
124	Auxiliary Feedwater Reliability	52.47/W
125.II.7	Reevaluation Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	Staff
128	Electric Power System Reliability	52.47/W
130	Essential Service Water Pump Failures at Multi-plant Sites	52.47/W
135	Steam Generator and Steamline Overfill	W
142	Leakage Through Electrical Isolators in Instrumentation Circuits	52.47/W
143	Availability of Chilled Water Systems and Room Cooling	52.47
153	Loss of Essential Service Water in LWRs	52.47
	Section 20.4, Three Mile Island Action Plan Items	
I.A.1.4	Long-Term Upgrade of Operating Personnel and Staffing	52.47
I.A.2.6(1)	Revise Regulatory Guide 1.8	52.47
I.A.4.1(2)	Interim Changes in Training Simulators	52.47
I.A.4.2	Long-Term Training Simulator Upgrade	52.47/W

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 14) relevant to the AP600 Design (Continued)

Issue	Title of Issue and Section of this report	Relevancy
I.C.1	Guidance for Evaluation and Development of Procedures for Transients and Accidents	Staff
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	52.47/W
I.C.9	Long-Term Program for Upgrading Procedures	52.47/W
I.D.1	Control Room Design Reviews	52.47/W
I.D.2	Plant Safety Parameter Display Console	52.47/W
I.D.3	Safety System Status Monitoring	52.47/W
I.D.5(2)	Control Room Design: Improved Instrumentation Research - Plant Status and Postaccident Monitoring	52.47/W
I.D.5(3)	Control Room Design: On-Line Reactor Surveillance Systems	52.47/W
I.F.1	Expanded Quality Assurance	52.47
I.F.2	Development of More Detailed QA Criteria	52.47/W
I.G.2	Scope of Test Program	52.47
II.B.1	Reactor Coolant System Vents	52.47/W
II.B.2	Plant Shielding to Provide Postaccident Access to Vital Areas	52.47/W
II.B.3	Postaccident Sampling Capability	52.47/W
II.B.8	Rulemaking Proceedings on Degraded Core Accidents Description	52.47/W
II.D.1	Performance Testing of PWR Safety and Relief Valves	52.47/W
II.D.3	Coolant System Valves: Valve Position Indication	52.47/W
II.E.1.1	Auxiliary Feedwater System Evaluation	52.47/W
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	52.47/W
II.E.1.3	Update Standard Review Plan and Development of Regulatory Guides	52.47
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Staff
II.E.3.1	Pressurizer Heater Power Supply	52.47/W
II.E.4.1	Dedicated Hydrogen Penetrations	52.47/W
II.E.4.2	Containment Isolation Dependability	52.47/W
II.E.4.4	Purging	52.47/W
II.E.5.1	Design Evaluation	W
II.E.6.1	In Situ Valve Testing, Test Adequacy Study	52.47
II.F.1	Additional Accident Monitoring Instrumentation	52.47/W
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	52.47/W
II.F.3	Instrumentation for Monitoring Accident Conditions	52.47/W
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	52.47/W
II.J.3.1	Organization and Staffing to Oversee Design and Construction	52.47/W
II.J.4.1	Revise Deficiency Reporting Requirements	52.47
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Staff
II.K.1(4d)	Review Operating Procedures and Training to Ensure that Operators Are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions	Staff
II.K.1(5)	Safety-Related Valve Position Description	52.47
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	52.47
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementing of all Bulletin Items	52.47
II.K.1(16)	Implement Procedures that Identify PZR PORV "Open" Indications and that Direct Operator to Close Manually at "Reset" Setpoint	Staff
II.K.1(17)	Trip Pressurizer Level Bistable so that Pressurizer Low Pressure Will Initiate Safety Injection	52.47

Table 20.1-1 USIs/GSIs in NUREG-0933 (Supplement 14) relevant to the AP600 Design (Continued)

Issue	Title and Section of this report	Relevancy
	Section 20.4, Three Mile Island Action Plan items	
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal System When Feedwater System Not Operable	W
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	52.47
II.K.1(25)	Develop Operator Action Guidelines	52.47
II.K.1(26)	Revise Emergency Procedures and Train Reactor Operators and Senior Reactor Operators	52.47
II.K.1(27)	Provide Analysis and Develop Guidelines and Procedures for Inadequate Core Cooling	52.47
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	52.47
II.K.2(10)	Hard-Wired Safety Grade Anticipatory Reactor Trip	W
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	W
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	W
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	52.47/W
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps During LOCA	Staff
II.K.3(6)	Instrumentation to Verify Natural Circulation	Staff
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of Steam Generators	Staff
II.K.3(9)	Proportional Integral Derivative Controller Modification	W
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for some Event Sequences	W
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	52.47/W
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	W
II.K.3(30)	Revised SBLOCA Methods to Show Compliance with 10 CFR Part 50, Appendix K	Staff
III.A.1.2	Upgrade Licensee Emergency Support Facilities	52.47/W
III.A.3.3	Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Communication Systems	52.47
III.D.1.1	Primary Coolant Sources Outside the Containment	52.47/W
III.D.3.3	In-Plant Radiation Monitoring	52.47/W
III.D.3.4	Control Room Habitability	52.47/W
	Section 20.5, Human Factors Issues	
HF1.1	Shift Staffing	52.47
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	W
HF4.4	Guidelines for Upgrading Other Procedures	52.47
HF5.1	Local Control Station	52.47/W
HF5.2	Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation	52.47/W

NOTES:

* 52.47: The resolution of the issue is required by 10 CFR 52.47(a)(1)(ii) and (iv).
W: Westinghouse submitted an evaluation.
Staff: The staff provided a resolution for the issue although Westinghouse did not provide an evaluation.

20.2 Task Action Plan Items

The task action plan items listed in Table 20.1-1 are evaluated against the AP600 design in this section. The majority of the items were chosen either because (1) 10 CFR 52.47(a)(1)(iv) or 10 CFR 50.34(f) require the design to comply with them, or (2) Westinghouse decided that the

item applied to the design and included a discussion of the item in the SSAR. In addition, the staff requested, and Westinghouse agreed, to address Issues A-17, A-29, and B-5 for the AP600 design.

The reference to SRP sections in this chapter is a reference to sections in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1982. The reference to the General Design Criteria (GDC) is a reference to the GDC in Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Issue A-1: Water Hammer

As discussed in NUREG-0933, Issue A-1 addresses the issue of water hammer in fluid systems in nuclear power plants. Water hammer can be caused by a number of conditions, such as voiding in normally filled lines, condensation in lines, entrainment of water in steam-filled lines, or rapid valve actuation. Issue A-1 addresses these probable causes, as well as possible methods for minimizing the susceptibility of systems to water hammer through design and operational considerations. This issue was resolved with the publication of NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," Revision 1, dated March 1984, which contained evaluation results of water hammer events, as well as details of recommendations and measures for water hammer prevention and mitigation.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design meets the guidance of applicable SRP sections in NUREG-0800 that provide criteria for mitigation of water hammer concerns and NUREG-0927, and addressed design features and system operation that mitigate or prevent water hammer damage. Westinghouse stated that design features are incorporated in the applicable systems, including the steam generator feedrings and piping, passive core cooling system, passive residual heat removal system, service water system, feedwater system, and steamlines. These features are summarized below.

The automatic depressurization system uses multiple, sequenced valve stages to provide a relatively slow, controlled depressurization of the reactor coolant system, which helps reduce the potential for water hammer. Once depressurization is complete, gravity injection from the refueling water storage tank is initiated by opening check valves, which reposition slowly. Gravity injection flow actuates slowly, without water hammer, as the pressure differential across the check valves equalizes, and the valves open and initiate flow.

The passive residual heat removal system exchangers are normally aligned with open inlet valves and closed discharge valves. This keeps the system piping at reactor coolant system pressure and prevents water hammer upon initiation of flow through the heat exchangers.

The core makeup tanks are normally aligned to the cold leg to keep the tanks at reactor coolant system pressure. The line is also normally kept filled with steam to prevent water hammer upon actuation of the core makeup tank. Section 6.3 of the SSAR provides additional information on the passive core cooling system.

The potential for water hammer in the feedwater line is minimized by the design and operation of the feedwater delivery system. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundles and below the

normal water level by a top discharge spray tube feedring. The layout of the feedwater line is consistent with industry standard recommendations to reduce the potential of a steam generator water hammer. In addition, operational limitations on flow to recover steam generator levels and on early feedwater flow into the steam generator minimize the potential for water hammer.

The startup feedwater system is a non-safety-related system that provides heated feedwater during plant startup, shutdown, and hot standby. The heated feedwater reduces the potential for water hammer in the feedwater piping and steam generator feedrings.

The main steamlines are designed to remove accumulated condensate from the main steamlines and to maintain the turbine bypass header at operating temperature during plant operation. The system is designed to accommodate flows during startup, shutdown, transients, and normal operation. This is to protect the turbine and turbine bypass valves from water slug damage.

DSER Open Item 20.2-1 resulted from the staff's preliminary review of Issue A-1. Subsequent staff evaluations of this issue have concluded that the above discussions, supplemented by the various measures to minimize the potential of water hammer described in SSAR Sections 1.9.4.2.2, 3B.2.3, 5.4.6, 6.3.2.5, 5.4.2.2, 5.4.7.2, 9.2.1.2.2, 10.4.7, and 14; and in the Topical Report WCAP-13054, "AP600 Compliance with the SRP Acceptance Criteria," provide acceptable commitments for the AP600 design to meet water hammer-related guidelines in applicable sections of the SRP and NUREG-0927. Therefore, Open Item 20.2-1 is closed.

DSER Open Item 20.2-2 identified staff questions relative to the results from a small-break loss-of-coolant accident (SBLOCA) test performed at Oregon State University (Reference Section 21.5.5 of this report). This open item has been subsumed by DSER Open Item 3.6.3.6-6. Although these results indicated that rapid condensation events have the potential to cause unanticipated dynamic loads in the reactor coolant system, the staff's evaluation found that the loads so induced are small and inconsequential to components and piping integrity (See Section 3.6.3.6 of this report). Therefore, DSER Open Item 20.2-2 is also closed, and Issue A-1 is resolved for the AP600 design.

Issue A-2: Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

As discussed in NUREG-0933, Issue A-2 addresses the concerns raised in 1975 by Virginia Electric Power Company that an asymmetric loading on the reactor vessel supports resulting from a pipe break at the vessel nozzle had not been considered by the utility or Westinghouse in the original design of the reactor vessel support system for North Anna Units 1 and 2. In the postulated event at the vessel nozzle, asymmetric loss-of-coolant accident (LOCA) loads could result from forces induced on the reactor internals by transient differential pressures across the core barrel, and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the more detailed analytical models, it became apparent to Westinghouse that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports.

The issue was resolved with the publication of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981. The asymmetric loads on the reactor vessel, internals,

primary coolant loop, and components should not exceed the limits imposed by the applicable codes and standards. The staff also issued Generic Letter (GL) 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," on February 1, 1984, to permit the application of leak-before-break (LBB) technology to eliminate the postulated pipe rupture from the design basis. Subsequently, the staff revised GDC 4 to permit the application of LBB.

In SSAR Section 1.9.4.2.2, Westinghouse states that the use of mechanistic pipe break (or LBB) criteria permits the elimination of the evaluation of dynamic effects of pipe breaks in the analysis of structures, systems, and components (SSCs). GDC 4 allows the use of LBB to eliminate from the design basis the dynamic effects of pipe ruptures postulated at locations defined in Section 3.6.2 of the SSAR. The dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, and traveling pressure waves from the depressurization of the system. The AP600 main reactor coolant loops are designed in accordance with LBB criteria. This is described in Section 3.6.3 and Appendix 3B of the SSAR. In the DSER, the staff stated that it could not accept LBB for the AP600 until the DSER Open Items in Section 3.6.3 of the DSER are resolved. This was DSER Open Item20.2-3.

Subsequent staff evaluations have resolved all of the Section 3.6.3 open items, and LBB has been accepted for certain AP600 systems, including the reactor coolant system. The staff's evaluations for LBB are discussed in Section 3.6.3 of this report. Thus the concern of Issue A-2 regarding asymmetric blowdown loads on reactor vessel, vessel supports, and across the core barrel, resulting from postulating a pipe break at the vessel nozzle is not applicable to the AP600. In addition, the COL applicant will verify that the actual material properties and final, as-built, piping analysis meet the acceptance parameters in the bounding analysis as committed in Section 3.6.4 of the SSAR. On the basis of this evaluation, the staff concludes that Open Item 20.2-3 is closed and Issue A-2 is resolved for the AP600 design.

Issue A-3: Westinghouse Steam Generator Tube Integrity

As discussed in NUREG-0933, Issue A-3 addresses staff concerns related to steam generator (SG) tube degradation. These concerns stemmed from the fact that the SG tubes are a part of the reactor coolant system (RCS) boundary, and that tube ruptures allow primary coolant into the secondary system where its isolation from the environment is not fully ensured. In 1978, Issues A-3, A-4, and A-5 were established to evaluate the safety significance of tube degradation in Westinghouse, Asea Brown Baveri-Combustion Engineering (ABB-CE), and Babcock and Wilcox (B&W) SGs, respectively. These studies were later combined into one effort because of the similarity of many problems among the pressurized-water reactor (PWR) vendors.

This issue was resolved and no new requirements were established (U.S. NRC, SECY-88-272, "Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 27, 1988; and NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 1988). However, the staff issued GL 85-02, "Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity," dated April 17, 1985, to provide

recommended actions from NUREG-0844. After reviewing responses to GL 85-02, the staff concluded that the large majority of licensees and applicants are following programs, practices, and procedures that are partially to fully consistent with, or equivalent to, the recommendations discussed in GL 85-02.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 steam generators are designed in accordance with GL 85-02 and NUREG-0844. The steam generators have features described in Section 5.4.2 of the SSAR to enhance tube performance and reliability. These features include the following:

- The design provides access to all tubes to perform inservice inspection.
- The tubes are fabricated from thermally treated nickel-chromium-iron Alloy 690.
- Support to the tubes is provided by ferritic stainless steel support plates.
- Contact between tubes and support plates is by the trifoil tube hole design, which provides a high sweeping velocity to reduce sludge accumulation in crevices.
- The portion of the tube within the tubesheet is fully expanded to close the crevices between the tube and tubesheet.
- The tube design provides for easy replacement of the tube bundle, if this is required.

The AP600 design was reviewed against the potential design features to reduce containment bypass leakage as specified in SECY-93-087 ("Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993), which are addressed in Section 15.6.3 of this report.

As discussed in Sections 5.2.4 and 5.4.2 of this report, the development of the SG tube preservice inspection (PSI) and inservice inspection (ISI) programs is the responsibility of the COL applicant. Steam generator tube integrity is verified in accordance with this surveillance program. The programs are plant specific and will be reviewed by the staff individually for each license application referencing the AP600 design certification, against the staff's regulatory criteria in place at the time of its review. In the DSER, Westinghouse was requested to address the role of the COL applicant in its discussion of the resolution of this issue for the AP600 design. This was Open Item 20.2-4. Westinghouse resolved all steam generator issues identified in Sections 5.2.3 and 5.4.2 of this report, and thus the role of the COL applicant concerning PSI and ISI has been addressed. Open Item 20.2-4 is closed.

The staff concludes that Issue A-3 is resolved for the AP600 design.

Issue A-9: Anticipated Transient Without Scram

As discussed in NUREG-0933, Issue A-9 addresses the issue of ensuring that the reactor can attain safe shutdown after incurring an anticipated transient with a failure of the reactor trip system (RTS). An anticipated transient without scram (ATWS) is an expected operational

occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power (LOOP) to the reactor) that is accompanied by a failure of the RTS to shut down the reactor.

The acceptance criteria for the resolution of Issue A-9 are as follows:

- Compliance with the mitigation requirement of 10 CFR 50.62(c)(1) that plant equipment must automatically initiate emergency feedwater (EFW) and turbine trip under conditions indicative of an ATWS. This equipment must function reliably and must be diverse and independent from the RTS.

- Compliance with the prevention requirement of 10 CFR 50.62(c)(2) that the plant must have a scram system that is diverse and independent from the existing RTS.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design complies with the requirements in 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," and includes a discussion of the design features to address the probability of an ATWS in Sections 1.9.5 and 7.7 of the SSAR.

Westinghouse indicates that the AP600 design complies with the requirements of 10 CFR 50.62 with a diverse actuation system that includes the AMSAC (ATWS mitigation system actuation circuitry) protection features mandated by 10 CFR 50.62 by tripping the turbine and diversely actuating selected engineered safeguards functions.

There are other AP600 design features aimed at minimizing the probability of ATWS occurrence and mitigating the consequences, as discussed in Section 1.9.5 of the SSAR. For the AP600 design with passive core cooling systems, the staff requires that an ATWS analysis be performed to demonstrate that its ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plant designs. Westinghouse provided, in response to request for additional information (RAI) 440.26, the analysis of a complete loss of normal feedwater without reactor trip using the LOFTRAN code.

The detailed discussion of this issue is found in Section 15.2.7 of this report. The staff reviewed the AP600 design and concluded that it meets the intent of 10 CFR 50.62 requirements. The staff, therefore, concludes that Issue A-9 is resolved for the AP600 design.

Issue A-11: Reactor Vessel Material Toughness

In SSAR Table 1.9-2, Westinghouse identifies that it considers Issue A-11 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue A-11 addresses the NRC concern that, because of the remote possibility of failure of nuclear reactor pressure vessels designed to the ASME Code, the design of nuclear facilities must provide protection against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. As plants accumulate more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This issue was resolved with the publication of NUREG-0744,

"Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Revision 1, October 1982 and GL 82-26, "NUREG-0744, Revision 1, Pressure Vessel Material Fracture Toughness," November 12, 1982.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 reactor vessel design complies with the requirements of Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50 and includes features to reduce neutron fluence, enhance material toughness at low temperature, and eliminate weld seams in critical areas. Material requirements are discussed in Sections 5.3.1 and 5.3.2 of the SSAR, and pressure and temperature limits are provided in Section 5.3.3 of the SSAR.

The AP600 reactor vessel design complies with the requirements of 10 CFR Part 50, Appendix G, and includes various features for the vessel to reduce neutron fluence, enhance material toughness at low temperatures, and eliminate weld seams in critical areas. The staff evaluation of the vessel material properties and fracture toughness is provided in Sections 5.3.1, 5.3.2, and 5.3.3 of this report.

The staff concludes that Issue A-11 is resolved for the AP600 design.

Issue A-12: Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

During the course of the licensing action for North Anna, Units 1 and 2, a number of questions were raised about the potential for lamellar tearing and low-fracture toughness of the SG and reactor coolant pump (RCP) support materials for these facilities. Concerns regarding the supports at North Anna were applicable to all PWRs. This was designated as Issue A-12 in NUREG-0933.

This issue was resolved and no new requirements were established (U.S. NRC, NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," Revision 1, October 1983). However, the staff recommended developing guidance for new plants on the basis of the fracture toughness requirements of Subsection NF of Section III of the ASME Code.

Westinghouse describes the SG and RCP supports in Section 5.4.10 of the SSAR. The supports are designed and fabricated in accordance with Subsection NF of Section III of the ASME Code. Westinghouse states that Subsection NF requirements provide acceptable fracture toughness for the support materials.

The staff concludes that the Westinghouse response to Issue A-12 addresses the structural integrity of SG and RCP supports. Therefore, Issue A-12 is resolved for the AP600 design.

Issue A-13: Snubber Operability Assurance

Snubbers are primarily used as seismic and pipe whip restraints at nuclear power plants. They function as rigid supports for restraining the motion of attached systems or components under such rapidly applied load conditions as earthquakes, pipe breaks, and severe hydraulic transients, while allowing free expansion of the systems and components during various operating conditions. Issue A-13 in NUREG-0933 addressed the concern of a substantial

number of snubber malfunctions, the most frequent of which were (1) seal leakage in hydraulic snubbers and (2) high rejection rate during functional testing of snubbers. This issue has been resolved and new guidelines were established with the revision of SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," in 1981. To provide the staff's basis for resolving Issue A-13 for the AP600, information in the SSAR must be consistent with the guidelines in SRP 3.9.3 relative to snubber operability.

In the DSER, the staff reported that additional information was required in the SSAR to demonstrate conformance with SRP 3.9.3. This was identified as DSER Open Item 3.9.3.3-1. Subsequent to issuance of the DSER, Westinghouse revised SSAR Section 3.9.3.4.3 to provide acceptable information which closed DSER Open Item 3.9.3.3-1. The staff's evaluation of this issue is in Section 3.9.3.3 of this report. On the basis of this evaluation, the staff concluded that the guidelines in SRP 3.9.3 relative to snubber operability have been met, and that Issue A-13 is resolved for the AP600 design.

Issue A-17: Systems Interactions in Nuclear Power Plants

As discussed in NUREG-0933, Issue A-17 addressed concerns regarding adverse systems interactions (ASIs) in nuclear power plants. Depending on how they propagate, ASIs can be classified as functionally coupled, spatially coupled, and induced-human-intervention coupled. As discussed in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," dated August 1989, and GL 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989, Issue A-17 concerns ASIs caused by water intrusion, internal flooding, seismic events, and pipe ruptures.

A nuclear power plant comprises numerous structures, systems, and components (SSCs) that are designed, analyzed, and constructed using many different engineering disciplines. The degree of functional and physical integration of these SSCs into any single power plant may vary considerably. Concerns have been raised about the adequacy of this functional and physical integration and the coordination process. The Issue A-17 program was initiated to integrate the areas of systems interactions and consider viable alternatives for regulatory requirements to ensure that the ASIs have been or will be minimized in operating plants and new plants. Within the framework of the program, the staff requested, as stated in NUREG-0933, that plant designers consider the operating experience discussed in GL 89-18 and use the probabilistic risk assessment (PRA) required for future plants to identify the vulnerability and reduce ASIs.

This issue identified the need to investigate the potential that unrecognized subtle dependencies, or systems interactions, among SSCs in a plant could lead to safety significant events. In NUREG-1174, intersystem dependencies are categorized on the basis of the way they propagate into functionally-coupled, spatially-coupled, and induced human-intervention coupled systems interactions. The occurrence of an actual ASI or the existence of a potential ASI, as well as the potential overall safety impact, is a function of an individual plant's design and operational features. For the AP600 with new or differently configured passive and active systems, a systematic search for ASIs is necessary.

Westinghouse submitted WCAP-14477, Revision 0, "The AP600 Adverse System Interaction Evaluation Report," dated February 1996, for staff review and approval. The purpose of the report was to identify possible adverse interactions among safety-related systems and between

safety-related and non-safety-related systems, and to evaluate the potential consequences of such interactions. The staff reviewed WCAP-14477 and provided Westinghouse with comments and questions. Westinghouse subsequently addressed the staff's questions and comments and issued a revision to WCAP-14477. The staff reviewed this issue as part of the regulatory treatment of non-safety systems (RTNSS) and documented its review in Chapter 22 of this report.

The staff concludes that Westinghouse has adequately assessed possible ASIs and their potential consequences in WCAP-14477, Revision 1. In addition, the staff conducted confirmatory testing involving potential systems interactions, and performed analyses of selected accident scenarios in which non-safety and/or safety systems could interact. Both the confirmatory tests and analyses showed that potential systems interactions did not have significant adverse effects on overall safety performance. Additionally, no additional unanticipated ASIs were observed. Therefore, Open Items 20.2-5 and 20.2-6 are closed and Issue A-17 is resolved for the AP600 design.

Issue A-24: Qualification of Class 1E Safety-Related Equipment

Construction permit (CP) applicants for which safety evaluation reports (SERs) were issued after July 1, 1974, were required by the NRC to qualify all safety-related equipment to Institute of Electrical and Electronics Engineers (IEEE)-323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." From the time this standard was originated, the industry developed methods that were used to qualify equipment in accordance with the standard. Some of these methods had not been resolved to the satisfaction of the NRC. To assess the adequacy of the equipment qualification methods and acceptance criteria used by nuclear steam supply system (NSSS) and balance-of-plant (BOP) vendors, the NRC determined that a generic approach was required. This was designated as Issue A-24 in NUREG-0933 and was resolved with the publication of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated July 1981.

In Section 1.9.4 of the SSAR, Westinghouse states that the AP600 environmental qualification methodology described in Appendix 3D of the SSAR is founded on the generic Westinghouse qualification program approved by the NRC. Westinghouse also states that this methodology addresses the requirements of GDC 4 and 10 CFR 50.49, as well as the guidance of Regulatory Guide (RG) 1.89 and IEEE Standard 323-1974.

On the basis of the staff's review, which is discussed in Section 3.11 of this report, the staff concludes that Westinghouse's approach to environmental qualification of Class 1E equipment is in compliance with 10 CFR 50.49 and Issue A-24 is resolved for the AP600 design.

Issue A-25: Non-Safety Loads on Class 1E Power sources

As discussed in NUREG-0933, Issue A-25 addressed a review of whether non-safety-related loads should also be allowed to share Class 1E power sources. The Class 1E power sources provide the electric power for the plant systems that are essential to reactor shutdown, containment isolation, reactor core cooling, containment heat removal, and preventing significant release of radioactive material to the environment. As discussed in NUREG-0933,

this issue was resolved in Revision 2 to RG 1.75, "Physical Independence of Electric Systems," with minor exceptions (see SSAR Appendix 1A and IEEE 384-1974).

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design conforms with the criteria in RG 1.75. The safety-related power supply is the Class 1E dc and uninterruptable power supply (UPS) system, which supplies power to the ac inverters for the plant instrumentation and control systems. The system also provides power to dc loads associated with the four protection channels and the accident monitoring system. This is discussed in Section 8.3.2.1.1 of the SSAR.

The 125 Vdc emergency lighting in the main control room (MCR) and in the remote shutdown area is non-Class 1E and is fed from a Class 1E UPS through two series fuses that are coordinated for isolation. Present regulatory practice allows the connection of non-safety loads to Class 1E (emergency) power sources if it can be shown that the connection of non-safety loads will not result in degradation of the Class 1E system. In the AP600 design, either of these fuses is able to interrupt any fault current before initiation of a trip of any upstream fuse. No credible failure of non-Class 1E equipment or systems will degrade the Class 1E system below an acceptable level. Therefore, Issue A-25 is resolved for the AP600 design.

Issue A-26: Reactor Vessel Pressure Transient Protection

Since 1972, there have been many reported pressure transients that have exceeded the pressure-temperature limits specified in technical specifications (TS) for PWRs. The majority of these events occurred at relatively low reactor vessel temperatures, at which the material has less toughness and is more susceptible to failure through brittle fracture. This is Issue A-26 in NUREG-0933, which was resolved with the issuance of SRP Section 5.2.2, "Overpressure Protection." Applicants for construction permits and operating licenses were requested to design an overpressure protection system for light-water reactors (LWRs) following the guidance provided in SRP Section 5.2.2.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design conforms to the criteria in Branch Technical Position (BTP) RSB 5-2, "Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures," of SRP Section 5.2.2. The pressurizer is sized to accommodate most pressure transients, and overpressure protection for the RCS is provided by either the pressurizer safety valves or the normal residual heat removal relief valves, as described in Section 5.2.2 of the SSAR.

The staff concludes that the AP600 design satisfies the BTP RSB 5-2 requirements and, therefore, considers Issue A-26 resolved for the AP600 design.

Issue A-28: Increase in Spent Fuel Pool Storage Capacity

Westinghouse identifies, in SSAR Table 1.9-2, that it considers Issue A-28 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

Issue A-28 of NUREG-0933 addressed the development of consistent and formalized acceptance criteria regarding the conversion of existing spent fuel storage pools to higher density storage racks, to increase storage capacity. This issue was resolved with the NRC

letter to licensees on April 17, 1978, which provided in a single document the criteria used by the staff to evaluate applications for spent fuel pool storage modifications.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the AP600 design incorporates the NRC criteria and the heat load is evaluated for the stated spent fuel storage capacity.

The staff evaluated the conformance of the AP600 spent fuel pool design to the NRC criteria in Section 9.1.2 of this report and, on the basis of the staff's conclusions in this section, Issue A-28 is resolved for the AP600 design.

Issue A-29: Nuclear Power Plant Design for Reduction of Vulnerability to Sabotage

As discussed in NUREG-0933, Issue A-29 addressed alternatives to the basic design of nuclear power plants with the emphasis primarily on reducing the vulnerability of reactors to radiological sabotage. In the past, reduction in the vulnerability of reactors to such sabotage has been treated as a plant physical security function and not as a plant design requirement. This issue, however, is not required for the AP600 design to meet 52.47(a)(1)(ii) or (iv).

Extensive efforts and resources are expended in designing nuclear plants to minimize the risk to public health and safety from equipment or system malfunction; however, reduction of the vulnerability to industrial sabotage is treated as a plant security function and not as a plant design requirement. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage during the preliminary design phase. Because emphasis is being placed on standardizing plants, it is especially important to also consider measures to reduce vulnerabilities to sabotage in designing the plant.

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because the issue had no new requirements. The staff requested that Westinghouse address how the AP600 is designed to prevent or mitigate plant vulnerabilities to sabotage. This was Open Item 20.2-7.

Westinghouse added a discussion of this issue to Section 1.9.4.2.2 of the SSAR. They state that the protective strategy, as described in the AP600 Security Design Report and the AP600 Security Design Vulnerability Analysis Report, is sufficient to prevent or mitigate radiological sabotage. The staff agrees with this statement. Therefore, Open Item 20.2-7 is closed and Issue A-29 is resolved for the AP600 design.

Issue A-31: Residual Heat Removal Shutdown Requirements

As discussed in NUREG-0933, Issue A-31 addressed the ability to transfer heat from the reactor to the environment after shutdown, which is an important safety function. It was resolved in 1978 with the issuance of SRP Section 5.4.7, "Residual Heat Removal (RHR) System."

The safe shutdown of a nuclear power plant following an accident not related to a LOCA has typically been interpreted as achieving "hot-standby" condition. The NRC has placed considerable emphasis on the hot-standby condition of a power plant in the event of an accident or other abnormal occurrence and, similarly, on long-term cooling, which is typically

achieved by the RHR system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than the hot-standby-condition values. Even though it may generally be considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that certain events have occurred that required eventual cooldown or long-term cooling until the RCS is cold enough for personnel to inspect the problem and repair it.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design includes passive safety-related decay heat removal systems that establish and maintain the plant in a safe-shutdown condition following design-basis events, and it is not necessary that these passive systems achieve cold shutdown as defined in RG 1.139.

The passive core cooling system is designed to maintain plant safe-shutdown conditions indefinitely. Cold-shutdown condition is necessary only to gain access to the reactor coolant system for inspection, maintenance, or repair. For the AP600 design, cold-shutdown conditions can be achieved using highly reliable, but non-safety-related systems, which have similar redundancy as current generation safety-related systems and are supplied with ac power from either onsite or offsite sources. Passive core cooling capability is discussed in Section 6.3 of the SSAR.

Westinghouse states that the passive RHR system can achieve hot-standby conditions immediately and can reduce the reactor coolant temperature to 215.6 °C (420 °F) within 36 hours. The reactor pressure is controlled and can be reduced to 1.72 MPa (250 psig). The passive RHR system also provides a closed cooling system to maintain long-term cooling. Therefore, the AP600 complies with GDC 34 by using a more reliable and simplified system for both hot-standby and long-term cooling modes, and it is not necessary that these passive systems achieve cold shutdown as defined by RG 1.139.

In GDC 34, the NRC requires a residual heat removal system to be provided with suitable redundancy in components and features to assure that, with or without onsite or offsite power, it can accomplish its safety functions so that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. No definition is specified as the safe-shutdown condition for which the RHR system should accomplish this. The EPRI URD proposed that the safe-shutdown condition be defined as 215.6 °C (420 °F) for the passive ALWR designs. The staff concluded that cold-shutdown is not the only safe stable shutdown condition that can maintain the fuel and reactor pressure boundary within acceptable limits. In SECY-94-084, Section C, "Safe Shutdown Requirements," the staff recommended, and the Commission approved, that the EPRI-proposed 215.6 °C (420 °F) criteria or below, rather than the cold-shutdown condition required by RG 1.139, be accepted as a safe stable condition, which the passive RHR system must be capable of achieving and maintaining following non-LOCA events. This acceptance is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of RTNSS. The SECY paper also states that the passive safety system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including the following:

(1) A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition, that no transients will result in the specified acceptable fuel design limits and pressure boundary design limit being violated, and that

no high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria.

(2) A probabilistic reliability analysis, including events initiated from the safe-shutdown conditions, to ensure conformance with the safety goal guidelines. The PRA would also determine the reliability/availability missions of risk-significant systems and components as a part of the effort for RTNSS.

The staff discusses the performance of the passive system capability in Chapters 4 and 5 of this report and the RTNSS issue regarding the availability of the RNS system during shutdown and refueling conditions in Section 19.3. Therefore, the staff considers Issue A-31 resolved for the AP600 design.

Issue A-33: NEPA Review of Accident Risks

Westinghouse initially identified, in Table 1.9-2 of the SSAR, that it considered Issue A-33 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

Issue A-33, of NUREG-0933, addressed the determination that, consistent with NEPA, the environmental assessment of requests for construction permits and operating licenses should include consideration of the possible impacts from accidents. This issue was resolved by a Commission decision on SECY-80-131 to implement a policy in which accidents are considered on a risk-specific basis. The policy was published in the *Federal Register* on June 13, 1980 (45 FR 40101). The assessment of accidents in the environmental impact statements for a plant to be operated is different from the evaluation conducted by the staff on the plant design in the safety evaluation report.

In Section 1.9.4 of the SSAR, Westinghouse stated that the consideration of possible impacts from accidents on the environment is included in the AP600 PRA Report ("Simplified Passive Advanced Light Water Reactor Plant Program," AP600 Probabilistic Risk Assessment, Revision 1, July 22, 1994, proprietary and non-proprietary versions) for a generic site. A site-specific assessment would be performed by the COL applicant.

The staff does not agree that the PRA report addresses this issue. The staff asked Westinghouse to provide the appropriate PRA report sections on the environmental impact of accidents and discuss the results of the review for the generic site that apply to the resolution of this issue. The assessment of possible environmental impacts from accidents should also be stated in the resolution of this issue. This was Open Item 20.2-8.

In a subsequent revision to the SSAR, Westinghouse removed the discussion of Issue A-33 because it is not required to be addressed for the design. The staff agrees that the issue is not required. Therefore, Open Item 20.2-8 is closed.

A-35: Adequacy of Offsite Power Systems

In GDC 17 of Appendix A to 10 CFR 50, the NRC requires that an offsite electric power system be available to assure that (1) the fuel and reactor boundary are maintained within specified

acceptable limits, and (2) core cooling, containment integrity, and other vital safety functions are maintained during accident conditions.

The AP600 design includes an offsite power source; however, the AP600 design does not require any offsite ac power source to achieve and maintain safe shutdown and, therefore, this issue is not applicable to the AP600 design.

Therefore, Issue A-35 is not applicable and considered resolved for the AP600 design.

Issue A-36: Control of Heavy Loads Near Spent Fuel

At all nuclear plants, overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, radioactivity could be released to the environment. Such an occurrence would also have the potential for overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases to the environment could exceed the exposure guidelines of 10 CFR Part 100. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there was a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary. This was designated as Issue A-36 in NUREG-0933.

The issue was resolved with the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," dated July 1980, and SRP Section 9.1.5, "Overhead Heavy Load Handling Systems."

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the AP600 design conforms to NUREG-0612 and Section 9.1.5 of the SRP. The light-load handling systems are described in Section 9.1.4 of the SSAR and the overhead heavy-load handling systems are described in Section 9.1.5 of the SSAR.

The staff evaluated the conformance of the AP600 design to NUREG-0612 and Section 9.1.5 of the SRP in Sections 9.1.4 and 9.1.5 of this report and, on the basis of the staff's conclusions in these sections, Issue A-36 is resolved for the AP600 design.

Issue A-40: Seismic Design Criteria Short-Term Program

As discussed in NUREG-0933, Issue A-40 addressed short-term improvements in seismic design criteria. The objectives of Issue A-40 were the following:

- investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites
- investigate alternative approaches, where desirable
- quantify the overall conservatism of the design sequence
- modify the NRC criteria in the SRP, where justified

This issue was initiated in 1978 to identify and quantify conservatism in the seismic design process, and to develop a basis for revising SRP Section 3.7 on seismic design analyses.

To resolve this issue, the staff revised SRP Sections 2.5.2, "Vibratory Ground Motion," 3.7.1, "Seismic Design Parameters," 3.7.2, "Seismic System Analysis," and 3.7.3, "Seismic Subsystem Analysis," to address areas of vibratory ground motion; design time-history criteria; development of floor response criteria, damping values, and soil-structure interaction (SSI) uncertainties; and combination of modal responses. The revisions also addressed seismic analysis of the above-ground tanks and Category 1 buried piping. The revised SRP Section 3.7 provided guidelines for the (1) site-specific ground response spectra, (2) justification of the use of single synthetic ground motion time-history by power spectral density function, (3) basis for location and limitation of input ground motion reduction for SSI analysis, and (4) design of above-ground vertical tanks and buried piping.

The DSER stated that for sites having specific characteristics outside the range of the selected parameters, the COL applicant would demonstrate the acceptability of the AP600 design to the site-specific seismic characteristics. This was COL Action Item 20.2-1. In Revision 24 of SSAR Sections 2.5.2.2 and 2.5.4.6, Westinghouse committed that the COL applicants referencing the AP600 design will perform site-specific evaluation and demonstrate the acceptability of the AP600 design to the site-specific characteristics. On the basis of its evaluation discussed in Sections 2.5.2 and 2.5.4, the staff concludes the SSAR commitment is acceptable and COL Action Item 20.2-1 is resolved.

An acceptable resolution of Issue A-40 is that future nuclear power plants should conform to the seismic design guidance of Revision 2 to SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. The AP600 response to Issue A-40 in SSAR Section 1.9.4.2.2 references the criteria and methodology described in SSAR Section 3.7 as the basis for resolving this issue. The staff's review of Section 3.7 of the SSAR is discussed in Sections 3.7.1, 3.7.2, and 3.7.3 of this report. On the basis of its evaluations in these sections, the staff concludes that the AP600 design is consistent with the guidelines in Revision 2 of SRP Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. Therefore, Issue A-40 is resolved for the AP600 design.

Issue A-43: Containment Emergency Sump Performance

As discussed in NUREG-0933, Issue A-43 concerns the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump or BWR RHR system suction intake must be initiated and maintained to provide adequate core cooling. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired, thereby seriously degrading long-term recirculation flow capability.

On December 3, 1985, GL 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage" was issued. This generic letter informed licensees of operating reactors, applicants for operating licenses and holders of construction permits of the resolution of Issue A-43.

The staff's evaluation of this issue is in Section 6.2.1.8 of this report. On the basis of this evaluation, Issue A-43 is resolved for the AP600 design.

Issue A-44: Station Blackout

Generic Issue A-44 was resolved with the publication of 10 CFR 50.63, which provides requirements that LWRs be able to withstand for a specified duration and recover from a station blackout (SBO). It addresses the likelihood of the loss of all ac power at the site, and the potential for severe core damage after the SBO.

In SSAR Revision 9, Westinghouse stated that ac electrical power is not needed to establish or maintain a plant safe-shutdown condition for the AP600 design. But, the design includes two redundant, non-Class 1E diesel generators to provide electrical power for non-safety-related active systems that provide a defense-in-depth function. In its DSER, the staff tied the resolution of the SBO issue for the AP600 design with the process defined for resolving the RTNSS issue. The non-Class 1E diesel generators are identified as risk-significant in the scope of the design-reliability assurance program and are identified as RTNSS important. Westinghouse forwarded a copy of the markup of AP600 Reliability Assurance Program (RAP), SSAR Section 16.2, to the NRC on October 3, 1996. Table 16.2-1, "Risk Significant SSCs Under the Scope of D-RAP" lists non-Class 1E diesel generators as RTNSS important. The RTNSS issue is resolved in Section 8.6.2.4 of this report, therefore, Issue A-44 is resolved for the AP600 design.

Issue A-46: Seismic Qualification of Equipment in Operating Plants

Issue A-46, of NUREG-0933, addressed the need to establish an explicit set of guidelines to verify the seismic adequacy of mechanical and electrical equipment at operating plants instead of backfitting the current design criteria for new plants. Requirements for resolution of this issue were issued in GL 87-02, "Verification of Seismic Adequacy of Mechanical and electrical Equipment in Operating Plants, Unresolved Safety Issue (USI) A-46," on February 19, 1987.

The AP600 response to Issue A-46 in Section 1.9.4.2.2 of the SSAR states that this issue applies to operating plants, not to plants to be constructed, and as such does not apply to the AP600 design, which is designed in accordance with current seismic qualification (not verification) requirements. It also stated that the seismic Category 1 mechanical and electrical equipment in the AP600 design will be qualified in accordance with the AP600 qualification methodology discussed in Section 3.10 of the SSAR. The staff agrees that this issue is not applicable to the AP600. The staff's evaluation of seismic qualification of equipment for the AP600 design is in Section 3.10 of this report. On the basis of this evaluation, Issue A-46 is resolved for the AP600 design.

Issue A-47: Safety Implications of Control Systems

As discussed in NUREG-0933, Issue A-47 concerns the potential for accidents or transients becoming more severe as a result of control systems failures, including power supply faults. Within this issue, the staff performed an in-depth review of non-safety-related control systems and assessed the effect of control system failures on plant safety.

Non-safety-grade control systems are not relied on to perform any safety functions, but they are used to control plant processes that could have a significant impact on plant dynamics. For the resolution of Issue A-47, the NRC evaluated the effects of control system failures on PWR reference plants, including a design subjected to single and multiple control system failures during automatic and manual modes of operation. The staff's two concerns related to the design were: (1) SG overfill and (2) reactor core heat removal to cold shutdown after a small-break LOCA, without overcooling the reactor vessel. The NRC issued GL 89-19, "Request for Action Related to Resolution of USI A-47, Pursuant to 10 CFR 50.54(f)," dated September 20, 1989, which required all operating PWR plants and plants under construction to provide the following:

- automatic protection from SG overfill by the main feedwater system (MFWS) and separate from the MFWS control system

- plant procedures and TS surveillance requirements to periodically verify the operability of the overfill protection during power operation

The resolution of Issue A-47 is that the plant shall have, as a minimum, control-grade protection against SG overfill by the MFWS, and TS and plant operating procedures to ensure in-service verification of the availability of the overfill protection, in accordance with GL 89-19.

In SSAR Section 1.9.4.2.2, Westinghouse states that, for the AP600 design, control system failures are considered as potential initiating events. The analyses of transients resulting from these failures demonstrated that the consequences are bounded by American Nuclear Society (ANS) Condition II criteria and no design-basis failure for a control system is expected to violate this criteria.

The integrated control system for the AP600 design was stated to obtain certain of its control input signals from signals used in the integrated protection system. With the integrated control and protection system, functional independence of the control and protection systems is maintained by providing a signal selection device in the control system for those signals used in the protection system. The purpose of this device is to prevent a failed signal, caused by the failure of a protection channel, from resulting in a control action that could lead to a plant condition requiring that protective action. The signal selection device provides this capability by comparing the redundant signals and automatically eliminating an aberrant signal from being used in the control system. This capability exists for bypassed sensors or for sensors whose signals diverge from the expected error tolerance.

The AP600 plant control system is stated to incorporate design features as redundancy, automatic testing, and self-diagnostics to prevent challenges to the protection and safety monitoring systems. Chapter 7 provides a discussion of the AP600 instrumentation and controls.

In Sections 7.2.1.1.6, 7.3.1.2.6, 7.7.1.8 and Figure 7.2-1, sheet 10, of the SSAR, Westinghouse addresses SG overfill protection. Westinghouse states that SG overfill protection is provided by a safety-grade SG high-water-level signal with a two-out-of-four initiating logic. This signal closes the MFWS control valves and isolation valve. This is provided in the RTS logic, which is sufficiently separated from the MFWS control system. However, in DSER Open Item 20.2-9 the

staff requested that the applicability of the TS to the SG overfill protection design be addressed. The AP600 TS (Chapter 16 of the SSAR), TS 3.3.1, "Reactor Trip System Instrumentation," and TS 3.7.3, "Main Feedwater Isolation and Control Valves," provide requirements that adequately address the surveillance requirements to verify the operability of the SG overfill protection. Therefore, DSER Open Item 20.2-9 is resolved and Issue A-47 is resolved for the AP600 design.

Issue A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

In December 1984, the staff concluded that rulemaking with regard to hydrogen control for LWRs with large, dry containments could be safely deferred because of the inherent capability of these containments to accommodate large quantities of hydrogen. This concern is covered under Issue 121. In the staff's plans for resolving Issue 121, any recommendations for further modifications to 10 CFR 50.44 related to LWRs with large, dry containments were provided at the conclusion of ongoing research. In April 1989, SECY-89-122 was forwarded to the Commission documenting the results of the staff's efforts in resolving USI A-48. Thus, this issue was resolved and new requirements were established.

The Commission promulgated new requirements on hydrogen control in 10 CFR 50.34(f), which requires a hydrogen control system predicated on a 100-percent fuel-cladding metal-water reaction and a hydrogen concentration limit of 10 percent on uniformly distributed hydrogen in the containment, or a post-accident atmosphere that will not support hydrogen combustion. The ability of the AP600 design to comply with the hydrogen control requirements of 10 CFR 50.34(f) are documented in Section 6.2.4 of the SSAR. The staff's evaluation of the ability of the AP600 design to comply with the hydrogen control requirements of 10 CFR 50.34(f) are documented in Section 6.2.5, "Combustible Gas Control Inside Containment," of this report.

SECY-93-087 indicated that the staff would evaluate the ALWR vendor's identification of equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Global and local hydrogen deflagrations are two of the conditions that are to be considered. In SECY-93-087, the staff recommended that the Commission approve the staff's position that design features provided only for severe-accident mitigation need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. In the SRM dated July 21, 1993, the Commission approved the staff's position. Westinghouse addressed equipment survivability in Appendix D to the AP600 PRA. The staff's evaluation of Appendix D is documented in Section 19.2.3.3.7 of this report.

On the basis of the staff's evaluation, as documented in Sections 6.2.5 and 19.2.3.3.7 of this report, Issue A-48 is resolved for the AP600 design.

Issue A-49: Pressurized Thermal Shock

The issue of pressurized thermal shock arises in PWRs because unanticipated transients or design-basis postulated accidents could result in severe overcooling (thermal shock) of the reactor pressure vessel concurrent with or followed by repressurization. In these events, rapid cooling of the internal surfaces of the reactor vessel results in thermal stresses with a maximum

thermal tensile stress at the inside surface. The magnitude of the thermal stress depends on the temperature profile across the vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stress if the vessel is pressurized.

As discussed in NUREG-0933, Issue A-49 addressed the concern that neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that might threaten vessel integrity. The staff's concern is the possibility of vessel failure as a result of a severe pressurized overcooling event, or pressurized thermal shock (PTS). As long as the fracture toughness of the reactor vessel material is relatively high, such events are not expected to cause vessel failure. However, the fracture toughness decreases during the operating life of a nuclear power plant from the fast neutron flux. The rate of decrease is dependent on the chemical composition of the material and the amount of irradiation. If the fracture toughness has been reduced significantly, severe high pressure-low temperature events could cause propagation of small flaws that could exist near the inner surface of the vessel. The assumed initial flaw might propagate into a crack through the vessel wall to threaten vessel integrity and core cooling capability.

This issue was resolved and new requirements were established in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events". The rule establishes screening criteria that are related to the fracture toughness of the reactor vessel. The risk from pressure and temperature (P/T) events is acceptably low for reactor vessel materials that are projected to be below the PTS screening criteria.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 design complies with the requirements of 10 CFR 50.61. The material requirements and P/T limits for the AP600 are discussed in Section 5.3.2 of the SSAR.

The staff discussed this issue in Section 5.3.2 of this report and concluded that the reactor vessel beltline materials proposed for the AP600 design are projected to be below the screening criteria in 10 CFR 50.61. Compliance with this rule is an acceptable basis for resolving this issue. Therefore, Issue A-49 is resolved for the AP600 design.

Issue B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

In NUREG-0933, this issue was divided into the following two parts, which were evaluated separately:

Part I – Ductility of Two-Way Slabs and Shells

Part I of Issue B-5 was defined in NUREG-0471, "Generic Task Problem Descriptions," dated June 1978, and addressed the lack of information related to the behavior of two-way reinforced-concrete slabs loaded dynamically in biaxial tension, flexure, and shear. The objective was to develop design requirements for concrete two-way slabs to resist loading caused by a LOCA or high-energy line break (HELB). An acceptable resolution to this issue is

to apply the two-way reinforced-concrete slab analysis methods to adequately address dynamic loading in biaxial membrane tension, flexure, and shear due to a LOCA or HELB.

Part II – Buckling Behavior of Steel Containments

Part II of Issue B-5 was also identified in NUREG-0471 and addressed the lack of a well-defined approach for design evaluation of steel containment vessels subject to asymmetrical dynamic loadings that may be limited by the instability of the shell. An acceptable resolution to this issue is to address adequately the design loads, the asymmetrical vessel configurations associated with the presence of equipment hatches, and the factor of safety in determining allowable loadings.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because the issue had no new requirements. Although this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv), the staff requested that Westinghouse address how the AP600 is designed for the behavior of two-way reinforced-concrete slabs loaded dynamically in biaxial tension, flexure, and shear, and to prevent buckling of the steel containment. In Section 3.8.2 of the DSER, the staff also requested Westinghouse to consider the guideline issued by the staff and address its concerns about buckling of the steel containment. This was DSER Open Item 20.2-10.

With respect to Part I of this generic issue, Westinghouse stated in Revision 12 of SSAR Section 3.8.4.3.1.4 that pressure and thermal loads within or across a compartment (such as main steam isolation valve and steam generator blowdown compartments) are generated on the basis of postulated high-energy breaks. The SSAR also stated that, for structural elements including compartment walls and floor slabs, the analysis and design of concrete elements (reinforced concrete structural elements and steel structural modules) conform to American Concrete Institute (ACI) ACI-349 code. The use of ACI-349 code, which provides design criteria and design procedures for the design of reinforced concrete walls and floor slabs under bending and biaxial tension, is acceptable to the staff as discussed in Section 3.8.4 of this report (Open Item 3.8.4.2-2). On this basis, the concern of Part I is resolved.

As for Part II of this generic issue, Revision 11 of SSAR Section 3.8.2.4.1.1 states that the buckling evaluation under external pressure uses the criteria in NE-3133 of Section III of the American Society of Mechanical Engineers (ASME) Code. The potential buckling under overall seismic loads are evaluated in accordance with ASME Code, Case N-284, Revision 0 supplemented by Appendix 3G to the SSAR. The staff's evaluation and review conclusions for the containment shell buckling under various loads and combined load conditions are discussed in Section 3.8.2 of this report (Open Items 3.8.2.4-3, 3.8.2.4-4, 3.8.2.4-5, 3.8.2.4-9 and 3.8.2.4-14). On the basis of the discussion above, the concern of Part II is resolved, and DSER Open Item 20.2-10 is closed.

Therefore, Issue B-5 is resolved for the AP600 design.

Issue B-17: Criteria for Safety-Related Operator Actions

As discussed in NUREG-0933, Issue B-17 involves the development of a time criterion for safety-related operator actions (SROAs), including a determination of whether automatic

actuation is required. This issue also concerns PWR designs that require manual operations to accomplish the switchover from the injection mode to the recirculation mode following a LOCA.

Current plant designs are such that reliance on the operator to take action in response to certain transients is necessary. Consequently, it becomes necessary to develop appropriate criteria for SROAs. The criteria would include a determination of actions that should be automated in lieu of operator actions and development of a time criterion for SROAs.

The review criteria for this issue are contained in American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.8-1984, "Time Response Design Criteria for Nuclear Safety Related Operator Actions,". Plants should perform task analysis, simulator studies, and analysis and evaluation of operational data to assess engineered safety features (ESFs) and safety-related control system designs for conformance to the criteria. Where nonconformance is identified, modification of the design and hardware may be required.

In SSAR Section 1.9.4.2.2, Westinghouse states that, for the AP600 design, the safety-related actions required to protect the plant during design-basis events are automatically initiated. The plant systems are designed to provide the required information to the operator so that plant conditions can be monitored and the performance of the safety-related passive systems and the non-safety-related active systems can be evaluated. The non-safety-related active systems are stated to be designed to automatically actuate, provide defense-in-depth for plant events, and preclude unnecessary actuation of the safety-related passive systems. There is stated to be a backup manual initiation for both the passive and active systems.

Westinghouse further states that, as described in SSAR Chapter 15, the safety systems maintain the plant in a safe condition following design-basis events. This is discussed above in Issue A-31. For most design-basis events, this is accomplished without operator action for up to 72 hours. Operator action is stated to be planned and expected during plant events to achieve the most effective plant response consistent with the event conditions and equipment availability. For events where operator action is taken, the plant design maximizes the time available for operators to complete required actions. For example, Westinghouse states that, during a steam generator tube rupture, no operator action is required to establish safe-shutdown conditions or prevent steam generator overfill.

As shown in Section 18.3, "Element 2: Operating Experience Review," of the AP600 DSER, the staff did not complete its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, "Element 2: Operating Experience Review," of this report, Westinghouse, in WCAP-14645, "Human Factors Engineering Operating Experience Review Report for the AP600 Nuclear Power Plant", Revision 2, has satisfactorily addressed this item. Therefore, Issue B-17 is resolved for the AP600 design.

Issue B-22: LWR Fuel

Westinghouse identified in Table 1.9-2 of the SSAR that it considers Issue B-22 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-22 addressed the staff concerns that individual reactor fuel rods sometimes failed during normal operations and many fuel rods are expected to fail during severe core accidents. Failure of fuel rods results in radioactive releases within a plant and is a potential source of release to the public. The resolution of this issue was to ensure that these fuel failures did not result in unacceptable releases to the public. Several problems were identified in the staff analysis to improve the predictability of fuel performance and these were addressed in the revision to SRP Section 4.2, "Fuel System Design," in 1981. The staff concluded that the then-existing requirements on fuel were adequate to ensure continued low fuel defect rates and additional requirements would not significantly increase the number of fuel defects. This issue was then dropped from further consideration.

Westinghouse states that the AP600 reactor core design complies with SRP Section 4.2 and the discussion on the fuel system design is in Section 4.2 of the SSAR.

The staff completed its review of the VANTAGE-5H fuel for the AP600 design. The details of fuel design and acceptance criteria are discussed in Section 4.2 of this report. The staff concludes that Westinghouse has satisfactorily resolved all questions raised during the staff review of the issue, and therefore, the staff considers this issue resolved.

Issue B-29: Effectiveness of Ultimate Heat Sinks

Westinghouse identified in Table 1.9-2 of its SSAR that it considered Issue B-29 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-29 addressed the staff concerns identified in NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," June 1978, that the validity of the mathematical models used to predict the performance of dedicated ponds, spray ponds, and cooling towers had not been confirmed, and that better guidance was needed regarding the criteria for the selection of weather data to define the design-basis meteorology. The vulnerability and need for further improvement to the design and operation of ultimate heat sinks (UHS) are addressed in Section 20.3 of this report in Issues 51, 130, and 153. This issue regarded confirming the validity of the NRC mathematical models for prediction of UHS performance and providing guidance regarding the criteria for weather record selection to define UHS design-basis meteorology. This issue was resolved by studies completed by the staff, which confirmed the capabilities of NRC models and provided assurance that the existing guidance was adequate. No new requirements were issued. However, the adequacy of the models to simulate the performance of a plant-specific UHS must be justified on a case-by-case basis.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the passive containment cooling system for the AP600 design complies with SRP Section 9.2.5, "Ultimate Heat Sink," by providing passive decay heat removal that transfers heat to the atmosphere, which is the UHS for accident conditions. The passive containment cooling system is described in Section 6.2.2 of the SSAR.

The staff evaluated the conformance of the AP600 design to Section 9.2.5 of the SRP in Section 6.2.2 of this report and, on the basis of the staff's conclusions in this section, Issue B-29 is resolved for the AP600 design.

Issue B-32: Ice Effects on Safety-Related Water Supplies

Westinghouse identifies in Table 1.9-2 of the SSAR that it considers Issue B-32 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-32 addressed the staff concerns identified in NUREG-0471 that additional information was needed on the potential effects of extreme cold weather and ice buildup on the reliability of plant water supplies. Experience gained during past severe winters indicated that a more thorough understanding of the potential effects of severe ice conditions was necessary to confirm that the design and operation of safety-related water supplies would ensure adequate operation of safety systems. Guidance for the review of licensee submittals regarding ice effects is in SRP Section 2.4.7, "Ice Effects."

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that Section 6.2.2 of the SSAR describes the UHS design and discusses the features that prevent freezing in the passive containment cooling system. This issue was addressed and resolved through the resolution of Issue 153, which is discussed for the AP600 design in Section 20.3 of this report. Therefore, on the basis of the staff's conclusions in this section, Issue B-32 is resolved for the AP600 design.

Issue B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and Normal Ventilation Systems

As discussed in NUREG-0933, Issue B-36 addressed the staff concern that the then-current guidance and staff technical positions regarding ESF and normal ventilation system air filtration and adsorption units needed to be revised. This issue was resolved by the issuance of Revision 2 of RG 1.52 for ESF ventilation filter units in March 1978, and Revision 1 of RG 1.140 for normal ventilation filter units in October 1979.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that there are no safety-related air filtration systems in the AP600 design. The specific functions of the normal ventilation systems are outlined in Sections 6.4 and 9.4.1 of the SSAR, with a discussion on the conformance with RG 1.140 in SSAR Appendix 1A.

The staff determined that Issue B-36 is closed for the AP600 design because the non-radioactive ventilation system (VBS) and the containment air filtration system (VFS) conform to RG 1.140. For the defense-in-depth filtration function of the VBS and VFS, Appendix 1A of the SSAR provides a comparison of the AP600 design to RG 1.140. In addition, Section 9.4 of the SSAR provides direct reference to Appendix 1A of the SSAR. Therefore, Issue B-36 is resolved for the AP600 design.

Issue B-53: Load Break Switch

GDC 17 of Appendix A to 10 CFR 50 requires that two offsite circuits be available to supply vital plant loads following a loss of all onsite ac power supplies. For those plants with designs that rely on a generator load break switch (or circuit breaker), the switch (or breaker) is relied on to isolate the main generator from the main transformer following a turbine trip to allow power to

be fed from the grid through the main transformer as a second offsite power source to the onsite Class 1E power system.

The AP600 design incorporates a generator load circuit breaker to provide a reliable source of ac power to the electrical systems; however, the AP600 design does not require ac power sources for design-basis accidents.

Therefore, Issue B-53 is not applicable and is considered resolved for the AP600 design.

Issue B-56: Diesel Reliability

Issues that result in a loss of offsite power necessitate reliance on the onsite emergency diesel generators for successful accident mitigation. Improvement of the starting reliability of onsite emergency diesel generators would reduce the probability of events that could lead to core-melt accident.

The AP600 diesel generators are non-Class 1E and are not required for accident mitigation, and their reliability is founded on industry standards and practices.

Therefore, Issue B-56 is not applicable and is considered resolved for the AP600 design.

Issue B-61: Allowable ECCS Equipment Outage Periods

As discussed in NUREG-0933, Issue B-61 addresses establishing surveillance test intervals and allowable equipment outage periods, using analytically based criteria and methods for the TSs. The present TS-allowable equipment outage intervals and test intervals were determined primarily on the basis of engineering judgment. Studies performed by the NRC on operating reactors indicated that from 30 to 80 percent of the emergency core cooling system (ECCS) unavailability was the result of testing, maintenance, and allowed outage periods. Therefore, by optimizing the allowed outage period and the test and maintenance interval, the equipment unavailability and public risk can be reduced.

In its May 28, 1993, letter, Westinghouse stated that the surveillance test intervals and allowable outage times are in the AP600 TSs and are determined from a combination of deterministic, analytical, and PRA-based evaluations.

In the initial submittal from Westinghouse, most of the allowable outage times were not specified in the AP600 TS (they are listed as "to be determined") because Westinghouse had not completed its evaluations. Also, the RTNSS tentatively identified several non-safety systems for the AP600 design that may warrant outage times listed in the TSs. This was Open Item 20.2-11.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 surveillance test intervals and allowable outage times help to meet plant safety goals while maximizing plant availability and operability. In determining these limits for the AP600 TSs, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

The staff's evaluation of the AP600 TSs is in Section 16 of this report. On the basis of this evaluation and Westinghouse's response to this issue, Open Item 20.2-11 and Issue B-61 are resolved for the AP600 design.

Issue B-63: Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary (RCPB)

Several systems connected to the RCPB have design pressures that are considerably below the RCS operating pressure. The NRC has required that valves forming the interface between these high- and low-pressure systems have sufficient redundancy to ensure that the low-pressure systems are not subjected to pressures beyond their design limits; however, there have been discussions about the adequacy of the isolation of low-pressure systems that are connected to the RCPB.

The resolution of this issue has been subsumed by the resolution of GSI-105, which is discussed in Section 20.3 of this report. Therefore, Issue B-63 is resolved for the AP600 design.

Issue B-66: Control Room Infiltration Measurements

The control room area ventilation systems and control building layout and structures are reviewed to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases, and that the control room can be maintained as the backup center from which technical personnel can safely operate during an accident. A key parameter affecting control room habitability is the rate of air infiltration into the control room. Current estimates of these rates are dependent on data relating to buildings that are substantially different from typical control room buildings in nuclear power plants.

As discussed in NUREG-0933, Issue B-66 was intended to facilitate compliance with staff requirements and guidance on control room habitability, specifically (1) GDC 19 and (2) SRP Sections 6.4, "Control Room Habitability Systems," and 9.4.1, "Control Building Ventilation Systems." Additional experimentally measured air exchange rates of operating reactor control rooms resulted in Revision 2 of SRP Section 6.4. See also the resolution of Issues 83 and III.D.3.4 for the AP600 design in Sections 20.3 and 20.4, respectively, of this report.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the MCR for the AP600 design is essentially leak-tight. Unfiltered air in-leakage is minimized by maintaining the MCR at a slightly positive pressure and the verification of the design infiltration rate is in accordance with SRP Section 6.4, "Control Room Habitability Systems." Control room habitability is discussed in Section 6.4 of the SSAR.

In Sections 6.4.5.1 and 14.2.9.1.6 of the SSAR, Westinghouse committed to performing preoperational testing for in-leakage during main control room emergency habitability system (VES) operation in accordance with ASTM E741, "Standard Test Method for Determining Air Leakage Rate by Tracer Dilution." In addition, in Section 6.4.5.4 of the SSAR, Westinghouse committed to conducting testing for MCR in-leakage during VES operation in accordance with ASTM E741 once every 10 years. Issue B-66 is resolved because the staff concluded that the testing described above will ensure that the AP600 design meets the dose limits of GDC 19.

Issue C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

As discussed in NUREG-0933, Issue C-1 addressed staff concerns regarding the long-term capability of hermetically sealed instruments and equipment that must function in postaccident environments. Certain classes of instrumentation incorporate these seals. When safety-related components within the containment must function during post-LOCA conditions, their operability is sensitive to the ingress of steam or water. This issue was resolved with the NRC Memorandum and Order CLI-80-21, docketed May 27, 1980, that ordered that applicants must meet NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," July 1981, to satisfy those aspects of Appendix A to 10 CFR Part 50 that relate to the environmental qualification of safety-related electrical equipment.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the AP600 environmental qualification program described in response to Issue A-24 addresses qualification of safety-related instrumentation and electrical equipment that must function under accident conditions. This program confirms the integrity of seals employed in the design of Class 1E equipment. There is discussion on this program in Issue A-24 above, in this section, and in Section 3.11 of the SSAR.

On the basis of the staff's review, which is discussed in Section 3.11 of this report, the staff concludes that the AP600 environmental qualification program adequately confirms the integrity of seals used in the design of Class 1E equipment and Issue C-1 is resolved for the AP600 design.

Issue C-4: Statistical Methods for ECCS Analysis

As discussed in NUREG-0933, Issue C-4 addressed the statistical methods used for performance evaluation of the ECCS during a LOCA. In accordance with the requirements of 10 CFR 50.46 as amended on September 16, 1988, the NRC requires that the LOCA analyses for license applications use either the 10 CFR Part 50 (Appendix K) evaluation models or the statistical (realistic) models, including the uncertainty of calculation in the adverse direction. The realistic models must be supported by applicable experimental data. Uncertainties in the realistic models and input must be identified and assessed so that uncertainty in the calculated results can be estimated.

In SSAR Section 1.9.4.2.2, Westinghouse states that the AP600 methodology applied for LOCA analysis is discussed in SSAR Chapter 15.

Appendix K of 10 CFR Part 50 specifies the requirements for LWR ECCS analysis, which call for specific conservatism to be applied to certain models and correlations used in the analysis to account for data uncertainties at the time Appendix K was written. USI C-4 addresses NRC development of a statistical assessment of the uncertain level of the peak cladding temperature limit. In 1988, 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," was revised to allow the realistic ECCS evaluation model, in addition to the evaluation model conforming to the Appendix K requirements. This best estimate evaluation model will use an analytical technique that realistically describes the behavior of the reactor system during a LOCA, with comparisons to applicable experimental data. The realistic evaluation model must identify and account for uncertainties in the analysis method and inputs

so that when the calculated ECCS cooling performance is compared to the acceptance criteria, there is a high level of probability that the criteria would not be exceeded.

As described in SSAR Chapter 15, computer codes WCOBRA/TRAC and NOTRUMP, respectively, are used for the large- and small-break LOCA analyses. WCOBRA/TRAC is a realistic code, and the uncertainties will be included in the analysis. NOTRUMP is a code using the Appendix K requirements. Therefore, Issue C-4 is resolved for the AP600 design.

Issue C-5: Decay Heat Update

As discussed in NUREG-0933, Issue C-5, addressed the specific decay heat models for the LOCA analysis models. In accordance with the requirements of 10 CFR 50.46, as amended on September 16, 1988, the LOCA analyses for license applications should use either the 10 CFR Part 50 (Appendix K) models, or the realistic models supported by applicable experimental data and including uncertainty of calculation in the adverse direction. When Appendix K models are used, the decay heat generation function should be determined by ANS 5.0, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," plus a 20-percent uncertainty factor. When realistic models are used, the decay heat function in ANS 5.1, "Decay Heat Power in Light Water Reactors," is acceptable for licensing applications.

In SSAR Section 1.9.4.2.2, Westinghouse states that the large-break LOCA analyses for the AP600 design, discussed in Section 15.6.5 of the SSAR, use the decay heat model identified in the 1979 ANSI 5.1 standard.

This issue involved following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations. Appendix K of 10 CFR Part 50 requires the use of 1971 ANS Standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," times 1.2 be used for the heat generation rates from the radioactive decay of fission products in the ECCS calculation. The staff has determined that the 1979 ANSI 5.1 is technically acceptable and has allowed this use in the realistic evaluation model. For the AP600 application, the 1971 ANS decay heat model and the 1979 ANSI decay heat model are used in NOTRUMP and WCOBRA/TRAC, respectively, for small- and large-break LOCAs. The staff has completed and documented its review of WCOBRA/TRAC and NOTRUMP in Chapter 15 of this report. The staff considers Issue C-5 resolved for the AP600 design.

Issue C-6: LOCA Heat Sources

As discussed in NUREG-0933, Issue C-6 addressed the issue identified in NUREG-0471 that involved staff evaluations of vendors' data and approaches for determining LOCA heat sources and the need for developing staff positions. The contributors to LOCA heat sources, along with their associated uncertainties and the manner in which they are combined, have an impact on LOCA calculations. The staff informed the Commission in SECY-83-472, "Emergency Core Cooling System Analysis Methods," November 17, 1983, that statistical combination of LOCA heat sources would be allowed to justify the relaxation of non required conservatism in ECCS evaluation models.

In SSAR Section 1.9.4.2.2, Westinghouse states that the discussion of LOCA heat sources for the AP600 design is included in Section 15.6.5 of the SSAR. The staff completed and documented its review of WCOBRA/TRAC and NOTRUMP in Chapter 15 of this report. The staff considers Issue C-6 resolved for the AP600 design.

Issue C-10: Effective Operation of Containment Sprays in a LOCA

As discussed in NUREG-0933, Issue C-10 addressed the effectiveness of various containment sprays to remove airborne radioactive material that could be present within the containment following a LOCA. This was expanded to include the possible damage to equipment located within the containment as a result of an inadvertent actuation of the sprays.

The AP600 relies on natural mechanisms, which are enhanced by the PCS, for the removal of airborne radioactive material post-LOCA. The staff's evaluation of these natural removal mechanisms (such as holdup, sedimentation, and diffusion) can be found in Section 15.3 of this report. In a staff requirements memorandum, dated June 30, 1997, the Commission approved the staff's recommendation that the AP600 include a containment spray system or equivalent for accident management following a severe accident. The containment spray system is described in Section 6.5.2 of the SSAR and the staff's evaluation of the system is in Section 19.2.3.3.9, "Non-safety-Related Containment Spray System," of this report. Westinghouse concluded that inadvertent actuation of the containment spray system was not credible in Section 6.5.2 of the SSAR. The staff's evaluation of this conclusion is in Section 6.2.1.1.4 of this report. On the basis of the staff's evaluations in Sections 6.2.1.1.4 and 15.3 of this report, Issue C-10 is resolved for the AP600.

Issue C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

As discussed in NUREG-0933, Issue C-17 was intended to develop criteria for the acceptability of radwaste solidification agents to properly implement a process control program for packaging diverse radioactive plant wastes for shallow land burial. There are no current criteria for a finding of acceptability of solidification agents.

As stated in NUREG-0933, the Commission issued 10 CFR Part 61 on licensing requirements for land disposal of radioactive waste, including Section 61.56, which addresses acceptable waste characteristics. Also, the staff developed BTP ETSB 11-3 to be part of SRP Section 11.4, "Solid Waste Management Systems," and provide design guidance for solid waste management systems (SWMSs) to be used at LWRs. Therefore, this issue has been resolved for implementation at nuclear power plants.

In Section 1.9.4.2.2 of the SSAR, Westinghouse states that the solid radwaste system for the AP600 design transfers, stores, and prepares spent ion exchange resins for disposal. The system also provides for disposal of filter elements, and sorting, shredding, and compaction of compressible dry active wastes. The solid radwaste system does not provide for liquid waste concentration or solidification. This will be provided using mobile systems. Solidification of waste is not performed by permanently installed systems.

The staff evaluated the conformance of the AP600 design to Section 11.4 of the SRP in Section 11.4 of this report. On the basis of the staff's conclusions in this section, Issue C-17 is resolved for the AP600 design.

20.3 New Generic Issues

The new generic issues of NUREG-0933 listed in Table 20.1-1 are evaluated against the AP600 design in this section. The majority of the items were chosen either because (1) 10 CFR 52.47(a)(1)(iv) or 10 CFR 50.34(f) require the design to comply with them, or (2) Westinghouse decided that the item applied to the design and included a discussion of the item in the SSAR. In addition, the staff requested, and Westinghouse agreed, to address Issues 14, 22, 29, 43, and 82 for the AP600 design. The staff also decided to include a discussion of Issue 122.2 in this section.

The references to SRP sections in this chapter are references to sections in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1982. The references to GDC are references to the GDC in Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Issue 14: PWR Pipe Cracks

As discussed in NUREG-0933, Issue 14 addressed cracking in PWR non primary (i.e., secondary) piping systems as a result of stress corrosion, vibratory and thermal fatigue, and dynamic loading. Cracking in PWR non primary system piping could lead to a decrease of the system functional capability and could possibly result in such situations as degraded core cooling. This issue deals with occurrences of MFW line cracking in certain Westinghouse and Combustion Engineering PWRs. In September 1980, the PWR Pipe Study Group completed its investigation of the issue and published its findings in NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping of Pressurized Water Reactors," dated September 1980. This report provides conclusions regarding systems safety and recommends technical solutions to the issue. The staff considered augmented inspections and inspection requirements, and concluded that they had low risk-reduction value. Therefore, this issue was resolved and no new requirements were established. This issue, however, is not required for the AP600 design to meet 52.47(a)(1)(ii) or (iv).

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because the issue had no new requirements. Although this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv), the staff requested in its DSER that Westinghouse address pipe cracks for the AP600 design. The staff also requested Westinghouse to discuss the AP600 compliance with the regulatory guidance provided in SRP Section 6.6, "Inservice Inspection of Class 2 and 3 Components," for all ASME Class 2 and Class 3 piping systems. This was Open Item 20.3-1.

In SSAR Section 1.9.4.2.3, Westinghouse states that the design and inspection requirements for feedwater lines are in Section 10.4.7 of the AP600 SSAR. Further, the issue of inservice inspection of Class 2 and 3 components is addressed in SSAR Section 6.6, "Inservice Inspection of Class 2 and 3 Components." All issues identified with inservice inspection of Class 2 and 3 components have been resolved. On that basis, Open Item 20.3-1 is closed and Issue 14 is resolved for the AP600 design.

Issue 15: Radiation Effects on Reactor Vessel Supports

As discussed in NUREG-0933, Issue 15 addresses the potential for radiation embrittlement of reactor vessel support structures. Neutron irradiation of structural materials causes embrittlement that may increase the potential for propagation of flaws that might exist in the materials. The potential for brittle fracture of these materials is typically measured in terms of the material's nil ductility transition temperature (NDTT). As long as the operating environment in which the materials are used has a higher temperature than the material's NDTT, failure by brittle fracture is not expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT, that is, they become more susceptible to brittle fracture at the operating temperatures of interest. This effect has to be accounted for in the design and fabrication of reactor vessel support structures.

As discussed in NUREG-0933, this issue had a high-priority ranking; but after extensive evaluation, the staff concluded that no new requirements needed to be issued by the NRC.

In SSAR Section 1.9.4.2.3, Westinghouse states that the supports for the AP600 reactor vessel are designed for loading conditions and environmental factors, including the neutron fluence. The material requirements are stated to include fracture toughness requirements and impact testing requirements in compliance with ASME Code, Section III, Subsection NF. These supports are not in the region of high neutron fluence where neutron radiation embrittlement of the supports would be a significant concern.

On the basis of the above, the staff considers the reactor vessel supports for the AP600 design to be adequately designed for radiation effects, and Issue 15 is resolved for the AP600 design.

Issue 22: Inadvertent Boron Dilution Events

As discussed in NUREG-0933, Issue 22 addressed the possibility of core criticality during cold-shutdown conditions from inadvertent boron dilution events. Although this issue was resolved with no new requirements, the acceptance criterion is that plants shall minimize the consequences of such events by meeting SRP Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)." Specifically, the plant shall respond in such a way that the criteria regarding fuel damage and system pressure are met, and the dilution transient is terminated before the shutdown margin is eliminated. If operator action is required to terminate the transient, redundant alarms must be in place and the following minimum time intervals must be available between when an alarm announces an unplanned dilution and when shutdown margin is lost:

- 30 minutes during refueling (Mode 6)
- 15 minutes during all other operating modes

In Section 15.4.6 of the SSAR, Westinghouse provides a safety analysis that demonstrates that redundant alarms are available to enable operators to detect and terminate an inadvertent boron dilution event within the above required time intervals, before shutdown margin is lost.

In addition to the events in this issue, the staff identified the following two boron dilution scenarios where a deborated water slug may accumulate in the RCS and a restart of the RCPs will cause this slug to pass through the core, resulting in criticality or a power excursion:

- The first scenario occurs during a plant startup when the reactor is deborated as part of startup procedures. A loss of offsite power will result in tripping the RCPs and charging pump. The subsequent startup of the diesel generator will restart the charging pump and cause the accumulation of deborated water in the reactor lower plenum. The RCP restart with recovery of offsite power will cause this deborated water to pass through the core.

- The second scenario is related to transients or accidents, such as a small-break LOCA with heat removal by reflux condensation natural circulation that may result in an accumulation of deborated water in the RCS loop. This water will pass through the core with an inadvertent restart of the RCPs.

The staff completed and documented its review of inadvertent boron dilution issue in Section 15.2.6.5.4 of this report. The staff considers Open Item 20.3-2 closed and Issue 22 resolved for the AP600 design.

Issue 23: Reactor Coolant Pump Seal Failures

As discussed in NUREG-0933, Issue 23 addressed the concerns about RCP seal failures that could cause a SBLOCA. PRA analyses have indicated that the overall probability of core damage as a result of a small break could be dominated by RCP seal failures. This issue includes improving the reliability of RCP seals by reducing the probability of seal failure during normal operations and under abnormal conditions. Specifically, acceptable resolutions to this issue include an RCP seal design that ensures the RCP seal integrity following SBO for an extended period.

In SSAR Section 1.9.4.2.3, Westinghouse states that the AP600 reactor coolant pumps are canned motor pumps that contain the motor and all rotating components inside a pressure vessel designed for full reactor coolant system pressure. They state that the shaft for the pump impeller and rotor is within this vessel; therefore, seals are not required. Further discussion on the canned motor pump design is in Section 5.4.1 of the SSAR. Westinghouse concludes that because the reactor coolant pumps do not rely on seals as being part of the reactor coolant pressure boundary, Issue 23 is not applicable to the AP600 design.

The staff agrees that the AP600 design uses canned motor reactor coolant pumps, which contain the motor and all rotating components inside a pressure vessel designed for full RCS pressure. The shaft for the impeller and rotor is contained within the pressure boundary; therefore, the staff concludes that seals are not required to restrict leakage out of the pump into containment and Issue 23 does not apply to the AP600 design. On the basis of the above, Issue 23 is resolved for the AP600 design.

Issue 24: Automatic ECCS Switchover To Recirculation

Issue 24 addresses the staff's concerns following a review of operating events that indicated a significant number of ECCS spurious actuations, particularly the four events that occurred at the

Davis Besse plant during 1980. Switchover from injection to recirculation involves realignment of several valves and may be achieved by (1) manual realignment, (2) automatic realignment, or (3) a combination of both. Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures. The safety significance of the issue is that switching suction to the sump prematurely could adversely affect the accident because the containment sump may not have enough inventory to provide pump suction. In NUREG-0933, this issue was classified as medium-safety priority but had not been generically resolved.

In SSAR Revision 9, Section 1.9.4.2.3, Westinghouse's response to the issue states that "The AP600 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened. The AP600 does provide for automatic opening of the recirculation line on a low level signal from the in-containment refueling water storage tank." The staff notes that the AP600 does not have safety-related pumps, as do the plants originally addressed by Issue 24. Furthermore, if the recirculation line were opened in the AP600, the flow path from the IRWST to the reactor vessel would still exist, whereas for conventional PWRs the flow path from the RWST would be closed when recirculation mode is entered. Therefore, for the AP600, the situation is not analogous to that addressed by Issue 24 for operating PWRs.

In its DSER, the staff requested in Open Item 20.3-3 that Westinghouse address Issue 24. Westinghouse has done this in SSAR Revision 9, as discussed in the preceding paragraph. Therefore, the staff finds that Open Item 20.3-3 is closed and Issue 24 is resolved for the AP600.

Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

As discussed in NUREG-0933, Issue 29 addressed staff concerns about the number of events involving the degradation of threaded fasteners (such as bolt cracking, corrosion and failure) in operating plants from 1964 to the early 1980s. Many of the events were related to components of the RCPB and support structures of major components. This raised questions about the integrity of the RCPB and the reliability of the component support structures following a LOCA or a seismic event. The licensees reported failures involving a variety of threaded fasteners and most frequently reported degradation mechanisms were wastage (corrosion) from boric acid attack and stress corrosion cracking (SCC). The former occurred more often at RCPB joints; the latter in structural bolting.

This issue was resolved and no new requirements were established on the basis of (1) operating experience with bolting in both nuclear and conventional power plants; (2) actions already taken through bulletins, generic letters, and information notices since 1982; and (3) industry-proposed recommendations and actions, which are documented in the EPRI Reports NP-5769 ("Degradation and Failure of Bolting in Nuclear Power Plants," April 1988) and NP-5067 ("Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel," Volume 1: "Large Bolt Manual," 1987 and Volume 2: "Small Bolts and Threaded Fasteners," 1990). The resolution of this issue is documented in GL 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants," dated October 17, 1991; and NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," dated June 30, 1990.

In SSAR Section 1.9.4.2.3, Westinghouse states that the elements of resolution of this issue pertain to operational and maintenance practices, which will be addressed by the COL applicant. It also states that conformance to the ASME Code, Section III requirements for pressure boundary components and related supports, which the AP600 design meets, will provide safe operation in the event of bolting degradation. Further, because of the emphasis in the AP600 design on access for maintenance and inspection, the recommended maintenance practices can be readily implemented.

The staff concludes that Westinghouse has adequately addressed this issue for the AP600 design; therefore, Issue 29 is resolved for the AP600 design.

Issue 43: Reliability of Air Systems

As discussed in NUREG-0933, Issue 43 is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv); however, the staff believed it should be addressed for the AP600 design because the issue dealt with all causes of air system unavailability. The issue addressed the incident at Rancho Seco where desiccant particles in the valve operator caused the slow closure of a containment isolation valve. Desiccant contamination in the instrument air system (IAS) was also found to be a contributing cause of the loss of the salt water cooling system at San Onofre in March 1980; this incident resulted in Issue 44, "Failure of the Saltwater Cooling System." Because the only new generic concern found in the evaluation of the San Onofre event was the common-cause failure of safety-related components as a result of contamination of the IAS, Issue 44 was combined with Issue 43.

Issue 43 was broadened to include all causes of air system unavailability because U.S. LWRs rely upon air systems to actuate or control safety-related equipment during normal operation even though they are not safety-grade systems at most operating plants. Safety system design criteria require (and plant accident analyses assume) that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform its intended function with the assistance of backup accumulators. An NRC Office for Analysis and Evaluation of Operational Data (AEOD) case study highlighted 29 failures of safety-related systems that resulted from degraded or malfunctioning air systems. These failures contradict the requirement that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or will perform its intended function with the assistance of backup accumulators. Some of the systems that may be significantly degraded or failed are decay heat removal, auxiliary feedwater, boiling-water reactor scram, main steam isolation, salt water cooling, emergency diesel generator, containment isolation, and the fuel pool seal system. The end result of degradation or failure of safety or safety-related systems is an increase in the expected frequency of core-melt events and, therefore, an increase in public risk.

This issue was resolved by the issuance of GL 88-14, "Instrument Air Supply Problems Affecting Safety-Related Equipment," dated August 8, 1988, which required licensees and applicants to review the recommendations of NUREG-1275 ("Operating Experience Feedback Report – Air Systems Problems," two volumes, dated July and December 1987, respectively) and perform a design and operations verification of the IAS. The following is a discussion of the

purposes for which Westinghouse considered the recommendations in NUREG-1275, Volume 2, for the AP600 design:

(1) Ensure that air system quality is consistent with equipment specifications and is periodically monitored and tested.

In Section 9.3.1 of the SSAR, Westinghouse states that in accordance with NUREG-1275, instrument air quality meets the manufacturer's standards for pneumatic equipment supplied as part of the plant. In addition, periodic checks are made to assure high-quality instrument air as specified in ANSI/ISA-S7.3, "Quality Standard for Instrument Air."

(2) Ensure adequate operator response by formulating and implementing anticipated transient and system recovery procedures for loss-of-air events.

In Section 9.3.7 of the SSAR, Westinghouse states that the COL applicant will address SSAR 1.9.4.2.3, Issue 43 as part of training and procedures identified in Section 13.5 of the SSAR.

(3) Improve training to ensure that plant operations and maintenance personnel are sensitized to the importance of air systems to common mode failures.

In Section 9.3.7 of the SSAR, Westinghouse states that the COL applicant will address SSAR 1.9.4.2.3, Issue 43, as part of training and procedures identified in Section 13.5 of the SSAR.

(4) Confirm the adequacy and reliability of safety-related backup accumulators.

In Section 9.3.1 of the SSAR, Westinghouse states that there are no safety-related air operated valves that rely on safety-related air accumulators to actuate to the fail safe position upon loss of air pressure.

(5) Verify equipment response to gradual losses of air to ensure that such losses do not result in events that fall outside FSAR analysis.

In Section 9.3.1.4 of the SSAR, Westinghouse states that during initial plant testing before reactor startup, safety systems utilizing instrument air will be tested as part of the safety system test to verify fail-safe operation of air-operated valves upon sudden loss of instrument air or gradual reduction of air pressure as described in RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems."

In a May 28, 1993 letter, Westinghouse concluded that this issue was not relevant to the AP600 design because the issue is the responsibility of the COL applicant. In the DSER, the staff requested that Westinghouse address this issue for the AP600 design because some of the recommendations in NUREG-1275, Volume 2, pertain to the quality of the design of the air systems, the adequacy and reliability of safety-related backup accumulators, and to verify that equipment response to gradual losses of air do not result in events outside the accident analyses, which are not the responsibility of the COL applicant. This was designated as DSER Open Item 20.3-4. Westinghouse revised the SSAR as described above to address all of the

recommendations. The staff finds Westinghouse's response acceptable. Therefore, DSER Open Item 20.3-4 is closed and Issue 43 is resolved for the AP600 design.

Issue 45: Inoperability of Instruments Due to Extreme Cold Weather

As discussed in NUREG-0933, Issue 45 addressed the potential for safety-related equipment instrument lines to become inoperable as a result of freezing or reaching the precipitation point of the sensing fluids. Typical safety-related systems employ pressure and level sensors that use small-bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to ambient temperature conditions. These lines generally contain liquid (e.g., borated water) that is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to subfreezing temperatures, heat tracing or insulation or both is used to minimize the effects of cold temperatures. These sensing lines are of concern because, should they freeze, they may prevent a safety-related system or component from performing its safety function.

To resolve this issue, the staff issued RG 1.151, "Instrument Sensing Lines," to supplement the existing guidance and requirements in the SRP, applicable GDC, and Instrument Society of America (ISA) standard ISA-67.02, "Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants." RG 1.151 addresses the prevention of freezing in safety-related instrument-sensing lines and includes such design issues as diversity, independence, monitoring, and alarms. In February 1984, SRP Sections 7.1, "Instrumentation and Controls – Introduction," Revision 3; Appendix A, Revision 1, to Section 7.1; and 7.7, "Control Systems," Revision 3 were revised to incorporate the resolution of this issue. Thus, this issue was resolved and new requirements were issued.

In SSAR Section 1.9.4.2.3, Westinghouse states that the AP600 design complies with SRP Sections 7.1; Appendix A to Section 7.1; 7.5, "Information Systems Important to Safety"; and 7.7. The conformance of the AP600 design to RG 1.151 is addressed in SSAR Appendix 1A.

On the basis of this, the staff concludes that the AP600 design complies with the relevant sections of RG 1.151 and the updated SRP sections. Therefore, Issue 45 is resolved for the AP600 design.

Issue 51: Improving the Reliability of Open-Cycle Service Water Systems

As discussed in NUREG-0933, Issue 51 addressed fouling of safety-related open-cycle service water systems by either mud, silt, corrosion products, or aquatic bivalves. This problem has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation in nuclear power plants. This issue was originally to address only aquatic bivalves. However, the issues on flow blockage in essential equipment caused by Corbicula (Issue 32) and service water system flow blockage caused by Blue Mussels (Issue 52) were incorporated into this issue, and Issue 51 was expanded to consider if the NRC staff should develop new requirements for improving the reliability of open cycle water systems. New requirements were issued in GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, on baseline fouling programs for nuclear power plants.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that the service water system for the AP600 design provides cooling water to the component cooling water system and has no safety-related functions. It is stated that none of the safety-related equipment requires water cooling to effect a safe shutdown or mitigate the effects of design-basis events. Heat transfer to the UHS is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell.

The design of the service water system and the provisions for minimizing long-term corrosion and organic fouling are discussed in Section 9.2.1 of the SSAR.

On the basis of the staff's review, which is discussed in Section 9.2.1 of this report, the staff concludes that the service water system is adequately designed to minimize fouling, and Issue 51 is resolved for the AP600 design.

Issue 57: Effects of Fire-Protection Systems Actuation on Safety-Related Equipment

NUREG-0933, "Generic Issues," Issue 57, and NUREG-5580, "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety Related Equipment," addressed fire protection system (FPS) actuations that have caused adverse interactions with safety-related equipment at operating nuclear power plants. Experience has shown that safety-related equipment subjected to water spray, as from the FPS, could be rendered inoperable and that numerous spurious actuations of the FPS have been initiated by operator testing errors or by maintenance activities, steam, or high humidity in the vicinity of FPS detectors.

SSAR Section 9A.3.1.1, "Containment/Shield Building," specifies that inadvertent operation of an automatic suppression system is prevented by the normally closed containment isolation valve in the water supply line. Operator action is required to open this valve and admit water to the system. Therefore, because the AP600 design does not provide automatic fire suppression in safety-related areas, Issue 57 for the AP600 design is considered resolved.

Issue 67.3.3: Improved Accident Monitoring

As discussed in NUREG-0933, Issue 67.3.3 addressed weaknesses in reactor system monitoring that could inhibit correct operator responses to events similar to the steam generator tube rupture (SGTR) event at the Ginna Power Plant on January 25, 1982. During the event, weaknesses in accident monitoring were apparent including (1) non redundant monitoring of RCS pressure, (2) failure of the position indication for the SG relief and safety valves, and (3) limited range of the charging pump flow indicator. As stated in NUREG-0933 and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980, (Supplement 1, January 1983), the implementation of the recommendations described in RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Revision 2, December 1980, resolved this issue.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because the issue was superseded by other issues. However, the AP600 post-accident monitoring system is described in Section 7.5 of the SSAR.

The staff concludes, as stated in Section 7.5 of this report, that the post-accident monitoring system conforms to Revision 3 of RG 1.97 and is acceptable. However, in DSER Open Item 20.3-5 the staff requested that Westinghouse include reference to the post-accident monitoring system and its capability when addressing this issue. Westinghouse has adequately addressed this issue in SSAR Section 1.9.4.2.3. This section states that the guidance of RG 1.97 is followed in the AP600 design. Therefore, DSER Open Item 20.3-5 is closed and Issue 67.3.3 is resolved for the AP600 design.

Issue 70: PORV and Block Valve Reliability

Power-operated relief valves (PORVs) and block valves were originally designed as non-safety components in the reactor pressure control system for use only when plants are in operation; the block valves were installed because of expected leakage from the PORVs. Neither valve type was needed to safely shut down a plant or mitigate the consequences of accidents. In 1983, the staff determined that PORVs were relied on to mitigate design-basis SGTR accidents and questioned the acceptability of relying on non-safety-grade components to mitigate DBAs. NUREG-0933, Issue 70, addressed the assessment of the need for improving the reliability of PORVs and block valves.

In Section 1.9.3 of the SSAR, Item (1)(iv), Westinghouse states that the AP600 design does not include PORVs. Overpressure protection is provided by two totally enclosed pop-type safety valves. If the pressurizer pressure exceeds the set pressure, the safety valves lift. A temperature indicator in the discharge piping for each safety valve alarms on high temperature to alert the operator to when the valves open. The staff concludes that because the AP600 design does not include PORVs and block valves, Issue 70 is not applicable. Therefore, Issue 70 is resolved for the AP600 design.

Issue 73: Detached Thermal Sleeves

As discussed in NUREG-0933, Issue 73 addressed the staff concerns, during the period 1978 to 1980, about reports of fatigue failures of thermal sleeve assemblies in the piping systems of both PWRs and BWRs. There have been five generations (0 through 4) of thermal sleeves used in Westinghouse reactors. Only "Generation 3" thermal sleeves have been found to be susceptible to high-cycle stresses due to flow-induced vibrations because of the particular weld attachments used in that design. The vibrations caused fatigue failures at the attachment welds and subsequent cracking and tearing away of the thermal sleeves. This issue was applicable to the design and operation of approximately 20 Westinghouse plants that used that generation thermal sleeve. This issue was resolved for Westinghouse plants with the publication of NUREG/CR-6010, "History and Current Status of Generation 3 Thermal Sleeves in Westinghouse Nuclear Power Plants," July 1992.

In the DSER, the staff requested that Westinghouse include a discussion of this issue in the SSAR. This was DSER Open Item 20.3-6. In Revision 7 to the SSAR, Issue 73 was added to Section 1.9.4.2.3, which states that the AP600 does not use Generation 3 thermal sleeves. Therefore, Open Item 20.3-6 is closed and Issue 73 is resolved for the AP600 design.

Issue 75: Generic Implications of ATWS Events at Salem Nuclear Plant

As discussed in NUREG-0933, Issue 75 addressed the generic implications of two events at Salem Unit 1 where there were failures to scram automatically because of the failure of both reactor trip breakers to open on receipt of an actuation signal. This issue was expanded to include a number of issues raised by the staff that were closely related to the design and testing of the reactor protection system (RPS). The requirements for this issue were stated in GL 83-28, "Required Actions Based on Generic Implications of Salem ATWS Event," dated July 8, 1983.

The actions covered by GL 83-28 fell into the following four areas:

(1) Post-Trip Review–This action addresses the program, procedures, and data collection capability to ensure that the causes for unscheduled reactor shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

(2) Equipment classification and vendor interface–This action addresses the programs for ensuring that all components necessary for performing required safety-related functions are properly identified in documents, procedures, and information-handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

(3) Post-maintenance testing–This action addresses post-maintenance operability testing of safety-related components.

(4) RTS reliability improvements–The intent of this action is to ensure that (a) vendor-recommended reactor trip breaker modifications and associated RPS changes are completed in PWRs, (b) a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, (c) the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their RTS, and (d) online functional testing of the RTS is performed on all LWRs.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because this was not a design issue. The staff did not agree with this conclusion in that this issue is related to recording and displaying all system parameters for subsequent use by plant personnel and online testing of the RTS, which are design issues. In DSER Open Item 20.3-7, the staff requested that Westinghouse address this issue for the AP600 design. Westinghouse has addressed this issue in Section 1.9.4.2.3 of the SSAR. In addition, the discussion in SSAR Sections 7.1, 7.2, and 7.5 describe relevant design aspects of the RTS. Therefore, DSER Open Item 20.3-7 is closed and Issue 75 is resolved for the AP600 design.

Issue 79: Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

As discussed in NUREG-0933, Issue 79 addressed the concern for an unanalyzed reactor vessel thermal stress during natural convection cooldown (NCC) of PWR reactors. The concern emerged from a preliminary evaluation of the voiding event that occurred in the upper

head of the St. Lucie Unit 1 reactor on June 11, 1980. On the basis of several conservative assumptions, Babcock and Wilcox (B&W) tentatively concluded that during natural convection cooling, axial temperature gradients could develop in the vessel flange area, which could produce thermal stresses in the flange area or in the studs that might exceed values allowed by the code when added to the stresses already considered (such as boltup loads or pressure loads).

The staff's efforts to resolve this issue were founded on a review of a B&W NCC analysis and the results of a NCC analysis by a NRC contractor, both of which were performed for the B&W 177 fuel assembly reactor vessel. The staff's evaluation and resolution of Issue 79 is documented in NUREG-1374, "An Evaluation of PWR Reactor Vessel Thermal Stress During NCC," dated May 1991, and GL 92-02. On the basis of conservative analyses and qualitative extrapolation of the results, the staff concluded the following in NUREG-1374:

- The B&W 177 is considered analyzed for NCC events that are bounded by the NCC transient profile shown in Figure 3 of NUREG-1374. The bounding profile in this figure was generated by the staff's contractor by using a conservative assumption of a maximum cooldown rate of 100° F per hour during the NCC event. This profile was used by the contractor in its conservative confirmatory stress analysis of the B&W 177.

- Adequate geometric similarity exists between the B&W 177 and other U.S. PWRs to support extending the findings and conclusions in NUREG-1374 to all U.S. PWRs.

- It is extremely unlikely that a single NCC event will cause the failure of any existing U.S. PWR reactor vessel, even if a cooldown rate of 100° F per hour is exceeded.

- NCC events of the type analyzed (i.e., NCC events that result in the plant being brought to a cold-shutdown condition) have a low frequency of occurrence. The staff is aware of only one such event, which occurred at St. Lucie as discussed above.

This issue was resolved and no new requirements were established because (1) NCC events that result in the plant being brought to a cold-shutdown condition occur infrequently and (2) the actual severity of a specific NCC event will determine the need for actions (if any) and the extent of actions that may be required of any licensee following certain NCC events that may place a reactor vessel in an unanalyzed condition or outside its documented design-basis.

The AP600 response to this issue in SSAR Section 1.9.4.2.3 references SSAR Section 3.9.1.1.2.11 and states that the response to GL-92-02 is the responsibility of the COL. In DSER Open Item 20.3-8, the staff requested Westinghouse to verify that the analyses to account for NCC events applicable to the AP600 reactor vessel integrity were evaluated and bounded by the generic assumptions and conclusions presented in NUREG-1374 and GL 92-02. In Section 3.9.1.1 of the SSAR, Westinghouse presents the AP600 design transients that are considered in the design and fatigue analysis of ASME Class 1 components. As discussed in Section 3.9.1.1 of this report, all of these transients have been adjusted for a 60-year plant life. In Section 3.9.1.1.2.11 of the SSAR, the total number of NCC transients used in the reactor vessel design for its 60-year life span is specified. In addition, in Figure 5.3-3 of the SSAR, a generic curve presenting operating temperature, pressure, and cool down rate (not exceeding 100 °F/hr) for the reactor vessel is provided, which is consistent

with recommendations stated in GL 92-02 and NUREG-1374. On the basis of above information, the staff has concluded that the AP600 analyses to account for NCC events are bounded by the analyses discussed in NUREG-1374, and are acceptable. Therefore, DSER Open Item 20.3-8 is closed.

In the DSER, the staff requested that Westinghouse acknowledge the responsibility of the COL applicant to follow the requirement in 10 CFR 50.73 to report a NCC event that has placed the reactor vessel outside its design-basis and to provide confirmation that no applicable regulatory design or fracture toughness criteria has been exceeded. This was DSER Open Item 20.3-9.

Westinghouse responded to this DSER open item by stating that the COL applicant is responsible for implementing a Licensee Event Report (LER) in conformance with 10 CFR 50.73. The response further stated that because this LER system will include NCC events that place the reactor pressure vessel either outside (1) its design basis, (2) applicable regulatory design criteria, or (3) fracture toughness criteria, a separate COL action item is not necessary. The staff agrees with this response. Therefore, DSER Open Item 20.3-9 is closed.

On the basis of the above discussions, Issue 79 is resolved for the AP600 design.

Issue 82: Beyond-Design-Basis Accidents in Spent Fuel Pools

The risks of beyond-design-basis accidents in the spent fuel storage pool were examined in WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975, and it was concluded in the report that these risks were orders of magnitude below those involving the reactor core. Issue 82 in NUREG-0933 reexamined accidents in the spent fuel storage pool for two reasons. First, spent fuel is being stored instead of reprocessed. This issue, however, is not required for the AP600 design to meet 52.47(a)(1)(ii) or (iv). This has led to the expansion of onsite fuel storage by means of high-density-storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have offered evidence of the possibility of fire propagation between assemblies in an air-cooled environment. These two reasons, in combination, provide the basis for an accident scenario that was not previously considered.

As stated in NUREG-0933, because of the large inherent safety margins in the design and construction of spent fuel pools, this issue was resolved and no new requirements were established.

In Section 1.9.4.2.3 of the SSAR, Westinghouse stated that the AP600 includes design provisions that preclude draining of the spent fuel pool. Also, provisions are available to supply water to the pool in the event the water covering the spent fuel begins to boil off. In the DSER, the staff requested that Westinghouse address PRA for accidents in the spent fuel pool for the AP600 design. This was designated as DSER Open Item 20.3-10.

Westinghouse provided additional information for NRC staff review that concluded that the likelihood of the initiating event, for the open item postulated, is extremely small for the AP600 design, and that a PRA of this type of event is unnecessary. Westinghouse further concluded that the deterministic design evaluation of the spent fuel pool discussed in Section 9.1 of the SSAR provides sufficient information for the staff to resolve Issue 82. The NRC staff reviewed

the additional information provided by Westinghouse and the updated information in Section 9.1 of the SSAR. As a result of this review, the staff concluded that DSER Open Item 20.3-10 is closed and Issue 82 is resolved for the AP600 design.

Issue 83: Control Room Habitability

As discussed in NUREG-0933, Issue 83 addressed the significant discrepancies found during a survey of existing plant control rooms before 1983. These discrepancies included the inconsistencies between the design, construction, and operation of the control room habitability systems and the descriptions in the licensing-basis documentation. In addition, the staff determined that total system testing was inadequate and that the control systems were not always tested in accordance with the plant TS. Issues related to Issue 83 include (1) Issue B-36, on criteria for air filtration and adsorption units for atmospheric cleanup systems, (2) Issue B-66, on control room infiltration measurements, and (3) Issue III.D.3.4, also on control room habitability. These three issues are discussed in Sections 20.2 and 20.4 of this report.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that habitability of the MCR during normal operation is provided by the non-safety-related nuclear island VBS. In the event of a design-basis accident involving a radiation release or a loss of all ac power event, the non-safety-related nuclear island VBS is automatically terminated, the MCR pressure boundary is isolated, and the safety-related VES is actuated.

The safety-related VES supplies breathable quality air for the MCR operators while the main control room is isolated. In the event of external smoke or radiation release, the non-safety-related nuclear island VBS provides for a supplemental filtration mode of operation, as discussed in Section 9.4 of the SSAR. In the event of a Hi-Hi radiation level, the safety-related VES is actuated. In the unlikely event of a toxic chemical release, the safety-related VES has the capability to be manually actuated by the operators. Further, a 6-hour supply of self-contained portable breathing equipment is stored inside the MCR pressure boundary.

In the DSER, the staff requested that Westinghouse address control room habitability and Issue 83 for the AP600 design. This was designated as DSER Open Item 20.3-11. DSER Open Item 20.3-11 is closed because (1) the staff concluded that the AP600 design meets the dose limits of GDC 19 except for use of the TID14844 source term (the justification for exempting the AP600 design from the requirements to use TID 14844 is provided in Section 20.6 of this report) and (2) the COL applicant will address procedures and training to meet the intent of Issue 83. Therefore, Issue 83 is resolved for the AP600 design.

Issue 87: Failure of High-Pressure Coolant Injection Steamline Without Isolation

Issue 87, in NUREG-0933, addressed the staff concerns about a postulated break in the high-pressure coolant injection (HPCI) steam supply line and the uncertainty regarding the operability of the isolation valves for the HPCI steam supply line under these conditions. A break in the line could lead to high flow and high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested *in situ* for the high design flow rates and pressures. Therefore, subsequent to installation of these valves, it is not feasible to

demonstrate the capability of the valves to close when exposed to the forces created by the flow resulting from a break downstream. This issue was resolved by the issuance of GL 89-10 and its supplements on safety-related motor-operated valve (MOV) testing, GL 96-005, and SECY-93-087, which recommended these valves be periodically tested inservice, under full flow and actual plant conditions where practical. Furthermore, in SECY-94-084 and SECY-95-135, additional guidelines are provided for testing MOVs.

In SSAR Section 1.9.4.2.3, Westinghouse states that for the AP600 design, safety-related MOVs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against the maximum differential pressure and flow. The requirements for MOV qualification testing are outlined in SSAR Section 5.4.8. In SSAR Section 3.9.8.4, Westinghouse further states that the inservice testing (IST) program for safety-related valves is to be submitted by the COL applicant. This IST program will be developed on the basis of the requirements outlined in SSAR Sections 3.9.6 and 5.4.8. The staff concluded that the information related to Issue 87 in the above SSAR sections is acceptable. The staff's evaluation of all MOV-related issues is provided in Section 3.9.6 of this report. On this basis, Issue 87 is resolved for the AP600 design.

Issue 93: Steam Binding of Auxiliary Feedwater Pumps

As discussed in NUREG-0933, Issue 93 addressed the potential for a common-mode failure of the auxiliary feedwater system (AFWS) or the emergency feedwater system (EFWS) resulting from steam binding of the AFW pumps caused by heated MFW leaking back through check valves. The AFWS is used to supply water to the SGs should the MFW system be lost, and steam binding of the AFW pumps could result in the loss of the AFWS.

The AFWS may be isolated from the MFW system by a check valve or one or more isolation valves (depending upon the specific design) to keep hot MFW from entering the AFWS. However, operating experience has shown that check valves tend to leak, thus, permitting hot MFW to enter the AFWS. This hot feedwater can subsequently flash to steam in the AFW pumps and discharge lines, causing steam binding of the pumps.

In addition, the AFW piping is sometimes arranged so that each AFW pump is connected through a single check valve (which is used to prevent back leakage) to piping that is common to two or three pumps. This arrangement creates the potential for common-mode failures as the hot feedwater leaks back through the check valves into other AFW pumps.

The staff issued GL 88-03 ("Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps," dated February 17, 1988) to the industry as the resolution of this issue. The letter implements monitoring and corrective procedures to minimize the likelihood of steam binding of the AFWS pumps. One of the corrective actions to be taken is the monitoring of AFW pump discharge piping temperatures to ensure that the fluid temperatures remain at or near ambient temperature.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that the AP600 design does not have a safety-related auxiliary feedwater system. The passive core cooling system is stated to provide the safety-related function of cooling the RCS in the event of loss of feedwater. The startup feedwater system (SUFS) is stated to provide the SGs with feedwater during startup, hot

standby, cooldown, and when the main feedwater pumps are not available, and have no safety-related function other than containment isolation.

The SUFS includes temperature instrumentation in the pump discharge for monitoring of the temperature of the SUFS. The system also includes a normally closed isolation valve and a normally closed check valve for each pump, limiting potential back leakage.

The staff concluded that steam binding is not a problem for the AP600 design because the passive core cooling system does not have any pumps that could fail as a result of steam binding, and the SUFS is not safety-related. Therefore, Issue 93 is resolved for the AP600 design.

Issue 94: Additional Low-Temperature Overpressure Protection for LWRs

As discussed in NUREG-0933, Issue 94 addressed low-pressure overpressurization events since the resolution of Issue A-26, which is discussed in Section 20.2 of this report. Therefore, this issue was intended to address the additional guidance for RCS low-temperature overpressure protection (LTOP) to ensure reactor vessel integrity beyond the requirements specified for Issue A-26 in SRP Section 5.2.2, "Overpressure Protection," and BTP RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperature." Issue 94 was resolved with the additional requirements to have the TS for overpressure protection consistent with those specified in Enclosure B to GL 90-06, "Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors, Pursuant to 10 CFR 50.54(f)," dated June 25, 1990.

In SSAR Section 1.9.4.2.3, the Westinghouse response to Issue 94 states that the reactor vessel for the AP600 design is designed to be less susceptible to brittle fracture during an LTOP event. Material requirements and welding processes are developed to enhance resistance to embrittlement. Fracture toughness of the reactor vessel is discussed in Section 5.3.2 of the SSAR.

As discussed in SSAR Sections 1.9.4.2.3 and 5.4.7, one of the safety-related functions of the RNS is to provide LTOP for the RCS during refueling, startup, and shutdown operations. The AP600 design contains a relief valve to provide this safety-related LTOP function. It is designed to limit the RCS pressure within the limits specified in Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50. In accordance with SSAR Table 3.2-3 and Figure 5.4-7, this relief valve and its associated piping are classified as safety-related ASME Class 2, seismic Category 1 components. SSAR Tables 3.2-1 and 3.9-16 identify these components as being subjected to inservice inspection and testing in accordance with the requirements of the ASME Code, Section XI.

On the basis of the above information, the staff concluded in the DSER that the AP600 reactor vessel has been adequately designed for LTOP; however, in the discussion of this issue in Revision 0 of SSAR Section 1.9.4.2.3, Westinghouse incorrectly referred to GL 90-016 instead of GL 90-06. The staff also concluded that Westinghouse should revise Section 1.9.4.2.3 to include appropriate LTOP TS addressing GL 90-06 in the AP600 design TSs instead of GL 90-016. This was DSER Open Item 20.3-12. Revision 9 of the SSAR included the correct

reference to GL 90-06. In addition, WCAP-13559, Revision 1, "Operational Assessment Report" contains a commitment to GL 90-06, which references SSAR Section 1.9.4.2.3. Therefore, DSER Open Item 20.3-12 is closed.

GL 90-06 addressed the establishment of additional guidance for RCS LTOP to ensure reactor vessel and RCS integrity beyond that identified in the resolution to Issue A-26, which is discussed in Section 20.2 of this report. As a resolution for Issue 94, GL 90-06 requires a revision to plant TSs for capability of the LTOP system. Other possible solutions identified in GL 90-06 include hardware modifications including the use of the RHR system relief valves, and requiring the LTOP system to be fully safety-related.

GL 90-06 states that the LTOP availability should be ensured by limiting the allowable outage time to 24 hours for a single LTOP channel while operating in Modes 5 and 6. In response to this open item, the AP600 Technical Specification LCO 3.4.15 for the LTOP system requires that, with the accumulators isolated, either the RNS suction relief valve or the RCS depressurized with an open RCS vent of greater than or equal to 34.8 cm^2 (5.4 in^2) be operable. If the RNS suction relief valve is inoperable, Action Item C of LCO 3.4.15 requires either that the relief valve be restored to operable status or that the RCS be depressurized and the RCS vent be established within 8 hours. Westinghouse states in response to RAI 440.81 that with the RCS depressurized and the vent path opened, the Appendix G pressure limits for the AP600 reactor vessel will not be exceeded for the design-basis LTOP events. The staff concludes that the AP600 TS is consistent with GL 90-06, and is acceptable.

Therefore, Issue 94 is resolved for the AP600 design.

Issue 103: Design for Probable Maximum Precipitation

As discussed in NUREG-0933, Issue 103 addressed the acceptable methodology for determining the design flood level for a particular plant site. The use of the most recent National Oceanic and Atmospheric Administration (NOAA) procedures for determining the probable maximum precipitation for a site was questioned after a licensee disputed the use of two of NOAA's hydrometeorological reports. The issue was resolved with the revisions to SRP Sections 2.4.2 and 2.4.3 in 1989 to incorporate the probable maximum precipitation procedures and criteria contained in the latest National Weather Service publications. This was documented in the *Federal Register* Notice 54 *FR* 31268 on July 27, 1989, and GL 89-22, "Potential for Increased Roof and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service," dated October 19, 1989.

In SSAR Section 1.9.4.2.3, Westinghouse states that the probable maximum precipitation is a site-related parameter and the AP600 design is designed for the air temperatures, humidity, precipitation, snow, wind, and tornado conditions specified in SSAR Table 2.0-1. Westinghouse states that the COL applicant has the responsibility to demonstrate that the specific site parameters are within the limits specified for the standard AP600 design. The specific site is acceptable if the site characteristics are within the AP600 plant site design parameters in SSAR Table 2.0-1. For cases where a specific site characteristic is outside the Table 2.0-1 parameters, Westinghouse states that the COL applicant must demonstrate that the site characteristic does not exceed the capability of the AP600 design. Additional information on the site interface parameters is provided in SSAR Chapter 2.

The COL applicant must use site-specific environmental data for determining the probable maximum precipitation in accordance with SRP Sections 2.4.2, "Floods," and 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers." This is to ensure the maximum flood level for the AP600 design specified in SSAR Table 2.0-1 shall not be exceeded by the site-specific flood level. This is discussed in Section 2.4 of this report.

Therefore, Issue 103 is resolved for the AP600 design.

Issue 105: Interfacing Systems LOCA at BWRs

Issue 105, in NUREG-0933, was limited to pressure isolation valves (PIVs) in BWRs and was resolved by requiring leak-testing of the check valves that isolate low-pressure systems that are connected at the RCS outside of containment. It is related to Issue 96, which addressed PIVs between the RCS and RHR systems in PWRs. As stated in NUREG-0933, the staff issued Information Notice (IN) 92-36, "Intersystem LOCA Outside Containment," dated May 7, 1992, on this subject. The individual plant examinations required by the staff on operating plants included analyses of these sequences. This issue was resolved without any new requirements for operating plants.

For advanced reactor design, the staff position regarding intersystem LOCA protection, as stated in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirement," as well as SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," is that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure. The phrase "to the extent practicable" is a recognition that all systems must eventually interface with atmosphere and that it would be difficult or prohibitively expensive to design certain large tanks and heat exchangers with the URS equal to full RCS pressure. Piping runs should be designed to meet the URS criteria, as should all associated flanges, connectors, and packings, including valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines. The designer should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

In Section 3.9.3.1 of this report, the staff discusses its evaluation that establishes the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure, should the low-pressure system be subject to full RCS pressure. The subsection within Section 3.9.3.1, "AP600 Design Criteria for ISLOCA," contains the design criteria proposed by Westinghouse for the low-pressure portion of the RNS. On the basis of this evaluation, the staff concludes that these criteria are acceptable to assure that the low-pressure side of any applicable system has been designed to meet the full RCS URS criteria.

For all interfacing systems and components that do not meet the full RCS URS criteria, the applicant must justify why it is not practicable to reduce the pressure challenge any further, and also provide compensating isolation capability. For example, applicants should demonstrate for each interface that the degree and quality of isolation or reduced severity of the potential pressure challenges compensate for and justify the safety of the low-pressure interfacing

systems or components. The adequacy of pressure relief and the piping of relief back to primary containment are possible considerations. As identified in SECY-90-016, each of these interfacing systems that has not been designed to withstand full RCS pressure must also include the following protection measures:

(1) the capability for leak testing of the pressure isolation valves

(2) valve position indication that is available in the control room when isolation valve operators are de-energized

(3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure system and both isolation valves are not closed

Westinghouse initially stated that information on this issue was in Section 1.9.1.5 of the SSAR and Section 5.4.7 of the SSAR for information on the RNS. In the DSER, the staff identified Open Item 20.3-13, stating no Section 1.9.1.5 existed in the SSAR. In fact "Section 1.9.1.5" was a typographic error, and the Intersystem LOCA is addressed in Section 1.9.5.1.7 of the SSAR. Therefore, Open Item 20.3-13 is closed.

SSAR Section 1.9.5.1.7, "Intersystem LOCA," provides Westinghouse's response regarding compliance of the AP600 design with the staff position on an intersystem loss-of-coolant-accident (ISLOCA). An ISLOCA evaluation of the RNS, chemical and volume control system (CVS), primary sampling system (PSS), and liquid radwaste processing system (WLS) was provided in Westinghouse's responses to RAIs 440.30, 440.132, and 440.158. The staff evaluation of these RAI responses found the ISLOCA issue was not satisfactorily addressed, and identified Open Item 20.3-14, stating that, for the systems discussed, there was insufficient discussion regarding the design pressure of the components, such as flanges, connectors, packings, valve stem seals, pump seals, valve bonnets, and the drain and venting lines. Subsequently, Westinghouse submitted topical report WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated July 1995, which provides a systematic evaluation of the design responses of various systems interfacing the RCS of the ISLOCA challenges. The staff reviewed WCAP-14425 as discussed below, and Open Item 20.3-14 is closed.

The systematic evaluation process includes (1) a review of the AP600 piping and instrumentation diagrams to identify these primary interfacing systems or subsystems directly interfacing with the RCS, and the secondary interfacing systems or subsystems interfacing with the primary interfacing systems, and (2) identification of primary and secondary systems and subsystems having a URS less than the RCS pressure. For those systems or subsystems not meeting the criterion of the URS greater than the RCS pressure, a design evaluation is made considering whether it is inside containment, whether it meets the three criteria specified in SECY-90-016, and whether it includes other design features specific to them that prevents an ISLOCA to the extent practicable. The report also provides the reasons why it is not practical to design large, low-design pressure tanks and tank structures that are vented to the atmosphere to the high pressure criterion. Interfacing systems or subsystems that connect directly to an atmospheric tank are excluded from further ISLOCA consideration. This is limited to the piping connected directly to the atmospheric tank, up to the first isolation valve other than a locked-open, manual isolation valve. The staff evaluation of these systems follows.

Normal Residual Heat Removal System

The portions of the RNS from the RCS to the containment isolation valves (CIVs) outside containment are designed to the RCS operating pressure, and the portions downstream of the CIV and upstream of the discharge line CIV are designed so that its URS is not less than the RCS operating pressure. The mechanical shaft seal of the RNS pump, which has a design pressure of 6,200 kPa (900 psig), is the only portion of the RNS having a URS lower than the RCS pressure. Section 3.1.3.2 of WCAP-14425 discusses the difficulties of designing the RNS pump seal to full RCS operating pressure. A fundamental problem is that any type of seal that can withstand RCS pressure will likely have abnormally fast wear of the seal faces during normal plant operation at low seal pressure. This increased wear at normal plant operating conditions could well prevent the seal from maintaining the pressure boundary if ever exposed to the full RCS pressure. Use of high pressure seals will also require more frequent maintenance during normal operation. Therefore, it is impractical to design a seal that would maintain the RCS pressure boundary with no leakage, and also operate satisfactorily at low-pressure conditions. The AP600 RNS pump mechanical seal is designed to minimize the amount of leakage if exposed to full RCS pressure. An Idaho National Engineering Laboratory (INEL) study on the Davis Besse Nuclear Power Station decay heat removal pump seal, with a design pressure of 3,100 kPa (450 psig), found that the rotating seal would maintain its structural integrity at pressures in excess of 17,200 kPa (2,500 psi), and the mechanical seals can withstand a pressure of 8,300 - 8,600 kPa (1,200 -1250 psi) without leaking. The AP600 RNS pump mechanical seal is similar to the Davis Besse DHR pumps, but its design pressure is twice as high. The AP600 RNS pump also has a disaster bushing that limits the leakage from the pump to within the capabilities of the normal makeup system in case of catastrophic mechanical seal failure. Leakage can be controlled with the seal leakoff line routed to a floor drain that is routed to the auxiliary building sump. This is more favorable than a seal specially designed for full RCS pressure at the expense of normal-condition reliability.

In SSAR Section 5.4.7.2.2, Westinghouse discusses the AP600 design features in the RNS specifically aimed at reducing the likelihood of an intersystem LOCA. On the suction side, there is a normally closed, motor-operated isolation valve in the common suction line outside containment, and two normally closed, motor-operated isolation valves in each parallel suction line inside the containment. There is a relief valve with a set pressure of 3,880 kPa (563 psig) connected to the RNS pump suction line inside containment, which is designed to provide low-temperature overpressure protection of the RCS and will reduce the risk of overpressurizing the RNS. On the discharge side, the common discharge line has a safety-related containment isolation check valve inside containment and a safety-related motor-operated isolation valve outside containment. The MOVs inside the containment are interlocked to prevent them from opening when the RCS pressure is above the RNS operating pressure of 3,100 kPa (450 psig). The power to these isolation valves is administratively blocked at the valve motor control centers to prevent inadvertent opening. In addition, the discharge header contains a relief valve, which discharges to the WLS effluent holdup tanks, to prevent overpressure in the RNS pump discharge line that could occur if the three check valves and the motor-operated CIV leaked back to the low pressure portions of the RNS.

Also, the RNS design includes an instrumentation channel that indicates pressure in each RNS pump suction line, and a high pressure alarm is provided in the MCR to alert the operator to a condition of rising RCS pressure that could eventually exceed the RNS design pressure. The

motor-operated pressure isolation valves also have remote position indications in the MCR. In addition, these pressure isolation valves are specified in SSAR Table 3.9-18 to be subject to technical specification LCO 3.4.16, which requires the leakage of each RCS PIV to be within limits with leak testing in accordance with surveillance requirement 3.4.16.1. The staff concludes that the RNS design meets the requirements of SECY-90-016.

Chemical and Volume Control System/ Makeup Systems

SSAR Section 9.3.6 provides a detailed description of the design, functions, and operations of the chemical and volume control system (CVS). The purification flow path of the CVS is a high-pressure closed-loop design, which is entirely within the containment. The potential contributors to an ISLOCA are the portions of the CVS located outside the containment, i.e., the letdown line to the liquid radwaste system, and the makeup system.

The CVS makeup pumps operate intermittently to make up for RCS leakage. The pumps start and stop automatically when the pressurizer level reaches the bottom and the top of the normal level band, respectively. The makeup pumps take suction from either the boric acid tank, or the demineralized water storage tank (DWST), and inject into the CVS purification loop return stream. The makeup pumps can also take suction from the waste holdup tanks or the spent fuel pool. The makeup line from the makeup pump discharge to the RCS has a design pressure greater than or equal to the RCS design pressure. However, the pump suction line piping and associated components have a design pressure of 1,030 kPa (150 psig) with the URS less than the RCS operating pressure.

In Section 3.3.3 of the WCAP-14425, Westinghouse contends that it is not practicable to design the low-pressure portions of the makeup suction piping to higher design pressure. It is not practicable to have a high design pressure for large tanks such as the boric acid tank, which are vented to the atmosphere, as well as the piping directly connected to these atmospheric tanks up to the first isolation valve. The suction lines each contain a check valve that separates the suction piping from a large atmospheric tank. These check valves are designed to open on low differential pressure, and have a high tendency to leak. The suction lines contain relief valves that protect the low-pressure portions of the piping from overpressure in the event of leaking check valves in the discharge line or thermal expansion in case of a loss of miniflow cooling. The relief valves direct any leakage from the discharge line check valves to the WLS effluent holdup tanks (EHTs), which are designed to handle radioactive fluids, and their level is monitored by remote instrumentation.

The passage of the high pressure reactor coolant to the CVS makeup suction is possible only when the makeup pumps are not running, and only if failures or leakage of multiple check valves on the makeup pump discharge side occurs. There is a high-pressure alarm in the pump suction line to alert the operator of overpressurization. In the event of a suction-side overpressurization, the makeup pumps can be operated to terminate overpressurizing the suction piping. If the makeup pumps did not start, the makeup line containment isolation valves would automatically close to terminate the ISLOCA. In addition, the purification loop inlet isolation valves would also be closed on a safeguards actuation signal. These multiple, safety-related isolation valves prevent an ISLOCA in the makeup suction line. As specified in Table 3.9-16 of the SSAR, the purification inlet stop valves, and the purification return line stop valve and check valve are subject to leak testing. These stop valves are provided with position indication in the control room. In addition, the makeup line CIVs also have the capability for

leak testing, and are provided with valve position indication in the control room at all times. The staff finds that protection measures meet the intent of SECY-90-016 ISLOCA position.

CVS Letdown/Liquid Radwaste System

The CVS letdown line connects to the high-pressure purification loop inside containment. Immediately downstream of this connection is a high-pressure, multistage letdown orifice, which reduces pressure in the letdown line from the RCS operating pressure to below the design pressure of the low-pressure portion of the letdown line. Around the letdown orifice is a bypass line containing a locked-closed manual isolation valve that is opened only at shutdown when the RCS is depressurized to provide sufficient letdown flow when required. The letdown line is then equipped with two safety-related, normally-closed, fail-closed CIVs where it penetrates containment to the WLS degasifier package and EHTs. The letdown line down to and including the outboard CIV has a design pressure of 17,130 kPa (2485 psig). Downstream of the outboard CIV, the WLS letdown line has a design pressure of 1,030 kPa (150 psig), and therefore does not meet the RCS URS criteria.

In Section 3.2.3 of WCAP-14425, Westinghouse contends that it is not practicable to design the low-pressure portions of the letdown line to a higher design pressure. The WLS EHTs are large atmospheric tanks, and are therefore not practicable for higher design pressure. Nor is the letdown line, which is routed to the degasifier package or the EHTs, and the degasifier package, which discharges directly to the WLS EHTs. The CVS letdown system has the following features to meet the ISLOCA criteria:

(1) the pressure drop across the CVS letdown orifice protects the WLS from overpressurization during letdown operations by reducing the pressure in the WLS

(2) in case of an inadvertent valve closure in the WLS during letdown, a relief valve, which discharges directly to the EHT, is provided that would protect the WLS from overpressurization

(3) due to the letdown orifice, a break in the WLS during letdown from the CVS would result in an RCS leak that is within the capability of the normal makeup system

(4) if an ISLOCA should occur, it would be terminated by automatic isolation of the two purification loop isolation valves and two letdown isolation valves on low pressurizer level or a safeguards actuation signal

(5) the letdown line CIVs have the capability for leak testing and have valve position indication in the control room at all time

(6) the WLS degasifier column contains a high-pressure alarm that would warn the control room operators that the WLS pressure was approaching the design pressure

In addition, as discussed previously, the purification inlet and return stop valves and check valve are subject to leak testing. The staff finds the CVS letdown piping meets the SECY-90-016 ISLOCA position.

Primary Sampling System

The primary sampling system (PSS) collects representative samples of fluids from the RCS and associated auxiliary system process streams, and the containment atmosphere for analysis by the plant operating staff. Section 3.4 of WCAP-14425 provides an ISLOCA evaluation of the PSS. The PSS lines consist of small, 0.64 cm (0.25 in) pipes. The whole PSS is designed to full RCS pressure and temperature, with the exception of the eductor water storage tank (EWST) and its drainage and level indication lines, eductor supply pump seal, and demineralized water supply line. These low-pressure portions have design pressures with URS below the RCS operating pressure. The applicant contends that it is not practical to design the low-pressure portion of the PSS to a higher design pressure because they are at atmospheric pressure and connect to the low-pressure demineralized water transfer and storage system (DWS). Designing the EWST to high pressure to meet ISLOCA criteria would require the DWS to be designed for high pressure, which is not practicable.

The PSS is connected to the RCS through the local sample points in the RCS hot legs, pressurizer vapor and liquid spaces, and the core makeup tanks. Each of these sampling connection lines contains a flow-restricting orifice that limits the flow from the RCS in the event of a sample line break, and also reduces the pressure in the sampling lines during sampling operations. Each sampling line also contains a normally closed isolation valve before being connected to a common header. The common header then penetrates the containment with two normally closed CIVs, which are also PIVs and will be isolated on safeguards signal if open for sampling operation. The sampling line then connects to a sample cooler and the sample bottles. In addition, one of the two lines connected to the low-pressure portion of the PSS contains two check valves, and the other contains one check valve and one normally closed isolation valve. In the event that these valves leaked, the leak would not overpressurize the low-pressure portions of the system, but would flow directly to the EWST. In the unlikely event of a gross failure of the high-pressure check valves, the maximum flow rate from the RCS would be within the capability of the normal makeup system. The water level in the EWST is monitored, and a high level alarm in this tank will alert the operator to a potential leak into the tank from the PSS sampling lines. The operator will then be able to isolate the leak by closing the CIVs. The CIVs have remote position indication in the control room and are subject to the CIV leakage test. Also, the leakage from these CIVs through the 0.64 cm (0.25 in) pipes would be small. Therefore, the PSS design meets the intent of the ISLOCA criteria.

Solid Radwaste System

The solid radwaste system (WSS), which provides storage facilities for both wet and dry solid wastes prior to and subsequent to processing and packaging, is connected to the high-pressure CVS demineralizers to facilitate transfer of spent resin from the CVS demineralizers to the spent resin storage tanks (SRSTs). The spent resin header connects to each of the three CVS demineralizers with an individual, normally closed isolation valve, and then penetrates containment with two normally closed locked-closed CIVs to the SRSTs outside. A manual valve is placed downstream of the outboard CIV to isolate the downstream piping to facilitate CIV leak testing. The portion of piping downstream of the manual isolation valve is a low-pressure design with a URS below the RCS operating pressure. Section 3.5.2 of WCAP-14425 asserts that it is not practical or necessary to design the WSS to a higher design pressure because the system contains many low-pressure components such as the SRST and resin transfer and mixing pumps.

The WSS spent resin line is normally isolated by locked-closed manual CIVs, which are administratively controlled, have position indications in the control room, and are leak tested in accordance with the inservice testing plan of SSAR Section 3.9.6. The CVS demineralizers are inside containment and normally circulate reactor coolant at RCS pressure. As such, resin transfer operations are conducted only during refueling operations when the RCS is fully depressurized. During normal power operation, the only pathway to the low-pressure portion of the WSS is for all three closed isolation valves to fail. Should that extremely unlikely event happen, the recirculation loop isolation valves can be closed to isolate the purification loop and the WSS from the RCS. In addition, downstream of the inboard CIV in the resin transfer line, there is a relief valve which discharges to the WLS containment sump inside containment. Therefore, the WSS spent resin lines are not required to be designed to a higher design pressure.

Demineralized Water Transfer and Storage System

The demineralized water transfer and storage system (DWS) receives water from the demineralized water treatment system, and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. The design and functional details of the DWS are provided in Section 9.2.4 of the SSAR. The demineralized water transfer pumps take suction from the DWST and supply water through a catalytic oxygen reduction unit to the demineralized water distribution header. From this header, demineralized water is supplied to various systems in the plant. One DWS supply line penetrates containment to a supply header inside containment, which serves as the DWS interface with the PSS and the CVS demineralizers. The DWS provides demineralized water to the PSS to flush the PSS lines prior to RCS sampling, and to the CVS demineralizers to sluice resin to the WSS.

The DWS is a low-pressure system design with a URS below RCS operating pressure. However, the only possible overpressurization pathways from the RCS are the connections to the PSS and the CVS demineralizers inside containment. Overpressurization of the DWS can only occur if there are multiple failures and misalignments of isolation valves and check valves in the high-pressure systems. A relief valve has been added to the DWS header inside containment to preclude the possibility of overpressurizing the DWS. In addition, an overpressurization of the DWS would most likely result in the rupture of the DWS header inside containment, and therefore is not a concern for ISLOCA.

Summary and Conclusion

The staff concludes that the AP600 design is consistent with the staff position discussed in SECY-90-016 regarding ISLOCA. Therefore, Issue 105 is resolved for the AP600 design.

Issue 106: Piping and Use of Highly Combustible Gases in Vital Areas

NUREG-0933, Issue 106, addressed the release of combustible gases from leaks or pipe breaks resulting in combustible gas accumulation in buildings containing safety-related equipment. NUREG-1364, "Regulatory Analysis for the Resolution of Generic Safety Issue 106: Piping and the Use of Highly Combustible Gases in Vital Areas," specifically addressed GSI 106, and provided alternatives for prevention, detection, and protection against

hazards associated with the release of combustible gases used, stored and piped through safety related areas and areas that expose safety-related equipment.

As discussed in NUREG-0933 and NUREG-1364, except for hydrogen, most combustible gases are used in limited quantities and for relatively short periods of time. Hydrogen is stored in high-pressure storage vessels and is supplied to various systems in the auxiliary building through small-diameter piping. A leak or break in this piping could result in an explosive mixture of air and hydrogen, posing a potential loss of safety-related equipment.

In SSAR Section 1.9.4.2.3, Issue 106, Westinghouse specifies that the AP600 design uses small amounts of combustible gases for normal plant operations. The plant gas system is discussed in Section 9.3.2 of the SSAR. Most such gases are stated to be used in limited quantities and associated with plant functions or activities that do not jeopardize safety-related equipment. These gases are found in areas of the plant that are removed from the nuclear island except the hydrogen supply line to the CVS inside containment, which is the only system on the island that uses hydrogen gas.

Hydrogen gas is supplied to the CVS from a single hydrogen bottle. The release of the contents of an entire bottle of hydrogen in the most limiting building volumes (both inside containment and in the auxiliary building) would not result in a volume percent of hydrogen large enough to reach a detonable level. Section 1.9.4.2.3 of the SSAR also specifies that the CVS hydrogen supply piping is routed through the turbine building and into the auxiliary building and then into containment. The hydrogen supply line is routed through the piping/valve room on Elevation 100' of the auxiliary building. The piping valve penetration room in the auxiliary building on Elevation 100' is designed as a 3-hour fire zone. Section 9.3.2 of the SSAR specifies that the hydrogen gas portion of the plant system is a packaged system consisting of a liquid hydrogen storage tank and vaporizer to supply hydrogen gas to the main generator for generator cooling, and to the demineralized water transfer and storage system to support removal of dissolved oxygen, and to other miscellaneous services. The hydrogen supply package system is located outdoors at the hydrogen storage tank area. The turbine building does not house any safety-related systems or equipment. The containment has hydrogen sensors to detect hydrogen leaks. The containment hydrogen concentration monitoring subsystem is designed as Class 1E and seismic Category 1 (SSAR Section 6.2.4.1).

The BTP 9.5-1, Section C.5.d, "Control of Combustibles," specified that care should be taken to locate high-pressure storage containers with the long axis parallel to building walls. In addition, BTP 9.5-1 specified that hydrogen lines in safety-related areas be either designed to seismic Class 1 requirements, or sleeved such that the outer pipe is directly vented to the outside, or should be equipped with excess flow valves so that in case of a line break, the hydrogen concentration in the affected area will not exceed 2 percent. Westinghouse specified in Table 9.5-1, Section C.5.d of the SSAR that the AP600 design complies with the BTP 9.5.1.

In addition, in Section 9.5.1 of the SSAR, the applicant references National Fire Protection Association (NFPA), Standard 50A, "Gaseous Hydrogen Systems at Consumers Sites," 1994 Edition. Table 9.5.1-3 identifies no exceptions to the referenced NFPA standard. Therefore, on the basis of compliance with the guidance provided in BTP 9.5.1 and the applicable NFPA Standard, GSI 106 is considered resolved.

Issue 113: Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

As discussed in NUREG-0933, Issue 113 addressed the staff's concerns in 1985 that there were no requirements for dynamic qualification testing or surveillance testing of large-bore hydraulic snubbers (LBHSs) (i.e., > 345 MPa (50 kips) load rating). The safety concern was the integrity of the SG lower support structures when subjected to a seismic event; however, this issue was applicable to all components, structures, and supports that rely on this type of snubbers for restraint from seismic loads and other dynamic loads, such as high-energy line breaks and water hammers.

LBHSs are active mechanical devices used to restrain safety-related piping and equipment during seismic or other dynamic events, yet also allow sufficient piping component flexibility to accommodate system expansion and contraction from such thermal transients as normal plant heatups and cooldowns. Dynamic testing and periodic functional testing are important to verify that the LBHSs are properly designed and maintained for the life of the plant. Issue 113 was resolved with no new requirements, although it was recommended that a regulatory guide be developed for future plants.

In SSAR, Section 1.9.4.2.3, Issue 113, Westinghouse states that the AP600 plant uses significantly fewer hydraulic snubbers than do current operating plants. It further states that, in addition to the recommendations in the NRC draft regulatory guide, SC-708-4, "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety," testing requirements have been established in ANSI/ASME OM Code-1990, "Code for Operation and Maintenance of Nuclear Power Plants." Because ANSI/ASME OM, Part 4, is referenced as a requirement for inservice testing of snubbers in ASME Section XI, IWF-5000, this is an acceptable commitment for periodic functional testing of LBHSs, and is in accordance with applicable portions of 10 CFR 50.55a(f)(4). However, DSER Open Item 20.3-15 requested additional commitments relative to dynamic qualification testing of LBHSs. The staff requested similar information as part of DSER Open Item 3.9.3.3-1. Revision 11 to SSAR Section 3.9.3.4.3 added a commitment to include dynamic testing as a part of the production operability tests for all snubbers. As discussed in Section 3.9.3.3 of this report, this was an acceptable commitment, and Open Item 3.9.3.3-1 was closed. Therefore, on the basis of the above discussions relative to periodic functional testing and dynamic qualification testing, Open Item 20.3-15 is also closed, and Issue 113 is considered resolved for the AP600 design.

Issue 120: On-Line Testability of Protection Systems

As discussed in NUREG-0933, Issue 120 addressed requirements for at-power testing of safety system components without impairing plant operation. The staff raised this issue because it was found in the review of several plant TS in 1985 that some older plants did not provide as complete a degree of on-line testing as other plants then undergoing staff review. The requirement for on-line testing of protection systems is in GDC 21. These requirements apply to both the RPS and engineered safety features actuation system (ESFAS). A protection system with two-out-of-four (2/4) logic that can operate with one channel in bypass, and the remaining three channels in a two-out-of-three (2/3) logic configuration meets this requirement. This issue was resolved with no new requirements.

Guidance for this issue is provided in RG 1.22, "Periodic Testing of Protection System Actuation Functions," RG 1.118, "Periodic Testing of Electric Power and Protection Systems," and IEEE Standard 338. Conformance to these documents ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is at power without adversely affecting plant operation.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because this issue had been prioritized as low or drop, or was not prioritized. The staff informed Westinghouse that this issue incorporates requirements of GDC 21 and, therefore, addresses an important safety function. In DSER Open Item 20.3-16 the staff requested that Westinghouse address this issue for the AP600 design. Although, the Westinghouse response in Section 1.9.4.2.3 of the SSAR still contains a statement that this item does not apply to the AP600, Westinghouse further stated that conformance with the applicable guidance of GDC 21 is found in SSAR Section 7.1. The staff finds that Section 7.1 of the SSAR adequately addresses compliance with GDC 21 and related aspects in this issue for the AP600 design. Therefore, DSER Open Item 20.3-16 is closed and Issue 120 is resolved for the AP600 design.

Issue 121: Hydrogen Control for Large, Dry PWR Containments

Staff studies showed that hydrogen combustion was not a significant threat to the dry containments evaluated, and the potential for global detonation was considered to be very unlikely for large, dry containments, such as that provided for the AP600; these findings were documented in NUREG-1150. Issue 121 was resolved for existing PWR dry containments on the basis of the NUREG-1150 findings of no problems on hydrogen control for the two PWR plants investigated, the individual plant-specific examinations for vulnerabilities to hydrogen risk, and the long-term implementation of an accident management program. No new regulatory guidance on hydrogen control, such as the need for a hydrogen control system, was required for existing PWR dry containments.

In spite of this conclusion for existing plants, the AP600 has been provided with a system for hydrogen control to meet the hydrogen control requirements of 10 CFR 50.34(f). The ability of the AP600 design to comply with the hydrogen control requirements of 10 CFR 50.34(f) is documented in Section 6.2.4 of the SSAR. The staff's evaluation of the ability of the AP600 design to comply with the hydrogen control requirements of 10 CFR 50.34(f) are documented in Section 6.2.5, "Combustible Gas Control Inside Containment," of this report.

SECY-93-087 indicated that the staff would evaluate the ALWR vendor's identification of equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Global and local hydrogen deflagrations are two of the conditions that are to be considered. In SECY-93-087, the staff recommended that the Commission approve the staff's position that design features provided only for severe-accident mitigation need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. In the SRM dated July 21, 1993, the Commission approved the staff's position. Westinghouse addressed equipment survivability in Appendix D to the AP600 PRA. The staff's evaluation of Appendix D is documented in Section 19.2.3.3.7 of this report.

Westinghouse did not address the preoperational and operational testing of the systems discussed, and the instructions in the emergency operating procedures (EOPs) for the operators. The staff was confused about the reference to analyses and the PRA evaluation report as to where the documentation was for the statements made. The staff asked Westinghouse to address these concerns for the AP600 design. This was Open Item 20.3-17.

The AP600 has been provided with a system for hydrogen control that meets the requirements of 10 CFR 50.34(f) as evaluated in Sections 6.2.5 and 19.2.3.3.7 of this report. Therefore, Open Item 20.3-17 is closed and Issue 121 is resolved for the AP600 design.

Issue 122.2: Initiating Feed and Bleed

As discussed in NUREG-0933, Issue 122.2 investigated the findings of the NRC inspection in 1985 of the loss-of-feedwater event at Davis-Besse on June 9, 1985. The issue dealt with the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following the loss of the standby steam generator heat sink (i.e., loss of feedwater). In an analysis of the loss-of-feedwater event, the staff found that operators were hesitant to initiate feed-and-bleed operations, and that the control room instrumentation was inadequate to alert operators to the need to initiate feed and bleed. A loss-of-feedwater in combination with a failure to diagnose and take corrective actions (i.e., initiate feed and bleed) would result in a loss of core cooling.

The staff completed its review of Westinghouse-provided information relating to the feed-and-bleed emergency guidelines AFR -H.1, "AP600 Response to Loss of Heat Sink," and concluded that the feed-and-bleed emergency guidelines are acceptable. Therefore, Open Item 20.3-18 is closed and Issue 122.2 is resolved for the AP600 design.

Issue 124: Auxiliary Feedwater System Reliability

Following the loss-of-feedwater event at Davis-Besse plant in 1985, Generic Issue 124 from NUREG-0933 addressed increasing reliability of the auxiliary or emergency feedwater system to 1E-04 unavailability/demand. In 1985, operating experience as well as staff and industry studies indicated that these systems failed at a high rate. The function of the auxiliary feedwater system (AFWS) in the majority of operating plants is to supply feedwater water to the secondary side of the SGs during system fill, normal plant heatup, hot standby, and cold shutdown conditions. The AFWS also functions following loss of normal feedwater flow, including loss resulting from an offsite power failure; additionally, it supplies feedwater to the SGs following accidents such as a main feedwater line break or a main steamline break. Therefore, the reliability of the AFWS is important to plant safety.

The NRC investigation of the Davis-Besse event indicated that the potential inability to remove decay heat from the reactor core was the result of the questionable reliability of the EFWS caused by any or all of the following:

- loss of all EFW as a result of common-mode failure of the pump discharge isolation valves to open

- excessive delay in recovering EFW because of a difficulty in restarting the pump steam-driven turbines once they tripped

- interruption of EFW flow because of failures in steamline break and feedwater line break accident mitigation features

In addition, the investigation of the event indicated that (1) a two-train system with a steam turbine-driven EFW pump may not be able to achieve the desired level of reliability and (2) provisions to automatically isolate EFW from a SG affected by a main steamline or feedwater line break may tend to increase the risk that adequate DHR is not available, rather than decrease it.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that this issue is not applicable to the AP600 design because the design does not have a safety-related auxiliary feedwater system. The passive core cooling system will provide the safety-related function of cooling the RCS in the event of loss of feedwater to the SGs. The startup feedwater system (SUFS), which has no safety-related function beyond containment isolation, provides the SGs with feedwater during plant conditions of startup, hot standby, cooldown, and when the main feedwater pumps are not available.

This issue required the use of a plant PRA to ensure the reliability/availability of the AFWS or EFWS by having a minimum reliability of 1E-04 unavailability/demand. Westinghouse contended that this issue is not applicable to the AP600 design because the SUFS has no such safety-related function.

In SECY-94-084, the staff established a process for RTNSS to identify the risk significance of the non-safety-related active systems. The staff determined that the SUFS should be subject to this RTNSS evaluation. Therefore, the SUFS reliability was designated as DSER Open Item 20.3-19.

Westinghouse provided the RTNSS evaluation in WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process." The SUFS was not identified as risk-important by the RTNSS process. No further analysis is required to resolve Issue 124.

The staff reviewed the AP600 design against Issue 124 and finds that this generic issue is not applicable to the SUFS because of design difference. The SUFS is not a safety-related system and does not have to perform the same safety function as the AFWS. Therefore, the staff concludes that DSER Open Item 20.3-19 is closed and Issue 124 is resolved for the AP600 design.

Issue 125.II.7: Reevaluate Provisions to Automatically Isolate Feedwater From Steam Generator During a Line Break

As discussed in NUREG-0933, Issue 125 addressed the long-term actions from NUREG-1154 and the Executive Director for Operations (EDO) memorandum dated August 5, 1985, on the loss-of-feedwater event at Davis-Besse on June 6, 1985. Issue 125.II.7 addressed the need for licensees to reassess the benefits of automatically isolating the EFWS after a break in the secondary side of the SG. For a typical PWR with automatic isolation (AI) of the EFW (AI-EFW), a low-SG-pressure signal causes closure of the main steam isolation valves (MSIVs)

and isolation of EFW from the faulted SG during a steamline break. AI-EFW minimizes blowdown from the SG secondary-side line break and limits primary system overcooling and the potential for return to criticality owing to positive moderator reactivity feedwater caused by overcooling of RCS inventory. If the EFW were not isolated, the peak containment pressure for secondary-side breaks would exceed that caused by a large break LOCA, the design-basis event for the containment.

However, AI-EFW has a disadvantage. If both channels of the controlling isolation logic system were to spontaneously actuate, the availability of EFW would be lost and the MSIVs would close. For the plants using turbine-driven main feedwater pumps, these pumps would be lost following the closure of the MSIVs and the loss of steam, and this loss would result in the loss of the secondary side heat sink. The capability to lock out the isolation logic is necessary to preclude this event.

The staff determined (as stated in NUREG-0933) that, for a new plant, the design does not need to include automatic isolation of EFW following a steamline break or feedwater line break, provided that the results of the analysis of the secondary-side line break and the containment analysis meet the criteria in the appropriate SRP section of NUREG-0800.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because this issue was resolved without new requirements for plants.

Although the startup feedwater system for the AP600 design is not a safety-related system, the staff requested that Westinghouse address Issue 125.II.7 and the question of automatic isolation of EFW for the AP600 design. This was Open Item 20.3-20.

In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the AP600 design. Therefore, Open Item 20.3-20 is closed and Issue 125.II.7 is resolved for the AP600 design.

Issue 128: Electrical Power Reliability

GSI 128 addresses the reliability of onsite electrical systems and encompasses GSIs 48, 49, and A-30. The staff reviewed Westinghouse's submittal and concluded that the AP600 design addresses Issue 48, "LCO for Class 1E Vital Instrument Buses in Operating Reactors"; Issue 49, "Interlocks and LCOs for Class 1E Tie breakers"; and Issue A-30, "Adequacy of Safety Related DC Power Supplies" as follows:

- Issue 48 – Westinghouse provided the limiting condition for operation (LCO) in the event of a loss of one or more Class 1E 120 Vac vital instrument buses and associated inverters. The staff finds this LCO acceptable.

- Issue 49 – The AP600 design does not include Class 1E tie breakers.

- Issue A-30 – The staff evaluated the Class 1E dc distribution system design for the aspects addressed by A-30 in Section 8.3.2.1 of this report and concluded that it is acceptable.

Therefore, Issue 128 is resolved for the AP600 design.

Issue 130: Essential Service Water Pump Failures at Multi-plant Sites

As discussed in NUREG-0933, Issue 130 addressed the vulnerability of Byron Unit 1 to core-melt sequences in the absence of the availability of Unit 2 (not yet operational). While Unit 2 was under construction, it was necessary to make a third service water pump available to Unit 1 via a cross-tie with one of the two Unit 2 essential service water (ESW) pumps. This issue raised concerns relative to multi-plant units that have only two ESW pumps per plant but have cross-tie capabilities. A limited survey of Westinghouse plants helped to identify the generic applicability of vulnerabilities of multi-plant configurations with only two ESW pumps per plant. In the multi-plant configurations identified (approximately 16 plants), all plants can share ESW pumps via a cross-tie between plants. Additional efforts to resolve this issue included (1) a limited survey of Westinghouse plants to determine the generic applicability of similar multi-plant configurations with two ESW pumps per plant and whether cross-tie capabilities existed, (2) a survey of Babcock and Wilcox, and ABB-CE plants to identify similar multi-plant configurations, and (3) a survey of single-unit plants to determine if similar ESW vulnerabilities existed.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that this issue is not applicable to the AP600 design because the plant design is for a single independent plant that does not share or cross-tie systems or components with another plant. In Section 3.1.1 of the SSAR, Westinghouse states that if more than one unit is built on the same site, none of the safety-related systems will be shared.

Therefore, Issue 130 is resolved for the AP600 design.

Issue 135: Steam Generator and Steamline Overfill

Westinghouse identified in Table 1.9-2, of its May 28, 1993, letter, that it considered Issue 135 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue 135 was initiated in 1986 to integrate various SG programs and related issues, including water hammer, eddy current testing, and steamline overfill consequences. Overfill is defined as water entering the main steamline caused by excessive feedwater flow resulting from control system failure or a SGTR. This issue was expected to provide a better understanding of SG and secondary-side integrity, including the effects of water hammer on secondary system components and piping, as well as the resultant radiological consequences. Because the staff concluded that SGTR and steamline overfill events are relatively low risks, this issue was resolved and no new requirements were established. This is documented in NUREG-0933 and NUREG/CR-4893, "Technical Findings Report for Generic Issue 135: Steam Generator and Steamline Overfill Issues," dated May 1991.

A subissue in Issue 135 was the improved eddy current testing of SG tubes. The staff deferred this subissue to the development of a revision to RG 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

In SSAR Section 1.9.4.2.3, Westinghouse states the following regarding the four tasks that comprise this issue, as discussed in NUREG-0933:

- Task 1 on code and regulatory requirements – SSAR Appendix 1A, which discusses the level of conformance with RG 1.83, states that the AP600 design essentially conforms with the regulatory guidance except where state-of-the-art advances have enhanced inservice inspection techniques. Specifically, as stated in Section 5.4.2.5 of the SSAR, the SGs permit access to the tubes for inspection, repair, and plugging per RG 1.83. The AP600 SGs include features to enhance robotics inspection of tubes without manned entry. The AP600 design will conform with RG 1.83, except in cases where state-of-the-art advances will provide better inspection techniques. As discussed in Sections 5.2.4 and 5.4.2 of this report, the development of the SG tube preservice inspection program and inservice inspection program is the responsibility of the COL applicant. Steam generator tube integrity is verified in accordance with this surveillance program. The program is plant-specific and will be reviewed by the staff individually for each license application referencing the AP600 design certification against the staff's regulatory criteria in place at the time of its review. In addition, the AP600 design will adhere to the water chemistry guidelines provided in EPRI NP-5960 and EPRI NP-6239 reports. See Section 5.4.2.1 of this report for additional information concerning AP600 steam generators.

- Task 2 on SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure" – In Section 15.6.3.1.4 of the SSAR, there is a discussion of anticipated operator recovery actions and the effects of those actions in the mitigation of a SGTR event. The automatic SG overfill protection is described in Section 15.6.3.2 of the SSAR and the control logic is described in Section 7.2 of the SSAR.

- Task 3 on several generic issues – A compilation of the generic issues are addressed by the following SSAR sections:

  - radiological consequences are discussed in Section 15.6.3 of the SSAR

  - SGTR design basis is discussed in Section 15.6.3 of the SSAR

  - supplemental tube inspections are discussed in Section 5.4.2.5 of the SSAR and Appendix 1A

  - denting criteria are discussed in Section 5.4.2.4.3 of the SSAR

  - safety-related display information is discussed in Section 7.5 of the SSAR

  - reactor coolant pump trip is discussed in Section 7.3.1.1.3.3 of the SSAR

- control room design and design process are discussed in Sections 7.5 and 18.9 of the SSAR
- development of EOPs is discussed in Section 18.9.8 of the SSAR
- organization responses as part of the COL application is discussed in Section 18.12 of the SSAR
- reactor coolant pressure control is discussed in Section 7.7.1.6 of the SSAR

- Task 4 on SG overfill, carryover and water hammer – SG overfill, water carryover and water hammer are discussed in Section 15.6.3.2 of the SSAR and the control logic is discussed in Section 7.2 of the SSAR.

Therefore, Issue 135 is resolved for the AP600 design.

Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits

As discussed in NUREG-0933, Issue 142 addressed observations in 1987 during safety parameter display system (SPDS) evaluation tests that, for electrical transients below maximum credible levels, a relatively high level of noise could pass through types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy transmitted through the isolator could damage or seriously degrade the performance of the Class 1E components; in other cases, the electrically generated noise on the circuit may cause the isolation device to give a false output. This issue addressed electrical isolators used to maintain electrical separation between Class 1E and non-Class 1E electrical systems and prevent malfunctions in the non-Class 1E circuits from degrading the performance of Class 1E circuits.

In resolving this issue, the staff determined from operating experience that isolation devices perform satisfactorily in the operating environment and have not been exposed to failure mechanisms that resulted in signal leakage. This determination was made, however, on the basis that current plants predominantly use electromechanical controls and may not be applicable to instrumentation and control (I&C) systems with digital or electronic components. This issue was resolved with no new requirements established.

In its May 28, 1993, letter, Westinghouse stated that the use of isolation devices in the AP600 instrumentation and control architecture is described in Sections 7.1.2.11, 7.1.4.2.7, and 7.7.1.11 of the SSAR. The isolation devices are tested to conform to design requirements and this testing will identify the devices potentially susceptible to electrical leakage. Westinghouse further stated that the plant operational program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems, as well as identify the specific isolation devices used in the design, is the responsibility of the COL license holder. This is COL Action Item 20.3-1.

Westinghouse uses isolation devices to preserve electrical independence of divisions and to prevent interactions between Class 1E and non-Class 1E systems. These devices are part of the protection system and meet Section 4 of IEEE Standard 279-1971. They are tested to confirm that credible failures at the output of the isolation device do not prevent the associated

protection system channel from meeting the minimum performance requirements. This testing meets the requirements for an inspection and test program, and identifies those devices that are potentially susceptible to electrical leakage. Optical coupling offers improved physical and electrical isolation and separation because it eliminates electrically conductive paths from receiving terminal to transmitting terminal.

The diverse actuation system (DAS), which is described in Section 7.7.1.11 of the SSAR, uses sensors that are separate from those being used by the protection system and the control system. This prohibits failures from propagating to the other plant systems through the use of shared sensors.

Load isolation is also required for the DAS. For the interface with air-operated valves, which are normally actuated through a solenoid valve, the AP600 design provides separate solenoid valves for each path. For the motor control centers, coil to contact isolation is used for separation.

In DSER Open Item 20.3-21, the staff requested that Westinghouse provide additional information on the isolation devices, I&C architecture, consequences of a communication error, and diagnostic tests to isolate the cause of the error. Revision 9 of the SSAR includes this additional information. The staff concludes that Westinghouse has adequately addressed this issue in Section 1.9.4.2.3 of the SSAR. Communication links are described as using extensive testing and error checking to minimize erroneous transmission. These data links are discussed in SSAR Section 7.1.2.9.

Therefore, the staff considers DSER Open Item 20.3-21 closed and Issue 142 resolved for the AP600 design.

Issue 143: Availability of Chilled-Water Systems and Room Cooling

As discussed in NUREG-0933, Issue 143 addresses problems experienced in recent years at several nuclear plants with safety system components and control systems that have resulted from a partial or total loss of HVAC systems. Many of these problems exist because of (1) the desire to provide increased fire protection and (2) the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been accomplished by enclosing the affected equipment in small, isolated rooms. However, the result has been a significant increase in the impact of the loss of room cooling. Another problem resulting from loss of room cooling is the advancement in control circuit design. With the introduction of electronic integrated circuits, plant control and safety have improved; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as RHR pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once these failures occur, the Advisory Committee on Reactor Safeguards believed that the impact of these failures on the proper functioning of air cooling systems has not been

reflected in the final PRAs of plants. Thus, plants with similar, inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled-water systems to remove heat from the rooms containing the components. If chilled-water and HVAC systems are unavailable to remove heat, the ability of the safety equipment within the rooms to operate as intended cannot be assured.

Issue 143 has not been generically resolved and is classified in NUREG-0933 as a high-safety priority. A possible solution to this issue would require a reevaluation of each plant's room heat load and heatup rate in order to locate areas in which the dependence of equipment operability on HVAC and room cooling may be reduced. Although the total elimination of this dependence may not be possible at all plants, this analysis would locate areas in which this dependence is critical. The critical dependencies and the ability to reduce them could be determined through the use of a plant-specific PRA. After the critical dependencies are identified, each plant would implement procedural changes (to provide alternate cooling) to eliminate or reduce the dependencies where possible. Hardware modifications may be needed for situations in which a procedure change cannot be implemented to reduce a critical dependency.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that this issue does not apply to the AP600 design because the design does not rely on active safety systems to provide safe shutdown of the plant. A total loss of HVAC systems will not prevent a safe shutdown. The staff agrees with this statement. Therefore, Open Item 20.3-22 is closed and Issue 143 is resolved for the AP600 design.

Issue 153: Loss of Essential Service Water in LWRs

As discussed in NUREG-0933, Issue 153 addressed the reliability of ESW systems and related operating problems. In a comprehensive NRC evaluation of operating experience related to ESW systems (NUREG-1275, Volume 3, "Operating Experience Feedback Report," dated November 1988), a total of 980 operational events involving the ESW system were identified, of which 12 resulted in complete loss of the ESW system. Among the causes of failure and degradation are (1) various fouling mechanisms (sediment deposition, biofouling, corrosion and erosion, foreign material and debris intrusion); (2) ice effects; (3) single failures and other design deficiencies; (4) flooding; (5) multiple equipment failures; and (6) human and procedural errors.

At each plant, the ESW system supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink. The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe plant shutdown or to mitigate the consequences of the accident. Under normal operating conditions, the ESW system provides component and room cooling. During shutdowns, it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to the fire protection system, cooling towers, and water-treatment systems at a plant.

The design of the ESW system varies substantially from plant to plant and the ESW system is highly dependent on the NSSS. As a result, generic solutions (if needed) are likely to be

different for PWRs and BWRs. The possible solutions are (1) installation of a redundant intake structure including a service water pump, (2) hardware changes to the ESW system, (3) installation of a dedicated RCP seal cooling system, or (4) changes to TS or operational procedures.

In the resolution of Issue 130 on ESW pump failures at multi-plant sites, discussed earlier in this section, the staff surveyed seven multi-plant sites and found that loss of the ESW system could be a significant contributor to core-damage frequency. The generic safety insights gained from this study supported previous perceptions that ESW system configurations at other multi-plant and single-plant sites may also be significant contributors to plant risk and should also be evaluated. As a result, Issue 153 was identified to address all potential causes of ESW system unavailability, except those that had been resolved by implementation of the requirements in GL 89-13.

The staff resolved Issue 153 with no new requirements established for operating and new plants.

In Section 1.9.4.2.3 of the SSAR, Westinghouse states that this issue does not apply to the AP600 design because the design does not rely on the service water and component cooling water systems to provide safety-related safe shutdown. The staff agrees with this statement. Therefore, DSER Open Item 20.3-23 is closed and Issue 153 is resolved for the AP600 design.

20.4 Three Mile Island Action Plan Items

The TMI action plan items listed in Table 20.1-1 are evaluated against the AP600 design in this section. The majority of the items were chosen either because (1) 10 CFR 52.47(a)(1)(iv) or 10 CFR 50.34(f) requires the design to comply with them, or (2) Westinghouse decided that the item applied to the design and included a discussion of the item in the SSAR. In addition, the staff requested, and Westinghouse agreed, to address Issue II.K.3(5) for the AP600 design. Also, the staff decided to include a discussion of Issues I.C.1, II.E.2.2, II.K.1(3), II.K.1(4d), II.K.1(16), II.K.3(6), II.K.3(8), II.K.3(30), and II.K.3(31).

The references to SRP sections in this chapter are references to sections in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1982. The references to General Design Criteria (GDC) are references to the GDC in Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

Issue I.A.1.4: Long-Term Upgrading of Operating Personnel and Staffing

As discussed in NUREG-0933, Issue I.A.1.4 addressed changes to 10 CFR 50.54, "Conditions of Licenses," concerning shift staffing and working hours of licensed operators. The final rule that amended 10 CFR 50.54 was approved on April 28, 1983. This issue was resolved and new requirements were established.

The staff, however, considers this issue not relevant to the AP600 design because it is an operational issue outside the scope of AP600 design certification. The organizational structure of the site operator is discussed in Section 13.1 of this report. The COL applicant will be

responsible for addressing this issue as part of the licensing process and is COL Action Item 20.4-1.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue was issued with no new requirements. Although Westinghouse is correct as to the design of the plant, the responsibility of the COL applicant should be identified. The staff requested that Westinghouse address this issue for the AP600 design. This was Open Item 20.4-1. As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse, in WCAP-14645, "Human Factors Engineering Operating Experience Review Report for the AP600 Nuclear Power Plant", Revision 2, has satisfactorily addressed this item. Open Item 18.3.3.1-2 is closed; therefore Open Item 20.4-1 is closed and Issue I.A.4.1 is resolved for the AP600 design.

Issue I.A.2.6(1): Revise Regulatory Guide 1.8

As discussed in NUREG-0933, Item I.A.2.6(1) addressed the revision of RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," following the publication of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The revisions to the regulatory guide were to address acceptable means to meet new requirements for long-term upgrading of training and qualifications for operational personnel. The revisions to RG 1.8 were approved by the Commission and published in May 1987 (*Federal Register* Notice 52 FR 16007). This issue is resolved with new requirements established.

Westinghouse did not address this issue in its SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant.

The staff also considers this issue not relevant to the AP600 design because it is an operational issue outside the scope of AP600 design certification. The organizational structure of the site operator is discussed in Section 13.1 of this report. The COL applicant will be responsible for addressing this issue as part of the licensing process and is COL Action Item 20.4-2. Therefore, Issue I.A.2.6(1) is resolved for the AP600 design.

Issue I.A.4.1(2): Interim Changes in Training Simulators

As discussed in NUREG-0933, Issue I.A.4.1(2) addressed the specific training simulator weaknesses identified in the short-term study, of Issue I.A.4.1(1), NUREG/CR-1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," dated August 1980. This issue was resolved with the revision to RG 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," in April 1981 and new acceptance requirements were established.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant.

The staff also considers this issue not relevant to the AP600 design because it is an operational issue outside the scope of AP600 design certification. Training materials are discussed in Section 13.2 of this report. The COL applicant will be responsible for addressing this issue as part of the COL process and is part of COL Action Item 20.4-3. Therefore, Issue I.A.4.1(2) is resolved for the AP600 design.

Issue I.A.4.2: Long Term Training Simulator Upgrade

As discussed in NUREG-0933, Issue I.A.4.2 addressed the capabilities of training simulators. This issue was resolved by Revision 1 to RG 1.149 ("Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations"), 10 CFR 55.45(b) on approved or certified simulation facility in licensed operator operating tests, and NUREG-1258 ("Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55," dated December 1987). New requirements were established.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. This is part of COL Action Item 20.4-3.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse, in WCAP-14645, has satisfactorily addressed this item. Open Item 18.3.3.1-2 is closed; therefore, Issue I.A.4.2 is resolved for the AP600 design.

Issue I.C.1: Guidance for Evaluation and Development of Procedures for Transients and Accidents

As discussed in NUREG-0933, Issue I.C.1 addressed the preparation of emergency operating procedures (EOPs). The information in EOPs should provide assurance that operator and staff actions are technically correct and the procedures are easily understood for normal, transient, and accident conditions. The EOPs must be function-oriented procedures to mitigate the consequences of the broad range of mitigating events and subsequent multiple failure or operator errors, without the need to diagnose specific events. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing and surveillance must be in compliance with the guidance provided in NUREG-0737 and its Supplement 1.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant.

Westinghouse responded to the staff's question indicating that information addressing this requirement is provided in SSAR Section 18.9.8 and in WCAP-14075, Revision 0, "AP600 Design Differences Document for Development of Emergency Operating Guidelines Report". Section 18.9.8 of the SSAR summarizes high level operator actions associated with transients and accidents. WCAP-14705 (Rev 0), discusses design differences between a Westinghouse reference plant and the AP600 design. The applicant also indicated that Westinghouse low pressure (LP) emergency response guidelines (ERGs) are applicable to the AP600 design.

In SSAR Section 18.9.8, Westinghouse provides only the descriptions of methodologies used to develop the ERGs and an outline of high-level mitigation strategies, not the detailed ERGs necessary for resolution of TMI action items. There are fundamental differences between the LP reference plant and AP600 design in the safety system design, operation, and philosophy of emergency mitigation and recovery. Unlike the LP reference plant where the safety systems are active systems, the safety systems in AP600 are passive. Active systems are non-safety-related systems providing defense-in-depth functions.

Even though the passive safety systems perform similar functions as the active safety systems in the LP reference plants, the AP600 mitigation sequences, including the actuation of the active defense-in-depth systems and passive safety systems, and plant conditions at which these systems will be actuated and will remain operating, differ from the LP reference plants. For AP600, the active systems, though not actuated by safeguard signals, are manually actuated and relied upon as first line of defense to avoid unnecessary actuation of passive safety systems.

The plant responses, including possible adverse systems interactions between the active and passive systems, may also differ significantly from the LP reference plants. Certain issues where operator actions play key roles in the accident scenarios require the AP600-specific ERGs as a basis for resolution. For example, in a SGTR event, operator's actions to isolate the faulted SG and other mitigation and recovery actions to minimize the possibility of radioactive releases through the main steam safety valves will be important for the resolution of the issue of containment bypass resulting from a SGTR event. Additionally, the ERGs should include guidance for low power and shutdown operations, when many systems will be out for maintenance and the plant is in a configuration different from the normal operation, and for severe accident management.

The staff, therefore, concluded that the AP600-specific ERGs are needed to satisfy these requirements. Supporting analyses necessary to demonstrate the effectiveness of operator actions in response to transients and accidents should also be provided by Westinghouse. This was Open Item 20.4-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse, in WCAP-14645, has satisfactorily addressed this item. Therefore, Open Item 20.4-2 is closed and Issue I.C.1 is resolved for the AP600 design.

Issue I.C.5: Procedures for Feedback of Operating Experience to Plant Staff

As discussed in NUREG-0933, Issue I.C.5 addressed the quality of procedures for feedback of experience at operating plants. This issue was clarified in NUREG-0737 and requirements were issued there.

In SSAR Section 1.9.3, Item (3)(i), Westinghouse addresses this issue and states that AP600 design engineers are continually involved in reviewing industry experiences from sources such as NRC bulletins, licensee event reports, NRC requests for information letters to licensees, *Federal Register* information, and NRC generic letters. It is stated that lessons-learned experience has been incorporated into the AP600 design through the participation in developing Volume III of the ALWR URD and in ALWR Utility Steering Committee activities.

Westinghouse addressed the responsibility of the designer of the plant; however, the COL applicant will also be responsible for site-specific information at the COL and operational phases. Development of detailed procedures is outside the scope of the AP600 design certification and is the responsibility of the COL applicant. This is COL Action Item 20.4-4. Westinghouse should address this and the methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures. This was Open Item 20.4-3.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in SSAR Chapter 18. Open Item 18.3.3.1-2 is closed; therefore Open Item 20.4-3 is also closed and Issue I.C.5 is resolved for the AP600 design.

Issue I.C.9: Long-Term Program for Upgrading Procedures

As discussed in NUREG-0933, Issue I.C.9 addressed the upgrading of procedures at operating plants. With the exception of EOPs, this issue was clarified in Supplement 1 of NUREG-0737 and resolved with Revision 1 of SRP Section 13.5.2. This issue was resolved with no new requirements.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue was issued with no new requirements. Although Westinghouse is correct, the responsibility of the COL applicant in procedure development should be identified. The staff requested that Westinghouse address this issue for the AP600 design. The methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures should be addressed. This was identified as Open Item 20.4-4 and COL Action Item 20.4-5.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Open Item 18.3.3.1-2 is closed; therefore, Open Item 20.4-4 is also closed and Issue I.C.9 is resolved for the AP600 design.

Issue I.D.1: Control Room Design Reviews

As discussed in NUREG-0933, Issue I.D.1 addressed licensees performing a detailed review of their control room using human factors engineering (HFE) techniques and guidelines to identify and correct design deficiencies. This issue was clarified in NUREG-0737 and NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981, and requirements were issued. This issue is considered resolved.

In Section 1.9.3 of the SSAR, Item (2)(iii), Westinghouse states that the AP600 MCR was designed by a multi-disciplined man-machine interface design team using state-of-the-art human factors principles. The team used a control room design process predicated on the functional decomposition of the plant, integrating the capabilities of both man and machine. SSAR Chapter 18 discusses the MCR design process and Section 1.9.1 of the SSAR provides information on the conformance of the design with applicable regulatory guides.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Open Item 18.3.3.1-2 is closed; therefore, Issue I.D.1 is resolved for the AP600 design.

Issue I.D.2: Plant Safety Parameter Display Console

As discussed in NUREG-0933, Issue I.D.2 addressed improving the presentation of the information provided to control room operators. The requirements for this issue are in Supplement 1 to NUREG-0737. This issue raised the need for a SPDS that clearly displays a minimum set of parameters defining the safety status of the plant. Paragraph (2)(iv) of 10 CFR 50.34(f) requires a plant SPDS console that will provide such a display to operators, and that is capable of displaying a full range of important plant parameters and data trends on demand and indicating when process limits are being approached or exceeded.

In SSAR Section 1.9.3, Item (2)(iv), Westinghouse states that the purpose of the plant SPDS is to display the important plant variables in the MCR to assist the operator in rapidly and reliably determining the safety status of the plant. The SPDS design is discussed in SSAR Chapter 18.

The SPDS requirements are stated to be specified during the MCR design process, discussed in Issue I.D.1, and are met by the MCR design, specifically as part of the alarms, displays, and controls. The requirements are met by grouping the alarms by plant process or purpose, as directly related to the critical safety functions.

The process data presented on the graphic displays is similarly grouped, facilitating an easy transition for the operators. The SPDS requirement for presentation of plant data in an analog fashion before reactor trip is met by the design of the graphic cathode ray tube (CRT) displays. Displays are available at the operator workstations, the supervisor workstation, the remote shutdown workstation, and at the technical support center.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Open Item 18.3.3.1-2 is closed; therefore, Issue I.D.2 is resolved for the AP600 design.

Issue I.D.3: Safety System Status Indication

As discussed in NUREG-0933, Issue I.D.3 addressed the need for those licensees and applicants who have not committed to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to install a bypass and inoperable status indication system to give operators timely information on the status of the safety systems. Resolution of this issue requires adoption of the guidelines in RG 1.47.

In Section 1.9.3 of the SSAR, Item (2)(v), Westinghouse states that the AP600 MCR meets RG 1.47, including automatic indication of bypassed and inoperable status of plant safety systems. This is described in SSAR Chapters 7 and 18, and Appendix 1A. Plant safety parameters, protection system status, and plant component status signals are processed by the protection and safety monitoring system and are available to the entire instrumentation and control system via the redundant monitor bus.

Class 1E signals are stated to be provided to the qualified data processor, which is part of the protection and safety monitoring system, for accident monitoring displays. The display of this data is incorporated in the process data displays, discussed in Issue I.D.2, on the graphic CRTs in the MCR.

In Section 7.1.4.2.13 of the SSAR, Westinghouse states that the AP600 design complies with the requirements in Paragraph 4.13 of IEEE Standard 279 and the guidelines in RG 1.47 which state that if the protective action of some part of the I&C system has been bypassed or deliberately rendered inoperable, this fact will be continuously indicated in the MCR. Westinghouse has provisions within the AP600 integrated protection system for bypasses (or blocks) of certain protective functions during operational modes such as test, maintenance, or if the channel sensor has failed and can not be immediately repaired. Westinghouse states that the control room design provides for automatic indication of bypassed and inoperable status of plant safety systems. The display of status information allows the operator to identify the specific functions that are bypassed, and also determine if the actuation logic has reverted to two-out-of-three or one-out-of-two. In addition, an alarm is sounded in the MCR if more than one bypass has been applied to a given protective function, thus causing a one-out-of-two logic.

The human factors details of this requirement are beyond the scope of the AP600 design review and should be addressed by the COL applicant. Westinghouse did not address this in its response for this issue. The staff requested that Westinghouse address the responsibility of the COL applicant in its response for Issue I.D.3. This was Open Item 20.4-5. The responsibility of the COL applicant is designated COL Action Item 20.4-6.

On the basis of the above information, the staff concludes that the AP600 design meets the guidance in RG 1.47 and, therefore, meets the requirements of Issue I.D.3 with respect to the I&C design for safety system status monitoring. Therefore, the staff considers DSER Open Item 20.4-5 closed and Issue I.D.3 is resolved for the AP600 design.

Issue I.D.5(2): Control Room Design – Improved Instrumentation Research Plant Status and Post-Accident Monitoring

As discussed in NUREG-0933, Issue I.D.5(2) addressed the need to improve the operators' ability to prevent, diagnose, and properly respond to accidents. This issue was originally raised in 1980, in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated May 1980, and led to new NRC requirements. Guidance for addressing the issue is in RG 1.47, which describes an acceptable method for implementing the requirements of IEEE 279-1971 ("Criteria for Protection Systems for Nuclear Power Generating Stations") and Appendix B (Criterion XIV) of 10 CFR Part 50, with respect to the bypass or inoperable status of safety systems; and RG 1.97, which defines an acceptable method for implementing NRC requirements to provide instrumentation and to monitor plant variables and systems during and following an accident.

The acceptance criteria for the resolution of this issue are:

- For ESF status monitoring, RG 1.47 recommends automatic bypassed or inoperable status indication at the system level for plant protection systems, safety systems actuated or controlled by protection systems, and their auxiliary and supporting systems. These features should indicate in the control room and should have manual input capability.

- For post-accident monitoring instrumentation, RG 1.97, Revision 2, gives criteria for design and qualification of the instrumentation. Three categories (designated 1, 2, and 3) provide a graded approach to requirements on the basis of the importance to safety of the variable being monitored. Criteria exist for equipment qualification, redundancy, power sources, channel availability, QA, display and recording range, equipment identification, interfaces, servicing, testing and calibration, human factors, and direct measurement. The actual variables to be monitored are tabulated by type, and the instrumentation design and qualification requirement (Category 1, 2, or 3) are identified for each variable.

In its May 28, 1993, letter, Westinghouse stated that the AP600 design conforms to RG 1.97, which provides acceptable guidance for post-accident monitoring of nuclear reactor safety parameters, including plant process parameters important to safety and the monitoring of effluent paths and plant environs for radioactivity. For the AP600 design, an analysis was conducted to identify the appropriate plant variables, and establish the appropriate design-basis and qualification criteria for instrumentation used by an operator for monitoring conditions in the RCS, secondary heat removal system, containment, and systems used for attaining a safe-shutdown condition. This is discussed in Section 7.5 of the SSAR.

Westinghouse addressed this issue in Section 1.9.4.2.1 of the SSAR. The instrumentation is used by the operator to monitor and maintain the safety of the plant during operating conditions, including anticipated operational occurrences, and accident and post-accident conditions. The plant parameters identified to satisfy RG 1.97 are processed and displayed by the qualified data processing system (QDPS) discussed in Section 18.8 of the SSAR. The verification and validation (V&V) of the QDPS complies with the V&V process described in SSAR Section 18.11.

In Section 7.5 of the SSAR, Westinghouse compares the AP600 design against the criteria in Revision 3 of RG 1.97 and addresses accident monitoring instrumentation. The design complies with Revision 3 of RG 1.97.

Issue I.D.5(2) was resolved with the issuance of Revision 2 of RG 1.97. On the basis of the information provided by Westinghouse and the fact that the AP600 design is in compliance with Revision 3 of RG 1.97, the staff concludes that this issue has been addressed. Therefore, Issue I.D.5(2) is resolved for the AP600 design.

Issue I.D.5(3): Control Room Design – On-Line Reactor Surveillance Systems

As discussed in NUREG-0933, Issue I.D.5(3) addressed the benefit to plant safety and operations of continuous on-line automated surveillance systems. Systems that automatically monitor reactor performance can benefit plant operations and safety by providing continuous diagnostic information to the control room operators, to predict anomalous plant behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise-monitoring in BWRs to detect vibrations in internal components, and pressure noise surveillance at TMI-2 to monitor primary loop degasification. On-line surveillance data have been used to assess loose thermal shields.

Continuous on-line surveillance of the NSSS involves the following areas for which acceptance criteria are separately defined:

- vibration monitoring of reactor internals
- RCPB leakage detection
- loose-parts monitoring

The acceptance criteria for the resolution of Issue I.D.5(3) for monitoring vibrations in internal components are in ANSI/ASME OM-5-1981, "Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors." This standard makes recommendations on the use of ex-core neutron detector signals for monitoring core barrel axial preload loss. This standard also documents a program containing baseline, surveillance, and diagnostic phases and makes recommendations for data acquisition frequency and analysis.

The acceptance criteria for leak monitoring are in RG 1.45, which documents acceptable methods for channel separation, leakage detection, detection sensitivity and response time, signal calibration, and seismic qualification of RCPB leakage detection systems. It defines the regulatory position for an acceptable design of these systems.

The acceptance criteria for loose-parts monitoring are in RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." This RG gives guidelines on such system characteristics as sensitivity, channel separation, data acquisition, and seismic and environmental conditions for operability. It also identifies alert levels, data acquisition modes, safety analysis reports, and TS pertaining to a loose-parts monitoring system.

The AP600 design includes a reactor coolant pressure boundary leakage detection system as required by 10 CFR 50, Appendix A, GDC 30, and it conforms to the staff regulatory positions,

as indicated in RG 1.133, for the design of the loose-parts monitoring system. The detailed system design discussions are in SSAR Chapters 5 and 7.

The staff reviewed the functional requirements of the metal impact monitoring system (MIMS), which monitors the reactor coolant system for the presence of loose metallic parts according to the regulatory position requirements as indicated in RG 1.133, Revision 1, May 1981, and concludes that the MIMS functional design requirements satisfied RG 1.133, with the exceptions of system surveillance and reporting requirements, which are most appropriately addressed by the COL for plant-specific design. Therefore, Issue I.D.5(3) is resolved for the AP600 design.

Issue I.F.1: Expanded Quality Assurance List

As discussed in NUREG-0933, Issue I.F.1 addressed improving the quality assurance (QA) program for the design, construction, and operation of nuclear power plants. The licensees were to identify those SSCs that were not labeled safety-related but were important to safety, to prioritize their importance to safety, and to prepare a generic QA list. In GDC 1, "Quality Standards and Records," the NRC requires that SSCs important to safety should be designed, built, and tested commensurate to their importance to safety. On January 5, 1984, the staff issued GL 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety-Related'," to clarify the use of the terms "important to safety" and "safety-related." This issue was considered resolved and no new requirements were established.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue had been resolved without new requirements. Although this is correct, the staff requested Westinghouse to address the classification of SSCs for the AP600 design. This was Open Item 20.4-6.

Westinghouse addressed this issue in Revision 16 to SSAR Section 3.2, "Classification of Structures, Components and Systems," which establishes the classification of SSCs for the AP600 design. Therefore, DSER Open Item 20.4-6 is closed and Issue I.F.1 is resolved for the AP600 design.

Issue I.F.2: Develop More Detailed Quality Assurance Criteria

As discussed in NUREG-0933, Issue I.F.2 addressed improvements to the QA program for the design, construction, and operation at nuclear power plants to provide greater assurance that these activities are conducted in a manner commensurate with their importance to safety. The subissues for Issue I.F.2 that must be addressed for 10 CFR 52.47(a)(1)(iv) are the following:

- Item 2 – include QA personnel in review and approval of plant procedures
- Item 3 – include QA personnel in all design, construction, installation, testing, and operation activities
- Item 6 – increase the size of the QA staff; and Item 9, clarify organizational reporting levels for the QA organization

Westinghouse did not address all of this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design, except for subparts I.F.2(2), "Include QA Personnel in Review and Approval of Plant Procedures," and I.F.2(3), "Include QA Personnel in all Design, Construction, Installation, Testing, and Operation Activities." The other subparts of I.F.2 were considered not relevant to the AP600 design because the subparts had been prioritized low, dropped, or not been prioritized, or were the responsibility of the COL applicant.

In Section 1.9.3 of the SSAR, Item (3)(iii), Westinghouse simply stated that the AP600 quality assurance plan was described in SSAR Chapter 17 and meets the requirements of this issue.

Westinghouse has only addressed the QA program for the design of the AP600. The QA program for the COL applicant's design, construction, and operation phases are outside the scope of AP600 design certification and are designated COL Action Item 20.4-7. The COL applicant will have the responsibility of addressing this issue for the design of the remaining parts of the plant, and for the modification and operation of the plant. Westinghouse should address the responsibility of the COL Applicant. This was Open Item 20.4-7.

Westinghouse subsequently addressed this issue in Section 1.9.3 of the SSAR. Item (3)(iii) "Quality Assurance Program (NUREG-0737 Item I.F.2)" states that the AP600 quality assurance plan was described in SSAR Chapter 17 and meets the requirements of Issue I.F.2. In SSAR Chapter 17, Section 17.4, "Combined License Information Items," Westinghouse included the need of the COL applicant to address its QA program for the design, construction, and operation phases. Accordingly, Open Item 20.4-7 is closed and Issue I.F.2 is resolved for the AP600 design.

Issue I.G.2: Scope of Test Program

As discussed in NUREG-0933, Issue I.G.2 addressed the need for licensees to develop a more comprehensive preoperational and low-power test program for their plant to find any anomalies in the responses of the plant to transients during the initial test program (ITP). With the revisions to the SRP Section 14, "Initial Test Program," and the NRC Office of Inspection and Enforcement Manual (June 1989 revision to NUREG-0933), this issue was considered resolved and new requirements were established.

The AP600 ITP is discussed in Chapter 14 of this report. The staff concluded that the design features and/or performance capability tests to be conducted and demonstrated on plant systems and components during the AP600 ITP described in SSAR Chapter 14 will comprehensively identify anomalies or unanticipated plant behavior in the responses of the plant to transients. Therefore, the staff requested Westinghouse to reconcile Table 1.9-2 with the scope of the AP600 ITP in SSAR Chapter 14. Also, Westinghouse should discuss the responsibility of the COL applicant who will implement the program at the plant. This was Open Item 20.4-8. The responsibility of the COL applicant is identified in COL Action Items 14.2.2-1, 14.2.3-1 and -2, 14.2.3.1-1, 14.2.3.2-1 and 14.2.5-1 in Chapter 14 of this report.

Westinghouse subsequently addressed this issue in Section 1.9.3 of the SSAR, Item I.G.2, "Scope of Test Program." In Item I.G.2, Westinghouse states that the program plan for preoperational and startup testing of the AP600 is in Section 14.2 of the SSAR. Also, in SSAR

Section 14.4, "Combined License Applicant Responsibilities," Westinghouse addresses the responsibility of the COL applicant who will implement the AP600 ITP.

The staff considers that the design features and/or performance capability tests to be conducted and demonstrated on plant systems and components during the AP600 ITP described in SSAR Chapter 14 will comprehensively identify anomalies or unanticipated plant behavior in the responses of the plant to transients. Therefore, DSER Open Item 20.4-8 is closed and Issue I.G.2 is resolved for the AP600 design.

Issue II.B.1: Reactor Coolant System Vents

As discussed in NUREG-0933, Issue II.B.1 addressed the requirements in 10 CFR Part 50 and NUREG-0737 to install reactor vessel and RCS high-point vents. These vents are designed to release noncondensable gases from the RCS to avoid loss of core cooling during natural circulation. The design of these vents must conform to the applicable GDC requirements of 10 CFR Part 50 (Appendix A), which are listed below, and meet the applicable codes and standards for the RCS pressure boundary. This issue was resolved and new requirements were issued in 10 CFR Part 50 and NUREG-0737.

In Section 1.9.3 of the SSAR, item (2)(vi), Westinghouse states that in the AP600 design the capability for remotely venting the high points of the RCS is provided by the safety-related automatic depressurization system (ADS) valves and the safety-related reactor vessel head vent system. Both discharge to the refueling water storage tank (RWST) inside containment. The ADS is stated to provide redundant groups of MOVs connected to the top of the pressurizer and air-operated valves connected to the top of each RCS hot leg. However, only the pressurizer MOVs, the first-stage ADS valves, are used for remote manual venting because they are the only ADS valves capable of being throttled. The reactor vessel head vent system removes steam and noncondensable gases directly from the reactor vessel head.

The reactor coolant system vents must meet the following design criteria:

- the system must be operable from the control room (GDC 19)
- the system must be testable (GDC 36)
- the system must be capable of functioning following a LOOP (GDC 17)
- the system must be able to withstand an operating-basis earthquake (RG 1.29)

Westinghouse further states that, during normal and moderate frequency events, noncondensible gases from the RCS accumulate in the pressurizer steam space, with very little accumulating in the reactor vessel head, because of the continuous recirculation of bypass spray flow through the pressurizer when the RCPs are operating. This bypass flow causes boiling in the pressurizer, making the pressurizer steam space the lowest static pressure region in the RCS. This causes off-gassing of the RCS to occur in the pressurizer. This gas accumulation can be removed by remote manual operation of the first-stage ADS valves.

During LOCAs, the ADS automatically depressurizes the RCS so that the passive core cooling system may operate and effectively deliver cooling flow to the core. This would not happen until the RCS pressure was brought down to the passive core cooling system operating level.

Westinghouse also states that it is possible that continued depressurization of the RCS by the ADS could result in creation of a gas-steam volume (or bubble) in the upper region, or head, of the reactor vessel. With only the ADS, this volume can expand, filling the head of the vessel until it reaches the inside of the hot leg and is vented through the hot leg and the surge line, and out of the RCS. At the hot leg, this volume either vents into the pressurizer through the surge line and enters the ADS, or enters the ADS through the hot leg. This will depend on which ADS valves are open. This venting provides an open injection and steam venting path through the reactor vessel, and maintains required core flow without needing to refill the reactor vessel and pressurizer.

The reactor vessel head vent system can be operated from the MCR to directly vent the top of the reactor vessel head and remove this volume. It is described in Section 5.4.12 of the SSAR.

The staff reviewed the high-point vents for the AP600 design. The design relies on the safety-related ADS valves and the safety-related reactor vessel head vent system to provide the capability of high-points venting of noncondensable gases from the RCS. Descriptions of the reactor vessel head vent system and the ADS valves are provided, respectively, in Sections 5.4.12 and 5.4.6 of the SSAR. These systems are operated from the control room, and associated valve position indications and alarms are provided. Their vent paths discharge to the IRWST. The vessel head vent system is entirely located inside containment. Because the system isolation valves do not serve a containment isolation function, containment integrity will not be compromised as a result of a loss of power to the valves. This is a design improvement relative to current operating and standard design plants, where the reactor vessel head vent system isolation valves also provide containment isolation. The system has the capability to remove noncondensable gases or steam from the RCS via remote manual operation of the redundant vent paths. It is designed to vent a volume of hydrogen equal to approximately one-half of the RCS volume at system pressure and temperature in 1 hour. The first-stage ADS valves are attached to the pressurizer and they provide the capability to vent noncondensable gases from the pressurizer steam space following an accident.

The staff concludes that AP600 design complies with Part 50.34(f)(2)(vi) requirements; therefore, Issue II.B.1 is resolved for the AP600 design.

Issue II.B.2: Safety Review Consideration – Plant Shielding To Provide Postaccident Access to Vital Areas

As discussed in NUREG-0933, Issue II.B.2 addressed having licensees perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The review would locate vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, where occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems. This issue was resolved and the requirements were provided in 10 CFR 50.34(f)(2)(vii) (II.B.2).

In Section 1.9.3 of the SSAR, Item (2)(vii), Westinghouse states that a plant shielding analysis was performed of the AP600 general plant arrangement. This included a review of the primary shielding surrounding the reactor; the secondary shielding that encloses the reactor coolant loops; the shielding for refueling operations, including the refueling canal walls and refueling water; the auxiliary shielding such as equipment compartments, valve galleries, piping tunnels, the CVS, and other equipment modules; and accident shielding, including the shielding provided by buildings and the shielding to minimize sky shine.

Westinghouse further stated that improvements were incorporated into the AP600 shielding design as they were identified. One specific improvement is the elimination of recirculated primary coolant outside of containment during design-basis events. The safety-related passive systems and emergency cooling and injection flow paths are located inside containment, even for long-term post-accident cooling. This design eliminates the necessity for flow paths outside containment with highly radioactive water from accident and post-accident core injection or recirculation cooling outside containment, which would require additional shielding. The non-safety-related systems can be used to recirculate coolant outside of containment following an accident; however, these systems would not be operated when high radiation levels exist in the reactor coolant.

Initially, the AP600 SSAR did not adequately address the relationship between the source term used for accident analysis and the plant shielding. The staff identified this deficiency as Open Item 20.4-9. Westinghouse has since amended Section 12.2 of the SSAR to address post-accident radiation sources used in the shield design and assessment of post-accident access to vital areas. The post-accident source term used for the AP600 is predicated on the core release model from NUREG-1465, which supersedes the TID-14844 source term assumptions as reflected in RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." Section 12.2 of the SSAR contains tables that list the post-LOCA instantaneous and integrated source strengths as a function of time. Section 12.3 addresses vital areas for post-accident access and includes radiation zone maps that show projected dose rates in these areas and access routes for the various post-accident actions requiring access to vital areas.

In Section 12.4.1.8 of the SSAR, Westinghouse provided a listing of the seven vital plant areas that will require post-accident accessibility. For each of these areas, Westinghouse calculated the dose rates on the basis of the AP600 post-accident source term. Using these calculated dose rates, along with the time estimates for ingress, egress, and performance of actions at the vital area locations, Westinghouse has shown that the personnel radiation doses for individuals accessing these areas following an accident do not exceed 5E-02 sieverts (5 rem) whole body or its equivalent to any part of the body and that the requirements of 10 CFR 50.34(f)(2)(vii) (II.B.2) are met, except for using the TID-14844 source term. The justification for exempting the AP600 design from the requirements to use TID-14844 is provided in Section 20.6 of this report.

The supplemental information on post-accident source terms and plant shielding added to Chapter 12 of the AP600 SSAR by Westinghouse resolves the staff's concerns in this area. Therefore, the staff considers Open Item 20.4-9 closed and Issue II.B.2 is resolved for the AP600 design.

Issue II.B.3: Postaccident Sampling Capability

As discussed in NUREG-0933, Issue II.B.3 addressed upgrading postaccident sampling at plants. The requirements are in 10 CFR 50.34(f)(2)(viii). The reactor coolant and containment atmosphere sampling-line systems should permit personnel to take a sample under accident conditions promptly and safely. The radiological spectrum analysis facilities should be capable of quantifying certain radionuclides that are indicators of the degree of core damage promptly. In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions.

In Section 1.9.3 of the SSAR, item (2)(viii), Westinghouse states that the AP600 plant PSS, described in SSAR Chapter 9, is designed to perform the post-accident sampling functions required by NUREG-0737 and recommended in RG 1.97. The staff evaluates post-accident sampling in Sections 9.3.3 and 13.3 of this report. The staff concludes that, based on the evaluation in Sections 9.3.3 and 13.3 of this report, Issue II.B.3 is resolved for the AP600 design, except for using the TID-14844 source term. The justification for exempting the AP600 design from the requirements to use TID-14844 is provided in Section 20.6 of this report.

Issue II.B.8: Rulemaking Proceedings on Degraded Core Accidents Description

Item II.B.8 of NUREG-0933 discussed the need to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis. The Commission expects that new designs will achieve a higher standard of severe accident safety performance than previous designs. In an effort to provide this additional level of safety in the design of advanced nuclear power plants, the NRC developed guidance and goals for designers to strive for in accommodating events that are beyond what was previously known as the design basis of the plant.

For advanced passive nuclear power plants, like the AP600, the staff concluded that vendors should address severe accidents during the design stage to take full advantage of the insights gained from such input as probabilistic safety assessments, operating experience, severe accident research, and accident analysis by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. Incorporating insights and design features during the design phase has been demonstrated to be much more cost effective than modifying existing plants.

The NRC issued guidance for addressing severe accidents in the following documents:

- the "NRC Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants"
- the "NRC Policy Statement on Safety Goals for the Operations of Nuclear Power Plants"
- the "NRC Policy Statement on Nuclear Power Plant Standardization"

- 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants"
- SECY-90-016, and the corresponding staff requirements memorandum (SRM) dated June 26, 1990
- SECY-93-087, and the corresponding SRM dated July 21, 1993

Whereas, the first three documents provide guidance as to the appropriate course for addressing severe accidents, 10 CFR Part 52 contains general requirements for addressing severe accidents, and the SRMs relating to SECY-90-016 and SECY-93-087 give Commission-approved positions for implementing features in new designs for preventing severe accidents and mitigating their effects.

The basis for resolution of severe accident issues for the AP600 is 10 CFR Part 52 and SECY-93-087. 10 CFR Part 52 requires (a) compliance with the TMI requirements in 10 CFR 50.34(f), (b) resolution of unresolved safety issues and generic safety issues, and (c) completion of a design-specific probabilistic risk assessment. The staff evaluates these criteria in Sections 20.3, 20.1 and 20.2, and 19.1 of this report, respectively.

The Commission-approved positions on the issues discussed in SECY-93-087 form the basis for the staff's deterministic evaluation of severe accident performance for the AP600. The staff evaluates the AP600 relative to these criteria in Section 19.2 of this report. Issue II.B.8 is resolved and Open Item 20.4-10 is closed for the AP600 design on the basis of the staff's evaluation of the probabilistic and deterministic analyses in the AP600 PRA, as documented in Chapter 19 of this report.

Issue II.D.1: Performance Testing of PWR Safety and Relief Valves

As discussed in NUREG-0933, Issue II.D.1 addressed the requirements in NUREG-0737 for qualification testing of RCS safety, relief, and block valves under expected operating conditions for design-basis transients and accidents, including ATWS. This issue was resolved by requiring licensees to conduct testing to qualify reactor coolant relief valves, safety valves, block valves, and associated discharge piping.

EPRI conducted a safety and relief valve test program for a group of PWR licensees to respond to the staff recommendations in NUREG-0587 and as clarified in NUREG-0737. The purpose of the program was to develop sufficient documentation and test data so that the participating licensees could demonstrate compliance with the II.D.1 requirements. The results were documented in the EPRI report, EPRI-NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982. The staff used the test results documented in EPRI-NP-2770-LD and summarized in EPRI-NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Report," December 1982, as a part of its acceptance criteria in its evaluations of the resolution of Issue II.D.1 for all current operating plants.

In Section 1.9.3 of the SSAR, item (2)(x), Westinghouse states that the AP600 design does not include power-operated relief valves and their associated block valves on the RCS. The safety valve and discharge piping used will either be of similar design as those valves tested and documented in EPRI Report NP-2770-LD or will be tested in accordance with the guidelines of

Issue II.D.1 in NUREG-0737. In DSER Open Item 20.4-11, the staff requested that Westinghouse clarify an apparent inconsistency in the references to the EPRI test data in Sections 1.9.3 and 1.9.7 of the SSAR. Revision 18 to the SSAR revised these two SSAR Sections so that they both reference EPRI Report NP-2770-LD. This is an acceptable response. Therefore, Open Item 20.4-11 is closed. The commitment in SSAR Section 1.9.3, which is summarized in this paragraph above, is consistent with the acceptance criteria used by the staff in it's evaluations of Issue II.D.1 for operating plants, and is acceptable for the AP600. Therefore, Issue II.D.1 is resolved for the AP600 design.

Issue II.D.3: Coolant System Valves – Valve Position Indication

As discussed in NUREG-0933, Issue II.D.3 addressed the requirements in NUREG-0737 for positive indication in the control room of RCS relief or safety valve position. The acceptance criterion for the resolution of this issue is that the plant design shall include safety and relief valve indication derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe in accordance with the requirements in NUREG-0737.
This indication shall have the following design features:

- Unambiguous safety and relief valve indication shall be provided to the control room operator.
- Valve position should be indicated within the control room and should be alarmed.
- Valve position indication may be either safety or control grade; if it is control grade, it must be powered from a reliable (e.g., battery backed) instrument bus (see RG 1.97).
- Valve position indication should be seismically qualified consistent with the component or system to which it is attached.
- Valve position indication shall be qualified for the appropriate operating environment which includes the expected normal containment environment and an operating basis earthquake (OBE).
- Valve position indication shall be human-factors engineered.

In Section 1.9.3 of the SSAR, item (2)(xi), Westinghouse simply states that the AP600 design does not include power-operated relief valves and their associated block valves, and the direct indication of the position of the relief and safety valves in the AP600 design is provided in the MCR.

This issue requires reactor coolant relief and safety valves be provided with positive indication in the control room. The applicant states that the AP600 design complies with the requirements in that positive position indication is provided for the pressurizer safety and the normal residual heat removal system relief valves. Section 5.4.9 of the SSAR describes the reactor coolant system relief and safety valve position indications. Position indication for these valves is accomplished through electrical reed switches, and is powered by the safety-related Class 1E dc and uninterruptable power system. These indications are provided in the control room by the protection and safety monitoring system (PMS).

Additionally, the main steam safety valves and power-operated relief valves also are provided with positive position indication in the control room by the PMS. The position sensors, and channels are seismically and environmentally qualified for a harsh environment. The power supply is Class 1E.

Confirmatory Item 20.4-1 stated that the resolution of this issue was dependent on the implementation and verification of SSAR changes to reflect the licensee's commitments. Positive position indication for the valves above was to be added to SSAR Table 3.11-1. The staff confirmed that these valves have been added to Table 3.11-1 of SSAR, Revision 9, and reflect the licensee's commitments on position indication. The staff considers Confirmatory Item 20.4-1 closed.

Therefore, Issue II.D.3 is resolved for the AP600 design.

Issue II.E.1.1: Auxiliary Feedwater System Evaluation

As discussed in NUREG-0933, Issue II.E.1.1 addressed improving the reliability of the auxiliary feedwater system or the emergency feedwater system. The issue addressed the following requirements in NUREG-0737:

- a simplified EFW system reliability analysis to determine the potential for system failure under various loss-of-main-feedwater transients
- the acceptance criteria in SRP 10.4.9 and BTP ASB 10-1
- evaluated EFW flow rate design basis and criteria

In Section 1.9.3 of the SSAR, Items (1)(ii) and (2)(xii), Westinghouse states that the AP600 design does not utilize an auxiliary feedwater system. A non-safety-related startup feedwater system (SUFS) is provided to remove the core decay heat after the reactor trip during postulated non-LOCA events. Flow indication of the SUFS is provided in the MCR. The SUFS pumps automatically start following anticipated transients resulting in low SG level. The startup feedwater control valves automatically control feedwater flow to the SGs during operation. They can also be manually operated from the MCR. Operation of the SUFS is not credited to mitigate licensing design-basis accidents, which are discussed in SSAR Chapter 15. The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal non-safety-related heat removal paths are unavailable; it is described in Section 6.3 of the SSAR.

On the basis of the staff's review, which is discussed in Section 10.4.9 of this report, the staff concludes that Issue II.E.1.1 is resolved for the AP600 design because the SUFS is non-safety-related.

Issue II.E.1.2: Auxiliary Feedwater System Automatic Initiation and Flow Indication

As discussed in NUREG-0933, Issue II.E.1.2 addressed improving the reliability of the auxiliary feedwater or emergency feedwater system. It addressed the requirement in NUREG-0737 for plants to install a control-grade system for automatic initiation of the EFWS. The acceptance criteria are in NUREG-0737 and in the design requirements of IEEE 279-1971. Specifically, the

system shall incorporate such design features as automatic system initiation, protection from single failure, and environmental and seismic equipment qualification. The issue requires provisions for automatic and manual auxiliary feedwater system initiation, and for flow indication in the control room.

In Section 1.9.3 of the SSAR, Items (1)(ii) and (2)(xii), Westinghouse states that the AP600 design includes the non-safety-related startup feedwater system (SUFWS) and not an auxiliary feedwater system. Flow indication of the SUFWS is provided in the MCR. The SUFWS pumps automatically start following anticipated transients resulting in reactor trips and the control valves automatically control feedwater flow to the SGs during operation. They can also be manually operated from the MCR. The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable.

The AP600 design does not use an auxiliary feedwater system. The design uses a non-safety-related SUFWS to remove the core decay heat after a reactor trip during non-LOCA events. Because the SUFWS is non-safety-related and not taken credit for in an accident, the system does not have to meet all of the requirements of IEEE Standard 279-1971. However, flow indication is provided in the MCR, and the pumps automatically start following anticipated transients resulting in a reactor trip and automatically control feedwater flow to the SGs during power operation. They can also be manually operated from the MCR. The safety-related passive core cooling system provides for emergency core cooling during transients and accidents, where the normal heat removal paths are not available.

Although the AP600 design does not have a safety-related auxiliary feedwater system, it provides the SUFWS, which adequately addresses the requirements in this issue.

Therefore, Issue II.E.1.2 is resolved for the AP600 design.

Issue II.E.1.3: Update Standard Review Plan and Development of Regulatory Guide

As discussed in NUREG-0933, Issue II.E.1.3 addressed improving the reliability of the auxiliary feedwater system (AFWS) or the EFWS. Section 10.4.9 of the SRP was to be updated, and RG 1.26 was to be revised to include these systems and possibly endorse certain standards. The SRP section was updated in July 1981; however, no additional public and occupational risk reduction was identified to support the need to revise the regulatory guide and it was not revised. This issue is resolved and the requirements were established in the changes to the SRP. However, Westinghouse should discuss how the AP600 design meets the SRP and RG 1.26 for the SUFWS. This was designated as Open Item 20.4-12 in the DSER.

Westinghouse states that this issue was a requirement to update SRP 10.4.9 to address the requirements of Item II.E.1.1 and Item II.F.1.2 for the AFWS. The SRP is written for the safety-related AFWS with a seismic Category 1 water source. The system also functions as an EFWS to remove heat from the primary system when the main feedwater system is not available during emergency conditions. The AP600 does not have an EFWS and does not include a seismic Category 1 water source for either the main or startup feedwater systems. The passive residual heat removal system provides the safety-related function to remove heat from the primary system when the main feedwater is not available. The design criteria for the

SUFWS are predicated on operational and investment protection considerations and not the requirements of SRP 10.4.9 or RG 1.26.

The staff reviewed the AP600 design against Issue II.E.1.3 and finds that the SUFS does not have to meet the requirements of SRP 10.4.9 and RG 1.26 because of design difference. The SUFS is a non-safety system and does not perform the safety function as the EFWS. The staff evaluated the SUFWS using SRP 10.4.9 as guidance and finds it acceptable. Therefore, the staff concludes that DSER Open Item 20.4-12 is closed and Issue II.E.1.3 is resolved for the AP600 design.

Issue II.E.2.2: Research on Small Break LOCAs and Anomalous Transients

As discussed in NUREG-0933, Issue II.E.2.2 addressed the NRC research programs focused on small-break LOCAs (SBLOCAs) and reactor transients. The programs included experimental research in the loss-of-flow tests (LOFT), Semiscale LOFT, Babcock and Wilcox integral systems test facilities, systems engineering, and material effects programs, as well as analytical methods development and assessments in the code-development program.

The programs called for in this issue were completed by the NRC and showed that ECCSs will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single-failure criteria of Appendix K to 10 CFR Part 50. The application of the experimental data from the research programs to validate the conservatism of the licensing codes used in the SBLOCAs are addressed in Issue II.K.3(30) in this section.

Westinghouse did not address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue was resolved with no new requirements.

Because the AP600 design is the first passive ALWR design to be reviewed by NRC, the staff is considering how the research for the non-passive LWRs apply to this design. The distinguishing feature of the AP600 is a dependence on safety systems whose operation is driven by natural forces, such as gravity and stored mechanical energy.

While passive systems may be conceptually simpler than conventional active systems, they may be potentially more susceptible to system interactions that can upset the balance of forces upon which the passive systems depend on for their operation. It should be noted that these "passive" systems still rely on some active operation to place them in operation.

For a design with passive safety systems and without a prototype plant that will be tested over an appropriate range of normal, transient, and accident conditions, the following requirements, are required by 10 CFR 52.47(b)(2)(i)(A):

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.

- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

- Sufficient data exist on the safety features to the design to assess the analytical tools used for safety and analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

Westinghouse developed test programs for the AP600 design to investigate the passive reactor and containment safety systems, including component phenomenological (separate-effects) test, and integral-systems tests. The staff completed and documented its review of the AP600 testing programs in Chapter 21 of this report. Therefore, Issue II.E.2.2 is considered resolved for the AP600 design.

Issue II.E.3.1: Pressurizer Heater Power Supply

Issue II.E.3.1 requires that emergency power be available to ensure that natural circulation can be maintained in the RCS if offsite power is lost, and pressurizer heater motive and control power shall interface with emergency buses through qualified devices.

The safety-related passive core cooling system can establish and maintain natural circulation cooling using the passive residual heat removal heat exchangers, transferring the decay heat to the containment refueling water storage tank water and to the passive containment cooling system without the pressurizer heaters. Pressurizer heaters are not required for safety and do not require power from the Class 1E system.

Therefore, Issue II.E.3.1 is resolved for the AP600 design.

Issue II.E.4.1: Dedicated Hydrogen Penetrations

TMI Action Plan Requirement II.E.4.1 of NUREG-0737 states that plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should have containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of GDC 54 and 56 of Appendix A to 10 CFR, and that are sized to satisfy the flow requirements of the recombiner or purge system. The AP600 design does not use external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere and this issue on dedicated containment penetrations for hydrogen control is not applicable to the AP600 design. Instead, the AP600 design includes passive autocatalytic recombiners (PARs) to meet the requirements of 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors."

The ability of the AP600 design to comply with the requirements of 10 CFR 50.44 is documented in Section 6.2.4 of the SSAR. The staff's evaluation of the ability of the AP600 design to comply with the hydrogen control requirements of 10 CFR 50.44 is documented in Section 6.2.5, "Combustible Gas Control Inside Containment," of this report. Because the AP600 design does not rely on external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere, Issue II.E.4.1 is resolved for the AP600 design.

Issue II.E.4.2: Containment Isolation Dependability

As discussed in NUREG-0933, Issue II.E.4.2 addressed improving the reliability and capability of containment structures to reduce the radiological consequences to the public from accidents, including degraded core events. The issue specifically addressed the need for dependable isolation of containment penetrations.

In Section 1.9.3 of the SSAR, item (2)(xiv), Westinghouse states that the AP600 CIS design satisfies NRC requirements, including post-TMI requirements. It further explains that two barriers are provided, one inside containment and one outside. These barriers are usually valves, but in some cases they are closed, seismic Category 1 piping systems not connected to the RCS or to the containment atmosphere. The design incorporates a reduction in the number of containment penetrations compared to previous plant designs and the majority are normally closed. Those few that are normally open use "automatically closed," failed-close isolation valves. The penetrations do not automatically reopen on the resetting of the isolation signal. Containment isolation is automatically actuated by diverse signals, and can be manually initiated from the MCR. Section 6.2.3 of the SSAR provides additional information.

In the DSER, the staff asked Westinghouse to provide a clearer explanation of how the AP600 design meets the requirements of this issue. The staff stated that the initial discussion in the SSAR was too brief and did not clearly address the requirements identified for this issue. This was Open Item 20.4-13.

The II.E.4.2 requirements are encompassed in the acceptance criteria for SRP 6.2.4. The staff therefore considered the relevant requirements in its review of the containment isolation system. See Section 6.2.4 of this report for additional discussion. The staff concludes that Open Item 20.4-13 is closed and Issue II.E.4.2 is resolved for the AP600 design.

Issue II.E.4.4: Purging

Issue II.E.4.4 served to improve the vent/purge valve isolation reliability of pre-TMI facilities. The vent/purge isolation valve operators at many of those facilities were not originally selected with consideration of torque capability to close against LOCA dynamic forces. Also, II.E.4.4 restricted containment vent/purge operations to safety-related purposes, thus reducing the likelihood that the valves would be open in the event of a LOCA.

The staff asked Westinghouse to provide a clearer explanation of how the AP600 design meets the requirements of this issue. This was Open Item 20.4-14.

Westinghouse states that the AP600 will meet the II.E.4.4 requirement. SSAR Section 6.2.3.1.3.F states that "Isolation valves are designed to have the capacity to close against the conditions that may exist during events requiring containment isolation." TSs will preclude unnecessary venting. Debris screens will be provided to protect the isolation valves from LOCA blowdown debris. Therefore, the staff concludes that Open Item 20.4-14 is closed and Issue II.E.4.4 is resolved for the AP600 design.

Issue II.E.5.1: Design Evaluation

As discussed in NUREG-0933, Issue II.E.5.1 addressed the requirement for Babcock & Wilcox Co. (B&W) licensees to propose recommendations on hardware and procedural changes relative to the need for methods for damping primary system sensitivity to perturbations in the once-through SG. 10 CFR 50.34(f)(2)(xvi) states that a design criterion should be established for the allowable number of actuation cycles of the ECCS and RPS consistent with the expected occurrence rate of severe overcooling events, considering anticipated transients and accidents.

Westinghouse identified in Section 1.9.3 of the SSAR that it considered Issue II.E.5.1 relevant to the AP600 design and stated that although this issue applies only to B&W designs, the AP600 design uses the passive core cooling system to provide emergency reactor coolant inventory control and emergency decay heat removal. Component design criteria have been established for the number of actuation cycles for the passive core cooling system. The identified actuation cycles include inadvertent actuation, as well as the system response to expected plant trip occurrences, including overcooling events. Operation of the ADS is not expected for either design-basis or best-estimate overcooling events. Section 3.9.1 of the SSAR has additional information.

In the DSER, the staff stated that Westinghouse should reference, in Table 1.9-2, its evaluation of this issue in Section 1.9.3 of the SSAR, as it does for other TMI Action Plan items. This was Open Item 20.4-15.

The staff reviewed the updated Table 1.9-2, which provides the status of TMI and USI/GSI related items discussions, including Item II.E.5.1 in Section 1.9 of the SSAR. Therefore, Open Item 20.4-15 is closed and Issue II.E.5.1 resolved for the AP600 design.

Issue II.E.6.1: *In Situ* Testing of Valves, Test Adequacy Study

As discussed in NUREG-0933, Issue II.E.6.1 addressed the adequacy of the requirements for safety-related valve testing. Valve performance is critical to the successful functioning of a large number of plant safety systems. This issue was divided into the following four parts during resolution of the issue. As a result of this division, the resolution of Issue II.E.6.1 was subsumed by the resolution of the following:

(1) testing of PIVs

(2) *in-situ* testing and surveillance of check valves

(3) reevaluation of the thermal-overload protection provisions for MOVs in RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"

(4) Operability verification for MOVs in accordance with GL 89-10

The staff's evaluations of the first two parts above, testing of PIVs and check valves, are discussed in Section 3.9.6.2 in Chapter 3 of this report (reference DSER Open Items 3.9.6.2-7 and 3.9.6.2-8 for PIVs, and DSER Open Items 3.9.6.2-4 and 3.9.6.2-5 for check valves). The

last two parts in the list above were also addressed by the staff in Section 3.9.6.2 of this report as a part of the resolution of Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance" (reference DSER Open Items 3.9.6.2-1 and 3.9.6.2-2). Since the resolution of Issue II.E.6.1 was subsumed by the resolution of the four parts listed above, and based on the staff's evaluations and resolution of these issues as discussed above, the staff concludes that Issue II.E.6.1 is resolved for the AP600 design.

Issue II.F.1: Additional Accident Monitoring Instrumentation

As discussed in NUREG-0933, Issue II.F.1 addressed providing instrumentation to monitor plant variables and systems during and following an accident. The issue addressed the need for plants to include instrumentation to measure, record, and read out in the control room the following containment parameters:

- pressure
- water level
- hydrogen concentration
- high-range radiation
- noble gas effluents

The staff clarified Issue II.F.1 in NUREG-0737 and requirements were issued. The radiation and noble gas effluent instrumentation is required to provide for continuous sampling of radioactive iodine and particulates at all potential accident release points, and for onsite capability to analyze and measure these samples. The acceptance criterion is the guidance in RG 1.97. NUREG-0660 also provides the requirements for a human factors analysis, which is to include the use of the indicators listed above by the operator during normal and abnormal plant conditions, integration of these indicators in plant EOPs and operator training, the use of other alarms, and the need for prioritization of alarms.

In Section 1.9.3 of the SSAR, Item (2)(xvii), Westinghouse states that the AP600 post-accident monitoring system is described in SSAR Chapter 7 and is designed to meet the requirements in NUREG-0737 and the guidelines in RG 1.97.

The system provides indication of the following plant parameters:

- Containment pressure, provided as a Type B key variable (per RG 1.97)

- Containment water level, provided as a Type B key variable

- Containment hydrogen concentration, provided as a Type C key variable

- Containment radiation (high level), provided as a Type B key variable

- Noble gases effluents – containment radiation is provided as a Type B key variable to ascertain RCS integrity and other noble gas effluents are Type E variables

In Section 7.5 of the SSAR, Westinghouse compares the AP600 design against the criteria in Revision 3 of RG 1.97 and addresses accident monitoring instrumentation. To determine whether or not the AP600 design is in compliance with RG 1.97, the staff requested in DSER

Open Item 20.4-16 that Westinghouse provide additional information for the noble gas effluent instrumentation and primary sampling system. COL Action Item 20.4-8 stated that the COL applicant should address the human factors aspects of accident monitoring instrumentation in this issue. Sections 11.5.5 and 9.3.3.1 of the SSAR contain the additional information needed on the design for the noble gas effluent instrumentation and the primary sampling system. This closes DSER Open Item 20.4-16. The human factors aspects of this issue are addressed in Chapter 18 of this report. Accident monitoring instrumentation is discussed in Section 12.4.4 of this report.

Therefore, Issue II.F.1 is resolved for the AP600 design.

Issue II.F.2: Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

10CFR 50.34(f) requires that instruments be provided in the control room that have unambiguous indication of inadequate core cooling (ICC), such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs. NUREG-0737, TMI Action Plan Item II.F.2, discusses the ICC phenomena and the need to have a reactor water level indication system that provides indication of reactor coolant void fraction when the RCPs are operating, and reactor vessel water level when the RCPs are tripped.

Before the TMI accident, an accepted operational practice of PWRs was to operate the RCPs, if they were available, during a LOCA to provide continued core cooling. During the TMI LOCA event with the stuck-open PORVs, the reactor coolant continued to leak through the open valves, the pressurizer level indicated high, and subsequent ICC occurred because the reactor coolant was highly voided. Nevertheless, core cooling was maintained with the continued operation of the RCPs. Subsequently, the RCPs were tripped, and because of high void content in the coolant, the water level dropped below the top of the core causing fuel damage. As a result of the TMI lessons learned, the reactor vessel water level indication system was added, specifically for PWRs, to ensure operator action to trip the RCPs following a LOCA, rather than later in the LOCA sequence to prevent an ICC event. NUREG/CR-5374, "Summary of Inadequate Core Cooling Instrumentation for U.S. Nuclear Power Plants," discusses acceptable approaches to instrumentations used to address ICC.

In response to staff RAI #440.162, Westinghouse explained that the AP600 design concept is different from current operating plants in that the AP600 design automatically trips the RCPs and initiates safeguard injections through the passive safety systems such as core makeup tank (CMT), ADS, passive residual heat removal (PRHR) and IRWST to maintain core cooling in the event of a SBLOCA. It does not rely on a reactor vessel level indication system as do existing reactors, where reactor vessel level indication is important for operator actions to trip the RCPs, to monitor coolant mass in the vessel, and to manually depressurize the RCS in the event of ICC. There is no need in the AP600 for the operator to trip the RCPs, to inject water into the core, or to manually depressurize the plant during a SBLOCA.

The instruments typically used in current PWRs include subcooling margin monitoring capability, core-exit thermocouples, and reactor vessel level indication system, which together would provide the operator with the ability to monitor the coolant conditions and to appropriately

take actions to ensure core cooling during the approach to, and to recover from, the inadequate core cooling conditions. The AP600 design includes subcooling margin monitoring capability, core-exit thermocouples, and the hot-leg level indication system. The AP600 hot-leg level indication system is different from the reactor vessel level indication systems currently used in Westinghouse plants.

The AP600 hot-leg level indication is a safety-related level indication system, which consists of separate pressure taps that connect to the bottom of the hot leg, and to the top of the hot-leg bend leading to the steam generator. This system has the ability to provide indication of reactor water vessel level for a range spanning from the bottom of the hot leg to approximately the elevation of the vessel mating surface.

In addition, during the operation of the ADS to depressurize the plant, the reactor vessel water level will vary greatly and will not provide a reliable indication of ICC. The AP600 hot-leg water level indication is not used to direct operator actions, even when the water level may potentially drop below the hot-leg level. Therefore, the water level is not an important indication for mitigation of ICC in the AP600 design. The hot-leg level indication system is used, however, as a verification of reactor water inventory to terminate the recovery action in the ERGs for the ICC event.

Because the AP600 design automatically trips the RCPs during a SBLOCA event, and because the operators are not prone to be misled by forced two-phase flow, the core exit temperature is an important and sufficient indication of an approach to ICC condition. The temperature reading provided by core-exit thermocouples has been appropriately included in the ERGs for plant recovery.

The staff reviewed the Westinghouse response and determined that for a SBLOCA event, a safeguard signal would automatically trip the RCPs, passive safety systems such as the CMT would automatically inject water into the core, the ADS would automatically initiate to depressurize the plant, the reactor coolant would automatically be cooled by the PRHR, and subsequent injection from the IRWST would occur. The staff also determined that for the AP600 design, the core-exit thermocouples and the subcooling margin monitoring together would provide unambiguous indication of an approach to ICC, and the safety-related hot-leg level indication is only used to terminate the recovery action in the ERGs for the ICC event.

Therefore, the requirements for ICC, as discuss in 10CFR 50.34(f), have been satisfied and Issue II.F.2 is resolved for the AP600 design.

Issue II.F.3: Instrumentation for Monitoring Accident Conditions

As discussed in NUREG-0933, Issue II.F.3 addressed the adequacy and availability of instrumentation that monitors plant variables and systems during and following an accident that includes core damage. Before the TMI-2 accident, nuclear power generating stations were equipped with accident monitoring instrumentation using the guidance identified in RG 1.97 (Revision 1) and ANSI/ANS Standard 4.5, "Criteria for Accident Monitoring Functions in Light Water Cooled Reactors."

The acceptance criterion for the resolution of this issue is that there shall be instrumentation of sufficient quantity, range, availability, and reliability to permit adequate monitoring of plant

variables and systems during and after an accident. Specifically, the instrumentation shall conform to the guidance in RG 1.97 (Revision 3) and ANSI/ANS Standard 4.5 and should provide sufficient information to the operator for (1) taking planned manual actions to shut the plant down safely; (2) determining whether the reactor trip, engineered-safety-feature systems, and manually initiated safety-related systems are performing their intended safety functions (i.e., reactivity control, core cooling, and maintaining RCS and containment integrity); and (3) determining the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, RCPB, and containment) and determining if a gross breach has occurred.

In Section 1.9.3 of the SSAR, Item (2)(xix), Westinghouse states that the AP600 post-accident monitoring system is designed using RG 1.97 as a guidance document. It is a Class 1E microprocessor-based system that provides instrumentation to monitor plant variables and systems during and following an accident. It consists of two independent, electrically isolated, physically separated channels, each channel containing a remote data processing unit cabinet, a data processing unit cabinet, and qualified displays. The remote data processing unit cabinet accepts analog signals from plant sensors, provides signal conditioning, and transmits the conditioned signal via fiber optic "data links" to the data processing unit cabinet. The latter cabinet receives Class 1E signals from the remote data processing unit cabinet and the protection system, and optically isolated non-Class 1E signals from certain non-Class 1E instrumentation and control systems. These signals are processed into accident monitoring formats and presented on qualified displays in the MCR, on the remote shutdown workstation, and to the monitor bus for use by other systems and the dedicated recorders. Additional information is provided in the previous response for Issue II.F.1 above and in Section 7.5 of the SSAR.

In Section 7.5 of the SSAR, Westinghouse states that the post-accident monitoring system is designed to Revision 3 of RG 1.97. However, Westinghouse had not provided the ranges expected during core damage events for certain plant variables, which resulted in DSER Open Item 20.4-17. The process parameters and their ranges are now provided in Table 7.5-1 of the SSAR resolving DSER Open Item 20.4-17. Because this system is stated to comply with Revision 3 of RG 1.97, the AP600 design meets the requirements of this issue. Therefore, Issue II.F.3 is resolved for the AP600 design.

Issue II.G.1: Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators

As discussed in NUREG-0933, Issue II.G.1 addressed upgrading the emergency power for the pressurizer relief and block valves, and pressurizer level indicators. In accordance with the requirements in NUREG-0737, the pressurizer equipment must be supplied from an emergency source of power in the event of a LOOP.

In Section 1.9.3 of the SSAR, Item (2)(xx), Westinghouse states that the AP600 design does not include PORVs and their associated block valves. Pressurizer level indication is provided by instrumentation powered from the Class 1E dc power system. This system provides safety-related, uninterruptable power for Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components essential for safe shutdown of the plant. The system is designed such that these essential plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are not available. SSAR Chapters 7 and 8 have additional information.

The AP600 design does not include power-operated relief valves and their associated block valves from the reactor coolant system. Pressurizer level indication is provided by instrumentation powered from the Class 1E dc power system. The system provides safety-related, uninterruptable power for the Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components that are essential for safe shutdown of the plant.

The Class 1E dc system is designed such that these critical plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are available.

The staff agrees that the AP600 design includes pressurizer equipment that is different from operating PWR plant designs. The pressurizer does not include power-operated relief valves and their associated block valves. It includes, however, the safety-related ADS valves and the pressurizer level indications, which are part of the post-accident monitoring system. Descriptions of these systems are provided, respectively, in Sections 5.4.6 and 7.5 of the SSAR. The pressurizer level and the ADS valve indications are provided by instrumentation powered from the Class 1E dc power system. This system provides safety-related, uninterruptable power for the Class 1E plant instrumentation, control, monitoring, and other vital functions. The Class 1E dc system is designed such that these critical plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are unavailable.

The staff reviewed the ADS design and concluded that the pressurizer equipment, including the ADS valves and pressurizer level indication system, is supplied with the Class 1E dc power and, therefore the requirements of this issue are satisfied.

Therefore, Issue II.G.1 is resolved for the AP600 design.

Issue II.J.3.1: Organization and Staffing to Oversee Design and Construction

As discussed in NUREG-0933, Issue II.J.3.1 addressed requiring license applicants and licensees to improve the oversight of design, construction, and modification activities so that they will gain the critical expertise necessary for the safe operation of the plant. This issue was included in Issue I.B.1.1, "Organization and Management Long-Term Improvements," which was resolved with changes to RG 1.8, "Personnel Selection and Training," and RG 1.33, "Quality Assurance Program Requirements (Operation)."

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because it was an issue that had been superseded by one or more other issues. This is correct in that this issue was included in the resolution of Issue I.B.1.1; however, Westinghouse should have addressed the organization and staff involved with AP600 design. Although the construction of the reactor plant design is a function of the COL applicant; the design of the plant is a function of both the designer and the COL applicant.

In Section 1.9.3 of the SSAR, Item (3)(vii), Westinghouse states that it has developed a management plan for the AP600 project that consists of a "properly" structured organization with open lines of communication, "clearly defined" responsibilities, "well-coordinated" technical

efforts, and "appropriate" control channels. The procedures to be used in the construction, startup, and operation phases of the AP600 are to be provided by the COL applicant.

Westinghouse should list in Table 1.9-2 its discussion of this issue in Section 1.9.3 of the SSAR as it does for other TMI Action Plan items, explain what it means by the adjectives "properly," "clearly defined," "well-coordinated," and "appropriate," which are used in the discussion in Section 1.9.3 of the SSAR, and discuss the QA standards and organization it used for the design of the AP600. This was DSER Open Item 20.4-18.

The organization for the plant beyond the AP600 design, the construction of the plant, and the modification of the plant are outside the scope of design certification for the AP600 design. A part of these concerns involves the organization of the COL applicant; however, the concerns regarding design of the plant outside of the AP600 design and construction do not involve the organization of the site operation. Therefore, the COL applicant will have the responsibility for addressing these concerns as part of the COL licensing process. This is COL Action Item 20.4-9.

QA standards and the organization that Westinghouse used for the design of the AP600 are discussed and found acceptable in Chapter 17 of this report. furthermore, Westinghouse identifies in SSAR Table 1.9-2 that this item is the responsibility of the COL applicant. Therefore, the staff concludes that DSER Open Item 20.4-18 is closed and Issue II.J.3.1 is resolved for the AP600 design.

Issue II.J.4.1: Revise Deficiency Reporting Requirements

As discussed in NUREG-0933, Issue II.J.4.1 addressed assuring that all reportable items are reported promptly to the NRC and that the information submitted is complete. The issue was resolved when new requirements were issued in 10 CFR 21 and 10 CFR 50.55(e), on July 31, 1991 (56 FR 36091).

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because it was an issue that was addressed in another NRC program. This was not an acceptable response. The plant procedures for adequately reporting in accordance with 10 CFR Part 21 and 10 CFR 50.55(e) are outside the scope of AP600 design review. The COL applicant will have the responsibility for having the proper reporting procedures and addressing this issue as part of the licensing process. This is considered a part of the plant procedures development by the COL applicant. This is COL Action Item 20.4-10 and DSER Open Item 20.4-19.

Also, as indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Therefore, the staff concludes that DSER Open Item 20.4-19 is closed and Issue II.J.4.1 is resolved for the AP600 design.

Issue II.K.1(3): Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents

As discussed in NUREG-0933, Issue II.K.1(3) requested licensees to have operating procedures for recognizing, preventing, and mitigating void formation in the RCS during transients and accidents to avoid loss of the core-cooling capability during natural circulation.

The staff reviewed the resolution of Issue I.C.I and its related ERG AES-0.2, "Natural Circulation Cooldown," and has concluded the guidelines direct the operators to cooldown and depressurize the plant using natural circulation conditions by dumping steam and subsequent RNS operation. These steps are specified to preclude any possible upper head voids formation and also to direct the operators to verify that a steam void does not exit in the vessel. The staff concludes that the ERGs provide directions to plant operators to recognize and to preclude voids formation in the vessel. Therefore, the staff considers Issue II.K.1(3) resolved for the AP600 design.

Issue II.K.1(4d): Review Operating Procedures and training to Ensure that Operators are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions

As discussed in NUREG-0933, Issue II.K.1(4d) asked licensees to provide operating procedures to ensure that operators shall not rely on level indication alone in evaluating plant conditions. As stated in NUREG-0933, the staff determined that this issue was covered by Issues I.A.3.1, I.C.1, and II.F.2, and is resolved.

Issue I.A.3.1, "Revise Scope and Criteria for Licensing Examinations," was implemented by NRC by a rule change to 10 CFR Part 55, "Operators Licenses," to require a simulator as part of the reactor operator licensing examinations. The staff will impose the requirements of 10 CFR 55.45 on simulators on the COL applicant referencing the AP600 design; therefore, Westinghouse and the staff does not have to address Issue I.A.3.1 for compliance with 10 CFR 52.47(a)(1)(iv).

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. However, in response to the staff request for additional information, Westinghouse stated that the design portion of this item is addressed in the proposed resolution to Issues I.C.1 and II.F.2.

The staff completed its review of Issues I.C.I and II.F.2 and concluded that AP600 ERGs do not instruct the operators to rely on level indication alone in evaluating plant conditions. The status of core cooling is determined by indications of core-exit thermocouple temperature, RCS subcooling, and RCS hot-leg temperature in addition to RCS level. The staff considers these issues resolved. Therefore, Issue II.K.1(4d) is resolved for the AP600 design.

Issue II.K.1(5): Safety-Related Valve Position Description

As discussed in NUREG-0933, Issue II.K.1(5) addressed the need to (1) review all valve positions and positioning requirements and positive controls, along with all related test and maintenance procedures to assure proper ESF functioning, if required, and (2) verify that

auxiliary feedwater valves are in the open position. This issue was resolved and requirements were issued in NUREG-0737.

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because it is the responsibility of the COL applicant.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse has satisfactorily addressed this item in WCAP-14645. Therefore, Issue II.K.1(5) is resolved for the AP600 design.

Issue II.K.1(10): Review and Modify Procedures for Removing Safety-Related Systems From Service

As discussed in NUREG-0933, Issue II.K.1(10) addressed the requirement that licensees review and modify, as needed, the procedures for removing safety-related systems from service, and restoring them to service, to assure that the operability status of the systems is known.

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because it was the responsibility of the COL applicant.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Therefore, Issue II.K.1(10) is resolved for the AP600 design.

Issue II.K.1(13): Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items

As discussed in NUREG-0933, Issue II.K.1(13) addressed the requirement that operating plants had TSs reflecting the requirements in the bulletins issued by the Commission for the TMI Action Plan. This issue is resolved and a new requirement was established.

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because it was the responsibility of the COL applicant. This was not an acceptable response because Westinghouse was required to address TSs for the AP600 in SSAR Chapter 16. These TSs should include requirements from the TMI Action Plan bulletins and should be addressed in the response to this issue. There may also be a responsibility for the COL applicant because the final TSs for the plant will be the responsibility of the COL applicant. The staff requested that Westinghouse address these items in a response to this issue. This was Open Item 20.4-20.

In SSAR Chapter 1, Westinghouse states that the AP600 TSs (Section 16.1) are predicated on and were reviewed against the Westinghouse Standard TSs, which incorporated the requirements of the bulletins for the TMI Action Plan.

The AP600 TSs are evaluated in Chapter 16 of this report. The staff reviewed the TSs against the Westinghouse standard TSs, which incorporated all the requirements of the bulletins for the TMI Action Plan. Therefore, the approved AP600 TSs incorporate all the appropriate bulletin requirements from the TMI Action Plan. The incorporation of operating experience in bulletins in the AP600 design is discussed in Section 20.7 of this report.

Therefore, Open Item 20.4-20 is closed and Issue II.K.1(13) is resolved for the AP600 design.

Issue II.K.1(16): Implemented Procedures that Identify Pressurizer PORV "Open" Indications and that Direct Operators to Close valve Manually at "Reset" Setpoint

As discussed in NUREG-0933, Issue II.K.1(16) addressed requiring procedures that identified pressurizer PORV "open" indications and directed operators to close the valve manually at the "reset" setpoint. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1 and II.D.3. The resolutions of Issues I.C.1 and II.D.3 for the AP600 design are discussed in this section.

In the SSAR, Table 1.9-2, Westinghouse states that this issue is not applicable to the AP600 design because the issue is the responsibility of the COL applicant.

This issue required procedures for PORVs and the AP600 design does not include these valves. To provide functions equivalent to that of PORVs (releasing noncondensable gases and mitigating consequences of beyond-design-basis events), Westinghouse includes the manually operated safety-related ADS, which is discussed in SSAR Chapter 5. The staff does not have adequate information regarding operational guidance and requirements of the ADS during severe accident events. The staff concluded that detailed operational strategies of the ADS should be included in the ERGs for the AP600 design for the severe mitigating accidents. ERGs are discussed in Issue I.C.1. Westinghouse should provide this information. This was Open Item 20.4-21.

In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the AP600 design. Therefore, Open Item 20.4-21 is closed and Issue II.K.1(16) is resolved for the AP600 design.

Issue II.K.1(17): Trip Pressurizer Level Bistable so that Pressurizer Low Pressure Will Initiate Safety Injection

As discussed in DSER Section 20.4, TMI Action Plan Item II.K.1(17) addresses the requirement for Westinghouse plants to trip the pressurizer level bistable so that the pressurizer low pressure, rather than the pressurizer low pressure and pressurizer low level coincidence, would initiate safety injection.

The AP600 design does not depend on pressurizer low pressure and pressurizer low level coincidence to initiate safety injection in the event of LOCAs. Safety injection in AP600 design is automatic. The safeguard signals that initiate safety injection are Low-1 pressurizer pressure or Hi-1 containment pressure or low compensated steamline pressure or Low-3 cold leg temperature. In addition, the AP600 design also gives the operator manual safety injection capability. The staff concludes that any single safeguard signals mentioned above would

initiate safety injection. Therefore, Open Item 20.4-22 is closed and Issue II.K.1(17) is resolved for the AP600 design.

Issue II.K.1(22): Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW is not Operable

As discussed in NUREG-0933, Issue II.K.1(22) addressed the requirement for BWR plants that auxiliary heat removal systems should be designed such that necessary automatic and manual actions can be taken to ensure proper functioning of the systems when the main feedwater system is not operable.

Westinghouse identifies in Table 1.9-2 of the SSAR that it considers Issue II.K.1(22) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, item (2)(xxi), Westinghouse states that, although this issue was applicable only to BWRs in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980, there are some considerations for the AP600 design. Following a loss of main feedwater (LMFW), there are a number of plant systems that automatically actuate to provide decay heat removal. The non-safety-related SUFS can be powered by the non-safety-related diesel generators, and is automatically actuated and controlled by low SG level. For design-basis events, the safety-related passive core cooling system includes passive residual heat removal heat exchangers, which automatically actuate to provide emergency core decay heat removal if the non-safety-related systems are not available. The MCR meets the NRC guidelines for manual actuation of protective functions, including those used in a LMFW event. Sections 6.3 and 10.4 of the SSAR provide additional information.

On the basis of the staff's review, which is discussed in Section 10.4.9 of this report, the staff concludes that Issue II.K.1(22) is resolved for the AP600 design because the SUFS is automatically actuated and controlled following a LMFW, and the passive core cooling system is automatically actuated if the non-safety-related systems are not available.

Issue II.K.1(24): Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip

Issue II.K.1(24), of NUREG-0933 required PWR licensees to perform a LOCA analysis for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip. The staff determined in NUREG-0933 that this issue for PWRs was covered by Issue I.C.1, "Short-Term Analysis and Procedures Revision."

Westinghouse provided, for staff review, the AP600 ERGs, which address Issue I.C.1. The staff reviewed the responses to Issue I.C.1 and specifically the ERG AE-0, "AP600 Reactor Trip or Safety Injection" for small-break LOCA that addresses Item II.K.1(24). The staff concluded that the AP600 design automatically trips the RCPs during a LOCA event. The guideline directs the operators to verify that all reactor coolant pumps have been tripped, and if not, the operators are directed to manually trip the reactor coolant pumps. On the basis of the plant design features and the appropriate operator actions using the ERGs, the staff considers Open Item 20.4-23 closed and Issue II.K.1(24) resolved for the AP600 design.

Issue II.K.1(25): Develop Operator Action Guidelines

As discussed in NUREG-0933, Issue II.K.1(25) required PWR licensees to develop operator action guidelines on the basis of the analyses performed in response to Issue II.K.1(24), which is discussed above. The staff determined in NUREG-0933 that this issue was covered by Issue I.C.1.

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because the issue has been superseded by one or more other issues. Although this issue was covered by Issue I.C.1, as stated above, Westinghouse also did not address this latter issue because it considered Issue I.C.1 the sole responsibility of the COL applicant.

The final procedures would be the responsibility of the COL applicant; however, the LOCA analyses for a range of time lapses and the specific information to go into the procedures would be the responsibility of the designer, or Westinghouse in the case of the AP600 design. Westinghouse addresses accidents for the AP600 design in SSAR Chapter 15. The staff requested that Westinghouse address operator action guidelines, or ERGs, of I.C.1 and the role of the COL applicant in Issue II.K.1(25). This was Open Item 20.4-24.

The staff completed its review of Issue I.C.1 and concluded that Issue I.C.I is closed. Therefore, Open Item 20.4-24 is also closed and Issue II.K.1(25) is resolved for the AP600 design.

Issue II.K.1(26): Revise Emergency Procedures and Train Reactor Operators (ROs) and Senior Reactor Operators (SROs)

As discussed in NUREG-0933, Issue II.K.1(26) addressed requiring all operating PWRs to revise their EOPs and to train the ROs and SROs for the plant, on the basis of guidelines developed in response to Issue II.K.1(25), which is discussed above. The staff determined in NUREG-0933 that this issue is covered by Issues I.A.3.1, "Revise Scope of Criteria for Licensing Examinations," I.C.1, and I.G.1, "Training Requirements."

As stated in NUREG-0933, Issues I.A.3.1, I.C.1, and I.G.1 have been implemented in the staff review of reactor plant designs and do not have to be addressed by Westinghouse for compliance with 10 CFR 52.47(a)(1)(iv).

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because the issue has been superseded by one or more other issues. Although this issue was covered by Issues I.A.3.1, I.C.1, and I.G.1, as stated above, Westinghouse also did not address these issues because it considered them the sole responsibility of the COL applicant.

Issue I.A.3.1 was to revise the scope of examinations and criteria for licensing examinations and Issue I.G.1 was new training requirements for operators. They are the responsibility of the COL applicant and not the responsibility of Westinghouse in the AP600 design review. This is COL Action Item 20.4-11. Westinghouse subsequently revised SSAR Table 1.9-2 to identify issues I.A.3.1 and I.G.1 as the responsibility of the COL applicant.

The guidelines developed as part of Issue II.K.1(25) are discussed in the previous section. Also, as indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed issue I.C.1 of this item in WCAP-14645. Therefore, Issue II.K.1(26) is resolved for the AP600 design.

Issue II.K.1(27): Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling

As discussed in NUREG-0933, Issue II.K.1(27) addressed requiring PWR licensees to provide analyses and develop guidelines and procedures for an ICC condition. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1 and II.F.2. The resolution of Issues I.C.1 and II.F.2 for the AP600 design are discussed in this section.

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because the issue has been superseded by one or more other issues. Although this issue was covered by Issues I.C.1 and II.F.2, as stated above, Westinghouse also did not address Issue I.C.1 because it considered this issues the sole responsibility of the COL applicant. Westinghouse addressed Issue II.F.2 in SSAR Section 1.9.3.

The staff asked Westinghouse to describe the analyses on ICC conditions that it has performed for the AP600 design and the guidelines that it has developed for the design from these analyses in its response for this issue. Westinghouse addresses accident analyses in SSAR Chapter 15. This was Open Item 20.4-25.

Westinghouse submitted AP600 ERGs for staff review, and also responded to staff DSER Open Item 20.4-17 to address Action Items I.C.1 and II.F.2, respectively. The staff reviewed Westinghouse's response to Action Item II.F.2 and a detailed discussion of this item is documented in this section. In the AP600 ERG, Westinghouse provides high-level guidance to deal with inadequate core cooling conditions. The staff reviewed AFR-C.1, "AP600 Response to Inadequate Core Cooling Procedure and Analysis Bases," which describes how passive safety-related systems would automatically trip the RCS pumps, and depressurize the RCS to inject water into the core upon receiving a safeguard signal. In this procedure, the operators are instructed to monitor plant conditions using core-exit temperature and indicated hot-leg level, which is designed to provide indication of an approach to ICC and to recover from an ICC condition. The operators are also instructed to manually initiate injection when automatic passive safety injections fail. Passive safety-related system actuation indications of CMT, ADS, PRHR, and IRWST are integrated into the procedures, which provide operators with directions to ensure that adequate core cooling will be maintained. The staff concludes that Westinghouse has appropriately provided analyses and procedures to mitigate ICC conditions. Therefore, Open Item 20.4-25 is closed and Issue II.K.1(27) is resolved for the AP600 design.

Issue II.K.1(28): Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required

As discussed in NUREG-0933, Issue II.K.1(28) addressed the requirement that PWRs be designed to ensure automatic RCP trip for all circumstances where required. The staff

determined in NUREG-0933 that this issue was covered by Issue II.K.3(5), "Automatic Trip of Reactor Coolant Pumps During Loss-of-Coolant Accident."

As stated in NUREG-0933, Issue II.K.3(5) has been implemented in the staff review of reactor plant designs and does not have to be addressed by Westinghouse and the staff for compliance with 10 CFR 52.47(a)(1)(iv).

Westinghouse did not address this issue in its SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because the issue had been superseded by one or more other issues. Although this issue was covered by Issue II.K.3(5), as stated above, Westinghouse also did not address Issue II.K.3(5) because it considered this issue the sole responsibility of the COL applicant.

The design to provide an automatic RCP trip, such as during a LOCA, required to be addressed by PWR licensees in Issue II.K.3(5) is the responsibility of the designer, in this case Westinghouse for the AP600 design and Westinghouse should address Issue II.K.3(5). See the discussion on Issue II.K.3(5) later in this section.

On the basis of the approved resolution of Issue II.K.3(5) for the AP600 design, as discussed in this section, Issue II.K.1(28) is resolved for the AP600 design.

Issue II.K.2(10): Hard-Wired Safety-Grade Anticipatory Reactor Trips

As discussed in NUREG-0933, Issue II.K.2(10) addressed the requirement for B&W plants to provide a design and schedule for implementation of a safety-grade reactor trip on loss of main feedwater, turbine trip, and significant reduction in SG level. These requirements were listed as Item 5 in BL 79-05B, which was issued on April 21, 1979. This issue was resolved and new requirements were issued.

In Section 1.9.3 of the SSAR, Item (2)(xxiii), Westinghouse states that this issue is applicable to only B&W plants, but that the AP600 trip logic includes an anticipatory reactor trip for loss of main feedwater by using low SG water level. It also states that Section 7.2 of the SSAR has additional information.

Westinghouse further states that, because the AP600 design does not include PORVs and block valves, the anticipatory reactor trip on a turbine trip is not needed. The staff agrees with the Westinghouse statements and considers Issue II.K.2(10) resolved for the AP600 design.

Issue II.K.2(16): Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power

As discussed in NUREG-0933, Issue II.K.2(16) addressed, for B&W plants, the requirement in NUREG-0737 and NUREG-0660 that the licensees investigate the consequences of losing coolant through the RCP seals during loss of offsite power.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.2(16) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, Item (1)(iii), Westinghouse states that the AP600 design uses canned motor RCP pumps. This design does not have a seal that can fail and initiate RCS leakage.

This issue required licensees to perform an evaluation of the likelihood and consequences of reactor coolant pump seal damage following a small-break LOCA with a loss of offsite power. The staff determined that this issue is covered by Issue 23, "Reactor Coolant Pump Seal Failures." The staff approved the resolution of Issue 23 for the AP600 design in Section 20.3 of this report. Therefore, Issue II.K.2(16) is resolved for the AP600 design.

Issue II.K.3(1): Install Automatic PORV Isolation System and Perform Operational Test

As discussed in NUREG-0933, Issue II.K.3(1) addressed for PWR operating plants the requirement in NUREG-0737 and NUREG-0660 to provide a system that uses the PORV block valves to protect against small-break LOCAs. This system would automatically cause the block valves to close when the RCS pressure decays after the PORV has opened.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(1) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, Item (1)(iv), Westinghouse states that the AP600 design does not include PORVs. The pressurizer volume is about 30 percent larger than the pressurizer volume in current PWRs with a comparable power rating. This larger volume increases transient operation margins and prevents safety valve actuation in most accident situations. The AP600 pressurizer surge line is also larger to permit a more rapid transfer of coolant between the RCS and the pressurizer, and to accommodate the automatic depressurization system first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge (Condition II loss of load transient) to prevent exceeding the maximum RCS pressure limit.

It further states that overpressure protection is provided by two totally enclosed pop-type safety valves. They are spring-loaded and self-actuated, with back pressure compensation, and are designed to ASME Code, Section III. If the pressurizer pressure exceeds the setpoint, the safety valves start lifting. A temperature indicator in the discharge piping for each safety valve alarms on high temperature downstream of the valve. The AP600 design includes an automatic depressurization system consisting of six parallel sets of two valves in series connected to the pressurizer and two parallel sets of two valves in series, with one set connected to each RCS hot leg.

Westinghouse also states that a PRA analysis was used to determine the probability of a LOCA caused by failure of the automatic depressurization system. Results of this analysis were stated to be factored into the design process. SSAR Chapter 5 and Section 6.3 have additional information.

On the basis of this information, Issue II.K.3(1) is resolved for the AP600 design.

Issue II.K.3(2): Report on Overall Safety Effect of PORV Isolation System

As discussed in NUREG-0933, Issue II.K.3(2) addressed requiring PWR licensees to document the actions to be taken to decrease the probability of a SBLOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce challenges to the safety valves during a design-basis event. The requirements were issued in NUREG-0737 and NUREG-0660.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(2) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, Item (1)(iv), Westinghouse states that the AP600 design does not include PORVs. The pressurizer volume is about 30 percent larger than the pressurizer volume in current PWRs with a comparable power rating. This larger volume increases transient operation margins and prevents safety valve actuation in most accident situations. The AP600 pressurizer surge line is also larger to permit a more rapid transfer of coolant between the RCS and the pressurizer, and to accommodate the automatic depressurization system first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge (Condition II loss of load transient) to prevent exceeding the maximum RCS pressure limit.

It further states that overpressure protection is provided by two totally enclosed pop-type safety valves. They are spring-loaded and self-actuated, with back pressure compensation, and are designed to ASME Code, Section III. If the pressurizer pressure exceeds the setpoint, the safety valves start lifting. A temperature indicator in the discharge piping for each safety valve alarms on high temperature downstream of the valve. The AP600 design includes an automatic depressurization system consisting of six parallel sets of two valves in series connected to the pressurizer and two parallel sets of two valves in series, with one set connected to each RCS hot leg.

Westinghouse also states that a PRA analysis was used to determine the probability of a LOCA caused by failure of the automatic depressurization system. Results of this analysis were stated to be factored into the design process. SSAR Chapter 5 and Section 6.3 have additional information.

This issue required applicants to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. The design purpose of PORVs is to prevent RCS overpressure and to reduce challenge to spring-operated safety valves for design-basis events. Older Westinghouse plants have PORVs but the AP600 design does not have PORVs. This issue is, therefore, not applicable to the AP600 design.

To satisfy the staff requirements, Westinghouse addresses the AP600 pressurizer design, which has about 30 percent larger volume than the pressurizer in current plants. The larger pressurizer increases transient operation margins and prevents safety valve actuation in most accident situations. The pressurizer surge line is also larger to permit a more rapid transfer of coolant between the RCS and the pressurizer, and also to accommodate the ADS first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge to prevent exceeding the maximum RCS pressure limit. The ADS valves are dc power and air

operated, and will actuate on the core makeup tank low level signals. The ADS system uses two-out-of-four logic to prevent inadvertent valve actuations. SSAR Chapters 5 discusses the design and operations of the ADS.

On the basis of the above discussion, Issue II.K.3(2) is resolved for the AP600 design.

Issue II.K.3(5): Automatic Trip of Reactor Coolant Pumps During LOCA

As discussed in NUREG-0933, Issue II.K.3(5) addressed requiring PWR licensees to study the need for an automatic trip of the RCPs and to modify plant procedures or the design, as appropriate. Licensees should know how to operate the RCPs in order to mitigate transients and accidents. The preservation of the maximum RCS inventory should be considered in the SBLOCA mitigation and the most effective DHR strategy should be considered in the mitigation of other transients.

Initially, Westinghouse stated that this issue was not applicable to the AP600 design because the issue is the responsibility of the COL applicant. The staff believes, however, that the design to provide an automatic RCP trip, as during a LOCA, which was required to be addressed by PWR licensees in Issue II.K.3(5), is the responsibility of the designer. The staff asked Westinghouse to address the RCP pump trip in its response to this issue. This was Open Item 20.4-26.

Westinghouse added a discussion, in Section 1.9.4.2.1 of the SSAR, in which they state that the AP600 design provides for an automatic trip of the RCPs on actuation of the passive core cooling system. This trip is provided to prevent reactor coolant pump interaction with the operation of the core makeup tank. SSAR Section 6.3 provides additional information.

On the basis of this information, Open Item 20.4-26 is closed and Issue II.K.3(5) is resolved for the AP600 design.

Issue II.K.3(6): Instrumentation to Verify Natural Circulation

As discussed in NUREG-0933, Issue II.K.3(6) addressed requiring licensees to provide instrumentation to verify natural circulation during transient conditions. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1, II.F.2, and II.F.3.

Westinghouse provided the staff with pertinent information about the AP600 design, which addresses TMI Action Items I.C.1, II.F.2 and II.F.3. The staff reached a conclusion that those issues relevant to the resolution of the TMI Action Item II.K.3(6) have been resolved. The detailed discussion of the related issues are addressed in their respective TMI item discussions. Therefore, Issue II.K.3(6) is resolved for the AP600 design.

Issue II.K.3(8): Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of Steam Generator

As discussed in NUREG-0933, Issue II.K.3(18) addresses further staff consideration of the need for diverse decay heat removal methods, which are independent of the steam generators. The staff determined in NUREG-0933 that this issue was covered by Issues II.C.1, "Interim

Reliability Evaluation Program," and II.E.3.3, "Coordinated Study of Shutdown Heat Removal Requirements." In NUREG-0933, the staff also stated that Issue II.E.3.3 was addressed in Issue A-45, "Shutdown Decay Heat removal Requirements."

In SSAR Table 1.9-2, Westinghouse states that this issue is not applicable to the AP600 design because the issue is superseded by other issues. This is correct and, as stated in NUREG-0933, Issues A-45, II.C.1, and II.E.3.3 have been implemented in the staff review of reactor plant designs and do not have to be addressed by Westinghouse and the staff for compliance with 10 CFR 52.47(a)(1)(iv). However, Westinghouse should provide information regarding the diverse decay heat removal methods independent of the steam generators for the AP600 design. This was Open Item 20.4-27.

Westinghouse submitted the AP600 shutdown evaluation report for staff review. The report describes multiple decay heat removal capabilities independent of the steam generator. The detailed discussion of the multiple decay heat capabilities is included in Chapter 19.3 of this report. The staff, therefore, concludes that Open Item 20.4-27 is closed and Issue II.K.3(8) is resolved for the AP600 design.

Issue II.K.3(9): Proportional Integral Derivative Controller Modification

As discussed in NUREG-0933, Issue II.K.3(9) addressed requiring Westinghouse plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs. The requirements were issued in NUREG-0737 and NUREG-0660. This issue is similar to Issue II.K.1(17) discussed above in this section.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(9) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

Westinghouse addresses this issue in Section 1.9.4.2.1 of the SSAR where it states that this issue is not applicable to the AP600 design because the design does not have PORVs. Additional information is provided in Sections 5.1.2 and 5.2.2 of the SSAR.

The staff completed its review and considers Issue II.K.3(9) resolved for the AP600 design.

Issue II.K.3(18): Modification of ADS Logic – Feasibility Study and Modification for Increased Diversity for Some Event Sequences

As discussed in NUREG-0933, Issue II.K.3(18) addressed requiring BWR plants to modify the ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. A feasibility study and risk assessment study were required to determine the optimum approach. The requirements were issued in NUREG-0737 and NUREG-0660.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(18) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, Item (1)(vii), Westinghouse states that this issue is applicable to the AP600 design because the design uses an ADS with some similarity to that used on BWRs.

The ADS automatically actuates on Low-1 CMT level, coincident with a CMT actuation signal. It is stated that manual actuation of the ADS is not required to maintain core cooling. As discussed in this section in Issue II.B.8 regarding degraded core accidents, the AP600 PRA analysis confirms the reliability of the automatic ADS actuation. In DSER Open Item 20.4-28, the staff requested that the specific PRA section confirming the reliability of the automatic ADS actuation be provided. This information is incorporated throughout the PRA analysis and is evaluated by the staff in Section 19.1 of this report. Therefore, DSER Open Item 20.4-28 is closed. Section 6.3 of the SSAR provides additional information.

The actuation of ADS stages 2 and 3 occur on a set time delay after the actuation of the first stage, as discussed above. ADS stage 4 actuates on a lower CMT level. Therefore, the staff agrees that manual actuation of the ADS is not required to maintain core cooling.

On the basis of the staff review of the ADS design discussed in Section 6.3 of this report, Issue II.K.3(18) is resolved for the AP600 design.

Issue II.K.3(25): Effect of Loss of AC Power on Pump Seal

As discussed in NUREG-0933, Issue II.K.3(25) addressed requiring that BWR licensees determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the RCP seal coolers. Adequacy of the seal design to withstand a LOOP should be demonstrated. This position should prevent excessive loss of RCS inventory following an anticipated operational occurrence. The requirements for this issue in NUREG-0737 are that the consequences of a loss of cooling water to the pump seal coolers be determined, and that the pump seals should be designed to withstand a complete LOOP for at least 2 hours. If seal failure is the consequence of loss of cooling water for 2 hours, an acceptable solution would be providing emergency power to the seal cooling pump.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(25) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In Section 1.9.3 of the SSAR, Item (1)(iii), Westinghouse states that the AP600 design uses canned motor RCP pumps. The canned motor pump design does not have a seal that can fail and initiate RCS leakage.

This issue required licensees to determine, on a plant-specific basis, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The demonstrated adequacy of the seal design to withstand a loss of offsite power should prevent excessive loss of reactor coolant system inventory following an anticipated operational occurrence. The staff determined that this issue was covered by Issue 23, "Reactor Coolant Pump Seal Failures," which is discussed in Section 20.3 of this report.

On the basis of the approved resolution of Issue 23 for the AP600 design in Section 20.3 of this report, Issue II.K.3(25) is resolved for the AP600 design.

Issue II.K.3(28): Study and Verify Qualification of Accumulators on ADS Valves

As discussed in NUREG-0933, Issue II.K.3(28) addressed requiring assurance from BWR licensees that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. The requirements were issued in NUREG-0737 and NUREG-0660.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue II.K.3(28) relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In the DSER, the staff requested that Westinghouse provide the number of cycles the ADS valves may be opened at design pressure in the SSAR. This was designated as Open Item 20.4-29. However, after the DSER was issued, Westinghouse changed the design of the ADS valves to not rely on air or nitrogen accumulators. In Section 9.3.1 of the SSAR, Westinghouse states that there are no safety-related air-operated valves that rely on safety-related air accumulators to actuate to the fail-safe position upon loss of air pressure.

In Section 1.9.3 of the SSAR, Item (1)(x), Westinghouse states that although this issue is identified as applicable to BWRs only, the AP600 uses a safety-related automatic depressurization system that is different from that presently used on BWRs. The AP600 automatic depressurization system uses safety-related dc motor-operated valves and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. There is no non-safety-related equipment or instrumentation, including instrument air or nitrogen supply, relied on in the operation of these valves. These valves are designed and qualified to function in the conditions of an accident. They will also be subject to preoperational and inservice testing. They will be included in the reliability assurance program. Therefore, Open Item 20.4-29 is closed and Issue II.K.3(28) is resolved for the AP600 design.

Issue II.K.3(30): Revised SBLOCA Methods to Show Compliance with 10 CFR Part, Appendix K

As discussed in NUREG-0933, Issue II.K.3(30) required licensees to revise and submit analytical methods for small-break LOCA analyses for compliance with Appendix K to 10 CFR Part 50 for NRC review and approval. The revision was to account for comparisons with experimental data, including data from LOFT test and Semiscale test facilities. Alternatively, licensees were to provide additional justification for the acceptability of their SBLOCA models with LOFT and Semiscale test data. Clarifications were issued in NUREG-0737. The staff reviewed the Westinghouse NOTRUMP code and documented its discussions in Chapters 15 and 21 of this report. The staff, therefore, considers Issue II.K.3(30) resolved for the AP600 design.

Issue III.A.1.2: Upgrade Licensee Emergency Support Facilities

As discussed in NUREG-0933, Issue III.A.1.2 addressed requiring licensees to upgrade their emergency support facilities by establishing a technical support center (TSC), an operational support center (OSC), and a nearsite emergency operations facility (EOF) for command and control, support, and coordination of onsite and offsite functions during reactor accident situations. This issue was resolved and new requirements were issued in NUREG-0737 and

GL 82-33, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability," December 17, 1982.

Westinghouse identifies, in Table 1.9-2 of the SSAR, that it considers Issue III.A.1.2 relevant to the AP600 design. In Section 1.9.3 of the SSAR, Item (2)(xxv) Westinghouse states that the AP600 design provides for an onsite TSC, which is discussed SSAR Chapter 18. The offsite EOF and onsite OSC are stated to be site-specific and the responsibility of the COL applicant. The interface requirements are also discussed in SSAR Chapter 18.

Descriptions of the functions and location of the TSC are provided in Sections 1.2 and 18.2.1.1.2.5 of the SSAR. Westinghouse states that the nearsite EOF and onsite OSC is outside the scope of the AP600 design. The staff agrees that the EOF is the responsibility of the COL applicant and this is encompassed in COL Action Items 13.3-1 and 18.2-2. However, the staff asked Westinghouse to provide for the onsite OSC in the AP600 design because Section 18.2.1.1.2.6 of the SSAR describes the functions and location of the OSC. This should be reflected in Section 1.9.3 of the SSAR, Item (2)(xxv), to demonstrate resolution of this issue for the AP600 design. This was Open Item 20.4-30.

A description of the functions and location of the TSC and OSC are provided in Sections 18.2.1.1.2.5 and 18.2.1.1.2.6, respectively of the SSAR. Figures 1.2-18 and 1.2-19 indicate the location of the OSC and TSC, respectively. Westinghouse states that the nearsite EOF is the responsibility of the COL applicant and this is encompassed in COL Action Item 13.3-1. Therefore, Open Item 20.4-30 is closed and Issue III.A.1.2 is resolved for the AP600 design.

Issue III.A.3.3: Install Direct Dedicated Telephone Lines and Obtain Dedicated Short-Range Radio Communication Systems

As discussed in NUREG-0933, Issue III.A.3.3 addressed the need for licensees to upgrade their communications capability at the emergency support facilities at the plant listed in Issue III.A.1.2. Relevant guidance is contained in NUREG-0660.

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because this issue is covered by another NRC program. This is not correct in that the emergency preparedness program is the only program the staff is aware of that covers this issue. However, the staff considers this issue outside the scope of the AP600 design certification and will be addressed by the COL applicant. This is designated COL Action Item 20.4-12.

The staff requested that Westinghouse address the responsibility of the COL applicant in the resolution of this issue for the AP600 design. This was Open Item 20.4-31.

Westinghouse addresses the responsibility of the COL applicant for communications interfaces among the MCR, the TSC and the emergency planning centers in Section 13.3 of the SSAR. This issue is identified in the SSAR as COL Action Item 13.3-1. Therefore, Open Item 20.4-31 is closed and Issue III.A.3.3 is resolved for the AP600 design.

Issue III.D.1.1: Primary Coolant Sources Outside the Containment Structure

Westinghouse identified in Table 1.9-2, of its May 28, 1993, letter that it considered Issue III.D.1.1 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue III.D.1.1 addressed the requirement that licensees identify design features to reduce the potential for exposure to workers at plants and to offsite populations from the release of primary coolant following an accident. This issue has three subissues:

- III.D.1.1(1), "Review Information Submitted by Licensees Pertaining to Reducing Leakage From Operating Plants"
- III.D.1.1(2), "Review Information on Provisions for Leak Detection"
- III.D.1.1(3), "Develop Proposed System Acceptance Criteria"

In NUREG-0737, Subissue III.D.1.1(1) required licensees to implement a program to reduce leakage from systems outside the containment that would or could contain highly radioactive fluids during a serious transient, or following an accident, to as-low-as-practical levels.

For Subissue III.D.1.1(2), the staff also stated in NUREG-0933 that Issue II.F.1 addressed accident monitoring instrumentation and that the RCPB leak detection capability must be equivalent to that specified in RG 1.45. Issue II.F.1 is addressed for the AP600 design earlier in this section.

The need for requiring leak-detection systems and the development of new acceptance criteria for these systems in Subissue III.D.1.1(3) were pursued by the staff in other issues, as Subissue III.D.1.1(2). Therefore, work on Subissue III.D.1.1(3) did not provide any data for staff consideration and this issue was dropped from further consideration.

In Section 1.9.3 of the SSAR, Item (2)(xxvi), Westinghouse states that the safety-related passive systems for the AP600 design do not recirculate radioactive fluids outside the containment following an accident. A non-safety-related system can be used to recirculate coolant outside of containment following an accident, but this system is not operated when high containment radiation levels exist. Therefore, DSER Open Item 20.4-32 is closed and Issue III.D.1.1 is resolved for the AP600 design.

Issue III.D.3.3: In-Plant Radiation Monitoring

10 CFR 50.34(f)(2)(xxvii) (III.D.3.3) states that the licensee shall "provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

The AP600 will be provided with area and airborne radiation monitors to supplement the personnel and area radiation survey provisions of the AP600 health physics program described in Section 12.5 of the SSAR. These area and airborne radiation monitors, which are described

in Section 11.5 of the SSAR, will comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and RGs 1.97, 8.2, 8.8, and 8.12.

In addition, NUREG-0737, Item III.D.3.3 states that "each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident." Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments. Because the description of portable instrumentation, training, and procedures is outside the scope of the AP600 SSAR, Westinghouse addressed this as a COL item. In Section 12.3.5 of the SSAR, Westinghouse states that the COL applicant will address the criteria and methods for obtaining representative measurements of radiological conditions, including airborne radioactivity in work areas. In addition, the COL applicant will address the use of portable instruments and the associated training and procedures to accurately determine the airborne iodine concentrations in areas within the facility where plant personnel may be present during an accident.

The information on in-plant radiation monitoring in Chapter 12 of the AP600 SSAR addresses the requirements of 10 CFR 50.34(f)(2)(xxvii) (III.D.3.3) and the staff's concerns in this area are resolved. Therefore, Issue III.D.3.3 is resolved for the AP600 design.

Issue III.D.3.4: Control Room Habitability

As discussed in NUREG-0933, Issue III.D.3.4 addressed upgrading the habitability of the control room for the operators. The requirements were provided in NUREG-0737.

In Section 1.9.3 of the SSAR, item (2)(xxviii), Westinghouse states that normally a non-safety-related HVAC system keeps the AP600 MCR slightly pressurized to prevent infiltration of air from other plant areas. During accident conditions, a safety-related isolation of the MCR is automatically actuated. Upon the loss of non-safety-related ac power, the MCR environment is sufficient to protect the operators and support the man-machine interfaces necessary to establish and maintain safe-shutdown conditions for the plant following postulated design-basis accident conditions.

The MCR is stated to be sealed with safety-related connections to a safety-related compressed air breathing source. This compressed air system provides continued pressurization and a source of fresh air for operator habitability. The air supply is sized to last for 72 hours following an accident. The onsite non-safety-related normal HVAC system will be operational before the installed compressed air supply is exhausted.

It is further stated that the non-safety-related HVAC system, equipped with a refrigeration-type air conditioning unit and powered from the onsite diesel generators, normally provides MCR cooling. If the normal HVAC system is not available, outside air is not allowed into the MCR, and the safety-related compressed air storage system is actuated.

In the DSER, the staff requested Westinghouse to address the possibility of toxic gases and substances onsite and offsite affecting control room habitability; the signals, or procedures and operator action for actuation of equipment for control room habitability, and the responsibility of the COL applicant. In addition, the staff requested that Westinghouse discuss the potential

exposure of operators to radiation brought into the control room after the compressed air supply is exhausted, the filtration provided by the HVAC, and dose limits. This was designated as DSER Open Item 20.4-33. DSER Open Item 20.4-33 is closed because the staff has determined that the AP600 design meets the dose limits of GDC 19, except for the use of the TEDE criteria and the TID-14844 source term. The justification for exempting the AP600 design from these requirements is provided in Sections 15.3 and 20.6. Also, Section 6.4.7 of the SSAR states that the COL applicant referencing the AP600 design is responsible for (1) the amount and location of possible sources of toxic chemicals in or near the plant, (2) provision for seismic Category 1, Class 1E toxic gas monitoring, and (3) evaluating the conformance of the onsite and offsite toxic releases with the guidelines of RGs 1.78 and 1.95 in order to meet the TMI Action Plan Item III.D.3.4 and GDC 19 requirements.

Therefore, Issue III.D.3.4 is resolved for the AP600 design.

20.5 Human Factors Issues

The resolution of the human factors issues identified in NUREG-0933 for the AP600 design are discussed in detail in Chapter 18 of this report and are mentioned briefly below. These human factors issues were taken from NUREG-0985, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," (Revision 2) dated April 1986. In Chapter 18 of this report, the staff presents its evaluation of the HFE for the AP600 design.

Issue HF1.1: Shift Staffing

This issue addressed ensuring that the numbers and capabilities of the staff at nuclear power plants are adequate to operate the plant safely. This issue was to determine the minimum appropriate shift crew staffing composition. To meet this goal, consideration was given to the following:

- the number and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode
- the minimum qualifications of plant personnel in terms of education, skill, knowledge, training experience, and fitness for duty
- appropriate limits and conditions for shift work including overtime, shift duration, and shift rotation

The review criteria for this issue are contained in the 10 CFR 50.54, SRP Sections 13.1.2 through 13.1.3, and RG 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit."

Westinghouse does not address this issue in the SSAR. It concludes, in Table 1.9-2, that this issue is not relevant to the AP600 design because this issue is the responsibility of the COL applicant.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Therefore, Issue HF1.1 is resolved for the AP600 design.

Issue HF4.1: Inspection Procedure for Upgraded Emergency Operating Procedures

As discussed in NUREG-0933, Issue HF4.1 addressed the development of criteria by the NRC to provide assurance during inspections that operating plant EOPs are adequate and can be used effectively. Lessons learned by the staff from its inspections of EOPs at plants were published in NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," April 1989. Temporary Instruction (TI) 2515/92, "Emergency Operating Procedures Team Inspections, " was later issued containing guidance for conducting these inspections. The issue was resolved with no new requirements.

Westinghouse initially identified, in Table 1.9-2 of the SSAR, that it considered Issue HF4.1 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 10 CFR 52.47(a)(1)(ii) or (iv).

In SSAR Section 1.9.4, Westinghouse states that the design of the AP600 EOPs is consistent with NUREG-1358 and its supplements, as well as current regulatory guidance and standards. Section 18.9.8 of the SSAR has additional information.

The staff asked Westinghouse to address the regulatory "guidance and standards" that it used to write the EOPs for the AP600 design in addressing this issue. This was Open Item 20.5-1.

This issue is covered in SSAR Section 18.9, “Procedure Development,” and is a COL responsibility. Therefore, Open Item 20.5-1 is closed and Issue HF4.1 is resolved for the AP600 design.

Issue HF4.4: Guidelines for Upgrading Other Procedures

As discussed in NUREG-0933, this issue addresses efforts by the staff to evaluate the quality of, and the problems associated with, existing plant procedures to ensure that plant procedures (other than EOPs which are discussed in Issue HF4.1 above) are adequate and could be used effectively, and to guide operators in maintaining plants in a safe state under all operating conditions. The NRC was to evaluate the need to develop technical guidance for use by industry in upgrading normal operating procedures and abnormal operating procedures. The objective of this issue is to be met by (1) developing guidelines for preparing and criteria for evaluating normal operating procedures and other procedures that affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures. This issue is not resolved and has a high priority in NUREG-0933.

The review criteria for this issue are in SRP Sections 13.5.1, "Administration Procedures," and 13.5.2, "Operating and Maintenance Procedures," and in Information Notice (IN) 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures."

Westinghouse did not initially address this issue in the SSAR. It concluded, in Table 1.9-2, that this issue was not relevant to the AP600 design because this issue was resolved with no new requirements. However, as discussed above, this is not correct.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. In addition, this item is covered by Element 8, Procedures Development, of NUREG-0711. Therefore, Issue HF4.4 is resolved for the AP600 design.

Issue HF5.1: Man-Machine Interface – Local Control Stations

As discussed in NUREG-0933, Issue HF5.1 addressed assuring that the man-machine interface at local control stations and auxiliary operator interfaces is adequate for the safe operation and maintenance of a nuclear power plant. The concerns associated with this issue include the assurance that indications and controls made available to operators at local control stations outside of the control room and remote shutdown room are sufficient and appropriate for their intended use. The regulatory guidance has been limited to the control room and the remote shutdown panel. Control room crew activities should be analyzed to establish and describe communication and control links between the control room and the auxiliary control stations. Additionally, the potential impact of auxiliary personnel on plant safety should be analyzed. This issue was resolved and no new requirements were established.

In Section 1.9.4 of the SSAR, Westinghouse states that it uses techniques and experience gained in the design of the MCR and remote shutdown panel on the local control station panels. The methodology to analyze the job/tasks of the control room is stated to be applied to the job/tasks of auxiliary personnel to identify and describe communication and action links between the control room and the auxiliary control stations.

As shown in DSER Section 18.3, Element 2, Operating Experience Review, the staff had not completed its review of this issue and therefore this issue was part of Open Item 18.3.3.1-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645 and Open Item 18.3.3.1-2 is closed. Therefore, Issue HF5.1 is resolved for the AP600 design.

Issue HF5.2: Review Criteria for Human Factors Aspects of Advance Controls and Instrumentation

As discussed in NUREG-0933, Issue HF5.2 addressed the use of advanced I&Cs, in particular with respect to plant annunciators. The then-existing human engineering guidelines for control rooms addressed the control, display, and information concepts and technologies that were being used in process control systems. These guidelines were not believed to be adequate for advanced and developing technologies that could be introduced into future designs. Improved alarm systems using advanced technologies were expected to become available, and guidelines for the use and evaluation of these longer-term alarm improvements were to be developed. This issue focused on the potential risk that could result from the human error in the use of control room alarms. Work on this issue was stopped when the development of review guidance for advanced alarms was integrated into the Office of Nuclear Regulatory

Research (RES) program to develop an "Advanced Human-Interface Design Review Guideline." This issue is resolved with no new requirements.

In Section 1.9.4 of the SSAR, Westinghouse states that the AP600 advanced alarm design, described in Section 18.9.2 of the SSAR, conforms with current guidance and requirements on integrated human factors design. A description of the computerized procedures is in Section 18.9.8.6 of the SSAR. A detailed description of the qualified display processing system is in Section 18.9.5 of the SSAR. The plan for the V&V of the AP600 man-machine interface system (M-MIS) is in Section 18.8.2.3 of the SSAR.

The staff asked Westinghouse to identify and discuss the "current guidance and requirements on integrated human factors design" it used to design the advanced alarm system for the AP600 design. The relationship of the computerized procedures and qualified display processing system to the advanced alarm system should also be explained. This was Open Item 20.5-2.

As indicated in Section 18.3, Element 2, Operating Experience Review, of this report, Westinghouse satisfactorily addressed this item in WCAP-14645. Therefore, Open Item 20.5-2 is closed and Issue HF5.2 is resolved for the AP600 design.

20.6 Three Mile Island Action Plan Requirements

Pursuant to 10 CFR 52.47(a)(ii), an applicant for design certification must demonstrate compliance with any technically relevant TMI requirements in 10 CFR 50.34(f). The relevant TMI Action Plan items, the 10 CFR 50.34(f) requirements, and the section in which they are addressed are listed in Table 20.6-1 of this report.

The TMI requirements in 10 CFR 50.34(f)(2)(vii), (viii), (xxvi), and (xxviii) refer to the accident source term in TID 14844. Specifically, 10 CFR 50.34(f)(2)(xxviii) requires the evaluation of pathways that may lead to control room habitability problems "under accident conditions resulting in a TID 14844 source term release." Similar wording appears in requirements (vii), (viii), and (xxvi). Westinghouse has adopted the new source term technology summarized in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995, not the old TID 14844 source term cited in 10 CFR Part 50.34(f). Based on the NRC staff's review, as discussed in Section 15.3 of this report, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose because Westinghouse has adopted acceptable alternatives that accomplish the intent of the regulations that specify TID 14844. On this basis, the Commission concludes that a partial exemption from the requirements of paragraphs (f)(2)(vii), (viii), (xxvi), and (xxviii) of 10 CFR 50.34 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

Table 20.6-1 10 CFR 52.47(a)(1)(ii) TMI Action Plan Items

TMI REQUIREMENT	10 CFR 50.34(f)	FSER Section
I.A.4.2	(2)(i)	20.4
I.C.5	(3)(i)	20.4
I.C.9	(2)(ii)	20.4
I.D.1	(2)(iii)	18 (all), 20.4
I.D.2	(2)(iv)	18.7, 20.4
I.D.3	(2)(v)	20.4
I.F.1	(3)(ii)	17.1, 20.4
I.F.2	(3)(iii)	17.1, 20.4
II.B.1	(2)(vi)	20.4
II.B.2	(2)(vii)	12.3.3, 20.4
II.B.3	(2)(viii)	9.3.3, 13.3, 20.4
II.B.8	(1)(i) & (xii), (2)(ix), (3)(iv) & (v)	19.1, 19.2, 19.4, 20.4
II.D.1	(2)(x)	20.4
II.D.3	(2)(xi)	20.4
II.E.1.1	(1)(ii)	10.4.9, 19.1, 20.4
II.E.1.2	(1)(ii), (2)(xii)	20.4
II.E.3.1	(2)(xiii)	8.4, 20.4
II.E.4.1	(3)(vi)	20.4
II.E.4.2	(2)(xiv)	20.4
II.E.4.4	(2)(xv)	20.4
II.F.1	(2)(xvii)	7.5, 12.4.4, 20.4
II.F.2	(2)(xviii)	20.4
II.F.3	(2)(xix)	20.4
II.G.1	(2)(xx)	8.4, 20.4
II.J.3.1	(3)(vii)	20.4
II.K.2(16)*	(1)(iii)	20.4
II.K.3(2)	(1)(iv)	20.4
II.K.3(25)*	(1)(iii)	20.4

TMI REQUIREMENT	10 CFR 50.34(f)	FSER Section
III.A.1.2	(2)(xxv)	13.3, 18.2.3.1, 20.4
III.D.1.1	(2)(xxvi)	20.4
III.D.3.3	(2)(xxvii)	12.3.4, 20.4
III.D.3.4	(2)(xxviii)	6.4, 20.4

* Although these TMI Action Plan items did not apply to Westinghouse PWRs in NUREG-0737, they are applied to all PWR designs in 10 CFR 50.34(f)(1)(iii).

20.7 Incorporation of Operating Experience

20.7.1 Background

As part of its program to disseminate information on operational reactor experience to the nuclear industry, the NRC issues generic communications (bulletins, generic letters, and information notices) when a significant safety-related event or condition at one or more facilities is believed to potentially apply to other facilities. A bulletin or generic letter is typically issued when the NRC staff determines that licensees should be required to inform the NRC what actions have been or will be taken to address an event, condition, or circumstance that is both potentially safety-significant and generic. An information notice is typically issued when the NRC staff determines that licensees should be informed of an event, condition, or circumstance that may be both potentially safety-significant and generic, but the event, condition, or circumstance is not sufficiently significant to warrant requiring licensees to confirm in writing that actions have been or will be taken. Potential safety issues highlighted in NRC generic communications have resulted in the establishment of a USI or GSI, and have also been incorporated into formal regulatory requirements.

The Commission requested, in its SRMs dated July 31, 1989, and February 15, and March 5, 1991, that an applicant submitting plant designs for standard plant design certification provide a discussion of how operating experience has been incorporated into the design.

A review of the AP600 design for incorporation of important lessons learned from operating plant experience was accomplished by reviewing the bulletins and generic letters issued between January 1, 1980, and December 31, 1997, and determining whether Westinghouse properly incorporated in the AP600 design the staff positions in those documents that were applicable to the design. In the NRC programs that account for operating experience, the bulletins and generic letters issued to the nuclear industry convey the most safety-significant lessons distilled from numerous sources of information on operating plant malfunctions (e.g., Licensee Event Reports), issue staff positions on resolving problems in these malfunctions, and request actions to be taken by the licensees. As a contrast, information notices do not request actions on the part of the licensees. Thus, reviewing how applicable bulletins and generic letters have been incorporated into the AP600 design is a sufficient basis for reviewing the design against operating experience.

In the resolution of bulletins and generic letters for the AP600 design, the staff went outside the specific purpose of the documents to determine their resolution for the AP600 design (e.g., BL 80-02 requested BWR licensees to determine the equipment in their plants supplied by a specific contractor with quality assurance and quality control problems, and report on what assurance the licensees' have that the equipment is sound. The staff would review the QA procedures for the AP600 design to determine that contractors do not have such problems). In addition, some of the bulletins and generic letters involved issues that will be the responsibility of the COL applicant during the construction or operation of the plant. This will be identified in the resolution of these documents for the AP600 design.

20.7.2 *Application Content Review*

Westinghouse submitted an application, including the SSAR, for standard plant design certification of the AP600 design in its letter of June 26, 1992. In that document, Westinghouse states that the design engineers continually review industry experience from sources such as NRC bulletins, Licensee Event Reports, NRC requests for information, and generic letters. It further states that operating plant experience has been incorporated in the AP600 design by virtue of its participation in developing Volume III of the EPRI ALWR Utility Requirements Document and in the activities of the ALWR Utility Steering Committee. Because of the brevity and generality of these statements, the staff requested, in its letter of July 24, 1992, that Westinghouse provide details on how operational plant experiences from specific sources was incorporated in the AP600 design.

On December 15, 1992, Westinghouse submitted WCAP-13559, "Operational Assessment for AP600," dated December 1992. In that document, Westinghouse addressed the manner in which it incorporated operating plant experience into the AP600 design and stated that it reviewed the NRC bulletins, generic letters, circulars, information notices, and Office of Analysis and Evaluation of Operational Data (AEOD) reports for the time period January 1, 1980 to December 31, 1991. Westinghouse discussed the applicability of these NRC documents to the AP600 design by referring to the appropriate SSAR sections or explaining the AP600 design does not have the equipment discussed in the NRC document. The disposition of the individual documents were broken down into the following categories:

- not applicable to the AP600 design (e.g., BWR only, B&W or CE facilities only, or not applicable to commercial reactors)
- not applicable for other reason (e.g., procurement issue, administrative communication, procedural issue, maintenance or surveillance issue, plant specific or isolated event)
- applicable to AP600 design certification

Westinghouse reviewed over 500 bulletins and generic letters for applicability to the AP600 design. Approximately 300 were determined not to be applicable. Of the remaining documents, a number were associated with USIs and GSIs discussed in Sections 20.2 through 20.4 of this report. As such, these bulletins and generic letters have already been incorporated into the staff's review of AP600 design and do not have to be considered as part of the staff review for incorporation of operating experience in the AP600 design.

The staff considered the bulletins and generic letters Westinghouse concluded were applicable to the AP600 design in determining the list of such documents that the staff should review for how operating experience was incorporated in the design.

20.7.2.1 Westinghouse WCAP-13559 Report

At the time of the DSER, the Westinghouse report was approximately three years out of date and needed to be updated for bulletins and generic letters issued since December 31, 1991. Although only two bulletins and three generic letters issued in 1992, 1993, and 1994 were identified as applicable to the AP600, there were a total of 26 issued in those years and Westinghouse should address their applicability to the AP600 design. Westinghouse should revise WCAP-13559 to include the bulletins and generic letters that were issued after December 31, 1991. This inclusion of new bulletins and generic letters should continue until the draft FSER for the AP600 design is issued. This was Open Item 20.7-1.

Westinghouse submitted Revision 1 to WCAP-13559 in September 1996. This revision included a review of the bulletins and generic letters issued up through April 20, 1996. Therefore, Open Item 20.7-1 is closed.

In reviewing Westinghouse's comments on applicable generic letters in WCAP-13559, there were incorrect comments on generic letters 80-070, 81-037, and 82-030; however, as explained below, none of these generic letters are considered applicable to the AP600 design review. Nevertheless, Westinghouse should correct its comments in WCAP-13559. Also, Westinghouse should change the incorrect statements that BL-83-04 and BL-90-01 applied only to BWR licensees. This was Open Item 20.7-2.

Westinghouse corrected the above comments in Revision 1 to WCAP-13559. Therefore, Open Item 20.7-2 is closed.

For GL 80-070, "Failure of Mercury Wetted Matrix Relays Through Centrifugal Charging Pumps Following Secondary Side HELB," July 1980, Westinghouse stated that Bulletin 80-019 was for GE plants only; however, the bulletin applied to GE mercury-wetted relays in reactor protection systems and was also sent to Westinghouse plant licensees. However, this GL was concerned with specific equipment hardware and was more than ten years old, and, therefore, was not considered applicable to the AP600 design.

For GL 81-037, "ODYN Code Reanalysis Requirements," Westinghouse stated that the GL was not issued; however, the GL was issued on December 29, 1981, and pertained to ODYN Code reanalysis requirements for BWRs. Because the GL applied to BWRs, it is not applicable to the AP600 design.

For GL 82-030, "Filings Related to 10 CFR Part 50 Production and Utilization Facilities," December 1982, Westinghouse stated that the GL was not applicable to commercial nuclear power plants; however, the GL concerns filings related to 10 CFR Part 50 utilization facilities that are commercial nuclear power plants. The GL informed reactor licensees that requests for exemptions and license amendments, and responses to staff requests for information, should be filed with the Director for Nuclear Reactor Regulation. This GL would not be considered

applicable to the AP600 design because it provides information or guidance to the licensees on matters that do not involve plant operations or design.

Westinghouse subsequently submitted Revision 2 to WCAP-13559 in March 1998. This revision includes Westinghouse's evaluation of bulletins and generic letters issued through December 31, 1997.

20.7.3 Regulatory Review

The SRP (NUREG-0800) guides the NRC staff for its review of a reactor facility design. This document states requirements, acceptance criteria (some of which are predicated on operating reactor experience), and findings that the staff must make. This document was last revised in April 1982. Significant issues raised before January 1981 were incorporated into the April 1982 revision. Accordingly, the staff concludes that it is appropriate to focus its review on issues of operating experience identified by the NRC since January 1980. As stated above, Westinghouse reviewed and reported on the bulletins and generic letters issued by the NRC between January 1, 1980, and December 31, 1997, as to their applicability to the AP600 design.

As discussed in Section 20.7.1 above, the bulletins and generic letters address the issues that are of sufficient safety significance to warrant requiring licensees to inform the NRC of the actions they have taken or will take, whereas information notices do not require a response. Accordingly, the NRC staff reviewed the bulletins and generic letters issued between January 1, 1980, and December 31, 1997, applicable to the AP600 design.

Upon initial review, certain bulletins and generic letters were excluded from the review because they were not relevant to the design of the AP600 plant, or because they were associated with TMI Action Plan items, USIs or GSIs, or existing rules and regulations and, thus, were already an integral part of the staff's AP600 design review process. The resolution of the technically relevant generic issues in NUREG-0933 (i.e., TMI Action Plan items, USIs, and GSIs) for the AP600 design are addressed in Sections 20.2 through 20.4 of this report. As examples, GL 88-04, "Distribution of Gems Irritated in Research Reactors," applies only to research reactors; GL 86-14, "Operator Licensing Examinations," applies to operator licensing exam schedules which are the responsibility of the owner/operator; GL 86-10, "Implementation of Fire Protection Requirements," is associated with 10 CFR 50.48 and 10 CFR Part 50, Appendix R; GL 89-06, "Task Action Item I.D.2 — Safety Parameter Display System," is associated with a TMI Action Plan item; and GL 84-15, "Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability," is associated with a USI/GSI. There are additional generic letters that transmitted previously issued bulletins and, therefore, were considered duplicates of the bulletins and not included in this operational experience review for the AP600 design.

However, there were cases where a bulletin was issued only to plants other than Westinghouse PWRs (e.g., BWRs or B&W and CE PWRs) and were still considered applicable to the Westinghouse AP600 PWR design. One, for example is BL-80-001, "Operability of ADS Valve Pneumatic Supply," which was restricted to BWR licensees. This bulletin is considered applicable to the AP600 design because the design has an ADS similar to that for BWRs, as discussed in the resolution of Issue II.K.3(28) in Section 20.4 of this report.

The remaining 44 bulletins and 122 generic letters, listed in Tables 20.7-1 and 20.7-2, respectively, were considered sufficiently applicable to the AP600 design for the staff to conduct an evaluation of document against the design. These documents were evaluated to assure that the issues identified had, if appropriate, been incorporated into the staff's AP600 design review. In some cases, as explained in the tables, it was determined that the document was, in fact, not applicable to the AP600 design. The resolution of the issues in these bulletins and generic letters is summarized in Tables 20.7-1 and 20.7-2, respectively, of this report.

20.7.4 Conclusion

Of the more than 550 bulletins and generic letters issued by the NRC between January 1, 1980, and December 31, 1997, 166 were identified to be incorporated into the staff review of the AP600 design because the issues involved in these documents were not already required by rule, regulation, or policy statement. These are listed in Tables 20.7-1 and 20.7-2 of this report.

In the DSER the staff noted that there were questions concerning the responses by Westinghouse to several of the bulletins and generic letters listed in Tables 20.7-1 and 20.7-2, respectively. The DSER tables included the questions for the specific bulletins and generic letters that Westinghouse needed to address to close out the review by the staff. This was identified as DSER Open Item 20.7.4-1. Tables 20.7-1 and 20.7-2 of this report detail the resolution of the DSER Open Items. All DSER Open Items in tables 20.7-1 and 20.7-2 have been resolved, therefore, DSER Open Item 20.7.4-1 is closed.

On the basis of its review of the bulletins and generic letters issued between January 1, 1980, and December 31, 1997; and Westinghouse's report (WCAP-13559) on how these bulletins and generic letters apply to the AP600 design, the staff concludes that Westinghouse has adequately addressed the incorporation of operational data into the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design

Bulletin No. and Title	Staff Resolution
BL-80-01, Operability of ADS Valve Pneumatic Supply	This bulletin was issued to only BWR licensees to determine the operability of the pneumatic operator for the ADS; however, the AP600 design has an ADS similar to BWRs. In Section 1.9.3 of the SSAR, item (1)(x), Westinghouse states that the AP600 uses a safety-related automatic depressurization system that is different from that presently used on BWRs. The AP600 automatic depressurization system uses safety-related dc motor-operated valves and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. There is no non-safety-related equipment or instrumentation, including instrument air or nitrogen supply, relied on in the operation of these valves. These valves are designed and qualified to function in the conditions of an accident. They will also be the subject of pre-operational and in-service testing, and they will be included in the reliability assurance program. Therefore, Open Item 20.7-3 is closed. Therefore, this bulletin is not applicable to the AP600 design.
BL-80-02, Inadequate Quality Assurance for Nuclear Supplied Equipment	Westinghouse states that the bulletin is not applicable to the AP600 design because the bulletin was issued only for BWR plants; however, the bulletin concerned contractors (in this case a BWR supplier) having quality control/quality assurance problems and the licensee programs to identify these problems. The staff asked Westinghouse to address this bulletin. This was Open Item 20.7-4. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Open Item 20.7-4, therefore, is closed and this bulletin is resolved for the AP600 design.
BL-80-03, Loss of Charcoal from Standard Type II, 2-inch, Tray Absorber Cells	Westinghouse does not consider this bulletin applicable to the AP600 design because it involves procurement issues. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-80-04, Analysis of a Pressurized-water Reactor (PWR) Main Steamline Break with Continued Feedwater Addition	Westinghouse stated that this bulletin was addressed in Section 15.1.5 of the SSAR on steam system piping failures. Westinghouse should address this in Section 6.2 of the SSAR and consider the containment pressure and temperature response to the event. The staff considered this bulletin in its review of Section 15.1.5 of the SSAR, and Section 6.2.1.4 of the SSAR on mass and energy release analysis for a postulated pipe rupture inside containment. This was Open Item 20.7-5. This bulletin asked addressees to review their containment pressure and temperature response analysis to determine if the main steamline break accident inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources such as continuation of feedwater or condensate flow. It also asked addressees to consider the ability to detect and isolate the damaged SG from these sources. In SSAR Sections 6.2.1.4.1.3 and 6.2.1.4.3.2, Westinghouse indicates that the effects of startup feedwater flow are maximized in the main steamline break mass and energy release by assuming maximum (runout) startup feedwater flow to a fully depressurized steam generator starting from the safeguard system signal or low steam generator level reactor trip and continuing until automatically terminated. Regarding normal feedwater, Westinghouse indicated in SSAR Section 6.2.1.4.1.2, that the unisolated feedwater line volumes between the steam generator and isolation valves has been accounted for in the mass and energy release. The feedwater flow rates are based on steam and main feed system design. Feedwater is isolated on a containment pressure signal. Because normal and startup feedwater addition have been maximized and because the AP600 has means to automatically isolate feedwater flow, the staff finds that the licensee has adequately addressed the containment-related issues in Bulletin 80-04. The containment-related aspects of Bulletin 80-04 are therefore resolved. The other aspect of the feedwater addition issue addressed by this bulletin, namely the reactivity addition that would occur as a result of a main steamline break, is addressed in Section 15.2.1.5 of this report. The reactivity-related aspects of this bulletin are considered resolved based on the staff's acceptance of the analyses provided in SSAR Section 15.1.5. Therefore, Open Item 20.7-5 is closed.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-80-05, Vacuum Condition Resulting in Damage To chemical Volume Control System (CVCS) Holdup Tanks	This bulletin addressed the collapse of CVCS tanks under partial vacuum. Westinghouse stated that the bulletin is not applicable to the AP600 design because the design has no hold-up tanks in the CVCS. In the DSER, the staff requested that Westinghouse address the potential of having a partial vacuum in a tank important to safety in the AP600 design. This was designated as Open Item 20.7-6. Open Item 20.7-6 is resolved because this bulletin is addressed by the staff in Section 11.2 of this report. Therefore, this bulletin is resolved for the AP600 design.
BL-80-06, Engineered Safety Feature (ESF) Reset Controls	Westinghouse stated in WCAP-13559 that this bulletin is addressed in Sections 7.3.1.1, and 13.5, and Chapter 14 of the SSAR. This bulletin listed the following two actions that apply to the AP600 design: (1) review the I&C system schematics to verify the ESF equipment remains in its emergency mode upon reset of the ESF actuation signal and (2) verify the as-built I&C system configuration conforms with schematics. For the AP600 design, resetting the ESF signal does not reposition any ESF equipment. Verification of the as-built I&C system is the responsibility of the COL applicant during the plant pre-operational tests. This is COL Action Item 20.7-1. The last action required by the bulletin is plant-specific and does not apply to the AP600 design. This bulletin is resolved for the AP600 design.
BL-80-08, Examination of Containment Liner Penetration Welds	Westinghouse stated that the bulletin is not applicable to the AP600 design because the design has no containment liner. The staff agrees with this assessment. This bulletin does not apply to the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-80-10, Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to Environment	The staff noted in the DSER that the issues in this bulletin would be considered during staff reviews of Section 6.2 and 11.5 of the SSAR. However, the issues in this bulletin were considered during the staff review of Sections 9.2.9 and 9.3.5. In Section 9.3.5 of the SSAR, Westinghouse states that there are no permanent connections between the WRS and non-radioactive piping. However, provisions are included for temporary diversion of contaminated water from normally nonradioactive drains to the WLS. Therefore, the WRS is designed to prevent the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal. In Revision 1 of WCAP-13559, Westinghouse stated that this bulletin was not applicable to the AP600 design and was the responsibility of the COL applicant. The event at Brunswick Nuclear Facility was caused by the use of a temporary heating hose, which resulted in contamination of a nonradioactive system and an unmonitored, uncontrolled release of radioactivity to the environment. On the basis of the above, the staff does not believe that such an event is caused by poor system design but because of poor system operation and maintenance programs. Therefore, the staff agrees with Westinghouse that the COL applicant should address this event in its plant operating and maintenance procedures. Therefore, this bulletin is resolved for the AP600 design.
BL-80-11, Masonry Wall Design	As stated in Revision 12 of SSAR Section 3.8.4.6.1.4, there are no safety-related masonry walls used in the nuclear island. Also, in WCAP-13559, Westinghouse stated that this bulletin is not applicable to the AP600 design because the design has no safety-related masonry walls. The staff agrees that this bulletin is not applicable to the AP600.
BL-80-12, Decay Heat Removal Operability	This bulletin dealt with reducing the likelihood of losing the decay heat removal capability. Westinghouse stated that this bulletin is addressed in Section 7.4.1 of the SSAR. The staff evaluated this issue in Section 6.3 of this report. This bulletin is resolved.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-80-15, Possible Loss of Emergency Notification System with Loss of Offsite Power	Westinghouse stated that this bulletin was not applicable to the AP600 design and was the responsibility of the COL applicant. Westinghouse indicates that the issues of this bulletin are discussed in Section 9.5.2.2.5 of the SSAR and refers to the resolution of Issue II.A.3.3 in Section 20.4 of the SSAR. The staff agrees that this bulletin is not applicable to the AP600 design. The reminder to the COL applicant to review this bulletin for recommendations related to loss of offsite power and a consequential loss of the emergency notification system is COL Action Item 20.7-2.
BL-80-16, Potential Misapplication of Rosemount Inc. Models 1151 and 1152 Pressure Transmitters	Westinghouse stated that this bulletin was not applicable to the AP600 design because it involved procurement and maintenance issues. The staff agrees with this conclusion. This was identified in the DSER as part of COL Action Item 20.7-2. It is redesignated as part of COL Action Item 20.7-3. Therefore, this bulletin is resolved for the AP600 design.
BL-80-18, Maintenance of Adequate Minimum Flow Through Centrifugal Charging Pumps Following Secondary-Side, High-Energy-Line Rupture	Westinghouse stated that this bulletin was not applicable to the AP600 design because the design has no charging pumps as part of safety injection. Westinghouse, however, should address the design of mini-flow lines for safety-grade pumps. This was Open Item 20.7-7. The staff agrees that this bulletin is not applicable to the AP600 design. Therefore, Open Item 20.7-7 is closed and this bulletin is resolved.
BL-80-20, Failures of Westinghouse Type W-2 Spring Return to Neutral Control Switches	Westinghouse stated that this bulletin was not applicable to the AP600 design because it involved a procurement issue. The staff agrees that the issues in this bulletin involve procurement and are the responsibility of the COL applicant. This was identified in the DSER as part of COL Action Item 20.7-2. It is redesignated as part of COL Action Item 20.7-3. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-80-24, Prevention of Damage Due to Water Leakage Inside Containment	Westinghouse stated that, as defined in Bulletin 80-24, there are no open systems in the AP600 containment. Bulletin 80-24 defines an open system as one that utilizes an indefinite volume, such as a river, so that leakage from the system could not be detected by inventory decrease. Cooling water for the AP600 design is supplied by closed systems, including the component cooling water system (SSAR Section 9.2.2) and the chilled water system (SSAR Section 9.2.7). Fire protection water used inside containment is stored in the passive containment cooling water storage tank (PCCWST), and isolated by containment isolation valves during operation. Water level in the PCCWST is alarmed in the MCR and excessive flow from the tank can be terminated. Monitoring containment sump level is a key part of AP600 leakage detection, which provides assurance that an increasing sump water level will be detected. SSAR Section 5.2.5 provides a description of leakage detection and SSAR Section 3.4.1.2.2.1 addresses containment flooding events. Therefore, this bulletin is resolved for the AP600 design.
BL-81-01, Surveillance of Mechanical Snubbers	This bulletin requested prompt examinations and testing of all mechanical snubbers installed in both operating plants and plants under construction. In Revision 0 of WCAP-13559, Westinghouse stated that this bulletin was not applicable to the AP600 design because the issues involved procurement. The DSER stated that Westinghouse should address the surveillance aspects of this bulletin. This was DSER Open Item 20.7-8. However, subsequent to issuance of the DSER, the staff has concluded that it agrees that this bulletin is not applicable to the AP600 because, as stated above, it only involved surveillance and testing of installed snubbers. Therefore, DSER Open Item 20.7-8 is closed. The staff concluded that the surveillance aspects of this bulletin are resolved for the AP600 design by (1) commitments to design criteria, production and operability tests for snubbers in SSAR Section 3.9.3.4.3, (2) the commitments in SSAR Sections 3.9.6, 5.2.4, and 6.6 to inservice testing and inspections in accordance with ASME Section XI, and (3) the commitment in SSAR Section 1.9.4.2.3 to perform inservice testing of snubbers in accordance with ANSI/ASME OM Code-1990. Article IWF 5300 of Section XI references ASME/ANSI OM-Part 4. As discussed in Section 3.9.3.3 of this report, the commitment in (1) above is acceptable. The commitments in (2) and (3) above are in accordance with 10 CFR 50.55a(b)(2)(viii) and 50.55a(f)(4), and are also acceptable. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-81-02, Failure of Gate-Type Valves to Close Against Differential Pressure	In WCAP-13559, Revision 3, Westinghouse references SSAR Sections 3.9.6.2, 5.4.8.1.2, and 5.4.8.2 as the basis for the resolution of this bulletin. The staff agrees with this basis. Since the subject of this bulletin led, in part, to the issuance of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," the staff's position is that the basis for disposition of BL-81-02 should be the information in the SSAR relative to GL-89-10. As discussed in Section 3.9.6.2 in Chapter 3 of this report, the staff has concluded that the commitments in SSAR Section 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis to resolve this issue.
BL-81-03, Flow Blockage of Cooling Water to Safety System Components by Corbicula SP (Asiatic Clams) and Mytilus SP (Mussels)	Westinghouse stated that this bulletin was not applicable to the AP600 design because the AP600 does not depend on a site water intake structure for safety-related heat removal. The staff requested in the DSER that Westinghouse address how the AP600 intake structure prevents flow blockage from potential sources. This was Open Item 20.7-9. The staff agrees with Westinghouse that this bulletin is not applicable to the AP600 because the component coolant system and service water system are not used for cooling safety-related components and therefore, Open Item 20.7-9 is closed. In addition, service water strainers and service water chemical injection are addressed in SSAR Section 9.2.1.2.2. Therefore, this bulletin is not applicable to AP600.
BL-82-02, Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	Westinghouse stated that this bulletin was not applicable to the AP600 design because the issues involved maintenance. The staff, however, requests that Westinghouse address the use of molybdenum disulfide lubricant within the reactor coolant pressure boundary, reactor coolant pump internals, reactor vessel closure studs and in any other service for the AP600 design. This was Open Item 20.7-10. The use of molybdenum disulfide lubricant is addressed in Section 5.2.3 of this report. Lubricants containing molybdenum disulfide are specifically prohibited for use in the AP600 design. Open Item 20.7-10 is closed. Therefore, this bulletin is resolved for the AP600 design.
BL-82-04, Deficiencies in Primary Containment Electrical Penetration Assemblies	This bulletin discusses the potential generic safety implications concerning electrical penetration assemblies supplied by the Bunker Remo company. Westinghouse stated that this bulletin was not applicable to the AP600 design because the issue involved procurement. The staff agrees that BL-82-04 will be resolved by the COL applicant.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-83-03, Check Valve Failures in Raw Water Cooling Systems of Diesel Generators	In Revision 0 to WCAP-13559, Westinghouse stated that this bulletin was not applicable to the AP600 design because the issues involved procurement. However, because this bulletin involved the inservice surveillance and testing of check valves and the location of check valves in diesel generators, the DSER requested Westinghouse to address these aspects of the bulletin. This was DSER Open Item 20.7-11. In Revision 1 to WCAP-13559, Westinghouse stated that this issue is not applicable because the AP600 diesel generators have not safety-related functions. The staff agrees with the Westinghouse position. BL-83-03 does not apply to the AP600 design. Therefore, Open Item 20.7-11 is closed.
BL-84-03, Refueling Cavity Water Seal	In Revision 1 of WCAP-13559, Westinghouse stated that this bulletin was not applicable to the AP600 design because the design does not use this type of seal; however, the bulletin requested licensees to address the probability and consequences of failure of these seals. In the DSER, the staff requested that Westinghouse address these aspects of the bulletin. This was designated as DSER Open Item 20.7-12. Westinghouse responded to this issue in the SSAR, the staff reviewed the response and found it acceptable. Therefore, Open Item 20.7-12 is closed and this bulletin is resolved for the AP600 design.
BL-85-02, Undervoltage Trip Attachments of Westinghouse DB-50 Type Reactor Trip Breakers (RTB)	Westinghouse stated in WCAP-13559 that this bulletin is addressed in Sections 7.1.2.2.4 and Chapter 16 of the SSAR. This bulletin (1) assured proper reactor trip breaker (RTB) testing in plants that had not yet installed the automatic shunt trip modification and (2) provided information about RTB reliability and technical specification (TS) operability. The AP600 design addresses this first part by providing automatic diverse trip actuation via the shunt trip attachment. Testing of the interface allows trip actuation of the breakers by either the undervoltage trip attachment or the shunt trip attachment. Westinghouse also provided sufficient information on RTB reliability and TS operability to adequately address the second part of the bulletin. DSER Open Item 20.7-13 incorrectly states that the referenced SSAR Chapter 16 and Surveillance Requirement (SR) 3.3.1.6 applies to RCPs. Table 3.3.1-1 of the AP600 TS indicates that SR 3.3.1.6 is the correct SR for the RTB undervoltage and shunt trip mechanisms, thus resolving DSER Open Item 20.7-13. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-86-01, Minimum Flow Logic Problems That Could Disable Residual Heat Removal (RHR) Pumps	Westinghouse stated that this bulletin was not applicable to the AP600 design because it applied only to BWR plants; however, this bulletin addressed the loss of RHR pumps caused by a single failure of the isolation valve in the mini-flow lines for the pumps. Westinghouse should address this aspect of this bulletin. This was Open Item 20.7-14. Westinghouse states, in its response for BL 86-03 below, that the AP600 does not have valves in minimum-flow (mini-flow) lines and, therefore, the issues of this bulletin should be resolved for the AP600 design. The staff agrees with Westinghouse. Therefore, Open Item 20.7-14 is closed and this bulletin is resolved for the AP600 design.
BL-86-02, Static "O" Ring Differential Pressure Switches	Westinghouse stated that this bulletin was not applicable to the AP600 design because it involved procurement issues. The staff agrees with this conclusion and concludes that these issues are the responsibility of the COL applicant. This was identified in the DSER as part of COL Action Item 20.7-2. It is redesignated as part of COL Action Item 20.7-3. Therefore, this bulletin is resolved for the AP600 design.
BL-86-03, Potential Failure of Multiple Emergency Core Cooling System (ECCS) Pumps Due to Single Failure of Air-operated Valve in Minimum-flow Recirculation Line	Westinghouse stated that this bulletin is not applicable to the AP600 design because the design does not have valves in mini-flow lines. The staff has reviewed this issue and agrees with Westinghouse that this bulletin is not applicable to the AP600 design. Therefore, this bulletin is resolved for the AP600 design.
BL-87-01, Thinning of Pipe Walls in Nuclear Power Plants	Westinghouse stated that the bulletin is a surveillance issue which is discussed in Sections 5.4.3.4 and 10.3.6 of the SSAR, and is partly the responsibility of the COL applicant. The issues concerning pipe wall thinning and erosion/corrosion of pipes are addressed in SSAR Sections 5.4.3.4 and 10.3.6. The staff completed its review of SSAR Sections 5.4.3.4 and 10.3.6 and all issues associated with pipe thinning and erosion/corrosion have been addressed and resolved. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-87-02, Fastener Testing to Determine Conformance with Applicable Material Specifications	In WCAP-13559, Revision 1, Westinghouse stated that this bulletin is not applicable to the AP600 design because it involves a procurement issue. The staff agrees that this GL is not applicable to the AP600 design certification phase. Westinghouse revised WCAP-13559 to identify this issue as a COL action. This is part of COL Action Item 20.7-3.
BL-88-01, Defects in Westinghouse Circuit Breakers	This bulletin provides information on the Westinghouse Series DS circuit breakers and safety concerns associated with their use. Westinghouse stated that this bulletin was not applicable to the AP600 design because the issue involved procurement. The staff agrees that BL-88-01 will be resolved by the COL applicant.
BL-88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes	Westinghouse stated that this bulletin was not applicable to the AP600 design because there are stainless steel support plates in the AP600 steam generators. The staff agrees with this assessment. See also Section 5.4.2.1 of this report for additional information concerning AP600 steam generators. Therefore, this bulletin is resolved for the AP600 design.
BL-88-03, Inadequate Latch Engagement in HFA Type Latching Relays Manufactured by General Electric Company	Westinghouse stated that this bulletin was not applicable to the AP600 design because it involved procurement issues. The staff agrees with this conclusion and concludes that these issues are the responsibility of the COL applicant. This was identified in the DSER as part of COL Action Item 20.7-2. It is redesignated as part of COL Action Item 20.7-3. Therefore, this bulletin is resolved for the AP600 design.
BL-88-04, Potential Safety-Related Pump Loss	Westinghouse stated that the bulletin was not applicable to the AP600 design because the design has no safety-related pumps. The safety-related cooling systems are passive systems. The staff reviewed this issue and agrees with this conclusion. Therefore, this bulletin is not applicable to the AP600 design.
BL-88-08, Thermal Stresses in Piping Connected to Reactor Cooling Systems	WCAP-13559 stated that this bulletin is addressed in Sections 3.9.3.1.2 Revision 2 of the SSAR. The staff concluded that the information in SSAR Section 3.9.3.1.2 provides an acceptable basis for resolving BL-88-08 for the AP600. The staff's evaluation of this issue is in Section 3.12.5.9 of this report.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-88-09, Thimble Tube Thinning in Westinghouse Reactors	Westinghouse stated that this bulletin does not apply to the AP600 design because the design does not have the thimble tube design discussed in the bulletin. The staff requested that Westinghouse address why the thimble tube design of the AP600 is acceptable in the resolution of this bulletin. This was Open Item 20.7-15 and 3.9.5-5. DSER Open Item 20.7-15 was subsumed by DSER Open Item 3.9.5-5. The staff reviewed the Westinghouse response to Open Item 3.9.5-5 in Revision 10 to SSAR Section 3.9.7.2, and found the description of the improved design of the AP600 thimble tubes to be acceptable. The staff's evaluation of this issue is in Section 3.9.5 of this report. Therefore, Open Item 20.7-15 is closed and this bulletin is resolved.
BL-88-11, Pressurizer Surge Line Thermal Stratification	WCAP-13559 stated that this bulletin is addressed in Section 3.9.3.1.2 Revision 2 of the SSAR. The staff has concluded that the information in SSAR Section 3.9.3.1.2 provides an acceptable basis for resolving BL-88-011 for the AP600. The staff's evaluation of this issue is in Section 3.12.5.10 of this report.
BL-89-01, Failure of Westinghouse Steam Generator Tube Mechanical Plugs	Westinghouse stated that this bulletin is not applicable to the AP600 design because the issues involve procurement; however, this bulletin involves certain steam generator mechanical plugs supplied by Westinghouse. Westinghouse should address why the AP600 design should not have the problems discussed in the bulletin. This was Open Item 20.7-16. Upon reconsideration, the staff concluded that the use of steam generator mechanical plugs is a procurement issue and, thus, the staff agrees with the Westinghouse assessment. Therefore, Open Item 20.7-16 is closed and this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-89-03, Potential Loss of Required Shutdown Margin During Refueling Operations	Westinghouse stated that this bulletin is not applicable to the AP600 design because the issues involved procedures. The staff agrees that this bulletin involved procedures; however, these procedures would involve movement and placement of highly reactive fuel during refueling within the core designed by Westinghouse. The staff requested that Westinghouse discuss what responsibility it has because it is designing the AP600, including the core and control systems, and what is the responsibility of the COL applicant. This was DSER Open Item 20.7-17 and DSER COL Action Item 20.7-3. SSAR Section 9.1 discusses fuel storage and handling, including the refueling equipment used to safely move and store fuels. Additionally, the IRWST provides large quantities of borated water that maintains the required shutdown margin. Therefore, Open Item 20.7-17 is closed and this bulletin is resolved. DSER COL Action Item 20.7-3 is redesignated as COL Action Item 20.7-4.
BL-90-01, Loss of Fill-oil in Transmitters Manufactured by Rosemount	Westinghouse states in WCAP-13559 that this bulletin is not applicable to the AP600 design because it involves a procurement issue. This resolves DSER Open Item 20.7-2 because it corrects the earlier statement that this issue only concerned BWR plants. Supplement 1 to this bulletin states that transmitters manufactured after July 11, 1989, are not subject to the fill-oil leakage problems identified in the bulletin. In DSER Open Item 20.7-18 the staff requested that Westinghouse address the on-line monitoring capability of the AP600 design because this is an effective method to address the loss of fill-oil in the Rosemount transmitter issue. However, this is no longer of concern because the staff believes that Rosemount transmitters manufactured before July 11, 1989 will not be used. In addition, the staff agrees that the issues in this bulletin involve procurement and considers DSER Open Item 20.7-18 resolved. Therefore, this bulletin is resolved for the AP600 design.
BL-92-01, Failure of Thermo-lag 330 Fire-barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage	In the DSER, the staff stated that Westinghouse did not address this bulletin in WCAP-13559. This was identified as Open Item 20.7-1 in the DSER. Open Item 20.7-1 is closed because Thermo-Lag is not used in the AP600 design. Therefore, this bulletin is not applicable to the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-93-02, Debris Plugging of Emergency Core Cooling Suction Strainers	This bulletin deals with the installation or storage of fibrous air filters or other temporary sources of fibrous material in containment, not designed to withstand a LOCA. Westinghouse states that the AP600 has no air filters inside containment with the exception of portable filters which should not be stored inside containment. This satisfies the intent of the bulletin. Therefore, BL-93-02 is resolved for the AP600 design.
BL-95-02, Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode	This bulletin deals with the need for BWR licensees to ensure that the suppression pools are relatively free from debris which could interfere with the operation of the safety-related pumps which take suction from the suppression pool by clogging the suction strainers for these pumps. In addition, the bulletin requests BWR licensees to determine whether there are adequate controls to ensure that foreign material exclusion (FME) procedures are effective. The IRWST in the AP600 provides a function similar to that of a BWR suppression pool. It provides a source of cooling water along with the core makeup tanks and the accumulators. In addition, the ADS discharges to the IRWST. The IRWST is made of stainless steel and thus will not be a source of corrosion products. All the lines leading to the IRWST are also of stainless steel or stainless steel clad. SSAR Section 6.3.2.2.7.3 states that a COL cleanliness program controls foreign debris introduced into the containment during maintenance and inspection operation. This satisfies the FME aspects of this bulletin and, therefore, resolves it for the AP600 design.
BL-96-01, Control Rod Insertion Problems	This bulletin was issued because of incomplete control rod insertion (IRI) evaluation at the South Texas and Wolf Creek plants. It has been determined that the IRI was caused by thimble tube distortion resulting from excessive load. Because this is a fuel design problem, and Westinghouse has not committed to any fuel manufacturers, the staff concluded that Westinghouse does not have to address this issue, unless it has committed to certain fuel designs discussed in the bulletin. This issue should be appropriately addressed by the COL applicant. This is part of COL Action Item 20.7-3, therefore, this bulletin is resolved for the AP600 design.

Table 20.7-1 Resolution of Applicable Bulletins Issued Between January 1,1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Bulletin No. and Title	Staff Resolution
BL-96-02, Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety-Related Equipment	This bulletin reminded licensees of their responsibilities for ensuring that activities involving the movement of heavy loads are performed safely. It also requested that licensees review their plans and capabilities for handling heavy loads and assure that their load handling operations are in accordance with existing regulatory guidelines and the licensing basis. In Revision 1 of WCAP-13559, Westinghouse stated that this issue is addressed by the resolution of USI A36 in Section 1.9.4.2.2 of the SSAR. In Section 1.9.4 of the SSAR, Westinghouse states that the AP600 design conforms to NUREG-0612 and Section 9.1.5 of the SRP. The staff determined that ensuring the safe movement of heavy loads is the responsibility of the COL applicant. Westinghouse revised WCAP-13559 to specify that this is a procedural issue and, therefore, is the responsibility of the COL applicant. Therefore, this bulletin is resolved for the AP600 design.
BL-96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	This bulletin provides the final resolution of the issue of blockage of ECCS pump suction strainer blockage for operating BWRs. The resolution includes the option of installing large passive suction strainers. As discussed in Section 6.2.1.8 of this report, the staff is satisfied that the design of the AP600 will prevent the clogging of the IRWST and the recirculation screens to the extent that the safety function of providing abundant water to the reactor core would be impaired. Therefore, this bulletin is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design

Generic Letter No. and Title	Staff Resolution
GL-80-001, Report on ECCS Cladding Models	Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication from NRC to the nuclear industry. The staff requested that Westinghouse address the clad swelling models (as described in NUREG-0630) that have been incorporated into evaluation models used for the AP600 design. This was Open Item 20.7-19. Westinghouse addressed this issue in Chapter 15 of the SSAR. Open Item 20.7-19 is closed and GL-80-001 is resolved for the AP600 design.
GL-80-002, Quality Assurance Requirements Regarding Diesel Generator Fuel Oil	This generic letter was concerned with requirements on diesel generator fuel oil in the quality assurance program. The staff requested in the DSER that Westinghouse be more specific and explain where in the SSAR that this generic letter is addressed. This was designated as Open Item 20.7-20. In Revision 1 of WCAP-13559, Westinghouse stated that this generic letter is not applicable to the AP600 design because the AP600 does not have safety-related diesel generators, as discussed in Section 8.3.1 of the SSAR. This resolves Open Item 20.7-20. Therefore, this generic letter is not applicable to the AP600 design.
GL-80-009, Low-Level Radioactive Waste Disposal	This generic letter concerned the requirements for solid waste shipments from a plant. Westinghouse stated in a response to RAI Q650.25F that to the extent that GL-80-009 applies to the design of AP600, it is addressed in SSAR Section 11.4.2. In addition, to ensure the COL applicant conforms GL-80-009, SSAR Section 11.4.6," Combined License Information for Solid Waste Management System Process Control Program," will be modified to explicitly identify the GL as a part of COL Action Item 11.4-1. The staff finds this response acceptable. Therefore, this generic letter is resolved for the AP600 design.
GL-80-013, Qualification of Safety-Related Equipment	This generic letter concerned the adequacy of the electrical equipment environmental qualification program. Westinghouse stated that this generic letter involved backfit environmental qualification issues and is discussed in Sections 3.11 and 1.9 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-80-014, LWR Primary Coolant System Pressure Isolation Valves	Westinghouse stated that this generic letter is addressed in Section 1.9 of the SSAR. The staff requested that Westinghouse be more specific and explain where in Section 1.9 of the SSAR that this generic letter is addressed. This was Open Item 20.7-21. In WCAP-13559, Revision 1, Westinghouse stated that Section 1.9.4.1.2, Issue B-63, discusses this issue. The staff evaluated this issue in the discussion of Issue 105, ISLOCA. Therefore, Open Item 20.7-21 is closed and this issue is resolved for the AP600 design.
GL-80-016, IEB 79-01b Environmental Qualification of Class 1E Equipment	This generic letter concerned meetings held on Bulletin 79-01 and questions about environmental qualification. Westinghouse stated that this generic letter involved backfit environmental qualification issues and is discussed in Sections 3.11 and 1.9 of the SSAR. The staff requested that Westinghouse be more specific and explain where in Section 1.9 of the SSAR that this generic letter is addressed. This was Open Item 20.7-22. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-22 is closed and this generic letter is resolved for the AP600 design.
GL-80-019, Resolution of Enhanced Fission Gas Release Concern	Westinghouse stated that no action was required by this generic letter; however, the fission gas release models for the AP600 design are accounted for in the fuel performance code discussed in WCAP-10851-P-A and WCAP-11873-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod design and Safety Evaluations." This generic letter is resolved for the AP600 design.
GL-80-026, Qualification of Reactor Operators	This generic letter set forth revised criteria to be used by the staff in evaluating reactor operator training. Westinghouse stated that this generic letter is the responsibility of the COL applicant. This generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-80-030, Clarification of the Term "Operable" as it Applies to Single-failure Criterion for Safety Systems Required by TS	Westinghouse stated that the definition of operable is addressed in Section 16.1 of the SSAR on plant TSs. By adoption of the improved Westinghouse Standard TS (STS), Westinghouse should adequately address the TS issues in this GL for the AP600 design. The TSs for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-80-035, Effect of a dc Power Supply Failure on ECCS Performances	Westinghouse stated that this generic letter is not applicable to the AP600 design because it concerned only BWR plants; however, the effect of dc power supply failure on ECCS performance could apply to PWRs. Westinghouse should address this effect for the AP600 design. This was Open Item 20.7-23. This GL addresses the concerns that the loss of a dc power supply could disable several emergency core cooling system components and, thereby, could result in a limiting single failure conditions for some breaks. In WCAP-13559, Revision 1, Westinghouse stated that this GL is addressed in SSAR Section 8.3.2, Table 8.3.2.7, "Failure Modes and Effects Analysis." The staff evaluated Section 8.3.2 and concluded that the effect of a dc power supply on ECCS is accordingly addressed in Table 8.3.2.7. Therefore, Open Item 20.7-23 is closed and GL 80-35 is resolved for the AP600 design.
GL-80-045, Fire Protection Rule	This generic letter requested comments on a proposed rule adding a new Section 50.48 and Appendix R, which was attached to the letter. Westinghouse stated that this generic letter is addressed in Section 9.5.1 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-80-048, Revision to May 19, 1980 Letter on Fire Protection (GL-80-045)	See the resolution of GL-80-045 above.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-80-053, Decay Heat Removal Capability	This generic letter requested TS amendments concerning decay heat removal capability. Westinghouse stated that this generic letter is addressed in Section 16.3.5 of the SSAR, on plant TSs. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-80-056, Commission Memorandum and Order on Equipment Qualification	This generic letter provided Commission Memorandum and Order dated May 27, 1980, on the Union of Concerned Scientists petition of May 27, 1978, on fire protection for electrical cables and environmental qualification of electrical components. Westinghouse stated that this generic letter is addressed in Section 3.11 and Appendix 3D of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-80-077, Refueling Water Level	This generic letter requested TS and procedures to assure that exposure of fuel assemblies and control rods can not occur during transfer while refueling. DSER Open Item 20.7-24 requested Westinghouse to address procedures concerning exposure of fuel assemblies and control rods during refueling. In Revision 1 of WCAP-13559, Westinghouse stated that this generic letter is not applicable to the AP600 design and was the responsibility of the COL applicant. See Sections 13.5.1 and 13.5-2 of the SSAR. The staff agrees with Westinghouse that this issue is the responsibility of the COL applicant. Therefore, Open Item 20.7-24 is closed and this generic letter is resolved for the AP600 design.
GL-80-082, IEB 79-01b, Supp. 2, Environmental Qualification of Class 1E Equipment	See the resolution of GL-80-016 above.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-80-088, Seismic Qualification of Auxiliary Feedwater System	WCAP-13559, Revision 2 states that this issue is not applicable to the non-seismic portion of the AP600 startup feedwater system (inside the turbine building), and that the safety-related portion of this system in the containment and auxiliary building is seismically qualified (Ref. SSAR Section 10.4.9). SSAR Section 10.4.9.1.1 and Figure 10.3.2-1 identify the startup feedwater line from the turbine building to the steam generator inlets as AP600 Class B and C (ASME Class 2 and 3). SSAR Table 3.2-1 identifies Class B and C components as Seismic Category I. The staff agrees with these safety classifications, and concludes that since the safety-related portion of the startup feedwater system is Seismic Category I, GL-80-088 is resolved for the AP600 design.
GL-80-098, IEB 80-24, Prevention of Damage Due to Water Leakage Inside Containment	See the resolution of BL-80-24 in Table 20.7-1 of this report.
GL-80-099, TS Revisions for Snubber Surveillance	This generic letter released Revision 1 to the Inservice Surveillance Requirements for snubbers in the Standard Technical Specifications (STS) in 1980. The Westinghouse Improved STS eliminated this requirement. It was replaced by ANSI/ASME OM Part 4, which is now referenced in ASME Section XI for inservice testing and inspection of snubbers. As discussed under Bulletin 81-001 in this table and in Section 3.9.3.3 of this report, Westinghouse committed to ASME Section XI and ANSI/ASME OM Part 4. Therefore, GL-80-099 is not applicable to the AP600.
GL-80-100, Appendix R to 10 CFR Part 50 Regarding Fire Protection	This generic letter stated that the Commission published a new Appendix R, fire protection, to 10 CFR Part 50. Westinghouse stated that this generic letter was discussed in Section 9.5.1 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-80-102, Commission Memorandum and Order of May 23, 1980, (Regarding IEB 79-01b Supp. 2)	See the resolutions of GL-80-082 and GL-80-016 above.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-80-106, Report on ECCS Cladding Models, NUREG-0630	See the resolution for GL-80-001 above.
GL-80-109, Guidelines for SEP Soil-structure Interaction Reviews	This generic letter provided guidelines for performing soil-structure interaction reviews for Systematic Evaluation Plants (SEP). Westinghouse stated that this generic letter was not applicable to the AP600 design because it is the responsibility of the COL applicant. Although the COL applicant is responsible for the site-specific data, the buildings of the AP600 design were designed to some range of soil-structure criteria for potential sites. Westinghouse should address this in its response to this generic letter. This was DSER Open Item 20.7-25. This generic letter provides guidelines for old operating plants such as SEP plants to perform a simplified soil-structure interaction evaluation, because detailed guidelines for the SSI analysis were not available during the design of these plants. As discussed in Chapter 3.7 of this report, the analysis method and criteria used by Westinghouse for the SSI analysis, as documented in SSAR Section 3.7.2, meet the SRP guidelines and are acceptable. The staff conclusion for the SSAR Section 3.7.2 review was also confirmed by the staff's confirmatory analysis. On this basis, there is no need for Westinghouse to respond to this generic letter and, therefore, the concern of GL-80-109 is resolved, and DSER Open Item 20.7-25 is closed.
GL-81-14, Seismic Qualification of Auxiliary Feedwater Systems	See the resolution of GL-80-088 above.
GL-81-019, Thermal Shock to Reactor Pressure Vessels	Westinghouse stated that this generic letter is addressed by addressing Generic Issue A-49 in SSAR Section 1.9.4.2. The staff's review of Issue A-49 is documented above. Therefore, GL-81-019 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-81-021, Natural Circulation Cooldown	This generic letter addressed procedures and training to prevent, recognize, and react to reactor vessel voiding during natural circulation cooldown. Westinghouse stated that this generic letter is not applicable to the AP600 design because it involved procedure and training issues that are the responsibility of the COL applicant. Westinghouse should address the emergency procedure guidelines (ERGs) for this event. See the resolution of GL-83-22. This was Open Item 20.7-26. Westinghouse submitted AP600 ERG-GW-GJR-100, Rev 3 dated 5/97 for staff review. The staff reviewed this submittal and its related natural circulation ERG and determined that guidelines are sufficiently given to the operator to cool down the plant using natural circulation means. Therefore, Open Item 20.7-26 is closed and this generic letter is resolved for the AP600 design.
GL-81-38, Storage of Low-Level Radioactive Wastes at Power Reactor Sites	DSER Open Item 20.7-27 requested that Westinghouse address onsite storage space as part of its description of the radwaste system. This generic letter provided guidelines for the storage of low-level radioactive wastes at plant sites. Westinghouse stated that this generic letter was not applicable to the AP600 because it is the responsibility of the COL applicant. This is a site-specific issue because it will depend upon the available offsite storage space for low-level radioactive waste from the plant. This will be identified by the COL applicant if it proposes an onsite low-level radioactive waste storage facility to the NRC. The NRC would then evaluate the proposed facility against the criteria in GL-81-38. In a response to Q650.33F (Revision 2), Westinghouse indicated that the AP600 radwaste building has provisions for temporary storage of low-level radioactive wastes. The COL applicant using this storage will have to comply with GL-81-38 in handling of low-level radioactive wastes. This is identified in a revised WCAP-13559 regarding GL-81-38 as a COL action. The staff finds the above approach acceptable and Open Item 20.7-27 is closed. This was identified in the DSER as COL Action Item 20.7-4. It is redesignated as COL Action Item 20.7-5. Therefore, this generic letter is resolved for AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-81-39, NRC Volume Reduction Policy	This generic letter provided the Commission policy statement on reduction of low-level radioactive wastes at plant sites. Westinghouse stated in a response to RAI Q650.26F that, to the extent that GL-81-39 applies to the design certification of AP600, it is addressed in SSAR Section 11.4.2.1. To ensure that the COL applicant will conform with GL-81-39, SSAR Section 11.4.6 was modified to identify this Generic Letter as a COL action. This is COL Action Item 20.7-6. The staff finds this response acceptable. Therefore, this generic letter is resolved for the AP600 design.
GL-82-04, Use of INPO [Institute of Nuclear Power Operations] SEE-IN Program	This generic letter recommended the INPO Significant Event Evaluation and Information Network (SEE-IN) program to screen the large volume of raw data pertaining to operational experience throughout the industry. Westinghouse stated that this generic letter is not applicable to the AP600 design because it involved administrative communication, and not plant design. Westinghouse should address the use of the SEE-IN program in including operational experience in the AP600 design. This was Open Item 20.7-28. Westinghouse included a discussion of the review of operating experience in the discussion of the resolution of TMI Action Plan Item I.C.5 in SSAR Chapters 1 and 18. The staff found this discussion acceptable. Therefore, Open Item 20.7-28 is closed and this generic letter is resolved for the AP600 design.
GL-82-08, Transmittal of NUREG-0909 Relative to Ginna Tube Rupture	Westinghouse stated that the generic letter is not applicable to the AP600 design because it involved administrative communication, and not plant design. The GL did transmit NUREG-0909 to licensees; however, Westinghouse should address how the AP600 design conforms to the findings in the NUREG. This was Open Item 20.7-29. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-29 is closed and this generic letter is resolved for the AP600 design.
GL-82-09, Environmental Qualification of Safety-Related Electrical Equipment	The generic letter requested comments on the then new rule on environmental qualification and the proposed RG 1.89 to implement the new rule. Westinghouse stated that the generic letter is addressed in Sections 1.9 and 3.11 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-82-17, Inconsistency Between Requirements of 10 CFR 50.54(t) and Standard TS for Performing Audits of Emergency Preparedness Plans	Westinghouse stated that this generic letter is not applicable to the AP600 design and is not a TS requirement. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TS for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-82-23, Inconsistency Between Requirements of 10 CFR 73.40(d) and Standard Technical Specifications (STS) for Performing Audits of Safeguards Contingency Plans	Westinghouse stated that this generic letter is not applicable and is not a TS requirement. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TS for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-82-39, Problems with Submittals of 10 CFR 73.21 Safeguards Information for Licensing Reviews	Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to the licensees. This generic letter is not a design issue because site security is within the scope of the COL applicant. This includes the reporting of safeguards information for licensing reviews. Therefore, this generic letter is resolved for the AP600 design.
GL-83-07, The Nuclear Waste Policy Act of 1982	This generic letter provided a copy of the Nuclear Waste Policy Act of 1982 and explained the requirements. Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to licensees. The Nuclear Waste Policy Act of 1982 requires licensees to have a contract with the Department of Energy (DOE) before receiving a license and is within the scope of the COL applicant. This was designated in the DSER COL as Action Item 20.7-5. It is redesignated as COL Action Item 20.7-7. Because GL-83-07 only notifies licensees of a regulation, COL Action Item 20.7-7 is not necessary and is hereby dropped. Also this issue is not a design issue and, therefore, GL-83-087 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-83-11, Licensee Qualifications for Performing Safety Analyses in Support of Licensing Actions	This generic letter stated the then-current staff practices regarding licensee qualifications for performing safety analyses using computer codes to support licensing actions. Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to licensees. The staff stated that Westinghouse should address the qualifications for performing safety analyses for the AP600 design. This was Open Item 20.7-30. In a revision to WCAP-13559 Westinghouse changed their response to indicate that the AP600 design is performed under a QA program which is reviewed by the NRC. Chapter 21 of this report presents the staff's evaluation of Westinghouse's testing program and computer code verification. Therefore, Open Item 20.7-30 is closed and this generic letter is resolved for the AP600 design.
GL-83-13, Clarification of Surveillance Requirements for Charcoal Adsorber Units in STS on ESF Cleanup Systems	Westinghouse stated that this generic letter is not applicable to the AP600 design because the design has no safety-related ventilation systems. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TS for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-83-14, Definition of Key Maintenance Personnel	This GL has been satisfactorily addressed by Westinghouse in their submittal of WCAP-13559 (Revision 1) as not applicable and administrative procedure. Therefore, this generic letter is resolved for the AP600 design.
GL-83-015, Implementation of RG 1.150, Revision 1 During Preservice and Inservice Examinations	Westinghouse has indicated in Appendix A of the AP600 SSAR that the AP600 design conforms to the recommendations of RG 1.150. Thus this generic letter is resolved for the AP600 design.
GL-83-21, Clarification of Access Control for Law Enforcement Visits	Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to the licensees. The staff agrees that the COL applicant will address specific access control measures. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-83-22, Safety Evaluation of "Emergency Response Guidelines"	This generic letter stated that the Westinghouse ERG program was acceptable and provided improved guidance for development of plant emergency operating procedures. Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL applicant. Westinghouse should address the ERG program for the AP600 design. This was Open Item 20.7-31. The staff reviewed the Westinghouse AP600 ERG-GW-GJR-100, Revision 3, dated May 1997, and documented its evaluation in Section 18.9.3 of this report. Open Item 20.7-31 is closed and this generic letter is resolved for the AP600 design.
GL-83-26, Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests	This generic letter provided guidance about TS for surveillance of the fuel impurity levels for diesel generators. Westinghouse stated that this generic letter is not applicable to the AP600 design as discussed in Section 1.9, and Appendix 1A, about the inapplicability of RG 1.137, "Fuel-Oil Systems for Standby Diesel Generators." Westinghouse states that the onsite diesel generators and associated fuel-oil systems are non-safety-related. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-83-27, Surveillance Intervals in STS	Westinghouse stated that this generic letter is addressed in SSAR Chapter 16 and Section 3.0. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TS for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-83-28, Required Actions Based on Generic Implications of Salem ATWS Event	This generic letter addressed certain intermediate-term actions to be taken by licensees as a result of the Salem ATWS events on the basis of NUREG-1000. Westinghouse stated that this generic letter is addressed in the Section 7.1.2.2.4 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-83-30, Deletion of STS Surveillance Requirement 4.8.1.1.2.D.6 for Diesel Generator Testing	Westinghouse stated that this generic letter is not applicable to the AP600 design as discussed in Section 1.9, and Appendix 1A, about the inapplicability of RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants." Westinghouse states that the onsite diesel generators are non-safety-related. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-83-32, NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS	Westinghouse stated that this generic letter is addressed in the Section 18.8.2.1.2 of the SSAR. The staff has completed its review of Section 18.8.2.1.2 of the SSAR and finds the Westinghouse position acceptable. Therefore, this generic letter is resolved for the AP600 design.
GL-83-33, NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	Westinghouse stated that this generic letter is addressed in Section 9.5.1 of the SSAR. The staff included, in its review of the AP600 design, the positions of the generic letter. Therefore, this generic letter is resolved for the AP600 design.
GL-83-41, Fast Cold Starts of Diesel Generators	This generic letter requested information on fast cold starts of diesel generators. Westinghouse stated that this generic letter is addressed in Section 16.2 of the SSAR, on plant TSs. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TSs for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. There were widespread concerns from the industry that fast cold start surveillance testing of EDGs may result in premature EDG degradation, and the staff issued this GL to address the possible deleterious effects of fast cold start on EDGs. Westinghouse stated that this GL is not applicable because the diesel generators in the AP600 design are not safety-related. The staff agrees. Therefore, GL 83-41 is not applicable and is considered resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-84-01, NRC Use of the Terms "Important to Safety" and "Safety-Related"	This generic letter explained the staff's position on important to safety in determining what equipment is covered by the quality assurance program for plant design and construction, and plant operation. Westinghouse stated that this generic letter is addressed in SSAR Chapter 17. The staff requested a reference to the specific sections where the GL-84-01 issues are discussed. This was Open Item 20.7-32. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-32 is closed and this generic letter is resolved for the AP600 design.
GL-84-04, Safety Evaluation of Westinghouse Topical Reports on Elimination of Postulated Pipe Breaks in PWR Primary Main Loops.	WCAP-13559, Revision 1, references USI A-2 for the response to this issue. The staff agrees that its evaluation of USI A-2 above in this chapter provides the basis for an acceptable resolution of GL-84-04.
GL-84-09, Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii)	Westinghouse stated that this generic letter is addressed in Section 6.2.4 of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-84-12, Compliance with 10 CFR Part 61 and Implementation of Radiological Effluent TS, Attendant Process Control Program	This generic letter addressed the concern of the compliance with 10 CFR Part 61 and implementation of radiological effluent TS, attendant process control program. Westinghouse stated in a response to RAI Q650.27F (Revision 1) that GL-84-12 has been superseded by GL-89-01, which has been incorporated into Technical Specification 5.5.3, "Radioactive Effluent Control Program," in a manner consistent with the guidance provided in NUREG-1431, Westinghouse Standard Technical Specification. In addition, SSAR Section 11.4.1.3 references 10 CFR 61 for radioactive waste disposal containers and SSAR Section specifies a COL requirement that "The Combined License applicant will develop a process control program in compliance with 10 CFR Section 61.55 and 61.56 for wet solid waste." The staff finds the above response acceptable. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-84-13, Technical Specifications for Snubbers	The purpose of GL-84-013 was to authorize the elimination of a table in plant specific Technical Specifications (TS) which required a list of all snubbers in the plant. Westinghouse stated that this generic letter is addressed in the inservice testing program, Section 3.9.3.4.3 of the SSAR. In the DSER, Open Item 20.7-13 requested a more direct response to this GL. By adoption of the improved Westinghouse STS, Westinghouse should adequately address the TS issues in this generic letter. The TS for the AP600 design are in SSAR Chapter 16 and discussed in Chapter 16 of this report. Subsequent to the issuance of the DSER, the staff determined that this GL is not applicable to the AP600 because the Westinghouse Improved Standard Technical Specifications eliminated all of the requirements relative to snubbers. It was replaced by ANSI/ASME OM Part 4, which is now referenced in ASME Section XI for inservice testing and inspection of snubbers (reference discussions under GL-80-099 and BL-81-001 in this table, and Section 3.9.3.3 of this report). Therefore, DSER Open Item 20.7-13 is closed and GL-84-13 is resolved for the AP600 design.
GL-84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	This GL addresses that the reliability of the EDG has been identified as being one of the main factors affecting the risk from SBO. Thus, attaining and maintaining high reliability of EDGs was a necessary input to the resolution of USI A-44. Westinghouse stated that this GL is not applicable because the diesel generators in the AP600 design are not safety-related and are not required for accident mitigation. The staff agrees, therefore, this GL is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-84-21, Long-Term, Low-Power Operation in PWRs	This generic letter was concerned with core peaking factors being greater than assumed in safety analyses for extended low-power operation followed by a return to full-power operation. Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication to the licensees. The staff does not agree with Westinghouse. During the review of the AP600 safety analysis in Chapter 15 of this report, the staff will consider the effect of extended low power operation on core peaking factors. Westinghouse should address core peaking factors for this event. This was Open Item 20.7-33. The safety evaluation of this issue is in discussed in Chapter 15 of this report. Open Item 20.7-33 is closed and this generic letter is resolved for the AP600 design.
GL-84-24, Certificate of Compliance to 10 CFR 50.49, Environmental Qualification of Equipment Important to Safety	This generic letter required certification from the licensees that the plant environmental qualification program satisfies 10 CFR 50.49, has at least one path to safe shutdown with qualified equipment (or a justification for continued operation), and has all other equipment qualified (or a justification for continued operation). Westinghouse stated that this generic letter is addressed in Sections 3.11 and 7.4, and Appendix 3D of the SSAR. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-85-02, Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issue Regarding Steam Generator Tube Integrity	Westinghouse stated that this generic letter is addressed by Generic Issue A-3 in SSAR Section 1.9.4.2.2. The staff's review of Generic Issue A-3 is documented above. Therefore, GL-85-02 is resolved for the AP600 design.
GL-85-05, Inadvertent Boron Dilution Events	Westinghouse discussed this issue in SSAR Section 15.4.6. The staff evaluated and discussed this issue in Chapter 15 of this report. This GL is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-85-06, Quality Assurance for ATWS Equipment that is not Safety-Related	In WCAP-13559, Revision 1, Westinghouse stated that this generic letter is addressed in Section 1.9.4.2.2 of the SSAR because it is included in the resolution of Issue A-9 for the AP600 design. While it is acceptable for Westinghouse to address AP600 design aspects of this GL through its response to Issue A-9, the staff asked Westinghouse to update WCAP-13559 to identify this issue as a COL action item. This is part of COL Action Item 20.7-3. In Revision 2 to WCAP-13559, Westinghouse identified procurement issues as the responsibility of the COL applicant. Therefore, this GL is resolved for the AP600 design.
GL-85-13, Transmittal of NUREG-1154 Regarding the Davis-Besse Loss of Main and Auxiliary Feedwater Event	DSER Open Item 20.7-34 requested Westinghouse to address NUREG-1154 regarding the Davis-Besse loss of all feedwater event and protecting the AP600 plant against that event. Westinghouse stated that the AP600 does not have an auxiliary feedwater system. The startup feedwater system is not a safety-related system and is not relied on to provide safety-related cooling for the reactor coolant system. The passive core cooling system, including the passive residual heat removal heat exchangers are the safety-related means of providing emergency cooling for the reactor coolant system. Therefore, this GL is not applicable to the AP600 design. The cause of the loss of main and auxiliary feedwater event on June 9, 1985 at Davis-Besse plant was (1) the licensee's lack of attention to detail in the care of plant equipment; (2) the licensee's history of poor performance in troubleshooting, maintenance, and testing of equipment; (3) the fact that licensee's evaluation of operating experience related to equipment did not always find the root causes of problems and corrected; and (4) the licensee's ineffective or not utilized engineering design and analysis effort to evaluate equipment problems. On the basis of the above, the staff finds that the Davis-Besse event is caused by inadequate system maintenance program. Therefore, the staff determined that Westinghouse should address this event in its plant operating and maintenance procedures for the main and startup feedwater systems. Westinghouse states that SSAR Section 13.5 identifies that the COL applicant is responsible for preparation of plant operating procedures. Therefore, DSER Open Item 20.7-34 is closed and GL-85-13, transmittal of NUREG-1154 regarding the Davis-Besse loss of main and auxiliary feedwater event is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-85-16, High Boron Concentrations	This generic letter encouraged reevaluating the need for high boron concentration (about 20,000 ppm boron) in the boron injection tank. Westinghouse stated that this generic letter is not applicable to the AP600 design because the design does not have a boron injection tank. The staff agrees because the design only has a coolant makeup tank with a maximum boron concentration of 3300 ppm boron. Therefore, this generic letter is resolved for the AP600 design.
GL-85-19, Reporting Requirements on Primary Coolant Iodine Spikes	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL applicant. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-86-04, Policy Statement – Engineering Expertise on Shift	This GL has been satisfactorily addressed by Westinghouse in SSAR Section 18.7, "Staffing" and has been identified as a COL responsibility. This was identified as part of COL Action Item 18.6-1. This generic letter is resolved for the AP600 design.
GL-86-07, Transmittal of NUREG-1190 Regarding the San Onofre Unit 1 Loss-of-Power and Water-Hammer Event	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is a plant-specific issue. Westinghouse should address NUREG-1190 and the protection in the AP600 against this type of event. This was Open Item 20.7-35. The resolution of this issue is subsumed by the resolution to Generic Issue A-1, "Water Hammer," which is discussed in Section 20.2 of this chapter. Therefore, Open Item 20.7-35 is closed and this generic letter is resolved for the AP600 design.
GL-86-10, Implementation of Fire Protection Requirements	This generic letter provided guidance on meeting Appendix R to 10 CFR Part 50, which took precedence over GL-83-13. Westinghouse stated that this generic letter is addressed in Section 9.5.1 of the SSAR. The staff included this generic letter in its review of the AP600 design in Section 9.5.1 of this report. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-86-13, Potential Inconsistency Between Plant Safety Analyses and TS	Westinghouse stated that this generic letter is not applicable to the AP600 design because it was only for B&W and CE plants. This is not acceptable. Westinghouse should address the consistency between plant safety analyses and the TS for the AP600. This was Open Item 20.7-36. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-36 is closed and this generic letter is resolved for the AP600 design.
GL-86-15, Information Relating to Compliance With 10 CFR 50.49, "Environmental Qualification of Equipment Important to Safety for Nuclear Power Plants	This generic letter provided guidance on licensee actions in cases where the environmental qualification of equipment is suspect and on then-current NRC policy with regard to enforcement of 10 CFR 50.49. Westinghouse stated that this generic letter is addressed in Section 3.11 and Appendix 3D of the SSAR. The staff requested that Westinghouse address if the AP600 meets the requirements in 10 CFR 50.49. This was Open Item 20.4-37. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-37 is closed and this generic letter is resolved for the AP600 design.
GL-86-16, Westinghouse ECCS Evaluation Models	This generic letter concerned the need for additions and corrections to ECCS evaluation models for certain computer codes. Westinghouse stated that this generic letter was not applicable to the AP600 design, as explained in SSAR Chapter 15. The staff requested that Westinghouse discuss the ECCS evaluation models for the AP600 design and list the specific SSAR section(s) involved with the issues in this generic letter. This was Open Item 20.7-38. Westinghouse discussed this issue in SSAR Sections 6.3.5 and 15.0.11. The staff evaluated the Westinghouse ECCS models and discussed this issue in Chapter 15 of this report. Open Item 20.7-38 is closed and this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-87-03, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, USI A-46.	This GL is not applicable to the AP600 design. Reference the staff's evaluation of Issue A-46 in this chapter.
GL-87-06, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	WCAP-13559, Revision 2, Westinghouse states that this generic letter is addressed in LCO3.4.16, "RCS Pressure Isolation Valve (PIV) Integrity." of the plant TS in SSAR Chapter 16. The staff's evaluation of this issue is discussed in Section 3.9.6.2 of this report as a part of the resolution of DSER Open Item 3.9.6.2-8. In this evaluation, the staff concluded that SSAR Table 3.9-18 contains an acceptable list of PIVs, and LCO 3.4.16 in the TS contains acceptable leak testing criteria for these PIVs. On the basis of this evaluation in Section 3.9.6.2, the staff concludes that GL-87-006 is resolved for the AP600 design.
GL-87-09, Sections 3.0 and 4.0 of STS on Limiting Conditions for Operation and Surveillance Requirements	Westinghouse stated that this generic letter is addressed in SSAR Chapter 16 and Section 3.0. The staff requested that Westinghouse identify the specific sections in SSAR Chapter 16. This was Open Item 20.7-39. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-39 is closed and this generic letter is resolved for the AP600 design.
GL-87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements	Westinghouse stated that this generic letter is addressed in Section 3.6.2 of the SSAR. This generic letter issued Revision 2 to BTP MEB 3-1 of SRP 3.6.2 to eliminate the guidelines for postulating arbitrary intermediate pipe ruptures. SSAR Section 3.6.2 provides information relative to postulating pipe ruptures that is consistent with MEB 3-1, Revision 2. Therefore, GL-87-11 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-87-12, Loss of Residual Heat Removal While the Reactor Coolant System Is Partially Filled	As a result of the loss of the decay heat removal function occurring in operating plants, this generic letter requested licensees to provide information regarding midloop operation, and GL-88-17 provided guidance to licensees. Westinghouse stated that this generic letter is addressed in Section 1.9.5.1 of the SSAR. The generic letter is addressed in midloop operation for SECY-90-016 issues in the section, and requested information from and provided guidance to ALWR vendors to ensure high reliability of the shutdown decay heat removal system. The SRM to SECY-90-016 provided four additional recommendations for decay heat removal during midloop operation. Westinghouse submitted WCAP-14837, Rev 2 (11/97) that discusses shutdown risk concerns, including potential loss of RNS. The staff resolved this issue and its evaluation is discussed in Chapter 19 of this report. This generic letter is resolved for the AP600 design.
GL-88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary in PWR Plant Components	This generic letter requested assurance that licensees had implemented a program to ensure that boric acid corrosion does not degrade the reactor coolant pressure boundary. Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL applicant. The staff agrees that this is a maintenance issue and within the scope of the COL applicant. This was identified in the DSER as part of COL Action Item 20.7-2. It is not a design issue. It is redesignated as part of COL Action Item 20.7-3. Therefore, this generic letter is resolved for the AP600 design.
GL-88-07, Modified Enforcement Policy Relating to 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants"	Westinghouse stated that this generic letter is addressed in Sections 1.9 and 3.11 of the SSAR. The staff requested that Westinghouse be more specific and explain where in SSAR Section 1.9 that this generic letter is addressed. This was Open Item 20.7-40. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-40 is closed and this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations	Westinghouse stated that this generic letter is addressed in Section 1.9.4 and Appendix 1A of the SSAR because it involves Issue A-47 and RG 1.99, "Radiation Embrittlement of Reactor Vessel materials." Issue A-47 is on safety implications of control systems and is discussed for the AP600 design in Section 20.2 of this report. Issues A-11 and 15 involve reactor vessel materials and radiation, and are discussed in Sections 20.2 and 20.3, respectively, of this report. Therefore, this generic letter is resolved for the AP600 design.
GL-88-12, Removal of Fire Protection Requirements from TS	Westinghouse stated that this generic letter is addressed in Section 5.7 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-41. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-41 is closed and this generic letter is resolved for the AP600 design.
GL-88-14, Instrument Air Supply System Problems Affecting Safety-Related Equipment	Westinghouse stated that this generic letter is addressed in Section 9.3.1 of the SSAR. See resolution of Issue 43 in Section 20.3 of this report. Therefore, this generic letter is resolved for the AP600 design.
GL-88-15, Electric Power Systems — Inadequate Control over Design Process	This GL informs the licensees of the various problems with electrical systems being identified with increasing frequency at nuclear power plants. This refers to the problems of onsite distribution system voltages lower than required for proper operation of safety equipment, diesel generator loading exceeding design, inadequate diesel generator response to actual loading, overloading Class 1E buses, inadequate breaker coordination, and inadequate fault current interruption capability. For GL-88-15, Westinghouse referred to SSAR Section 8.3.1.1.2.1 in WCAP-13559, Revision 1, "Operational Assessment For AP600." SSAR Section 8.3.1.1.2.1 stated that the onsite diesel generators are non-safety-related. The final resolution of the RTNSS issue resulted in the determination that administrative availability controls of ac power from the diesel generators are needed in all modes. See Section 8.6.2.4 of this report for additional discussion on the RTNSS resolution of this item. The breaker coordination and fault current interruption capability will be considered by Tier 1 and ITAAC. Therefore, GL 88-15 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-88-16, Removal of Cycle-specific Parameter Limits from Plant TS	Westinghouse stated that this generic letter is addressed in Section 5.9 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-42. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-42 is closed and this generic letter is resolved for the AP600 design.
GL-88-17, Loss of Decay Heat Removal	This generic letter concerned loss of decay heat removal during nonpower operation. Westinghouse stated that this generic letter is discussed in Section 1.9.5.1 of the SSAR. This generic letter and GL-87-12 are addressed in midloop operation for SECY-90-016 issues in the section, and requested information from and provided guidance to licensees. The SRM to SECY-90-016 provided four additional recommendations for decay heat removal during midloop operation. Westinghouse submitted WCAP-14837, Rev. 2, (11/97) that discusses shutdown risk concerns, including potential loss of RNS. The staff resolved this issue and its evaluation is discussed in Section 19.3 of this report. Therefore, this GL is resolved for the AP600 design.
GL-88-18, Plant Record Storage on Optical Disks	This generic letter approved the use of this method of record keeping at nuclear plants when appropriate quality assurance controls are applied. Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL Applicant. This issue is not a design issue because the use of optical disks for storage of records is optional and within the scope of the COL applicant. If this method is selected, it should be addressed in the applicable quality assurance program. Therefore, this generic letter is resolved for the AP600 design.
GL-88-20, Individual Plant Examination for Severe-Accident Vulnerabilities	Westinghouse stated that this generic issue is addressed in the AP600 Probabilistic Risk Assessment Report. This report is separate from the SSAR. Risk insights are already an integral part of the staff's AP600 design review process as discussed in Chapter 19 of this report on severe accidents and PRA for the design. This generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-89-01, Implementation of Programmatic and Procedural Controls for Radiological Effluent TS	Westinghouse stated that this generic letter is addressed in Section 5.7 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-43. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-43 is closed and this generic letter is resolved for the AP600 design.
GL-89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products	Westinghouse stated that this generic letter is not applicable to the AP600 design because procurement of equipment and parts is the responsibility of the COL applicant. The staff agrees that this generic letter is not applicable to the AP600 design because it involves the procurement of vendor products, which is within the scope of the COL applicant. This was identified in the DSER as part of COL Action Item 20.7-2. It is not a design issue. It is redesignated as part of COL Action Item 20.7-3. This generic letter is resolved for the AP600 design.
GL-89-04, Guidance on Developing Acceptable Inservice Testing Programs	In WCAP-13559, Revision 2, Westinghouse stated that this generic letter is addressed in Section 3.9.6.2 of the SSAR. The staff's evaluation and acceptance of the AP600 inservice testing program was based on the information in SSAR 3.9.6, and is documented in Section 3.9.6 of this report. Therefore, GL-89-004 is resolved for the AP600.
GL-89-07, Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is a procedural issue, which is the responsibility of the COL applicant. Industrial security and sabotage protection for the AP600 plant is discussed in Section 13.6 of this report and in the resolution of Issue A-29 in Section 20.2 of this report. However, the issues involved in this generic letter are the responsibility of the COL applicant. This was identified in the DSER as COL Action Item 20.7-6. It is redesignated COL Action Item 20.7-8. This generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-89-08, Erosion/Corrosion Induced Pipe Wall Thinning	This generic letter requested information on the long-term erosion/corrosion monitoring program that provided assurance that the structural integrity of all high energy carbon steel systems will be maintained. Westinghouse stated that this generic letter is a surveillance issue, which is the responsibility of the COL applicant and is discussed in Sections 5.4.3.4 and 10.3.6 of the SSAR. The staff agrees with this assessment. Further, all open issues associated with erosion/corrosion have been resolved for the AP600 design. Therefore, this generic letter is resolved for the AP600 design.
GL-89-10, Safety-Related Motor Operated Valve Testing and Surveillance	WCAP-13559, Revision 2 references SSAR Section 3.9.6.2 as the basis for resolution of this GL. As discussed in Section 3.9.6.2 in Chapter 3 of this report, the staff has concluded that the commitments in SSAR Sections 3.9.6 and 5.4.8 relative to inservice and qualification testing of motor-operated valves provides an acceptable basis to resolve GL-89-10 for the AP600 design.
GL-89-13, Service Water System Problems Affecting Safety-Related Systems	This generic letter requested information about compliance of service water systems with certain GDC and quality assurance requirements, as test control. Westinghouse stated that this generic letter is addressed in Section 16.2 of the SSAR on plant TSs. This response did not appear to address the issues raised on the GDC and quality assurance requirements in the generic letter. This was Open Item 20.7-44. Westinghouse states in Revision 1 to WCAP-13559 that the service water system is not used for safety-related cooling in the AP600. Therefore, this generic letter is not applicable to the AP600. The staff agrees with Westinghouse on its assessment. Therefore, Open Item 20.7-44 is closed and this generic letter is resolved for the AP600 design.
GL-89-14, Line-item Improvements in Technical Specifications — Removal of 3.25 Limit on Extending Surveillance Intervals	Westinghouse stated that this generic letter is addressed in Section 3.0 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-45. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-45 is closed and this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-89-15, Emergency Response Data System	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL applicant. In the DSER the staff requested that Westinghouse address what data within the AP600 design should be emergency response data. This was DSER Open Item 20.7-46. The staff now agrees that this will be the responsibility of the COL applicant. It is noted, however, that Appendix E to 10 CFR Part 50, Section VI 2(a)(i) provides the selected plant parameters for PWRs. Due to the unique design of the AP600, the plant parameters required for the Emergency Response Data System (ERDS) will be similar but not all inclusive. The reminder to the COL applicant to review this Generic Letter and Appendix E to assure that the necessary and sufficient AP600 plant parameters are available to the ERDS is part of COL Action Item 20.7-9. Therefore, DSER Open Item 20.7-46 is closed and this generic letter is resolved for the AP600 design.
GL-90-02, Alternative Requirements for Fuel Assemblies in the Design Features Section of TS	Westinghouse stated that this generic letter is addressed in Section 4.2 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-47. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-47 is closed and this generic letter is resolved for the AP600 design.
GL-90-09, Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	This generic letter involved a line-item improvement to plant TSs. For the same reasons discussed in the staff's evaluations of GL-80-099 and GL-84-013 above in this table, the staff determined that GL-90-09 is not applicable to the AP600 design.
GL-91-01, Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from TS	Westinghouse stated that this generic letter is addressed in SSAR Chapter 16. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-48. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-48 is closed and this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-91-04, Changes in TS Surveillance Intervals to Accommodate a 24-month Fuel Cycle	Westinghouse stated that this generic letter is addressed in Sections 3.1 to 3.9 and Chapter 16 of the SSAR. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-49. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-49 is closed and this generic letter is resolved for the AP600 design.
GL-91-05, Licensee Commercial-Grade Procurement and Dedication Programs	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is a procurement issue which is the responsibility of the COL applicant. In WCAP-13559, Revision 1, Westinghouse states that this generic letter is not applicable to the AP600 design because it addresses a procurement issue. While the staff agrees that this generic letter is not applicable to the AP600 design certification phase, Westinghouse needs to revise WCAP-13559 to identify this issue as a COL action item. This is part of COL Action Item 20.7-3. In Revision 2 of WCAP-13559, Westinghouse stated that procurement issues are the responsibility of the COL applicant. Therefore, this generic letter is resolved for the AP600 design.
GL-91-07, GSI-23, "Reactor Coolant Pump Seal Failure" and its Possible Effect on Station Blackout	Westinghouse stated that this generic letter is not applicable to the AP600 design, as discussed in Section 5.1.3.3 of the SSAR. Issues A-44 and 23 (discussed in Sections 20.2 and 20.3, respectively, of this report) are involved with station blackout and reactor coolant pump seal failure; however, this generic letter is outside these issues. The staff resolved GSI-23 because the AP600 design does not have RCP seals. This generic letter is not applicable to AP600 canned pump design. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-91-08, Removal of Component Lists from TS	Westinghouse stated that this generic letter is addressed in SSAR Chapter 16. The staff requested that Westinghouse identify the specific section in SSAR Chapter 16. This was Open Item 20.7-50. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-50 is closed and this generic letter is resolved for the AP600 design.
GL-91-09, Modification of Surveillance Interval for the Electrical Protection Assemblies in Power Supplies for the Reactor Protection System	This generic letter provided guidance for the plant TS surveillance interval for these assemblies because the then-current TS only required the channel check at 6-month intervals instead of each time the plant is in cold shutdown for greater than 24 hours, unless performed in the last 6 months. Westinghouse stated that this generic letter is not applicable to the AP600 design because it was addressed only to BWR plants. Westinghouse should address the proper surveillance interval for these assemblies for the AP600 design. This was Open Item 20.7-51. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, Open Item 20.7-51 is closed and this generic letter is resolved for the AP600 design.
GL-91-15, Operating Experience Feedback Report, Solenoid-Operated Valve Problems at U.S. Reactors	This generic letter informed licensees of a case study on solenoid-operated valves by AEOD. No specific action was requested. Westinghouse stated that this generic letter is not applicable to the AP600 design because it involved procurement and maintenance issues, which are the responsibility of the COL applicant. The staff agrees with Westinghouse as stated in the resolution of Issue I.C.5 in Section 20.4 of this report. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-91-16, Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	This generic letter described the then-current federal legislation on fitness for duty and changes to 10 CFR Part 2, Appendix C, and 10 CFR Part 55, regarding fitness for duty. No specific actions were requested. Westinghouse stated that this generic letter is not applicable to the AP600 design because it was an administrative communication with licensees. Westinghouse should discuss the responsibility of the COL applicant. This was Open Item 20.7-52. Westinghouse, in SSAR Chapter 13, states that the COL applicant will address site-specific information relative to its security, contingency, and training and qualification plans and its fitness-for-duty program. This issue was identified as part of COL Action Item 13.6.13-1. These measures will satisfy the conditions of the generic letter. Therefore, Open Item 20.7-52 is closed and this generic letter is resolved for the AP600 design.
GL-91-17, GSI 29, "Bolting Degradation or Failure in Nuclear Power Plants"	Westinghouse stated that this generic letter is not applicable to the AP600 design because the letter addresses a maintenance issue within the responsibility of the COL applicant. The staff agrees that this is a maintenance issue and within the scope of the COL applicant. This issue is part of COL Action Item 20.7-3. Therefore, this generic letter is resolved for the AP600 design.
GL-92-01, Revision 1, Reactor Vessel Structural Integrity	This generic letter requested information to assess compliance with the requirements and commitments regarding reactor vessel integrity in view of concerns raised by the staff review of such integrity for the Yankee Nuclear Power Station. Westinghouse did not address this generic letter. As stated in Section 20.7.2.1 of this report, Westinghouse should revise WCAP-13559 to include the bulletins and generic letters that were issued after December 31, 1991. This was Open Item 20.7-1. Westinghouse responded to this item by stating that the reactor vessel integrity is addressed in SSAR Sections 5.3.2 and 5.3.3. All open issues associated with the SSAR sections have been resolved. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-92-02, Resolution of Generic Issue 79, "Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown"	The resolution of this issue is discussed under Issue 79 in Section 20.3 of this report. This generic letter is resolved for the AP600 design.
GL-92-08, Thermo-Lag 330-1 Fire Barriers	This is discussed for BL-92-01 in Table 20.7-1 of this report. Westinghouse did not address this generic letter. In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-93-01, Emergency Response Data System Test Program	Westinghouse stated that this generic letter is not applicable to the AP600 design because it is the responsibility of the COL applicant. The staff agrees that this will be the responsibility of the COL applicant. The reminder to the COL applicant to review this generic letter for requirements related to ERDS testing and to coordinate the testing of the ERDS with the NRC ERDS Project Manager is part of COL Action Item 20.7-9.
GL-93-04, Rod Control System Failure and Withdrawal of Rod Cluster Assemblies	This generic letter addressed the single failure vulnerability within the Westinghouse solid state rod control system that could cause inadvertent withdrawal of control rods in a sequence resulting in a power distribution not considered in the design-basis analyses. Westinghouse addresses this issue in Section 3.9.4 of the SSAR. The staff's evaluation is discussed in Chapter 4 of this report. This generic letter is resolved for the AP600 design.
GL-93-05, Line-item TS Improvements to Reduce Surveillance Requirements for Testing During Power Operation	In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-93-07, Modification of the TS Administrative Control Requirements for Emergency and Security Plans	In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-93-08, Relocation of TS Tables of Instrument Response Time Limits	In a letter dated September 21, 1995, the staff informed Westinghouse that this issue was no longer considered relevant to the design review for the AP600. Therefore, this generic letter is resolved for the AP600 design.
GL-95-03, Circumferential Cracking of SG Tubes	Westinghouse stated that this generic letter is addressed by addressing Generic Issue A-3 in SSAR Section 1.9.4.2.2. The staff's review of Generic Issue A-3 is documented above, and Generic Issue A-3 is resolved for the AP600 design. Further, additional discussion concerning steam generator materials for the AP600 steam generators is provided in Section 5.4.2.1 of this report, and all open items associated with this section have been addressed and resolved. Therefore, this generic letter is resolved for the AP600 design.
GL-95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	This GL requested all holders of operating licenses or construction permits for nuclear plants to identify all safety-related power-operated gate valves in their plants that may be susceptible to pressure locking or thermal binding, and take necessary corrective actions to ensure operability of applicable valves. For the AP600 design certification, the staff's position is that, in the design of applicable valves, a commitment to incorporate provisions to prevent these situations from occurring is sufficient to resolve this GL. WCAP-13559, Revision 2 references SSAR Sections 5.4.8.1.2 and 5.4.8.2 for resolution of this GL. The staff has concluded that these two SSAR sections contain sufficient commitments to design applicable valves so that there is reasonable assurance that pressure locking and thermal binding will not occur. Therefore GL-95-007 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-96-01, Testing of Safety-Related Logic Circuits	This generic letter addressed problems with the testing of safety-related logic circuits. A number of NRC regulations document the requirements to test safety-related systems to ensure that they will function as designed when called upon. Westinghouse addresses testing of safety-related logic circuits in Section 7.1.2 of the SSAR. However, the action of comparing electrical schematic drawings and logic diagrams against plant surveillance test procedures to ensure that the surveillance procedures fulfill the technical specification requirements are the responsibility of the COL applicant. This issue is identified as part of COL Action Item 13.6.13.1-1. The staff considers this generic letter resolved for the AP600 design.
GL-96-02, Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat	Westinghouse states, in Section 13.6 of the SSAR, that the comprehensive security program is the responsibility of the COL applicant and will be addressed in the security plan, contingency plan, and guard training plan. The staff is in agreement, and these measures will satisfy the conditions of the generic letter. Therefore, this generic letter is resolved for the AP600 design.
GL-96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves	WCAP-13559, Revision 2 references SSAR Sections 3.9.6.2 and 5.4.8.5 for resolution of this GL. As discussed in Chapter 3, Section 3.9.6.2 of this report, the staff concluded that SSAR Section 3.9.6 and SSAR Table 3.9-16 contain commitments for the AP600 to develop an inservice test program consistent with the recommendations in GL-89-10 and its supplements, and GL-96-05 for MOVs and power-operated valves other than MOVs to demonstrate their design basis capability throughout the plant life (Ref. resolution of DSER Open Items 3.9.6.2-1 and 3.9.6.2-3). SSAR Sections 3.9.8.4 and 5.4.8.5 contain COL commitments relative to inservice testing in conformance with SSAR Section 3.9.6 and Table 3.9-16, and insitu testing to confirm the capacity of the valve to operate under design conditions. Therefore, GL-96-05 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and december 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	This generic letter addressed concerns associated with water hammer, two-phase flow, and thermally induced overpressurization. The generic letter requested that licensees evaluate systems that were found to be vulnerable to these conditions, perform operability determinations as appropriate per the guidance contained in GL 91-18, and take necessary corrective actions per TS and 10 CFR 50 Appendix B requirements. If corrective actions were required, the generic letter reminded licensees of their responsibility to ensure that systems remained operable and could continue to perform their safety functions in the interim while corrective actions were being implemented. In Revision 2 of WCAP-13559, Westinghouse stated that Section 6.2.2 of the SSAR specifies that the cooling water to the containment fan coolers is not safety-related. Westinghouse also stated that Section 6.2.3.1.3 of the SSAR specifies that the containment penetrations are protected from overpressurization. The staff finds this response acceptable. Therefore, this generic letter is resolved for the AP600 design.
GL-97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	For resolution of this GL, WCAP-13559, Revision 2 references SSAR 5.2.3.1 which states that the use of nickel-chromium-iron alloy in the AP600 reactor coolant pressure boundary is limited to Alloy 690, and Alloy 600 is used only for buttering applications. This is acceptable and, therefore, GL-97-001 is resolved.
GL-97-04, Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	This generic letter deals with assurance of adequate net positive suction head for emergency core cooling system pumps. Since the AP600 does not use emergency core cooling system pumps, this generic letter is not applicable to the AP600 design.
GL-97-05, Steam Generator Tube Inspection Techniques	WCAP-13559, Revision 2 states that this generic letter is not applicable to the AP600 design. The staff agrees because this generic letter relates to steam generator tube inspections that are conducted in accordance with approved inspection procedures, and as such, are outside the scope of design certification. On this basis, GL-97-05 is resolved for the AP600 design.

Table 20.7-2 Resolution of Applicable Generic Letters Issued Between January 1, 1980, and December 31, 1997, for the Westinghouse AP600 Design (continued)

Generic Letter No. and Title	Staff Resolution
GL-97-06, Degradation of Steam Generator Internals	WCAP-13559, Revision 2, states that this generic letter is not applicable to the AP600 design because tube supports are fabricated from stainless steel, as discussed in SSAR Section 5.4.2.2. The staff agrees with this assessment. Since the support plates are made from stainless steel, no degradation of these components is expected. This is acceptable. Therefore, GL-97-06 is resolved for the AP600 design.

21 TESTING AND COMPUTER CODE EVALUATION

21.1 Introduction

The AP600 is the first "passive" advanced light-water reactor (ALWR) design reviewed by the NRC. Its distinguishing feature is a dependence on safety systems whose operation is driven by "natural" forces, requiring no continuously operating, electrically (alternating current (ac)) powered, mechanical components such as pumps. In the context of the AP600, the natural forces include gravity and stored mechanical energy. Also included in the definition of "passive" systems are components that may require electrical (direct current (dc)) power, supplied by batteries, to change state; however, these components are allowed to change state only once (i.e., a valve can change from "closed" to "open," but must remain open thereafter). Check valves, which require no power and open or close according to the differential pressure across the valve are also included in the passive systems.

An initial review of the Westinghouse AP600 testing program was conducted and documented in the draft safety evaluation report (DSER) in November 1994. In the DSER, the staff identified open items that were required to be resolved to enable to the staff to complete its review. In April 1996, the staff issued a supplement to the DSER (SDSER) regarding the testing program and computer code development and qualification efforts. In the SDSER, the staff identified additional open and confirmatory items and updated the status of and closed existing DSER open items. This report encompasses the staff's evaluation of the entire review.

21.1.1 Passive Emergency Injection Systems

The AP600's emergency core cooling (ECC) injection systems include the following:

- core makeup tanks (CMTs), which initially inject cold, borated water into the reactor coolant system (RCS), and which can operate either in a natural-circulation-driven recirculatory mode or by gravity drain

- accumulators, pressurized by compressed nitrogen gas, which also inject cold, borated water into the RCS if RCS pressure is less than 700 psig

- in-containment refueling water storage tank (IRWST), which provides borated water by gravity drain to the RCS

The accumulators are isolated from the RCS during normal operation by means of check valves in their injection lines that are held shut by RCS pressure. The CMTs are prevented from injecting into the RCS during normal operations by fail-open air-operated valves in the CMT discharge lines which open on an actuation signal. The IRWST is isolated from the RCS by a check valve and squib valve in series in each IRWST discharge line.

21.1.2 Ultimate Heat Sink

The ultimate heat sink for core decay heat in the AP600 is provided by the passive containment cooling system (PCS), which condenses steam released into containment on the inside surface of the steel containment shell. The steam can result from efflux from an RCS breach or a steamline break, or from boiling in the IRWST as a result of long-term passive residual heat removal operation. The condensate returns to the IRWST or to the containment sump and from either of which, may be recirculated to the RCS if necessary.

21.1.3 Passive Residual Heat Removal System

The passive residual heat removal (PRHR) system cools the core in the event that heat removal via the steam generators is not available. The PRHR system, which can operate up to full RCS pressure, contains one heat exchanger (HX) tube bundle submerged in the IRWST. Water flows by natural circulation from one hot leg through the HX and returns to one of the steam generator cold-leg channel heads. This system can also function with the reactor coolant pumps (RCPs) operating.

21.1.4 Automatic Depressurization System

The automatic depressurization system (ADS) consists of two independent sets of depressurization valves, arranged in four stages. The first three stages of each set are connected to the RCS at the top of the pressurizer and exhaust through pipes and a sparger into the IRWST where steam contained in the effluent is condensed. The fourth stage is connected to the hot leg (one bank per hot leg) and exhausts directly to the containment. The first stage of the ADS is actuated when the liquid volume of either of the CMTs reaches 67.5 percent full; timers are also actuated, which then control the opening of the second and third ADS stages. The fourth stage opens when either of the CMTs reaches 20 percent full.

Operation of the ADS is crucial to the performance of the passive safety injection systems. While the CMTs are connected directly to RCS cold legs and operate at full RCS pressure, the gas pressure of the accumulators is approximately 4.8 MPa (700 psig), and the IRWST is at containment pressure and cannot drain into the RCS until the RCS pressure is reduced to less than approximately 0.1 MPa (14 psi) above containment pressure.

21.1.5 Unique Characteristics of the Passive Design

All active systems in the AP600, such as pumped normal residual heat removal (RNS), startup feedwater, and chemical and volume control system (CVS), and including the diesel generators that provide onsite ac power, are classified as non-safety-related components. The AP600 thus represents a significant departure from both the current generation of operating reactors and the "evolutionary" light-water reactors (LWR) in its dependence on passive safety systems.

Although passive systems may be conceptually simpler than conventional active systems, they are potentially more susceptible to systems interactions that can upset the balance of forces on which passive safety systems depend for operation. In addition, with the exception of compressed-gas-driven accumulators, there is very little experience with these types of safety systems. The unique characteristics of these types of systems are explicitly recognized in the regulations governing the evaluation of standard plant designs. In 10 CFR 52.47(b)(2)(i)(A),

NRC states that the following three requirements, among others, must be met for a plant that differs significantly from those designs that are evolutionary changes from LWR designs of plants which have been licensed and in commercial operation before the effective date of 10 CFR Part 52 or a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions":

(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.

(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

(3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

These requirements have been interpreted to require that a passive plant vendor must develop and perform design certification test programs of sufficient scope. This includes both separate-effects and integral-systems experiments to provide data to assess the computer codes used to analyze plant behavior over the range of conditions described in Item 3 above.

To satisfy the requirements of 10 CFR 52.47(b)(2)(i)(A), Westinghouse has developed test programs to investigate the behavior of the passive reactor and containment safety systems, including both component and phenomenological (separate-effects) tests and integral-systems tests. In this chapter, the NRC staff evaluates the capability of the AP600 test programs to satisfy these regulatory requirements.

For systems-related testing, the staff required that Westinghouse submit, as a minimum, the following material for each test program:

- a test specification, describing the test facility, test objectives, and test matrix
- for those facilities testing scaled systems or components, a scaling report, demonstrating that the test facility met appropriate geometric and thermal-hydraulic similarity criteria, and that the data would cover a parametric phenomenological range comparable to that expected in the AP600 plant
- a report or reports containing a record of data from all tests in the facility, an uncertainty/error evaluation of the test data, and analyses of selected tests (as approved by the NRC staff) demonstrating an understanding of the sequence of events and key phenomena influencing system (or component) behavior

The next section describes the major issues pertaining to passive safety system performance that Westinghouse needed to address in the design certification testing programs. An overview of each of the test programs is then presented, followed by a summary of the NRC's activities in reviewing each of the test programs. The remainder of this chapter is devoted to individual evaluations of each program, a description and review of Westinghouse's code validation

program, and the staff's determination regarding the compliance of the design certification test programs with the regulatory requirements previously described.

21.2 Issues of Concern

Most of the safety systems in the AP600 design represent new concepts in plant safety system design. As a result, there are numerous questions and issues connected with the performance of these unique components and systems. These issues range from relatively straightforward phenomenological questions to complex, system-related concerns. Major issues related to each safety system are discussed below.

21.2.1 Core Makeup Tanks

The AP600 has two CMTs, each with a volume of about 56.6 m^3 (2000 ft^3), which are initially filled completely with cold, borated water. The top of each tank is connected by means of a pressure balance line (PBL) to an RCS cold leg; this connection maintains the tanks at RCS pressure. A discharge line connects the bottom of each tank to a direct vessel injection (DVI) line, which provides ECC flow to the vessel downcomer. In many transients and accidents, after the CMT discharge line isolation valves open, the CMTs begin to recirculate, with the cold, borated water flowing into the RCS being replaced by RCS water flowing up through the PBLs. When either RCS pressure or inventory is reduced to the point at which vapor is introduced into the PBLs, the recirculatory loop is broken and the CMTs begin to drain into the RCS. Specific issues related to CMT operation include the following:

- recirculation and gravity drain behavior, including condensation during draining
- thermal stratification in the CMTs
- effects of system depressurization on heated CMT behavior

The thermal-hydraulic response of the CMTs also affects the ADS. The first stage of the ADS is actuated when the water level in either of the two CMTs reaches a level corresponding to 67.5-percent volume, the subsequent two stages then actuate on timers, and the fourth stage actuates again on level, corresponding to 20-percent volume.

When the DSER and SDSER were issued, the reference design for the level detectors in the CMTs that produce the actuation signals was a heated resistance temperature device (RTD). The heated RTD was supposed to discriminate between being covered by liquid and being surrounded by vapor. The capability of this instrument to function properly, both in terms of actuating the ADS when needed and keeping the ADS from actuating when it is not required, was a key issue in the SDSER and was investigated as part of the CMT testing program. The results from the test program raised questions about the ability of the instrument to accomplish its design functions. As a consequence of these results, Westinghouse has changed the reference CMT level instrument to narrow-range differential pressure level switches. The evaluation of the CMT level instrumentation is found in Section 7.3.6 of this report. The staff considers the "key issue" with regard to testing the performance of the CMT level sensor to be resolved.

21.2.2 Automatic Depressurization System

The ADS comprises four stages of depressurization valves. The first three stages are connected to a piping network off the top of the pressurizer; the fourth stage is connected to the RCS hot leg. There are two complete networks connected to the pressurizer and one fourth-stage assembly on each hot leg. Each of the first three stages consists of two valves in series, an "isolation" valve, which opens first, and a "control" valve, which opens shortly thereafter. Effluent from the RCS travels through the pressurizer, exhausts into a pipe, and flows into the IRWST through a sparger.

Each fourth-stage assembly contains four valves, which are arranged in two parallel flow paths, each path with two valves in series. The two valves in series are again designated as "isolation" and "control." Effluent from the fourth stage of ADS exhausts directly into the containment.

At the time the ADS design certification tests were performed, valve performance characteristics for each stage were specified in terms of such parameters as throat area and effective resistance because the particular valve type (e.g., gate, globe, squib) had not been specified. In Section 21.3.2 of this report, the staff discusses how the ADS test information will be used to develop ADS valve specifications for the actual ADS valves. The staff notes that Revision 5 of the standard safety analysis report (SSAR) has specified valve types for all four stages of ADS valves.

The following issues are important to ADS operation:

- ADS valve performance and reliability (discussed in Section 19.1.8 of this report)
- critical flow through the various components of the ADS network (valves, pipes, spargers), including transition from critical to sub-critical flow
- the effects of condensation in the IRWST on ADS system performance, including mechanical and thermal loading on the IRWST and submerged mechanical components such as pipes, spargers, and heat exchangers
- the severity of IRWST vibration due to ADS system performance (discussed in Section 3.2.8 of this report)

21.2.3 Passive Residual Heat Removal System

The main component of the PRHR system is a heat exchanger, comprising a C-shaped bundle of several hundred tubes. The bundle is completely submerged in the IRWST. The PRHR system can cool the RCS either by forced or natural circulation in the event of a complete loss of feedwater (main and startup) to the steam generators. The heat exchanger bundle is designed to remove about 2 percent of AP600 full power using natural circulation, and about twice that amount using forced convection. Heat transfer on the inside (primary side) of the tubes is initially single-phase (liquid) convection, although during some accidents, steam may also enter

the tubes after some time; heat transfer on the outside (IRWST side) of the tubes is either single-phase natural convection (liquid) or boiling, depending on the temperatures of the primary and IRWST sides. The main issues related to PRHR system operation follow:

- natural convection heat transfer in the tube bundle
- heat transfer on the IRWST side of the tubes, especially in two-phase flow, where critical heat flux and vapor blanketing of the tubes may be of concern
- mechanical loads on heat exchanger tubing and supports, including consideration of potential steam hammer load caused by phase separation in tubes' primary side under accident conditions (This issue is discussed in Section 3.6.3.6 of this report)

21.2.4 Check Valves

The check valves in the AP600's safety systems are generally of conventional design. At the CMT discharge, the check valves are installed so that they are "biased open" (i.e., the disc's normal position, with no differential pressure across the valve, is partly open). These components are not unique to the AP600; however, the conditions under which they will be operating in the plant are substantially different from those in conventional plants. The major issues pertaining to these valves follow:

- operation with the very low differential pressures characteristic of natural circulation and gravity drain systems, especially the ability to open on demand after long periods of being held closed by fluid at RCS temperature, pressure, chemistry (This issue is discussed in Section 3.9.6 of this report)
- integrity of valve disc, especially at its supporting hinge, under flow-induced vibratory loads (This issue is discussed in Section 3.9.6 of this report)

21.2.5 Interdependency of Systems

In addition to component and safety system performance on a system-by-system basis, there are systems interactions that must also be considered in the AP600. The safety systems in the AP600 are highly interdependent in their influence on plant response. For instance, it is expected that during a small-break, loss-of-coolant accident (SBLOCA), a low pressurizer level or pressure signal will generate a safety injection ("S") signal. CMT and PRHR discharge valves then open. The rate at which the RCS is cooled by the CMT and PRHR influences the depressurization rate, CMT recirculation and draining, and possibly accumulator injection. CMT level then actuates the ADS, which proceeds to reduce plant pressure to near containment pressure, ultimately allowing IRWST injection; the plant then makes a transition into long-term recirculatory behavior that also involves the PCS. Interactions may also occur if operators actuate non-safety-related active systems to attempt to cope with accidents. The close coupling between AP600 safety systems and the multiple flow paths that can develop, especially with the small pressure differentials characteristic of natural circulation flows, makes the AP600 a complex system to analyze. It is, therefore, necessary to understand integral-systems response over the entire pressure range of the AP600, and to have data from such experiments to test on a system-wide basis, as well as component and phenomenological models developed from separate-effects tests.

21.2.6 Containment Performance

There are also critical issues with respect to the PCS and containment performance that must be evaluated as part of the AP600 test program. The AP600 steel containment is the path for ultimate long-term heat removal after an accident. Steam released from a break, from boiling in the IRWST after several hours' operation of the PRHR system, or from actuation of stage 4 of the ADS, flows into the containment atmosphere to the inside wall of the containment. When containment pressure increases sufficiently, discharge valves open on a tank of water located above the top of the containment dome, and water is sprayed on the outer surface of the dome and flows down the sides. Heat from the inside of the containment vaporizes the water, which is then carried off by natural convection of air upward along the outer surface of the containment. The air enters through ports near the top of the shield building and flows downward in a downcomer formed by the shield building and a baffle; it then flows up along the containment in the space between the baffle and the dome, exhausting to the environment at the top of the dome.

The following key issues affect PCS performance:

- condensation heat transfer in the presence of non-condensables in the dome
- condensate flow on the interior surface of the containment
- cooling water distribution, flow, and evaporation on the outer surface of containment
- the effect of wind on air flow into and out of the PCS

21.2.7 Application of Existing Models and Correlations

There are also issues related to the application of existing models and correlations to the operating, transient, and accident conditions that can exist in the AP600. These issues are discussed in the following sections.

21.2.7.1 Departure From Nucleate Boiling

A specific area of interest involves models for the departure from nuclear boiling (DNB). Most DNB correlations are developed for limited ranges of thermal-hydraulic and geometric parameters, and AP600 conditions may extend beyond those ranges. It is, therefore, necessary to develop data for thermal-hydraulic and geometric ranges that encompass predicted AP600 conditions.

21.2.7.2 Shutdown Operations

An additional area of concern relates to shutdown operations in the AP600. The test program focused on events occurring in conjunction with power operations. No testing was performed to deal specifically with the issue of shutdown events, and the staff must assess the capability of the testing performed to address this issue. In the DSER, the staff stated that Westinghouse must demonstrate that the AP600 analysis codes, as validated by its testing program, can be relied upon to accurately represent shutdown conditions in the AP600. This was identified as DSER Open Item 21.2.7.2-1. Westinghouse submitted WCAP-14837, Revision 1, "AP600 Shutdown Evaluation Report," in June 1997. In Section 4.1.2 of WCAP-14837, Revision 1,

Westinghouse discusses the applicability of test data and the AP600 accident evaluation computer codes to shutdown events.

The AP600 phenomena identified in the phenomena identification and ranking table (PIRT) that are most significant for accidents that begin at high power remain almost entirely applicable at shutdown conditions. Items that were ranked "high" for the full-power cases remain so for accidents during shutdown, and none of the phenomena that are not ranked "high" for full-power cases become highly important during shutdown, except for the reliance on the CMTs to mitigate an event when the accumulators are isolated. This is not a code or testing issue, but is instead a system operability issue that is addressed through the technical specifications.

In validating the NOTRUMP code, Westinghouse used experiments which were initiated from a simulated decay power level, and the typical initial fluid pressures and temperatures are similar to the AP600 RCS values which will be present during shutdown operations. Furthermore, the gravity drain phenomena that are important for SBLOCA events remain important during shutdown. As a result, the existing test program has included all of the phenomena that are important to the analysis of accidents that begin during shutdown.

The LOFTRAN-AP and LOFTTR2-AP codes are not needed for AP600 safety analyses during shutdown conditions, because the accidents that they would analyze, such as turbine trip or loss of feedwater, are either (1) not possible, or (2) the system response is bounded by the full-power scenarios.

On the basis of these considerations, the staff finds that Westinghouse adequately addressed the need for testing to justify the use of its analytical tools for accidents that occur during shutdown, per the requirements of 10 CFR 52.47(b)(2)(i)(A). The staff finds this acceptable, and therefore DSER Open Item 21.2.7.2-1 is closed.

21.2.8 Summary

Each of the major areas described above has been addressed by testing in one or more experimental facilities, with the objectives of resolving the key issues and complying with the requirements of 10 CFR 52.47(b)(2)(i)(A). In Section 21.3 of this report, the staff describes each of the major AP600 test programs.

21.3 Overview of Westinghouse Testing Programs

Westinghouse developed a design certification test program utilizing both separate-effects and integral-systems facilities to investigate the behavior of the AP600 passive safety systems and to develop a database for the validation of the computer codes used to perform the transient and accident analyses presented in the SSAR. The test programs can be broadly characterized as programs that are related to reactor systems, programs that are related to containment systems, and component testing.

Each test program is described briefly in this section. The staff's detailed evaluation of each program is given in Section 21.5 of this report.

21.3.1 Core Makeup Tank Test Program

The CMT test program was a separate-effects test program developed to characterize the CMT over the range of thermal-hydraulic (pressure, temperature, flow) conditions that it will experience in the plant. Important phenomena to be studied included thermal stratification in the CMT and the effects of recirculation, draining, and plant depressurization on CMT behavior. Tests were performed at the facility located at Westinghouse's Waltz Mill site, near Madison, Pennsylvania. The test article was approximately 3.3 m (10 ft) in height and 0.49 m (1.6 ft) in diameter. Compared to the actual AP600 component, the test article was one-half of the height and 1/7.77 of the diameter. The reactor vessel was simulated by a steam/water reservoir (SWR). The CMT test article was connected to the SWR by a pipe simulating the cold-leg/CMT PBL, which came directly off the SWR, and by a drain line, simulating the direct vessel injection (DVI) line. The RCS cold leg was not represented in this facility. A steam distributor of a design similar to that proposed for the AP600 CMT was installed in the upper head of the test article at the PBL nozzle.

The CMT test facility was capable of operating up to approximately 17.2 MPa (2500 psia) and 364 °C (688 °F). Data acquisition was accomplished using a personal computer (PC)-based data acquisition system (PC-DAS). The facility was instrumented with:

- thermocouples, including those to obtain concentrated measurements of CMT fluid and wall temperatures
- pressure transducers, including differential pressure transducers to measure CMT level
- flowmeters

In addition, four of the prototypical heated RTDs for level (steam-water interface) detection, of the design originally proposed by Westinghouse for use in the AP600 for automatic depressurization system actuation, were installed in the test article. An objective of the test program was to investigate the response to changes in CMT conditions, including temperature and liquid level.

The CMT test facility design and instrumentation were originally described to the staff in meeting presentations in 1990. The test facility design and instrumentation are described in WCAP-13345, Revision 3 ("AP600 Core Makeup Tank Test Specification") and in WCAP-14132 ("Facility Description Report for AP600 CMT Test Program"). A preliminary test matrix is included in the test specification. The staff began its review of the CMT test program in 1991, and provided numerous comments to Westinghouse regarding the facility design, instrumentation, and the test matrix. Additional comments were made by the Advisory Committee on Reactor Safeguards (ACRS). The facility design evolved considerably from its initial configuration and, in fact, continued to change even after the beginning of testing. This was done to address specific technical issues and to conform to changes in the AP600 design affecting CMT design and operation. Many of these changes to the facility design, instrumentation, and test matrix were made by Westinghouse in response to staff and ACRS suggestions.

Westinghouse documented its scaling analysis of the CMT facility in WCAP-13963, Revision 0 ("Scaling Logic for the Core Makeup Tank Test"). The staff and the ACRS provided comments on this scaling analysis to Westinghouse, and the staff issued RAI 440.52, which indicated in general that WCAP-13963, Revision 0, did not provide sufficient information on the scaling of the CMT test facility, and discussed several specific technical issues. These concerns were identified as DSER Open Item 21.3.1-1. Westinghouse responded to RAI 440.52 and committed to revise WCAP-13963 to address more comprehensively the staff's and ACRS's concerns. The revised scaling report, WCAP-13963, Revision 1, was issued in January 1995. The staff reviewed the response to RAI 440.52 and WCAP-13963, Revision 1, and concluded that the additional information demonstrated the acceptability of the CMT test facility scaling logic; details of the review are discussed in Section 21.5.1 of this report. The staff finds this acceptable, and therefore, DSER Open Item 21.3.1-1 is closed.

The CMT test program began with cold preoperational tests in May 1993. The program proceeded though several series of tests. The "100" series investigated condensation of steam on the CMT walls, with and without the effects of non-condensable gases. The "300" series looked at mixing, condensation behavior, and CMT draining when steam was injected into cold water, such as might occur during a large SBLOCA or a large-break LOCA (LBLOCA) (i.e., no recirculation to heat the CMT water. The "400" test series was similar to the "300" series, but the system was depressurized gradually during the tests to assess the effect of changing pressure on the draining behavior. The final "500" series of tests included a period of recirculation between the SWR and the CMT to establish a desired temperature profile in the CMT. After the recirculation period, the SWR water level was reduced to allow steam to flow to the CMT, and the CMT was depressurized and drained. This series most closely represented conditions in the CMT expected during non-LOCA transients (CMT recirculation) and most SBLOCAs (recirculation, followed by draindown and depressurization). The test program was completed in September 1994.

The staff identified submittal of all remaining CMT test program documentation as DSER Open Item 21.3.1-2. Westinghouse submitted quick-look reports (QLRs) documenting hot and cold preoperational tests and each series of matrix tests. Westinghouse also submitted the CMT test program final data report (WCAP-14217) and test analysis report (WCAP-14215) in November 1994 and December 1994, respectively. All CMT test program documentation has been submitted. Therefore, DSER Open Item 21.3.1-2 is closed.

21.3.2 Automatic Depressurization System Test Program

The ADS test program was a separate-effects test program consisting of two phases, with somewhat different objectives. The first part, Phase A, was an investigation of steam flow through a prototypical AP600 ADS sparger, into a large, water-filled tank representing the IRWST. Parameters of interest are as follows:

- sparger flow and pressure drop
- tank thermal-hydraulic and structural response, including condensation, thermal stratification, and condensation-induced pressure loads on the tank walls

The second part, Phase B1, was a test of the thermal-hydraulic behavior of the ADS piping network that extends from the pressurizer into the IRWST. Tests were performed at the facility,

called "VAPORE," located at the Central Research Establishment of ENEA (Ente per le Nuove technologie, l'Energia e l'Ambiente, the Italian Energy Agency) at Casaccia, outside of Rome, Italy. The Phase B1 tests were identified as Phase B tests in the DSER. Originally, the Phase B1 test program was conceived as an investigation of the "integral" behavior of the ADS valves and their operators, including both thermal-hydraulic and mechanical characteristics of valve performance. After discussions with the staff, however, Westinghouse modified the test objectives to focus only on the thermal-hydraulic behavior of the ADS valves, piping, and sparger. The main parameter of interest was the flow from the pressurizer to the simulated IRWST with various combinations of ADS stages open, including choking at various locations through the valve/piping/sparger network. The major thermal-hydraulic variable for these tests was the quality of the fluid entering the ADS network. Steam-only blowdowns were performed using a discharge line from the top of the steam-water supply tank. This discharge line contained a separator to remove entrained liquid. A discharge line was also provided from the bottom of the supply tank to obtain two-phase mixtures through the ADS network. A control valve in the bottom discharge line from the supply tank was adjusted to allow the flow to be varied from saturated liquid conditions to two-phase flow over a range of qualities. In addition, a series of tests was performed with cold water flowing through the valve/piping network to aid in determining the hydraulic characteristics of the valve/piping/sparger system under single-phase, non-choked conditions.

For Phase A, the actual test article was only the AP600 sparger, installed in the large water quench tank. Saturated steam was supplied to the sparger from a large supply tank.

The test facility was modified extensively for Phase B1. The supply tank, the large water quench tank, and the sparger were retained, and a piping network representing one complete group of ADS valves (stages 1, 2, and 3, with two valves in series per stage) was added. One ADS valve in each stage was represented by an actual valve. The other valve in each stage was represented by a spool piece containing an orifice to simulate the throat area and loss characteristics of various potential valve designs. Exhaust piping led from the ADS piping network to the same quench tank and sparger assembly used in Phase A.

In Phase A, the sparger pedestal above the floor of the tank was not entirely prototypic with respect to the AP600, while in Phase B1, an actual AP600-design pedestal was used. The ADS piping, the valves and simulated valves, and the sparger were full-size components, and could operate up to full AP600 pressure. The facility was instrumented with:

- flowmeters
- pressure transducers
- thermocouples, including rakes in the simulated IRWST to measure stratification

Instrumentation to measure pressure loads and tank response in the IRWST were included. Data acquisition was accomplished using a computer-controlled PC-DAS.

The ADS facility and test program were first described to the staff in 1990. The program then underwent several modifications, partly as a result of ADS design and logic changes, and also because the objectives of the test program changed from those of a combined thermal-hydraulic and valve performance test to those of a thermal-hydraulic test only. The ADS test program is described in WCAP-13342 ("AP600 Automatic Depressurization System Test") and the Phase A

facility configuration is described in WCAP-14149 ("VAPORE Facility Description Report, AP600 Automatic Depressurization System Phase A Test"). The staff identified submittal of the ADS Phase B1 facility description report as DSER Open Item 21.3.2-1. The Phase B1 facility description (WCAP-14303) was submitted in March 1995. Therefore, DSER Open Item 21.3.2-1 is closed.

The staff began its review of the ADS test program in 1991 and provided comments to Westinghouse regarding facility design, instrumentation, and the test matrix. Additional comments were made by the ACRS. As noted previously, the Phase B1 test configuration and test objectives changed substantially after 1990, and these changes have affected the scope of the staff's review. Westinghouse responded to the staff's comments and either justified its design approach or incorporated changes to the facility design, instrumentation, and test matrix to respond to staff (and ACRS) comments.

Because the VAPORE facility incorporated full-scale components, a formal scaling report was not required for this test program. However, Westinghouse committed to provide a "road map" to demonstrate how the information from this test program will be used as a basis for further development and testing of the actual ADS valves. This was identified as DSER Open Item 21.3.2-2. This item required the information to be submitted by Westinghouse, and the staff's review thereof. Westinghouse submitted an initial "road map" in a letter to the NRC responding to RAI 952.96. The staff provided its comments on this document, and Westinghouse submitted a revised "road map" by letter dated April 30, 1997. The staff reviewed the revised "road map" and finds it acceptable in response to the staff's concerns, as detailed in Section 21.5.2 of this report. Therefore, DSER Open Item 21.3.2-2 is closed.

Phase A of the ADS test program began in June 1992, and concluded in December 1992. The Phase A program and selected test data are described in WCAP-13891 ("AP600 Automatic Depressurization System Phase A Test Data Report").

Phase B1 of the ADS test program began in July 1994, and was completed in November 1994. Westinghouse documented the tests initially in QLRs. The final data report (WCAP-14234) was issued in April 1995, and the test analysis report (WCAP-14305) was submitted in June 1995. This report was subsequently revised, and Revision 2 was submitted in February 1998. Details of the staff's evaluation are discussed in Section 21.5.2 of this report.

21.3.3 Passive Residual Heat Removal Heat Exchanger Test Program

The objective of the PRHR separate-effects test program was to generate data on heat transfer to be used in the design and characterization of the PRHR heat exchanger in the AP600. The test facility was located at Westinghouse's Science and Technology Center near Pittsburgh, Pennsylvania. The test article consisted of three vertical (straight) type 304 stainless steel tubes, approximately 5.49 m (18 ft) in length, with an outer diameter of 19.0 mm (0.75 in.) and wall thickness of 1.69 mm (0.0665 in.). The tubes were placed in a water tank approximately 1.2 m (4 ft) in diameter and 9.8 m (32 ft) tall, with a nominal water depth inside the tank of 7.3 m (24 ft) (this depth was varied as a test parameter). The tubes were placed in a straight line near a wall of the tank, with a center-to-center spacing of 38.1 mm (1.5 in.). A baffle was placed near the tubes, almost at the same height as the tank to limit natural convection flow around the tubes.

The tube dimensions and spacing were characteristic of the PRHR system design as of about 1989, when the first series of tests was performed; however, the heat exchanger design changed significantly in 1992, as explained below.

The PRHR facility could operate with primary (tube side) conditions up to full AP600 pressure and temperature; the tank-side conditions were initially ambient, since the tank was not inside a building. However, the tank temperature could be varied by operating the heat exchanger tubes before the test to establish desired initial conditions. The facility was instrumented primarily with thermocouples to measure local temperatures and to determine heat transfer coefficients. Flow to each tube could also be measured.

Testing was conducted in two phases for the PRHR test program (1) Phase 1 in 1989 and (2) Phase 2 in 1990. However, the facility design and basic objectives of the two test series were essentially identical. The main purpose of the second phase was to extend the range of conditions over which PRHR heat transfer was measured to aid in the development of heat transfer correlations.

Westinghouse described the test program in two reports: WCAP-12980 ("AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report") and WCAP-13368 ("Passive RHR Heat Exchanger Test Extension"). These reports contain full descriptions of the test facility, instrumentation, data collection procedures, and test matrices, as well as selected data. Since the heat exchanger tubes were nominally full length, no scaling report was submitted on this test facility.

The staff began its review of the PRHR test program in 1991. After testing was completed, Westinghouse changed the PRHR heat exchanger (PRHRHX) design from straight vertical tubes to vertical C-shaped tubes. The initial focus of the review was on the applicability of the three-tube test data to a full-size heat exchanger. This issue increased in importance when the heat exchanger design changed in 1992. The staff issued RAIs requesting detailed justification on the applicability of the straight-tube PRHRHX test data to the new C-tube configuration. This was identified as DSER Open Item 21.3.3-1. Although Westinghouse responded to the staff's RAIs, further review by the staff raised additional technical concerns. Westinghouse committed to revise the PRHR test report to address data applicability to resolve this open item.

Westinghouse provided substantial additional information to address the issues raised in DSER Open Item 21.3.3-1. The PRHR test report was completely revised and reissued as WCAP-12980, Revision 3, in April 1997. Westinghouse also provided additional justification regarding applicability of correlations derived from the straight-tube PRHR test data to the C-tube AP600 PRHRHX design in responses to RAI 952.94 and in a letter response to DSER Open Item 21.3.3-1. After reviewing Westinghouse's additional information, the staff concluded that additional confidence was needed in the ability of Westinghouse's PRHR heat transfer model to calculate actual C-tube performance data. Consequently, the staff provided selected data from the NRC's confirmatory test program in the Rig of Safety Assessment/ Large Scale Test Facility (ROSA/LSTF) loop, which employs a simulated C-tube PRHRHX of prototypic bundle dimensions (with approximately 1/30 the number of heat exchanger tubes) immersed in a tank of water simulating the IRWST. The data provided to Westinghouse included the PRHRHX inlet flow and inlet temperature, and the IRWST temperature profile, for two ROSA tests at several times in each test. The staff then asked Westinghouse to use this information to perform

a "blind" calculation of the PRHRHX outlet temperature and tube temperatures at several locations along the length of the HX. The staff reviewed Westinghouse's calculations and found that Westinghouse's correlation predicted tube and outlet temperatures within a few degrees of the data at all times and locations. The staff finds this to be acceptable and concludes that the PRHRHX model does an adequate job of predicting PRHRHX tube and outlet temperatures. The staff therefore concludes that Westinghouse has demonstrated the adequacy of the straight-tube-based correlations for analysis of the C-tube PRHRHX, and therefore, DSER Open Item 21.3.3-1 is closed. Further discussion of the staff's evaluation of the PRHR test program, including details of the ROSA/LSTF data evaluation, is in Section 21.5.3 of this report.

21.3.4 Departure From Nucleate Boiling Test Program

The DNB separate-effects test program differs from those described above in that it was not directly related to the performance of a passive safety system. Rather, the objective of this program was to acquire data on DNB in fuel rod bundles over the flow range relevant to the AP600 during transients. Data from these tests were used to extend the range of the correlations used to predict DNB in Westinghouse's reactor system computer codes. The DNB test program was carried out during 1993 and 1994 in the Heat Transfer Research Facility (HTRF) of Columbia University, in New York City. Similar tests were performed at this facility for other reactor and fuel designs. Steady-state DNB tests were performed on three fuel bundles, representing Westinghouse's reference fuel design for the AP600.

Westinghouse conducted these critical heat flux (CHF) tests at the Columbia University HTRF. A total of 372 data points were collected from four configurations of 5 X 5 rodded test bundles. The data collected pertained to inlet pressure, inlet mass velocity, inlet temperature, average bundle heat flux, and identification of the thermocouples for reference.

The four configurations tested differed with respect to cell type, use of intermediate flow mixers (IFMs), and grid rotation. Each configuration consisted of electrically heated rods arranged in a 5 X 5 rectangular array. Rod spacing was maintained by Zircaloy mixing vane grids, IFMs, and/or simple support grids. Each array was encased in a ceramic-lined shroud box that was positioned within the pressure boundary of the test section housing. In operation, the flow entered the test section housing near the bottom and then traveled vertically upward. The rod inner diameters were tapered to produce a nonuniform (cosine) axial flux distribution and a representative radial power distribution.

In the DSER, the staff stated that it had not completed its review of the DNB test program and identified this as DSER Open Item 21.3.4-1. The staff has now reviewed the DNB test program documentation for the AP600 reference case VANTAGE 5-H fuel assemblies as discussed is Sections 4.4 and 21.5.4 of this report. Therefore, DSER Open Item 21.3.4-1 is closed.

21.3.5 Oregon State University Advanced Plant Experiment Test Program

The Oregon State University (OSU) Advanced Plant Experiment (APEX) test program was one of two major integral test programs conducted by Westinghouse for AP600 design certification. The objective of the test program was to obtain integral-systems data for the validation of computer codes used for AP600 safety analyses. Particular emphasis was placed on low-pressure and long-term cooling behavior in design-basis SBLOCAs. Tests were performed at the facility located on the OSU campus in Corvallis, Oregon.

APEX was a low-pressure, ¼-height representation of the AP600, including the RCS and related components and all safety systems in direct communication with the primary system. Although containment cooling systems were not represented, the containment sump was simulated by two tanks. The "primary" sump tank simulated containment sump volumes from which fluid could be recirculated back to the RCS during the long-term cooling phase of a LOCA, and was connected to direct vessel injection (DVI) lines. The "secondary" sump tank simulated volumes in the AP600 containment from which fluid could not be recirculated back to the RCS; fluid that entered the secondary sump was unrecoverable for long-term cooling, as would be the case in the AP600 plant. The volume scale of the facility was 1/192. The layout of the facility was similar to the AP600, with two cold legs and one hot leg per loop and vertically mounted reactor coolant pumps with no loop seals. In addition to the safety injection systems (CMTs, accumulators, and IRWST) and the four-stage ADS, the PRHR heat exchanger was simulated by a scaled heat exchanger bundle in the IRWST. The facility was instrumented with over 700 thermocouples, flowmeters, pressure transducers, and void detectors. Data were collected using three linked PCS that could record data and display real-time plots of selected channels.

Westinghouse documented the facility design and test program specifications in WCAP-14124 ("AP600 Low Pressure Integral Systems Test at Oregon State University - Facility Description Report") and WCAP-13234, Revision 1 ("Long Term Cooling Test"), respectively. The staff identified the submittal of the final scaling report as DSER Open Item 21.3.5-1. An extensive scaling analysis was performed for the APEX facility and documented in WCAP-14270 ("Scaling Analysis for the OSU Integral System and Long Term Cooling Test Facility"), which was submitted in January 1995. The final report resolved a number of questions and comments on the draft scaling report; however, the staff issued several additional RAIs on the final report. Westinghouse provided responses to all of the outstanding RAIs related to the scaling report; these have been included in WCAP-14727, Revision 2, "AP600 Scaling and PIRT Closure Report," Appendix A. The staff reviewed these responses and finds them to be acceptable, as detailed in Section 21.5.5 of this report. Therefore, DSER Open Item 21.3.5-1 is closed.

Cold preoperational testing began in January 1994; the matrix tests began in May 1994 and ended in August 1994. An additional series of tests beyond those in the original test matrix was performed in August and September 1994. Most of the matrix tests run in the APEX facility simulated design-basis accidents for the AP600, primarily SBLOCAs of various sizes and at different locations in the RCS. All of the tests included an extended period after the loop was fully depressurized to investigate integral system thermal-hydraulic behavior during injection from the IRWST, transition from IRWST to sump injection, and long-term recirculatory cooling from the simulated sump. The two major variables affecting system behavior were break size and location. Break size varied from (scaled) 12.7-mm (0.5-in.) to approximately 203 mm (8 in.). Break locations tested included the cold leg, hot leg, CMT pressure balance line, and DVI line. Other effects studied included interactions with non-safety-related systems and the effect of elevated containment pressure, which was simulated by increasing the pressure of the IRWST and sump tanks.

Staff review of the OSU APEX program began in 1991 and evolved as the test facility design and experimental program grew in complexity and scope. The staff provided its evaluation of the test facility design, instrumentation, and test matrix to Westinghouse and made several comments about recommended changes in these aspects of the test program. Westinghouse

was responsive in incorporating staff recommendations into the facility and the test matrix. Additionally, Westinghouse made changes in the APEX facility to incorporate changes in the AP600 design.

The staff identified the submittal of all OSU APEX test documentation as DSER Open Item 21.3.5-2. All QLRs for the cold preoperational, hot preoperational, hot functional, and matrix tests were provided to the staff. Westinghouse submitted the OSU test program final data report (WCAP-14252) in May 1995 and the complete test analysis report (WCAP-14292, Revision 1) for the program in September 1995. All OSU APEX test documentation has been submitted. The staff has completed its review of the data and analysis for the test program; Westinghouse has also responded to the staff's RAIs on these reports. The staff considers submittal of the Final Data Report and the Test Analysis Report as adequate to close DSER Open Item 21.3.5-2. The staff's review of the OSU/APEX final data report (FDR), test analysis report (TAR), and selected RAIs is discussed further in Section 21.5.5.

21.3.6 SPES-2 High-Pressure, Full-Height Integral-Systems Test Program

The SPES-2 test program was the second integral-systems test program performed for design certification of the AP600. The objective of this test program was similar to that of the OSU APEX program: acquisition of integral-systems data for the validation of computer codes used to perform AP600 safety analyses. Unlike the APEX facility, however, SPES-2 could operate at pressures and temperatures up to prototypic AP600 values and was approximately full vertical scale. Because of this unique capability, tests in SPES-2 focused primarily on integral system behavior in the period from accident initiation (at prototypic pressure and temperature and scaled full power) to the establishment of stable injection from the IRWST. The test matrix included a range of SBLOCAs from scaled 25-mm (1-in.) cold leg breaks to double-ended guillotine (DEG) ruptures of a DVI line and of a CMT balance line. In addition, non-LOCA transients were simulated in SPES-2, including single steam generator tube ruptures (SGTRs) and a main steamline break. Other parameters tested included interactions with non-safety-related systems (cold leg SBLOCA and SGTR) and inadvertent actuation of the ADS during an SGTR event. Tests were performed at the facility located at the Societa' Informazioni Esperienze Termoidrauliche (SIET) laboratories in Piacenza, Italy, between February and November 1994.

SPES-2 was a full-height representation of the AP600. The volume scale was approximately 1/395. However, SPES was not designed from the beginning as an AP600 test facility. Rather, SPES-2 was a modification of the existing SPES-1 facility, which represented a 1/427-volume-scale of a Westinghouse three-loop pressurized water reactor (PWR). As a result, some distortions and atypicalities existed in SPES-2 compared to the AP600 design. The most significant of these were as follows:

- SPES-2 had only one pump per loop, rather than the two pumps per loop of the AP600, so that outlet flow from (or through) the pump had to be split between the two cold legs.

- SPES-2 had an external piped downcomer, rather than an annular downcomer. The design was modified, however, so that there was an annular section at the top of the simulated reactor vessel, which then fed into the piped downcomer below the elevation of the DVI lines.

- SPES-2 had a much larger surface-area-to-volume ratio than the AP600. This caused distortions in two ways. In the initial stages of a transient, high heat losses occurred. This was compensated by increasing the rod bundle power during the period from accident initiation to beginning of ADS blowdown. In the later stages of a transient, the effect was reversed, and excessive heat input to the system occurred from the structure. To relieve the excess steam resulting from this effect, the vent area of the simulated ADS-4 valves in SPES-2 was significantly larger than its nominal scaled (1/395) value, to achieve a depressurization rate calculated to be approximately the same as in the AP600.

All AP600 safety systems were represented, including CMTs, accumulators, IRWST injection, four-stage ADS, and the PRHR system, in which the PRHR heat exchanger was simulated by three C-tubes in the IRWST tank. (For most tests, only one of the three C-tubes was used.) Sump recirculation was not simulated in the SPES-2 loop. The facility contained over 300 instruments, including thermocouples, flowmeters, pressure transducers, and void instrumentation. Data were collected using a computer-controlled PC-DAS.

Westinghouse, with the aid of SIET, has documented the SPES-2 test program in several reports, including WCAP-13277 ("Scaling, Design, and Verification of SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant"); WCAP-13277, Revision 1 ("Scaling, Design, and Verification of SPES-2, the Italian Experimental [sic] of the AP600; Scaling Update"); WCAP-14053 ("AP600 FHFP Integral Systems Test Specification"); and WCAP-14073 ("SPES-2 Facility Description"). Several other documents that include instrumentation descriptions and design drawings were also provided by Westinghouse to assist in the staff's review of the SPES-2 program.

Staff review of the SPES-2 test program is somewhat different from its review of other AP600 design certification tests. The SPES-2 program was not part of the original AP600 test program. When the staff began the review of the AP600 test programs in 1991, it came to the conclusion that the proposed integral-systems testing in the OSU test facility was not sufficient for the AP600 design. The staff proposed, in SECY-92-030 ("Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992), that Westinghouse be required to perform high-pressure, full-height integral-systems tests as part of the AP600 design certification test program. In response to the staff's recommendation, Westinghouse proposed to modify the SPES-1 facility at SIET. The staff reviewed the development of the modified facility design and the test program from their initial stages and provided extensive comments on all aspects of the facility and testing.

Westinghouse was generally responsive to the staff's recommendations as the SPES-2 program developed. In addition, Westinghouse made changes in the facility design to incorporate changes in the AP600 plant design; the latest of these changes came after matrix testing had begun and required the rerunning of one matrix test. The staff also reviewed the scaling analysis of the SPES-2 facility given in the above-referenced documents.

Cold preoperational testing in the SPES-2 facility began in May 1993; matrix testing began in February 1994 and was completed in November 1994. The staff identified the submittal of the remaining documentation, including the final test report, for the SPES-2 program as DSER Open

Item 21.3.6-1. QLRs were submitted on the cold and hot preoperational test programs and on all of the matrix tests. Westinghouse submitted the final data report (WCAP-14309, Revision 1) for the SPES-2 program in July 1995 and the test analysis report (WCAP-14254) was submitted in May 1995. Pre-test predictions of the SPES-2 tests were also performed by Ansaldo for Westinghouse. However, these predictions were performed with the RELAP5/MOD3 computer code, rather than with any of the codes that Westinghouse qualified for design certification analyses. Validation of the Westinghouse design certification codes against SPES-2 data are found in Sections 21.6.1 and 21.6.2 of this report. All test program documentation for the SPES-2 program has been submitted. Therefore, DSER Open Item 21.3.6-1 is closed.

21.3.7 Wind Tunnel Test Program

One of the principal design objectives for the PCS is that external wind conditions not resist the buoyant air flow in the annular region between the containment vessel and the baffle wall. Counterflow in this region could diminish the natural convective cooling which constitutes a portion of the total heat removal from the post-accident containment atmosphere. The containment can be characterized as either wind-neutral, wind-positive, or wind-negative depending on whether external winds have no effect, assist, or hinder, respectively, buoyant flow in the annulus. To better understand the effects of adverse weather conditions, severe terrain, adjacent structures, and building design variations on the air flow within the annulus, Westinghouse conducted a series of wind tunnel tests on scale models of the AP600 containment.

The wind tunnel tests consisted of four phases, 1, 2, 4A, and 4B (there were no Phase 3 tests), each with the following objectives:

- Phase 1 - to determine the effect of external winds and external structures on the flow around the containment and in the annulus, and determine any modifications to the design necessary for better wind neutrality

- Phase 2 - to determine the flow induced loads on the baffle wall separating the shield building and the containment building, and investigate the effects of a cooling tower and various chimney designs on flow within the annulus

- Phase 4A - to study the sensitivity of the tests to the wind tunnel Reynold's number, further examine the effects of a hyperbolic cooling tower on flow in the annulus, and examine the effects of a tornado velocity profile on flow in the annulus.

- Phase 4B - to examine the effects of severe terrain on flow in the annulus and on baffle wall loading

As of March 1, 1996, final data for all test phases had been submitted via WCAP-13294, "Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor;" WCAP-13323, "Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor;" WCAP-14068, "Phase I.A. Wind

Tunnel Testing for the Westinghouse AP600 Reactor;" and WCAP-14091, "Phase I. Wind Tunnel Testing for the Westinghouse AP600 Reactor." An analysis of the test data was submitted in PCS-TAC-059, "Analysis of AP600 Wind Tunnel Testing for PCS Heat Removal."

21.3.7.1 Test Models - Phase 1 and 2

For the Phase 1 tests, 1:96.67 scale models of the containment shield building and surrounding structures, including the turbine building and a hyperbolic cooling tower, were used. The complete model assembly was located on a turntable which allowed different wind directions to be simulated. The height of the turbine building was adjustable.

The Phase 2 tests used the same models as those in Phase 1, with the exception of the following two modifications:

(1) the flow path between the shield building and containment vessel was modeled by including the baffle wall

(2) a second chimney design was investigated (in addition to the Phase 1 design)

For both tests, the flow in the wind tunnel was representative of that of a turbulent boundary layer and corresponded to ANSI A58.1 exposure category C, as documented in WCAP-13323. This exposure category is typical of open terrain with scattered obstructions with heights generally less than 9.14 m (30 ft), and includes terrains such as flat regions, open country regions, and grasslands (American National Standard A58.1-1982, pp. 12-16)

21.3.7.2 Test Matrix - Phase 1 and 2

Phase 1 tests were conducted to determine the optimal inlet vent configuration (for a wind-positive or wind-neutral design) and to investigate the effects of the turbine building, cooling tower, and chimney height on flow in the annulus. The effects of a chimney cap and de-aerator were also examined. To characterize the flow, the pressure was measured at selected circumferential locations at the inlet to the annulus and at the outlet of the chimney, and a pressure difference calculated. This pressure difference, which represents the driving force for wind-induced flow, was expressed as a dimensionless pressure coefficient in the test reports.

In the Phase 2 tests, baffle wall loads were determined by measuring the differential pressure across the baffle. As in the Phase 1 tests, the inlet-minus-chimney pressure difference was determined. Flow visualization studies were also performed to determine the flow characteristics in the annulus.

The Phase 1 test report presented data in the form of pressure coefficients for various inlet vent configurations, adjacent building configurations, and wind directions. Data presented from the Phase 2 tests consisted of plots of pressure coefficients versus wind direction for various chimney/top configurations. Note that for all test phases, wind angles of attack varied from 0° to 360°. A summary of the test matrix is provided in Table 21.3-1.

Table 21.3-1 Wind Tunnel Test Phases 1 and 2 Matrix

Test Configuration	Phase 1	Phase 2
Turbine building elevation varied	Yes	No
Effect of cooling tower tested	Yes	Yes
Chimney height varied	Yes	No
Chimney cap and deaerator tested	Yes	No
Different chimney designs tested	No	Yes
Inlet-chimney ΔP measured	Yes	Yes
Baffle ΔP measured	No	Yes

21.3.7.3 Test Models - Phase 4A and 4B

The Phase 4A tests comprised three subtests to address tornado wind loading, the effect of a cooling tower, and the sensitivity of Phase 1 and 2 test results to the wind tunnel Reynold's number. Various combinations of models and wind tunnels were used for each subtest.

Tornado Wind Loadings - Phase 4A

The model used in the Phase 4A tests to study tornado wind loadings was the same 1:96.67 model used in the Phase 1 and 2 tests. The tests were performed at the University of Western Ontario (UWO) boundary layer wind tunnel. Modifications to the model used in the Phase 1 tests involved the addition of additional circumferential rings of pressure taps at the following locations:

- the exterior of the main building at 2/3 the height of the inlets
- the main building exterior just below the inlets
- directly inside the inlet manifold
- on the exterior half-way up the chimney
- at the top of the containment annulus

Effects of the Cooling Tower - Phase 4A

The same 1:96.67 model was used to investigate the effects of the cooling tower. However, these tests were conducted in the National Research Council of Canada's 9.14 m x 9.14 m (30 ft x 30 ft) wind tunnel in Ottawa, Canada.

Effects of High Reynold's Number - Phase 4A

A 1:30 scale model was chosen with the aim of achieving the highest Reynold's number possible to study the effects of high Reynold's number on the results. The model was not capable of being rotated and did not include internal flow paths. The tests were performed at the National Research Council of Canada's 9.14 m x 9.14 m (30 ft x 30 ft) wind tunnel in Ottawa, Canada.

Effect of Severe Terrain - Phase 4B

The Phase 4B tests were performed to investigate the effect of severe terrain on baffle wall loads and used a 1:800 scale model of the containment and surrounding buildings. This scaling was chosen to allow a large area of terrain to be included in the proximity of the plant. The complete assembly was placed on a turntable and could be rotated to investigate the effect of different wind azimuths.

For both of the wind tunnels, a scaled boundary layer wind profile corresponding to ANSI A 58.1 exposure category C, as in the Phase 1 and 2 tests, was used.

21.3.7.4 Test Matrix - Phase 4A

The tests conducted at the UWO facility represent the most complete set of data of all the Phase 4A tests. Data taken from Phase 4A tests conducted at the National Research Council of Canada facility were obtained primarily to address concerns with the UWO data regarding Reynold's number scaling, and to determine if any adjustments to prior tests in Phase 1 and 2 were necessary.

Data collected in all of the 4A tests included air inlet and outlet pressure measurements and the pressure difference across the baffle wall. For the Phase 4A 1:96.67 scale tests at UWO, data were taken for a full range of wind speeds and azimuths, including those representative of tornado conditions to help determine the design basis loading of the baffle wall. The Phase 4A 1:30 scale tests were conducted over a range of wind speeds, up to the maximum attainable in the wind tunnel. Table 21.3-2 summarizes the Phase 4A test matrix.

Table 21.3-2 Wind Tunnel Test Phase 4A matrix

Configuration/ Effects Modeled	**Phase 4A 1:96.67, National Research Council**	**Phase 4A 1:96.67, Univ. of Western Ontario**	**Phase 4A 1:30, National Research Council**
Surroundings	Yes	Yes	No
Cooling tower	Yes	No	No
Chimney	Opened/closed/ roughened/smooth	Closed/smooth/ roughened	Closed/smooth/ roughened
No. of Azimuths	2	full 360°	1
No. of speeds	3	1	6
Tornado	No	Yes	No

21.3.7.5 Test Matrix - Phase 4B

The Phase 4B tests measured pressures around the throat and in the wake of the cooling tower to examine the effects of severe terrain on flow in the annulus and around the containment

building. Wind speeds in the wake of the cooling tower were also measured. The wake characteristics used to determine the placement of the cooling tower for the 4B tests were taken during the 4A tests.

Tests were performed using the following configurations:

- the base case, which consisted of a complete AP600 plant model and one cooling tower
- the base case with two cooling towers
- the base case with an escarpment nearby
- the base case with an escarpment and mountain backdrop
- a river valley with the mountain backdrop, another mountain on the other side of the shield building, and with the escarpment filled in
- a river valley with two cooling towers

21.3.7.6 Scaling Validation Tests

In WCAP-12394, Westinghouse presented the results of validation tests run to show that the aerodynamics in the range from the model to full scale was not significantly affected by the Reynold's number (Re). The Reynold's number affects where flow separation will occur, thereby affecting the pressure on the structure. These tests were considered necessary because strict dynamic similarity between the model and the AP600 necessitated wind tunnel speeds where compressible flow phenomena become important, adding considerable complexity to the problem of modeling the flow around the model. To remove the necessity of matching the Reynold's number, the flow separation points in the full and model scales were matched by roughening the model surface. The experimenters' goal was to show that the pressure variation versus wind speed was considerably less for a roughened model surface than for a "smooth" model surface, and therefore, there were few Reynold's number induced effects in the roughened model. The amount of surface roughening was found by determining a relative roughness for the model from knowledge of the full-scale Reynold's number.

Results from the tests of the smooth and roughened models showed that the pressure varied 10-percent to 30-percent less with wind speed for the roughened model, with fluctuations in pressure of less than 12-percent. This was interpreted in the test report as an indication that effects of the Reynold's number were significant for a smooth model and consequently, roughening was necessary to minimize these effects. The experimenters further concluded that only chimney roughening was necessary, and any residual effects as a result of the Reynold's number would amount to a maximum of 20 percent of the pressure fluctuations.

In WCAP-13323, Westinghouse outlined its approach for scaling the wind tunnel test data to the full scale AP600. In it, dynamic pressures scaled to the height of the AP600 inlet vents were

used to determine design loads for a number of design wind speed cases. Sample calculations were provided. The various wind cases covered were as follows:

(1) fastest mile wind speed in ANSI Exposure C
(2) fastest mile wind speed in ANSI Exposure D (less gusty than exposure C)
(3) probable maximum hurricane speed
(4) tornado speeds

Flow losses in the model annulus were modeled by matching, at various points in the flowpath, loss coefficients in the model with losses representative of the AP600. Since no full scale test data was available to determine the AP600 loss coefficients, coefficients determined from tests conducted with a 1/6-scale, 14° sector model were used.

21.3.8 Large-Scale Passive Containment Cooling System (PCS) Test Program

The purpose of the PCS Large-Scale Test (LST) program was to investigate anticipated thermal-hydraulic phenomena in a large-scale facility. As stated in WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," Revision 1, dated April 1997:

> "The purpose of the passive containment cooling system (PCS) heat transfer test was to examine anticipated thermal-hydraulic phenomena on a large scale: the interior natural convection and steam condensation the exterior water film evaporation, air cooling heat removal, and water film behavior. This experiment is designed to induce similar containment dome heat transfer processes and circulation/stratification patterns inside containment as in the AP600; however it is not meant to simulate specific AP600 accident scenarios. The large scale test data is used to validate the WGOTHIC computer code, which will be used to analyze the AP600 containment."

21.3.8.1 Test Objectives

The LST tests were conducted in three phases. The objectives of each phase were as follows:

- Phase 1 - These were baseline tests performed at constant pressure conditions to investigate the effect of different levels of water coverage, various external air flow rates, and the presence of internal structures on containment heat removal.

- Phase 2 - The Phase 2 tests were conducted to validate heat and mass transfer correlations over a range of prototypical internal conditions. Transient heat transfer, the distribution of non-condensables, and the effect of non-condensables on heat transfer were examined.

- Phase 3 - The Phase 3 tests examined the effect of break location and non-condensable concentration on mixing in the vessel and on overall heat removal. These tests were intended to aid in the overall understanding of the containment cooling phenomena.

As of July 31, 1994, all planned phases of the LST had been completed and test data reports had been submitted by Westinghouse. The base line tests were documented in WCAP-13566, "AP600 1/8th Large-Scale Passive Containment Cooling System Heat Transfer Baseline Data

Report," dated October 1992. The Phase 2 and Phase 3 tests were documented in WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and 3," dated July 1994. These reports contained primarily test data, and did not provide any evaluation or interpretation of the test results. Submittal of this outstanding LST documentation was identified as DSER Open Item 21.3.8.1-1. Westinghouse, in letter NTD-NRC-95-4463, dated May 15, 1995, submitted "AP600 Testing Program Report: Large-Scale Test Data Evaluation (PCS-T2R-050)." This closed DSER Open Item 21.3.8.1-1.

Westinghouse's evaluation of the large-scale PCS test data yielded the following information and conclusions:

- Evaporation was the primary mode of heat removal from the outside of the vessel (approximately 75 percent of the total), followed by sensible heating of the subcooled liquid film (approximately 17 percent of the total). The remainder of the heat was transferred to the environment by convection and radiation.

- The heat removal rate was proportional to the film area in the quadrant-coverage cases, but had a weak dependence on the coverage area in striped-coverage cases. For the same film coverage area, striped coverage provided better heat removal than quadrant coverage.

- The heat removal rate appears to be more strongly dependent on ambient air temperature than on initial liquid film temperature.

- The heat removal rate has a relatively weak dependence on annulus air velocity, which indicates that, for the conditions tested, the resistance to heat transfer on the inside of the vessel is greater than on the outside.

- For all of the wetted large-scale tests (except the horizontal, high-velocity steam jet injection case), the highest heat fluxes occurred near the top of the dome at the elevation where the external film was applied. Although the dome represents about 30 percent of the heat transfer surface area, approximately 40 percent of the total heat removal occurred on the dome and 60 percent on the cylindrical sidewalls.

- Injection of low-velocity steam resulted in relatively good mixing above, but, stratification below [the injection location], causing air to be concentrated below the operating deck. The heat removal flux (or rate per unit area) increased as the axial steam concentration gradient was increased (by raising the injection location).

- Injection of high-velocity steam resulted in a well-mixed vessel (both above and below the operating deck).

- Injection of a light, non-condensable gas did not degrade the condensation heat transfer or affect the overall heat removal. The gas did not stratify (collect at the top of the vessel), but was well mixed above the injection location and eventually, well mixed throughout the entire vessel.

21.3.8.2 Test Facility

The LST facility was located at the Westinghouse Science and Technology Center in Churchill, Pennsylvania. The vessel used in all test phases was a 1/8-linear-scale version of the actual AP600 containment vessel constructed of steel, with a height of approximately 6.09 m (20 ft), a diameter of 4.57 m (15 ft), and a shell wall thickness of 2.2 cm (0.875 in.).

A transparent plexiglass shell surrounded the containment vessel to form a 7.6 cm (3 in.) annular cooling air flow path. Air enters the annulus at an elevation near the simulated operating deck level. An axial fan was mounted at the exit of the annulus to establish prototypic air flow velocity in the riser. For the Phase 2 and 3 tests, the axial fan speed was set to a predetermined value to establish an air flow rate in the LST riser section typical of that expected in the prototypical AP600. The downcomer portion of the AP600 PCS was not modeled in the LST. The lack of a downcomer region was a concern to the staff, and is further addressed in Section 21.5.8 of this report.

Water was applied to the external surface of the containment vessel by a water distribution system consisting of four independently controlled sectors of J-tubes. The system delivered water to the containment dome at two radial locations. The flow rate was adjustable to allow investigation of water film striping effects. The delivery system geometry used in the LST differed from the delivery system geometry used in the prototypical AP600. In addition, the AP600 delivery system is gravity feed while the LST water delivery system was tied to the municipal water system. The LST water delivery system tended to provide a fluctuating flow rate for most tests. A predetermined valve position was set at the start of a test, based on unheated surface tests and the target water coverage condition specified in the test matrix. Of the 13 priority tests (analyzed with WGOTHIC in WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995) only 6 (including 2 dry tests) achieved the target water coverage condition of 75 percent. In all but one of the remaining tests, the actual coverage was higher. In one test the coverage was well below the target condition.

Superheated steam was supplied at a controlled pressure to the steam generator compartment below the operating deck of the test vessel's interior. A flow distributor was used to provide low velocity steam at a height scaled to that of the actual AP600 operating deck. In some tests, the steam flow was too low to be measured with the existing instrumentation and post-test analysis of the condensation collection data were used to determine the steam boundary condition for use in WGOTHIC analyses of these tests. The blind test (LST test 220.1) was one such case.

Gutters mounted around the inner circumference of the vessel collected condensed steam from the inner surface of the test vessel dome and sidewall. The condensate drained into a handling system that measured the mass of the liquid collected. In addition, there were four other condensate collection points. The five collection points are as follows:

(1) inside surface sidewall condensate discharge system
(2) polar crane girder condensate discharge system
(3) "rainfall" condensate discharge system
(4) inside vessel condensate discharge system (below deck)
(5) steam generator compartment condensate discharge system

Certain LST Phase 1 (baseline) tests included internal partitions to model open, closed, and steam generator volumes below the operating deck.

Vertical communication existed between these open volumes and the area above the operating deck, while the closed areas consisted of dead-end spaces with one entrance and no exit. The off-center steam generator compartment, into which steam was injected, communicated vertically with the test vessel volume above the operating deck. However, unlike the prototypical AP600 configuration, this compartment did not communicate with other compartments below the operating deck. Therefore, it is difficult to directly assess global natural circulation and below deck behavior in the prototypical AP600 on the basis of the LST tests. This distortion was addressed by Westinghouse in WCAP-14845 and is further discussed in Section 21.6.5.5 of this report.

To study the effect of hydrogen generation and distribution in the Phase 2 and 3 tests, helium was injected into the vessel through a port in the steam-supply line near the bottom of the vessel. Instrumentation was supplied to monitor and measure the parameters of interest. Table 21.3-3 provides an instrumentation summary.

Table 21.3-3 Large-Scale Test (LST) Facility Instrumentation

Parameter Measured	Instrument
Steam flow	Vortex shedding flow meter, Variable orifice flow meter, Rate of condensate collection.
Steam inlet pressure	Pressure transducer
System pressure	Pressure transducer
Wind speed/direction	Anemometer/weather vane
Cooling water flow (on and off shell)	Magnetic flow meter
Annulus differential pressure	Pressure transducer
Internal velocity	Pacer and Höntzsch anemometers
Internal fluid temperatures	Thermocouple rake
Vessel wall temperatures	Thermocouples
Gas sampling	Sample tube and sample bomb
Annulus wall temperatures	Thermocouples
Annulus air flow and temperature (inlet/outlet)	Anemometers and radiation shielded thermocouples

21.3.8.3 Modeling of Internal Heat Sinks

Heat sinks in the AP600 containment consist of equipment and structural materials, and can be divided into short-term sinks, with relatively small time constants, and long-term sinks, with a

relatively slow response to thermal transients and larger time constants. The modeling of internal heat sinks in the Phase 1 (baseline) tests consisted of the steel superstructure which supports the subcompartment partitions and operating deck grating. The Phase 2 and 3 tests included these subcompartment partitions and also used 0.975 cm (0.375 in) thick aluminum plates installed in the open and dead-end compartments to more realistically simulate the short-term heat absorption of the AP600 equipment and structures. The modeling of long-term heat sinks in the LST was accomplished by removing portions of the insulation surrounding the open and dead-end volume compartments near the button of the vessel. This was done to study the effects of the heat sinks on the distribution of non-condensables within the vessel.

21.3.8.4 Test Matrix

The tests and target conditions covered are summarized in Table 21.3-4. The 16 Phase 1 (baseline) steady-state tests were performed at three constant pressures of 170, 308, and 377 kPa (10, 30, and 40 psig). The effects of water coverage, external air flow, and internal structures on heat removal were investigated.

Table 21.3-4 Large-Scale Test (LST) Tests and Target Conditions

Test Configuration	Phase 1	Phase 2	Phase 3
Number of tests	16	24	10
Internals modeled	Partially	yes	yes
Steam pressure varied	yes	yes	yes
Steam flow rate varied	no	yes (includes modeling of transients)	yes (includes modeling of transients)
Annulus air flow varied	yes	no	no
Helium added	no	yes	no
Water coverage varied	yes (75% to full)	yes (50% to full)	no (75%)
Blowdown modeled	no	yes	yes (steam discharge point varied)
Effect of non-condensables	no	yes	yes (amount of air varied)

The 24 Phase 2 tests included both steady state and transient investigations, and were intended to span a range of operating parameters (non-condensables concentration, steam flow, and steam pressure) sufficient to validate the ability of WGOTHIC to predict the AP600 containment temperature and pressure response to a design-basis accident (DBA). External water flow rates

were intended to represent those expected on the AP600. The air flow in the annulus was not varied.

The 10 Phase 3 tests studied the effect of rapid pressurization, steam discharge location, and initial vessel air pressure on condensation mass transfer. The effect of non-condensables was studied by conducting tests at pressurized and evacuated initial conditions.

21.3.8.5 Scaling

Westinghouse performed a number of experiments to obtain data for qualifying the WGOTHIC computer program for performing AP600 containment design calculations. The most significant test for the WGOTHIC computer program validation was the LST, which was the only integral systems test of the passive containment cooling system. This section presents the background of the development of the scaling studies performed by Westinghouse. The final scaling analysis and the staff's evaluation is presented in Section 21.6.5.5 of this report.

There was a concern regarding whether the LST tests included conditions sufficient to simulate the phenomena identified in the PIRT and whether they covered a range of parameters representative of AP600 design-basis events. The tests and facility also had a large number of non-prototypical features which could distort the data and affect scale-up of the tests to prototypical size:

- PCS flow established before start of transient (except test 219 which had low heat flux)
- Because of the variability in water pressure to the sprays hooked up to the facility's municipal water supply, the PCS flow rate fluctuated and required additional consideration in evaluating and understanding some test results
- Different water distribution system on shell dome (J-tubes in the LST versus water distribution bucket and weirs in the AP600)
- Relatively high heat removal via sensible heat addition (raising subcooled liquid to saturation as opposed to evaporation)
- Lack of internal heat sinks and non-prototypical internal flow paths (steam generator subcompartment isolation from remainder of below deck regions)
- The height of the test vessel is 6.1 m (20 ft), which was shorter than the height dictated by the 1/8 linear scale used for the LST facility
- The distribution of free volume below the operating deck in the test vessel was different than that in the prototypical AP600:

AP600	LST
19-percent by open volumes	21-percent by open volumes
73-percent by dead-end volumes	70-percent by dead-end volumes
8-percent by steam generator subcompartment	9-percent by steam generator subcompartment

- Annulus riser too narrow; lack of a downcomer; forced air flow
- Vessel shell too thick

In the "AP600 Passive Containment Cooling System Preliminary Scaling Report," letter NTD-NRC-94-4264, dated July 1994 Westinghouse submitted the first iteration of a scaling analysis which used the mass, energy, and momentum conservation equations to derive dimensionless parameters for the heat and mass transfer processes occurring inside containment. Specifically, the analysis addressed heat and mass transfer between the steam emitted from the break and the pools and the solid surfaces in containment. This preliminary scaling study did not model the heat sink effect of the concrete, nor did it address the evaporative cooling of the outer vessel wall or the scale-up of the LST to the AP600. Westinghouse indicated in the report that these issues would be covered in the second iteration of the scaling effort.

The Westinghouse scaling methodology utilized guidance from the Severe Accident Scaling Methodology (SASM) as described in NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," Draft Report for Comment, dated November 1991. The resulting analysis was applied to two limiting transients, the large double-ended, cold-leg LOCA and the large main steam-line break. The similarity between the Westinghouse scaling analysis and the SASM is limited to starting with the identification of phenomena using a PIRT approach. The remainder of the Westinghouse approach, which was based on preserving normalized (and therefore dimensionless) heat fluxes resulting from different processes, departed significantly from the approach presented in the SASM.

The scaling analysis was unconventional and based on normalization rather than on similitude. In using the principle of similitude, basic dimensionless variables (referred to as π-groups) are identified, and are treated as the governing variables for the phenomena (i.e., the phenomena are functions of the π-groups and preserving the value of the π-groups assures that the phenomena are unchanged under a change in scale). There was no justification given by Westinghouse to establish that phenomena are invariant when ratios of heat fluxes are preserved under a change in scale.

The initial documentation of the Westinghouse scaling approach was contained in WCAP-14190, "Scaling Analysis for AP600 Passive Containment Cooling System," dated October 1994. This approach was presented to the staff in a November 1994 meeting (letter NTD-NRC-94-4346, "Presentation Material from the November 15-17, 1994 Meeting on AP600 Passive Containment Cooling System Analyses," dated November 21, 1994), and the staff raised a number of issues and concerns. In response to those comments, the revised Westinghouse scaling approach was presented at the March 1995 ACRS thermal-hydraulics subcommittee meeting (Advisory Committee on Reactor Safeguards, Thermal-Hydraulic Phenomena Subcommittee, Rockville, Maryland, March 29 and 30, 1995). The following description is on the basis of the staff's interpretation of the March 1995 ACRS presentation:

- Changes in the top-down PIRT were made to reflect discussions and lessons learned from the November 1994 meeting.

- The π-groups were consolidated in a different manner to convey information more clearly.
- An attempt was made to relate external π-groups in the annulus to containment pressure.
- A more complete separate effects analysis of stratification in the dome region was presented, which suggests that both the LST and AP600 were clearly operating with stratified atmospheric conditions induced by DBA releases.

Westinghouse divided their scaling analysis into an energy and momentum analysis. In the case of the energy analysis, a separate computer program had been developed to analyze the AP600 containment pressure response to a LOCA and MSLB event. The computer program, referred to as the "scaling model" was used to calculate transient π-groups for internal and external energy transport processes. These groups represent energy partitioning ratios. The momentum analysis addressed the air/steam mixing process that may be occurring within the containment and the downcomer, and also considered the riser buoyancy forces responsible for maintaining air flow in the PCS.

Energy scaling

The Westinghouse energy scaling method was developed in five steps:

(1) construction of a simple computer program to calculate an AP600 containment pressure sequence

(2) generation of π-groups using an energy rate equation for the containment and PCS (coupled)

(3) calculation of the time dependent scaling π-groups for the AP600 with the simple computer program

(4) application of measured dependent and independent parameters to calculate the LST π-groups, as defined in Step 2 above

(5) comparison of the AP600 and LST π-groups to assess whether the LST scales appropriately

The scaling model, or computer program, employed a simple, three-volume containment model. WGOTHIC was not used for the scaling studies. Volumes were associated with the containment vessel, riser, and downcomer. A number of assumptions were made in the model:

- the containment is well-mixed
- the air/vapor density profile is linear in the riser and downcomer
- a fixed evaporation rate is chosen for all calculations (30 lb/sec)
- the internal convection process is free convection

- air and steam are ideal gases
- the break supplies steam and water at a saturation pressure equal to the pressure in the containment
- steam is saturated at the liquid film surface temperature
- the external convection in the annulus is forced
- all heat sinks are lumped as one node

The π-groups were defined using an energy conservation equation, relating the containment atmospheric pressure rate of change to various energy transport processes. These π-groups were therefore formulated as energy rate ratios for each transfer process. The driving terms and resistances required to develop the energy rates were evaluated implicitly within the scaling model. The important groups relating mass-transfer processes did not explicitly include containment characteristic dimensions.

An energy transport mechanism such as condensation from the containment atmosphere to the film surface was represented by a π-group that is a ratio of a condensation energy rate to steam injection energy rate. All π-groups for a volume, referred to as a module, must add to one. The relative magnitudes of the π-group for a volume (containment, riser, or downcomer) indicate the relative importance of that energy transport process in affecting the time rate of change of the containment pressure.

An accident sequence was defined according to a specified steam injection history. The scaling model predicted all the time dependent states of the system (containment, riser, and downcomer) and calculated the time dependent π-groups. Since the AP600 π-groups are time-dependent quantities developed for a specific sequence of events, or steam injection history, the groups are themselves dependent on the time history of the event.

Once the AP600 π-groups were determined, the values were compared to the identical groups, in terms of formulation, for the LST facility test. A specific test was used to determine relevant test π-groups. It is important to note that the scaling model or computer program was not run for the LST test to develop time dependent groups. Experimentally derived quantities in the π-group definitions were substituted directly into these group definitions (see Step 2 above) to establish the π-group value for the specific test. The justification for scaling was on the basis of a comparison of the ratios of the AP600 and LST π-groups. If the ratio is one, then the transport process was considered to be scaled between the facilities.

Momentum scaling

The containment internal momentum scaling, was presented in letter NTD-NRC-94-4264, dated July 28, 1994, "AP600 Passive Containment Cooling System Preliminary Scaling Report," and followed a derivation and correlation application reported by Petersen (P.F. Peterson, "Scaling and Analysis of Mixing in Large Stratified Volumes," International Journal of Heat Transfer, Volume 37, Supplement 1, pp. 97-106, 1994) for mixing by jet and buoyant plumes in non-condensing atmospheres. The geometry for the derivation is one dimensional, that is, the

mixing layers are assumed to be semi-infinite in the horizontal dimension. A ratio of the height (H) of the bounding ceiling above the injection to the jet diameter (d) represents the only geometrical scaling variable for the analysis. A criterion developed by Petersen for stable or unstable stratification was applied to the AP600 and LST steam injections. For the external momentum scaling analysis, the scaling equation was a simplified static mechanical energy balance equation for air flow in the riser. The π-groups were developed to relate gravitation energies in the downcomer, riser, and chimney to the energy losses from air flowing in the riser. An experimentally determined form and frictional loss coefficient for the AP600 PCS downcomer, riser, and chimney geometry was used to define the energy loss term. The form and friction loss coefficients for the AP600 downcomer, riser, and chimney were experimentally determined in a scaled, dry, unheated test facility. These loss coefficients were used in the model to define the energy loss term.

The scaling method was still under development in early 1996 and the staff expected a final report in May 1996. The only discussion available when the SDSER was written which described the Westinghouse scaling analysis was in the form of a presentation to the ACRS in March 1995. The March 1995 ACRS presentation was used as a reference in WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995. The staff requested, in RAI 480.396 (October 1995), that Westinghouse document the ACRS presentation on the scaling approach to support the PIRT provided in WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995. Westinghouse informed the staff that the PIRT was revised in letter NSD-NRC-96-4643,"Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," dated February 15, 1996. Westinghouse informed the staff that the scaling methodology was revised in letter NSD-NRC-96-4790, "Scaling Analysis for AP600 Containment Pressure During Design-Basis accidents," dated August 8, 1996. Therefore, the response to RAI 480.396 which referenced these two reports was acceptable.

In a December 6 and 7, 1995 meeting with the staff, Westinghouse provided a status report on the scaling study ("Summary of AP600 Design Review Meeting Regarding the Passive Containment Cooling System and WGOTHIC Computer Code," letter from D.T. Jackson (NRC) to Westinghouse Electric Corporation, December 27, 1995). The staff determined that Westinghouse was working on a scaling analysis which was not based on normalization but rather relied on a more traditional evaluation on the basis of similitude. In using the principle of similitude, basic dimensionless variables (referred to as π-groups) are identified which are the governing variables for the phenomena; that is, the phenomena are functions of the π-groups and preserving the value of the π-groups assures that the phenomena are unchanged by a change in scale. The staff expected this report in May 1996. The revised scaling methodology was provided in letter NSD-NRC-96-4790, "Scaling Analysis for AP600 Containment Pressure During Design-Basis accidents," dated August 8, 1996.

Westinghouse developed a PIRT for the AP600 PCS, in letter NSD-NRC-96-4643, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," dated February 12, 1996. This report was reviewed by the staff. Earlier versions of the PIRT were in WCAP-14190 "Scaling Analysis for AP600 Passive Containment Cooling System," dated October 1994 and WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995, and superseded. Several phenomena may not have been considered in the February 1996 PIRT or may have been identified as inconsequential; therefore, there was a continued need to obtain closure of the PIRT and the scaling analysis to support design certification.

The following important issues (these are discussed in more detail in Section 21.6.5 as part of the WGOTHIC computer program review) were not considered in the February 1996 PIRT:

- thermal resistance between steel liner and concrete
- fogging in the annulus
- condensation heat transfer to the baffle

The categories presented in letter NSD-NRC-96-4643, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," February 12, 1996 did not appear to be chosen in a systematic manner by considering phenomena and components, rather the categories were selected to fit a preconceived, tabular scheme (generally the same nine phenomena for each component). This necessarily meant that the phenomenological categories were combined into large areas identified only by very general descriptions.

The justification for not considering certain phenomena or for the low ranking of phenomena in the PIRT provided in letter NSD-NRC-96-4643, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," dated February 12, 1996, appeared to be incorrect. For example, the argument given for ranking atmospheric humidity low incorrectly sighted the effect of humidity on saturation pressure, and did not address the effects of humidity on heat capacity, fogging and air flow rate. In Section 3.3.12, "External Atmosphere," of the enclosure to letter NSD-NRC-96-4643,"Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," dated February 12, 1996, Westinghouse stated the following:

> "3.3.12.2 Humidity - The peak containment pressure is limited by mass transfer rates, that are functions of steam partial pressure differences. The peak internal pressure is approximately 40 psia, while the ambient steam partial pressure is limited to the saturation pressure, 1 to 2 psia. In terms of steam partial pressure difference between the inside and outside of containment and riser, the maximum variation expressed as a fraction of the total can be used as a first order estimate of the containment pressure sensitivity to humidity. The resulting variation is less than 3%, so the ranking is L [low] during the post-reflood and 24 hour time phases. The phenomena is not relevant during blowdown and reflood because the PCS has no effect then."

Arguments also provided by Westinghouse in "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System" to dismiss fogging and recirculation cite references which did not seem to totally support the conclusions.

Areas of Concern with the Previous Scaling Analysis

The staff had the following specific areas of concern with regard to the previous scaling analysis:

- The Westinghouse scaling approach was non-standard, relying on preserving ratios of fluxes rather than classical similarity groups. It needed to be demonstrated that preserving ratios of normalized heat fluxes assured that the results of scaled tests were representative of the prototype. Mass and energy transport phenomena are functions of basic similarity dimensionless groups (e.g., Reynolds Number). These basic similarity

groups are regarded as independent variables for the phenomena. It seemed that the fluxes were dependent variables and it was not clear that the Westinghouse approach of preserving these quantities under a change in scale was an appropriate scaling analysis.

- Westinghouse made the statement (in NTD-NRC-95-4561, "Scaling Role in AP600 PCS DBA Analysis," dated September 19, 1995) that scaling analysis helps with the bounding calculations by providing enough insight into the actual physics to show that bounding modifications can be made to the evaluation model. Westinghouse needed to demonstrate that each bounded process had a bound that is independent of scale or covers the prototype (AP600) scale.

- The LST had a significant number of non-prototypical features which may limit its applicability in the area of validation of best-estimate computer programs. These non-prototypical feature of the LST required detailed evaluation by Westinghouse to determine what distortions they introduce when LST data is scaled-up in support of AP600 analysis.

The scaling analysis has been completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The staff's review of the revised scaling analysis, and its use to support the design certification, is provided in Section 21.6.5 of this report.

Areas of Concern with the Previous PIRT

Closure of the PIRT and scaling analysis results was needed. That is, the information developed in the tests needs to be used to confirm the PIRT rankings, and the π-groups identified in the scaling analysis as being important needs to relate to the phenomena identified in the PIRT.

The PIRT has been completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff review of the revised PIRT, and its use to support the design certification, is provided in Section 21.6.5 of this report.

Resolution of the overall LST scaling issue and PIRT was a primary concern of the staff. This was identified as DSER Open Item 21.3.8.5-1. Specific issues that needed to be addressed included the following:

- SDSER Open Item 21.3.8.5-1a – Westinghouse needs to demonstrate that its scaling approach assures that the results of the tests are representative of the prototype. It is not obvious that preserving normalized heat flux ratios assures similitude or that phenomena are not affected by the change in scale. The scaling analysis was completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The staff's review of the revised scaling analysis, and its use to support the design certification, is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1a is closed.

- SDSER Open Item 21.3.8.5-1b - The role played by scaling in a bounding analysis needs to be clarified. Westinghouse needs to relate proposed bounding values to the test data. The scaling analysis was completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The PIRT was completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff's review of the revised scaling analysis as it relates to validation of the revised PIRT and to the development of the bounding analysis (the evaluation model), and its use to support the design certification is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1b is closed.

- SDSER Open Item 21.3.8.5-1c - Westinghouse needs to address phenomena which could be important, but which may not have been included or may have been incorrectly determined to be inconsequential in the PIRT. This includes condensation heat transfer on the inside of the baffle and fogging in the annulus. The PIRT was completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff review of the revised PIRT, and its use to support the design certification is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1c is closed.

- SDSER Open Item 21.3.8.5-1d - Westinghouse needs to extend the PIRT to the next lower level of detail. The current cursory level does not tie the categories to specific phenomena, making the current PIRT of limited value for its intended purpose as a modelers' guide. The PIRT was completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff's review of the revised PIRT, and its use to support the design certification is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1d is closed.

- SDSER Open Item 21.3.8.5-1e - Closure with the PIRT needs to be achieved on the basis of the scaled testing experience. That is, the importance of phenomena as determined by the scaled testing needs to be factored back into the PIRT. Westinghouse needs to (1) address how each of the items identified in the PIRT was handled in the experiments and the scaling process, (2) identify all of the significant non-prototypical features of the LST which could distort the data and which could affect scale-up of the tests to prototypical size, and (3) evaluate their effect on the applicability of the data for computer program validation. The PIRT was completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff's review of the revised PIRT as it relates to items (1), (2) and (3), and its use to support the design certification, is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1e is closed.

- SDSER Open Item 21.3.8.5-1f - The scaling method is still under development and the staff expects a final report in May 1996. Westinghouse revised the scaling analysis, which is now not unconventional (on the basis of normalization), and reflects a more

traditional evaluation using the principles of similitude. The revised scaling analysis is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The staff's review of the revised scaling analysis, and its use to support the design certification is presented in Section 21.6.5 of this report. Therefore, SDSER Open Item 21.3.8.5-1f is closed.

DSER Open Item 21.3.8.5-1, with its sub-parts, is closed as the issues identified are addressed in the revised scaling report and in the revised PIRT report.

21.3.9 Water Distribution Testing Program

21.3.9.1 Introduction

The water distribution tests investigated the effectiveness of the PCS at delivering a uniform water film to the outer surface of the containment vessel. The tests were performed on both a 1/8 sector of a full-scale AP600 containment dome with a shortened sidewall and on a "cap" representing a section of the full scale containment. The facility was located at Westinghouse's Waltz Mill facility in Madison, Pennsylvania. Data gathered from the tests were used to help determine water coverage fractions used as input to WGOTHIC code. Testing was conducted in three phases, with the objectives of each phase as follows:

- Phase 1 - to determine the effectiveness of the water distribution system in delivering water to the containment dome
- Phase 2 - to investigate the ability of the weir system to provide a uniform flow distribution over the containment dome and sidewall
- Phase 3 - to continue the Phase 2 tests to determine the flow off the bottom and sides of the model, and determine the water film thickness

As of July 31, 1994, all of the water distribution tests had been completed and the test data submitted to NRC via the following Westinghouse topical reports (1) WCAP-13353, "Passive Containment Cooling System Water Distribution, Phase 1 Test Data Report, (2) WCAP-13296, "PCS Water Distribution Test Phase II Report," and (3) WCAP-13960, "PCS Water Distribution Phase 3 Test Data Report."

These reports contained only a matter-of-fact presentation of the data and presented no evaluations or conclusions. Westinghouse has since submitted reports PCS-GSR-003, "A Method for Determining Film Flow Coverage for the AP600 Passive Containment Cooling System" and NSD-NRC-96-4646, "Conservatism in Modeling of the PCS Film in the DBA Evaluation Model and Comparison of the Range of Film Parameters in the PCS Test Data with AP600." This closed DSER Open Item 21.3.9.1-1 concerning the documentation of test data.

21.3.9.2 Test Models

The Phase 1 test model consisted of a full-scale, 6.1 m (20 ft) diameter circumferential section (or "cap") of the containment dome fabricated from steel. The complete model assembly was

capable of being tilted horizontally to investigate the effect this had on the surface film distribution.

The model for both the Phase 2 and 3 tests was a full-scale, 1/8 sector of the containment dome attached to a 1/16 sector of the sidewall fabricated from steel. However, while the scale of the sidewall was the same as that of an actual AP600, the full height of the sidewall was not modeled. The height of the sidewall in the model was 3.048 m (10 ft), whereas the height of an actual AP600 sidewall is 25.29 m (83 ft).

Certain welds on the surfaces were made using worst case welding tolerances to represent surface irregularities which would likely be present in an actual AP600. Attempts were made to enhance water coverage by adding surfactants to the water and scoring the surface, but were unsuccessful. However, a commercially available inorganic-zinc paint was used to coat the surface of the model to enhance its wettability.

For all test phases, the water distribution system consisted of a distribution bucket attached to the top of the dome. The distribution bucket had equally spaced slots cut around its perimeter to regulate water flow onto the containment dome. Adjustable weirs were installed on the surface of the dome at two radial locations to promote water distribution over the vessel surface. Water runoff from dome and sidewall was collected using a system of gutters located around the lower circumference of the test structure. The gutters drained into individual buckets, thereby allowing the volume of water in each bucket to be measured and used to determine the azimuthal distribution of water on the model surface. For each test, water flow rates ranged from 26.1 L/min. to 832.8 L/min. (6.9 gpm to 220 gpm). Information provided in the data reports indicate that all tests were run with water at near ambient temperature.

A scaling analysis was not performed for these tests.

21.3.9.3 Test Matrix

A summary of the tests performed in Phases 1 through 3 is given in the Table 21.3-5 of this report.

As indicated in Table 21.3-5, the effect of tilting the containment model on water distribution and the azimuthal water distribution over the dome were investigated in the Phase 1 tests. The performance of various weir and distribution bucket designs was investigated and water film thickness and width measurements were made in the Phase 2 tests. The Phase 3 tests were similar to those conducted in Phase 2, but in addition the PCS baffle wall structural supports were modeled and the effect of tilting the distribution weirs on film coverage was investigated.

The staff noted that none of the tests in Phase 1, 2, or 3 were conducted with a heated dome or sidewall. A heated surface would be expected to affect water coverage differently than would an unheated surface. Westinghouse was made aware of the staff's concern regarding the use of an unheated surface before commencement of the test program. The applicability of water

Table 21.3-5 Summary of Phases 1 through 3 Water Distribution Tests

Effects investigated	Phase 1	Phase 2	Phase 3
Model/Weir tilt	Model	No	Weir
Heated surface	No	No	No
Surfactant in water	Yes	No	No
Worst-case surface welds modeled	No	Yes	Yes
Baffle wall supports modeled	No	No	Yes
Water film thickness/width measurements	No	Yes	Yes
Water runoff measurements	Yes	Yes	Yes

coverage data taken from an unheated surface is discussed in Section 21.5.9 of this report and was identified as SDSER Open Item 21.5.9-2. DSER Open Item 21.3.9.3-1, which addressed the lack of a heated surface, no longer exists since the issue was addressed by SDSER Open Item 21.5.9-2.

21.4 Overview of NRC Activities

The NRC staff has observed the onsite performance of selected tests at many of Westinghouse's test facilities. On the basis of the staff's observations of these tests, as documented in formal test observation reports, the staff has concluded that the AP600 test programs were performed in a competent, professional manner, with due consideration for meeting test specifications and acceptance criteria. The staff believes that the test programs have provided useful data for evaluating AP600 safety system performance; however, a detailed review of the test results has been required for the staff to be able to make a final judgment on the adequacy of the vendor's test programs.

The NRC staff performed a comprehensive review of each Westinghouse design certification test program. The review activities are carried out by several different branches within the Office of Nuclear Reactor Regulation, with support from the Office of Nuclear Regulatory Research. These activities cover a broad range of issues related to the following:

- test facility design, instrumentation, and scaling
- test data and analyses
- quality assurance (QA)

The staff has taken an active role in observing and auditing the applicant's performance in these areas. The staff's activities for each test program are described briefly below. Section 21.7 of this report provides a comprehensive summary of the QA inspections conducted by the staff at Westinghouse's design certification test facilities.

21.4.1 Core Makeup Tank Test Program

The staff reviewed the test facility design, instrumentation, scaling, and test matrix for the CMT test program. Several matrix tests were observed onsite by the staff and its contractors, and the test data and their analysis were reviewed. The review covered both the thermal-hydraulic behavior of the CMT and associated equipment, and the performance of the prototypical heated RTD-level sensors that were originally proposed for use in the AP600 to control ADS actuation.

21.4.2 Automatic Depressurization System Test Program

The staff reviewed the test facility design, instrumentation, and test matrix for Phases A and B1. Scaling was not considered directly in the staff's review, since the major AP600 components under test are full size. However, the staff reviewed Westinghouse's plans to use nozzles or orifices or both to simulate the presence of one of the two valves in each ADS stage in Phase B1. The staff reviewed the ADS test data and Westinghouse's analyses.

21.4.3 Passive Residual Heat Removal Heat Exchanger Test Program

The review of this test program focused mainly on the data, since testing was completed before the staff began its review of AP600 testing. Facility design was evaluated, as previously discussed, in light of the substantial change made to the PRHR system design subsequent to the completion of testing. The staff reviewed Westinghouse's test data and analyses. The audit analyses of testing in this program raised questions about Westinghouse's conversion of the data from electronic signals to engineering units, as well as about the subsequent analysis. These issues were discussed with Westinghouse, and Westinghouse acknowledged the errors in the original data conversion. To address these issues, Westinghouse committed to reanalyze the data and revise and reissue the final test report for this program. WCAP-12980, Revision 3 was issued in April 1997 and resolved the data conversion and analysis issues raised by the staff. Further evaluation of this test program is discussed in detail in Section 21.5.3 of this report.

21.4.4 Departure From Nucleate Boiling Test Program

The staff observed testing in the HTRF at Columbia University and reviewed the test data. As previously discussed, the staff's review of the DNB test program was limited, for the most part, to an evaluation of the test data and its application to extend computer code DNB correlations. Because the HTRF is a well-established facility for such tests, and the tests are similar in character to those that have been done in the past, an extensive review of this test facility has not been needed.

The staff prepared the DSER before it reviewed the data from the Columbia University DNB tests for the flow conditions in the AP600. Westinghouse provided the test data in WCAP-14371. The evaluation of this data is discussed in Section 21.5.4 of this report.

21.4.5 OSU/APEX Test Program

The staff reviewed the OSU/APEX test program extensively. The test facility design was assessed at an early stage of the program, and numerous comments were made to Westinghouse regarding modifications and additions to the loop design and instrumentation. The test matrix was also reviewed as the facility design and capabilities changed. Because this is a reduced-height, low-pressure facility representing a reactor in which natural circulation is particularly important, a scaling analysis showing proper facility design and giving guidance in how to apply test data to analysis of the full-scale plant was considered essential. The staff commented on the draft scaling report produced by OSU and has reviewed the final version of the scaling analysis. The staff issued a number of RAIs on this report as discussed in Sections 21.3.5 and 21.5.5 of this report.

The staff observed selected tests in the facility and reviewed the test data and associated analyses. The staff issued a large number of RAIs related to the test reports, and has received and reviewed Westinghouse's responses. Westinghouse's pre- and post-test calculations were also reviewed, and NRC audit analyses were performed.

21.4.6 SPES-2 High-Pressure, Full-Height Integral Systems Test Program

The staff reviewed the SPES-2 design modifications as proposed by Westinghouse and SIET when the program was established in 1992, and recommended a number of additions and changes to the facility design and instrumentation. The test matrix was also the subject of extensive discussion between the staff and Westinghouse. Scaling is somewhat more straightforward in SPES-2 than in APEX because of the full-height configuration; however, the staff reviewed the scaling of key components to determine if SPES-2 would be able to capture important thermal-hydraulic phenomena. As previously detailed, the staff noted that since SPES was an existing facility requiring some modifications, there were design-related differences from the AP600 configuration and other deviations from ideal scaling, including use of the existing SPES pressurizer, which is somewhat shorter and about 25 mm (1 in.) larger in diameter than a properly scaled AP600 component. Major modifications, such as the addition of the CMTs, IRWST, and pressurizer surge line, conform to the SPES-2 volumetric scale. The staff evaluated the SPES-2 scaling analysis, including the use of the shorter pressurizer, and determined that, despite the distortions, the scaling analysis showed that SPES-2 should provide thermal-hydraulic data over a range of conditions applicable to the AP600 design. The SPES-2 scaling analysis met the objectives described in Section 21.1.5 of this report, and therefore, is acceptable to the staff. Additional discussion of SPES-2 scaling can be found in the evaluation of Westinghouse's AP600 Scaling and PIRT Closure Report found in Section 21.5.10 of this FSER.

The staff observed several SPES-2 tests; additional observation was performed on the NRC's behalf by Agenzia Nazionale per la Protezione dell'Ambiente (ANPA-DISP), the Italian nuclear regulatory agency, as part of an international agreement with that agency. The staff has completed its review of SPES-2 data, and also performed analyses of several SPES-2 tests.

21.4.7 Wind Tunnel Test Programs

The staff witnessed portions of Phase 4A and 4B tests at the National Research Council of Canada's boundary layer wind tunnel in Ottawa, Canada, and at the University of Western

Ontario in London, Canada. The staff's activities included a complete facility tour, which allowed close-up examination of the models and the wind tunnels, observation of confirmatory testing conducted before installing the actual test model, and observation of a test performed to characterize the flow past the cooling tower. Overall, the tests were conducted by competent personnel practicing sound experimental procedures.

21.4.8 Large-Scale PCS Test Program

The staff witnessed Phase 2 matrix test 220.1, "Transient Blowdown Steam Flow, Reduced Water Flow and Coverage Area, Non-Condensible Gas Samples Taken," and test 221.1, "Transient Blowdown Test with Helium Addition." The staff performed a top-level check of the facility QA, testing performance, and test results. While only a fraction of the tests in the complete program were witnessed, it was reasonable to expect that the general observations made of the viewed tests were representative of the other tests in the program, and were generally indicative of the complete program.

A more thorough QA review was performed in May 1995 and was documented in NRC Inspection No. 99900404/95-01, letter from R. M. Gallo (NRC) to N. J. Liparulo (WEC), dated August 8, 1995. The failure to accurately specify instrumentation requirements and the test specification not accurately reflecting the instrumentation that was procured and installed in the PCS LST facility was identified as Nonconformance 95-01-01. The failure to provide instructions or procedures for the verification of critical as-built dimensions was identified as Nonconformance 95-01-02.

It was expected that these two nonconformance issues would be addressed during the WGOTHIC computer program QA audit.

During NRC Inspection No. 999900404/97-02, held from November 17 to November 21, 1997, the staff determined that the test specification was revised to properly reflect the instrumentation installed in the LST. In addition, it was determined that the verification of the critical as-built dimensions were re-verified using acceptable instructions and procedures.

21.4.8.1 Phase 2 Test 220.1

The staff discussed the test objectives, instrumentation, and operational problems with the test supervisor. The objectives were clearly defined, the crew was well-versed in its duties and, because this was a later test in a long series, most testing problems were resolved. Transitional points where problems were likely to arise were discussed, and the staff watched carefully during these periods to ensure they were handled properly. The test was guided by a step-by-step procedure that included test prerequisites and instructions, target test parameters, the necessary instrumentation, and gas sample data sheets. The operators were knowledgeable as to the facility layout and the location of instruments and controls.

Before running test 220.1, the crew had practiced with the steam boiler and was well-versed in its operating idiosyncrasies. However, problems with the boiler almost invalidated the test. The test procedures called for three rounds of data measurements; however, after the second round was completed, the boiler shut down on low feedwater flow and testing ceased. Westinghouse test engineers determined that a third set of data measurements was not necessary. In

examining the first two rounds of measurements, it was found that the pressure readings were within 0.69 kPa (0.1 psi) at each sampling point. Normally a 2.07 - 2.76 kPa (0.3 - 0.4 psi) discrepancy would be considered acceptable for readings that were supposed to be the same. Therefore, a third round of data measurements was not considered to be necessary. The acceptability of Westinghouse's decision to conclude test 220.1 with only two sets of data measurements was identified as DSER Open Item 21.4.8.1-1.

During the test, condensate was collected at five different locations and was removed through a manifold under the vessel. The rate of condensate collection provided a check of steam flow rate measurement during steady-state operation.

Temperatures were measured by a total of 315 chromel-alumel thermocouples. These were batch-sample calibrated by the Westinghouse testing laboratory. Multiple thermocouples were provided at various locations to provide data to determine through-wall vessel heat flux, and to measure internal and external atmospheric temperatures. Only one thermocouple was provided to measure incoming steam temperature. Five steam pipe thermocouples provided a check of steam temperature under steady-state conditions. Containment atmosphere gas sample lines were provided at four locations, and extended into the bulk fluid of the vessel. The gas sample data were collected to be used to assess the ability of computer programs to predict steam stratification and non-condensable gas concentrations within the vessel. The staff observed one sample being taken. A valve was opened allowing vessel atmosphere to flow through a heated collection line into a heated collection bottle. The staff expressed concern as to the uniform temperature of the collection apparatus. Non-uniform temperatures might cause condensation within the sample bottles, distorting the measurement. Furthermore, no sample lines were routed to the dead-end compartments within the test vessel where non-condensable levels were expected to be the highest.

In WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," Revision 1, dated April 1997, Westinghouse discusses the gas sampling apparatus. The procedures used to collect a sample included heating of the tube to 205°C (400°F) (although the internal temperature might be 18°C (65°F) warmer). The probe tube inside the containment was also heated, and could be as high as 167°C (300°F) above the air and steam temperature. This procedure was designed to avoid any problems with measurements resulting from steam condensing in the sampling system by maintaining the gas sample apparatus temperature higher than the 150°C to 175°C (300°F to 350°F) containment atmosphere temperature.

Instrument manufacturers' calibrations and instructions concerning periodic re-calibrations were followed. When required, instrument readings were zeroed following each test. Instrument errors were provided along with the data to the analyst. Re-calibration of steam flow instrumentation was necessary after the installation of the high-capacity steam source. The staff questioned the accuracy of the calibration procedures and was told that accuracy would be discussed in the final test report.

In WCAP-14135, Revision 1, Westinghouse discusses the steam flow measurements for test 220.1. A comparison of condensate and vortex meters indicates that the vortex meter consistently performed at a 15 to 20 percent lower flow than indicated by the condensate over the steady-state period. The vortex meter was operating at the lower end of its operational range during the steady-state period. During the low-flow, steady-state portion of the test the discrepancy noted was about 1.9 percent of full scale but about 15 percent lower than the

recorded condensate readings. (The meter accuracy was quoted as 1 percent of full scale with the range extending from 2.68 to 0.20 kg/sec (5.9 to 0.45 lb/sec) at the meter's test operating conditions.) The Westinghouse test engineering group recommended that 15 percent should be added to the steam flow rate for all times greater than 10.9 hours to compensate for this difference. Post-test calibration showed no discrepancies in performance of the flow meter although the calibration was limited to the lower 15 percent of its span.

Test 220.1 was particularly important because it was the only PCS blind test. A blind test is a test for which data is withheld from test analysts. The analysts predict the test's outcome only on the basis of initial test conditions. The analysis is then compared to the actual test data. In the case of test 220.1, only the test boundary conditions (inlet steam flow rate and temperature; steamline pressure and temperature; air annulus exit velocity and pressure drop; the PCS water flow rate and temperature; wind speed, wind direction and ambient temperature) were provided to the Westinghouse analysts to predict the resulting parameters (internal pressures, temperatures, air-steam flow rates, and non-condensable concentrations) using the WGOTHIC computer program. The boundary data were provided in WCAP-14135, Revision 1. The blind test boundary conditions to be used for the WGOTHIC analysis were provided to the staff in letter NTD-NRC-95-4422, "Mass and Energy Tables for AP600 Large Scale Containment Test 220.1," dated March 27, 1995, and included the 15 percent increase in the steam flow as recommended by the Westinghouse test engineering group.

After the pre-test calculation was performed, the Westinghouse analysis group obtained the post-test data from the Westinghouse test engineering group. The post-test condensation rate data showed that the average steam flow used in the pre-test prediction was too low and that significant dips in the pre-test steam flows, taken from the vortex meter, were in error. Westinghouse modified the mass and energy release rates for the blind test evaluation, and provided the new boundary conditions to the staff in letter NTD-NRC-95-4456, "Revised Mass and Energy Tables for the AP600 Large Scale Containment Test 220.1." The analyses of blind test 220.1, based on the revised steam flow, were provided in WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995.

After a review of the test analysis results presented in WCAP-14382, dated May 1995, and the problems associated with the low steam flow, it was apparent that a third test would not have provided any additional data that would improve the boundary data set for use in the WGOTHIC analyses. The steam inlet flow was too low to be adequately measured by the available instrumentation for the steady-state portion of the test and the boundary conditions used in the analyses would still contain uncertainty and the need to rely on the analysis of the post-test condensation collection data to back out the boundary conditions. Therefore, DSER Open Item 21.4.8.1-1 is closed and the data from Test 220.1 were acceptable for the blind test evaluation.

The submission of all outstanding LST documentation (including the final LST test report) had been previously identified as DSER Open Item 21.3.8.1-1. The submittal of Westinghouse letter NTD-NRC-95-4463,"AP600 Testing Program Report: Large-Scale Test Data Evaluation (PCS-T2R-050)," dated May 15, 1995, closed DSER Open Item 21.3.8.1-1.

21.4.8.2 Phase 2 Matrix Test 221.1

This test was important in that it was the first time that all significant parameters were incorporated into a single test. As with previous staff observations, the staff discussed the test objectives, instrumentation, and operational problems with the test supervisor. The objectives were clearly defined, the test crew was well versed in its duties, and most problems were resolved. The written test procedures and prerequisites were correctly followed by the test personnel. Facility instrumentation seemed satisfactory with the exception that only five anemometers were provided to measure the flow of the containment vessel internal atmosphere. Section 21.5.8 of this report more fully discusses the staff's concerns associated with the PCS testing program.

21.4.9 Water Distribution Test Program

The staff witnessed three of the Phase 3 water distribution tests. These particular tests were chosen for observation on the basis that they spanned the range of design flow rates for the PCS. The staff's activities included a pre-briefing with the test supervisor which included a review of the test objectives, identification of any operational problems, and a discussion of the instrumentation. Because this was a later test in a long series, most operational and equipment problems were resolved. The staff observed that test personnel were well versed in their duties and that the test operators appeared very knowledgeable as to facility layout and the location of instruments and controls.

In general, the instrumentation appeared sufficient for the purposes of the experiment. A particularly difficult measurement was that of the film thickness using a capacitance probe. The difficulty in making the measurements was that the film was not uniform in thickness; rather, its surface on the sidewall exhibited wavy laminar flow. Because of the non-uniformity of the film and the relatively small number of readings taken, the staff noted that the measurements were probably inadequate to accurately characterize the AP600 film flow. The staff further noted that the model contained several representations of welds and surface defects, per WCAP-13290, "Passive Containment Cooling System Water Distribution Test Specification," that could affect the film coverage.

The staff concludes that the water distribution tests were performed by competent personnel following clear, complete procedures. Issues regarding the validity of the test data and test model non-prototypicalities are further discussed in Section 21.5.9 of this report.

21.5 Evaluation of Vendor Testing Programs

This section contains a brief, program-by-program evaluation of Westinghouse's design certification testing. The staff completed its review of all documentation related to reactor-systems-related test programs, including scaling reports, test matrices, quick-look reports, final data reports, and test and analysis reports. The staff requested, in RAI 440.566, that Westinghouse develop a report to demonstrate that the testing programs adequately addressed thermal-hydraulic phenomena and system responses determined to be "important" in Westinghouse's PIRTs for the AP600, and that the parametric range of these phenomena and systems effects were appropriately scaled and captured by the AP600 test facilities. Westinghouse issued the "AP600 Scaling and PIRT Closure Report," WCAP-14727, Revision 1, to address these issues and to resolve outstanding RAIs on many of the test programs, and

subsequently issued WCAP-14727, Revision 2 to respond to staff and ACRS comments and questions. The staff's evaluation of each test program is discussed in the sections that follow. The staff has also completed its review of WCAP-14727; this is discussed in Section 21.5.10 of this report.

The staff notes that the evaluations of the test programs in this section address only such issues as test facility design and scaling, adequacy of the test matrices, test performance and data acquisition, and evaluation of the data with respect to QA, coverage of an adequate thermal-hydraulic range of conditions, and the vendor's discussion of the important thermal-hydraulic phenomena exhibited during the tests. These test program evaluations do not include review of the capability of Westinghouse's accident analysis computer codes to model and represent analytically the phenomena and system responses observed during the testing. The staff's review and evaluation of the adequacy of those computer codes are described in Section 21.6 of this report.

21.5.1 Core Makeup Tank Test Program

Initially, the staff focused its review on the design, instrumentation, and scaling of the test facility, and on the test matrix. As discussed in Section 21.3.1 of this report, Westinghouse modified the original facility design, including test article size, facility configuration, and instrumentation, as a result of comments provided by the NRC staff and the ACRS. The staff determined that the final design of the test facility provided an adequate representation of the key features of the RCS and connecting piping that would affect CMT performance, such as the relative elevations of the steam-water reservoir (representing the reactor vessel) and the test article (representing the CMT) and the resistances of the pressure balance line and CMT drain line. The test article was sufficiently large to provide data representative of actual CMT performance. The staff noted that the aspect ratio of the test article provided a largely one-dimensional representation of CMT thermal-hydraulic behavior; the adequacy of this approach was a scaling issue to be resolved through Westinghouse's scaling analysis (see below). The staff also found that the number, type, and location of the facility instrumentation was adequate to provide data on CMT temperatures, pressures, and flows during the tests.

The staff reviewed the CMT scaling report, and issued RAI 440.52, covering several issues that required additional discussion, such as modeling and scaling of heat transfer, recirculatory and draining behavior (including possible multi-dimensional thermal-hydraulic effects), stratification, and behavior of the CMT steam distributor. The staff and members of the ACRS met with Westinghouse to discuss the scaling report. Westinghouse responded to the specific questions in RAI 440.52 and issued the revised scaling report in January 1995. The staff reviewed Westinghouse's response to RAI 440.52 and the revised scaling report, and found the following:

- Westinghouse demonstrated that the heat transfer between the fluid and the CMT wall during the recirculation phase would be adequately represented in the test article over a similar range of thermal-hydraulic conditions.

- Westinghouse showed that the recirculation and draining rate of the CMT was sufficiently slow during events of interest to make its behavior largely one-dimensional, thus permitting the use of a one-dimensional scaling approach. The effects of

multi-dimensional thermal-hydraulic phenomena on conclusions drawn from a one-dimensional scaling analysis were, therefore, determined to be relatively minor.

- The approach for scaling the steam diffuser in the CMT provided an adequate means of representing steam condensation phenomena near the diffuser.

- While the transition from the recirculatory mode to the draining mode of operation was not represented in a prototypic manner, the process for doing this was sufficiently rapid as to preserve the key thermal-hydraulic phenomena of interest, particularly the temperature profile in the tank.

- Thermal-hydraulic parameters important to condensation phenomena were adequately represented.

- The separate-effects data developed in this test program would be used in conjunction with and augmented by data on CMT performance acquired in the two integral-systems tests programs.

On the basis of these reviews, the staff concludes that Westinghouse has addressed scaling issues in an acceptable manner as addressed in Section 21.3.1 of this report. Further discussion of scaling and post-test "validation" of the PIRT and scaling analysis are found in Section 21.5.10 of this report.

The staff reviewed the final test matrix, and determined that the testing program would permit acquisition of data over most of the operating range of the CMT in the AP600, with respect to temperature, pressure, and flow, and would address the "important" phenomena, as determined from the AP600 PIRT. The design and operation of the facility did not permit acquisition of data at very low pressures. However, this was judged to be acceptable, since both of the integral test facilities would operate down to those pressures and would provide additional data for code validation in that range.

The staff observed testing in the facility to aid in its determination of the acceptability of the data for code validation. Test 306, a constant-pressure drain-down test, was observed on June 1, 1994, and documented in an internal NRC memorandum (Levin to McPherson, dated August 1, 1994). Test 507 was observed on September 16, 1994, and documented in a memorandum (Levin to McPherson, September 28, 1994). This test involved recirculation of the CMT to heat 20 percent of the tank, then depressurization and drain-down of the tank. The staff found that the test was conducted in a competent, professional manner, and that the data were appropriately acquired and then checked after the test was completed. These findings provide confidence that the data are valid and acceptable for use for AP600 code validation.

The staff also reviewed the FDR and the TAR for the CMT test program. The staff issued RAIs 440.259-440.262 and 440.359-440.362 requesting further discussion on the test data and their interpretation, with particular attention to the issues of concern, such as recirculation and draining behavior, stratification, and flashing previously described in this chapter. The staff also requested that Westinghouse address such test performance and analysis questions as post-test review and acceptance criteria, facility performance, and data uncertainty and error analysis. Westinghouse responded to the RAIs by providing further details on the noted technical issues. The recirculation and draining behavior of the test article were compared to

results from the integral test programs, with acceptable results. Flashing was determined to not pressurize the CMT and thus, did not control draining behavior; rather, draining was controlled by the differential density head between the CMT and the SWR. Westinghouse also corrected minor errors identified by the staff, and fully explained the process for qualifying data against acceptance criteria. The reports and RAI responses also satisfactorily discussed overall test performance and analysis issues. The staff concluded that the RAIs, together with the FDR and TAR, acceptably addressed the staff's questions concerning the thermal-hydraulic aspects of the testing program.

The staff also conducted QA inspections to verify conformance with QA commitments for design certification testing. The staff found QA implementation on the CMT test program to be acceptable, with no identified non-conformances. Further information on these inspections is provided in Section 21.7 of this report.

The staff's overall assessment of the CMT test program is that the data acquired during the test program and verified to meet Westinghouse's acceptance criteria are valid and applicable for use to validate computer models for the AP600. The staff concludes that the CMT test program has fulfilled its objectives to provide a separate-effects database to assess AP600 analytical models, per the requirements of 10 CFR 52.47(b)(2)(i)(A).

The staff's post-test review of the data from Test 507, during the test observation previously discussed, identified concerns related to the capability of the RTD level sensors to perform their safety-related functions in a reliable manner. After completion of the CMT test program, Westinghouse submitted WCAP-14442 ("AP600 Core Makeup Tank Level Instrument Test Data and Evaluation Report"), dated August 1995 which contained level sensor performance data over a wide range of CMT conditions. The staff reviewed WCAP-14442 and determined that it provided insufficient information to resolve staff concerns about the ability of the instrument to perform its functions in the AP600. This issue was identified as DSER Open Item 21.5.1-1. Westinghouse has changed the reference design of the CMT-level instrumentation from the original heated RTDs to a set of differential-pressure level switches. The staff considers the change in the reference sensor design sufficient to close the original open item, which was related to the specific issue of temperature and level discrimination by the RTDs. Thus, DSER Open Item 21.5.1-1 is closed. Further discussion of the staff's evaluation of the current reference design of the CMT-level sensor can be found in Section 7.3.6 of this report.

21.5.2 Automatic Depressurization System Test Program

The staff's original review of the ADS test program in the VAPORE facility focused primarily on the Phase B1 facility design, instrumentation, and test matrix. Scaling was not considered a significant issue because of the full-size configuration of the ADS piping network, exhaust pipe, and sparger. The staff determined that the facility design and instrumentation were generally acceptable, but the staff noted that only one of the two ADS valves in each stage would actually be installed in the facility for these tests, with the second valve represented by nozzles and/or orifices designed to give flow area (in choked flow) and flow resistance (in unchoked flow) similar to those of a prototypic valve. The staff requested, in RAI 952.96, that Westinghouse provide a detailed explanation as to how the information derived from the design certification tests are used to develop design performance criteria for actual ADS valves. In response to RAI 952.96, Westinghouse submitted a "road map," describing the specific relationship between the design

certification test program and the qualification (qualification, in this context, refers to demonstration of a valve's capability to meet its design performance criteria) of the actual valves. The staff reviewed the "road map" and concluded that it adequately discussed the range of data acquired during the Phase B1 test program, how the data from the design certification test program are used to validate code models for prediction of ADS performance, and how these models are used to analyze the results of follow-on testing outside of design certification and to thereby establish performance criteria for the AP600 ADS valves. The staff further concluded that the design certification test program test matrix provided adequate coverage of thermal-hydraulic conditions as indicated in the "road map", and addressed the "important" phenomena related to ADS performance identified in the AP600 PIRT, and was therefore acceptable.

In addition, the staff recognizes the importance of the information contained in the ADS "road map" in establishing a process for ensuring that the performance of the actual ADS valves in an AP600 plant meets functional requirements consistent with those determined from the design certification test program and reflected in design-basis analyses performed for the plant. Accordingly, the staff determined that the ADS "road map" should be incorporated into the AP600 SSAR. In addition, the steps in the "road map" leading from the design certification test program to the qualification of the actual AP600 valves should be incorporated into the inspections, tests, analyses, and acceptance criteria (ITAAC) for the AP600 (see Section 14.3 of this report), and cross-referencing between the ITAAC, SSAR, and other appropriate documentation should be included, to ensure that the process is properly and consistently implemented. Westinghouse submitted its response to this RAI by letter dated February 27, 1998. The response included a new section, Section 5.4.6.3, "Design Verification," to be added to the SSAR, which summarizes the process described in the valve qualification "road map." Modifications were also made in the Tier 1 design description of the ADS and in the ITAAC for the valves, to ensure that documentation exists demonstrating that each ADS valve was appropriately qualified and that the valve performance is bounded by qualification tests. The staff reviewed this response and determined that it adequately addresses the open item. Therefore, this issue is closed.

The staff visited the VAPORE facility on several occasions and was able to inspect the test equipment and PC-DAS, and discuss the conduct of testing with the facility staff. None of those visits coincided with the performance of a test. The staff did, however, conduct a QA inspection at the VAPORE facility, during which the data files from the test program were reviewed. The staff concluded that the testing was conducted in a competent, professional manner, and that the data acquired during testing and verified to meet Westinghouse's acceptance criteria, should be acceptable for use to validate computer models. The staff identified one nonconformance concerning as-built drawings of the VAPORE facility, and one unresolved issue concerning instrument calibration. Westinghouse and ENEA provided additional information to address the nonconformance and resolve the calibration issue; these issues were determined to be closed in a subsequent QA inspection. See Section 21.7 of this report for additional details on these issues.

The staff reviewed the FDR and the TAR report for the Phase B1 test program, and issued a number of RAIs concerning the test articles and the use of the data for modeling thermal-hydraulic aspects of valve performance in the AP600. Specific issues included the demonstration that the tests covered an adequate range of thermal-hydraulic conditions, and that the data were adequately characterized to permit their use in the validation of computer

code models predicting ADS performance. The completion of responses to the staff's RAIs on this testing program was identified as DSER Open Item 21.5.2-1. Westinghouse responded to the staff's RAIs. Upon review, the staff concluded that Westinghouse had demonstrated that the test conditions provided sufficient coverage of the operating conditions expected in the AP600, and that the assumptions made in analyzing the data were reasonable. Therefore, DSER Open Item 21.5.2-1 is closed. However, the staff requested Westinghouse to provide additional documentation with regard to these issues, as discussed below.

The ADS test program was discussed in a meeting between Westinghouse, the staff, and the ACRS Subcommittee on Thermal-Hydraulic Phenomena on December 9 and 10, 1997. In that meeting, questions were raised regarding Westinghouse's evaluation of the data from the test program, particularly the calculation of key thermal-hydraulic parameters. The staff concluded that Westinghouse needed to revise the TAR for the ADS test program to more completely discuss how the data were evaluated, including assumptions made with respect to thermal-hydraulic conditions in the test facility (e.g., assumed negligible or unimportant effects), and how these assumptions affect the inferred performance characteristics of the ADS valves in the AP600 plant. In addition, these revisions needed to provide complete justification as to why the range of thermal-hydraulic conditions covered by the ADS test program, and the data acquired therefrom, comprise an adequate basis for validation of code models for ADS performance analysis. Westinghouse submitted Revision 2 of WCAP-14305, dated February 1998, and provided additional information by letter dated February 27, 1998. The staff reviewed WCAP-14305, Revision 2 and the information provided by the letter, and determined that the TAR demonstrates that the range of thermal-hydraulic conditions, particularly flowrate and quality, covered by the ADS Phase B1 tests was adequate to represent ADS operational conditions. In addition, Westinghouse clarified the process used for the analysis of the test data to determine the performance characteristics of the ADS valves and the overall system of valves, piping, and related hardware. The staff determined that this process is acceptable since it incorporates the performance characteristics of the valves as tested plus a substantial margin to ensure that design basis analyses are conservative. Therefore, the program is adequate to provide the basis for validation of computer code models of ADS performance and the staff finds the ADS test program acceptable.

21.5.3 Passive Residual Heat Removal Heat Exchanger Test Program

The staff's evaluation of the test program focused on the applicability of the three-tube test data to the current design of the PRHR heat exchanger for the AP600. Results from selected integral tests and preliminary NRC analyses have shown that the behavior of the PRHR system has a significant — possibly dominant — effect on RCS behavior over a wide range of DBAs. The capability to predict this behavior is, therefore, an important aspect of AP600 analyses. The differences between the test facility configuration and the current AP600 design affect both primary (tube side) and secondary (tank side) heat transfer. Of specific concern are flow distribution and behavior in the tubes and two-phase flow behavior in the IRWST, especially within the tube bundle. High heat transfer rates could cause violent boiling on the outer surface of the tube, resulting in vapor blanketing of some portion of the heat exchanger surface and drastic reduction in heat transfer. Westinghouse analyzed the PRHRHX performance and concluded that it is unlikely that vapor blanketing would occur on the PRHRHX tubes, and that if it did occur, such behavior would be limited to a very short length near the inlet of the tube bundle, leaving sufficient heat transfer area to meet its design performance requirements. The

staff reviewed Westinghouse's analyses; in addition, the staff notes that vapor blanketing was not observed on the simulated PRHRHXs in Westinghouse's OSU/APEX and SPES-2 integral test facilities. Vapor blanketing was also not observed in the NRC's confirmatory test program in the ROSA/LSTF loop. The staff concludes that Westinghouse has resolved this concern.

Although the test configuration of straight vertical tubes represented the in-plant configuration at the time of the test, Westinghouse changed the AP600 heat exchanger design to a vertical C-shaped tube bundle with several hundred tubes. The average tube length is substantially greater than that of the original, straight-tube design; the tube diameter is the same as the earlier design. The nominal design capacity of the heat exchanger is calculated by Westinghouse to be approximately 2 percent of AP600 full power using natural circulation. The staff specified in RAIs (e.g., RAI 440.567 and RAI 952.94) that Westinghouse must justify in detail the applicability of the straight tube data to the C-tube configuration. Furthermore, the staff determined that Westinghouse's data conversion software for this test program contained an error in converting a key flow signal to engineering units. To respond to these issues, Westinghouse committed to reanalyze the data from the PRHR test program and reissue the PRHR test report in a format similar to that used for the other design certification test programs. This was done in WCAP-12980, Revision 3.

After reviewing WCAP-12980 and Westinghouse responses to RAIs and DSER Open Items, the staff determined that there was still insufficient evidence to demonstrate that the data from the PRHR test program was applicable to the C-tube HX design. Therefore, the staff informed Westinghouse that justification of the use of the straight-tube data required analysis of data from a C-tube HX. The staff provided Westinghouse with data from two tests from the confirmatory test program in the ROSA/LSTF at the Japan Atomic Energy Research Institute. The test facility included a scaled C-tube PRHRHX of prototypic tube dimensions and spacing, but with a reduced number of tubes in the HX bundle. The HX was submerged in a large tank of water, simulating the IRWST. The tests included a simulated station blackout (SBO) and a simulated 12.7-mm (0.5-in.) SBLOCA. For both of these tests, the fluid entering the PRHRHX remained a single phase liquid for an extended period. Data provided to Westinghouse included the HX inlet flow and temperature and the temperature profile of the IRWST tank. Westinghouse was required to use its PRHR analysis model, developed from the straight-tube tests, to calculate the PRHRHX outlet temperature and tube wall temperatures at three locations along the length of the tubes. These calculations were performed at five discrete times over the course of the SBO test and at three times over the course of the SBLOCA test.

Westinghouse provided its calculated results by letter to the NRC dated September 18, 1997. Calculated temperatures were on the basis of an equivalent single HX tube and an average IRWST temperature. The staff reviewed the calculated HX outlet and tube wall temperatures. To compare tube wall temperatures, the staff averaged the data from three instrumented tubes in the ROSA/LSTF HX to get an equivalent single-tube temperature. Westinghouse's results predicted the ROSA/LSTF data quite well, with calculated temperatures within a few degrees of measured temperatures. Some of the disagreement between the data and Westinghouse's calculations were attributed to the way the staff averaged the ROSA/LSTF data. In addition, the staff performed confirmatory calculations, using Westinghouse's heat transfer correlations, for the same tests and times as Westinghouse. The staff's results were consistent with Westinghouse's results. In addition, improved agreement between the data and calculated results was observed when the IRWST model was modified to explicitly represent the temperature stratification in the tank, as compared to a single average tank temperature. The

staff concluded that the remaining discrepancies between the data and Westinghouse's calculations were largely attributable to IRWST modeling and not to the performance of the correlations themselves. On the basis of the staff's analysis and the review of Westinghouse's results, as described above, the staff concluded that the straight-tube-based heat transfer model did an adequate job of predicting C-tube HX performance. In view of Westinghouse's analyses of ROSA/LSTF data, all other RAIs and open items became less significant in terms of the test program review and are considered to be satisfactorily resolved. No further issues were found during the staff's review of the Scaling and PIRT Closure Report (see Section 21.5.10 of this report). Thus, the staff concludes that the PRHRHX test program met its objectives and fulfilled regulatory requirements of 10 CFR 52.47(b)(2)(i)(A).

21.5.4 Departure from Nucleate Boiling Test Program

As previously noted, the staff prepared the DSER before it reviewed the data from the Columbia University DNB tests for the flow conditions in the AP600. Westinghouse provided the test data in WCAP-14371. The staff has now completed its review of the VANTAGE 5-H data. The reference case for the AP600 is DNB ratio (DNBR) test data for the VANTAGE 5-H fuel assemblies.

Westinghouse provided the system conditions that bound the low-flow conditions encountered in the AP600 loss-of-flow and locked-rotor accident analyses in Section 5.0 of WCAP-14371. Test data are given in Tables 6-1 and 6-2 of WCAP-14371.

A test data analysis and comparison of the measured CHF with predicted CHF data, showed that the WRB-2 correlation tends to overpredict CHF at the low-flow conditions (less than 4.9E6 kg/hr-m^2 (1E6 lbm/hr-ft^2)). Westinghouse reported data which showed that the magnitude of the overprediction depended primarily on the local mass flux and slightly on the local pressure. As a consequence of these results, Westinghouse applied a multiplier to the WRB-2 correlation to account for the CHF overprediction in Section 8.0 of WCAP-14371. In the low flow regions, Westinghouse refers to the WRB-2 as the Adjusted WRB-2 correlation.

The staff reviewed the DNB testing data submitted in WCAP-14371 and the correction factor applied by Westinghouse. The staff finds the range of test conditions covers the conditions anticipated for AP600 transients. The staff also agrees that the correlation correction applied by Westinghouse for low flow conditions, as discussed above, will result in a conservative estimate for the onset of DNB on the basis of the test data. The analyses and data are therefore acceptable. The following three conditions, which were raised in the DSER, are now closed. A detailed discussion of these issues is included in Section 4.4 of this report.

(1) If the local mass flux in the hot channel is outside the range of the WRB-2 correlation and between 2.34E+06 and 5.08E+06 kg/hr-m^2 (4.8E+05 and 1.04E+06 lbm/hr-ft^2), the adjusted WRB-2 correlation must be used for the DNB Ratio (DNBR) calculation. This was identified as DSER Confirmatory Item 21.5.4-1. In response, Westinghouse revised Section 4.4.2.2.1 of the SSAR to include this condition. The staff finds this acceptable, and therefore, DSER Confirmatory Item 21.5.4-1 is closed.

(2) If the local mass flux is between 4.4E+06 and 1.8E+07 kg/hr-m^2 (0.9E+06 and 3.7E+06 lbm/hr-ft^2), the WRB-2 correlation must be used for the DNBR calculation. This

was identified as DSER Confirmatory Item 21.5.4-2. In response, Westinghouse revised Section 4.4.2.2.1 of the SSAR to include this condition. The staff finds this acceptable, and therefore, DSER Confirmatory Item 21.5.4-2 is closed.

(3) This acceptance applies only to VANTAGE 5-H fuel assemblies. This was identified as DSER Confirmatory Item 21.5.4-2. In response, Westinghouse revised Section 4.4.2.2.1 of the SSAR to include this condition. The staff finds this acceptable, and therefore, DSER Confirmatory Item 21.5.4-3 is closed.

21.5.5 Oregon State University/Advanced Plant Experiment Test Program

The staff completed its review of the OSU/APEX test program. In the DSER, four outstanding RAIs were identified to be addressed by Westinghouse. Completing this review was identified as DSER Open Item 21.5.5-1; this item remained open, pending submittal and review of responses to RAIs 440.573, 440.575, 440.576 and 440.580. Westinghouse submitted responses to these RAIs by letters in February and March 1998. RAIs 440.573, 440.575, and 440.580 all requested minor clarifications related to Westinghouse's interpretation of data as provided in WCAP-14292, Revision 1, the OSU/APEX TAR. The staff reviewed these responses and has determined that they address the staff's questions and provide plausible explanations regarding data interpretation. RAI 440.576 requested that Westinghouse provide an uncertainty analysis focusing on derived quantities (i.e., information that was not directly measured by the test facility instruments but involved a combination of the data from several instruments to calculate the value of a parameter). (An example of this is derivation of actual liquid levels from collapsed liquid level data on the basis of differential static pressure, modified by fluid temperature data.) Westinghouse's response to RAI 440.576 provided a detailed uncertainty analysis, demonstrating by using mass and energy balances for the facility that the uncertainties in derived quantities were effectively bounded by instrumentation errors as discussed in the TAR. The staff has reviewed this response and considers it acceptable to answer the staff's questions. Since Westinghouse has responded acceptably to all of the staff's outstanding RAIs, DSER Open Item 21.5.5-1 is now closed.

In addition to Westinghouse's design certification testing, the NRC conducted a confirmatory testing program in the OSU/APEX facility. Insights from those confirmatory tests that bear upon integral system behavior in general, and facility response in particular, have been factored into the test program review.

Staff review of the facility design, instrumentation, test matrix, and scaling of the OSU/APEX facility evolved with the test program. As originally conceived, the facility was to be a low-pressure (approximately 345 kPa (50 psia)) loop to investigate the last part of the plant depressurization and long-term cooling behavior in SBLOCA events. Westinghouse and OSU gradually changed the design to increase the maximum pressure to about 2.7 MPa (400 psia), which improved scaling and allowed a wider range of test conditions to be explored. The staff determined that the final design was acceptable. Essentially, the entire primary system was represented, including all safety-related systems, in a geometry very similar to the actual plant design. While the containment itself was not simulated, the two sump tanks provided a scaled representation of the volumes into which flow from the break and ADS-4 discharge, and condensate from the passive containment cooling system would drain and recirculate to the RCS. The facility was extensively instrumented to provide temperature, pressure, flow, and void fraction data throughout the system. The test matrix focused on SBLOCAs for two reasons

(1) within the design basis, LOCAs are the only events to cause the ADS to actuate and to progress to long-term cooling; (2) LBLOCA response in the AP600 was calculated to be similar in many ways to conventional designs, and Westinghouse asserted that important phenomena in LBLOCAs related to long-term cooling would be similar to SBLOCA behavior. The staff agreed with Westinghouse's approach and found this reasoning acceptable, and determined that the OSU/APEX test matrix provided adequate coverage of break size and location to address important system-related phenomena identified in the AP600 PIRT.

Westinghouse submitted the final scaling report for the OSU/APEX facility, WCAP-14270, in January 1995. The staff issued a number of RAIs, which, as discussed in Section 21.3.5 of this report, have been adequately answered by Westinghouse, with the responses included in Appendix A of WCAP-14727, Revision 2. The staff requested clarification of assumptions made in deriving scaling parameters and verification that important hydraulic characteristics, such as the distribution of frictional pressure losses, were properly scaled. The staff reviewed Westinghouse's responses to the RAIs and determined that the information provided demonstrated that the scaling had accounted for these issues, and that the scaling analysis for the APEX facility appropriately reflected consideration of key AP600 thermal-hydraulic phenomena, and therefore completes the review of the scaling report. This was identified as part of DSER Open Item 21.3.5-1 and supports the closure of this item, as discussed in Section 21.3.5 of this report.

The staff observed design certification testing in the OSU/APEX facility on two occasions. Test SB19, performed July 14, 1994, was a simulated 50.8 mm (2 in.) cold leg SBLOCA. In addition, an elevated containment pressure, as might exist during such an event in the plant, was simulated by increasing the pressure in the tanks simulating the sump and the IRWST. The staff's observations were reported in a memorandum, Levin to McPherson, dated July 21, 1994. Two tests were observed on a subsequent trip. Test SB6, a 101.6 mm (4 in.) cold leg break, was performed on July 19, 1994, and Test SB12, a double-ended guillotine break of a direct vessel injection (DVI) line, was performed on July 21, 1994. The staff's observations were reported in a memorandum, diMarzo and Bessette to McPherson, dated September 16, 1994. In all three tests, the staff observed that the tests were performed in a competent and professional manner. The PC-DAS allowed selected instruments to be monitored in real time to provide insight into facility response. The staff's observations indicated that the data acquired meeting Westinghouse's acceptance criteria would be acceptable for use in developing and validating computer code thermal-hydraulic models. The staff also performed a QA inspection at OSU to determine compliance with QA commitments for safety-related testing. Minor nonconformances were identified concerning instrument calibration and verification of critical as-built facility dimensions. An unresolved item as also identified concerning disposition of test results that did not meet Westinghouse's acceptance criteria. The records indicating disposition were not kept at OSU, but rather at Westinghouse. A follow-up QA inspection verified proper disposition of these results; Westinghouse also provided information on instrument calibration and facility dimensions to close the two nonconformances. Additional details on the QA inspections is in Section 21.7 of this report.

Westinghouse submitted the FDR (WCAP-14252) and the TAR (WCAP-14292, Revision 1) for the OSU/APEX test program in May 1995 and September 1995, respectively. This was identified as part of DSER Open Item 21.3.5-2 and supports the closure of this item as discussed in Section 21.3.5. However, the staff discussed specific concerns with Westinghouse,

and issued many RAIs related to these two reports, the majority of which requested clarification of unclear points or correction of errors. Technical concerns raised by the staff included the following:

- A check valve was installed improperly in the facility. This error permitted flow between the two DVI lines, which was not quantified. The effect of this unmeasured flow required evaluation. This was the subject of RAI 440.570.

- Potential scaling distortions related to the break size and simulated ADS valve sizes were identified. Westinghouse was requested to provide further discussion of the reasons for and the effects of these distortions. This was the subject of RAIs 480.274 and 440.376.

- Several types of oscillations, during various phases of the tests, were noted during Westinghouse's program. Further investigation of these phenomena indicated that some may be related to specific aspects of the facility's design, and that others could be representative of AP600 phenomena. The oscillatory phenomena, their causes, and any implications with regard to reactor safety system performance required evaluation. This was the subject of RAI 480.275.

- A higher-than-expected break flow was measured in Test SB5. Westinghouse was requested in RAI 440.584 to determine why this anomaly occurred and what the effect could be on the acceptability of the test.

Westinghouse provided responses to RAIs 480.275 and 440.584 and included the responses in Appendix A of the Scaling and PIRT Closure Report, WCAP-14727, Revision 2. RAIs 440.570, 440.274 and 440.376, regarding the check valve issue and the scaling of the break and ADS valves, were handled as potential distortions and were discussed in WCAP-14727, Revision 2. (see Section 21.5.10 of this report for further discussion of this issue.)

The oscillations identified during tests were determined to occur for several reasons. In some cases, such as oscillations related to CMT draining during the long-term cooling phase, the behavior was determined to be specific to the OSU/APEX loop and was related to CMT refill during the test. CMT refill was able to occur because of the reduced height of the test facility and is not expected to occur in the AP600. The impact of this distortion was considered to be relatively minor. Other oscillations occurred because of the interaction between various volumes of water in the facility at different periods of the transient. The staff believes that these oscillations could potentially occur in the AP600. However, the oscillations are self-limiting, and cease when the mixture level in the reactor vessel drops below the top of the hot leg. This level still provides a large margin to uncovering the core; thus, the staff concluded that oscillations do not represent a safety concern in the AP600. The reason for the anomalous break flow in Test SB5 could not be determined. The amount of excess flow was consistent with a leak around the orifice used to simulate the SBLOCA for this test. Since the break flow was measured directly, far downstream of the orifice, determination of the conditions for the test is not dependent on trying to model directly the flow through the orifice. Thus, rather than the nominal conditions for the test (simulated 25.4 mm (1 in.) break), the actual flow was closer to that expected for a 50.8 mm (2 in.) break. The data are still applicable for code validation, and the staff determined that the coverage of test matrix was adequate despite the anomaly. Thus, staff review of Westinghouse's responses found them to be acceptable to resolve the issues

raised. The key technical issues related to the OSU/APEX program are discussed above. However, the staff asked numerous RAIs related primarily to clarification of information provided in the FDR and the TAR. The staff reviewed Westinghouse's responses to all other OSU/APEX RAIs and questions from other sources (e.g., meetings with the ACRS), and found them acceptable. In addition, while the staff's confirmatory test program provided substantial additional data from the OSU/APEX test facility, and contributed to the staff's understanding of potential AP600 integral system behavior in design-basis events, the staff did not identify any significant new phenomena that call into question the adequacy or applicability of the design certification test program at OSU.

21.5.6 SPES-2 High-Pressure, Full-Height Integral Systems Test Program

The completion of the staff's review of the SPES-2 testing program was identified as DSER Open Item 21.5.6-1. The only remaining issue for this program was the review of Westinghouse's scaling closure report. The staff designated completing the review of the scaling closure report as SDSER Open Item 21.5.10.1-1. DSER Open Item 21.5.6-1 was subsumed into SDSER Open Item 21.5.10.1-1, and therefore, DSER Open Item 21.5.6-1 is closed.

Early staff review of the SPES-2 program focused on test facility modifications (since SPES was an existing facility), instrumentation, scaling, and the test matrix. As discussed in Section 21.3.6, SPES-2 included several distortions that arose either as a result of its 1/395 scaling ratio (e.g., heat loss, metal heat addition) or the necessity of modifying an already-existing facility (e.g., external downcomer, one pump per loop). The staff reviewed the impact of these distortions. The major design distortions were found to be acceptable. In the case of the downcomer, the addition of an annular section to accommodate the cold legs and DVI lines was found to adequately characterize flow behavior within the context of the one-dimensional system representation provided by the SPES-2 facility. The single pump per loop, rather than the two pumps in the AP600, was considered to be a relatively minor distortion, since the safety system response in the types of events simulated in SPES-2 includes tripping the reactor coolant pumps early in the accident, coincident with CMT actuation. The resistance to natural circulation flow through the AP600 piping and two pumps was represented in SPES-2, and the elevation of the cold legs with reference to the steam generators was maintained. The thermal-hydraulic distortions resulting from the extra power in the initial stage of the transients and the oversized ADS-4 vent are discussed further in Section 21.5.10 of this report.

Because the SPES-2 facility was full-height and operated at full pressure and scaled full power conditions, the scaling analysis was a relatively simple comparison of the configuration (piping and key elevations), component flow areas, and pressure losses in SPES-2 to the AP600 plant. Except for those distortions identified above, the facility compared well to the plant. Innovative designs were employed to minimize distortions where possible. For example, the SPES-2 CMTs were full-pressure and full-height, but were about 1/20 the diameter of the AP600 components. As explained above, this can cause a distortion in the structural heat content, which could affect CMT draining behavior. To minimize the distortion, the SPES-2 CMTs were designed with thin walls. Since these tanks could not withstand differential pressures up to full RCS pressure, they were placed inside larger vessels that were pressurized with air to maintain acceptable stresses in the tank walls. As the loop (and CMTs) depressurized during a test, air

was exhausted from the vessels to maintain acceptable structural conditions. The staff found the scaling approach and modified facility design acceptable.

The staff reviewed the test matrix and determined that it covered an adequate range of SBLOCA sizes and locations; the simulation of the SGTR and main steamline break events was also found to be adequate. The staff also determined that the test program would address many "important" items related to integral system behavior as identified in the AP600 PIRT, especially at elevated system pressures and temperatures outside the range of conditions covered in the OSU/APEX tests. All of the tests represented "design-basis" accident scenarios (including a single active failure), with the exception of one SGTR test, and one SBLOCA test that included the use of (simulated) non-safety systems to assess potential adverse systems interactions. The testing procedure for all SBLOCA tests was to consider the test as concluded when stable IRWST injection was established (i.e., there was no attempt to represent long-term cooling, as was the case in the OSU/APEX tests). The only exception was the test with non-safety systems simulated, which was not predicted to fully depressurize; use of the simulated RNS system as a low-pressure injection system prevented the CMTs from draining to the ADS-4 level setpoint. For the non-LOCA tests (SGTRs and main steamline break), the criterion for test termination was for the system to be stabilized at elevated pressures, since within the design basis, these events were not predicted to result in ADS actuation.

Westinghouse's program was guided in part by pre-test predictions performed by Ansaldo. As previously discussed, the staff did not formally review those pre-test predictions, since they were performed using RELAP5/MOD3. The NRC used the RELAP code to perform audit calculations of both test facility experiments and AP600 accidents and transients. However, it is not one of the codes being used by Westinghouse for AP600 design certification analyses, and the pretest predictions are thus, to some extent, irrelevant as far as review of the AP600 testing and analysis programs.

The staff observed several tests in the SPES-2 facility. Matrix Test 3, a 50.8 mm (2 in.) cold leg SBLOCA, was actually the first test performed in the program on February 5, 1994. The staff's report is contained in a memorandum, Levin to McPherson, dated February 22, 1994. Matrix Test 6, a double-ended guillotine DVI line break, was performed on June 10, 1994, and is reported in a memorandum, Jensen to McPherson, dated July 18, 1994. Matrix Test 12, a simulated main steamline break at hot, zero-power conditions, was performed on October 11, 1994, and reported in a memorandum, Levin to McPherson, dated October 25, 1994. In addition to NRC staff observations, the NRC also received reports of observation of selected tests by staff members of the ANPA, the Italian nuclear and environmental regulatory agency. The staff found that the tests were conducted in a competent and professional manner and that data were properly acquired and verified. Reports by ANPA were consistent with staff observations. On the basis of these observations, the staff found that the data meeting Westinghouse's acceptance criteria are appropriate for computer code validation. The staff also performed a QA inspection at SIET Laboratories to verify compliance with commitments related to QA implementation for safety-related testing. The staff found that SIET had an excellent QA program, which was fully implemented for the SPES-2 program. One minor nonconformance was identified concerning verification of critical as-built dimensions, and an unresolved item was identified concerning disposition of test data that did not meet acceptance criteria. During a follow-up QA inspection, SIET and Westinghouse provided additional information demonstrating that critical dimensions had been accurately obtained, and

that unacceptable test data were properly dispositioned. The staff concluded that these items were acceptably closed.

The staff's and ANPA's on-site observations, and examination of preliminary ("quick-look") test reports, indicated that the facility appeared to operate predictably. The tests simulating design-basis SBLOCAs ended with the facility fully depressurized and with the core covered and cooled using IRWST injection. The SBLOCA test that included use of non-safety systems also proceeded as expected, with injection from the IRWST using the NRHR system pumps allowing the system to stabilize at a slightly elevated pressure. For the non-LOCA transients, the system stabilized at elevated pressures with no ADS actuation. CMT injection in the recirculation mode was successful in adding coolant inventory without reducing CMT levels to the ADS-1 setpoint. No core heatup was observed during any of the tests. Oscillations were observed during several of the tests, and were especially persistent during the SGTR tests. However, the reactor vessel liquid levels were well above the top of the simulated core. The staff requested Westinghouse to address this behavior in RAIs 440.417 and 480.242. Upon review, the staff found that Westinghouse's RAI responses provided plausible explanations for the observed phenomena, relating them to density-wave oscillations that ceased when the steam generators drained. Since in a LOCA this would occur well before any possibility of uncovering the core, the behavior did not raise safety concerns. In the SGTR test, the steam generators did not drain for an extended period, causing the oscillations to persist. However, this did not affect the ability of the CMTs to recirculate and stabilize the system pressure and temperature without ADS actuation. The staff finds Westinghouse's responses to these questions adequate to resolve the staff's concerns.

The primary issues to be addressed by the SPES-2 tests were related to integral systems behavior, especially at elevated pressures beyond those covered in the OSU/APEX experiments. Examples of these issues are as follows:

- system response and systems interactions in the early stages of design-basis accidents, including transition from CMT recirculation to CMT draining; accumulator injection; and effects of early stages of depressurization

- for non-LOCA transients, ability of the system to come to a stable condition at elevated pressures

The staff reviewed in detail the FDR and the TAR, and issued RAIs 440.417 – 440.430, 440.528 – 440.540, and 480.225 – 440.242. Many of the RAIs concerned clarification of Westinghouse's descriptions of the sequence of events in the tests and were not critical technical issues. In response to staff questions, Westinghouse provided additional discussion of the events in the tests both in RAI responses and in revisions to the FDR and TAR. The staff determined that these clarifications resolved the staff's concerns and are acceptable. Several of the RAIs also dealt with issues related to distortions from identified sources (e.g., excess power to compensate for heat losses) and "validation" of those aspects of the AP600 PIRT relevant to the SPES-2 tests. These were addressed by Westinghouse in WCAP-14727. (See Section 21.5.10 of this report.) The major technical issue not related to distortions was the oscillatory behavior, which was acceptably resolved as discussed above. Therefore, the staff's reviews of the FDR and TAR and of Westinghouse's RAI responses support the conclusion that the SPES-2 test program accomplished its objectives and that Westinghouse developed a

database that addressed the issues raised by the staff. Also, as with the APEX test program, insights gained from NRC-sponsored AP600 confirmatory testing in both the ROSA-AP600 facility and in the OSU/APEX facility were considered in the SPES-2 evaluation. Therefore, on the basis of its review of the test program reports and RAI responses, the staff has determined that the SPES-2 testing program is acceptable and meets the requirements of 10 CFR 52.47(b)(2)(i)(A). Resolution of SDSER Open Item 21.5.10.1-1 is discussed in Section 21.5.10.1 of this report.

21.5.7 Wind Tunnel Test Programs

21.5.7.1 Phase 1

As indicated in WCAP-13294, pressure measurements taken during Phase 1 indicated that there was a net positive pressure difference, with substantial fluctuations, between the inlets and the chimney for many of the test cases. Experimentation determined that this difference could be somewhat reduced by adjusting the inlet vent configuration. The pressure difference was relatively insensitive to the height of the chimney and adjacent turbine building, but installation of the chimney cap resulted in noticeably smaller mean pressure differences, still with substantial fluctuations. The presence of the cooling tower increased the fluctuating component of the pressure difference for those tests in which it was upstream of containment.

21.5.7.2 Phase 2

The main results of the Phase 2 tests, documented in WCAP-13323, indicated that flow non-uniformities existed around the building circumference and as far as 2/3 of the way down the shield building annulus for one of the chimney designs. However, very little annular flow was observed for the other chimney design tested. The peak pressure differences between the inlets and chimneys occurred for 1 to 5 seconds, and the cooling tower caused a significant increase in the fluctuations of the measurements.

21.5.7.3 Phase 4A

As indicated in WCAP-14068, data from these tests showed that at higher wind tunnel flows, the test results became insensitive to the Reynold's number. To account for differences in results taken in the range between the Reynold's number used for the main UWO tests and the Reynold's number above which the results do not change, the mean and peak baffle wall and inlet-minus-chimney pressure coefficients for the Phase 2 tests were adjusted by a factor determined from the phase 4A data. The data also indicated that the presence of the cooling tower resulted in substantially lower mean loads but larger peak loads on the baffle. However, the effect of the cooling tower was not considered large enough to warrant any adjustment of the Phase 1 and 2 data.

Baffle loading under simulated tornado conditions was found to be lower than that under normal conditions because of a reduction in the unsteady component of the pressure difference across the baffle. The tornado loads were also enveloped by the hurricane induced loads determined in the Phase 2 tests. Westinghouse concluded the hurricane-induced loads were conservative for determining baffle-wall loading under tornado conditions.

21.5.7.4 Phase 4B

As indicated in WCAP-14091, for most configurations of the Phase 4B tests, the largest peak inlet-minus-chimney pressure varied little from the base case. In the river valley case with a cooling tower, the inlet-minus-chimney pressure difference increased by a factor of 1.14 over the base case for a small range of wind angles. It was concluded in WCAP-14091 that the baffle loads determined from the base case were bounding, with the exception of the river valley with two cooling towers case.

The report also stated that the effect of mountains or an extra cooling tower was to reduce the inlet-minus-chimney pressure difference for those wind directions with the mountains or cooling towers upstream of the plant. In all such cases, the mean difference remained positive.

The most negative fluctuation in the inlet-minus-chimney pressure difference occurred for certain incoming wind angles in the case of a river valley terrain with one cooling tower, with the pressure difference negative 68-percent of the time versus 4.5-percent for the base case. Westinghouse letter PCS-T2C-059, "Analysis of AP600 Wind Tunnel Testing for PCCS Heat Removal," indicated that the addition of a second cooling tower had little effect on the fluctuations or pressure difference.

The staff previously identified closure of the issue of whether the AP600 containment is wind neutral, wind-positive, or wind-negative as DSER Open Item 21.5.7.4-1. According to the American Society of Heating, Refrigerating and Air Conditioning Engineers, Inc. (ASHRAE), pressure fluctuations on the outside of a structure can be caused by other upwind structures and terrain irregularities. Such fluctuations generally have time periods from one to several seconds. The magnitude and direction of the differential pressure across openings in a structure (the downcomer and annulus in the case of the AP600) can depend on the configuration and size of the openings (ASHRAE Handbook, 1985 Fundamentals, pp. 14.5 - 14.10).

In document PCS-T2C-059, "Analysis of AP600 Wind Tunnel Testing for PCS Heat Removal," dated May 1995, Westinghouse presented an analysis of the effect of oscillating inlet-minus-chimney pressure differences on the AP600 containment LOCA response. The Westinghouse analysis reasoned that the time constant which characterized the vessel shell response to heat transfer in the annulus, and the time constant which characterizes the shell heat capacity relative to its conductivity, were both longer than the period of pressure fluctuations observed in the wind tunnel tests (255 and 69 seconds, respectively, versus several seconds for the period of pressure fluctuations, as indicated in W letter PCS-TAC-059). The stated period of the pressure fluctuations were on the order of those cited from the ASHRAE reference in the preceding paragraph. Westinghouse concluded from these observations that the response of the containment shell and the temperature on the inside surface of the vessel would be relatively unaffected by external pressure fluctuations.

For cases involving lower wind speeds, corresponding to lower frequency pressure oscillations, the inlet-minus-chimney pressure differences were correspondingly much lower. Westinghouse concluded that such pressure differences would not have a significant impact on PCS heat removal, since lower pressure differences result in reduced forced flow in the annulus.

The analysis also included an investigation of the effect of pressure oscillations applied to the chimney of the AP600 WGOTHIC model. The data used for input came from the river valley with one cooling tower test configuration, which contained the highest percentage of negative inlet-minus-chimney pressure differences (on a time basis) of all the test cases. The containment pressure response to a LOCA was calculated with WGOTHIC for various external wind speeds, most of which used a forced convection heat transfer correlation in the annulus. This was considered conservative since, as stated in letter PCS-TAC-059, it would tend to underpredict heat transfer when flow in the annulus reversed and the velocity in the annulus temporarily passed through zero because of pressure oscillations.

The analysis indicated that oscillating inlet-minus-chimney pressure differences slightly increased heat removal from the containment, thereby slightly decreasing the calculated pressure. For wind profiles representative of different terrains, such as open country terrain, the report indicated that the mean pressure difference was above zero and would therefore tend to enhance forced convection heat transfer. Westinghouse concluded from these studies that a conservative calculation of the containment response to a LOCA or MSLB should include the assumption of no imposed wind conditions, and stated that this methodology is used in the SSAR analysis (starting with Revision 5).

The staff reviewed Westinghouse's methodology for assessing the wind-neutrality of the AP600 containment and finds, based on the above, both the conclusion that no imposed wind conditions is conservative for the AP600 containment analysis and the decision to use this methodology in design-basis analyses is acceptable. Therefore, DSER Open Item 21.5.7.4-1 was closed.

The staff emphasized that the acceptability of the convection heat transfer model used in the annulus, and the review of the WGOTHIC code, were still open items as identified Sections 21.5.8 and 21.6.5 of this report. The staff determined that the convection heat transfer model used in the annulus, for the licensing analyses to support the AP600 design certification, is acceptable, as discussed in Section 21.5.8 of this report. The acceptability of WGOTHIC is discussed in Section 21.6.5 of this report. Therefore the associated open items are closed.

21.5.8 PCS Test Program

Non-Condensables, Mixing and Stratification

During the witnessing of LST test 220.1, the staff noted that there were only 4 gas sampling ports for measuring and sampling the internal vessel atmosphere for non-condensable concentrations. The samples were taken at a location approximately 15.2 cm (6 in.) from the vessel wall. These experiments were not able to yield data that could be used to validate and verify detailed 3-D spatial distribution of non-condensables as calculated by computer programs such as COBRA-NC or COMMIX. Only coarse axial, integral results were available (e.g., a grab-sample non-condensable concentration in and near the vessel dome versus near and below the operating deck). Furthermore, the bomb sampling technique for sorting out the ratios of steam concentration to those of air and helium was subject to its own experimental uncertainties. In the test specification WCAP-13267, "Test Specification: Large-Scale Containment Cooling Test," AP600 Document Number PCS-T1P-002, Revision 1,

December 1991, Westinghouse described a 4 percent uncertainty band in the air/steam ratio devices. An evaluation of the sampling error was provided in WCAP-14135, Revision 1, dated April 1997. The reported error in the measurement of the partial pressure of air was ± 7.10 kPa (1.03 psi).

Westinghouse has provided comparisons of the measured to WGOTHIC calculations of the air pressure ratios (the air partial pressure divided by the total vessel pressure) for 9 of the 11 priority tests which were analyzed with the distributed parameter model in WCAP-14382, dated May 1995. In two of the tests data were not taken. The helium pressure ratio comparison is also provided for the test (219.1) with helium addition. Two other Phase 2 tests (217.1 and 218.1) with helium injection were not analyzed as part of the priority test matrix. Only three cases using the WGOTHIC lumped parameter model were provided for comparison of the measured to calculated air pressure ratios. Two of these did not have companion distributed parameter model analysis. No uncertainty bands were provided on the plots for the measures values. In response to RAI 480.392, the air pressure ratio predictions from the distributed parameter model for the nine priority tests were re-plotted with the uncertainty bands shown on the data. Westinghouse stated in the response that the conclusions regarding mixing and stratification were not altered because of the uncertainty in the test data. While the comparisons provided seem to show, in a qualitative manner, reasonable agreement between the test data and the analyses for the distributed parameter model, issues concerning non-prototypicalities in the LST and the revised DBA evaluation model approach needed to be resolved. Non-prototypical features of the LST design, for example, the isolation of the steam generator subcompartment from the other below deck regions, which could lead to different flow patterns (both forced and natural circulation) and non-condensables distributions were identified as part of SDSER Open Item 21.3.8.5-1e.

In August 1995, Westinghouse informed the staff that the AP600 (DBA) evaluation model would be based on a conservative assessment of mixing and stratification. An earlier report on mixing and stratification, letter NTD-NRC-95-4459, "Report on Stratification and Mixing Effects on AP600 Passive Containment Cooling System DBA," dated May 8, 1995, was based on the Westinghouse methodology before the change in August 1995 to a conservative, bounding DBA evaluation model. In letter NSD-NRC-96-4763, dated July 1, 1996, Westinghouse submitted a report titled "Assessment of Mixing and Stratification Effects on AP600 Containment."

In keeping with the new DBA approach, mixing and stratification is to be treated in a conservative manner. As previously identified in SDSER Open Items 21.3.8.5-1b and 21.3.8.5-1e, the treatment of non-condensables in the LST program and the assessment of the impact of non-condensables on the prototypical AP600 were still under staff review. The issue of the treatment of mixing and stratification in the AP600 DBA evaluation model, and the use of the LST data to support the evaluation model, was identified as DSER Open Item 21.5.8-1. The use of the LST data to support the evaluation model were presented in the revised scaling report, WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The treatment of mixing (circulation) and circulation to support the evaluation model is presented in Section 9 of WCAP-14407, "WGOTHIC Application to AP600," Revision 3, dated April 1998. The staff review of these documents is presented in Section 21.6.5. DSER Open Item 21.5.8-1 is therefore considered to be closed.

PCS Water Coverage Flow Rate

The staff also observed that the 50-percent water coverage specified for the first few hours of test 220.1 was only a target. A 90-to 95-percent coverage was more representative of what was actually, on the average, achieved for test 220.1. Even this number is somewhat misleading, since during the test large portions of the exterior dried out from time to time, such that the range of water coverage was probably in the range of 40 percent to 100 percent. This seemed partially because of what was occurring inside containment, and partially as a result of the variability in water pressure to the j-tubes hooked up to the facility's municipal water supply. Additionally, water striping measurements were simply made by a technician with a ruler during the steady-state end portion of the run. The staff was concerned with the uncertainty of experimental results which simply state that the test was run with, for example, 95-percent striping.

During the May 1995 QA audit of the LST facility (NRC Inspection No. 99900404/95-01, letter from R. M. Gallo (NRC) to N. J. Liparulo (WEC) dated August 8, 1995), the staff discussed water coverage with the Westinghouse test engineering group. Test 219.1 was discussed during the audit. As with other tests, the staff noted that the water flow rate was not constant.

The water distribution flow control valve was set to a predetermined position (for test 219.1 this was 10 percent of full flow for a target coverage of 50 percent) at the start of a test on the basis of a data table developed early in the test program relating the valve position to observed striping. This table was developed for water coverage fraction specified in the test target data matrix. No changes to the valve position were made during a test to compensate for any change in the water coverage flow rate. The test procedures and specifications did not address variations in the water coverage flow rate as a potential problem.

The Westinghouse test engineering group stated that Westinghouse's Containment and Radiological Analysis Group acknowledged the disposition of test deviations such as this and other deviations, including, for example, the lowering of the steam flow from 0.09 kg/sec (0.2 lb/sec) to 0.05 kg/sec (0.12 lb/sec) for test 219.1. As noted in the audit, the disposition of test deviations between the Westinghouse test engineering group and the Westinghouse analysis group were to be reviewed during a future inspection.

During NRC Inspection No. 999900404/97-02, held from November 17 to November 21, 1997, the test acceptance criteria were discussed with appropriate Westinghouse staff. The following four test acceptance criteria were established for the LST:

(1) Data on forcing functions available, (i.e., steam flow rate, fan speed, water flow rates, inlet temperature of steam, water and air). Strict adherence to the specific absolute pressures and flow rates is not necessary but values should be nearly constant as defined in the test matrix.

(2) Data on response variables available, (i.e., condensate flow rates, excess water flow rates, air, water and steam outlet temperatures, vessel pressure, 80 percent of the vessel and fluid temperature instrumentation, and vessel water coverage measurements were taken).

(3) Unplanned excursions must be evaluated on a case-by-case basis. Failures that may result in faulty data outputs are not acceptable.

(4) The vessel pressure is maintained within specified pressure limits during the constant pressure portions.

Variations in the PCS water coverage flow rate was not considered in the development of the test acceptance criteria. The important criterion was the target coverage area, as specified in the test matrix. Even then, the acceptance criteria did not require strict adherence to the target value, only that a nearly constant value could be determined for a test. With respect to the specific steam flow rate for test 219.1, the test acceptance criteria were followed.

As was identified in the evaluation of the water distribution testing program, the applicability of the water coverage test results from both the water distribution and large-scale tests, as they relate to the input used for the WGOTHIC computer program for the AP600 Safety Analysis Report, was still under staff review. This was identified as DSER Open Item 21.5.8-2. The fluctuations in the water coverage flow rate was identified as part of SDSER Open Item 21.3.8.5-1e. The Zuber-Staub model used for establishing water coverage for use in the WGOTHIC computer program was also under review as discussed in WGOTHIC Computer Program Review Status Report (Section 21.6.5 of the SDSER). The water coverage model is described in Section 7 of WCAP-14407, Revision 3, "WGOTHIC Application to AP600," dated April 1998. The staff's review of this document and the water coverage model is presented in Section 21.6.5. DSER Open Item 21.5.8-2 is closed.

Internal Velocities

The staff expressed concern with the instrumentation provided in the LST to measure the velocity of the atmosphere internal to the containment vessel. This instrumentation was:

- 3 Anemometers - Pacer, Model APT 275 Vane, 0 to 2419 m/min (7935 ft/min)
- 2 Anemometers - Höntzsch Instruments, 0 to 914 m/min (3000 ft/min)

Of these instruments, the three Pacer models failed during testing, and it was explained that they had generally measured the same range of values: 152 cm (5 ft) to 244 cm (8 ft) per second. Furthermore, the Höntszch meters failed during the high-capacity blowdown testing. Since the internal velocity was important to the condensation of steam on the containment wall, the staff believed that more measurements were needed to quantify the mixed-convective flow field. This was an issue that could be of importance in the scaling of results to predict the prototypical containment performance and was identified as DSER Open Item 21.5.8-3. Non-prototypical features of the LST design, for example the isolation of the steam generator subcompartment from the other below deck regions, which could lead to different flow patterns (both forced and natural circulation) and internal velocity fields were identified as part of SDSER Open Item 21.3.8.5-1e.

A QA review was performed for the LST in May 1995 and was documented in NRC Inspection No. 99900404/95-01, letter from R. M. Gallo (NRC) to N. J. Liparulo (WEC) dated August 8, 1995. During this review it was determined that Westinghouse's analysis group's judgment on internal velocities was that the measurements obtained give an indication of local

bulk velocity along the vessel wall which were useful, although not necessary, in the validation of WGOTHIC. It was also known that the instrumentation might not survive the test conditions.

In August 1995, Westinghouse informed the staff that the AP600 DBA evaluation model would no longer be on the basis of the use of the mixed-convection correlation. The analysis would now be based on a bounding, conservative approach and the use of free convection for mass and heat transfer inside containment. A multiplier is to be applied to the correlations on the basis of the experimental database to ensure that the correlation bounds the data, letter NTD-NRC-95-4570, "Bases for AP600 PCS Mass Transfer Correlation Biases," dated September 28, 1995.

Since the August 1995 change in the AP600 DBA evaluation model, which uses free convection correlations for the mass and heat transfer on the vessel interior wall, the need to accurately predict the interior velocities for validation of the WGOTHIC computer program is not a significant concern. The concern remained that the LST provided sufficient information to assist in the development of a conservative, bounding analysis, as identified in SDSER Open Item 21.3.8.5-1b and SDSER Open Item 21.3.8.5-1e. As such, DSER Open Item 21.5.8-3 remained open until the scaling and PIRT issues were resolved. The scaling analysis has been completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. The PIRT has been completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The staff review of the revised scaling analysis as it relates to validation of the revised PIRT and to the development of the bounding analysis (the evaluation model), and its use to support the design certification is presented in Section 21.6.5. Therefore, DSER Open Item 21.5.8-3 is closed.

Condensate Formation Inside Containment

Another area of staff concern was that of the measurement of condensate forming within the containment vessel. There were five regions from which condensate was collected (1) girder, (2) inside vessel sidewall, (3) rain (collected on a lower horizontal plane), (4) inside vessel (below deck) and (5) a simulated steam generator section. While these were intended to be measured individually, their flows were combined into two measuring tanks only, thus losing the identification of their sources. The measurement of condensate forming within the containment vessel must be considered when addressing the question of the ability of the WGOTHIC computer program to predict containment performance. This was identified as DSER Open Item 21.5.8-4.

The LST final data report WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," dated July 1994, summarized the condensation collection and data. (WCAP-14135 was revised in April 1997 however, the summary of the condensation collection data did not change.) Condensation collection for the Phase 2 tests directed the condensate from the regions below the simulated operating deck going to one collection system and the remaining condensate, from the three locations above the simulated operating deck to the second collection system. Phase 3 tests 222.1, 222.2, 222.3 and 222.4 were used to obtain additional insights into the condensation. The condensate discharges were re-aligned during various steady-state portions of the tests. In test 222.1, it was found that about 3 to 4 percent of the condensate was from the combined rainfall and regions below deck with the remainder

evenly divided between the dome and vessel sidewall. It was determined that 61 percent of the condensate collected on the dome in test 222.2 during the final collection period. 56 to 58 percent of the condensate was generated on the sidewalls in the last two steady-state portions of test 222.3. In test 222.3 is was noted that erratic steam flow lead to a 18- to 26-percent difference between the condensate and average steam flow measurements during the first three periods of this test. Finally, in test 222.4, the Westinghouse test engineering group provided a caution on the use of the condensate distribution data. During two collection periods, the mismatch between the steam flow and condensate flow showed 18 percent more condensate flow than steam flow.

In letter NTD-NRC-95-4463, dated May 15, 1995, Westinghouse submitted "AP600 Testing Program Report: Large-Scale Test Data Evaluation (PCS-T2R-050)." There were no conclusions presented on the condensate collection or measurements in this report.

In WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995, the comparison of the total measured to predicted condensate flow rates were provided for the priority test analyzed with the distributed parameter model. No data comparisons were provided for the individual measurements for test 222.4, the only Phase 3 test which isolated the condensate flows. In the LST data report, WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," dated July 1994 there was an assessment of the overall data quality that provided a rough comparison of the system heat balance as evaluated using three methods:

- condensate mass flow rate
- external heat loss (water, air and radiant)
- heat flux across the vessel wall

The comparison suggested that the condensate mass flow rate overestimates the heat removal when compared to the other two methods. This was a topic of discussion at the December 6 and 7, 1995, meeting with the staff, as documented in "Summary of AP600 Design Review Meeting Regarding the Passive Containment Cooling System and WGOTHIC Compute Code," letter from D.T. Jackson (NRC) to Westinghouse Electric Corporation, December 27, 1995, Docket No. 52-003. Westinghouse indicated that the heat balance analysis would be reviewed to address the concern. The staff considered this commitment to be part of DSER Open Item 21.5.8-4.

In letter NSD-NRC-97-5299, dated August 29, 1997, Westinghouse provided the response to this concern. The referenced heat balances were only prepared to provide a comparison of the test data. Review of the data in Table 3.3-1 of WCAP-14135 indicated that the estimate of the heat balance from the condensate was lower than the estimates from the other calculational methods. Each calculation was on the basis of certain assumptions and simplifications:

(1) Condensate - Relies on the accuracy of the condensate measurements and assumes that there is a negligible heat loss between the vessel and the measurement location of the condensate temperature. It assumes that there is no holdup of condensate within the vessel. This is seen to be the most accurate of the heat balances.

(2) External heat loss - This calculation relies on the accuracy of the water flow on and off the vessel. The uniformity of the water and air temperature, the exit air velocity measurement, and the estimate of heat loss through the baffle assembly are all simplification used in obtaining the estimate from this heat balance.

(3) Heat Flux - This calculation assumes that the water coverage is constant from the top of the vessel to the bottom outside gutter. It applies the minimum and maximum temperatures assumed over an entire area, centered at the thermocouple elevations to the estimated wet and dry areas, respectively.

Numerous attempts were made during the course of testing to measure rainfall within the test vessel in addition to the events detailed in the test report. In no case was any rainfall measured and conditions only resulted in a backup of condensate within the vessel at high steam flows because of the relative capacity of each system.

The heat balance was performed to assess the overall quality of the LST. The condensate measurement-based heat balance was used as the reference for the comparison of the various methods and was judged by Westinghouse as the best measure. No appreciable rainfall was identified during any test. DSER Open Item 21.5.8-4 is therefore closed, as the heat balance analyses were only used to judge the overall performance of the test.

Air Annulus Instrumentation

Lack of adequate instrumentation in the annular region was a concern since initial results obtained by the staff using the CONTAIN computer program show that evaporation from the exterior liquid film controlled the performance of LST 212.1. The lack of adequate instrumentation in the LST annular region made it difficult to evaluate the evaporation rate from the containment shell. This was DSER Open Item 21.5.8-5.

In WCAP-14382, "WGOTHIC Code Description and Validation," dated May 1995, the comparison of the measured to predicted external excess flow rates were provided for the priority test analyzed with the distributed parameter model. The external excess water was collected in an external gutter mounted on the LST vessel and measured with a flow meter. No uncertainty bands were provided on the plots for the measured values. The fluctuating PCS water flow rate is seen in the data, adding an additional level of complexity to the interpretation of these data.

In letter NTD-NRC-95-4463, dated May 15, 1995, Westinghouse submitted "AP600 Testing Program Report: Large-Scale Test Data Evaluation (PCS-T2R-050)." Westinghouse's evaluation of the large-scale PCS test data concluded that evaporation was the primary mode of heat removal from the outside of the vessel (approximately 75 percent of the total), followed by sensible heating of the subcooled liquid film (approximately 17 percent of the total). The remainder of the heat was transferred to the environment by convection and radiation. DSER Open Item 21.5.8-5 was closed, as Westinghouse had provided the requested summary evaluation report.

Air Annulus Design

It was also noted that the LST model did not include a downcomer region, and used a chimney-installed fan to model circulation in the annular region, both of which were non-prototypical of the AP600. Assessment of the impact of the non-prototypicalities was identified as DSER Open Item 21.5.8-6. The scaling analysis has been completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. Distortions and non-prototypicalities of the LST were addressed in that report. The PIRT has been completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The development of the evaluation model, including the treatment of distortions and non-prototypicalities of the LST were addressed in that report. The staff's review of the revised scaling analysis as it relates to validation of the revised PIRT and to the development of the bounding analysis (the evaluation model), and its use to support the design certification, is presented in Section 21.6.5. Therefore, DSER Open Item 21.5.8-6 is closed.

Internal Heat Sinks

The modeling of the long- and short-term heat sinks, flow paths, and internal volumes in containment, especially in relation to their representation in WGOTHIC, is discussed in Section 21.6.5.7 of this report. This was identified as DSER Open Item 21.5.8-7. The scaling analysis has been completely revised and is documented in WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998. Distortions and non-prototypicalities of the LST were addressed in that report. The PIRT has been completely revised and is documented in WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for the AP600 Passive Containment Cooling System," dated April 1998. The development of the evaluation model, including the treatment of distortions and non-prototypicalities of the LST were addressed in that report. The staff review of the revised scaling analysis as it relates to validation of the revised PIRT and to the development of the bounding analysis (the evaluation model), and its use to support the design certification, is presented in Section 21.6.5. Therefore, DSER Open Item 21.5.8-6 is closed.

With regard to scaling, the LST facility and data needed to be linked to the AP600 prototype, and the effects of concrete as a heat sink and heat rejection through the shell needed to be addressed. The staff recognized that Westinghouse was still in the process of completing its scaling efforts. Resolution of the LST scaling and PIRT issue had been previously identified as DSER Open Item 21.3.8.5-1. The LST scaling and PIRT are discussed in Section 21.6.5.5 of this report.

21.5.9 Water Distribution Test Program

In the DSER and in Section 21.5.8 of this report, the staff identified the treatment of uncertainty in the water distribution test data as part of DSER Open Item 21.5.8-2. Measurements of water "striping" (distinct strips of water flow) were made at steady state (approximately 10 minutes after initiation of flow) by measuring the width of dry and wet patches on the test model with a ruler. Westinghouse counted an area as wet only if active flow could be seen, and considered damp areas dry since they most likely would be if the shell were heated.

The method of measurement and the assumptions made concerning wet and damp areas could result in significant uncertainty in the experimental results. Since water distribution test data was used to validate the coverage model used as input to WGOTHIC, uncertainty in the coverage model could result in errors in WGOTHIC predictions of AP600 DBA containment response. Although Westinghouse's treatment of wet and damp areas appeared to be conservative, this conservatism would only be acceptable if WGOTHIC were a validated code (i.e., confidence in the code's predictive capabilities exists and any conservatism in code input parameters (such as coverage fractions) would add to the conservatism of the code's predictions). The use of the water distribution test data in the development of the water coverage model, as part of the limited PCS flow model, used in WGOTHIC for AP600 licensing analyses in support of design certification is discussed in Section 21.6.5.

In the DSER, the staff expressed concern over the validity of the coverage fractions used in the SSAR, Revision 2, WGOTHIC analysis at times much later than the 11 minute delay assumed in the analysis. In SSAR, Revision 2, the coverage model assumed a 40-percent, 60-percent, 70-percent water distribution (i.e., 40-percent on the center of the containment dome, 60-percent on the region between the first and second rings of weirs, and 70-percent on the sidewall) and further assumed that water coverage did not exist until 11 minutes into the transient, allowing time to initiate the signal and to fill the headers and weirs. The coverage fractions were then kept constant for the duration of the analysis. The validity of the water coverage at much later times following a DBA was identified as DSER Open Item 21.5.9-1. The use of the water distribution test data in the development of the water coverage model, as part of the limited PCS flow model, used in WGOTHIC for AP600 licensing analyses in support of design certification is discussed in Section 21.6.5. The initial PCS flow rate has been increased to 1,660 liters/minute (about 440 gpm), from 830 liters/minute (220 gpm), and the delay time is now taken to be 337 seconds. The limited PCS flow model now considers the changing wetted area coverage as the PCS flow rate decreases over time.

DSER Open Item 21.5.9-2 involved the issue that coverage fractions obtained from the water distribution tests may not be representative of actual DBA conditions, since the tests were conducted with an unheated surface. Under actual DBA conditions, the vessel shell is heated because of the energy release from the accident possibly resulting in different coverage fractions than those used in the SSAR DBA analysis. While the LST employed a heated surface, the fact that the tests were not full scale and that considerable fluctuations were present in the water supply both tend to question the credibility of the data. The use of the water distribution test data in the development of the water coverage model, as part of the limited PCS flow model, used in WGOTHIC for AP600 licensing analyses in support of design certification is discussed in Section 21.6.5.

In the DSER, the staff also expressed its concern that the supporting arms of the baffle wall of the PCS, surface irregularities, and the possible effect of weir clogging with foreign material were not accounted for in the WGOTHIC analysis. This was identified as DSER Open Item 21.5.9-3.

In document PCS-GSR-003, " A Method for Determining Film Flow Coverage for the AP600 PCS," dated July 1994 Westinghouse provided a model for determining coverage values to be used in the WGOTHIC analysis. The model, developed by Zuber and Straub approximately 30 years ago ("Stability of Dry Patches Forming in Liquid Films Flowing Over Heated Surfaces," International Journal of Heat and Mass Transfer, Vol. 9, 1966) included terms which account for

the film momentum, surface tension, thermocapillary effects, film potential energy, and static pressure (added by Westinghouse). The Westinghouse adaptation of the model attempted to mechanistically predict the onset of flow instability, a phenomenon which could result in redistribution of the water film on the surface. To apply the model to the AP600, Westinghouse introduced an arbitrary parameter, referred to as the reference stability margin R_{ref}, to determine the onset of flow instability and to account for surface irregularities. The model was applied to four water distribution tests and all of the LST tests to yield a value of R_{ref} that predicted the tests. To determine coverage fractions for input to WGOTHIC, Westinghouse used expected initial PCS flow rates and shell heat flux values obtained from WGOTHIC (assuming a 40-60-70 percent coverage distribution) as input to the coverage model.

According to the model predictions, Westinghouse concluded that coverage fraction input to the WGOTHIC model should be higher in the dome region and lower on the sidewall. Letter NTD-NRC-94-4286, "Supplemental Information on AP600 PCS Film Flow Coverage Methodology," dated August 31, 1994, addressed questions that were raised about the model, and forwarded the results of sensitivity studies which showed that coverage can be reduced by a factor of two without reaching the design pressure if the value of R_{ref} is appropriately adjusted. The coverage values used in the SSAR (Revision 5) model were the same as those presented in report PCS-GSR-003, with the exception of the upper and middle dome where lower coverage fractions were used.

Report NSD-NRC-96-4646, "Conservatism in Modeling of the PCS Film in the DBA Evaluation Model and Comparison of the Range of Film Parameters in the PCS Test Data with AP600," dated February 15, 1996, addressed conservative assumptions used in the modeling of the evaporating film in WGOTHIC. In particular, the following issues were covered in the report:

- the conservatism of waiting 660 seconds following a DBA before applying PCS water to the shell
- a comparison of the range of parameters of the water distribution and other tests used to validate the coverage model to those expected in the AP600
- the method used to determine a conservative, bounding value of the parameter R_{ref} used in the coverage model
- the method used to determine a conservative, bounding minimum PCS flow rate for the DBA evaluation model

The staff reviewed document NSD-NRC-96-4646 in the context of the Open Items mentioned in this section. The staff recognized that the information contained in document NSD-NRC-96-4646 may have addressed some or all of the items, and was aware of Westinghouse's August 1995 decision to perform a bounding analysis of the AP600 containment response in lieu of the best estimate approach of earlier efforts. Given that context, the staff reviewed the Open Items and the information provided by Westinghouse paying particular

attention to Westinghouse's conservative approach. The following list summarized the Open Items of this section and the minimum information required for progress toward their closure:

- DSER Open Item 21.5.8-2 – The acceptability of the water distribution, LST, and other test data for validation of the coverage model used in the AP600 DBA analysis remained a concern to the staff. Westinghouse needed to show that experimental uncertainty in the data from these tests does not invalidate their use for determining coverage fractions, or should show that sufficient conservatism exists in the coverage model such that uncertainties in the data do not invalidate the model's use. The current licensing model, including the limited PCS flow model, and the use of the water distribution test, the LST and other test data, is discussed in Section 21.6.5. DSER Open Item 21.5.8-2 closed.

- DSER Open Item 21.5.9-1 – With regard to the WGOTHIC DBA analysis, Westinghouse needed to demonstrate the validity and conservatism, if applicable, of using the same coverage fractions for all times and of the 660-second time delay before initiation of PCS flow in the WGOTHIC DBA analysis. The initial PCS flow rate has been increased from 830 liters/minute (220 gpm) to 1,660 liters/minute (about 440 gpm) and the delay time to credit PCS is 337 seconds in the current model. The limited PCS flow model, as discussed in Section 21.6.5, considers the changing coverage fractions as the PCS water flow rate is decreased over time. DSER Open Item 21.5.9-1 is therefore closed.

- DSER Open Item 21.5.9-2 – Westinghouse needed to demonstrate how water coverage data obtained from an unheated surface is extrapolated to represent film behavior on the AP600. The development of the limited PCS flow model, as discussed in Section 21.6.5, considers the LST as well as other heated separate effects experiments in evaluating the expected film behavior on the AP600. DSER Open Item 21.5.9-2 is closed.

- DSER Open Item 21.5.9-3 – The staff was interested in understanding if and how the baffle wall standoffs were treated in the coverage model. These seem to have a destabilizing effect on the film flow and may change the stability margin, R_{ref}. The staff was also interested in understanding whether the degree of conservatism present in the coverage fractions includes a reduction in PCS flow because of clogging of weirs with debris, a situation which could be postulated to occur during the course of plant operation. Baffle standoffs were modeled in the water distribution test and in the LST. Therefore, Westinghouse concluded that the effects of the baffle standoffs were included in the basis for the limited PCS flow model. Westinghouse further concludes that the design of the AP600 in combination with an appropriate surveillance program, as part of the inservice inspection program, minimizes the possible of clogging the weirs. Periodic inspections of the PCS air flow path from the shield building annulus inlet to the exit ensures that it is unobstructed, the baffle plates are properly installed, and the upper annulus safety related drains are unobstructed. As specified in technical specification 3.6.6, these inspections are preformed every 24 months. The limited PCS flow model, as discussed in Section 21.6.5, includes a conservative value for R_{ref} and for the PCS flow credited when compared to the actual available flow. Therefore, DSER Open Item 21.5.9-3 is closed.

21.5.10 AP600 Scaling Issues

During the AP600 design certification review, the staff reviewed the scaling reports for the individual testing programs, as discussed in Sections 21.5.1, 21.5.2, 21.5.3, 21.5.5, and 21.5.6. Furthermore, the evaluation of the test programs and associated data included a review of the PIRTs developed for the AP600. The PIRTs identify and classify, in importance, the thermal-hydraulic phenomena that must be included in Westinghouse's analytical models, and for which data must, therefore, be available to evaluate those analytical models. As a result of these assessments, the staff determined that an overall "closure" document was needed to integrate Westinghouse's test results, demonstrating that the test data cover an appropriate thermal-hydraulic range, address the pertinent phenomena identified in the PIRTs, and are consistent with assumptions made in deriving the scaling parameters used to design the test facilities and develop the test matrices. Accordingly, the staff issued RAI 440.566, requesting that Westinghouse provide such a closure document. This was identified as SDSER Open Item 21.5.10.1-1.

Westinghouse submitted Revision 0 of "AP600 Scaling and PIRT Closure Report," WCAP-14727, in October 1996. The staff reviewed this report, and raised a number of issues and concerns requiring additional information for resolution. The staff and Westinghouse also met with the ACRS to discuss the staff's initial review of WCAP-14727. The ACRS independently asked several questions similar to those of the staff, and also identified new issues and concerns that Westinghouse was requested to resolve.

Westinghouse issued Revision 1 of WCAP-14727 in July 1997. The report was substantially revised to incorporate the answers to the staff's and the ACRS's questions. In addition, Westinghouse included several appendices with the report. Appendix A contains the responses to all of the staff's and the ACRS's questions on WCAP-14727, Revision 0 and many other RAIs concerning various aspects of the individual design certification test programs. Following additional questions from the staff and the ACRS, as detailed below, Westinghouse issued Revision 2 of WCAP-14727 in February 1998.

In WCAP-14727, Revision 1, Westinghouse described the scaling bases for the design certification test facilities, and used the equations of mass, momentum, and energy conservation to derive dimensionless groups (called "pi" groups). These "pi" groups consist of ratios of fluid properties (e.g., density, enthalpy), system physical characteristics (e.g., pipe diameters), and thermal-hydraulic characteristics (e.g., pressure, temperature, velocity), and are used to (1) determine the relative importance of associated phenomena and system parameters, and (2) provide a basis for comparison of the range of parametric coverage of the test facilities, compared to the AP600 plant. Good agreement between a "pi" group value and the AP600 value for the same "pi" group (within a factor of about 2) indicates acceptable scaling of the phenomena or parameter represented by the "pi" group. The values of the "pi" groups were evaluated for the test facilities in two ways (1) using values on the basis of actual test data, and (2) by "hand calculations," using thermal-hydraulic models to derive approximate "steady-state" values of fluid properties and thermal-hydraulic parameters. The "pi" group values for the AP600 plant were derived using the "hand calculation" method. The comparison of the "pi" group numerical values for the test facilities and the AP600 was on the basis of the "hand calculation" values.

The staff reviewed WCAP-14727, Revision 1, including Westinghouse's responses to staff and ACRS questions. The staff determined that most of the questions were acceptably addressed. However, the staff concluded that additional discussion was needed in the report to explain the differences between the "data"-based and "hand calculation"-based values of the "pi" groups. These values often differ by up to an order of magnitude, and use of the "pi" values on the basis of data could give somewhat different results in comparing the response of the test facilities to that of the AP600. The staff understands that the hand-calculated values of the "pi" groups were used to keep the AP600 on a consistent basis with the test facility (since there are no "data" values, aside from nominal geometric parameters) for the actual plant. However, the "hand calculation" method requires the use of simplified models (e.g., two-phase flow and pressure drop), which may not reflect actual facility (or plant) behavior. One of the staff's primary objectives in requesting the "closure" report was to evaluate the actual data produced by the facilities in terms of consistency with the importance (rank) assigned to local and system phenomena in the PIRTs. The staff requested that Westinghouse provide additional discussion of the data-related values of dimensionless groups, both in comparison to the "hand-calculated values" and in relation to the ranking in the PIRTs. Westinghouse provided additional information on this issue in a letter dated February 13, 1998, and also incorporated new information in WCAP-14727, Revision 2. The "hand-calculated" values of the dimensionless groups are on the basis of parameters derived from simplified models of the test facilities. Some assumptions are required for these models to be used. For example, parameters related to SBLOCAs generally ignored the flow from the break and assumed quasi-steady (or slowly changing) conditions. In contrast, some data-based values were derived during rapidly changing conditions and may have been influenced to some extent by parameters neglected in the simplified models. Thus, the disagreements between hand-calculated values and data-based values are greatest during the initial part of the transient, when the break flow may be relatively large (before ADS actuation, which overwhelms the flow from the break) and facility conditions are changing rapidly. The values derived for the AP600 plant, for comparison to the test facilities, were also derived using simplified models, with assumptions similar to those used for the test facilities. There is an implied assumption of similarity of those parameters that were neglected in the simplified facility and plant models. However, this is a reasonable assumption, since these parameters, such as break flow, comprise boundary conditions that were established using the facility scaling information. In view of the above, the staff finds that Westinghouse's response to this open item is acceptable; and therefore, SDSER Open Item 21.5.10.1-1 is closed.

The scope, organization, and content of WCAP-14727 was also discussed at the December 9 and 10, 1997, meeting between the staff, Westinghouse, and the ACRS Subcommittee on Thermal-Hydraulic Phenomena (see Section 21.5.2). As a result of ACRS questions and comments at this meeting, Westinghouse was requested to revise the report to address the following ACRS comments:

- correction of errors in equations
- correction (if necessary), complete discussion, and justification of assumptions and/or specific models used in data assessment, determination of scaling parameters, and "pi" group evaluations
- in the specific case of OSU/APEX scaling, performance of a multi-loop scaling analysis, or an alternative quantitative assessment to augment the single-loop analysis discussed

in Revision 1 of WCAP-14727 and, to demonstrate that the facility is appropriately scaled for the thermal-hydraulic phenomena and system behavior occurring during the transition from the end of the ADS blowdown period to the inception of IRWST injection (e.g., flow split between ADS-1/2/3 and ADS-4 valves, pressurizer draining behavior, counter-current flow limitation in the pressurizer surge line, overall system depressurization behavior), to provide confidence that these data can be used to assess design-basis-accident analysis computer codes used for the AP600 plant

As previously noted, Westinghouse submitted Revision 2 of WCAP-14727, which contained corrections to Revision 1 and new information to address the issues discussed above. Revision 2 also contains information provided by Westinghouse to close SDSER Open Item 21.5.10.1-1 and incorporate, in Appendix A, responses to several RAIs. The most significant new information in Revision 2 of WCAP-14727 was a multi-loop scaling analysis of the period in SBLOCAs between the end of ADS blowdown and establishment of stable IRWST injection, during which the minimum mixture level in the reactor vessel is expected to occur. The multi-loop analysis was undertaken in response to these issues to demonstrate that the integral test facilities were well-scaled for this critical period, and that the test data were appropriate for use in validating AP600 computer codes for analyses of the events in question. The staff reviewed Westinghouse's multi-loop analysis, and found that (1) the "pi" group comparison indicates good scaling for this period, and (2) the results were consistent with the previous single-loop analysis. The staff therefore concludes that the multi-loop analyses demonstrates that the AP600 integral facilities are acceptably scaled for this period in an SBLOCA and that the data are appropriate for computer code validation. The staff also reviewed the other changes made to WCAP-14727 and determined that they are acceptable. Therefore, these issues are closed.

21.5.11 Compliance With 10 CFR 52.47(b)(2)

The three requirements for design certification testing and analysis programs, as stated in the Introduction to this chapter, can be summarized as demonstration of

(1) performance of each safety feature

(2) effects of systems interactions and

(3) existence of an adequate database for code validation

For the reactor systems, Westinghouse has completed a test and analysis program to address each of these requirements. Separate-effects tests address the performance of AP600 safety systems, including CMTs, ADS, and PRHR. Additional separate-effects tests have been performed to extend the range of DNB correlations in the AP600 analysis codes, to comply with the last of the three requirements above. Integral systems tests also produce data on performance of these safety systems, with the addition of both accumulator (compressed-gas-driven) injection and gravity-drain safety injection from the CMTs and IRWST. The integral systems tests also produced data on the effects of systems interactions. Demonstration of the existence of an adequate database for code validation was the objective of the code qualification effort and has been verified by the staff's acceptable conclusions regarding the code qualification effort.

Compliance of the containment systems testing program with the requirements of 10 CFR 52.47(b)(2) are addressed in Section 21.6.5.3.3 of this report.

On the basis of the specific evaluations provided in the various sections of Chapter 21 of this report, the staff has concluded that the individual test programs and the code qualification program are acceptable. The staff reviews of test facility designs, instrumentation, scaling, and test matrices in the AP600 testing and analysis programs concluded that the requirements of 10 CFR 52.47(b)(2) have been met.

21.6 Code Development and Qualification Efforts

Westinghouse has specified three reactor systems computer codes and one containment system computer code for use in performing AP600 safety analyses. The codes are as follows:

- LOFTRAN/LOFTTR2, for non-LOCA transients, such as SGTRs and main steamline breaks
- NOTRUMP, for small-break LOCAs
- WCOBRA/TRAC, for large-break LOCAs
- WCOBRA/TRAC, for long-term cooling
- WGOTHIC, for containment systems performance

Five generic open items were identified in the DSER relating to the status of the code review at the time the DSER was issued:

(1) DSER Open Item 21.6-1 stated that Westinghouse was expected to submit details of the NOTRUMP validation by the end of 1994. Westinghouse has submitted a substantial amount of information and documentation on NOTRUMP since the DSER was issued. Because the staff reached an acceptable conclusion regarding the application of NOTRUMP for AP600 analyses as detailed in Section 21.6.2 of this report, DSER Open Item 21.6-1 is closed.

(2) Westinghouse was expected to submit details of the LOFTRAN validation by the end of 1994 per DSER Open Item 21.6-2. This open item has been subsumed by the new LOFTRAN evaluation discussed in Section 21.6.1 of this report. Therefore, DSER Open Item 21.6-2 is closed.

(3) DSER Open Item 21.6-3 requested Westinghouse to submit all remaining documentation on WCOBRA/TRAC by the end of 1994. Westinghouse has now submitted all documentation for the WCOBRA/TRAC base code "Best Estimate" methodology along with its application to AP600 as discussed in Section 21.6.3 of this report. Therefore, DSER Open Item 21.6-3 is closed.

(4) The staff stated in the DSER that it was not aware of any specific reactor systems analyses connected to the design certification test programs using the AP600 computer codes. Subsequent to issuance of the DSER, Westinghouse has performed test

program analyses using the AP600 computer codes. These analyses have been provided in the final verification and validation reports for LOFTRAN and NOTRUMP, and in the AP600 code applicability document for WCOBRA/TRAC. DSER Open Item 21.6-4 requested Westinghouse to provide information on the validation of AP600 analyses codes as part of the test program reports. The staff now acknowledges that this information was provided by the code verification and validation and applicability reports and that the staff's final evaluation provided in Section 21.6 of this report subsume this concern. Therefore, DSER Open Item 21.6-4 is closed.

(5) Data from the wind tunnel tests is used to evaluate the boundary conditions input to WGOTHIC for the AP600 containment analysis, which is discussed in Section 21.5.7 of this report. No predictions between the wind tunnel test data and any computer code analysis will be made. The use of the data from these tests as input to the AP600 safety analysis is acceptable. The staff reviewed Westinghouse's methodology for assessing the wind-neutrality of the AP600 containment and finds both the conclusion that no imposed wind conditions is conservative for the AP600 containment analysis and the decision to use this methodology in design basis analyses is acceptable. Therefore, Open Item 21.6-5 is closed.

Data from the water distribution tests will be used to determine the water coverage fractions provided as input to the WGOTHIC code for analysis of the AP600 containment, which is discussed in Section 6.2 of this report. There has been no attempt to match specific test data and any computer code predictions. The validity of the coverage values assumed in the SSAR WGOTHIC analysis is discussed in Section 21.6.5.

Data from the LST facility has been used to validate predictions made by the WGOTHIC code. The containment analytical model is discussed in Section 6.2 of this report. The staff's review and final evaluation of WGOTHIC is discussed in Section 21.6.5 of this report. Therefore, DSER Open Item 21.6-6 is closed

The staff's assessment of each computer code is provided as follows.

21.6.1 LOFTRAN/LOFTTR2 Computer Code for non-LOCA Transients

Westinghouse has modified its LOFTRAN computer code for use in AP600 transient and accident analysis. The staff has reviewed the application of LOFTRAN methodology to the AP600, as discussed below.

21.6.1.1 Background

The "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800 (SRP), identifies numerous transient and upset conditions anticipated to occur in an operating nuclear power plant. The effects of each of the transients and accidents must be analyzed for a reactor design to ensure that the nuclear reactor does not exceed limiting conditions for each class of accident or transient. The different accidents and transients are classified with respect to the anticipated frequency of occurrence and severity of result permitted.

Generally, computer codes used for safety analyses are demonstrated to conservatively predict important transient phenomena to ensure plant safety limits are satisfied. Such demonstrations require an analysis and review of the construction of the computer code and its comparison to experimental data. The computer code should have a faithful representation of the plant insofar as possible, with a minimum of engineering judgement used to interpret plant phenomena. The computer code should also demonstrate the correct or conservative trends of an analysis when viewed from a licensing perspective. This can be done by comparing the code to experimental data whenever possible. Such comparisons demonstrate the conservatism and fidelity of the approach. The comparisons may also indicate any weaknesses or shortcomings that need to be addressed or compensated for.

Typically, different computer codes must be used to simulate the various classes of upset conditions. One set of codes will be used for neutron kinetics-based excursions, another set for non-LOCAs, and another for accidents resulting in major loss of coolant inventory with accompanying two-phase flow in the reactor coolant system.

Transient analyses performed with LOFTRAN are conducted in conjunction with the appropriate support code, such as FACTRAN for detailed fuel and heat flux modeling, and THINC or WESTAR for calculating the DNBR. These supporting codes have been previously reviewed and approved and are applicable to the AP600 passive reactor design because the AP600 design conditions fall within the codes' range of validity.

These component models have not been added to the flow and pressure network solution implicitly in LOFTRAN, but rather, they communicate through explicit boundary connections to the code. Thus, these models need to be reviewed and checked for their stability and accuracy in regard to their interaction with the main LOFTRAN numerics, which include implicit as well as explicit differencing, along with numerical analysis techniques. Additionally, their comparison to experimental data is extremely important in assessing the validity of the complete system model.

21.6.1.2 Summary of LOFTRAN and the Topical Reports

Westinghouse submitted LOFTRAN to the NRC by Letter NS-SL-453 (October 11, 1972). LOFTRAN is a thermal-hydraulic computer program developed for the analysis of operating reactor transient events discussed in Chapter 15 of the SSAR for which there is no loss of reactor coolant system inventory. The transients for which LOFTRAN is applicable are identified in the PIRT, Table 21.6-1, discussed below. LOFTRAN is designed to simulate the behavior of a multiloop PWR system by modeling the reactor core, the reactor pressure vessel, the hot- and cold-leg piping, the steam generators, the pressurizer, and the reactor coolant pumps. LOFTRAN was subsequently modified to improve the steam generator secondary side model, add a break flow model to represent a tube rupture, and improve the logic system to represent operator actions. This enhanced, or specialized, version of the code is referred to as LOFTTR2. LOFTRAN and LOFTTR2 are collectively referred to as LOFTRAN, unless a specific distinction is made elsewhere in this report. The staff approved LOFTRAN for licensing basis transient analyses in a letter to Westinghouse dated July 29, 1983.

Westinghouse confirmed the applicability of LOFTRAN to the AP600 passive plant design in WCAP-14234 ("LOFTRAN & LOFTTR2 AP600 Applicability Document"). Assessments of the component models added to LOFTRAN were submitted through preliminary verification and validation (PV&V) reports. The final assessment of the applicability of LOFTRAN to the AP600

plant design was submitted in the final verification and validation (FV&V) report, WCAP-14307 ("AP600 LOFTRAN-AP and LOFTTR2-AP Final Verification and Validation Report"). The staff issued numerous requests for additional information (RAIs) on the material in the PV&V reports and the FV&V report. The Westinghouse responses to the staff RAIs have been reviewed and considered in the preparation of this safety evaluation report.

21.6.1.3 Phenomena Identification and Ranking Table

It is important to identify all physical phenomena that will occur in the AP600 under the accident conditions of interest to ensure that the important physical processes and phenomena have been modeled. One method of identification is through the development of a phenomena identification and ranking table (PIRT). The PIRT methodology provides a framework where physical processes and phenomena in a specific hardware geometry under anticipated accident sequences are first identified and then ranked in terms of their importance to the course of the analysis. A PIRT is generally developed from expert opinions provided by a group of knowledgeable analysts. The use of a group of experts, rather than a single analyst, increases the chances that all important phenomena have been identified and included in the PIRT, and that the rankings have accurately characterized each specific phenomena as being of high, medium, or low importance to the integral quantities of interest. A properly established PIRT will serve as a road map through a transient, identifying and ranking the important phenomena and functions necessary to predict and deal with each phase of a transient. The PIRT for the AP600 non-LOCA transients submitted in WCAP-14234 is shown in Table 21.6-1. The Westinghouse PIRT agrees with the NRC PIRT, developed as part of the review and confirmation process, for the phenomena of importance in each transient. Accordingly, the staff finds the PIRT developed for the accidents and transients for which LOFTRAN is used to be applicable and acceptable.

Differences exist between the Westinghouse PIRT and the NRC PIRT. The Westinghouse PIRT is much more extensive in the depth of coverage of the non-LOCA transients. The staff finds the Westinghouse PIRT acceptable for non-LOCA transients.

Review of the SGTR transient shows that the NRC PIRT ranks the importance of upper head flashing as medium (M) while the Westinghouse PIRT ranks the importance of this phenomenon as low (L). The staff notes that calculations indicate that the upper plenum of the vessel stays subcooled with up to ten ruptured tubes, which is beyond the design-basis analysis. Therefore, this difference in the PIRTs is acceptable.

The NRC PIRT ranks the importance of balance line initial temperature distribution as medium (M), whereas the Westinghouse importance ranking is low (L). The code explicitly calculates the initial temperature distribution; therefore, the difference in ranking does not affect the analysis and is acceptable.

21.6.1.4 Evaluation of the LOFTRAN Analytical Models

The analytical models in the approved LOFTRAN have not been altered or modified for application of LOFTRAN to the AP600 passive reactor design. LOFTRAN, as approved, models the reactor primary and secondary coolant systems. The primary coolant system is modeled as a reactor core and vessel, hot- and cold-leg piping, pressurizer, and the reactor coolant pumps.

The secondary coolant system is modeled as a steam generator, tube and shell sides, with main steamline and feedwater line piping.

The reactor core kinetics model is a simple point neutron kinetics model, using six delayed neutron groups, capable of calculating reactor feedback effects and scram reactivity. The fuel model permits calculation of decay heat, matching the ANS Standard (1971) + 20-percent decay heat curve, and DNBR.

The reactor coolant thermal-hydraulic system is represented as a nodal model specified by the user. The momentum equation is solved for flow, friction pressure losses, elevation heads, pump head, and fluid momentum. The reactor coolant pump homologous curves are input by the code user. The pressurizer model treats the pressurizer as a two-region (steam and water) node. A variable volume model is used to simulate the potentially varying level during a transient.

The primary side of the steam generator is treated as a multiple node system, while the secondary side is treated as a single node with a saturated steam and water mixture. The steam and feedwater lines can be treated as isolated, or failed to isolate, or in the case of a break, the Moody correlation can be used to compute the break flow.

Major control systems are also included to simulate the response of the control systems and operator actions in transients and accidents.

During its review, the staff requested additional information on the applicability of several of the phenomenological models in the approved version of LOFTRAN to the thermal-hydraulic conditions anticipated in the AP600 design. The staff concerns included the pressurizer location, wall friction, global pressure location, compressibility effects, reverse flow, and heat transfer options. When the SDSER was issued, Westinghouse had not yet submitted responses to all the staff's RAIs. The staff could not complete its determination of the adequacy of the analytical models in LOFTRAN for application to the AP600 passive reactor design until it received and reviewed responses to the outstanding RAIs. Submittal of outstanding RAI responses was SDSER Open Item 21.6.1.4-1. Westinghouse has now submitted all outstanding RAI responses related to LOFTRAN. The staff has completed its review of these responses and found them to be technically complete and sound. Therefore, SDSER Open Item 21.6.1.4-1 is closed.

21.6.1.5 Evaluation of the LOFTRAN AP600 Component Models

Hardware component-specific models were added to the approved version of LOFTRAN to represent the AP600-unique hardware features. The following component models were added:

- automatic depressurization system (ADS)
- core makeup tank (CMT)
- passive residual heat removal heat exchanger (PRHRHX)
- in-containment refueling water storage tank (IRWST)

21.6.1.5.1 Automatic Depressurization System

The ADS is designed to depressurize the reactor coolant system (RCS) to values near the prevailing containment pressure to enable gravity injection from the IRWST. Three stages of the ADS come off the top of the pressurizer; the fourth-stage ADS paths are connected to the hot legs. The first-stage ADS is actuated when 33 percent of the liquid in a CMT has drained, resulting in the initiation of depressurization of the plant via the ADS valves to the IRWST. The second- and third-stage ADS valves open according to timers that are started with the actuation of the first stage. If the CMTs continue to drain, the fourth-stage ADS valves will actuate when 80 percent of the liquid in a CMT has drained. The fourth-stage ADS valves, located on the hot legs, open directly to the containment to facilitate RCS depressurization to the containment pressure.

Westinghouse notes that the ADS is not required for non-LOCA and SGTR events. Detailed modeling of the ADS is not done except to simulate an inadvertent opening of an ADS stage. The slow opening response of the ADS valves is simulated in LOFTRAN by use of the pressurizer relief valve model, modified to represent a slower opening ADS valve. The relief valve model has been further modified to permit the analyst to specify not only the opening time, but also the valve flow area as a function of time.

21.6.1.5.2 Core Makeup Tank

Two CMTs are connected to the RCS by normally open isolation valves on the cold-leg balance lines and normally closed isolation valves on the CMT discharge lines. The CMTs provide high-pressure gravity-driven borated coolant injection into the RCS to provide reactivity control and core cooling. The CMT discharge valves open on a safety (S) signal and remain open. The tanks are maintained at system pressure by the cold-leg pressure balance lines. During normal operation, the CMTs and the cold-leg balance lines are completely filled with liquid.

The CMT system is represented in LOFTRAN by 15 fluid nodes for the CMTs, 8 nodes for the injection line, and 3 nodes for the balance line between the cold leg and the CMT. Heat transfer from the tank metal walls is modeled by one heat slab per fluid node (a total of 15). Boron is tracked on a node basis in the piping and on a tank average basis in the CMT fluid nodes. Mass and energy are transferred from node to node in the CMT model by use of the LOFTRAN slug flow model.

21.6.1.5.3 Passive Residual Heat Removal Heat Exchanger

The PRHR system is a C-shaped, single-pass, downflow heat exchanger submerged in the IRWST. The system inlet connects to the top of the horizontal hot-leg section containing the pressurizer loop. The system outlet is connected to the bottom of the pressurizer loop steam generator outlet plenum. Normally closed isolation valves open to actuate the system.

The LOFTRAN model of the PRHR can consist of 45 nodes in 5 regions. The PRHR piping from the hot leg is represented by two nodes, the inlet header piping and inlet channel head by a single node, the outlet channel head and outlet header piping by a single node, and the outlet piping to the steam generator outlet plenum by two nodes. The PRHR heat exchanger tubes

are broken down into 22 nodes (5 horizontal nodes at the top of the heat exchanger, 12 vertical nodes, and 5 horizontal nodes at the bottom of the heat exchanger).

21.6.1.5.4 In-containment Refueling Water Storage Tank

The IRWST is a source of water for gravity feed injection into the RCS once RCS pressure has been reduced to values near the containment pressure. The IRWST also serves as a heat sink for the removal of heat via the PRHR system discussed above and as a discharge reservoir for the first three stages of the ADS. Condensation of steam in the containment is a long-term source of water to the IRWST, which can then return to the RCS. Although not part of the IRWST, the containment sump is a second source of gravity fed coolant injection into the RCS over the long term.

The IRWST is modeled in LOFTRAN as a single fluid region. Initial conditions are input, and mass and energy are then tracked in the node as a homogeneous mixture. Once the fluid saturation temperature and pressure are reached, steaming to the containment node is accounted for.

The staff questioned the validity of using a homogeneous, mixed condition in the IRWST when some temperature stratification is likely. The staff was concerned that analysis sensitivity to PRHR could be impacted by stratification in the IRWST. In a letter dated October 2, 1997, Westinghouse provided the results of a sensitivity study comparing the analysis of non-LOCA transients with a temperature stratification in the IRWST (on the basis of data from the SPES and PRHR test programs) with the results that assume a homogeneous IRWST temperature. The study indicates that modeling a homogeneous IRWST produces conservatively low heat transfer from the PRHR and that LOFTRAN provides conservative results for non-LOCA transient analysis.

21.6.1.6 Code Qualification

Qualification and assessment of LOFTRAN and its models has been conducted in two areas (1) separate-effects tests and (2) integral-systems tests. The combination of these tests, when properly applied, leads to overall conclusions regarding the capability of a computer model to adequately predict the behavior of the AP600 when subjected to upset conditions.

21.6.1.6.1 Separate-Effects Tests

Assessment of LOFTRAN against separate-effects tests permits the isolation of individual models in the code in such a way that the capabilities of the model can be determined while remaining within the context of the code.

<u>CMT</u>

Assessment of the LOFTRAN CMT model was conducted by comparing code predictions to the CMT test facility data. Results of the predictions indicate that the LOFTRAN CMT model over-predicts injection flow rate in single-phase natural circulation. The staff has examined, on a case-by-case basis, the Chapter 15 transients analyzed with LOFTRAN to determine conservatism of this model. When the flow is two-phase, a buoyancy head penalty is used

because, as noted previously, LOFTRAN is not designed for significant two-phase flow. Thus, a conservative result is ensured.

PRHR

Assessment of the LOFTRAN PRHR model was conducted by comparing code predictions with PRHR test facility data. The LOFTRAN model simulated the three-tube test configuration, along with appropriate test conditions. On the basis of the PRHR test facility data, Westinghouse modified the heat transfer correlations LOFTRAN uses on the outside of the PRHR tubes. The staff raised concerns regarding the ability of three straight tubes to adequately represent the heat transfer phenomena occurring on the outside of the C-shaped PRHR tubes. Westinghouse performed further analyses of the PRHR heat transfer correlations by calculating the performance of the full height C-tube PRHR heat exchanger used in the ROSA AP600 confirmatory tests. The analyses demonstrated the heat transfer correlation conservatively predicted the heat transfer measured in the experiment. Details of the evaluation of the PRHR test program and applicability of the data are provided in Sections 21.3.3 and 21.5.3 of this report.

Westinghouse notes that the analyst can select "the appropriate PRHR heat transfer option in LOFTRAN" by either over-predicting or under-predicting PRHR heat transfer. The staff had a concern regarding this level of flexibility in the analyst's control of the predicted results. The staff asked Westinghouse to describe which PRHR heat transfer option it has selected for each analysis in Chapter 15 of the AP600 SSAR in which LOFTRAN is applied and explain why the option is conservative for that application. This was SDSER Open Item 21.6.1.6-1. Westinghouse provided details of the PRHR heat transfer options used by LOFTRAN for the Chapter 15 SSAR analyses in WCAP-14234. In addition, the staff has considered this information in its review of the applicable transients presented in Chapter 15 of the AP600 SSAR to ensure selection of the appropriate heat transfer options. Therefore, SDSER Open Item 21.6.1.6-1 is closed.

21.6.1.6.2 Integral-Systems Tests

Integral-systems tests permit an assessment of the entire code, or a representative set of the code's models, and its capability to predict the behavior of an entire system that is representative of a nuclear power reactor design.

Integral-systems tests conducted at the SPES-1 facility were used to assess LOFTRAN's capabilities for predicting natural circulation. Comparison with the results of Test SPNC-01 indicates that LOFTRAN was capable of predicting the natural circulation flow rate to within 4 percent of the measured flow rate.

A number of cases involving SGTR at the SPES-2 facility were predicted using LOFTRAN. The response of the parameters identified as having high importance in the PIRT, (Table 21.6-1), were predicted by LOFTRAN. The behavior of the CMT and PRHR systems during the SPES-2 SGTR tests were accurately predicted by LOFTRAN.

Main steamline break (MSLB) tests conducted at the SPES-2 facility were used for assessing LOFTRAN. Overall, LOFTRAN was able to predict test trends. However, features of the test

facility, such as steam generator thick metal heat capacity and system heat losses, were not predicted correctly. Also, the use of conservative rather than realistic break flow modeling resulted in over-prediction of the break flow rate and break flow quality. The staff notes that where LOFTRAN has difficulty with modeling the SPES results as a result of excessive heat losses, the AP600 design is not expected to have similar heat losses. Overall, LOFTRAN test predictions appear to be conservative.

21.6.1.7 Open Item Close-out

The staff could not complete its review of LOFTRAN until it had received and reviewed responses to all unanswered RAIs. This was SDSER Open Item 21.6.1.7-1. Westinghouse has now submitted all unanswered RAI responses related to LOFTRAN. The staff has reviewed the responses and found them to be technically sound and acceptable. Therefore, SDSER Open Item 21.6.1.7-1 is closed. The staff's evaluation of the Westinghouse responses to its specific concerns is discussed below. Westinghouse was also requested to identify which information in the RAI responses would be incorporated into the LOFTRAN FV&V report (WCAP-14307) or the code applicability document (WCAP-14234). This was SDSER Open Item 21.6.1.7-2. Westinghouse has incorporated all RAI responses related the LOFTRAN FV&V report into WCAP-1430 and all RAI responses related to the code applicability document into WCAP-14234. Therefore, SDSER Open Item 21.6.1.7-2 is closed.

In its responses to RAIs 440.279 and 440.449, Westinghouse indicated that auxiliary programs are used during transient analyses. It had not submitted these auxiliary programs to the staff for review but had furnished an outline of the methodology used for the auxiliary programs in its RAI responses. Westinghouse stated that the code's numerics would be too difficult to modify in order to model two pumps and cold legs for each loop. The inability to model the basic geometry of an AP600 plant differs from the use of LOFTRAN for conventional PWRs, for which modeling the plant geometry is straightforward. Therefore, Westinghouse has resorted to auxiliary programs, hand calculations, and conservative assumptions to model pump trips and startups. Although this is not the preferred method of transient analysis, the staff concludes that the analyses can be done in an acceptable and conservative manner. However, in order to confirm the acceptability of this methodology, the staff requested that Westinghouse submit a detailed example of use of the methodology for staff review. This was SDSER Open Item 21.6.1.7-3. Westinghouse submitted a report, "Methodology and Sample Verification Calculations for AP600 Partial Loss of Flow Transients and Locked Rotor/Broken Shaft Transients," dated September 1996. The referenced methodology provides a conservative calculation of the effects of a partial loss of flow or locked rotor/broken shaft and, therefore, SDSER Open Item 21.6.1.7-3 is closed.

RAIs 440.315 and 440.449 address the use of the "penalty" model for the CMT piping when "moderate" voiding takes place. Westinghouse is inconsistent in its RAI responses regarding the use or non-use of this model. Since LOFTRAN has no "true" capability of modeling two-phase flow, as can be seen in some of the SPES-2 comparisons when the system is two phase, the user must be made aware of the approach to such conditions. SDSER Open Item 21.6.1.7-4 requested Westinghouse to submit criteria for the use of the penalty model. In response to this concern, Westinghouse has stated that, should boiling be detected in an analysis, a sufficiently large equivalent stratified zone height will be applied to stop natural circulation of the CMT. However, Westinghouse notes that boiling does not occur in the CMT during the design basis non-LOCA and steam generator tube rupture transients and, therefore,

this model is not used. Because none of the transients analyzed with LOFTRAN results in boiling in the CMT piping, SDSER Open Item 21.6.1.7-4 is closed.

It is the staff's position that LOFTRAN be restricted from application to analysis involving actuation of the ADS since it has not been benchmarked against ADS actuation experiments. ADS actuation would involve global two-phase flow behavior for blowdown. LOFTRAN does not have the capability to model this behavior. This was SDSER Open Item 21.6.1.7-5. Westinghouse has stated that the ADS piping interactions and possible interactions with the IRWST have not been assessed in the LOFTRAN code since the ADS is not used for mitigation of any transients analyzed with the code. The inadvertent opening of the ADS valves is the only transient analyzed with the code in which the ADS plays a part. In this case, the ADS is treated in the same manner as an open PORV for which LOFTRAN has been found acceptable. Therefore, SDSER Open Item 21.6.1.7-5 is closed.

Westinghouse furnished information, in response to RAI 440.288, about mass and energy errors for the CMT solution scheme. A true indication of the mass error is the difference between the state density and the mixture density multiplied by the volume at each time step. Westinghouse, however, calculates mass error by comparing the mass of progressive time steps. The cumulative error is the sum at each time step. The correct cumulative error needs to be provided because of the relatively large time steps being taken in LOFTRAN and the CMT. Westinghouse does not perform a balance on mass, energy, momentum, or volume in LOFTRAN to conserve data storage. The staff requested that for each transient analyzed using LOFTRAN, Westinghouse should provide information on the impact of not conserving mass, energy, momentum, and volume. This was SDSER Open Item 21.6.1.7-6. Westinghouse has provided the maximum mass error and the maximum energy error for the limiting case of each of the six main design-basis accident classes analyzed with LOFTRAN. The maximum mass error that occurs is +0.035 percent and the maximum energy error that occurs is +0.108 percent. Therefore, SDSER Open Item 21.6.1.7-6 is closed.

Westinghouse's response to RAI 440.462 was not clear whether LOFTRAN can "unchoke" at an SGTR break or at a main steamline break. When a break unchokes, as will be the case in any break model, the form losses because of the geometry of the break and the piping become important, along with the frictional losses. It is not clear from the Westinghouse response whether the flow can unchoke and choke separately. If choked flow is applied throughout the transients, Westinghouse was requested to demonstrate that this approach is conservative in all cases for the AP600. This was SDSER Open Item 21.6.1.7-7. Three applications are expected to result in chocked flow (1) steam generator tube rupture, (2) main steamline break, and (3) feedwater line break. Use of LOFTRAN for these three events has been previously reviewed and approved. The feedwater line break application is different for the AP600 in that the PRHR partially mitigates the event by heat removal in addition to the fluid remaining in the steam generators. In the analysis the break is sized to result in complete loss of feedwater flow to both steam generators. This rapidly depletes the inventory, minimizing the heat removal from the faulted steam generator. Therefore, SDSER Open Item 21.6.1.7-7 is item is closed.

21.6.1.8 Conclusions

Westinghouse developed LOFTRAN to assess the consequences of a number of the SSAR Chapter 15 nuclear power plant transients and upset conditions in the current generation of

plants. The code was modified for application to the AP600 passive reactor design through addition of component models for the ADS, CMT, PRHR, and IRWST. On the basis of this review, the staff concludes that LOFTRAN has been modified to include the necessary models for the AP600 plant features and the behavior expected during an AP600 non-LOCA transient and, therefore, is applicable to the AP600 passive reactor design.

21.6.2 NOTRUMP Computer Code for Small-Break LOCAs

The staff has evaluated Westinghouse's application of the NOTRUMP computer code to the AP600 design, as described below.

21.6.2.1 Background

The acceptance criteria for emergency core cooling systems (ECCS) for light-water nuclear power reactors, given in 10 CFR 50.46, require that ECCS performance be calculated in accordance with an acceptable evaluation model. Two approaches may be taken to demonstrate that an acceptable model has been applied to an ECCS design. In one approach (commonly referred to as a "best estimate"), the evaluation model must contain sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. This necessitates comparisons to applicable experimental data along with identification and assessment of uncertainty in the analysis methods and inputs so that the uncertainty in the calculated results can be estimated. This uncertainty must then be accounted for in subsequent calculations. Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of 10 CFR Part 50, Appendix K, ECCS evaluation models. Westinghouse chose to demonstrate the acceptability of the SBLOCA response of the AP600 passive reactor design using an Appendix K ECCS evaluation model.

NUREG-0737 ("Clarification of TMI Action Plan Requirements") is a report transmitted by NRC to licensees of operating power reactors and applicants for operating reactor licenses forwarding Three Mile Island (TMI) Action Plan items that were approved by the Commission for implementation. In Section II.K.3.30 of Enclosure 3 to NUREG-0737, the staff outlines the Commission guidance for the licensee to demonstrate that its SBLOCA methods comply with the requirements of Appendix K to 10 CFR Part 50. The technical issues to be addressed are outlined in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants".

In this section, the staff documents its review of the NOTRUMP computer program for calculating the plant response to SBLOCAs occurring in the Westinghouse AP600 passive standardized plant design.

21.6.2.2 Summary of NOTRUMP and the Topical Reports

NOTRUMP is a thermal-hydraulic computer program developed for analysis of Safety Analysis Report (SAR) Chapter 15 SBLOCA events, as identified in the SRP. NOTRUMP was submitted to the NRC by Westinghouse in a letter dated November 12, 1982. NRC acceptance was given in a letter dated May 23, 1985.

The code models one-dimensional thermal hydraulics using control volumes interconnected by flow paths (junctions). The spatial- and time-dependent solution is governed by the integral forms of the conservation equations in the control volumes and flow paths. The code thermal-hydraulics account for nonequilibrium thermodynamics and apply drift flux models for calculating relative velocities between the steam and liquid phases. Reactivity feedback is modeled with point kinetic neutronics. The code incorporates special models to calculate responses of the reactor coolant pumps, steam separators, and the core fuel pins. Another significant code feature is a node stacking capability for calculating a single mixture elevation. This eliminates unrealistic layering of steam and liquid mixture in adjacent vertical control volumes (i.e., "pancaking effects"). A two-phase horizontal stratified flow model is also included.

The Westinghouse report on the applicability of NOTRUMP to the AP600 passive plant design was submitted in WCAP-14206, "Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses," dated November 1994. Assessments of the AP600 hardware-specific models added to the NOTRUMP code were submitted through numerous preliminary verification and validation reports pertaining to the various separate effects and integral systems test facilities employed in the AP600 test program. The following preliminary verification and validation reports identified numerous changes made to the models contained in the previously approved NOTRUMP computer code:

- MT01-GSR-001, "AP600 NOTRUMP Core Makeup Tank Preliminary Validation Report," October 1994
- LTCT-GSR-001, "NOTRUMP Preliminary Validation Report for OSU Tests," July 1995
- PXS-GSR-002, "NOTRUMP Preliminary Validation Report for SPES-2 Tests," July 1995
- RCS-GSR-003, "AP600 NOTRUMP Automatic Depressurization System Preliminary Validation Report," April 1995
- MT01-GSR-011, "AP600 NOTRUMP Core Makeup Tank Preliminary Validation Report for 500-Series Natural Circulation Tests," April 1995
- LTCT-GSR-004, "Addendum to NOTRUMP Preliminary Validation Report for OSU Tests," September 1995
- PXS-GSR-004, "Addendum to NOTRUMP Preliminary Validation Report for SPES-2 Tests," September 1995

Descriptions of the model changes were subsequently submitted for staff review in Westinghouse responses to RAIs 440.466 through 440.485. A comprehensive report providing an assessment of the model changes, and performance of the code, is documented in Westinghouse WCAP-14807, Revision 4 ("NOTRUMP Final Verification and Validation Report"), dated February 1998, along with the approximately 150 RAI responses provided to the staff for review.

21.6.2.3 Phenomena Identification and Ranking Table

It is important to identify all physical phenomena that will occur in the AP600 under accident conditions of interest to ensure that the important physical processes and phenomena have been modeled. One method of identification is through the development of a PIRT. The PIRT methodology provides a framework where physical processes and phenomena in a specific hardware geometry under anticipated accident sequences are first identified and then ranked in terms of their importance to the course of the analysis. A PIRT is generally developed from expert opinions provided by a group of knowledgeable analysts. The use of a group of experts, rather than a single analyst, increases the chances that all important phenomena have been identified and included in the PIRT, and that the rankings have accurately characterized each specific phenomena as being of high, medium, or low importance to the integral quantities of interest. A properly established PIRT will act as a road map through a transient, identifying and ranking the important phenomena and functions necessary to predict and deal with each phase of a transient. The PIRT for the AP600 SBLOCA, submitted in response to a RAI 440.325, is shown in Table 21.6-2.

The staff also developed a PIRT as part of the review and confirmatory process. The NRC PIRT for the AP600 SBLOCA is documented in a report from Idaho National Engineering Laboratory, INEL-94/0061, Revision 1. The PIRT prepared by Westinghouse divides the SBLOCA into four intervals (1) blowdown, (2) natural circulation, (3) automatic depressurization system (ADS) blowdown, (4)and IRWST injection cooling. Within each interval, the specific hardware and phenomena are evaluated as having high (H), medium (M), or low (L) importance. The NRC PIRT contains five intervals. The hardware functions and phenomena within two of the NRC PIRT intervals, "Passive Decay Heat Removal" and "CMT Drain to ADS Actuation," are accounted for in the Westinghouse PIRT interval "Natural Circulation." Therefore, all hardware functions and phenomena are accounted for. The Westinghouse PIRT and the NRC PIRT are comparable.

Westinghouse also submitted a list of the important phenomena and hardware items identified in the PIRT with a description of the test program, and planned benchmark and assessment calculations which will provide supporting validation for the plant analyses.

The staff compared the Westinghouse and NRC PIRTs and finds that all high- and medium-ranked phenomena are captured both in the PIRTs and in the testing program. Therefore, the staff finds the Westinghouse PIRT is applicable to the AP600 passive reactor design.

21.6.2.4 Evaluation of the NOTRUMP Analytical Models

NOTRUMP is a general (variable) nodalization code. Plant models are constructed from generalized control volumes (fluid and metal nodes), flow links, heat sources, and heat sinks. The nonequilibrium thermodynamics and hydraulics include several drift-flux options to calculate relative vapor/liquid velocities (slip). Fission heat is calculated using reactivity and reactor kinetics. The code uses the same water thermodynamic properties as used in WFLASH (WCAP-8200, "WFLASH — A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," July 1973). The code has an extensive number of forced- and natural-convection heat transfer correlations covering the spectrum of the boiling curve.

Critical flow correlations available include the Moody model, a modified Zaloudek model, and the Murdock-Baumann model. Special-purpose models include flooding, bubble rise, mixture level tracking, a continuous contact flow link, variable flow links, a horizontal stratified flow model, and externals which provide the user flexibility to "program" user specific modifications. Component models include an accumulator, a centrifugal pump, steam separators, and a fuel rod model. The user has available control volumes, flow paths, and heat slabs which can be used to control pressure, enthalpies, mixture levels, mass flows, and heat fluxes as a function of time. Simple valves are simulated as input flow loss coefficients.

Application of the approved NOTRUMP computer code to the AP600 passive reactor design required a number of modifications, or enhancements, to the basic NOTRUMP models. Nineteen modifications were made to the models as follows:

(1) Add the SIMARC (SIMulator Advanced Real-time Code) drift-flux model.
(2) Modify the drift-flux correlations.
(3) Recast the momentum equations for net volumetric flow.
(4) Add the NOTRUMP EPRI/Flooding Drift-Flux Model.
(5) Modify contact coefficients.
(6) Add internally calculated liquid reflux flow links.
(7) Add mixture overshoot logic.
(8) Add implicit treatment of bubble rise.
(9) Modify the pump model.
(10) Add implicit treatment of momentum equation gravitational head terms.
(11) Modify horizontal flow drift-flux levelizing model.
(12) Add an Unchoking Model.
(13) Add Shah condensation correlation.
(14) Add Zuber critical heat flux correlation.
(15) Change the two-phase friction multiplier.
(16) Add Henry/Fauske model and homogenous equilibrium model.
(17) Modify fluid node stacking logic.
(18) Modify transition boiling correlation solution.
(19) Code Numerics.

The following is a brief summary of the models changed in, or added to, the NOTRUMP code.

(1) SIMARC Drift-Flux Model

The NOTRUMP code uses a drift-flux model to calculate phase separation within a node. Use of the void propagation approach in the drift velocity models causes nonphysical behavior in cases of low-void fraction nodes above high-void fraction nodes. Liquid could be levitated above vapor, or a high-phase flow from a node would be predicted when the node had little of that phase to actually support the predicted flow. The design of the ECCS for the AP600 leads to low-pressure and low-flow conditions for an SBLOCA, which amplified these problems.

A methodology developed through the Westinghouse Simulators Department, the SIMulator Advanced Real-time Code (SIMARC), was adapted to the NOTRUMP drift-flux model. The SIMARC drift-flux methodology, although simplified in its simulator version, was modified for use in NOTRUMP. The basis of the methodology as applied is that the net fluxes at a transition from

concurrent up-flow or down-flow to countercurrent flow can be determined. Determination of the transition fluxes is by either mass or volumetric fluxes, on the basis of the code structure. Experience has shown that the methodology works best when applied on a volumetric flow basis. Westinghouse found that the SIMARC methodology can help solve drift-flux difficulties such as a liquid node above a gas node.

Before the review of the SIMARC methodology, the staff's opinion was that either a flux-weighted void fraction approach or a void propagation approach should be used. Although a flux-weighted approach may be nonphysical (i.e., the mathematical model yields a condition that is physically impossible), it smooths numerical problems and gives a well-behaved solution. Westinghouse pointed out that the SIMARC approach is more physical in that it is on the basis of transitions from concurrent up-flow or down-flow to countercurrent flow, and it is simpler to apply. Similarly, although the void propagation approach is physically sound, it has difficulties with cases of liquid nodes above gas nodes. The SIMARC approach is both physically based and able to deal with this case.

Several of the NOTRUMP models were modified for analyzing the AP600 design. These models must be evaluated through inference on the basis of code response to other phenomena. For the AP600, Westinghouse evaluated the SIMARC drift flux technology by observing the capability of the code with respect to two-phase level swell calculations. Through the results documented in WCAP-14807, Westinghouse demonstrates that the NOTRUMP code provides a conservative prediction of data from two-phase level swell experiments (G-2, GE, Achilles). The following documents support the conclusions in WCAP-14807:

- Westinghouse Electric Corporation-EPRI report, EPRI NP-1692, Volume 1,"Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336-Rod Bundle," January 1981.

- Battelle Northwest Laboratory report, BNWL-1411, "Experimental High Enthalpy Water Blowdown From a Simple Vessel Through a Bottom Outlet," June 1970.

- Battelle Northwest Laboratory report, BNWL-1463, "Coolant Blowdown Studies of a Reactor Simulator Vessel Containing a Perforated Sieve Plate Separator," February 1971.

- NEDO-10329, "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," April 1971.

- EPRI Report EPRI NP-1527, "BWR Refill-Reflood Program - Model Qualification Task Plan," October 1991.

- NUREG-CR-1899/GEAP-24898, "BWR Refill-Reflood Program - Model Qualification Task Plan," October 1981.

- NUREG-CR-2571/GEAP-22049, "Refill-Reflood Program Model Qualification for BWR Safety Analysis," February 1984.

- AEEW-R2339, "Achilles Unballooned Cluster Experiments Part 4: Low Pressure Level Swell Experiments," July 1989.

Westinghouse also concluded that the code results were reasonable for benchmark studies of vertical flow and horizontal flow. The staff agrees with this assessment and finds the use of the SIMARC drift-flux technology acceptable for analysis of the AP600 SBLOCA.

(2) Modified Drift-Flux Correlations

The drift-flux correlations in NOTRUMP were modified in three ways to make them work with the SIMARC technology discussed above. The three modifications involved (1) the Yeh void fraction, (2) the improved TRAC-P1 flow regime map, and (3) the distribution parameter used with the bubbly and slug-flow drift correlations.

The Yeh void fraction correlation was modified to produce reasonable interphase velocities as void fractions approach zero and one. The TRAC-P1 flow regime map was modified in its implementation in NOTRUMP to use a general droplet flow correlation in place of an interpolation scheme between the annular flow region and a region in which there is a void fraction of one. This produces more reasonable interphase velocities as the void fraction approaches one. The modification to the distribution parameter used with the bubbly and slug-flow correlations resolves creating a situation in which it can become impossible to solve for net flow through a flow link leading to unrealistic interphase velocities.

Through the results documented in WCAP-14807, Westinghouse demonstrates that the NOTRUMP code agrees reasonably well with data from two-phase level swell experiments. Westinghouse also concludes that the code predicts counter-current flow limitation (CCFL) well for the benchmark studies, and the results indicate that NOTRUMP tends to conservatively predict more holdup of liquid in small and large diameter pipes than the data indicate. In view of the above, the staff finds the use of the modifications made to the NOTRUMP drift-flux correlations acceptable for analysis of the AP600 SBLOCA.

(3) Re-cast Momentum Equation for Net Volumetric Flow

The NOTRUMP code was modified to add an option to use the momentum equation and the drift-flux equations, for a junction, in terms of net volumetric flow. This modification permits the mass flow in a junction to change as densities in the adjacent nodes change, improving the behavior of the node stacking and mixture level tracking. This will result in reduction in non-physical predicted pressure oscillations. In addition, as discussed with regard to the SIMARC drift-flux methodology, drift-flux models generally perform better in terms of volumetric flow since drift flux is itself a volumetric flow concept. The staff notes that the NRC supported codes TRAC and RELAP5 also use volumetric-based momentum equations.

In WCAP-14807, Westinghouse documents a benchmark analysis performed by simulating flow through a horizontal frictionless pipe. Because the code predicts that the velocity will remain constant while the density gradient propagates downstream without a significant change in pressure, Westinghouse concludes that the model is functioning properly. The staff agrees that the model is functioning conservatively and finds this modification to NOTRUMP for the analysis of the AP600 SBLOCA acceptable.

(4) NOTRUMP EPRI/Flooding Drift-Flux Model

A vertical drift-flux model, on the basis of the Electric Power Research Institute (EPRI) full-range drift flux correlation, a vertical flooding correlation, and the SIMARC drift flux methodology, was implemented in NOTRUMP. The model uses the EPRI correlations for concurrent upflow and concurrent downflow with a flooding limit at high void fractions. This model was used for all vertical pipes outside of the core region.

In WCAP-14807, Westinghouse compared the results of NOTRUMP calculations to accepted correlations for a benchmark countercurrent flow limitation problem. It is evident from the data that the NOTRUMP EPRI/flooding drift-flux model predicts a reduction in liquid downflow velocity at lower vapor velocities than in the accepted correlations. This tends to cause NOTRUMP to calculate more liquid holdup than is observed in experiments. This is a conservative result, and therefore, the staff finds the NOTRUMP EPRI/flooding drift flux model for use in AP600 SBLOCA analyses acceptable.

(5) Modified Contact Coefficients

Fluid nodes represented in the NOTRUMP code can contain both a lower mixture region and an upper vapor region. The mixture region can contain either subcooled water or a saturated steam-water mixture. The vapor region can contain either superheated vapor or a saturated steam-water mixture. For each flowlink connected to a fluid node, the code must determine what fraction of the link's phasic flows come from or go into that node's vapor region. In NOTRUMP, this flow partitioning is controlled by contact coefficients. Using the default model, a problem can occur when subcooled liquid enters a node with no mixture region so that all of the liquid is homogeneously mixed into the vapor region. This situation results in a sudden, large depressurization of the node. An option has been added to the flow partitioning model to allow the user to override the NOTRUMP default flow partitioning when it incorrectly represents the physical problem being analyzed. Specifically, this modification allows the user to force flow from a particular flowlink to always place liquid flow into the mixture region of the recipient node rather than the vapor region. Westinghouse notes that other model additions to NOTRUMP, such as the node stacking and mixture level tracking model (part of the mixture overshoot logic), have effects on flow partitioning. Mixture level tracking and node stacking are discussed in items (7) and (17), which follow.

The modification to the contact coefficient model is important for use at specific locations in the SPES, OSU, and AP600 models. The SPES and OSU analysis results, presented in WCAP-14807, indicate that the model modification was used to prevent sudden non-physical node depressurizations because of artificial homogenization of fluid conditions within a recipient node. Use of the flow partitioning model prevented sudden node depressurization while producing correct fluid thermal-hydraulic conditions, such as temperature and void fraction. The staff finds that the modifications added to the default flow partitioning model to prevent the model from producing non-physical results in the NOTRUMP calculation are acceptable. The non-physical behavior found in some of the OSU and SPES tests does not appear in the AP600 SBLOCA calculations.

(6) Internally Calculated Liquid Reflux Flow Links

The approved NOTRUMP code contained a model for calculating hot leg-to-reactor vessel reflux. The model for calculating hot leg-to-reactor vessel reflux has not been changed, but a generalization of the model was made to permit use of liquid reflux in other areas, such as the steam generator primary side, the CMTs, and in calculations for the SPES integral system test facility.

Generalization of the liquid reflux flow model for application to components in addition to the hot leg-to-reactor vessel flow region is acceptable for application to the AP600 SBLOCA since the flow conditions in those components are expected to be in the range of validity of the model.

(7) Mixture Overshoot Logic

The approved NOTRUMP code is capable of tracking mixture levels passing from node to node as a region drains or fills. When the code was first applied to the low pressures found in AP600, it did not perform well. A problem occurs during a time step in which a fluid node drains. During the time step, more mass and/or energy may be taken out of the lower region of the node than exists there. This causes the code to calculate negative values. Strategies to deal with this problem had to be modified for the AP600 calculations. The logic modifications will avoid the production of negative mass and energy when the mixture level crosses a node boundary. This is done by setting the mass or energy to zero, instead of allowing a negative value, and redistributing what would have been negative mass and energy to the recipient region, thus conserving both mass and energy. This permits the code to calculate valid temperature and pressure conditions in the donor region.

While performing the NOTRUMP assessment, calculations of SPES and OSU experiments, mixture levels crossed node boundaries numerous times. The results agreed reasonably well with the experimental data, and therefore, the staff finds the mixture level overshoot logic modifications for application of the NOTRUMP code to the AP600 SBLOCA acceptable.

(8) Implicit Treatment of Bubble Rise

The approved NOTRUMP code incorporated the bubble rise mass flow rate. On the one hand, maintaining a fixed mass flow rate throughout a time step can create instability if more liquid is convected out of a mixture region than was contained in the region at the beginning of the time step. On the other hand, implicit treatment of the bubble rise mass flow rate estimates the change in the bubble rise mass flow rate corresponding to the change in the fluid node's state variables during a time step. A further modification was made to prevent the bubble rise mass flow rate from becoming negative. Also, a modification was made to allow a user specified override of the internally-calculated interfacial area. Finally, an option was added to base the bubble rise on the volume flow into a node rather than on the mass flow. The latter is consistent with the recasting of the momentum equation and drift-flux equations in volumetric rather than mass flow form.

In the analysis of the behavior of an RCS undergoing an SBLOCA, one of the most important parameters calculated is the two-phase level swell. Proper assessment of the capability of the model to predict two-phase level swell is critical to determining the ability of the code to predict

such important information as the system mass inventory. The ability of a code to model two-phase level swell depends heavily on an accurate bubble rise model, as well as an accurate drift-flux model. A simulation was performed using a small two-node stack which showed that with the implicit bubble rise model a smooth transition occurred, as it should, as the mixture level crossed the node boundary. Westinghouse also assessed NOTRUMP at the low-flow and low-pressure conditions expected in the AP600 design by comparison with a number of two-phase level swell tests from the G-2 test series, tests from the GE level swell test series, and two Achilles tests. These tests are documented in the reports listed in Section 21.6.2.4(1) of this report.

Westinghouse also assessed NOTRUMP performance in a depressurizing system to characterize the ability of NOTRUMP to calculate system mass inventory under SBLOCA conditions in AP600. The staff finds the NOTRUMP performance acceptable in predicting the two-phase level based on these various comparisons - taken as a whole, and therefore, finds that the implicit treatment of bubble rise in NOTRUMP acceptable for the analysis of the AP600 SBLOCA.

(9) Modified Pump Model

The NOTRUMP pump model was adapted from the Westinghouse large-break LOCA (LBLOCA) model for use in operating plant analyses. The low-pressure and low-flow conditions of the AP600 design during the SBLOCA made parts of the generic pump model unnecessary after the initial pump coastdown. Accordingly, a set of simplifications was made to the pump model for application to the AP600 SBLOCA. First, the NOTRUMP option to use pump inlet density, from the donor-specific volume for the head-specific volume, rather than the equivalent density, is exercised. Second, a simplification is made in the pump discharge pressure calculation. Third, because choke flow is not possible, the pump critical flow calculation is not used, further simplifying the calculation. It is noted that with a larger flow area through the pump than through the break, critical flow through the pump is not expected.

Westinghouse performed two calculations modeling an AP600 SBLOCA. The only difference between the two calculations was the pump model. The results using the new pump model were identical to the results using old model. The new pump model makes the calculation more robust, in the sense of being more representative of the AP600 configuration, without changing the results. Accordingly, the staff finds the modified pump model acceptable for application of NOTRUMP to the AP600 SBLOCA.

(10) Implicit Treatment of Momentum Equation Gravitational Head Terms

The approved NOTRUMP code uses an explicit calculation of the fluid node gravitational head term. The gravitational head is then held constant throughout the time step. Use of the explicit form of the gravitational head in the code numerics can create numerical instabilities as well as flow instabilities when the fluid node density undergoes rapid changes. The gravitational head term determination was changed to an implicit form that estimates the change in the gravitational head corresponding to the change in the fluid node's state variables during a time step. This change is said to not affect the computation of the fluid node gravitational head from the fluid properties, while eliminating flow instabilities from rapid density changes in the fluid node at the low pressures anticipated in the AP600 SBLOCA.

Westinghouse performed two calculations modeling an oscillating manometer. One calculation used the original explicit treatment of gravitational head while the other used the new implicit treatment. These calculations, documented in WCAP-14807, agree reasonably well and indicate that the model using an implicit treatment of gravitational head behaves well. Therefore, the staff finds the modified model acceptable for application of NOTRUMP to the AP600 SBLOCA.

(11) Modified Horizontal Flow Drift-Flux Levelizing Model

The NOTRUMP drift-flux model, using the SIMARC drift-flux technology, was modified to define a drift velocity whose magnitude and direction are on the basis of the collapsed levels at each end of a horizontal flow link. The form of the correlation, Westinghouse has stated in response to RAI 440.477, was developed heuristically. The staff is of the opinion that models developed by use of heuristic methods must be justified and fully evaluated since heuristic methods, by nature, lack strong technical bases. However, Westinghouse notes that the appropriate dimensions and effects are incorporated.

The levelizing model is intended to provide a potential for countercurrent flow in which liquid flows toward the end of a link with the lower collapsed level. This would tend to bring the levels to the same point. Primary use of the levelizing model in the AP600 SBLOCA analysis is in the horizontal links at the top of the steam generator U-tubes, at the top of the hot leg-to-PRHR system line, and in the PRHR horizontal links.

Westinghouse performed a benchmark calculation and reported the results in WCAP-14807. The benchmark models a horizontal pipe with a dam at one end and a drain at the other end. The benchmark also compares NOTRUMP with well-known flow regime transitions modes, such as Taitel/Dukler and Wallis/Dobson. Liquid is injected into the bottom of the pipe while vapor is injected into the top. The calculated results indicate that NOTRUMP predicts that countercurrent flow will occur within the limits allowed by interfacial wave stability. The staff finds the horizontal flow drift-flux levelizing model acceptable for application of NOTRUMP to the AP600 SBLOCA.

(12) Unchoking Model

A revised model has been implemented in the NOTRUMP code which is used for the transition from choked to unchoked conditions for all critical flow models. This model provides better blending between choked and unchoked flow conditions and provides a continuous transition between fully choked flow (when the recipient stagnation pressure is less than or equal to the throat pressure) and fully unchoked flow (when the recipient stagnation pressure approaches the donor stagnation pressure).

Data from separate effects tests and integral system tests, such as upstream and downstream pressures and flow rates, were compared to NOTRUMP calculations and indicate that the NOTRUMP unchoking model operates adequately for the AP600 SBLOCA. Accordingly, the staff finds that the unchoking model is acceptable for the application of NOTRUMP to the AP600 SBLOCA.

(13) Shah Condensation Correlation

Westinghouse added the Shah condensation correlation, a general flow correlation for condensation inside pipes, to the NOTRUMP code to handle the anticipated steam condensation inside both the steam generator tubes and PRHR tubes. The correlation is a function of the Reynolds number, the tube hydraulic diameter, the Prandtl number, reduced pressure, and fluid quality in the tube. The recommended tube diameter for application of the correlation encompasses both the AP600 steam generator tubes and the PRHR tubes. The remaining recommended parameter ranges for application of the Shah correlation are predicted to be met under SBLOCA conditions in the AP600 design.

Based on the above, the staff finds the addition of the Shah condensation correlation to the NOTRUMP condensation correlations acceptable for analysis of the AP600 SBLOCA.

(14) Zuber Critical Heat Flux Correlation

The heat transfer package in NOTRUMP was extended by addition of the Zuber critical heat flux correlation for stagnant flow situations. In particular, in the AP600 design, stagnant flow on the IRWST pool side of the PRHRHX would have had NOTRUMP, with the original heat transfer package, calculating transition boiling rather than saturated natural circulation. That would have overpredicted the tubeside energy release to the poolside fluid. The Zuber correlation was installed to permit a more realistic prediction under stagnant flow conditions.

For the majority of the SBLOCA transients, the temperatures in the IRWST pool do not approach boiling because the loss of inventory in the RCS makes the PRHR inactive before the IRWST water can be sufficiently heated. IRWST boiling might only be reached for very small SBLOCAs. However, the Zuber correlation will play a role in all transients because it is used as a lower limit on the critical heat flux for the stagnant fluid in the IRWST for all cases.

The form of the Zuber critical heat flux correlation, modified by Lienhardt and Dhir, results in a more realistic, and more conservative, estimate of the energy released to the IRWST when boiling occurs. The PRHR heat transfer calculated by NOTRUMP for the SPES and OSU transients, documented in WCAP-14807, is lower than that measured in the experiments. This is a conservative result. Based on the above, the staff finds the use of the Zuber critical heat flux correlation, as modified by Lienhardt and Dhir, acceptable for use in NOTRUMP analysis of the AP600 SBLOCA.

(15) Changes to the Two-Phase Friction Multiplier

The approved NOTRUMP code used the Martinelli-Nelson two-phase friction multiplier, as modified by Thom, for pressures above 1.72 MPa (250 psia). Below 1.72 MPa (250 psia), the approved code used the multiplier for 1.72 MPa (250 psia). Since the AP600 design is expected to depressurize to atmospheric pressure via the ADS, the two-phase friction multiplier was extended to atmospheric pressure. The extension installed in NOTRUMP was an augmentation of the Thom-modified Martinelli-Nelson table with two data points, 689 kPa (100 psia) and 101.3 kPa (14.7 psia), from the unmodified Martinelli-Nelson data. Linear interpolation is used between the Thom-modified Martinelli-Nelson multiplier at 1.72 MPa (250 psia) and the Martinelli-Nelson multiplier at 689 kPa (100 psia) and 101.3 kPa (14.7 psia).

In addition, two potential discontinuities in two-phase friction multiplier determination were eliminated. The user can base the computation of the multiplier at a flow link undergoing transition from concurrent to countercurrent flow on the thermodynamic quality rather than on the flow quality. Also, the transition from two-phase to single-phase steam flow frictional pressure drop is smoothed by use of a linear interpolation of the friction loss between flow quality of 90 percent and saturated steam.

As with the SIMARC drift-flux modeling, the two-phase friction multiplier must be evaluated as part of the overall performance in prediction of related phenomena, such as two-phase level swell. Through results documented in WCAP-14807, Westinghouse demonstrates that the NOTRUMP code results agree reasonably well with data from two-phase level swell experiments (G-2, GE, and Achilles). Code results were also found to be reasonable for the SPES and OSU test assessments. The staff finds that the extension of the two-phase friction multiplier and smoothing techniques used in NOTRUMP are acceptable for evaluation of the AP600 SBLOCA.

(16) Henry/Fauske and HEM Critical Flow Model

The critical flow models contained in the approved NOTRUMP code have been augmented with the addition of the Henry/Fauske and Homogeneous Equilibrium Models (HEM). Appendix K, I.C.1.b, to 10 CFR Part 50 mandates the use of the Moody model to calculate the discharge rate from a break once the fluid has been calculated to be two-phase in composition. It was found in the ADS Phase B tests (Westinghouse report RCS-GSR-003), as well as in the integral systems tests performed in the SPES facility (Westinghouse report PXS-GSR-002) that the use of the Henry/Fauske and HEM discharge models for flow through the ADS valves provided a more realistic representation of the expected discharge rate, and thus the system depressurization, following ADS actuation.

Concerns were raised by the staff regarding the degree of conservatism in use of the Henry/Fauske and HEM models versus the Moody model for calculation of the discharge rate from the AP600 ADS valves during a SBLOCA. In response, Westinghouse demonstrated through a comparison calculation that the use of Henry/Fauske and HEM produced a longer system depressurization, with an accompanying greater system mass inventory reduction. Thus, the use of Henry/Fauske and HEM for the calculation of flow through the ADS valves is the more conservative approach and is acceptable.

(17) Modified Fluid Node Stacking Model

A node stacking and mixture level tracking model was implemented in the approved NOTRUMP SBLOCA computer code. In the application of the code to the AP600 SBLOCA, the node stacking logic was refined. Refinement was added to allow redefining mixture fractions under special conditions, and to pass mixture level control to the more general mixture level overshoot logic added to the code. When a moving mixture region reaches a point contact flow link, or junction, the mixture fraction is redefined to assist in forming a new region in the node into which the mixture region is moving. Under normal conditions the mixture level overshoot logic will handle passing the mixture level between the nodes, now making the stacking logic a backup scheme should the mixture level become "stuck" at the boundary of stacked nodes. As

described, the fluid node stacking logic assists when the stack mixture elevation is draining, filling, or stuck.

As noted above, several logic additions have been made to the NOTRUMP code, including the fluid node stacking, mixture level overshoot, and bubble rise logic. The proper functioning of these logic schemes while interacting can be demonstrated best in the analysis of the SPES and OSU integral system tests. These tests have been analyzed and documented in WCAP-14807. The results yield reasonable agreement between the experimental data and the NOTRUMP calculation except for the double-ended guillotine break of the direct vessel injection (DVI) line transients in SPES and OSU. In the double-ended guillotine break of the DVI line, the core level is overpredicted. The Westinghouse explanation of the differences in core and downcomer behavior in the DVI line break is primarily because of the one-dimensional nature of NOTRUMP. The test data indicate that a two-dimensional temperature pattern develops in the downcomer which NOTRUMP is not able to predict. This allows portions of the downcomer to remain saturated and to flash when ADS 1-3 open. Less mass is then stored in the downcomer. Also, the vapor generated in the core exits through the broken DVI line and not through the intact DVI line. NOTRUMP predicts vapor exiting by both paths. The staff believes that the AP600 will perform in a manner that is more similar to the behavior of the test facilities than to the behavior predicted by NOTRUMP. This discrepancy is resolved by the time ADS 1-3 blowdown is completed, as evidenced by good agreement between the measured and predicted core levels in both the SPES and OSU tests. Thus, the discrepancy does not adversely affect the prediction of core level. Although the NOTRUMP code is unable to predict the two-dimensional behavior of the design, the staff concludes that this discrepancy is acceptable because the core does not uncover in either the tests or the calculations. Core heatup does not occur in either case. A two-dimensional analytical capability would be desirable, but would not appreciably change the results.

(18) Modified Transition Boiling Correlation Solution

Westinghouse reports LTCT-GSR-001 ("NOTRUMP OSU Preliminary Verification and Validation Report") and PXS-GSR-002 ("SPES-2 Preliminary Verification and Validation Report") refer to changes that were made to the numerical solution scheme used in the NOTRUMP heat links when the Westinghouse Transition Boiling correlation is used. The successive substitution iterative scheme that was used previously was replaced by an interval halving iterative scheme. This scheme guarantees convergence. Westinghouse states that the heat links are not used to model core heat transfer and that no changes are made for the AP600 application. Transition boiling is not expected to occur since the AP600 core is not predicted to uncover.

Because transition boiling is not predicted to occur in the core, and the heat links solution techniques are not use for core heat transfer, the staff did not review the changes to the numerical solution techniques in the NOTRUMP heat links for calculating core heat transfer.

(19) Code Numerics

Of the numerous modifications to the NOTRUMP code models that were discussed above, some of the model changes also resulted in changes and modifications to the code numerics. In WCAP-14807, Westinghouse provides a discussion of the changes in the numerics that resulted from modifications, such as moving from a mass flow-based to net volumetric flow-based momentum equation, implicit treatment of bubble rise and droplet fall, and implicit treatment of

gravitational head. This discussion traces the development of the numerics changes from the source equations to their difference formulation to the modifications to the central matrix equation. The changes in the solution matrix are presented along with the modifications to the submatrices. The numerics changes, while complex, are presented in sufficient detail such that the staff is able to conclude that the numerics were modified in an acceptable fashion to accomplish an accurate solution to the physical processes being represented.

21.6.2.5 Evaluation of the NOTRUMP AP600 Component Models

In addition to the NOTRUMP model modifications discussed above, hardware-specific component models were added to represent AP600-unique hardware features. Component model additions comprise the following:

(1) ADS
(2) CMT
(3) PRHRHX
(4) IRWST

The following is a brief summary of the component models added to the NOTRUMP code. The component tests, shown in Table 21.6-3, were reanalyzed to demonstrate the NOTRUMP model performance for the ADS and CMT components and pertain to the discussion which follows in the respective sections evaluating the ADS and CMT models.

(1) Automatic Depressurization System

The ADS is designed to depressurize the RCS to values near the prevailing containment pressure to enable gravity injection from the IRWST. Three stages of the ADS come off the top of the pressurizer; the fourth-stage ADS paths are connected to the hot legs. The first stage ADS is actuated when 33 percent of a CMT liquid has drained, resulting in the depressurization of the plant via the ADS valves to the IRWST. The second and third stage ADS valves open on the basis of timers that are started with the actuation of the first stage and also discharge to the IRWST. If the CMTs continue to drain, the fourth stage ADS will actuate when 80 percent of the liquid has drained from a CMT. The fourth stage ADS valves, located on the hot legs, open directly to the containment to facilitate depressurization to the containment pressure.

Westinghouse models the first three stages of ADS in NOTRUMP as a single flow link. The fourth-stage ADS is modeled as a single flow path to the containment. The loss coefficient used for the fully open area is converted from the manufacturer supplied value of the valve coefficient. The four arms of the sparger located in the IRWST are modeled as a single node. Critical flow through the ADS valves is modeled using the Henry/Fauske subcooled critical flow model and the HEM for saturated fluid discharge. Discussion of the Henry/Fauske and HEM additions to NOTRUMP were previously discussed.

For critical flow, the NOTRUMP code compared well with flow data from the ADS tests, which indicates that the critical flow models in NOTRUMP perform acceptably for the calculation of flow through the ADS valves. This is a highly ranked PIRT item. However, NOTRUMP tends to underpredict the upstream piping pressure drop in the tests and overpredicts the pressure drop of the ADS valve. When the flow in the ADS valve is choked, NOTRUMP overpredicts the

pressure drop. However, the overall ADS system pressure drop is predicted well, resulting in correct prediction of ADS choked flow. This situation raised a concern about the models and how they affect the fluid conditions at the entrance to the ADS piping. Westinghouse reviewed the ADS 1-3 test data and determined that the data reduction was performed correctly. The staff reviewed a comparison of the pressure, flow rate, and timing results for the SPES and OSU tests, and the responses to the staff's concerns and found them acceptable.

(2) Core Makeup Tank

There are two CMTs connected to the RCS by normally open isolation valves on the cold-leg balance lines and normally closed isolation valve on the CMT discharge lines. The CMTs provide high-pressure, gravity-driven, borated coolant injection into the RCS to provide reactivity control and core cooling. The CMT discharge valves open on a safety (S) signal and remain open. During normal operation, the CMTs and the cold-leg balance lines are completely filled with liquid.

The NOTRUMP CMTs are represented by four volumes. The top node contains 10 percent of the CMT volume; each of the two middle nodes represent 15 percent of the CMT volume; the bottom node comprising the remaining 60 percent. The cold-leg balance line is connected to the top node, while the CMT drain line is connected to the bottom node of the CMT. Each of the four fluid volumes contain heat structures representing the metal walls of the CMT.

The AP600 CMTs are new hardware designed subsequent to the guidance of NUREG-0737. WCAP-14807 documents the results of comparisons between the data obtained in the tests listed in Table 21.6-3 and the NOTRUMP calculations. The comparisons indicate that NOTRUMP, when using a multi-node CMT model, gives a reasonable prediction of the temperature distribution within the CMT. Also, the CMT pressure is predicted reasonably well and the outflow of the CMT is predicted within the error bounds on the data. Because the NOTRUMP code does not have a thermal stratification model, the predicted temperature of the injected CMT fluid is usually higher than the measured temperature, and the start of CMT draining is frequently delayed. Each of these inaccurate code predictions is conservative, therefore, the staff finds the NOTRUMP model for the CMTs acceptable for evaluations of the AP600 SBLOCA.

(3) Passive Residual Heat Removal Heat Exchanger

The PRHR system is a C-shaped, single-pass, downflow heat exchanger, submerged in the IRWST. The system inlet connects to the top of the horizontal hot-leg section containing the pressurizer loop. The system outlet connects to the bottom of the pressurizer loop steam generator outlet plenum. Normally closed isolation valves open to actuate the system on a safety (S) signal.

The NOTRUMP model of the PRHR comprises four equally sized volumes comprising the two horizontal runs of the heat exchanger with two equal-sized vertical volumes. Single nodes represent the upper (inlet) and lower (outlet) horizontal piping. The PRHR HX is immersed in the IRWST. Heat transfer is modeled using the standard NOTRUMP heat transfer correlations plus, on the inside of the tubes, the Shah correlation, as discussed above for condensation modeling, and the Lienhardt and Dhir modified Zuber correlation for critical heat flux on the IRWST side of the tubes.

After review of the integral system assessments included in WCAP-14807, the staff notes that the PRHR HX heat transfer calculated by NOTRUMP for the SPES and OSU transients is lower than that measured in the experiments. This is a conservative result, and therefore, the staff accepts the NOTRUMP PRHR model for analysis of the AP600 SBLOCA.

(4) In-containment Refueling Water Storage Tank

The IRWST provides a source of water for gravity feed injection into the RCS once RCS pressure has been reduced to values near the containment pressure. The IRWST also serves as a heat sink for the removal of heat via the PRHR discussed above and a discharge reservoir for the first three stages of the ADS. Condensation of steam in the containment provides a long-term source of water to the IRWST which can then return to the RCS. Although not part of the IRWST, the containment sump provided a second source of gravity fed coolant injection into the RCS over the long term.

The IRWST is modeled in NOTRUMP as two fluid volumes. The upper volume contains the PRHR systems while the lower volume represents the remainder of the IRWST. The ADS stages connected to the IRWST are connected to the upper IRWST fluid volume.

The NOTRUMP analyses were performed for a range of SBLOCAs for both the SPES and OSU integral facilities. The comparisons between NOTRUMP calculations and experimental data for the SPES and OSU tests, documented in WCAP-14807, show reasonable agreement. The IRWST injection line flows, outlet flows, and PRHR inlet and outlet temperatures are predicted reasonably well. The staff finds the NOTRUMP IRWST model acceptable for analysis of the AP600 SBLOCA.

21.6.2.6 Code Qualification

Qualification, or assessment, of the NOTRUMP code and its models was carried out in three areas (1) benchmark calculations, (2) separate-effects tests, and (3) integral systems tests. The combination of benchmark calculations, separate-effects tests, and integral systems tests, when properly applied, leads to overall conclusions regarding the ability of a computer code to adequately predict the behavior of a nuclear power plant subjected to upset and accident conditions. Because no single test captures all of the relevant phenomena, it is necessary to utilize all three categories to adequately cover the phenomena of interest. The three categories are discussed below.

(1) Benchmark Calculations

Benchmark calculations are useful to demonstrate that logic interactions do not result in numeric instabilities, or physically unrealistic results. In general, these benchmark problems consist of thought problems (hypothetical problems not on the basis of data from an actual test facility) and simple nodal models to verify a particular code function or single phenomenological behavior. Assessment of the NOTRUMP code via benchmark calculations was performed for areas involving changes to the previously approved code such as the reactor coolant pump models, plus those areas involving logic changes and additions to the code. Extensive logic modifications were made, as previously discussed, involving mixture level overshoot, fluid node stacking, and bubble rise. Many of these logic models interact during the calculation of the

SBLOCA. Separate from the assessment of the overall performance of these models in predicting the integral test facility behavior, benchmark calculations were performed for the following models:

(1) NOTRUMP EPRI/flooding drift-flux model
(2) horizontal drift-flux levelizing model
(3) net volumetric flow-based momentum equation
(4) implicit treatment of gravitational head
(5) implicit treatment of bubble rise
(6) pump model
(7) fluid node stacking logic

In WCAP-14807, Westinghouse used the benchmark calculations to demonstrate that NOTRUMP calculated results for each case that agreed with the logical expectations for that case. No unrealistic model interactions were uncovered and no numeric instabilities were encountered. The overall performance of these models is discussed in Section 21.6.2.4 of this report.

(2) Separate Effects Tests - Two-Phase Level Swell

Assessment of the NOTRUMP code against separate-effects tests permits the isolation of individual models within the code such that the capabilities of the model can be determined while remaining within the context of the code.

The two-phase level swell can be an important phenomenon during an SBLOCA. Although the AP600 SBLOCA is not predicted to result in core uncovery, a two-phase mixture will exist in the upper vessel regions and therefore, the code must be capable of predicting the location of a two-phase level. The two-phase level swell model extensions to accommodate the low pressures anticipated in the AP600 SBLOCA were assessed by comparisons with data obtained in three test facilities. Westinghouse analyzed tests from the G-2 test program, from the General Electric (GE) level swell test program, and from the Achilles systems test program. The tests analyzed for validation of the two-phase level swell-related models in the NOTRUMP code are shown in Table 21.6-4.

The three test programs, given above, were chosen to encompass the anticipated pressure and flow conditions in the AP600 design for the assessment of the two-phase level swell model in NOTRUMP. The GE tests covered the intermediate pressure of 6,894.7 kPa (1,000 psia); the G-2 tests covered the range from 5,515.8 kPa (800 psia) to 101.3 kPa (14.7 psia); and the Achilles tests provided integral system data at 101.3 kPa (14.7 psia) and 202.6 kPa (29.4 psia) pressure. In addition to the coverage of the anticipated pressure range, the three test programs provided data for different scale facilities.

G-2 Core Uncovery Tests

The G-2 level swell tests were performed as part of the core uncovery tests in the Westinghouse G-2 test facility. The tests were performed as quasi-steady-state separate-effects core uncovery tests at constant pressure and power. The facility was designed to provide downflow film boiling and reflood heat transfer data under the power, flow, temperature, and pressure conditions representative of a PWR.

The G-2 tests that were selected for assessment of the NOTRUMP two-phase level swell model covered the expected AP600 small-break system pressure, from intermediate to atmospheric, and the power range. The 344.7 kPa (50 psia) and 101.3 kPa (14.7 psia) tests were especially important for modeling since they give the closest match to the conditions expected during IRWST injection in the AP600 (i.e., at the time of the minimum vessel inventory).

The NOTRUMP analysis of the G-2 tests identified leakage and recirculation paths of the bundle coolant that resulted in heat removal that were not accounted for in the break flow. Axially distributed leakage through the heater rod bundle baffle into the baffle/vessel annulus region setup an uncontrolled and unmeasured recirculation path within the test facility. Leakage tests performed in the G-2 facility in 1975 did not provide sufficient information on the baffle/vessel leakage to determine temperature dependence, variation over the test period, or nature of the flow. Thus, for the analyses of two-phase level swell, Westinghouse assumed that the leakage could be bidirectional, temperature-independent, constant flow area, and homogeneous when a two-phase mixture was leaking.

The NOTRUMP analyses of the G-2 tests were performed assuming both high and low leakage flow rates applied in the upper half of the test bundle. With the leakage bounds developed from the assumptions noted above, the NOTRUMP-predicted two-phase level is conservative relative to the test data. The NOTRUMP-predicted two-phase level with low baffle leakage is above the measured level for the early part of the tests and then predicts the level to be below the measured level. The level swell predicted for the high leakage bound remains below the measured level throughout the tests. Thus, indicating that when baffle leakage is accounted for, NOTRUMP can conservatively predict two-phase level.

General Electric Vessel Blowdown Tests

The GE small vessel blowdown facility was designed to provide two-phase level swell data during a BWR blowdown (NUREG/CR-2571/GEAP-22049 and AEEW-R2339). The facility consisted of a cylinder, 3.7-m (12-ft) in length and approximately 0.3 m (1 ft) in diameter, with five heater rods in the lower section. The tests were performed with and without a perforated plate installed to represent the internal resistance of a reactor core.

A significant problem with the data, which were collected during the tests, is the lack of break flow information. Information was not collected on the break flow rate or break flow void fraction. The tests were intended to involve only dry steam blowdown, but the analysis of the test results indicates that significant liquid entrainment occurred. This was pointed out in both the Westinghouse analysis of the test results and the analyses performed by GE using the TRAC computer code (NUREG/CR-2571/GEAP-22049).

Because of the break liquid entrainment noted above, the vessel did not contain the predicted mass of water, and therefore, gave a lower two-phase level than predicted by the computer codes. Westinghouse accounted for this mass difference in the NOTRUMP calculations by adding a lower vessel break system to remove the liquid mass to match the test program measured vessel mass. While this is an artificial mass removal method, the NOTRUMP code, when presented with a vessel mass consistent with the tests, produced a reasonable prediction of the two-phase level swell. The GE TRAC analyses indicated the same excess vessel mass as the NOTRUMP analyses, approximately 9.1 kg (20 lbs) of liquid.

The nodalization studies were performed using 6, 12, 24, and 48 nodes in the blowdown vessel. The studies found that the greatest effect occurred going from 6 to 12 nodes. Beyond that point, little change in the results occurred with the increased noding. The test analyses were performed with 12 nodes for the blowdown vessel.

Achilles Low-Pressure Level Swell Tests

The Achilles low-pressure level swell tests (Reference 24) were performed by the United Kingdom Atomic Energy Authority (UKAEA) at Winfrith as part of the general LOCA core heat transfer studies in support of the Sizewell B nuclear power station. The facility is scaled to represent a Westinghouse four-loop PWR. The core consists of 69 electrical heater rods of a diameter, pitch, and length typical of a PWR. Two of the Achilles tests, A1L066 and A1L069, were run at high power, 80 kW, and low pressure, 121.4 kPa (17.6 psia) and 202.7 kPa (29.4 psia), respectively.

The staff views the Achilles tests as important for the assessment of two-phase level swell since the calculation invokes multiple parts of the modifications made to the NOTRUMP code. The tests were run at constant pressure and power, resulting in a decreasing core mixture level with time. The calculations required use of the Yeh drift-flux correlation in the heater rod section, the SIMARC methodology in the flow links between the fluid nodes in the heater rod section, the implicit bubble rise model, the fluid node stacking logic, and the mixture level overshoot logic.

The tests were run starting with an initial water level above the top of the heated bundle. When the upper bundle thermocouple indicated heatup, the mass of the vessel was determined from differential pressure cells. The system mass was then taken as the time zero reference for the start of the NOTRUMP calculation. This technique is consistent with that used by the UKAEA in its RELAP5 analyses.

The NOTRUMP calculation of the two-phase level swell for test A1L066 indicates a conservative calculation for the majority of the test period. Late in the test, the NOTRUMP-calculated two-phase level is above the measured data, but within the data uncertainty. The NOTRUMP calculation of the two-phase level swell for test A1L069 indicates a conservative calculation for the entire period of the test.

The NOTRUMP model nodalization for the Achilles test was chosen to be consistent with that used in both the G-2 and GE tests. Axial nodalization sensitivity was examined using a range of nodding from 5 to 48 axial bundle nodes. The noding chosen for the reported calculations was 12 nodes, or one node per foot in the heated bundle. It was noted by Westinghouse that the 0.3-m (1-ft) axial node case yields a better, while conservative, fit to the test data for two-phase level swell.

Two-Phase Level Swell Conclusions

The assessment of two-phase level swell under the anticipated pressure and power conditions of the AP600 SBLOCA was difficult because of the lack of low pressure two-phase level swell data. All data currently available are from test facilities with flaws in the tests and data collection that make it necessary to make assumptions in the computer code modeling of the facilities and tests. Thus, there are no known ideal test results for assessment of the two-phase level swell capabilities of the code at low pressure. When reasonable assumptions are made to account for

the facility problems noted in this section, the NOTRUMP code does an acceptable job of predicting two-phase level swell. The results of the assessments, presented in WCAP-14807, indicate that NOTRUMP underpredicts the mixture level over a wide range of thermal-hydraulic conditions which may be found during AP600 SBLOCAs. The predicted level is consistently conservative or within the test data uncertainty. The staff finds that the tests used in the assessment of NOTRUMP two-phase level swell sufficiently test the code's capability to permit the judgment that the NOTRUMP code adequately predicts two-phase level swell at low system pressure.

(3) Integral Systems Tests

Integral systems tests permit an assessment of the entire code, including all pertinent models, acting as a unit to predict the full system behavior. Westinghouse analyzed selected SPES and OSU integral tests for the final verification and validation effort shown in Table 21.6-5. Comparison between several of the related tests assists in understanding the effects of scale on the analysis results.

Assessments were performed to compare the results of NOTRUMP calculations to data from SPES and OSU tests for a variety of transients covering a wide range of break sizes and locations. The NOTRUMP code was found to provide reasonable predictions of the highly-ranked PIRT phenomena, including the following:

- pressurizer pressure and level
- core inlet and outlet temperatures
- CMT injection flow rate and collapsed liquid level
- the steam generator collapsed liquid level as well as pressure and temperature
- the cold leg balance line levels
- the upper plenum and upper head collapsed liquid levels
- the PRHR inlet and outlet temperatures
- the break flow rate

An exception to the above-noted acceptable results is the double-ended guillotine break of a DVI line. The calculated results for core level during a double-ended guillotine break of a DVI line were nonconservative (higher) than the measured value. The staff agrees with the explanation of the root cause of this discrepancy presented by Westinghouse, that the nonconservative behavior of the code is due to the one-dimensional nature of NOTRUMP, while the phenomena taking place are two-dimensional. This issue is discussed in more detail in Section 21.6.2.4(17) of this report.

21.6.2.7 Regulatory Compliance

Following the accident at TMI, the NRC focused attention on the SBLOCA and proposed revisions to the methods and analyses performed to better demonstrate compliance with the requirements set forth in 10 CFR 50.46. With regard to Westinghouse- designed PWRs, the NRC outlined technical issues in NUREG-0611 ("Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plant") regarding the capabilities of the WFLASH computer program used to simulate the reactor coolant response to a SBLOCA. WFLASH was an early methodology that Westinghouse

developed for simulation of SBLOCA response. In NUREG-0611, the staff identified specific models in the WFLASH computer code that were considered deficient. Furthermore, the NRC issued NUREG-0737, Section II.K.3.30 to clarify the post-TMI requirements regarding SBLOCA modeling. In essence, Section II.K.3.30 of NUREG-0737 recommends that licensees of Westinghouse-design PWRs revise their SBLOCA models in accordance with the guidelines specified in NUREG-0611, or justify continued acceptance of the current model. Section II.K.3.31 further recommends that each licensee submit a new SBLOCA analysis using an approved evaluation model that meets the criteria of NUREG-0737, Section II.K.3.30.

In response to these requirements, Westinghouse developed the NOTRUMP code for reference in the new SBLOCA ECCS evaluation model calculations. As such, the NOTRUMP code was developed to overcome the deficiencies identified in the WFLASH computer program while also addressing the post-TMI requirements. Following NRC review, the NOTRUMP code was approved for evaluating SBLOCA response in Westinghouse-design PWRs.

10 CFR Part 50, Appendix K

Westinghouse modified the approved NOTRUMP code for application to the AP600 design for analysis of the SBLOCA in compliance with the requirements of 10 Part CFR 50, Appendix K. Of the many requirements specified in Appendix K, only one refers both to portions of NOTRUMP that have been modified and to phenomena that are anticipated in AP600 SBLOCAs. 10 CFR Part 50, Appendix K, Section C.2, requires that the frictional loss in pipes and other components, including the reactor core, be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. Appendix K then specifies acceptable correlations.

The friction factor calculations within the AP600 version of NOTRUMP are largely the same as found in the approved version of NOTRUMP with the exception of some smoothing and an extension of two-phase multipliers down to atmospheric pressure. As previously discussed in this report, the effectiveness of this modified model was evaluated and documented in WCAP-14807. The outcome of these evaluations show that the NOTRUMP code results agree reasonably well with data from two-phase level swell experiments (G-2, GE, and Achilles). The code results were also reasonable for the SPES and OSU test assessments. The staff finds the frictional loss model in NOTRUMP acceptable for the analysis of AP600 SBLOCA.

NUREG-0611 and NUREG-0737

Since the motivation to develop the NOTRUMP code arose from the guidance in NUREG-0611 and NUREG-0737, the modifications and applicability of the modified code to AP600 SBLOCA evaluations have been reviewed regarding the following TMI small break modeling concerns.

(1) Provide calculated validation of the SBLOCA model to adequately calculate the core heat transfer and two-phase coolant level during core uncovery conditions.

The NRC requested that the heat-up methodologies be compared to the core cooling tests performed by the Oak Ridge National Laboratory (ORNL) at its Thermal Hydraulic Test Facility (THTF). The ORNL tests provide a good database to assess the heat transfer capabilities of a

fuel rod subjected to uncovery and the resultant steam cooling conditions that can occur in the upper portion of the bundle. The tests cover a wide range of pressures and rod powers for both transient film boiling and bundle uncovery steam-cooling conditions. Predictions of two-phase level swell as well as the steam cooling convection heat transfer are essential to successful predictions of SBLOCA response.

Evaluation: The AP600 integral data do not indicate core uncovery and, therefore, the heatup model was not exercised. The assessments in WCAP-14807 of the G-2, GE, and Achilles tests provided verification of the steam cooling model and level swell in NOTRUMP during core uncovery conditions at the low pressure and low flow anticipated in the AP600 design. The staff finds the NOTRUMP code acceptable to evaluate core heat transfer and two-phase level swell in the AP600 SBLOCA.

(2) Validate the adequacy of modeling the primary side of the steam generators as a homogeneous mixture.

It is necessary to demonstrate that there is sufficient spatial detail to model the primary and secondary systems to properly account for forward and reverse heat transfer as liquid drains from the primary active tubes. Because the steam generators can act as a heat source following some SBLOCAs, proper accounting for the steam-water behavior and associated depressurization rates is required.

Proper accounting of the annular and slug flow regimes as they may occur in the steam generators should be incorporated into the modeling of this region. One must, therefore, ensure that the flow region behavior in the generators is consistent with the heat transfer conditions throughout the transient. If there is a potential for flooding or "hold-up" of the liquid in the generators, then the hydraulic model should also account for this behavior. Also, if a stratified flow model is used in the hot-leg piping, this flow regime should be justified.

Evaluation: Steam generator heat transfer has a secondary role in AP600 since the PRHR system functions along with the ADS to control system pressure and depressurize the plant. This low importance of steam generator heat transfer is reflected in the low PIRT ranking. nevertheless, the models contained in NOTRUMP are applicable to the AP600 plant performance.

In the AP600, the PRHR HX functions in a manner similar to the steam generators as a major heat removal system. The PRHR uses the same models; thus, the staff considered its modeling under this item. The PRHR heat transfer was not properly predicted in the SPES and OSU comparisons since the code was unable to predict the correct outlet temperature. The NOTRUMP code tends to overpredict the outlet temperature. This result is conservative; therefore, the staff finds the NOTRUMP model for the steam generator and PRHR system acceptable.

NOTRUMP contains provisions for stratified and dispersed flow regimes in the loop piping and steam generators. The flooding models in the code capture the potential for liquid hold-up in the loop and steam generators should steam velocities be sufficient to entrain and limit drainage in the loops. The PIRT ranking for flow-regime-related phenomena is low, so that this phenomenon does not appear to have a significant impact on AP600 performance.

Comparisons between the NOTRUMP calculated fluid conditions and the measured fluid conditions from SPES and OSU tests demonstrate that the models used in NOTRUMP for the stratified and dispersed flow regimes are appropriate. The staff finds the NOTRUMP models for the stratified and dispersed flow regimes in the hot leg piping acceptable for analysis of the AP600 SBLOCA.

(3) Validate the condensation heat transfer model and effects of non-condensable gases.

The condensation correlation used in the blowdown hydraulics code must be justified as to the applicability to the two-phase flow conditions in the active tubes of the steam generators. The need for a best-estimate correlation was stressed as opposed to empirical relationships containing "conservatisms." Particular emphasis should be placed on the applicability of the correlations to U-tube steam generators since most correlations used to date are on the basis of flat-plate geometries. Non-condensable gases should also be accounted for.

Evaluation: Westinghouse added the Shah condensation correlation to the NOTRUMP condensation correlation package previously approved. As discussed in Section 21.6.2.4(13) of this report, the staff finds the addition of the Shah correlation acceptable for analysis of the AP600 SBLOCA.

The NOTRUMP code can not calculate the effects of the non-condensable gases injected into the primary coolant system during the AP600 SBLOCA. The presence of non-condensable gases is of concern because of the possible degradation in performance of the PRHR HX for system depressurization and heat removal. During the conduct of the test program, non-condensable gases entered the system but were not tracked as they moved through the system. The gases either exited the system or were found to end up in the PRHR HX or the CMTs. It was noted that the non-condensable gases entered the PRHR HX late in the transient, when the PRHR system no longer had a significant role in heat removal. At this point in the transient, the non-condensable gases do not appear to have a detrimental effect on the system. Since the non-condensable gases do not play a role in the AP600 SBLOCA, the staff accepts the NOTRUMP code for evaluation of AP600 SBLOCA in spite of its inability to calculate the effects of non-condensable gases. This position will be re-evaluated if scenarios are found which cause non-condensable gases to reach the PRHR HX while it is actively removing heat from the primary system.

(4) Demonstrate, through noding studies, as part of the sensitivity studies, the adequacy of the SBLOCA model to calculate flashing during system depressurization.

Evaluation: The adequacy of the NOTRUMP code to model the system effects, as well as local fluid conditions, during the AP600 SBLOCA is demonstrated through the consistency in noding between the scaled, integral system test facilities (SPES-2 and OSU) and the AP600 design. The staff finds the noding acceptable for analysis of the AP600 SBLOCA on the basis of analyses performed on the noding differences used in the PRHR and downcomer models for the test facilities and AP600.

(5) Validate the polytropic expansion coefficient applied in the accumulator model.

Evaluation: The accumulator model was not changed from the approved NOTRUMP code to the AP600 version of the code. Also, the accumulator in the AP600 design is similar to that

employed in the current generation operating Westinghouse PWRs. The staff therefore accepts the model as it is applied to the AP600 SBLOCA.

(6) Break discharge model

Since Appendix K to 10 CFR Part 50 requires use of the Moody critical flow model, the blowdown hydraulic code must contain this methodology.

Evaluation: The Moody critical model is used in the NOTRUMP code as specified by Appendix K (10 CFR Part 50) for saturated break flow. It is noted that the flow rate out the ADS is being predicted through the addition of the Henry/Fauske and HEM critical flow models previously discussed. Because the ADS is a depressurization system and not an actual break, this is considered acceptable based in part on the precedents established by the depressurization system used in the boiling water reactors. The staff finds that Westinghouse demonstrated that the treatment of the ADS is conservative.

(7) Validate the SBLOCA model with loss-of-fluid test (LOFT) facility tests L3-1 and L3-7. In addition, validate the model with the Semiscale S-UT-08 experimental data.

There is a need for integral as well as separate-effects test comparisons. The NRC identified the LOFT and Semiscale integral-system tests which should be included as part of the code verification process. These tests include LOFT and Semiscale integral system tests addressing SBLOCA transients, including an examination of the continued operation of the main coolant pumps on the system response following initiation of a small break (L3-6). Semiscale S-07-10D was also identified as an integral test which could be used in the benchmarking of codes against a SBLOCA transient where long-term core uncovery was simulated.

Evaluation: The purpose of this requirement was to demonstrate the ability of the code to adequately deal with plugging and clearing of the steam generator to the reactor coolant pump loop seal. The AP600 design eliminated the loop seal, as well as placing the steam generators entirely above the hot-leg reactor vessel nozzle, which is above the reactor core. Accordingly, the staff finds that this requirement is not applicable to the analysis of the AP600 SBLOCA since there is no loop seal to prevent the steam generator tube contents from flowing into the reactor vessel. The staff notes that Westinghouse provided other integral effects assessments of NOTRUMP that address the highly ranked PIRT items.

The staff is aware that Westinghouse submitted modifications to the NOTRUMP code incorporating a condensation model on the basis of results of the COSI safety injection (SI)/steam condensation experiments. The COSI test facility is a scaled representation of the cold-leg and SI injection ports in a Westinghouse designed PWR. The pressure range covered by the COSI tests is outside of the range of interest for the low-pressure conditions expected in the AP600 SBLOCA. In addition, the AP600 design uses direct vessel injection for SI. Accordingly, the staff position is that the COSI condensation model is neither applicable nor acceptable for evaluation of the AP600 SBLOCA.

In addition to the above modeling concerns, NUREG-0737 recommendations indicate that the effect of the operation of the main coolant pumps on SBLOCA response should be assessed. In Section 7.3.2.2.2 of the SSAR, Westinghouse indicates that a safety-grade, single-failure- proof,

reactor coolant pump trip is provided. As such, Westinghouse is not required to evaluate AP600 performance with the main coolant pumps operating.

21.6.2.8 ACRS Review

Two meetings were held with the ACRS Thermal-Hydraulic Subcommittee for review of the NOTRUMP code. Those meetings resulted in numerous additional review items and concerns. As a result the staff required that Westinghouse fully document the code numerics, providing detailed derivations of all equations modified or changed from the source form to the difference form as applied in the code. This is in addition to fulfilling the commitments Westinghouse made during the meetings. Subsequently, Westinghouse documented responses to the following six issues:

(1) Momentum Flux - Deficiencies were benchmarked against additional detailed calculations using actual two-phase flow equations that include the effects of compressibility and the condition of constant entropy.

(2) ADS 1-3 - The test data analysis report was revised to show that the data reduction was performed correctly.

(3) Entrainment - Entrainment was considered as part of the overall scaling and IRWST-level penalty development.

(4) IRWST-Level Penalty - A multiloop scaling analysis was performed for the time period of ADS 4 and IRWST draining. The basis for ADS flow was justified, along with ADS 4 flow affected by entrainment of liquid and the corresponding effect on the pressure loss as a result of two-phase flow.

(5) Pressurizer Surge Line Flooding - An evaluation similar to that applied to the IRWST level penalty was performed.

(6) Noding - Additional justification was provided for the basis used which differs from the accepted approach developed under the CSAU work. This applies in particular to the PRHR and downcomer.

Several of these items involve phenomena that are not well represented or modeled in NOTRUMP, because of the structure of the code. Nonetheless, overall code calculations of the plant's performance show large margins to licensing limits, and Westinghouse has addressed the issues in a conservative fashion. Therefore, the staff concludes that these issues do not alter the staff's determination that NOTRUMP is suitable for analyzing the behavior of the AP600.

21.6.2.9 Resolution of Open and Confirmatory Items

A large number of open items were noted in the DSER and SDSER. In addition, more than 150 RAIs were issued during the review of the applicability of NOTRUMP to the AP600 SBLOCA. Westinghouse responded to all of the RAIs. The following is a discussion regarding the resolution of the open and confirmatory items identified in the SDSER.

Open Item Discussion

Open Item	Discussion
21.6.2.2-1	Westinghouse needs to identify which information from the NOTRUMP-related RAI responses will be formally incorporated into NOTRUMP-related documentation such as the final verification and validation report, the code applicability document (WCAP-14206), or the SSAR. Westinghouse appended the RAI responses to the NOTRUMP Final Verification and Validation Report, WCAP-14807, Revision 2. This item is closed.
21.6.2.2-2	Westinghouse needs to submit the final verification and validation report. The NOTRUMP Final Verification and Validation Report, WCAP-14807, Revision 2, was submitted. This item is closed.
21.6.2.4-1	Westinghouse needs to explain what provision will be used to ensure that volumetric-based momentum equations will be used for all AP600 calculations. The volumetric-based momentum equations is the default form of the momentum equations for the AP600 version of NOTRUMP. The volumetric-based momentum equations will be the default form of the momentum equations for the AP600 version of NOTRUMP, and Westinghouse will implement administrative controls to ensure that they are used. This item is closed.
21.6.2.4-2	Westinghouse needs to submit the NOTRUMP assessment cases to demonstrate the adequacy of the re-casting of the momentum equation and drift flux equations in net volumetric form. Westinghouse submitted assessment cases of the adequacy of the net volumetric form of the momentum equation as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the use of the net volumetric form of the momentum equation is discussed elsewhere in this report. This item is closed.
21.6.2.4-3	Westinghouse needs to submit the assessment cases to demonstrate the acceptability of modifications to the transient terms in the momentum equation of NOTRUMP. Westinghouse discontinued work on modifications to the transient terms in the momentum equation as applied in NOTRUMP. This item is closed.

21.6.2.4-4 Westinghouse needs to explain what provision will be used in NOTRUMP to ensure that options to override the default flow partitioning will be used for all AP600 calculations.

The volumetric-based momentum equations is the default form of the momentum equations for the AP600 version of NOTRUMP. The volumetric-based momentum equations will be the default form of the momentum equations for the AP600 version of NOTRUMP, and Westinghouse will implement administrative controls to ensure that they are used. This item is closed.

21.6.2.4-5 Westinghouse needs to complete all benchmark and assessment calculations (to be included in the FV&V report) to demonstrate the acceptability of the logic modifications for application of the NOTRUMP code to the AP600 SBLOCA.

Westinghouse submitted all the required benchmark and assessment calculations as a part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the assessments and of the logic modifications are discussed in Section 21.6.2.4, Items (7), (8), and (17), of this report. This item is closed.

21.6.2.4-6 Westinghouse needs to determine whether to use additional separate-effects level swell tests to support the qualification of the NOTRUMP code to address the code's capability to predict two-phase level swell and system mass inventory. (See Open Item 21.6.2.6-2.)

Westinghouse adopted the use of several assessment cases from the G-2, GE, and Achilles tests for assessment of the two-phase level swell and system mass inventory capability of NOTRUMP. The acceptance of the assessments and models are discussed in Section 21.6.2.6, Item (2), of this report. This item is closed.

21.6.2.4-7 Westinghouse needs to submit benchmark calculations to demonstrate that the modified pump model is reasonable for application of NOTRUMP to the AP600 SBLOCA.

Westinghouse submitted benchmark calculations as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2, which demonstrate the adequacy of the modified pump model for the AP600 design. The acceptance of the model is discussed in Section 21.6.2.4, Item (9), of this report. This item is closed.

21.6.2.4-8 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the changes made to the NOTRUMP gravitational head term, and applicability to the AP600 SBLOCA.

Westinghouse submitted benchmark calculations as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2, which

demonstrate the adequacy of the modifications made to the gravitational head terms. The acceptance of this model change is discussed in Section 21.6.2.4, Item (10), of this report. This item is closed.

21.6.2.4-9 Westinghouse needs to submit benchmark calculations (to be included in the FV&V) to demonstrate the acceptability of the model changes and additions.

Westinghouse submitted benchmark assessment calculations as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2, that demonstrate the adequacy of the NOTRUMP model changes and additions. The acceptance of the model changes and additions are discussed in Section 21.6.2.6, Item (1), of this report. This item is closed.

21.6.2.4-10 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the adequacy of the NOTRUMP birthing logic and its applicability to the AP600 SBLOCA.

Westinghouse discontinued the use of the birthing logic in the NOTRUMP code. This item is closed.

21.6.2.4-11 The NOTRUMP FV&V report and assessment calculations need to demonstrate the acceptability of the Zuber critical heat flux correlation for AP600 SBLOCA analysis.

Westinghouse demonstrated the adequacy of the Zuber critical heat flux correlation, as modified by Lienhardt and Dhir, in the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of this model is discussed in Section 21.6.2.4, Item (14), of this report. This item is closed.

21.6.2.4-12 The NOTRUMP FV&V report needs to demonstrate the acceptability of the smoothing logic.

Westinghouse demonstrated the adequacy of the NOTRUMP smoothing logic in the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the smoothing logic is discussed in Section 21.6.2.4, Item (1), of this report. This item is closed.

21.6.2.4-13 Westinghouse needs to submit the assessment calculations to demonstrate acceptable logic operation and logic interactions during the FV&V of the AP600 NOTRUMP code.

Westinghouse addressed the adequacy of logic model operation and interaction through the assessment calculations submitted as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. This item is closed.

21.6.2.5-1 Westinghouse needs to address the models affecting the fluid entering the ADS piping, particularly for the hot legs and pressurizer in the FV&V report.

Westinghouse addressed the reanalysis of the OSU and SPES-2 test data and demonstrated acceptable, conservative prediction of the ADS performance as discussed in Section 21.6.2.5(1) of this report. This item is closed.

21.6.2.5-2 Westinghouse needs to investigate the NOTRUMP code's inability to properly characterize CMT thermal stratification and to better explain some of the differences in CMT discharge flow comparisons.

The NOTRUMP code's inability to properly characterize CMT thermal stratification is discussed in Section 21.6.2.5, Item (2), of this report. Since the overall effect is believed to result in a conservative calculation of the SBLOCA for the AP600, this inability is accepted. This item is closed.

21.6.2.5-3 Westinghouse needs to submit its reanalysis of previously analyzed component separate-effects tests that are listed in Table 21.7 to demonstrate the acceptability of these tests.

Westinghouse reanalyzed the selected separate effects tests as documented in the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the analyses is discussed in Section 21.6.2.5 of this report. This item is closed.

21.6.2.6-1 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the NOTRUMP model changes and additions for which these benchmark calculations are to be performed.

Westinghouse submitted the benchmark calculations demonstrating the adequacy of the NOTRUMP model changes and additions as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the benchmark calculations is discussed in Section 21.6.2.6 of this report. This item is closed.

21.6.2.6-2 Westinghouse needs to demonstrate the overall adequacy of the separate-effects testing relating to level swell and void fraction distributions in the NOTRUMP code (see Open Item 21.6.2.4-6).

Westinghouse demonstrated through the assessment calculations submitted in the NOTRUMP Final V&V Report, WCAP-14807, Revision 2, that the two-phase level swell and void fraction distributions are adequate. The acceptance of these assessments and models are discussed in Section 21.6.2.6, Item (2), of this report. This item is closed.

21.6.2.6-3 Westinghouse needs to submit reanalysis of the integral systems tests listed in Table 21.10.

Westinghouse submitted reanalyses of the requested integral systems tests as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the analyses is discussed in Section 21.6.2.6, Item (3), of this report. This item is closed.

21.6.2.7-1 Westinghouse needs to submit PRHR primary side heat transfer comparisons between NOTRUMP and OSU/SPES-2 data in the FV&V report.

Comparisons were performed between the PRHR primary side heat transfer predicted by NOTRUMP and those calculated from the data for OSU and SPES-2. Acceptable prediction of the heat transfer calculated from test data was demonstrated. See discussions in Sections 21.5.3 and 21.6.2.5(3) of this report. This item is closed.

21.6.2.7-2 The NOTRUMP FV&V report needs to address the effects of non-condensable gases on PRHR heat transfer.

While the non-condensable gases cannot be tracked by the NOTRUMP code, the non-condensable gases were shown (through the test programs) to enter the system late in the SBLOCA transients when the PRHR is no longer a significant source of heat removal. Since the non-condensable gases do not play a role in the AP600 SBLOCA, the staff accepts the NOTRUMP code for evaluation of AP600 SBLOCA in spite of its inability to calculate the effects of non-condensable gases. This item is closed.

21.6.2.7-3 Westinghouse needs to clarify whether and how it will use the COSI condensation model in the AP600 NOTRUMP code.

The COSI condensation model is not used in the AP600 NOTRUMP code. This item is closed.

Confirmatory Item Discussion

21.6.2.4-1 The application of the SIMARC drift-flux technology is acceptable for the analysis of the AP600 SBLOCA pending confirmation of the model through the benchmark and assessment of the code to be provided in the FV&V report.

Westinghouse demonstrated the adequacy of the SIMARC drift-flux methodology for the analysis of the AP600 SBLOCA through assessment and benchmark cases submitted as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the methodology is discussed in Section 21.6.2.4, Item (1), of this report. This item is closed.

21.6.2.4-2 The modifications made to the NOTRUMP drift-flux correlations are acceptable for analysis of the AP600 SBLOCA pending confirmation of the model through the benchmark and assessment of the code to be provided in the FV&V report.

Westinghouse demonstrated the adequacy of the modifications made to the drift-flux correlation used in the NOTRUMP code through the assessment and benchmark calculations submitted as part of the NOTRUMP Final V&V Report, WCAP-14807, Revision 2. The acceptance of the model changes is discussed in Section 21.6.2.4, Item (2), and Section 21.6.2.6, Item (2), of this report. This item is closed.

21.6.2.4-3 Westinghouse needs to verify that the NOTRUMP code does not use the Bjornard and Griffith modification.

The Bjornard and Griffith modifications to the Zuber critical heat flux correlation is not used in the NOTRUMP code. NOTRUMP uses the Lienhardt and Dhir modifications to the Zuber critical heat flux correlation which is acceptable to the staff as discussed in Section 21.6.2.4(14) of this report. This item is closed.

21.6.2.4-4 Westinghouse needs to verify that heat link methodology for transition boiling is not used in AP600 NOTRUMP calculations.

Westinghouse verified that the heat link methodology for transition boiling is not and will not be used in the core region of the AP600 NOTRUMP calculations because the core is not predicted to be subjected to transition boiling. Should any future licensing calculations predict the occurrence of transition boiling in the core, review of this methodology will be re-opened. Administrative controls will be used to prevent use of the heat link methodology for AP600 core calculations. See discussion in Section 21.6.2.4(18) of this report. This item is closed.

21.6.2.5-1 The acceptability of the PRHR model used in the NOTRUMP code is contingent on a finding that the PRHR data are applicable (see Section 21.3.3 of the SDSER).

Comparisons were performed between the PRHR primary side heat transfer predicted by NOTRUMP and those calculated from the data for OSU and SPES-2. An acceptable prediction of the heat transfer calculated from test data was demonstrated. See discussions in Sections 21.5.3 and 21.6.2.5(3) of this report. This item is closed.

21.6.2.7-1 Comparisons of the NOTRUMP code simulations to the OSU and SPES-2 test data in the FV&V report should confirm the applicability or insensitivity of the NOTRUMP flow regime models to the key system response parameters.

The key systems parameters, such as system pressure and component levels, show generally good comparisons between the measured OSU and SPES-2 test data and the NOTRUMP code predictions. The parameters identified as being of high importance in the various tests in the PIRT are conservatively predicted. See discussion in Section 21.6.2.6(3) of this report. This item is closed.

21.6.2.10 Conclusions

The NOTRUMP computer code was developed by Westinghouse to assess the consequences of an SBLOCA. The code was modified through introduction of model additions and changes in 18 of the approved code's models. In addition, component models for the ADS, CMT, PRHR HX, and IRWST were added to make the code applicable to the AP600 passive reactor design. The staff review of the code's application to the AP600 SBLOCA, the component test program, and the integral systems tests resulted in a large number of RAIs. Westinghouse responded to the RAIs and documented the responses in the NOTRUMP Final V&V Report, WCAP-14807.

Additional assessment calculations were considered important to the assessment of the level swell models in NOTRUMP. The additional requests for benchmarking were on the basis of the lack of level swell benchmarks provided by Westinghouse in the documentation and the non-conservative predictions displayed by NOTRUMP in several of the SPES and OSU tests. The NOTRUMP code, in these cases, overpredicted the liquid inventory in the core and upper plenum regions of the reactor vessel. Because there was a basic lack of low-pressure data to qualify codes for level-swell phenomena, it was the staff's position that additional tests be analyzed for model qualification.

The staff expressed concerns regarding the assessment of many of the models modified in the approved NOTRUMP code. In particular, the changes to the drift-flux models, bubble rise model, and momentum equations significantly alter the two-phase level swell capabilities of the code. An adequate assessment of the two-phase level swell is essential to properly understand the predictions of the code in an SBLOCA situation since it is a depressurizing, two-phase condition. Westinghouse performed numerous assessments of the logic models and the two-phase level swell models to demonstrate the adequacy of the models in predicting two-phase level and void fraction distribution in the AP600 SBLOCA.

In addition, the staff expressed concern about the extensive logic models added to the code to control mixture level, region birthing, etc. Westinghouse was required to demonstrate that the interaction of the logic models does not lead to unrealistic results. Also, the staff required Westinghouse to demonstrate that mass and energy are conserved as mass and energy and are redistributed when mixture regions pass through flow links. The "mechanical" movement of mass and energy in these logic schemes suggested that the models be exercised through the benchmark calculations to assure that the conservation laws are not being violated.

Westinghouse added options to NOTRUMP to permit use of the momentum equation in volumetric form and flow partitioning in the analysis of the AP600 SBLOCA. The staff does not consider the "options" added to improve the performance of NOTRUMP in analyzing the AP600 SBLOCA to be options. The staff position is that the "options" added to NOTRUMP for AP600 SBLOCA analyses are required to be used for those analyses.

Because transition boiling is not expected to occur in the AP600 core under SBLOCA conditions, the changes in the numerical solution techniques used in the NOTRUMP heat links when transition boiling is predicted to occur were not reviewed. Therefore, this methodology may not be invoked in the application of the NOTRUMP code to AP600 calculations. Should NOTRUMP be applied to calculations for which this methodology is being invoked, the review of the modified transition boiling correlation solution scheme will be revisited.

The staff notes that the NOTRUMP code cannot calculate the effects of non-condensable gases injected into the primary coolant system during the AP600 SBLOCA. Non-condensable gases enter the PRHR late in the transient, when the PRHR HX no longer has a significant role in heat removal. Thus, the non-condensable gases do not appear to have a significant effect on the course of the event. The staff accepts the NOTRUMP code for evaluation of the AP600 SBLOCA in spite of this shortcoming. However, if scenarios are found which cause non-condensable gases to reach the PRHR HX while it is actively removing heat from the primary system, NOTRUMP cannot be used to analyze those scenarios.

Notwithstanding the limitations that the staff identified in its review of the application of the NOTRUMP code to analyses of the AP600 design and the conditions that Westinghouse must observe as it applies the code, the staff has confidence that the use of NOTRUMP is acceptable for AP600. This is because the phenomena expected during a SBLOCA are modeled reasonably well in the test facilities, code comparisons with the experiments are reasonable, and they indicate that there are large margins to licensing limits which are unlikely challenged by uncertainties in the code models.

In a letter dated February 27, 1998, Westinghouse submitted Revision 4 to WCAP-14807, "NOTRUMP AP600 Final Verification and Validation Report." Therefore, with the limitations and conditions described in this report, the staff concludes that the NOTRUMP code has been appropriately modified to include the features necessary to model the AP600 plant and the phenomena expected during an AP600 SBLOCA. Therefore, it can be applied to the AP600 passive reactor design.

21.6.3 WCOBRA/TRAC Computer Code for Large-Break LOCAs

The staff has evaluated Westinghouse's use of the WCOBRA/TRAC computer code in the AP600 design for LBLOCAs, as discussed below.

21.6.3.1 Introduction

In 1988, the NRC revised 10 CFR 50.46 to allow the use of realistic/best estimate (BE) computer models in calculating ECCS performance. The approach allowed realistic computer models to be used to calculate a nuclear power plant's response to a LBLOCA, provided the uncertainty in the calculated results was determined. The uncertainty was to be added to the calculated results, including the peak cladding temperature (PCT), when comparing the ECCS performance to the requirements of 10 CFR 50.46. Westinghouse submitted to the NRC a realistic methodology for performing LBLOCA analyses of Westinghouse three- and four-loop PWRs with cold-leg injection, WCAP-12945-P, "Code Qualification Document for Best Estimate LOCA Analysis," Volumes 1 to 5, dated June 1992 to June 1993. In the following discussion, WCAP-12945-P is referred to as the Code Qualification Document (CQD). NRC approved WCAP-12945-P for use in licensing analyses of three- and four-loop PWRs with cold-leg

injection in a letter from the NRC to Westinghouse, "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. Westinghouse has submitted WCAP-14171-P, " WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," to document the application of WCOBRA/TRAC, together with Westinghouse's modifications to the approved realistic BE methodology to the AP600 LBLOCA. Westinghouse's responses to staff RAIs and discussion items regarding their realistic BE LBLOCA modeling applications were also included in this evaluation.

The realistic BE LBLOCA models are designed to show conformance of the ECCS to 10 CFR 50.46 requirements. Westinghouse submitted the AP600 methodology to the NRC for review and acceptance as a method to analyze LBLOCAs in a manner that conforms to NRC requirements in 10 CFR 50.46; guidance contained in RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance, Regulatory Guide," dated May 1989; and the Code Scaling, Applicability, and Uncertainty (CSAU) methodology, NUREG/CR-5249, "Quantifying Reactor Safety Margins," EGG-2552, December 1989.

This safety evaluation documents the results of the staff review of the Westinghouse realistic BE LBLOCA methodology for licensing analyses of the Westinghouse AP600. The AP600 review includes the method for compliance with the guidance contained in RG 1.157, the CSAU methodology, and how the method is used to meet the requirements and acceptance criteria of 10 CFR 50.46.

21.6.3.2 Material Reviewed

The following AP600-specific material was considered in this review:

- WCAP-14171-P, " WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," Revision 2, dated March 24, 1998
- Westinghouse letter NTD-NRC-95-4598, dated November 17, 1995
- Westinghouse letter NSD-NRC-96-4908, dated December 10, 1996
- Westinghouse letter NSD-NRC-97-5171, dated June 10, 1997
- Westinghouse letter NSD-NRC-97-5240, dated July 18, 1997
- Westinghouse letter NSD-NRC-97-5291, dated August 27, 1997
- Westinghouse letter NSD-NRC-97-5332, dated September 8, 1997

WCAP-14171-P is the basis for the review. In that report, Westinghouse provided the AP600 LBLOCA PIRT and compared the AP600 LBLOCA response to that for the North Anna plant. The results of the AP600 specific WCOBRA/TRAC code assessments were also presented. Finally, Westinghouse described how it would apply the WCOBRA/TRAC realistic methodology to the AP600. In particular, Westinghouse described the changes and simplifications made to the approved methodology for the application to the AP600 analysis. The main difference from

the approved methodology is the simplification of the uncertainty analysis by including a larger number of bounding parameters. Westinghouse chose this approach because of the large margin available in the AP600 relative to the 10 CFR 50.46 PCT limit.

This review also considered the information provided by the Westinghouse letter responses listed above to clarify a number of discussion items related to WCAP-14171-P. These discussion items covered a wide range of topics including the AP600 PIRT; AP600 assessments; and the AP600 uncertainty methodology including new parameter distributions, methodology changes and simplifications, and applicability of the approved methodology to the AP600. WCAP-14171-P, Revision 2, was issued in February 1998 to incorporate the Westinghouse discussion item responses, as well as document the application and methodology restrictions listed in Section 21.6.3.17 of this report.

21.6.3.3 Summary of 10 CFR 50.46 Review

10 CFR 50.46 is the legal basis for realistic analysis of ECCS performance. This section summarizes how Westinghouse's AP600 methodology meets the requirements of 10 CFR 50.46, which describes the ECCS acceptance criteria for light water reactors.

The portion of 10 CFR 50.46(a)(1)(i), that was the focus of this review states the following:

> Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded.

This indicates the following four concerns:

(1) the analytical technique must realistically describe reactor LOCA behavior

(2) comparisons to applicable experimental data must be made

(3) uncertainties in the analysis method and inputs must be assessed so the uncertainties in the calculated results can be estimated

(4) the uncertainty must be accounted for when comparing the calculated ECCS performance to the criteria set forth in Paragraph (b) of 10 CFR 50.46 so that there is a high level of probability that the criteria would not be exceeded

In NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996, the staff found the Westinghouse methodology for three- and four-loop plants met the above requirements. First, The staff found that WCOBRA/TRAC realistically describes the behavior of a PWR during a LBLOCA on the basis of a review of the models and correlations

and the code assessment results. Also, The staff reviewed all uncertainty distributions, response surface generation, and their applications in determining the 95th percentile PCT. Uncertainty because of reactor input parameters was also reviewed.

In WCAP-14171-P, Westinghouse showed the applicability of the WCOBRA/TRAC code to the AP600 and described the modifications made to the approved methodology for AP600. Westinghouse showed the similarity of the AP600 and three- and four-loop plant LBLOCA response and provided DVI assessments. Because of the similarity in LBLOCA response, it was concluded that the previous NRC code and code assessment review for the approved methodology also applied to AP600. This fact and the DVI assessments lead to the conclusion that WCOBRA/TRAC realistically describes the behavior of the AP600 during a LBLOCA. The review of the similarity of the AP600 and three- and four-loop plant LBLOCA response is discussed in Section 21.6.3.4 of this report, and the DVI assessments are discussed in Section 21.6.3.7 of this report. Therefore, for the AP600, Westinghouse has satisfied the 10 CFR 50.46(a)(1)(i) requirements that the methodology realistically describe reactor behavior during a LBLOCA and that comparisons to applicable data be made.

On the basis of the information provided by Westinghouse concerning the modifications to the approved uncertainty methodology for AP600, the staff concluded that Westinghouse had adequately justified the changes. This review is discussed in Sections 21.6.3.9 and 21.6.3.12 of this report. Therefore, for the AP600, Westinghouse has satisfied the 10 CFR 50.46(a)(1)(i) requirements for quantifying and accounting for the uncertainty when comparing the calculated ECCS performance to the criteria of 10 CFR 50.46(b).

Section 50.46(b) states the five acceptance criteria for the ECCS. The ECCS must ensure the following:

- the PCT is less than 1204°C (2200°F) [10 CFR 50.46(b)(1)]
- the maximum local cladding oxidation does not exceed 17 percent of total cladding thickness before oxidation [10 CFR 50.46(b)(2)]
- the maximum core-wide hydrogen generation does not exceed 1 percent [10 CFR 50.46(b)(3)]
- the core geometry remains coolable [10 CFR 50.46(b)(4)]
- long-term cooling shall be ensured [10 CFR 50.46(b)(5)]

Westinghouse's realistic LBLOCA methodology for AP600 meets these criteria as follows. The review of the PCT methodology found it adequate to meet NRC requirements for realistic LBLOCA analyses. Westinghouse's methods for determining the 95th percentile PCT are the subject of the review summarized and discussed in a number of sections of this report. The AP600 calculated 95th percentile PCT is discussed in Section 15.2.6.5 of this report.

Westinghouse applies the approved methodology to AP600 to show compliance with 10 CFR 50.46(b)(2). Because the AP600 has similar fuel and a similar LBLOCA response to plants for which WCOBRA/TRAC was previously approved, and because of the low calculated

PCT, the staff has determined that the approach for the approved methodology in the area of local oxidation calculation is applicable to AP600.

For core-wide oxidation, the AP600 methodology provides for evaluation of whether oxidation is significant at the estimated PCTs. Oxidation in not significant below 982°C (1800°F). Because the AP600 has a low estimated PCT, at or below 941°C (1725°F), and corresponding low local oxidation estimates, the staff concludes that an evaluation approach for the core-wide oxidation calculation is applicable to AP600. Thus, it was concluded that no further action is required in this area.

Westinghouse stated that meeting 10 CFR 50.46(b)(1) and (2) ensures that a coolable core geometry is maintained per 10 CFR 50.46(b)(4). In view of the foregoing discussion on these criteria, the staff finds this acceptable. This is consistent with the approved methodology and consistent with 10 CFR Part 50, Appendix K, Evaluation Models, and use of a realistic LBLOCA methodology for the PCT and results of oxidation calculations would not change this conclusion.

On the basis of the above, the staff concludes that for AP600 Westinghouse has met those portions of 10 CFR 50.46, Part (a)(1)(i) that were the focus of this review. 10 CFR 50.46(b)(5), long-term cooling, was reviewed separately in Section 21.6.4 of this report.

21.6.3.4 Similarity of AP600 and North Anna LBLOCA Responses

One of the main points made by Westinghouse in its discussion of WCOBRA/TRAC applicability to AP600 LBLOCA is that AP600 responds similarly to three- and four-loop operating plants in regards to a LBLOCA. In WCAP-14171-P, Westinghouse made a comparison of the AP600 LBLOCA response to that for the North Anna plant. This comparison is important because many times during the course of the review this similarity is cited as the basis for the applicability of WCOBRA/TRAC and the approved methodology to AP600. This section discusses Westinghouse's comparison first, followed by the results of the review of that comparison.

21.6.3.4.1 Comparison of AP600 and North Anna LBLOCA Responses

In WCAP-14171-P, Westinghouse compared the AP600 LBLOCA response to that for the North Anna plant, a three-loop PWR with 17 x 17 fuel assemblies. Westinghouse stated this plant was chosen because the fuel assembly design and vessel size are similar to that for the AP600. The results presented showed that the AP600 LBLOCA response is similar for the blowdown, refill, and reflood phases of a North Anna LBLOCA. During blowdown, the initial fuel heatup is followed by a blowdown cooling period. Accumulator injection begins during blowdown for both plants, and both plants experience a period of ECC bypass. During refill, the lower plenum inventory is replenished by accumulator water while the core heats up. Reflood begins when the lower plenum level reaches the bottom of the core, and it is a bottom-up reflood for both plants.

Westinghouse did note some differences in the two plant responses. First, AP600 includes CMTs that begin to inject at approximately 2 seconds into the accident because of a safety signal generated at 1 second on the basis of high containment pressure. This flow lasts until 12.5 seconds when the accumulators start to inject. Regarding the CMT flow, Westinghouse notes that the amount of liquid injected is small (0.5 percent of the total CMT inventory), and the injection occurs during the ECC bypass period. Therefore, Westinghouse concluded the CMT injection does not impact core cooling in an AP600 LBLOCA.

Second, Westinghouse noted that the AP600 accumulators are designed to provide extended coverage of a LBLOCA. This was done by designing the AP600 accumulator to provide reduced flow and larger water volumes relative to North Anna. This provides injection flow for a longer period during a LBLOCA.

Finally, the calculated AP600 response during blowdown cooling is considerably better than the calculated North Anna response in that more core downflow occurs with significant rewetting of the fuel cladding. Westinghouse stated this was because of the larger flow area between the upper head and the upper plenum in AP600 relative to that in North Anna. The larger flow area causes more downflow through the core. As a result, WCOBRA/TRAC calculated quenching during blowdown of the low power and average channel rods in the WCAP-14171-P analyses.

21.6.3.4.2 Review of AP600 and North Anna LBLOCA Comparison

As noted above, Westinghouse compared the AP600 LBLOCA response to that for the North Anna plant, a three-loop PWR with 17 x 17 fuel assemblies. The comparison found that the AP600 LBLOCA response is similar for the blowdown, refill, and reflood phases of a North Anna LBLOCA. The following three issues were discussed by Westinghouse:

(1) During blowdown, the initial fuel heatup is followed by a blowdown cooling period. Accumulator injection begins during blowdown for both plants, and both plants experience a period of ECC bypass.

(2) During refill, the lower plenum inventory is replenished by accumulator water while the core heats up.

(3) Reflood begins when the lower plenum level reaches the bottom of the core, and it is a bottom-up reflood for both plants.

Of the differences noted by Westinghouse, the CMT injection was not a concern because the amount of liquid injected is small, and the injection occurs during the ECC bypass period which can only be conservative with respect to 10 CFR 50.46(b) requirements. Also, although the AP600 accumulators are designed to provide extended coverage of a LBLOCA through reduced flows and larger water volumes, it was concluded that this difference can be reasonably accommodated by appropriately modeling the accumulators in the WCOBRA/TRAC model. This is because the AP600 accumulator design is similar to that in an operating plant (except as noted below), for a tank of pressurized nitrogen over water. Accounting for the AP600 design changes in the WCOBRA/TRAC model would include representing the larger water volume and adjusting the accumulator model so that the calculated flow rate matches the design flow rate. Both of these can be done using standard modeling techniques for representing accumulator differences between operating plants.

The one accumulator design difference between the AP600 and current plants that needed to be addressed was the use of a spherical accumulator tank for AP600 versus a cylindrical tank for current plants. This geometric difference is not considered important because the accumulator discharge is driven by the expansion of the nitrogen cover gas, and the accumulator to primary system pressure difference. However, concerns arose that the nitrogen gas might not act like an ideal gas because of liquid-gas heat transfer through the vessel wall. Tests of cylindrical vs

spherical accumulators showed that there was practically no difference in the draining time because the discharge of 28 L (1 ft^3) of liquid from either a cylindrical or a spherical accumulator tank will cause the same pressure decrease in the driving force even though the geometry is different. This is the same result that would be seen in current plants with two cylindrical accumulator tanks of different cross-sectional area. Also, given the large pressure difference that develops between the accumulator tank and the primary system during discharge, other geometrical effects (such as surface area and liquid/gas interface area) are negligible.

The last issue noted by Westinghouse concerned differences during the blowdown cooling phase. The calculated AP600 response during blowdown cooling is considerably better than the calculated North Anna response. This is because more core downflow occurs with rewetting of the rods in the low power and average core channels. As noted by Westinghouse, the AP600 was designed to enhance the blowdown cooling phase by increasing the flow area between the upper head and the upper plenum. Therefore, the enhanced blowdown cooling for AP600 relative to North Anna is not unexpected because of the physical differences in plant design.

However, because the upper head flow first enters the upper plenum, there is still a question as to whether the code is calculating the correct flow split between the upper plenum and core and the upper plenum and loops. In the response to comment 1(d) of Westinghouse letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse noted that the pressurizer is modeled in a worst case location based on sensitivity runs, and the global model run matrix varies parameters that will affect this flow split. To determine what kind of variation in blowdown cooling is calculated as a result of the global model run matrix, the review considered the results Westinghouse provided in response to RAI 440.661 provided by a Westinghouse letter NSD-NRC-97-5292, dated August 27, 1997. In general, it was found that this information supported Westinghouse's statement. The global model run matrix calculations showed variations in the calculated AP600 blowdown PCT and blowdown cooling because of differences in the magnitude and timing of the calculated blowdown cooling flow. Further, the AP600 results provided in the response to RAI 440.661 were compared to North Anna global model run matrix results in Westinghouse letter NTD-NSA-SAI-95-391, dated October 13, 1995. This comparison showed the AP600 and North Anna results were consistent in that, when the same changes in parameter values were analyzed, similar trends regarding blowdown PCT and cooling were calculated. Therefore, it is concluded that the Westinghouse's global run matrix will adequately account for variation and uncertainty in the flow split from the upper plenum to the core and the upper plenum to the loops.

The review process also addressed the following issues related to the differences between the AP600 and three- and four-loop plant LBLOCA response related to (1) the canned motor pumps in AP600 versus shaft seal pumps in three- and four-loop plants and, (2) the affect of the 2 x 4 configuration on the break flow.

Regarding the pumps, the responses to comments 1(i)1 and 12(g) of Westinghouse letter NSD-NRC-97-5171, dated June 10, 1997, showed that the AP600 pumps performed similarly to those in three- and four-loop plants. Westinghouse showed that the homologous curves for the AP600 pumps are similar to those for three- and four-loop plants and the pumps experience similar inlet conditions, and it also noted that they are both vertical, single-stage centrifugal pumps. Thus, the behavior of the pumps should not have an impact on the AP600 LBLOCA response.

In Westinghouse letter NSD-NRC-97-5171 dated June 10, 1997, the response to comment 12(h) discussed the effect of the 2 x 4 configuration on the break response. The information provided showed that the flow in the intact cold leg of the broken loop remained positive during the first 8 seconds of the accident. During this period, the blowdown PCT occurred at approximately 6 seconds and had decreased approximately 55.5°C (100°F) by the time flow reversed at 8 seconds in the intact cold leg of the broken loop. This implies that the 2 x 4 configuration does not impact the blowdown PCT calculation. During the period of reverse flow in the intact cold leg of the broken loop (which lasts from 8 to 18 seconds), the flow through the intact cold leg is always less than 50 percent of the total loop side break flow. Thus, it never dominates the loop side break flow. The reverse flow through the intact cold leg of the broken loop could result in additional flow down through the core because the mass to feed this reverse flow comes from the vessel. At a maximum flow of 454 kg/s (1000 lbm/s), this additional flow is only 10 percent of the vessel side break flow during the 8 to 18 second period. If the core flow increased by the maximum 10 percent, then the core heat transfer coefficient (HTC) would increase by approximately 8 percent (using the Dittus-Boelter correlation). During this period, the core is in the blowdown cooling phase and the range of heat transfer multipliers used during the blowdown cooling phase will cover this type of uncertainty. Thus, the 2 x 4 configuration should not have a significant impact on the break response or core heat transfer.

21.6.3.5 Westinghouse Methodology and CSAU Comparison

In the Westinghouse response to comment 1 in its July 18, 1997 letter (NSD-NRC-97-5240), Westinghouse submitted a comparison of its realistic LBLOCA methodology on the basis of WCOBRA/TRAC to the CSAU methodology of NUREG/CR-5249. This section summarizes that comparison and, where appropriate, references the sections in this report that describe the Westinghouse methodology and the review in more detail. The information is presented by following the three elements and 14 steps in the CSAU methodology.

21.6.3.5.1 Element 1 - Requirements and Capabilities

This element consists of the first 6 steps of the CSAU methodology. These steps are intended to determine the scenario modeling requirements and compare them to computer code capabilities to determine the applicability of the computer code to the particular scenario. Element 1 is also used to identify potential limitations in the application of the code.

Step 1: Specify the Scenario

The capabilities of a computer code are scenario dependent. For example, the requirements to properly calculate a LBLOCA are different from an SBLOCA. This is because the dominant phenomena and processes are different. Therefore, the first step in the CSAU methodology is to specify the scenario being considered. The CSAU and Westinghouse AP600 realistic methodologies selected the LBLOCA. In identifying the specific scenario, Westinghouse fulfilled CSAU Step 1.

Step 2: Select the Nuclear Power Plant

The response of a particular PWR plant to a LBLOCA will vary from plant to plant. Therefore, the type of plant or plants being considered must be identified.

For CSAU, a Westinghouse four-loop PWR with 17 x 17 fuel bundles was selected. For this application, Westinghouse selected the AP600 plant with a 17 x 17 fuel bundle and passive ECC injection into the reactor downcomer.

By specifying the type of plant considered, Westinghouse fulfilled CSAU Step 2.

Step 3: Identify and Rank Phenomena

Not all phenomena are equally important in calculating a plant's response to a LBLOCA. Therefore, the phenomena must be identified and ranked relative to their importance in calculating the primary safety criteria for a LBLOCA. For the LBLOCA, the primary safety criterion is the PCT. Phenomena important to each phase of the LBLOCA are identified and ranked separately in the PIRT. By using a PIRT, the list of phenomena needing to be considered in the analysis is simplified and reduced to a manageable size.

In CSAU, expert opinion and user experience formed the basis for the PIRT developed during the review as described in NUREG/CR 5047.

Westinghouse developed a PIRT for the AP600 similar to that in the CSAU methodology. The Westinghouse PIRT is discussed in Section 21.6.3.6 of this report including how the important phenomena were identified and ranked. This was done for each phase of a LBLOCA. Since Westinghouse developed a PIRT similar to that developed in the CSAU study, Westinghouse fulfilled CSAU Step 3.

Step 4: Select a Frozen Code

Selecting a frozen code is important because it ensures that changes to the code after an evaluation is completed do not impact the conclusions of the study. Also, it ensures changes occur in an auditable and traceable manner.

CSAU used TRAC-PF1 MOD1, Version 14.3 (NUREG/CR 3858).

Westinghouse selected WCOBRA/TRAC, MOD7A. Revision 1, for AP600 analyses; the NRC approved this code version in the three- and four-loop plant methodology. As discussed in Sections 21.6.3.12.1 and 21.6.3.16 of this report, Westinghouse made several minor modifications to the code to allow for the modeling of AP600 passive safety features during the CMT injection phase of a LBLOCA. However, Westinghouse also indicated the changes had a negligible impact on the calculated PCT as stated in response to comments 4(a) and (b) in Westinghouse letter NSD-NRC-97-5240, dated July 18, 1997. Because the AP600 95th percentile PCT shows a large margin to the 10 CFR 50.46 PCT limit (the AP600 SSAR shows 913°C (1675°F) versus the 10 CFR 50.46 limit of 1204°C (2200°F)), this is not considered safety significant.

Because Westinghouse selected an approved code, and because the modifications made do not impact the calculated PCT, the staff considers that Westinghouse has met the intent of CSAU Step 4.

Step 5: Provide Code Documentation

This step provides documentation that is consistent with the frozen code version. Adequate documentation allows confirmation of the code's applicability to the specific scenario and evaluated plants. CSAU recommends the documentation include a user manual, user guide, developmental assessment reports, and a models and correlations quality evaluation report.

TRAC code documentation available to the CSAU methodology included a code user manual and code description, a models and correlations document, and developmental code assessment reports.

Westinghouse documented its realistic LBLOCA methodology for three- and four-loop plants in the five volume CQD. The CQD included a description of the WCOBRA/TRAC models and correlations, a series of code assessments, description of applying the methodology to a PWR, and the uncertainty evaluation. Significant documentation was also generated during the review of the approved methodology as a result of Westinghouse's responses to NRC questions. NRC review of the CQD and the other Westinghouse-generated documentation is found in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. In WCAP-14171-P, Westinghouse provided arguments for the applicability of WCOBRA/TRAC to the AP600 and described the modifications of its approved methodology as applied to the AP600. Review of the application of WCOBRA/TRAC and the modified methodology to the AP600 is the subject of this report.

Westinghouse provided the revised CQD in a submittal dated March 24, 1998, which documents the WCOBRA/TRAC code and realistic/best estimate methodology to incorporate Westinghouse's commitment and other changes made during the NRC review process.

Except for the user manual and user guide, documentation equivalent to that outlined in CSAU Step 5 was provided by Westinghouse. The user manual and guide were not included in the scope of the review.

Step 6: Determine Code Applicability

The applicability of a computer code is determined by evaluating the conservation equations, closure relationships, code numerics, and structure and nodalization relative to the important phenomena identified by the PIRT in Step 3. This step identifies the code's applicability and helps to identify areas needing modification or needing to be considered in the uncertainty evaluation.

To determine code applicability, CSAU used the PIRT to identify important phenomena and evaluated the capabilities of the chosen code, TRAC-PF1, to calculate those phenomena. CSAU concluded the TRAC-PF1 code was applicable to LBLOCA analyses.

In the CQD, Westinghouse performed a similar evaluation for three- and four-loop plants, but described the LOCA transient in terms of physical processes including fluid flow, structural heat transfer, and structural distortion. Westinghouse assessed the capabilities of WCOBRA/TRAC to predict the phenomena associated with the above processes by a direct review of the models

and comparisons to experimental data, and Westinghouse concluded the code was applicable to LBLOCA analyses. Westinghouse compared the AP600 response to a North Anna LBLOCA analysis in WCAP-14171-P, Section 2. As shown by Westinghouse, the AP600 LBLOCA response is very similar to that for North Anna (see Section 21.6.3.4 of this report). Westinghouse also developed a PIRT for the AP600 and showed it was similar to that developed for three- and four-loop plants with cold-leg injection. Therefore, Westinghouse concluded WCOBRA/TRAC could be applied to the AP600 LBLOCA. Review of WCAP-14171-P and the PIRT found nothing to contradict Westinghouse's conclusion in this area. Therefore, Westinghouse met the intent of CSAU Step 6.

21.6.3.5.2 Element 2 - Assessment and Ranging of Parameters

In Element 2, Steps 7 to 10 are used to determine the effects of the important parameters over the specified ranges. The effects to consider include those associated with code accuracy, effects of scale, and parameter ranges for the uncertainty evaluation.

Step 7: Establish an Assessment Matrix

In this step, the data set used to determine the code uncertainty on the basis of comparisons to test data is established. The PIRT table is used to help determine the assessment matrix, which should include both separate effects and integral tests. The assessment matrix is used to provide a database for evaluating (1) the code accuracy to calculate phenomena important to the scenario, (2) the code's capability to scale-up phenomena to the full-size plant, and (3) the influence of nodalization on the calculation.

The CSAU study reviewed prior TRAC-PF1 assessments to confirm they examined the dominant phenomena identified in the PIRT. Tests used in the CSAU study included the Upper Plenum Test Facility (UPTF) for ECC bypass; Marviken for break flow; LOFT for scaling and nodalization; Slab Core Test Facility (SCTF) and Cylindrical Core Test Facility (CCTF) for scaling, heat transfer, and steam binding; INEL film boiling tests for heat transfer, two-phase pump data, and rewet data.

The approved Westinghouse methodology for three- and four-loop plants included assessment of WCOBRA/TRAC against approximately 100 separate effects and integral tests. In WCAP-14171-P, Westinghouse demonstrated the similarity of the AP600 LBLOCA response to that for current generation plants. By showing this similarity, Westinghouse extended the three- and four-loop plant assessment from the approved methodology to the AP600. In response to comment 1 in its July 18, 1997 letter (NSD-NRC-97-5240), Westinghouse compared the WCOBRA/TRAC assessment for three- and four-loop plants against the highly ranked phenomena in the Westinghouse three- and four-loop plant PIRT. This comparison was reviewed by the NRC in the three- and four-loop plant review. The staff found that all important phenomena identified in the three- and four-loop plant PIRT were covered in the assessment tests (see NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996). In addition, Westinghouse in WCAP-14171-P provided assessments for those features specific to the AP600 LBLOCA such as DVI. These new assessments are discussed in Section 21.6.3.7 of this report.

Also, as discussed by Westinghouse in response to comment 16 of its June 10, 1997 letter (NSD-NRC-97-5171), the range of the tests was found to adequately cover the range of the conditions expected in the AP600 (see Section 21.6.3.13.2 of this report) for the important parameters defined in RG 1.157 and Westinghouse's PIRT. Therefore, Westinghouse established an assessment matrix consistent with CSAU Step 7.

Step 8: Define Nuclear Power Plant Nodalization

The nodalization studies discussed in this step are intended to define a PWR nodalization that is sufficient to provide needed detail yet economical to run full-scale PWR analyses.

CSAU used previous studies with developmental versions of TRAC-PF1 to define the noding detail for the PWR. The basic rule was to use the same number of nodes as in the LOFT code assessment work. The core model did not include a separate channel to represent the hot assembly.

Westinghouse established the AP600 noding in the vessel on the basis of system geometry (location of guide tubes and support columns) and the LBLOCA processes discussed in Step 3. This AP600 nodalization is similar to the three- and four-loop plant nodalization in the approved methodology. Some changes were made in the AP600 nodalization to account for AP600 geometry differences. To meet RG 1.157 recommendations, a hot assembly is represented as a separate channel; AP600 plant calculations are used to determine the hot assembly location in the core. The AP600 nodalization was applied to the AP600 specific experiment simulations.

To allow use of the CQD assessment calculations in determining the CQD WCOBRA/TRAC code uncertainty, the three- and four-loop plant PWR nodalization was applied to the CQD experiment simulations to ensure nodes of similar axial length. The similarity of the AP600 and three- and four-loop plant nodalizations allows for the application of the CQD code uncertainty to AP600. Review of the experiment and PWR nodalizations is discussed in Section 21.6.3.8 of this report. Therefore, it was concluded that Westinghouse developed the AP600 plant nodalization consistent with CSAU Step 8.

Step 9: Determine Code and Experiment Accuracy

This step discusses two approaches in determining the code accuracy. First, there is the direct comparisons to experimental data. Second, there is the use of experimental data to determine parameter ranges for use in PWR sensitivity studies.

In CSAU, this step consisted of two parts, ranging of parameters for the uncertainty evaluation and code and experiment accuracy. In the first part, models were assessed and ranges determined by comparing code predictions to data and use of scatter plots. The code bias was estimated and applied as a multiplier on a calculated result or as an additive term to correct the tendency of the model to overpredict or underpredict the data. The scatter about the bias line was used to develop the model uncertainty. In most cases, a uniform distribution was assumed because of a lack of data. In the second part, code calculations of PCT were compared to experimental data for separate and integral effects tests.

In the approved Westinghouse methodology, several thermal-hydraulic models ranges were determined for the uncertainty evaluation including critical flow, fuel rod parameters, heat transfer, minimum film boiling temperature (T_{MIN}), pump/nozzle resistance, and condensation. Other models were confirmed not important or were conservatively biased. Westinghouse, in response to comment 12(k) of its July 18, 1997 letter (NSD-NRC-97-5240), evaluated each of these items for applicability to the AP600. Further information on the review of this area is given in Sections 21.6.3.9 and 21.6.3.12 of this report. In the second part, Westinghouse performed the extensive code assessment discussed above and used it to determine an experimentally based code bias and uncertainty. Review of the experimentally based bias and uncertainty is discussed in Section 21.6.3.9.5 of this report.

Westinghouse included both types of approaches in its determination of the code and experiment accuracy; however, there are some differences such as the comparison of the model and code-based uncertainties discussed in Sections 21.6.3.9.1 and 21.6.3.9.5 of this report. The Westinghouse methodology is considered to be consistent with the intent of CSAU Step 9.

.Step 10: Determine the Effect of Scale

Step 10 recognizes that not all of the code assessment work will be performed on tests completed at full-scale test facilities. This step assesses the effects of the scale differences on the code uncertainty estimate.

In the CSAU study, it was concluded that power-to-volume scaled test facilities adequately simulate the PWR response except in the areas of the downcomer (ECC bypass) and upper plenum entrainment. Sensitivity studies were performed for the PWR to determine the effects of scale on the basis of the developed parameter ranges. For upper plenum entrainment, TRAC-PF1 calculations with the entrainment models altered were run to determine the bias for the effect of steam binding on PCT. In the case of ECC bypass, full-scale UPTF data were used to develop a bias applied to the calculated PCT. CSAU also identified critical flow and pump two-phase performance as needing additional review because of the lack of full-scale data and included variations in these models in the run matrix used to develop the PCT response surface.

Westinghouse determined that the CSAU conclusions on the applicability of power-to-volume scaled facilities also applied to its approved methodology. Thus, Westinghouse evaluated the effects of ECC bypass and upper plenum entrainment. However, full-scale UPTF test data was not available in these areas. To evaluate the effects of ECC bypass with DVI, WCOBRA/TRAC was assessed against UPTF Test 21, and Westinghouse found the WCOBRA/TRAC results were conservative relative to the test data (see WCAP-14171-P, Section 3.2). For upper plenum entrainment, UPTF Test 29B was evaluated as part of the three- and four-loop plant review, and Westinghouse also found a conservative (negative) bias in its WCOBRA/TRAC calculations (see NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996). In both cases, Westinghouse does not apply the negative bias. Regarding critical flow, Westinghouse accounts for this directly in the uncertainty analysis with the effect determined by PWR sensitivity calculations. For pump two-phase performance, Westinghouse evaluations in response to comment 12(g) of its June 10, 1997, letter (NSD-NRC-97-5171), showed that just as for three- and four-loop plants single-phase pump performance in AP600 is more important and included this parameter directly in the uncertainty evaluation. The variations for critical flow and single-phase pump performance are discussed in Section 21.6.3.9.2 of this report.

On the basis of the above, the staff concluded that Westinghouse has met the intent of CSAU Step 10 and that conservative biases for ECC bypass and upper plenum entrainment were not used by Westinghouse to adjust the final calculated PCT.

21.6.3.5.3 Element 3 - Sensitivity and Uncertainty Analysis

In this element, the effects of individual contributors to the total uncertainty are determined and combined to provide a statement on the total uncertainty of the analysis.

Step 11: Determine the Effect of Reactor Input Parameters and State

Uncertainty in the operating state of the PWR at the time of the accident results in uncertainty in the calculated PCT. This step assesses the effects of the plant initial conditions on the accident results.

The CSAU study evaluated the peaking factor and fuel stored energy to define an operating point. Plant inputs were on the basis of the assumption of base load operation.

Westinghouse considered the effects of a wide range of parameters on the calculated PCT and both plant initial conditions and boundary conditions were considered. For AP600, Westinghouse determines the impact of the following four conditions:

(1) Plant physical configuration - steam generator tube plugging, hot assembly location, and pressurizer location relative to the break

(2) Power distributions and operating history - to simplify the overall AP600 analysis, bounding parameters on the basis of AP600 sensitivity calculations are used in this area (e.g., peaking factors and axial power distribution)

(3) Initial fluid conditions - reactor pressure, reactor T_{avg}, and accumulator conditions

(4) Boundary conditions - break location, type, and size; containment pressure; and offsite power availability

The above items are discussed in more detail in Section 21.6.3.11 of this report, but it is noted here that the AP600 analysis uses bounding values for the items listed in Items (1), (3), and (4), above, in addition to those already noted for Item (2).

Through this bounding approach, the Westinghouse methodology accounts for the effects of the uncertainty in the initial plant operating conditions on the overall analysis. Therefore, the Westinghouse methodology is consistent with CSAU Step 11.

Step 12: Perform PWR Sensitivity Calculations

This step provides information on the effects of the plant input conditions and code model uncertainties on the code output (primarily PCT). This is done by performing code sensitivity calculations with the input varied to determine the effects on the calculated results.

For CSAU, a thermal-hydraulic run matrix that varied break flow, pump two-phase head degradation, T_{MIN}, core entrainment, and combinations of break flow and pump two-phase head degradation was developed. Local effects were calculated using the TRAC supplemental rod option. This allows different rods to be modeled, but the rods do not feed back into the thermal-hydraulic analysis. The CSAU study used these rods to determine the effect of peaking factor, fuel conductivity, gap heat transfer coefficient (HTC), forced convection HTC, and combinations of gap HTC and fuel conductivity and gap HTC and forced convection HTC.

For AP600, the effects of power distributions (both peaking factors and power shapes) and the effects of initial conditions are determined through WCOBRA/TRAC calculations. Westinghouse then uses a bounding approach to account for these uncertainties in the AP600 analysis. The Westinghouse thermal-hydraulic or global model run matrix looks at the effects of break flow, broken loop vessel nozzle loss coefficient, and condensation multiplier. Local effects of hot rod peaking factor, gap HTC, fuel density, fuel conductivity, cladding burst temperature, cladding burst strain, metal-water reaction, convection HTC, and various cross products of these parameters are accounted for through HOTSPOT simulations. These simulations use the HOTSPOT code with global model run matrix results as boundary conditions. These are discussed in more detail in Sections 21.6.3.9.4 and 21.6.3.12.4 of this report.

Westinghouse performs the sensitivity calculations discussed above to determine the effect of code input and models on the WCOBRA/TRAC calculated PCT. Thus, the Westinghouse methodology is consistent with CSAU Step 12.

Step 13: Combine Biases and Uncertainties

In this step, the uncertainties associated with the various parts of the methodology (e.g., code limitations, scale effects, and initial operating conditions) are combined. One approach is to use a Monte Carlo simulation to determine the PCT distribution.

In CSAU, the results of Step 12 were used to determine response surfaces for the blowdown and reflood PCTs using seven variables. The 95th percentile PCT was assessed through Monte Carlo simulation.

Westinghouse bounded the effects of power distributions (both peaking factors and power shapes) and initial and boundary conditions in the WCOBRA/TRAC analyses performed with the global model run matrix developed in Step 12. Westinghouse developed response surfaces from the results of these WCOBRA/TRAC analyses. The 95th percentile PCT was developed through Monte Carlo simulation. This is discussed in Section 21.6.3.9 of this report.

Westinghouse combines the various components of uncertainty. In doing so, the methodology is consistent with CSAU Step 13.

Step 14: Determine Total Uncertainty

In this step, a final statement of total uncertainty given as a probability for the limiting value of the primary safety criteria is made for the code. On the basis of RG 1.157, the staff used the 95th percentile as the basis for determining compliance with the high probability requirement of 10 CFR 50.46. Biases may be applied to account for uncertainty contributors that could not be quantified or because it was not economical to quantify the effect of uncertainty contributors.

The CSAU study estimated the 95th percentile PCT with a Monte Carlo simulation using the response surfaces developed in Step 13 and randomly sampling the assumed distributions for the seven parameters. A number of biases were then applied on the basis of the results of studies performed during the CSAU development.

The Westinghouse methodology also uses a Monte Carlo simulation to determine the 95th percentile PCT. In this way, the Westinghouse methodology is consistent with the CSAU approach in Step 14. The power distribution (peaking factors and power shapes) and the initial condition effects are bounded. The model response surfaces developed in Step 13 are used to calculate a bias and uncertainty associated with the global model run matrix parameters using randomly sampled variables. The uncertainties from the models response surfaces are used as discussed in Sections 21.6.3.9.1 and 21.6.3.9.5 of this report. No biases are applied after the PCT is calculated, although several negative biases were estimated.

In the approved methodology, the Monte Carlo analysis combined various uncertainties by superposition. To correct for inaccuracies in the superposition approach, Westinghouse developed a correction to the superposition approach that was applied during the Monte Carlo analysis. Because of the bounding of the power shape and initial condition effects for AP600, superposition and the superposition correction are not needed for AP600. However, Westinghouse does include a verification step for the response surfaces used in the AP600 methodology. On the basis of the above, the staff concluded that Westinghouse uses a methodology consistent with CSAU Step 14.

21.6.3.5.4 Summary of Review

This CSAU comparison summary shows the Westinghouse methodology closely follows the CSAU methodology, and that Westinghouse accounted for the 14 steps of the CSAU methodology. Specific similarities include using a PIRT to identify important phenomena, ranging parameters and sensitivity studies to determine code uncertainty propagation, and use of Monte Carlo simulations and response surfaces to determine the 95th percentile PCT.

However, some differences are noted in the details of the Westinghouse application of the CSAU methodology because of Westinghouse's need to address the concerns of operating plants, the justification of operating plant limits, the different codes involved, and additional full-scale data now available. For example in Step 11, CSAU assumed the plant to be analyzed was in baseload operation. Westinghouse, to justify AP600 operating limits, needed to consider Technical Specification limits for a number of parameters. Another example is the use of biases in Step 14. In some cases, the CSAU methodology applied biases at the end of the analysis to account for uncertainties it did not account for in other ways. In the Westinghouse methodology, some uncertainties were accounted for by showing the code had a conservative bias on the basis of comparing the code results to test data not available to the CSAU study, but the negative bias was not applied to the analysis. In the staff's judgement, these types of differences are not significant. This is because they reflect Westinghouse's need to justify plant operating limits that were not considered in the development of CSAU, or they represent a more conservative approach than that used in the development of CSAU. Overall, it was concluded that, on the basis of the discussion above, the Westinghouse methodology is consistent with the CSAU methodology.

21.6.3.6 PIRT Evaluations (CSAU Step 3)

As noted in Section 21.6.3.5.1, Step 3, of this report, a PIRT helps in identifying the important phenomena that control a specific accident scenario and ranking them for their relative importance. In this way, the important phenomena can be identified and accounted for in the uncertainty analysis. It also provides a means of reducing the phenomena needing to be considered to a manageable number. Westinghouse provided the WCOBRA/TRAC AP600 PIRT in WCAP-14171-P, and it is reproduced in Table 21.6-6. Review of the PIRT is discussed below.

Westinghouse's AP600 PIRT discussion included the company's own ranking of phenomena for AP600 and comparisons to the Westinghouse three- and four-loop plant PIRT and to that done by the experts panel during the CSAU review, as describe in NUREG/CR-5047. The Westinghouse PIRT used a ranking scale of 1 to 9 with 9 being the most important and 1 being the least important. This is the same ranking scale used in the CSAU review. Comparison of the three PIRTs found them very similar. In general, there was good agreement between the PIRTs, and only occasionally was a phenomenon ranked differently by more than two. On the basis of NUREG/CR-5047, the staff considered a ranking difference of three or more an indication of a significant difference of opinion between the PIRTs.

Differences between the three- and four-loop plant PIRT and the CSAU expert PIRT were covered during the staff's review of the approved methodology. In many cases, the Westinghouse AP600 PIRT showed similar differences with the CSAU expert PIRT, and the reasons Westinghouse gave for the differences were similar to that for the earlier three- and four-loop plant review. The staff considers this appropriate because of the similarity of the AP600 and three- and four-loop plant LBLOCA responses (see Section 21.6.3.4 of this report). Items handled in this way are marked by an asterisk in Table 21.6-6. Therefore, this review focused on the areas where the AP600 design and/or phenomena impacted the PIRT evaluation.

Cladding oxidation was rated lower by Westinghouse for the AP600 relative to CSAU and three- and four-loop plants. Westinghouse noted this is because of the significantly lower PCT in the AP600 LBLOCA analysis, and the lower PCT is the result of the lower peak power in the AP600 core design. The staff agrees with this position because the AP600 PCTs (WCOBRA/TRAC calculated PCTs are approximately 760 °C (1400 °F) in blowdown, 649 °C (1200 °F) in reflood, and 95th percentile PCT is 913 °C (1675 °F)). These PCTs are much lower than the cladding temperature (approximately 982 °C (1800 °F)) where oxidation becomes a concern.

Westinghouse's AP600 ranking for pump two-phase performance during blowdown was the same as that for three- and four-loop plants, and both were ranked lower than the CSAU expert PIRT. Because the AP600 pumps are different from those in three- and four-loop plants, Westinghouse was asked to justify the AP600 ranking. In response to comments 1(i)1 and 12(g) in its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse showed that the AP600 pumps performed similarly to those in three- and four-loop plants. For example, Westinghouse showed the homologous curves for the AP600 pumps are similar to those for three- and four-loop plants, and the pumps experience similar inlet conditions. On the basis of this information and its earlier acceptance for the three- and four-loop plants, the staff concludes that the AP600 pump two-phase performance ranking is appropriate.

The ranking by Westinghouse for non-condensable gases in the cold leg during reflood differed also, but this is because of the DVI used in AP600. In the downcomer, the accumulator nitrogen cover gas is also not important because the AP600 accumulators do not empty until after the core is completely quenched (see Figure 21.6-1 versus Figure 21.6-2). In addition, on the basis of Westinghouse's response to comment 1(g) in its letter NSD-NRC-97-5171, dated June 10, 1997, 60 percent of the initial liquid remains in the accumulators at the time of PCT so that small changes in the analysis results or the AP600 design will not cause the accumulators to empty before the time of PCT. In the approved methodology, Westinghouse results presented in response to Volume 1, question 134, of Westinghouse letter NTD-NSA-MYY-95-12, dated April 21, 1995, discussed the effects of dissolved nitrogen coming out of solution in the primary system, and Westinghouse showed that effect was negligible.

During the review Westinghouse was requested to clarify the phenomena where AP600 was ranked lower than the three- and four-loop plants since it was not clear based on the physical differences between the plant types. One of these phenomena was oxidation, which is discussed above. Others include reflood heat transfer, core entrainment/de-entrainment, and containment pressure.

Westinghouse discussed the reasons for these ranking differences in response to comments 1(a) and 1(b) in its letter NSD-NRC-97-5171, dated June 10, 1997, and noted that the AP600 methodology is conservative for each of these phenomena and/or the phenomena are covered in the uncertainty analysis. Therefore, the Westinghouse AP600 methodology accounts for these items even though they are ranked lower.

Overall the staff found good agreement between the Westinghouse AP600 PIRT and the Westinghouse three- and four-loop plant and the CSAU expert PIRTs. Where a phenomenon ranking differed by three or more, the staff found the difference appropriately justified by Westinghouse. The staff, therefore, concluded that the Westinghouse PIRT for the Westinghouse AP600 plant is adequate.

21.6.3.7 Code Assessment (CSAU Step 7)

Code assessment is required by 10 CFR 50.46(a), and may conform to the guidance of RG 1.157, and CSAU. Westinghouse uses the results in determining an experiment-based uncertainty for WCOBRA/TRAC.

In WCAP-14171-P, Westinghouse reviewed the AP600 plant LBLOCA processes relative to those in three- and four-loop plants. This included critical flow, heat transfer, ECC injection, and steam generator and pump behavior. As discussed in Sections 1, 2.1, and 2.2 of WCAP-14171-P, Westinghouse found few LBLOCA processes that are different for AP600 relative to current plants, except for DVI and the CMTs. Westinghouse found the CMTs did not contribute to core cooling in the AP600 LBLOCA early phase; therefore, the CMTs do not need to be addressed for LBLOCAs. Also, Westinghouse showed that AP600 and three- and four-loop plant LBLOCA responses are similar. In addition, the AP600 response to upper head flow during blowdown, structural heat transfer because of the core reflector, use of canned rotor pumps, and the longer accumulator discharge found in AP600 is different from current generation plants. However, Westinghouse concluded that specific assessment of these areas was not needed because they are not completely new areas for AP600 relative to current plants.

Westinghouse concluded that the extensive code assessment performed for three- and four-loop plants could be applied to AP600, but WCOBRA/TRAC also needed specific assessment against DVI data.

Regarding the above issues, in the review of WCAP-14171-P, the staff draws the following five conclusions:

(1) The CMTs inject during the ECC bypass period, and, therefore, the coolant injected does not contribute to core cooling. In fact, Westinghouse stated in WCAP-14171-P that an AP600 analysis was run without the CMTs and the passive residual heat removal heat exchanger, and the PCT differed by less than 5.5°C (10°F) from the base case.

(2) For items such as upper head flow during blowdown, core reflector structural heat transfer, use of canned rotor pumps, and longer accumulator discharge, the staff notes that these do not represent components or situations that are completely new or that cannot be represented by appropriate input modifications to models used for current plants. For example, the AP600 canned rotor pumps are vertical single-stage centrifugal pumps (the same as three- and four-loop plants) that are modeled by using the appropriate homologous curves and pump characteristic parameters. Also, while the AP600 experiences enhanced blowdown cooling relative to three- and four-loop plants, the staff notes this is an enhancement of a phenomenon present in operating plant LBLOCA. Westinghouse provided some assessment of the upper head to upper plenum flow in the LOFT assessments in its letter NSD-NRC-97-5332, dated September 8, 1997, and the global model run matrix varies the flow split from the upper plenum to the core and hot legs. For more information on this last item, see Section 21.6.3.4.2 of this report.

(3) As discussed in Section 21.6.3.4 of this report, Westinghouse showed the similarity of the AP600 and three- and four-loop plant LBLOCA response.

(4) Westinghouse's conclusion regarding what AP600-specific assessment needed to be performed is supported by the review of important LBLOCA processes in Section 1 of WCAP-14171-P which showed that there is little unique to the AP600 LBLOCA. Also, the PIRT comparisons provided by Westinghouse in Section 2.1 of WCAP-14171-P and reviewed in Section 21.6.3.5 of this report show few unique and important phenomena for AP600 LBLOCA.

(5) In an NRC letter entitled, "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996, the staff reviewed the code assessment work Westinghouse performed on WCOBRA/TRAC. That review found all the highly ranked phenomena for three- and four-loop plants were covered by the tests. This, combined with Items 3 and 4, means the three- and four-loop plant assessment in the approved methodology can be applied to AP600. A summary of the Westinghouse WCOBRA/TRAC assessment for three- and four-loop plants is provided in Section 21.6.3.7.1 of this report.

When applied to the AP600, Items (1) to (5), above, mean that the areas unique to AP600 can be addressed by appropriate input for models used in current plants, are addressed by the uncertainty evaluation, or addressed by AP600 specific assessments (only the AP600 use of

DVI was directly assessed). The DVI assessments performed by Westinghouse are discussed in Section 21.6.3.7.2 of this report.

21.6.3.7.1 Summary of Westinghouse WCOBRA/TRAC Assessment for Three- and Four-loop Plants

The details of the three- and four-loop assessment review are found in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996; however, they are summarized here. Westinghouse assessed WCOBRA/TRAC against a large number of separate effects, integral, and special tests. Facilities analyzed included a wide range of scales, small-scale to full-scale.

Separate effects tests: WCOBRA/TRAC calculated reasonably well the results from the blowdown tests in the Oak Ridge National Laboratory Thermal-Hydraulic Test Facility and Westinghouse tests from the G-1 and G-2 test facilities and G-2 refill tests. For forced reflood tests in various FLECHT facilities and the German FEBA facility, WCOBRA/TRAC gave a good to slightly conservative PCT calculation, but it had difficulty calculating turnaround time, cooldown rate, and quench time accurately. For PCT, Westinghouse addressed these effects by determining heat transfer multipliers on the basis of comparisons to test data that are applied in the uncertainty evaluation. Westinghouse addressed these issues for oxidation by applying a time shift to the oxidation calculations. Westinghouse also covered the effect of turnaround time in the methodology through the global model run matrix.

Integral effects tests: Although break flow mispredictions caused the WCOBRA/TRAC calculated PCTs for LOFT to be lower than the data, the WCOBRA/TRAC uncertainty evaluation directly accounts for the effect of break flow uncertainty on the calculated PCT. Therefore, the Westinghouse LOFT assessments were considered adequate. On the basis of conservative predictions, the staff considered the CCTF and SCTF assessments adequate. Also, Westinghouse's comparisons showed the WCOBRA/TRAC results were good to conservative, relative to UPTF test data.

Other assessments: Based on Westinghouse's comparisons, WCOBRA/TRAC results were conservative to good relative to the test data from the Westinghouse/EPRI one-third-scale steam/water mixing tests and the Creare-scaled ECC bypass tests. For the National Research Universal (NRU) Reactor comparisons, WCOBRA/TRAC adequately simulated the NRU tests because PCTs were reasonably calculated as were the burst parameters. However, NRU results showed there is some uncertainty in the transient rod internal pressure (RIP) calculation that will affect the burst temperature criterion in WCOBRA/TRAC analyses. Westinghouse showed the effect of the RIP uncertainty on the burst temperature criterion was small and well within the burst temperature uncertainty accounted for with the local effects models in HOTSPOT. Therefore, for local effects, the Westinghouse methodology accounts for the uncertainty in transient RIP. Westinghouse also calculates hot assembly (HA) rod burst in the full WCOBRA/TRAC analyses called for in its methodology. If WCOBRA/TRAC calculates a HA rod reflood PCT greater than 871 °C (1600 °F) but not rod burst, Westinghouse in the approved

methodology committed to increasing the initial RIP in the WCOBRA/TRAC HA rod until burst is calculated and choosing the more limiting of the burst and non-burst cases. This will adequately account for transient RIP uncertainties and their effect on rod burst in the full WCOBRA/TRAC runs.

21.6.3.7.2 Westinghouse DVI Assessments

The assessment of WCOBRA/TRAC's ability to calculate DVI is summarized below. Westinghouse analyzed two tests to evaluate WCOBRA/TRAC's ability to simulate DVI; those tests were CCTF Run 58 and UPTF Test 21.

CCTF Run 58

The cylindrical core test facility (CCTF) was a large-scale test facility designed to study the refill and reflood system response to a LBLOCA in a four-loop PWR. The flow area scale factor for CCTF was 1/21.4 of the reference PWR. The core represented a full height heated length of 3.66 M (12 ft), and there were three intact loops and one broken loop. In the core, three power levels could be simulated. Run 58 modeled downcomer injection as part of the ECCS, similar to the AP600. The test data collection began at 85 seconds, and, during portions of the test, accumulator water was also injected into the cold leg (from 85 to 116 seconds) and the lower plenum (from 85 to 103 sec). The experiment initiated with heat-up at t=0 seconds while reflood was initiated at t=93 seconds.

Westinghouse compared the calculated and measured results for a number of parameters. The cladding temperature response is illustrated in Figures 21.6-3 to 21.6-8. All the comparisons of cladding temperature response are found in Figures 3.1-16 to -30 of WCAP-14171-P. The comparisons showed the calculated results overpredicted the cladding temperatures with the greatest overprediction coming at the 2.4 m (8 ft) and 3 m (10 ft) elevations. While the cladding PCTs were higher, the calculated cooldown after the PCT was more rapid, and quench times were earlier than indicated by the data (Figures 21.6-9 to 21.6-11). Westinghouse attributed the more rapid quench progression in the WCOBRA/TRAC analysis relative to the data to the fact that the calculated results did not show the core/downcomer oscillations observed in the test. As discussed in the Westinghouse response to comment 8 of its July 18, 1997 letter (NSD-NRC-97-5240), this difference was because of the calculated downcomer level stabilizing below the downcomer injection port, whereas the downcomer level recovered to submerge the port in the test. The lower downcomer level was the result of a small overprediction of entrainment out of the vessel.

The upper plenum pressure is overpredicted for most of the test (Figure 21.6-12) because the more rapid quench advancement in the calculation results in higher steam production relative to the test. Downcomer and core differential pressure comparisons (Figures 21.6-13 and 21.6-14, respectively) show that the calculated downcomer pressure difference was within the band of the oscillations observed in the test, and the measured core pressure difference was well predicted. Regarding loop flows, the comparisons show that in general WCOBRA/TRAC results were relatively close to the test data except for the Cold Leg 1 and Hot Leg 4 steam mass flows, which were overpredicted (Figures 21.6-15 to 21.6-19).

Westinghouse concluded that overall WCOBRA/TRAC gave a good calculation of the system response measured in CCTF Run 58. The staff's review of the WCOBRA/TRAC results

provided show that the calculated results do overpredict the cladding temperatures especially at the upper elevations. The test PCT was overpredicted by approximately 17 °C (30 °F) (WCOBRA/TRAC 871 °C (1600 °F) and test 854 °C (1570 °F)). Evaluating the other code result/data comparisons, the staff agrees with Westinghouse's conclusions regarding the agreement between the code results and the data as discussed above. As noted by Westinghouse, one of the major factors causing the difference in the code results relative to the test data is the lack of oscillations in the calculated results. This was a result of the lower calculated downcomer level. Overall, the WCOBRA/TRAC results for CCTF Run 58 were in reasonable agreement with the data, and the PCT results were conservative relative to the data.

UPTF Test 21

UPTF is a model of a German four-loop PWR that was part of the international 2D/3D research program. The facility included a vessel with the top quarter of the core and upper plenum represented at full size. The core was not simulated but was represented by a steam/water injection system to set up the appropriate flow conditions in the vessel. The loops were also full size and represented three intact loops and one broken loop. They included steam generator simulators and adjustable flow resistances to simulate tripped pumps. Test 21 consisted of five runs, two of which simulated DVI. These runs were Run 272, Phase A, and Run 274, Phase B. The runs and their subphases differed in terms of the injection rate, steam flow rate, and ECC subcooling.

Westinghouse compared the calculated and measured results for a number of different parameters. The important ones included lower plenum liquid inventory and integrated break flow. The comparisons showed that WCOBRA/TRAC underpredicted the liquid penetration into the lower plenum (i.e., WCOBRA/TRAC underpredicted the liquid inventory during the test). Consistent with this, WCOBRA/TRAC also overpredicted the integrated break flow (i.e., WCOBRA/TRAC overpredicted the ECC bypass).

On the basis of the code/data comparisons, Westinghouse concluded that WCOBRA/TRAC gave a conservative calculation of ECC bypass for the UPTF DVI injection tests. The staff review of the UPTF Test 21, Run 272, Phase A, and Run 274, Phase B, WCOBRA/TRAC results found they support the Westinghouse conclusions. The calculations resulted in less liquid penetration into the lower plenum and higher ECC bypass relative to the test results. While this indicates WCOBRA/TRAC overpredicted ECC bypass in the test, the staff notes there is one issue relating to the DVI location in UPTF relative to AP600. In WCAP-14171-P, Westinghouse noted the DVI location in UPTF was at approximately the elevation of the hot and cold legs while the DVI location in AP600 is approximately 0.9 m (3 ft) below the cold leg centerline. Westinghouse addressed the impact of this difference on extending the UPTF results to AP600 in response to comments 8(g) and 8(i) in letter NSD-NRC-97-5171, dated June 10, 1997. While the AP600 configuration should result in a more favorable ECC bypass condition (see page 3-80, WCAP-14171-P), Westinghouse showed that ECC bypass ended in the AP600 analysis at a lower downcomer steam mass flux relative to the UPTF test data. This indicates that other factors in the AP600 calculations (e.g., code models or plant nodalization) are keeping the WCOBRA/TRAC analysis conservative. Therefore, the staff agrees the UPTF results support the fact that WCOBRA/TRAC gives a conservative ECC bypass calculation for DVI and AP600.

Conservatism in the ECC bypass calculation was accepted by the staff in its review of the approved Westinghouse realistic methodology for three- and four-loop plants. With the conservative ECC bypass calculation, which leads to a higher PCT, Westinghouse could have argued for the application of a negative bias in the final PCT calculation. Instead, Westinghouse chose to ignore the negative bias and retained the conservative ECC bypass calculation as a conservatism in the overall approved methodology. On this basis, the conservative ECC bypass calculation for DVI, with Westinghouse again ignoring the negative bias, is considered adequate.

21.6.3.8 Plant Nodalization (CSAU Step 8)

Westinghouse's methodology addressed the AP600 nodalization issue consistent with the guidance of RG 1.157, Section 3.5, and CSAU Step 8. Westinghouse developed the AP600 nodalization so that it is similar to that used for three- and four-loop PWRs. The three- and four-loop PWR nodalizations were used as the basis for the nodalizations developed for the separate effects and integral assessment analyses in the CQD. The AP600 nodalization was used for the DVI assessments in WCAP-14171-P. This was done because Westinghouse desired to apply the code bias and uncertainty developed from the assessment studies to the PWR calculations. Although there are geometric differences between PWRs, Westinghouse's methodology is flexible enough to accommodate differences in design and still maintain consistency with the assessment nodalizations. If a consistent nodalization methodology is used for the assessment and PWR calculations, then Westinghouse concluded, and the staff agrees, geometric differences/nodalization will not add significantly to the code bias and uncertainty.

The review of the three- and four-loop plant nodalization is discussed in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. The review found the Westinghouse methodology provided consistent nodalizations between the tests and plants.

In WCAP-14171-P, Westinghouse discussed the WCOBRA/TRAC nodalization for the AP600. It is important to note that Westinghouse included explicit modeling of the hot assembly (HA) in the location that leads to the least flow during blowdown. The intent is to limit the HA blowdown cooling and bound the effect on the PCT.

On the basis of the consistency between the AP600, PWR, and assessment nodalizations, the staff considers the AP600 nodalization adequate for realistic LBLOCA analyses. Also, review of the AP600 nodalization found Westinghouse considered important information such as upper plenum structures in determining the types of core channels and the location of the HA, low-power assemblies on the core periphery, and loop components in determining the nodalization. In particular, the staff considers placing the HA in the location minimizing blowdown flow an appropriate bounding assumption in the analyses.

21.6.3.9 Code/Experiment Accuracy (CSAU Step 9)

The need to determine the code and experiment accuracy is discussed in RG 1.157, Section 4, and CSAU Step 9. Westinghouse described its method of uncertainty evaluation for AP600 in WCAP-14171-P. The basic approach was to use the approved methodology but simplify where appropriate for AP600. Westinghouse's approach and the review of it are discussed below.

In the approved uncertainty evaluation, Westinghouse identified initial plant conditions and model effects as the important factors to consider. The plant conditions were further subdivided into initial conditions and core power distributions. In addition, a model or plant condition may be global or local in its effect. A parameter has a global effect if it can affect the entire thermal and hydraulic transient. It has a local effect if it only affects the local conditions at the PCT location. The interrelationship of the uncertainty parameters is shown in Figure 21.6-20.

The major simplification of the approved methodology for the AP600 was the decision to bound the plant conditions (both initial conditions and power distributions) in the WCOBRA/TRAC analyses. As a result, only the global model and local effects needed to be directly included in the AP600 uncertainty analysis.

There are a large number of phenomena and parameters in a LBLOCA analysis. To reduce the number involved in the uncertainty analysis only those ranked 7, 8, or 9 in the PIRT are addressed. In WCAP-14171-P supplemented by Westinghouse's response to comment 12(k) of letter NSD-NRC-97-5240, dated July 18, 1997, Westinghouse evaluated the important items from the approved methodology uncertainty analysis relative to their application for AP600. Because of the similarity of the AP600 and three- and four-loop plant responses to a LBLOCA, and because the same types of equipment/materials are used in both types of plants, Westinghouse showed that most of the same important parameters apply to AP600. These include the following:

- critical flow
- pump single-/two-phase performance (broken loop relative resistance)
- fuel rod conditions such as power distributions, stored energy, decay heat, cladding burst, cladding reaction, and gap conductance
- core heat transfer and T_{MIN}
- ECC bypass; entrainment and steam binding
- condensation

Westinghouse either conservatively calculated, bounded, or directly included the above items in the AP600 uncertainty methodology.

The following discussion clarifies how Westinghouse evaluated each item above. An overview or roadmap of the Westinghouse methodology is given first, and it is followed by more detailed discussions on the models and the plant conditions. Then, the impact of local effects on models and plant conditions will be discussed.

21.6.3.9.1 Westinghouse Realistic LBLOCA Methodology Roadmap

This section gives a brief overview or roadmap of Westinghouse's AP600 realistic LBLOCA methodology. References to those sections in this report that describe and summarize the review of the methodology are given.

The Westinghouse AP600 realistic LBLOCA methodology consists of several parts. To determine the 95th percentile PCT, Westinghouse uses the WCOBRA/TRAC code and then performs an uncertainty analysis. The WCOBRA/TRAC base code is described in CQD. Then, there is the analysis to determine the 95th percentile PCT. This analysis is discussed below and in Sections 21.6.3.9.2 to 21.6.3.9.5 of this report. Westinghouse's use of WCOBRA/TRAC is described first followed by the 95th percentile PCT determination.

For AP600, Westinghouse uses the WCOBRA/TRAC program to analyze the plant's response to changes in initial conditions and account for power distribution (peaking factor and power shape) effects through one-at-a-time sensitivity studies. The effect on PCT is measured by the PCT change (ΔPCT) as a result of the initial condition change or power distribution change. These sensitivity studies are discussed in Section 21.6.3.11 of this report. For AP600, the sensitivity studies determine the bounding conditions to be used in the WCOBRA/TRAC analyses for the global model run matrix. This is the simplification of the approved methodology noted above for the AP600.

Next, Westinghouse analyzes the same global model run matrix developed in the approved methodology for AP600. This provides the ΔPCT information needed to develop response surfaces that it uses to account for code model uncertainties in the uncertainty analysis. The response surfaces are also discussed in this and following sections of this report.

The next part of the uncertainty evaluation is to account for local or hot spot uncertainties. Westinghouse uses the HOTSPOT model, discussed in Section 21.6.3.12.4 of this report, to perform the local uncertainty evaluation. The parameters affecting the local uncertainty include the following:

- hot rod (HR) calculational uncertainty
- hot rod pellet diameter, enrichment, and rod bow uncertainties
- fuel density and conductivity
- gap and cladding HTC
- rod internal pressure
- cladding burst temperature and strain
- metal-water reaction; and fuel relocation

For selected WCOBRA/TRAC runs from the global model run matrix, Westinghouse performs a direct Monte Carlo analysis using the HOTSPOT model to determine the spread of the PCT distribution due to local uncertainties. The PCT distributions as a result of local uncertainties for the selected runs in the global model run matrix are then fit to two response surfaces, one for the biases of the distributions and one for the standard deviations of the distributions that are obtained from the HOTSPOT runs. The response surface variables are the models varied in the run matrix. Westinghouse combines the model global and local effects as discussed in Section 4.2.3 of its letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995. The second part of the uncertainty

evaluation uses data on the basis of experimental results and a nodalization uncertainty. This is divided into two parts. The first uncertainty, B1, is the uncertainty based on comparisons between the code calculated PCT and the measured PCT in all the applicable assessment cases plus a nodalization uncertainty (combined using square root sum of the squares). The second uncertainty is on the basis of experimental cladding temperature data scatter about the average cladding temperature at a given elevation (uncertainty B2). In the approved methodology, the B1 and B2 uncertainties are used to provide lower bounds to other uncertainty estimates. With the simplified AP600 uncertainty methodology, these two uncertainties are still used to provide lower bounds to other uncertainties but in a different way than in the approved methodology as follows:

(1) The uncertainty B1 is used in the approved methodology to establish a lower bound on the uncertainty determined from the superposition correction. The superposition correction is used to correct for inaccuracies in the superposition assumptions made in the approved methodology. Superposition is used to combine various uncertainties. For AP600, superposition is no longer used because of the simplifications in the AP600 methodology, and the uncertainty B1 is used to provide a lower bound to the correction required to ensure the accuracy of the global model response surface for ΔPCT previously developed.

(2) The uncertainty B2 is used in the approved methodology to establish a lower bound on the uncertainty determined from the global models/local uncertainty. For AP600, B2 provides a lower bound to the uncertainty in the global model response surface for the standard deviation.

Westinghouse now has the information needed to perform a Monte Carlo simulation to determine the 95th percentile PCT (i.e., the PCT greater than that expected to occur in 95 percent of possible LBLOCAS). The general steps of a single Monte Carlo iteration are described next, and the steps are illustrated in Figure 21.6-21. Westinghouse samples the model parameter distributions (Box 1) and uses them in the response surfaces for model bias and uncertainty (Box 5) to get the ΔPCT because of model uncertainties. This is added to the base PCT to get a preliminary PCT, $PCT_{p,i}$, for the iteration (Box 6). Westinghouse then uses HOTSPOT results from the selected global model run matrix analyses to determine the corrections for uncertainties in the global model response surfaces (both ΔPCT and standard deviation, Box 7). For ΔPCT, the correction is compared to the B1 uncertainty (discussed above), and the larger value is selected. The resulting distribution is sampled to get the model uncertainty ΔPCT. This is added to $PCT_{p,i}$ to get the preliminary corrected hot spot PCT for the Monte Carlo iteration, $PCT_{p,ci}$. Westinghouse then compares the B2 uncertainty (discussed above) to the standard deviation correction, and the larger uncertainty value is selected. The resulting distribution is sampled and added to $PCT_{p,ci}$ to get the final hot spot PCT for the Monte Carlo iteration, PCT_i. To determine the 95th percentile PCT, the above process is repeated many times. For each iteration, I, the calculated PCT_i is binned. The PCTs in each bin are counted starting from the highest PCT and when 5 percent of the PCTs are counted, the 95th percentile PCT is determined.

21.6.3.9.2 Models - Global Effects

For AP600, the global models of importance are the same as for three- and four-loop plants (i.e., critical flow, broken loop relative resistance, and condensation). Westinghouse justified the use of the same models for AP600 in response to comment 12(k) in letter NSD-NRC-97-5240, dated July 18, 1997. This is discussed in more detail in Section 21.6.3.12.2 of this report, but here it is noted that the same parameters apply because of the similarity of the different plant types regarding LBLOCA response, equipment, and materials.

The uncertainties in the above models are accounted for in Westinghouse's uncertainty methodology using the following process. The uncertainty distributions for each parameter are determined. Then, thermal-hydraulic sensitivity studies are performed where each parameter is varied (singularly and in combination with the other two parameters) over a range that appropriately bounds the expected range of model variation as follows. The ranges used for the sensitivity studies for each parameter are as follows:

(1) Break flow: 100 percent of the data in the break flow uncertainty distribution.

(2) Break flow resistance ratio: The uncertainty in this parameter is ranged as discussed in CQD Section 26-4 and Sections 3.1.2 and 4.4.2.1 of Westinghouse letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995.

(3) Condensation: For the run matrix, the maximum and minimum multipliers are on the basis of test data (see Figure 3.1.7-1 of Westinghouse letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995). Although the sampled range in the uncertainty evaluation is larger, the run matrix covers almost 77 percent of the sampled range. The staff considers this adequate because the extrapolation is small.

Westinghouse calls this the global model run matrix, and it is shown in Table 4.5-1 of WCAP-14171-P. The results of this run matrix are used as input to the HOTSPOT code to determine the local HR temperature uncertainty as discussed in Sections 21.6.3.9.4 and 21.6.3.12.4 of this report. The global model and HOTSPOT effects are then used to develop response surfaces for the Monte Carlo analysis.

21.6.3.9.3 Plant Conditions - Global Effects

Westinghouse accounted for the uncertainty in the initial plant conditions. As mentioned above, this includes power distributions (peaking factors and power shapes) and initial conditions. The uncertainty in plant initial conditions is accounted for in the analysis by assuming bounding conditions in the run matrix discussed in Section 21.6.3.9.2 of this report. The conditions selected are developed from plant sensitivity studies. As previously noted, this is different from the approved methodology, but it does provide conservative PCT results.

21.6.3.9.4 Local Effects Models/Parameters

Having established the global uncertainties for models and bounded plant conditions, Westinghouse addressed the effects of local HR uncertainties on the global model results. The

parameters involved for AP600 are the same ones in the approved methodology and include HR calculational uncertainty; hot rod pellet diameter, enrichment, and rod bow uncertainties; fuel density and conductivity; gap and cladding HTC; rod internal pressure; cladding burst temperature and strain; metal-water reaction; and fuel relocation. In Westinghouse's response to comment 12(j) in letter NSD-NRC-97-5171, dated June 10, 1997, it discussed the basis for this approach. Westinghouse noted that the AP600 uses a standard Westinghouse fuel product. This indicates the important items for local models/parameters are the same for AP600 relative to three- and four-loop plants.

These sources of HR local uncertainty are evaluated using the HOTSPOT model. This model (see Section 21.6.3.12.4 of this report) is a stand-alone calculation of local effects at the PCT locations (i.e., blowdown, first reflood, and second reflood) and the burst node location (see page 165 of Westinghouse letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995). The HOTSPOT model is run many times in a direct Monte Carlo simulation. Boundary conditions input to the HOTSPOT model are taken from WCOBRA/TRAC runs in the global model run matrix. For each HOTSPOT run, Westinghouse develops a bias and distribution that describes the effect of the local uncertainties on the PCT. In this way, local effects are combined with global effects for models. For the WCOBRA/TRAC runs taken from the global model run matrix, two response surfaces are fit to the distributions developed with HOTSPOT for the individual runs, one to the biases and one to the uncertainties. These response surfaces are used in the Monte Carlo simulation to account for local uncertainty propagation.

21.6.3.9.5 WCOBRA/TRAC Experiment Based Uncertainty

In Westinghouse's methodology, the code uncertainty is on the basis of comparisons to experimental results, a nodalization uncertainty, a data scatter uncertainty, and a code uncertainty on the basis of ranging of model parameters. For AP600, Westinghouse divides the experiment based uncertainty into two parts, which is the same as for the approved methodology. (See previous discussion in Section 21.6.3.9.1 of this report.) Because superposition is not used for AP600 due to Westinghouse's AP600 methodology simplifications, Westinghouse uses the experiment-based uncertainty differently, as follows. First, the code uncertainty on the basis of comparisons to experimental results, plus the nodalization uncertainty, is compared to the global model ΔPCT response surface uncertainty, rather than the superposition correction. These are items related to the uncertainty in the bias or the code's ability to calculate the average PCT. Second, the data scatter uncertainty is compared to the global model standard deviation response surface uncertainty rather than the global model/local uncertainty. These are related to the uncertainty associated with local effects. In both cases, the larger of the two uncertainties is used in the Monte Carlo calculation of the 95th percentile PCT.

21.6.3.9.6 Other Parameters/Factors Considered by Westinghouse

Westinghouse's list of important parameters in the approved methodology considered a number of items not discussed in the previous sections. These include ECC bypass, entrainment/steam binding, and accumulator nitrogen. NRC approval of Westinghouse's handling of each item is discussed in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated

June 28, 1996; however, that review is summarized here. For ECC bypass and entrainment/steam binding, Westinghouse showed WCOBRA/TRAC provided conservative calculations in these areas, but Westinghouse does not apply a negative bias to account for these conservatisms. The staff found this acceptable. For accumulator nitrogen, Westinghouse chose to limit the beneficial effects of accumulator nitrogen discharge on heat transfer in HOTSPOT calculations by setting a maximum HTC. For AP600, this is not important because accumulator nitrogen does not discharge until long after the PCT has been calculated and the core quenched.

Because of the use of DVI in AP600, Westinghouse directly evaluated ECC bypass when DVI is used. Westinghouse found a negative bias could be applied to analyses because of delayed and reduced ECC penetration relative to full-scale UPTF tests (see WCAP-14171-P). However, the bias is not applied at this time resulting in conservative PCT calculations. Given that a conservative bias is ignored, the staff concluded that Westinghouse's approach for the AP600 ECC bypass uncertainty will result in a conservative PCT calculation. As discussed earlier, this type of conservatism was accepted by the NRC in the review of the approved methodology for three- and four-loop plants.

21.6.3.9.7 Review of Uncertainty Propagation

As the above description shows, Westinghouse's uncertainty methodology is very complex. Therefore, it was carefully reviewed to assess whether it included the important parameters and whether the uncertainty distributions for the important parameters were justified.

The review considered the list of important parameters identified by Westinghouse in WCAP-14171-P. The highly ranked phenomena from the Westinghouse AP600 PIRT were considered, and this review found them to be ranged directly in the uncertainty analysis, modeled directly in WCOBRA/TRAC analyses, bounded in WCOBRA/TRAC analyses, covered in the variation of items ranged directly, or conservatively calculated in WCOBRA/TRAC analyses.

At the heart of the uncertainty methodology is the assumed uncertainty distributions. The review considered Westinghouse's justification for each parameter range and the associated uncertainty distribution in Section 21.6.3.12.2 of this report. That review found Westinghouse's justification for the distributions adequate on the basis of information provided.

In the approved methodology, the NRC, in a letter to Westinghouse, "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996, found Westinghouse's approach for uncertainty propagation for three- and four-loop plants acceptable because WCOBRA/TRAC and HOTSPOT calculations cover the LBLOCA through reflood. This allowed the effects of uncertainty in parameters and combinations of parameters to be calculated through the entire accident. The Staff also found run matrix development and response surface generation acceptable. These same approaches are applied to AP600. Given the similarity of the AP600 and three- and four-loop plant LBLOCA responses, the staff finds this approach acceptable for the AP600 application.

21.6.3.10 Effects of Scale (CSAU Step 10)

The effects of scale on the calculated results need to be evaluated to ensure the code models can appropriately calculate full-scale PWR behavior given that most of the code assessment matrix is on the basis of smaller-scale test facilities. This is discussed in RG 1.157, Sections 4.1 and 4.2, and CSAU, Step 10. Westinghouse evaluated the effects of scale on the WCOBRA/TRAC code in its letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995. As noted in Section 21.6.3.5.2, Step 10, of this report, power-to-volume scaled test facilities adequately simulate the PWR response except in the areas of ECC bypass and steam binding; also, critical flow and pump performance needed to be considered because of the lack of full-scale data. Staff review of the scaling issues for three- and four-loop plants (see NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996) concluded that Westinghouse had adequately addressed scaling issues in these areas. The NRC review is summarized here.

ECC Bypass
Using data at various scales, Westinghouse showed that WCOBRA/TRAC had a conservative bias in its ECC bypass calculation as facility size approaches full-scale. That is, the code overpredicted the amount of fluid bypassing the vessel and going out the break.

Steam Binding
Using data for several test facilities at different scales, Westinghouse showed WCOBRA/TRAC had a conservative bias as facility scale increased. This implies greater steam binding will be calculated by the code in plant calculations relative to the steam binding observed in the experiments.

Critical Flow
Westinghouse accounts for the uncertainty in break flow by directly including critical flow in the uncertainty evaluation.

Pump Performance
Westinghouse showed single-phase pump performance was more important than two-phase performance and included the single-phase pump performance directly in the uncertainty evaluation.

Because of the similarity of the AP600 and three- and four-loop plant LBLOCA response as discussed in Section 21.6.3.4 of this report, Westinghouse only needed to readdress ECC bypass because of the use of DVI in the AP600.

To address ECC bypass with DVI for AP600, Westinghouse compared WCOBRA/TRAC results to CCTF and full-scale UPTF data to evaluate WCOBRA/TRAC's capability to calculate this phenomenon. The comparisons show that WCOBRA/TRAC overpredicted the PCT for the CCTF tests, and it also overpredicted the ECC bypass in the UPTF DVI tests. That is, the code overpredicted the amount of fluid bypassing the vessel and going out the break in UPTF. These tests are discussed in more detail in Section 21.6.3.7 of this report. Westinghouse also

addressed test to plant differences that could affect the interpretation of the UPTF results relative to AP600.

The review considered Westinghouse's submittal for ECC bypass. The DVI ECC bypass calculation was conservative, and Westinghouse addressed the test to plant differences that could affect the interpretation of the test results relative to AP600. On the basis of the above, the staff considers Westinghouse to have adequately addressed issues relating to scaling.

21.6.3.11 Reactor Input Parameters/State (CSAU Step 11)

Both RG 1.157, Section 3.1, and CSAU Step 11 discuss the need to identify the effects of the reactor input and initial/boundary conditions on the calculated PCT. Westinghouse addressed this issue in WCAP-14171-P.

21.6.3.11.1 Westinghouse's Methodology

In WCAP-14171-P, Section 4.4, Westinghouse evaluated the approved three- and four-loop plant methodology relative to its application to the AP600 in the area of reactor input parameters and initial state. In general, Westinghouse proposed a simplification of the three- and four-loop plant approach for AP600 to reduce the analytical effort required. This simplification for AP600 was made by taking a bounding approach for a larger number of parameters relative to the uncertainty evaluation for three- and four-loop plants.

The parameters/models in the approved methodology were discussed in CQD Section 21; these include the plant physical configuration, plant initial operating conditions (including core power parameters and primary fluid conditions), accident boundary conditions, and WCOBRA/TRAC models needing further analysis because of a lack of prototypical assessment or the need to consider the effects of different transient time scales in the tests and a PWR.

In Table 4.4-1 of WCAP-14171-P, Westinghouse compared the AP600 methodology to that for three- and four-loop plants for plant physical configuration, initial operating conditions, and accident boundary conditions. As found in this comparison, Westinghouse's AP600 methodology will bound most of the parameters that were directly included in the uncertainty evaluation for the approved methodology. This includes, for example, core peaking factors and initial conditions. In the AP600 methodology, the bounding value for the parameters will be determined through one-at-a-time sensitivity studies using WCOBRA/TRAC. The sensitivity studies use plant design information or Technical Specification ranges to determine what ranges to consider as discussed in Westinghouse's response to comment 3 in letter NSD-NRC-97-5240, dated July 18, 1997. The methodology then uses these bounding values in the AP600 input model for the WCOBRA/TRAC global run matrix.

Westinghouse also reviewed for AP600 the approved methodology's approach for dealing with models not adequately assessed, models that were simplified or lacking basic knowledge, and phenomena that may have more important effects on a PWR LBLOCA than a scaled experiment. These included models for break flow, pumps, accumulator nitrogen, condensation, entrainment, ECC bypass, core heat transfer, and fuel rods.

On the basis of the information in the response to comment 12(k) of letter NSD-NRC-97-5240, dated July 18, 1997, Westinghouse justified that only certain models identified in Section 4.4 of

WCAP-14171-P needed additional review for application to AP600. For those areas not needing additional review, Westinghouse noted this was because of the similarity of the different plant types regarding LBLOCA response, equipment, and materials. For example, the different plants operate at the same pressure and temperature conditions; thus, the critical flow uncertainty from the approved methodology applies to AP600. The pumps in the different plants are both vertical, single-stage centrifugal pumps with similar homologous curves, and they experience similar inlet conditions. Thus, the approved methodology pump uncertainty applies to AP600. As a final example, the same fuel is specified for AP600 as in three- and four-loop plants. Thus, the uncertainties related to fuel parameters in the approved methodology apply to AP600.

Westinghouse's response to comment 12(k) in letter NSD-NRC-97-5240, dated July 18, 1997, also clarified how it dealt with the models needing additional review such as T_{MIN} and blowdown cooling heat transfer. Westinghouse proposed a new value for T_{MIN} to be used in WCOBRA/TRAC during blowdown and reevaluated blowdown cooling heat transfer to generate an AP600 specific uncertainty distribution. These two areas are discussed in more detail in Section 21.6.3.12.2 of this report.

In the approved methodology, Westinghouse also used sensitivity studies to evaluate the effects of time step control (CQD Section 22-5), break spectrum (CQD Section 22-6), and fuel rod burnup (CQD Section 22-7) on calculated PCT for three different plants. These studies are discussed below.

In its July 18, 1997, letter (NSD-NRC-97-5240) response to comment 2(a), Westinghouse noted and justified that the time step controls developed for the approved methodology applied to the AP600 model. This was because of the similarity of the controlling phenomena for the different plant types during the important LBLOCA periods.

Westinghouse discussed the break spectrum analyses for AP600 in its July 18, 1997, letter (NSD-NRC-97-5240) response to comment 2(b). On the basis of an earlier analysis for the AP600 SSAR, the limiting LBLOCA for AP600 was determined to be the double-ended cold-leg guillotine (DECLG) break with $C_D = 0.8$. Because the plant input model and the code version has remained basically the same, Westinghouse concluded, and the staff agrees, that the previous break spectrum study was still valid. In addition, the Westinghouse realistic methodology for AP600 does not directly identify the limiting break size associated with the 95th percentile PCT. This is because the effect of critical flow uncertainty on PCT is included in the uncertainty evaluation through the global model response surface and sampling the full range of discharge coefficients developed from the Marviken test comparisons in the Monte Carlo simulation to determine the 95th percentile PCT. Therefore, the AP600 realistic methodology accounts for the break flow uncertainty but in a different way than a 10 CFR Part 50, Appendix K, analysis. The AP600 methodology is also consistent with the approved realistic methodology for three- and four-loop plants.

To account for the uncertainty in the break type (i.e., split or guillotine), the determination of the 95th percentile PCT includes separate Monte Carlo simulations for the guillotine breaks and the limiting split break from the break spectrum study.

Fuel rod burnup studies were presented in Section 22-7 of the CQD for three- and four-loop plants. In these studies, beginning-of-life (BOL) fuel was found to be limiting. In its

July 18, 1997, letter response (NSD-NRC-97-5240) to comment 2(a), Westinghouse noted that the fuel type used in AP600 is equivalent to that of many three- and four-loop plants, and the same burnup criteria are applied to the HR and the HA rod. Therefore, the CQD burnup studies apply to the WCOBRA/TRAC AP600 application.

21.6.3.11.2 Review Summary

Review of WCAP-14171-P found that Westinghouse provided an exhaustive list of important LBLOCA parameters to be considered in the plant sensitivity studies and included AP600 specific parameters. This review did not identify any parameters to add to Westinghouse's list.

It is noted that Westinghouse performed AP600 sensitivity studies to determine the effect of the parameter or model variation on the calculated PCT. These analyses are presented in the AP600 SSAR. Using the results of these sensitivity studies, Westinghouse's AP600 methodology chooses bounding assumptions for use in the WCOBRA/TRAC global run matrix. Therefore, these studies are part of the AP600 methodology for the determination of boundary conditions and do not impact the WCOBRA/TRAC applicability evaluation.

Because of the similarity of the controlling phenomena between three- and four-loop plants and AP600, it is concluded that the time steps control strategy used in the approved methodology can be applied to the AP600 model.

The uncertainty in break flow is appropriately accounted for in the uncertainty analysis. Westinghouse's CQD study on rod burnup showed that BOL is the limiting time-in-life, and this can be applied to AP600 because of the equivalent fuel type.

On the basis of the review summarized above, the staff concludes that Westinghouse's methodology includes AP600 sensitivity studies to determine the effect of reactor input parameters and state. These are provided in the AP600 SSAR. Previous studies in the approved methodology for rod burnup and time steps also apply for the reasons stated above. Because the sensitivity studies show which parameters are important to PCT calculations and the effect of the parameter change on the calculated PCT, these sensitivity studies provide an adequate basis for determining bounding conditions for use in the AP600 methodology.

21.6.3.12 Additional Method Description and Review

This section describes those parts of Westinghouse's realistic LBLOCA methodology for AP600 that are important but do not fall directly into the steps of the CSAU methodology. First, the WCOBRA/TRAC code is discussed. This is followed by a discussion of the AP600 uncertainty distribution review; Westinghouse's handling of split breaks and the HOTSPOT model are also discussed. Conclusions are provided at the end of each subsection and summarized in Section 21.6.3.12.5 of this report.

21.6.3.12.1 LBLOCA Method Description/Review - Code Selection

The NRC approved WCOBRA/TRAC, MOD7A.Revision 1 for analysis of LBLOCAs in three- and four-loop plants in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. This approval came after a detailed review of the code's models and

correlations and the large assessment database provided by Westinghouse. As stated in Westinghouse's response to comment 4(a) in its July 18, 1997 letter (NSD-NRC-97-5240), this is the code version used by Westinghouse to analyze the AP600 LBLOCA. Westinghouse stated that several small modifications to the code were required to analyze AP600 and its passive safety features; however, these modifications only affect passive safety features that do not impact the initial AP600 response to a LBLOCA. Also, the AP600 and three- and four-loop plant LBLOCA responses are very similar as discussed in Section 21.6.3.4 of this report. Therefore, the staff considers WCOBRA/TRAC, MOD7A, Revision 1, adequate for AP600 LBLOCA analyses.

21.6.3.12.2 LBLOCA Method Description/Review - Uncertainty Distributions and Assumptions

In NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996, the staff reviewed and approved all of the uncertainty distributions assumed by Westinghouse in its uncertainty evaluation for three- and four-loop plants. In WCAP-14171-P, supplemented by its responses to comments 12(j) and 12(k) in letters dated June 10 and July 18, 1997, respectively (NSD-NRC-97-5171 and NSD-NRC-97-5240), Westinghouse discussed the basis for the application of these uncertainty distributions to AP600. Except for T_{MIN} and blowdown cooling, whose uncertainty distributions Westinghouse modified for application to AP600, Westinghouse was able to adequately justify these uncertainty distributions as discussed below using the three- and four-loop plant distributions for AP600. The justification was on the basis of similarity of the different plant types regarding LBLOCA response, equipment, and materials.

Critical flow
The different plants operate at the same pressure and temperature conditions; thus, the critical flow uncertainty from the approved methodology applies to AP600.

Pump uncertainty/break path resistance
The pumps in the different plants are both vertical, single-stage centrifugal pumps with similar homologous curves, and they experience similar inlet conditions. Thus, the approved methodology pump uncertainty applies to AP600. The broken loop cold-leg nozzle (BLCL) has a similar K-factor so the approved methodology BLCL nozzle uncertainty applies.

Fuel rod parameters
The same fuel is specified for AP600 as in three- and four-loop plants. Thus, the uncertainties related to fuel parameters in the approved methodology apply to AP600. For AP600, the bounding axial power shape is identified and used in the WCOBRA/TRAC analyses, and peaking factors are also bounded at Technical Specification values.

Heat transfer
(Blowdown cooling heat transfer and T_{MIN} are discussed separately.)

For reflood heat transfer, Westinghouse performed additional assessment to verify WCOBRA/TRAC for the lower initial cladding temperatures expected in the AP600 at the start of reflood. Westinghouse showed the reflood heat transfer uncertainty distribution from the approved methodology bounded the uncertainty distribution on the basis of this initial cladding

temperature data on the lower end of the range. The approved methodology uncertainty distribution also gave a lower median multiplier than the uncertainty distribution on the basis of the cold temperature data. These are both conservative because a lower multiplier results in lower heat transfer in the HOTSPOT calculations. At the upper end of the distribution, however, the approved methodology distribution extends slightly beyond the upper limit of the initial cladding temperature data. The staff does not consider this a problem for several reasons. First, only 3 percent of the approved methodology distribution extends beyond the cold temperature data distribution, so the effect is small. Second, the higher PCTs in the HOTSPOT PCT distribution are those that result from the application of smaller multipliers. With lower minimum and median multipliers, the approved methodology distribution will provide conservative results relative to the cold data distribution. Finally, the 95th percentile PCT reported by Westinghouse in the AP600 SSAR is 913°C (1675°F). This indicates the small extension of the approved methodology distribution beyond the upper end of the cold data distribution is not a safety issue.

During the blowdown heatup period, the AP600 plant responds in the same way as three- and four-loop plants. The heatup period is approximately 5 seconds, similar to three- and four-loop plants. The fuel specified for AP600 is a typical Westinghouse product; therefore, the LOFT and Oak Ridge tests used to develop the blowdown heatup uncertainty for three- and four-loop plants are also applicable to AP600. These support the application of the approved methodology blowdown heatup heat transfer uncertainty distribution to AP600.

Also, the staff used information in Westinghouse letter dated September 8, 1997, to verify that the lower core power density for AP600, which enters into the equation used to establish the blowdown heatup uncertainty, did not impact the application of the approved methodology distribution for blowdown heatup heat transfer to AP600.

During refill, the core is voided, core flow is minimal, and all plants with a given fuel design will experience similar heat transfer. Because AP600 is equipped with typical Westinghouse fuel and the refill heat transfer in the core is independent of other plant design features, the approved methodology refill heat transfer uncertainty distribution is applicable to the AP600.

ECC bypass

The DVI assessments in Section 21.6.3.7 of this report show WCOBRA/TRAC provides a conservative estimate of ECC bypass.

Steam binding/hot leg entrainment

Assessments performed in this area for the approved methodology show that WCOBRA/TRAC provides conservative predictions for steam binding. This is supplemented by the CCTF Test 58 (a DVI test) results that show hot leg liquid flows to be slightly overpredicted.

Noncondensable gases/accumulator nitrogen

For AP600, accumulator nitrogen cover gas is not important because the AP600 accumulators do not empty until after the core is completely quenched (see Figures 21.6-1 and 21.6-2 of this report). In addition, on the basis of the Westinghouse response to comment 1(g) in letter NSD-NRC-97-5171, dated June 10, 1997, there is 60 percent of the initial liquid remaining in the accumulators at the time of PCT so that small changes in the analysis results or the AP600 design will not cause the accumulators to empty before the time of PCT. In the approved methodology, Westinghouse results presented in response to Volume 1, Question 134, of its

letter NTD-NSA-MYY-95-12, dated April 21, 1995, discussed the effects of dissolved nitrogen coming out of solution in the primary system, and Westinghouse showed that effect was negligible.

Condensation

The approved methodology developed the condensation multiplier range on the basis of separate effects test data. The AP600 downcomer is dimensionally similar to a three-loop plant. Therefore, the condensation multiplier range from the approved methodology is considered applicable.

It is also noted that Westinghouse applied bounding assumptions for the power distributions (peaking factors and axial shapes) and initial conditions. The bounding assumptions were determined from WCOBRA/TRAC analyses where the input was varied over the expected operating range of the AP600 plant (see Section 4.4 of WCAP-14171-P). The use of bounding parameters for initial conditions and power shapes significantly reduced the number of parameter uncertainty distributions needing justification.

As mentioned above, Westinghouse modified the handling of T_{MIN} and the blowdown cooling HTC multipliers. The changes in these areas are discussed below. The T_{MIN} changes are discussed first followed by the blowdown cooling changes.

T_{MIN}

For AP600, Westinghouse elected to modify the T_{MIN} model in WCOBRA/TRAC during blowdown from the approved three- and four-loop plant methodology. The T_{MIN} model in the HOTSPOT code remains unchanged for application to AP600. The reason for the change in the WCOBRA/TRAC code is the need to assure a conservative T_{min} calculation during the AP600 blowdown cooling period.

Westinghouse chose to take a bounding approach for the blowdown T_{MIN} in WCOBRA/TRAC AP600 calculations. The value chosen and the basis for that selection is discussed in Section 4.1 of WCAP-14171-P. The review considered the data provided by Westinghouse and found the value chosen adequately supported. The bases for this include the following:

(a) Westinghouse chose appropriate data to use in the evaluation.

(b) Westinghouse selected a T_{MIN} value that is slightly lower than the 5th percentile value of the T_{MIN} data. The data base supports an average T_{MIN} greater than about 538°C (1000°F). Use of the value chosen by Westinghouse results in a T_{MIN} value that clearly is lower than the average of the blowdown data base.

Blowdown Cooling

Westinghouse reevaluated the ability of WCOBRA/TRAC to analyze blowdown cooling heat transfer in WCAP-14171-P, Section 4.2. The WCOBRA/TRAC calculations used the T_{MIN} value discussed above. The database Westinghouse analyzed comprised 10 Oak Ridge National Laboratory dispersed flow film boiling tests (NUREG/CR-2435). As shown in Section 4.2 of WCAP-14171-P, the 10 tests provide data that covers the range of conditions expected during

the AP600 blowdown cooling period. The approach to determine the blowdown cooling HTC multipliers used in the uncertainty analysis was consistent with the approach for the other LBLOCA phases.

It is considered that the new blowdown cooling HTC uncertainty distribution is adequate for AP600 analyses. This is because data that covered the range of AP600 conditions such as pressure and flow were used, and the uncertainty distribution was developed consistent with other LBLOCA phases. It is also noted that the new distribution will provide results that are conservative relative to a direct application of the blowdown cooling distribution from the approved methodology as discussed in Westinghouse letter NSA-SAI-96-102, dated March 25, 1996. This is because, although the minimum multipliers for the two distributions are similar, the AP600 median and maximum multipliers are less than those from the approved methodology. For this application, lower multipliers are more conservative because they result in lower heat transfer at the cladding surface in HOTSPOT calculations.

In addition to evaluating the uncertainty distributions, the review also considered the simplifying assumptions Westinghouse made to make the uncertainty evaluation manageable. In the approved methodology, Westinghouse made simplifying assumptions in the following areas:

(a) combining initial condition uncertainties
(b) break type (guillotine versus split) selection
(c) approach for the resistance ratio (R_B, this is the ratio of the resistances from the core to the break via the various possible flow paths)
(d) combining uncertainties in peaking factors, F_Q and $F_{\Delta H}$.

Of these simplifying assumptions, Items (a) and (d) are not applicable to AP600 because of the simplifications to the approved methodology implemented by Westinghouse. For Item (b), Westinghouse uses the same approach of analyzing each break type separately for AP600, but this is not impacted by the AP600 application. Only Item (c) was potentially affected by the AP600 application because of the canned motor pumps used in AP600.

The AP600 approach for R_B was discussed by Westinghouse in its June 10, 1997, letter (NSD-NRC-97-5171) responses to comments 12(g), 12(h), and 12(i). The information provided showed that the pumps for AP600 and three- and four-loop plants had similar homologous curves and were subject to similar conditions during a LBLOCA. Also, the AP600 broken loop cold leg nozzle has a radius of curvature similar to that found in three- and four-loop plants; this indicates the AP600 nozzle K-factor is similar to that developed for the approved methodology. Finally, Westinghouse showed that the AP600 2 x 4 configuration does not have an impact on the analysis. According to these considerations, it was concluded that Westinghouse had adequately justified the application of the approved methodology approach for R_B to AP600.

Because of the simplifying assumptions made by Westinghouse for AP600 analyses in the areas of initial conditions and power shapes, superposition is not used in the AP600 application. Therefore, the superposition validation step in the approved methodology (see Attachment 2 of Westinghouse letter NSA-SAI-96-019, dated January 24, 1996) does not need to be included in the AP600 analysis. However, Westinghouse does use a similar process to ensure that any uncertainties in the global model response surfaces are accounted for in the analysis.

21.6.3.12.3 LBLOCA Method Description/Review - Split Breaks

In RG 1.157, the staff indicates that split breaks should be considered in the break spectrum analysis used to determine the 95th percentile PCT. Westinghouse addressed split breaks for AP600 in the same manner established for three-and four-loop plants as discussed in its June 10, 1997, letter (NSD-NRC-97-5171) response to comment 13. Westinghouse addressed split breaks in the AP600 SSAR.

This review considered the AP600 split break results presented in AP600 SSAR and compared them where possible to North Anna split break calculations as discussed in Westinghouse letter NTD-NSA-SAI-95-391, "Revisions to Westinghouse Best-Estimate Uncertainty Methodology," dated October 13, 1995. The comparison showed that AP600 and North Anna similarly respond to the split break spectrum. During blowdown for example, the North Anna limiting split break occurred at C_D = 1.4, and the AP600 split break occurred at C_D = 2.0. But the North Anna C_D = 2.0 split break PCT was within 15°C (27°F) of the C_D = 1.4 case, and the AP600 C_D = 1.4 split break PCT was within 21°C (38°F) of the C_D = 2.0 case. In addition, the North Anna PCTs for CDS from 1.2 to 2.0 were within 26°C (47°F) of each other. In the AP600 cases for the same C_D range, the PCTs were within 39°C (70°F) of each other. Finally, both AP600 and North Anna have the limiting reflood PCT occur at a C_D = 1.0. This analysis demonstrates that although a different C_D was limiting for blowdown, the AP600 and North Anna responses for the different split break sizes were very similar.

As discussed in Section 21.6.3.4 of this report, Westinghouse also showed that AP600 and North Anna respond similarly to a double-ended cold leg guillotine break. On the basis of the similarity of the AP600 response to that for North Anna for split and guillotine breaks, the staff considers the application of the approved methodology approach to AP600 split breaks adequate.

21.6.3.12.4 LBLOCA Method Description/Review - Westinghouse HOTSPOT Model

Westinghouse developed a model to evaluate the effects of the uncertainties associated with local parameters on the calculated PCT called the HOTSPOT model (see Westinghouse letter NTD-NSA-SAI-95-5 dated March 6, 1995). It models a portion of a fuel rod at the PCT or burst location including fuel, gap, and cladding. The HOTSPOT model is a simple physical model that allows the effects of uncertainties to be calculated directly by running the model many times with parameter values that vary randomly according to specified distributions. The parameter uncertainties considered are linear power, fuel density after cladding burst, fuel conductivity before and after burst, metal-water reaction rate, gap pressure, gap conductance, HTC, T_{MIN}, burst temperature, and burst strain. The review of the uncertainty distributions for the above parameters is summarized in Section 21.6.3.12.2 of this report. Westinghouse letter NTD-NSA-MYY-95-5, dated March 6, 1995, provides additional details on the HOTSPOT model.

The staff review of the HOTSPOT model found it acceptable for realistic LBLOCA analysis in three- and four-loop plants. The staff determined that it is appropriate to extend this approval to the AP600 because the same fuel is specified for AP600 as for three- and four-loop plants as discussed in Westinghouse's response to comment 12(j) in letter NSD-NRC-97-5171, dated June 10, 1997. Thus, the models in HOTSPOT also apply to the AP600.

21.6.3.12.5 Summary of Review

This section summarizes the results of the review of the Westinghouse realistic LBLOCA methodology. The following conclusions were reached:

(1) NRC review of the WCOBRA/TRAC code models found them acceptable for realistic analysis of LBLOCA results. The details supporting this conclusion are found in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. Because the AP600 and three- and four-loop plant LBLOCA responses are very similar, the staff has determined that WCOBRA/TRAC, MOD7A, Revision 1 is adequate for AP600 LBLOCA analyses.

(2) Uncertainty distributions/assumptions - On the basis of the information provided, Westinghouse has adequately justified the uncertainty distributions used for AP600. Application of the approved methodology distributions was justified on the basis of similarity of the different plant types regarding LBLOCA response, equipment, and materials. Appropriate information was provided by Westinghouse for the AP600 specific distributions used for blowdown cooling and T_{MIN}.

(3) Split breaks - Westinghouse has addressed split breaks in the same manner as three- and four-loop plants. On the basis of the similarity of the AP600 response to that for North Anna for split and guillotine breaks, the staff considers the application of the approved methodology approach to AP600 split breaks adequate. Westinghouse addresses split breaks in the AP600 SSAR.

(4) HOTSPOT - The HOTSPOT model is acceptable for realistic LBLOCA analysis in three- and four-loop plants. The staff has determined that it is appropriate to extend this approval to the AP600 because the same fuel is specified for AP600 as for three- and four-loop plants. Thus, the models in HOTSPOT also apply to the AP600.

21.6.3.13 Comparison with Regulatory Guide 1.157

The recommended features of a realistic LBLOCA analysis are described in RG 1.157. Comparison of the Westinghouse methodology with RG 1.157 is summarized in Section 21.6.3.13.1 of this report. Additionally, in RG 1.157, the staff discussed determining the range of applicability for a number of different models. Consistency with the RG 1.157 recommendations in this area is discussed in Section 21.6.3.13.2 of this report.

21.6.3.13.1 Summary of Westinghouse Methodology/RG 1.157 Comparison

The staff review of the approved Westinghouse methodology found it either met the RG guidance in Part C, Section 3, "Best-Estimate Code Features," and Section 4, "Estimation of Overall Calculational Uncertainty," or else Westinghouse provided adequate justification for the approach presented. For example, Westinghouse uses a point kinetics model to analyze fission heat as allowed by RG 1.157, Section 3.2.2, and the fission product decay heat model is on the basis of the 1979 ANSI/ANS standard allowed by RG 1.157, Section 3.2.4. Where Westinghouse took an alternative approach to that outlined in RG 1.157, the staff found that Westinghouse provided adequate justification for the approach presented. For example,

Westinghouse uses the Cathcart-Pawel model (ORNL/NUREG-17) to calculate the Zircaloy-water reaction at temperatures above 816 °C (1500 °F). In RG 1.157, Section 3.2.5, the Cathcart-Pawel model is recommended but only for temperatures greater than 1038 °C (1900 °F). Westinghouse showed the Cathcart-Pawel model overpredicts the reaction rate at temperatures below 1038 °C (1900 °F); therefore, it is conservative to use it at the lower temperatures. Such conservatism is allowed by RG 1.157 (see Part C, Section 1).

For AP600, Westinghouse used a modified version of the approved methodology discussed in WCAP-14171-P. The main difference from the approved methodology is the simplification of the uncertainty analysis by including a larger number of bounding parameters. Westinghouse chose this approach because of the large margin available in the AP600 relative to the 10 CFR 50.46 PCT limit. The staff notes that this bounding approach adds conservatism to the WCOBRA/TRAC calculated results for AP600 and, as noted above, conservatism is allowed by RG 1.157.

It is also noted that one change made to a WCOBRA/TRAC model impacts the LBLOCA PCT calculation. This was to use a low estimate of T_{MIN} during blowdown in the WCOBRA/TRAC calculations. Again, this makes the AP600 application more conservative than the approved methodology because it would result in the AP600 analysis calculating rod quench during blowdown at a lower temperature than the approved methodology. The change to T_{MIN} is discussed in more detail in Section 21.6.3.12.2 of this report.

In RG 1.157, Section 4.5, the staff notes that the CSAU methodology was originally used to evaluate the overall calculational uncertainty in PCT predictions for NRC developed BE computer programs. In Section 21.6.3.5 of this report, the review found that the Westinghouse AP600 methodology was consistent with the intent of CSAU.

According to the previous NRC approval and the information provided during this review, Westinghouse has met RG 1.157 guidance or provided adequate justification for its alternate approach for the AP600 realistic methodology.

21.6.3.13.2 WCOBRA/TRAC Range of Conditions/Applicability

In RG 1.157, the staff identified certain models for which the range of applicability needed to be justified. This requirement meant that WCOBRA/TRAC should be assessed and/or the individual code models should be on the basis of data that cover the range expected in the AP600 analyses. In RG 1.157, the staff identified the following models as requiring justification for the range of applicability:

- critical flow
- ECC bypass
- frictional pressure drop
- critical heat flux
- transition and film boiling heat transfer
- single-phase vapor heat transfer
- level swell

Also, Westinghouse's PIRT identified the following models as important:

- critical flow
- pump
- condensation
- entrainment
- ECC bypass
- core heat transfer including T_{MIN}
- fuel rod including stored energy, decay heat, and gap conductance

To address these issues for AP600, Westinghouse in response to comment 16 in its letter NSD-NRC-97-5171, dated June 10, 1997, stated that the three- and four-loop plant assessment ranges are applicable to AP600 except for blowdown cooling and reflood heat transfer. Westinghouse's response also identified and addressed the following areas of the PIRT evaluation that were ranked high for AP600 but not for three- and four-loop plants: rewet, upper head phenomena, accumulator discharge during reflood, and DVI. It is in these areas that this review was focused.

(1) Blowdown cooling: In Table 4.2-1 of WCAP-14171-P, Westinghouse showed that the blowdown cooling assessment covered the appropriate range of conditions for AP600 in the areas of pressure, mass flux, inlet water temperature, and power.

(2) Reflood heat transfer: In response to comment 16 in its June 10, 1997 letter (NSD-NRC-97-5171), Westinghouse noted that it performed additional WCOBRA/TRAC assessments to cover the lower cladding temperatures calculated for the AP600 at the start of reflood. Additional review using information from Westinghouse's letter dated September 8, 1997, found that the previous WCOBRA/TRAC reflood heat transfer assessments in the CQD also appropriately bounded the AP600 results for pressure, reflood rate, initial core power, and inlet subcooling.

(3) Rewet: In Table 4.1-1 of WCAP-14171-P, Westinghouse showed that conditions appropriate for AP600 were covered in the T_{MIN} assessment in the areas of pressure, mass flux, inlet subcooling, and power. The staff also notes T_{MIN} is directly included in the uncertainty evaluation.

(4) Upper head phenomena: Westinghouse used a conservative upper head temperature per the response to comment 12(e) in its letter NSD-NRC-97-5171, dated June 10, 1997. In addition, the full-pressure, full-temperature LOFT test analyses in the CQD simulated upper head draining as discussed by Westinghouse's response to comment 16 in its letter NSD-NRC-97-5171, dated June 10, 1997, and the information provided in its letter NSD-NRC-97-5332, dated September 8, 1997.

(5) Accumulator discharge during reflood: Westinghouse noted that accumulator discharge is more important for AP600 relative to existing plants because the AP600 accumulator is larger and discharges longer, but existing assessments of accumulator discharge are adequate for AP600. The staff agrees with this because, except as noted below, the AP600 accumulators are similar in design to those in current plants, using pressurized nitrogen over water. Also, modeling the larger accumulator water volume and longer discharge in the AP600 can easily be done with standard modeling techniques used to

represent accumulator differences in existing plants. Finally, Westinghouse used a bounding accumulator discharge model in its AP600 analyses as discussed in the response to comment 12(f) of its letter NSD-NRC-97-5171, dated June 10, 1997.

The one accumulator design difference between the AP600 and current plants that needs to be addressed is the use of a spherical accumulator tank for AP600 versus a cylindrical tank for current plants. This geometric difference is not important because the accumulator discharge is driven by the expansion of the nitrogen cover gas, which acts like an ideal gas, and is controlled by the accumulator to primary system pressure difference. Because the nitrogen acts like an ideal gas, the discharge of 28 L (1 ft^3) of liquid from either a cylindrical or a spherical accumulator tank will cause the same pressure decrease in the driving force even though the geometry is different. This is the same result that would be seen in current plants with two cylindrical accumulator tanks of different cross-sectional area. Also, at the large pressure difference that develops between the accumulator tank and the primary system during discharge, the staff believes that any other geometrical effects (such as surface area and liquid/gas interface area) will be negligible.

(6) DVI: Comparison of Tables 3.2.7-1 and 3.2.7-2 in WCAP-14171-P, shows the appropriate range of conditions for DVI were covered in the areas of DVI flow, core steam flow rate, and ECC subcooling.

This comparison demonstrates that the WCOBRA/TRAC assessment covered the appropriate range of conditions for the important AP600 models and/or the methodology directly accounts for or bounds the uncertainty. Other parameters were covered through the approved methodology.

21.6.3.14 Other Technical Issues

In AP600, CCFL modeling is important because of the effect it has on the calculation of ECC bypass and fall-back at the upper core plate (UCP). For AP600, CCFL needed to be reviewed because of plant differences relative to three- and four-loop plants that affect the ECCS configuration and the potential for fall-back at the UCP due to the lower core power density. This area is discussed below. This section also discusses the issue of compensating errors as it relates to the AP600 analysis.

21.6.3.14.1 WCOBRA/TRAC CCFL Modeling Assessment

The proper calculation of CCFL is important to predict correctly the PCT in a LBLOCA, and the CCFL calculation can be evaluated by comparing it to accepted CCFL correlations and/or test data. This section discusses the WCOBRA/TRAC approach to CCFL, summarizes the staff evaluation of the WCOBRA/TRAC CCFL calculation during the review of the approved methodology, and discusses those CCFL calculations important to AP600 that needed to be reconsidered.

The main PWR areas of importance are the downcomer annulus for ECC bypass and the UCP for fall-back of liquid carried out of the core. WCOBRA/TRAC does not use empirical CCFL correlations; rather it calculates CCFL directly from the basic code equations and constitutive

relationships (for example, the interphase area, interphase heat transfer, and interphase drag correlations). As a result, the WCOBRA/TRAC approach is different from the application of a CCFL correlation. When a CCFL correlation is used, the correlation imposes a limit on the calculated flow solution that the code is not allowed to violate. For example, if a code uses the Wallis correlation to model CCFL, then the flow solution is not allowed to violate the Wallis equation $[j_g^*]^{1/2} + m[j_f^*]^{1/2} = C$, where m is often set equal to 1.0 and C would be selected by the code developer to represent the geometry being modeled. In WCOBRA/TRAC, no such limit is imposed on the flow solution, and the CCF calculated comes directly from the constitutive relations and the solution of the basic equations.

The evaluation of Westinghouse's calculation of CCFL with WCOBRA/TRAC for three- and four-loop plants is discussed in detail in the NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996, and summarized here. To determine whether WCOBRA/TRAC was properly calculating CCFL, that evaluation considered test facility and plant nodalization, the basic equations, the constitutive relationships, and comparisons to data. The CCFL test nodalizations and the plant nodalizations were consistent, which allows the assessment results to be applied to the plant calculations. The staff review found the code models and correlations were adequate in such areas as interfacial drag and condensation. The review also found WCOBRA/TRAC gave good to conservative results for the assessments on the basis of tests in facilities ranging from small- to full-scale.

For AP600, two areas of the previous review needed to be reconsidered. They were ECC bypass calculations with DVI and fall-back at the UCP due to the lower core power density in AP600. The review considered the comparisons between the WCOBRA/TRAC results and the data for the parts of UPTF Test 21 that represented DVI. For all cases, conservative results were found. Also, the PCT in CCTF Run 58 was conservatively calculated. These results are discussed in more detail in Section 21.6.3.7 of this report. Westinghouse also assessed the potential for fall-back from the upper plenum to the core at the UCP in the response to comment 1(e) in its letter NSD-NRC-97-5171, dated June 10, 1997. That response showed that little fall-back is calculated in AP600 analyses. The staff notes that even if this result is not realistic, it is conservative. On the basis of the review summarized above, the staff concludes that WCOBRA/TRAC's models conservatively represent CCFL in the downcomer and at the UCP for the AP600.

21.6.3.14.2 Compensating Errors

Compensating errors are defined as errors which taken one at a time may produce demonstrably incorrect results, but when combined provide acceptable but misleading results. For example, PCT could be calculated well but only because the heat transfer model overpredicted the local HTC, thus, compensating for a low core flow rate.

In the review of the approved methodology, Westinghouse provided a detailed evaluation of compensating errors in WCOBRA/TRAC as applied to three- and four-loop plants. In general, that evaluation is considered applicable to the AP600 analysis because of (a) the similarity of the AP600 and three- and four-loop plant LBLOCA responses, (b) the AP600 specific assessments for DVI showed that WCOBRA/TRAC gave conservative estimates of ECC bypass and core cooling, and (c) while AP600 has better blowdown cooling than three- and four-loop plants, the

global model run matrix ranges the calculated blowdown PCT and cooling as discussed in Section 21.6.3.4 of this report.

Westinghouse provided information supporting the conclusion that no compensating errors exist which might compromise the ability of WCOBRA/TRAC to predict the conditions during a large break LOCA in its response to RAI 440.739F dated February 1, 1998. The additional information related to AP600 is summarized as follows:

(1) In the ORNL steady-state film boiling test predictions, the non-equilibrium vapor conditions observed experimentally were also predicted. This provides confidence that an important component of the heat flux model, the vapor heat sink, is predicted correctly. While differences exist between the predicted and measured values, these differences are captured in the heat flux uncertainty which is applied in the overall uncertainty analysis.

(2) The validation of the CCFL model by the simulation of UPTF Test 21, considering effects of condensation, demonstrates that compensating errors do not exist in the prediction of ECC bypass for the direct vessel injection geometry.

(3) The detailed analysis of CCTF Test 58 indicates that the thermal-hydraulic response of the core to loop flow is proper, and not the result of a compensating error.

21.6.3.15 Resolution of DSER Open Items

A number of open items were identified in the staff's SDSER on the AP600 codes and testing program related to the applicability of WCOBRA/TRAC to AP600 LBLOCA analyses. The following is a discussion regarding the specific resolution of those open items. These items do not represent the only issues or concerns the staff had during the review; however, these were issues that could not be resolved at the time the SDSER was issued and can now be closed on the basis of the completion of the staff's review.

Open Item	Discussion
21.6.3-1	Westinghouse needs to submit the revision to Section 4 of WCAP-14171-P to demonstrate the applicability of the operating plant uncertainty evaluation to the AP600. Westinghouse revised WCAP-14171-P to include Section 4 on the applicability of the operating plant uncertainty evaluation to the AP600. This item is closed.
21.6.3-2	The staff is still reviewing the numerical ranking of the WCOBRA/TRAC AP600 LBLOCA PIRT parameters. Westinghouse provided additional information in Section 2.1 of WCAP-14171-P on the basis for the numerical rankings assigned in the AP600 LBLOCA PIRT. Review of the PIRT rankings was completed and found adequate in Section 21.6.3.6 of this report. Therefore, this item is closed.

21.6.3-3 The staff is still reviewing Westinghouse's response to RAI 440.347 on AP600 blowdown cooling because of downflow.

Westinghouse provided additional information on the AP600 blowdown cooling calculation in the response to comment 1(d) in its letter NSD-NRC-97-5171, dated June 10, 1997. Westinghouse noted the global model run matrix would vary the flow split from the upper plenum to the core and the upper plenum to the loops. This flow variation would result in the calculation of different blowdown cooling effects in the core. This was reviewed and found adequate in Section 21.6.3.4.2 of this report. The review considered the results of the global model run matrix provided in response to RAI 440.661. This response showed the calculated variation in blowdown PCT and blowdown cooling flow as a result of the global model run matrix input changes. Therefore, this item is closed.

21.6.3-4 The staff is assessing the absence of core and downcomer oscillations in the WCOBRA/TRAC analysis of Run 58 at the CCTF.

Westinghouse in response to comment 8 in its letter dated July 18, 1997, clarified the reasons for the lack of downcomer/core oscillations in its CCTF analysis. The calculated downcomer level stabilized below the DVI nozzle, while in the test the downcomer level recovered to resubmerge the DVI nozzle. In the analysis of the test results, the recovery of the downcomer level to resubmerge the DVI nozzle was stated to be the cause of the downcomer/core oscillations in the test. The calculated level did not recover because core entrainment was slightly overpredicted. Also, even though the downcomer/core oscillations were not calculated, Westinghouse noted in Section 3.1.6 of WCAP-14171-P that the calculated PCT was still higher than that measured in the test (i.e., calculated: 871°C (1600°F); test: 854°C (1570°F)). Therefore, this item is closed.

21.6.3-5 The staff is reviewing the pressure differences between the WCOBRA/TRAC analysis and CCTF Run 58 data in the CCTF in order to understand their significance.

In the response to comment 7(f) in its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse noted the core and downcomer differential pressures were calculated well as shown in Figure 3.1-35 and Figure 3.1-36 of WCAP-14171-P. On review of those figures, it was found that the calculated downcomer differential pressure was within the band of the oscillations observed in the test and the measured core differential pressure was well predicted. Therefore, this item is closed.

21.6.3-6 The staff is reviewing the loop flow differences between the WCOBRA/TRAC analysis and the CCTF in order to understand their significance.

In the response to comment 7(f) of its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse noted the loop steam flows were also predicted well. On review of the loop flow comparisons in WCAP-14171-P, it was concluded that flows were relatively well predicted except for the Cold Leg 1 and

Hot Leg 4 steam mass flows which were overpredicted. This is not considered significant given the overprediction of PCT discussed above. Therefore, this item is closed.

21.6.3-7 Westinghouse needs to address the impact of the elevation difference between the DVI location in the UPTF and the AP600, and on Westinghouse's conclusion that WCOBRA/TRAC will conservatively calculate ECC bypass in the AP600.

In the response to comments 8(g) and 8(i) in its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse discussed the impact of this difference. Westinghouse showed that while the AP600 configuration should more easily end ECC bypass (see page 3-80, WCAP-11471-P), the AP600 analysis shows that ECC bypass ends at a lower downcomer steam mass flux relative to the UPTF test data. Therefore, other factors in the code analysis such as code models or plant nodalization are keeping the code analysis conservative. Therefore, this is no longer a concern to the staff and this item is closed.

21.6.3.16 Compliance with 10 CFR 50.46 Requirements

This section discusses how Westinghouse's AP600 methodology meets the requirements of 10 CFR 50.46. This section of the CFR describes the ECCS acceptance criteria for LWRs.

The portion of 10 CFR 50.46(a)(1)(i), that was the focus of this review states the following, "Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded."

This indicates (a) the analytical technique must realistically describe reactor LOCA behavior, (b) comparisons to applicable experimental data must be made, (c) uncertainties in the analysis method and inputs must be assessed so the uncertainties in the calculated results can be estimated, and (d) the uncertainty must be accounted for when comparing the calculated ECCS performance to the criteria set forth in Paragraph (b) of 10 CFR 50.46 so that there is a high level of probability that the criteria would not be exceeded.

The staff determined that the Westinghouse methodology for three- and four-loop plants met the above requirements in as documented in NRC letter "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Analysis," dated June 28, 1996. The review included the following:

(1) The WCOBRA/TRAC code and code assessment - According to the review of CQD Volume 1 and Westinghouse's comparisons to over 100 tests described in CQD Volumes 2 and 3, the staff concluded that WCOBRA/TRAC realistically describes the behavior of a PWR during a LBLOCA.

(2) Uncertainty evaluation - In Volumes 4 and 5 of the CQD and Reference 19, Westinghouse described its methods for determining the uncertainty of the analysis methods and inputs and applying them to the calculated ECCS performance. NRC review of the uncertainty methodology included reviewing all uncertainty distributions, response surface generation, and their applications to determining the 95th percentile PCT. Uncertainty because of reactor input was also reviewed.

In WCAP-14171-P, Westinghouse demonstrated the applicability of the WCOBRA/TRAC code to the AP600 and described the modifications made to the approved methodology for AP600. Westinghouse's approach was to demonstrate the similarity of the AP600 and three- and four-loop plant LBLOCA response and to provide DVI assessments. Because of the similarity of the LBLOCA response, the staff concluded that the previous NRC code and code assessment review also applied to AP600. For the DVI assessments provided, Westinghouse demonstrated that WCOBRA/TRAC conservatively predicted the PCT for CCTF Test 58 and overpredicted the ECC bypass for UPTF Test 21. The conclusion of the previous NRC code and code assessment review plus the conservative DVI assessments supports the staff conclusion that WCOBRA/TRAC is adequate to provide realistic evaluations of the AP600 LBLOCA with the tendency toward conservative results. The review of the similarity of the AP600 and three- and four-loop plant LBLOCA response is discussed in Section 21.6.3.4 of this report, and the DVI assessments are discussed in Section 21.6.3.7 of this report. Thus, the staff concludes that Westinghouse has met, for the AP600, those parts of 10 CFR 50.46(a)(1)(i) that require justification that the methodology realistically describes reactor behavior during a LBLOCA and require comparisons to applicable data.

Modifications to the WCOBRA/TRAC code were made but are applied only to model passive safety equipment that does not impact the initial AP600 response to a LBLOCA. In response to comment 4(a) in its letter NSD-NRC-97-5240, dated July 18, 1997, Westinghouse asserted that the changes only negligibly affect the PCT, but they are needed to analyze the AP600 LBLOCA CMT injection phase which occurs after the accumulators empty and CMT injection begins. Because the AP600 95th percentile PCT shows a large margin to the 10 CFR 50.46 PCT limit (the AP600 SSAR shows 913°C (1675°F) versus 1204°C (2200°F)), this is not considered safety significant. On this basis, the staff concludes that the code changes do not impact the previous NRC review.

Westinghouse also discussed the modifications to the approved uncertainty methodology for AP600 in WCAP-14171-P as supplemented by responses in Westinghouse letters dated June 10, July 18, and September 8, 1997 (NSD-NRC-97-5171, NSD-NRC-97-5240, and NSD-NRC-97-5332 respectively). These modifications simplified the approved methodology by bounding a larger number of parameters relative to the approved methodology. The review of these changes is discussed in Sections 21.6.3.9 and 21.6.3.12 of this report. On the basis of the information provided, the staff has concluded that Westinghouse has adequately justified the changes. Therefore, Westinghouse has satisfied the requirements of 10 CFR 50.46(a)(1)(i), dealing with quantifying the uncertainty and accounting for the uncertainty when comparing the calculated ECCS performance to the criteria of 10 CFR 50.46(b), for the AP600.

10 CFR 50.46(b) of the regulations states the five acceptance criteria for the ECCS. The ECCS must ensure the following:

- The PCT is less than 1204°C (2200°F) [10 CFR 50.46(b)(1)]
- The maximum local cladding oxidation does not exceed 17 percent of the total cladding thickness before oxidation [10 CFR 50.46(b)(2)]
- The maximum hydrogen generation shall not exceed 1 percent of the amount that would be generated if all the cladding surrounding the fuel, except that around the plenum volume, were to react [10 CFR 50.46(b)(3)]
- Calculated changes in core geometry shall be such that the core remains amenable to cooling [10 CFR 50.46(b)(4)]
- After the successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat removed for the extended period of time required by the long-lived radioactivity in the core [10 CFR 50.46(b)(5)]

Westinghouse's realistic LBLOCA methodology for AP600 meets these criteria as follows:

(1) Westinghouse's method of determining the 95th percentile PCT is summarized in Section 21.6.3.9 of this report.

(2) To determine the maximum calculated local oxidation, Westinghouse used the same approach as for three- and four-loop plants in the approved methodology.

(3) Determination of the maximum core wide hydrogen generation is on the basis of an evaluation using the PCT and maximum local oxidation.

(4) According to their AP600 calculations, Westinghouse provided sufficient justification to conclude that the coolable geometry criterion is met when 10 CFR 50.46 PCT and oxidation criteria are met. This is the same approach as for three- and four-loop plants.

(5) Westinghouse meets the long-term cooling criterion for AP600 using methods that were reviewed separately.

The staff's review of the PCT methodology determined that it is adequate to meet NRC requirements of realistic LBLOCA analyses. As stated earlier, Westinghouse's methods for determining the 95th percentile PCT were the subject of the review summarized in this report and discussed in a number of sections.

The review considered the following items in concluding that the three- and four-loop plant local oxidation methodology was applicable to AP600. First, the fuel used in AP600 is not significantly different from that used in three- and four-loop plants. Second, the AP600 LBLOCA response is similar to that for three- and four-loop plants. Finally, in the response to comment 15 in its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse stated that the 95th percentile PCT was below 927°C (1700°F), the limit for significant metal-water reaction to

begin. The low PCT results in the 95th percentile of the peak local oxidation equaling 0.2 percent versus the 10 CFR 50.46 limit of 17 percent. These three factors (similar fuel, similar LBLOCA response, and low PCT calculation which results in a low local oxidation calculation) demonstrate that the approach for the approved methodology in the area of local oxidation calculation is applicable to AP600.

For the core-wide oxidation, Westinghouse determined that the small peak local oxidation and the low 95th percentile PCT provides substantial margin for the core-wide oxidation calculation. As noted above, the low PCT calculated for AP600 results in a negligible peak local oxidation calculation. These factors (low PCT and local oxidation calculations) demonstrate that an evaluation approach for the core-wide oxidation calculation is applicable to AP600.

For the coolable geometry criterion, the staff agrees that meeting 10 CFR 50.46(b)(1) and (2) ensures that a coolable core geometry will be maintained. Past NRC experience has shown that meeting the PCT criterion (10 CFR 50.46(b)(1)) ensures that changes in core geometry because of the LBLOCA transient (e.g., cladding swelling and burst and LOCA loads) do not prevent adequate core cooling as evidenced by the highest calculated PCT. This is consistent with 10 CFR Part 50, Appendix K, Evaluation Models and the use of a realistic LBLOCA methodology for the PCT and oxidation calculations would not change this. Also, a similar argument was used in the approval of the Westinghouse-realistic methodology for three- and four-loop plants. The staff judged that AP600-specific changes do not invalidate the same approach for AP600. In addition, in the response to comment 15 in its letter NSD-NRC-97-5171, dated June 10, 1997, Westinghouse noted that no fuel rod rupture is calculated in the AP600 LBLOCA analyses because of the low PCT calculated.

On the basis of the above, the staff concludes that for AP600, Westinghouse has met, the requirements of those portions of 10 CFR 50.46(a)(1)(i), that were the focus of this review. Westinghouse also has demonstrated methods adequate to show compliance with 10 CFR 50.46(b)(1) - (4) for the AP600. Compliance with 10 CFR 50.46(b)(5), long-term cooling, is separately reviewed.

21.6.3.17 Conclusions and Limitations

The realistic AP600 LBLOCA methodology submittal by Westinghouse was reviewed to determine its compliance with 10 CFR 50.46 and compatibility with NRC guidance in RG 1.157 and the CSAU methodology. On the basis of the review of the information provided by Westinghouse in WCAP-14171-P and responses to NRC questions, the staff finds the Westinghouse-realistic AP600 methodology acceptable subject to the following methodology and application restrictions.

The application restrictions include the following:

(1) Section 21.6.3.16 of this report discussed Westinghouse's methods for meeting the five criteria listed in 10 CFR 50.46(b). AP600 long-term cooling issues are being addressed separately. Therefore this evaluation, as documented in this section of the FSER, does not apply to the application of WCOBRA/TRAC to AP600 long term cooling. AP600 long term cooling is evaluated in Section 21.6.4 of this report. The staff's review of WCOBRA/TRAC in this section of the FSER is restricted to the initial phase of the

LBLOCA. As seen in Figures 21.6-1 and -2 of this report, this corresponds to the end of the reflood phase which is at about 130 seconds.

(2) Application of the Westinghouse methodology and WCOBRA/TRAC to SBLOCA was not considered in this review.

(3) Based on National Research Universal (NRU) reactor assessment results noted during the review of the approved methodology, there is some uncertainty in the transient rod internal pressure (RIP) calculation that will affect the burst temperature criterion in WCOBRA/TRAC analyses. The approved methodology review found Westinghouse's uncertainty methodology adequately accounts for the uncertainty in transient RIP for local effects. However, Westinghouse also calculates hot assembly (HA) rod burst in the full WCOBRA/TRAC analyses called for in its methodology. If WCOBRA/TRAC calculates a HA rod reflood PCT greater than 871°C (1600°F) but not rod burst, Westinghouse in letter NSA-SAI-96-156 dated April 30, 1996 (List II, Item 2), committed to increasing the initial RIP in the WCOBRA/TRAC HA rod until burst is calculated and choosing the more limiting of the burst and non-burst cases. This adequately accounts for transient RIP uncertainties and their effect on rod burst in the WCOBRA/TRAC runs.

(4) On CQD page 7-24, Westinghouse stated the fuel pellet thermal expansion model in MATPRO-11 (Revision 1, NUREG/CR-0497), used in WCOBRA/TRAC was simplified by omitting the corrections for mixed oxide (Pu) fuel. In Westinghouse letter NSA-SAI-96-156 dated April 30, 1996 (List II, Item 6), Westinghouse committed to resubmitting the relevant WCOBRA/TRAC models for NRC review if the code will be used to analyze US licensed plants with mixed oxide fuels.

(5) In the approved uncertainty methodology, a number of assumptions for distributions were supported using plant-specific data; therefore, in Westinghouse letter NSA-SAI-96-156 dated April 30, 1996 (Attachment 5), and in Westinghouse letter NSA-SAI-96-167 dated May 9, 1996 (Attachments 1 and 2), Westinghouse agreed to verify the following assumptions on a plant-specific basis:

(a) The process used to account for the response surface uncertainty assumes the data points are normally distributed, with constant variance, around a straight line. The normality must be checked for each phase of the accident for each plant.

(b) HOTSPOT PCTs are normally distributed. This must be checked at each point where the HOTSPOT PCT is varied in a Monte Carlo sample (i.e., the points used to build the response surface for the HOTSPOT standard deviation ($\sigma_{\alpha\beta\bullet}$) and the validation points).

(c) Response surface for $\sigma_{\alpha\beta\bullet}$ is accurate or conservative. This should be checked by comparing the response surface estimate with the Monte Carlo standard deviation at each validation point. Westinghouse identified the methodology to this on a plant-specific basis in Westinghouse letter NSA-SAI-96-167 dated May 9, 1996 (Attachment 1).

(6) The distributions corresponding to WCOBRA/TRAC uncertainty on the basis of experiments ($\sigma_{WC/T}$) and the uncertainty as a result of experimental data scatter (σ_{2p}) will be checked for normality if the code is modified or the assessment data base changes. See Westinghouse letter NSA-SAI-96-156 dated April 30, 1996, List III, Item 2.

(7) Westinghouse, in letter NSA-SAI-96-156 dated April 30, 1996 (List II, Item 10), committed to use the multiplier given in Attachment 4 of the same letter to account for rod-to-rod radiation effects in the heat transfer multiplier database.

(8) It is noted that the Westinghouse response in letter NSA-SAI-96-028 dated January 26, 1996 (Attachment 5), derived the expressions for the shear stress to the wall and to the vapor shown in CQD Equations 6-120 and 6-121. Westinghouse concluded that the wall shear stress equation used the incorrect friction factor. To assess the effect, Westinghouse reevaluated FLECHT-SEASET Test 31805 with a corrected version of WCOBRA/TRAC. There was little impact on the PCT, and the results from the corrected code version had slightly later quench times. Westinghouse concluded the effect was small, and the NRC agreed. Therefore, Westinghouse proposed that the error be tracked and corrected when other changes to the code are required.

Application restrictions on the AP600 methodology include the following criteria:

(1) Approval of Westinghouse's methodology depends on the time step sizes used to show small mass and energy errors and used in PWR time step convergence studies (see Volume 4, question 50, of Westinghouse letter NTD-NSA-MYY-95-15 dated May 12, 1995). If the time step sizes used in the methodology change, Westinghouse must justify results similar to those identified above are obtained with the new time step scheme.

(2) Westinghouse, in letter NSA-SAI-96-156 dated April 30, 1996 (List II, Item 8), committed to not changing the value and range of the broken loop cold leg nozzle loss coefficient for plant specific applications. Also, the values developed apply only to LBLOCA before core quench.

(3) If the 1 percent core wide oxidation limit is exceeded with the base methodology, Westinghouse in letter NSA-SAI-96-156 dated April 30, 1996 (List III, Item 5), committed to identifying in the licensing submittal or the engineering report which of the options described in its response to Volume 2, question 62, of Westinghouse letter NSA-SAI-96-156 dated April 30, 1996 (Attachment 12), were used to bring the calculated core wide oxidation below the 1 percent limit by reducing the margin in the calculated results.

(4) In WCAP-14171, Revision 1, Westinghouse documents the application of the its LBLOCA best-estimate methodology to the AP600. In developing that methodology some parameters were not included in the uncertainty analysis. Some of these parameters were conservatively bounded while others were judged to have an insignificant impact on the low-calculated PCT results and for simplicity were not considered. Among the latter set of parameters are (1) the AP600-unique passive safety system (CMTs and PRHR) modeling, and (2) fuel rod oxidation models. A bounding assessment of the potential impact of the PRHR and the CMT systems modeling on the

calculated PCT was obtained by eliminating those systems from the LBLOCA transient calculation. As noted on page 2-33 of WCAP-14171, the effect of these systems is minor. The blowdown phase PCT values for the WCOBRA/TRAC case in which these systems were eliminated each decreased by 3°C (5°F) from the base case blowdown PCT results. The reflood phase PCT results in the WCOBRA/TRAC case without PRHR and CMT systems modeled is the same as the base case reflood PCT results.

The AP600 LBLOCA 95 percent PCT values for blowdown and reflood are reported in Table 15.6.5-9 of the SSAR. The limiting PCT is that for the blowdown, at 913°C (1676°F). In the event that either the blowdown or reflood phase PCT exceeds 941°C (1725°F) in a future AP600 best estimate methodology large break LOCA analysis for any reason, Westinghouse shall perform the actions described below:

(a) Repeat the global model matrix of calculations and the final 95 percent uncertainty calculations.

(b) Address the sensitivity to the CMT and RHRS modeling parameters which are not included in the AP600 uncertainty methodology. Repeat the study that identifies the PCT sensitivity to PRHR/CMT elimination, and add the blowdown and reflood PCT impacts as a bias to their respective 95 percent PCT results.

(c) Perform an analysis of the maximum local oxidation using the techniques approved for 3/4 loop plant applications to show compliance with the applicable 10 CFR 50.46 criteria. A transient with PCT in excess of the 95 percent PCT value identified in step (a) above, augmented by the biases identified in step (b) above, would be used for the oxidation analysis. The core-wide oxidation analysis would also be performed using the methods approved for use in 3/4 loop plant applications.

(d) The results of these calculations will be submitted for staff review before their implementation.

21.6.4 WCOBRA/TRAC Computer Code for Long-Term Cooling

The staff has evaluated Westinghouse's use of the WCOBRA/TRAC computer code for long-term cooling in the AP600 design, as described below.

21.6.4.1 Introduction

The long-term cooling (LTC) phase of a LOCA initiates with the establishment of steady-state flow into the reactor vessel from the IRWST through two DVI lines. As the IRWST empties, the source of vessel injection flow switches to the containment sump. The cooling process has no direct parallel in operating reactors. No credit for pumps are used in any phase of the event, and the reactor coolant water boiloff in the core is condensed in the containment. The condensate is returned by gravity to the IRWST or containment sump, and the heat is transferred to the environment through the containment shell. The coolant then re-enters the reactor vessel from the IRWST or sump by gravity. These systems are designed to provide adequate reactor cooling for an extended period of time, without outside intervention or supply of power.

Westinghouse evaluated the LTC system performance of the AP600 using the WCOBRA/TRAC code using a discontinuous "window" methodology. A description of the general methodology and analysis of LTC events using WCOBRA/TRAC can be found in the AP600 SSAR for Chapter 15.

In this evaluation, the NRC staff reviewed the application of the WCOBRA/TRAC code and associated windows methodology for calculating the long-term cooling performance of the AP600 ECCS design for conformance to the requirements of 10 CFR 50.46 and Appendix K of 10 CFR 50.

Westinghouse validates the use the WCOBRA/TRAC code to predict the thermal hydraulic response of the AP600 design during the LTC phase of LOCAs, in WCAP-14776, "WCOBRA/TRAC, OSU Long-Term Cooling Final Validation Report." The LTC phase is defined as the time from the stabilization of the injection from the IRWST until the reactor is recovered. (In this context stabilization of the IRWST injection indicates quasi-steady state flow after opening of the squib valves. Reactor recovery indicates that the reactor is in a stable shutdown configuration under operator control). The primary coolant pressure during the LTC phase is near the containment pressure because the system depressurizes during the early part of the transient.

The AP600 design relies on passive safety systems to supply coolant to the core during the LTC phase, which is characterized by gravity driven flow from the IRWST into the reactor vessel. IRWST injection to the vessel follows vessel depressurization through the ADS and particularly ADS stage 4 (ADS-4) which vents directly into the containment. Hot water and steam flow out of the RCS through valves or the break depending on break location. The hot water flows through the ADS-4 valves (or the break) to the sump while the steam flows to the containment shell, where it is condensed by the PCS and returned to the IRWST. Heat transfer through the containment shell to the atmosphere via the PCS provides the ultimate heat sink for the core decay heat. The IRWST supplies most of the flow into the reactor vessel during the early portion of the LTC phase while the sump supplies most of the flow during the latter portion of the LTC phase.

WCOBRA/TRAC was previously validated for LOCAs in existing pressurized water reactors (PWRs) using a wide range of low-pressure gravity reflood tests. This validation is generally applicable to the AP600 LTC phase. However, additional validation of WCOBRA/TRAC was required for the AP600 LTC phase because of the passive safety systems in the AP600 design. This was accomplished using data from the APEX facility at OSU.

In WCAP-14776, Westinghouse describes the validation of WCOBRA/TRAC using the APEX/OSU data. The APEX facility at OSU is described briefly in Section 21.6.4.2 of this report. The issues identified during the review and evaluation are summarized in Section 21.6.4.4. Conclusions and references are presented in Section 21.6.4.5.

21.6.4.2 The APEX Facility at OSU

The OSU test facility is a 1/4-height (1/192 volume) scaled model of the AP600 design which simulates the following systems: RCS, steam generator (SGS), passive core cooling (PXS), ADS, lower containment sump (LCS), CVS, and normal residual heat removal (RNS). The facility is designed to simulate SBLOCAs in the AP600 design. However, in this review it is the

late part of the LOCA transients (i.e., the LTC phase) that is used for the validation of the WCOBRA/TRAC. The facility operates at reduced pressure in the early part of the SBLOCA but at prototypic pressures during the LTC phase of the transient. The HX function of the containment is simulated to the extent that an equal liquid mass to the condensate return is introduced into the system.

The facility is constructed to reflect the AP600 design geometry including piping routings and vertical and horizontal locations of all tanks and vessels. In addition, it includes the ability to measure single and two-phase break and ADS flows. The IRWST water volume and tank height are appropriately scaled to represent the AP600. In addition the IRWST simulation includes venting to the containment and overflowing to the sump. The sump can be pressurized to simulate the effect of a pressurized containment during the LTC phase.

21.6.4.3 Summary of the WCOBRA/TRAC Validation Report

In WCAP-14776, Westinghouse describes the validation of WCOBRA/TRAC for the LTC phase in the AP600. Westinghouse describes the important phenomena occurring during the LTC phase, the WCOBRA/TRAC code, the input model used to represent the OSU facility, and the validation approach. The PIRT is in Section 1 of the WCAP. The model description and nodalization are in Section 2. The verification approach and the WCOBRA/TRAC model changes and improvements are presented in Sections 3 and 4 respectively of WCAP-14776. Four OSU experiments were used in the validation:

(1) 50.8 mm (2 in.) cold leg break (Test SB01)

(2) Double-ended break in the pressure balance line of a CMT (Test SB10)

(3) Double-ended break in a DVI line (Test SB12)

(4) 12.7 mm (0.5 in.) cold-leg break (Test SB23)

The calculation method utilizes the "window" technique, which consists of analyses of judiciously chosen time segments in the evolution of the LTC. This is possible due to the slow evolution of the transient (in this phase) which is driven by the decay heat and the perturbations introduced by transition points. The windows are chosen at critical times in the evolution of the transient such as to represent the most conservative situations. The choice of window location is facilitated by an approximate solution of the LTC transient using the WGOTHIC code. WGOTHIC is used only to supplement physical reasoning and knowledge of the anticipated physical phenomena.

In Section 5 of WCAP-14776, Westinghouse describes the analysis of a window for each experiment. Each window was 1000 seconds long and corresponded to a time period near the end of IRWST injection and the switch over to sump injection.

The initial conditions for the calculations at the start of the windows were obtained using measured test conditions as a guide. These initial conditions included the liquid levels and fluid temperatures in the downcomer, core, and upper plenum components; the temperatures of the vessel internals and the heater rods; and the system pressure.

Sensitivity calculations of Tests SB01 and SB10 were performed to investigate the effects of the initial conditions used at the start of the window. The initial collapsed vessel level was varied from 75 percent of the heated length in the core to the top of the hot leg. The initial downcomer temperature varied from 60 to 88°C (140 to 190°F) in test SB01 and from 66 to 100°C (150 to 212°F) in Test SB10. These sensitivity calculations showed that the results during the first 400 seconds of the window were affected by the initial conditions. However, the quasi-steady state results obtained during the latter portion of the window were not affected by the initial conditions, indicating that the influence of the initial conditions was limited to the first 400 seconds of the window and that the boundary conditions determined the quasi-steady state results.

The boundary conditions applied to the calculations during the windows were also obtained from test data. The principal boundary conditions were the core decay heat, the pressure and temperature in the IRWST, the level, pressure and temperature in the sump and the break separator.

WCAP-14776 contains three appendices. Appendix A describes the results of additional calculations, including those in which the time duration of the window was extended beyond 1000 seconds. Appendix B provides an overview of the windows methodology, the validation of WCOBRA/TRAC for analysis of the LTC phase, and application of the windows methodology to AP600 design calculations. Responses to staff RAIs are contained in Appendix C.

21.6.4.4 Resolution of Issues

The staff's review of WCAP-14776 identified the following three major issues:

(1) the validity of the windows method for the analysis of the LTC phase

(2) the sufficiency of the range of validation compared to the intended application of WCOBRA/TRAC to the AP600

(3) the adequacy of the validation results

21.6.4.4.1 Validity of the Windows Method

The following four sub-issues arise with the windows approach:

(1) determination of appropriate initial conditions

(2) determination of appropriate boundary conditions

(3) the duration of the window

(4) the identification of the most limiting window during the transient

Relative to the determination of initial conditions, the number and location of instruments in the OSU facility was not sufficient to provide initial values of thermodynamic states and flow rates for all the nodes in the WCOBRA/TRAC model. However, Westinghouse was able to use the existing OSU measurements as a guide to determine the initial conditions. Sensitivity calculations described and summarized in Section 2 of WCAP-14776 showed that the initial

conditions affected the results in the first 400 seconds of the window but did not affect the quasi-steady state results obtained during the latter portion of the window. Because the influence of the initial conditions is limited to the early portion of the window, the actual values of the initial conditions are not critical. Therefore, the staff concludes that the sensitivity calculations performed by Westinghouse adequately address the sub-issue related to initial conditions.

The second sub-issue is related to the determination of appropriate boundary conditions, which govern the quasi-steady state results. Boundary conditions of core decay heat, pressure and temperature in the IRWST, and level, pressure, and temperature in the sump and the break separator were applied in the WCOBRA/TRAC validation calculations. These boundary conditions are all on the basis of OSU measurements. Thus, the staff concludes that these boundary conditions are appropriate for the validation calculations.

The third sub-issue is related to the duration of the window. The validation calculations described in Section 5 of WCAP-14776 were run for 1000 seconds. The staff questioned if the window length was sufficient to produce a quasi-steady state result since small differences in the calculated inflows and outflows could accumulate over time and cause a divergence between the calculated and measured collapsed liquid levels in the reactor vessel. Vessel levels are important because they determine if the core remains covered and coolable during the LTC phase.

Westinghouse addressed the question by performing extended time calculations, at least 3000 seconds long, which are described in Appendix A of WCAP-14776. The first extended window for Test SB01 began at 1260 seconds, corresponding to the start of IRWST injection, and continued until 4600 seconds. The vessel levels increased during this window because of the relatively high flow rates of cold water from the IRWST, which was nearly full at the start of the window. The second extended window for Test SB01 began at 8000 seconds, about midway between the start of IRWST injection and the switch over to sump injection, and continued until 11,000 seconds. This window covered a period when the collapsed levels in the vessel decreased because of steaming in the core. The third calculation simulated Test SB10 and extended the calculation described in Section 5 of WCAP-14776. This calculation began at 13,500 seconds and continued until 16,500 seconds, representing a time period from near the end of IRWST injection through the switch over to sump injection. No divergence in the solution was detected for any of the windows. The quasi-steady state solutions were as accurate in comparison to data at 3000 seconds as they were at 1000 seconds.

In Appendix A of WCAP-14776, Westinghouse also describes two additional sensitivity calculations of Test SB01 that investigated the possible divergence between the calculated and measured results. In the first calculation, the initial conditions at 1260 seconds were applied at 3600 seconds, the calculation was extended to 4600 seconds, and the results were compared to the calculation from 1260 seconds to 4600 seconds described above. The results at 4600 seconds were similar for both calculations. The second sensitivity calculation was run from 8000 seconds to 9000 seconds, but the IRWST level was set at 0.8 m (2.5 ft) above the measured value for 200 seconds and then returned to the measured value for the remainder of the window. The results from this sensitivity calculation were compared to those described above. While the early portion of the solution reflected the difference in boundary conditions,

essentially identical results were obtained for the base and sensitivity calculations well before the end of the 1000-second window. These sensitivity calculations indicated that perturbations in the initial conditions and boundary conditions did not cause a divergence in the calculated results. Therefore, the sub-issue concerning the duration of the window is resolved by the additional calculations described in Appendix A of WCAP-14776.

The last sub-issue concerns the identification of the most limiting period during the LTC phase. The most limiting period during this phase occurs because of the worst combination of decay power, injection flow rate, and injection temperature, which for the OSU experiments generally occurred shortly after the switch over to sump injection because of the relatively high core power, low injection flow rate, and high injection temperature. The analyses described in Section 5 of WCAP-14776 concentrated on the time period near the end of IRWST injection and the switch over to sump injection. The calculation of Test SB10 described in Appendix A of WCAP-14776 extended about 2500 seconds after the start of sump injection. Thus, WCOBRA/TRAC was validated at or near the most limiting conditions during the OSU experiments.

The review of WCAP-14776 indicates that the sub-issues related to the validity of the windows approach for the LTC phase are all resolved. What makes the windows method viable is the insensitivity of the analytical solution to changes in the initial conditions. Thus, the windows methodology is judged to be adequate for modeling the LTC phase.

21.6.4.4.2 Range of Validation

There are two major issues concerning the range of validation: (1) the relationship of the OSU facility parameter ranges to the AP600 design and (2) whether the validation cases analyzed were sufficiently broad in terms of break size, break location, and time to support the intended application of WCOBRA/TRAC for the LTC phase in the AP600 design.

The relationship of the OSU facility to the AP600 design is presented in WCAP-14270 "AP600 Low-Pressure Integral Systems Test at Oregon State University, Facility Scaling Report" which provides a scaling analysis of the OSU facility, including a comparison of the ideal and the as-built facility dimensions. In WCAP-14727, "AP600 Scaling and PIRT Closure Report," Westinghouse documents the process used to assure that the AP600 test program sufficiently addresses important thermal-hydraulic phenomena and systems interactions issues, and therefore, adequately supports code validation. The contents of WCAP-14270 and WCAP-14727 were reviewed separately and are discussed in other sections of this chapter.

Regarding the cases analyzed, the concern is whether the validation included parameter ranges sufficiently broad to support the application of WCOBRA/TRAC to the LTC phase of an AP600 design LOCA transient.

According to Appendix B of WCAP-14776, WCOBRA/TRAC, the windows method will be applied to a spectrum of break sizes and locations for the AP600 design. The break spectrum will include both SBLOCAs and LBLOCAs.

Window analyses were performed for four OSU SBLOCA experiments. These experiments included a range of break sizes (from a 12.5 mm (0.5 in) break to a double-ended break in a DVI

line) and a range of break locations (e.g., cold leg, pressure balance line, and DVI line). The break sizes in these experiments adequately cover the range of break sizes for SBLOCAs.

LBLOCA experiments were not performed in the OSU facility, and hence, no large-break validation calculations were performed for the LTC phase. The OSU experiments, taken in total, indicate that behavior during the LTC phase was similar in all the tests. The similarity of behavior occurs because of the similarity of the once-through flow from the IRWST and/or sump and out the ADS-4 lines (or the break in some instances) during the LTC phase. On the basis of these results, the staff expects that the system response during the long-term phase of a LBLOCA will be similar to that observed for a small-break LOCA. Thus, the SBLOCA validation calculations described in WCAP-14776 are judged to be applicable to a large-break LOCA.

The analyses in WCAP-14776 cover a wide range of times, from the beginning of IRWST injection through the switch over to sump injection and beyond. The window choices covered all combinations of maximum decay heat with minimum cooling flow. Likewise it examined conservative initial and boundary conditions. In all instances it was found that there is adequate reactor cooling and adequate core flushing to avoid boron concentration. These validation cases are adequate to demonstrate the capability of WCOBRA/TRAC to calculate the important phenomena occurring during the LTC phase.

On the basis of the above arguments, the staff concludes that the validation cases described in WCAP-14776 are sufficiently broad to support the intended application of WCOBRA/TRAC for the LTC phase in the AP600 design.

21.6.4.4.3 Adequacy of Validation Results

The third major issue is related to the adequacy of the validation results; specifically, whether WCOBRA/TRAC was capable of calculating the important phenomena during the LTC phase.

In WCAP-14776, Westinghouse includes a PIRT that identifies the important phenomena occurring during the LTC phase. In WCAP-14776, Westinghouse also presents validation results for parameters related to the highly ranked PIRT phenomena. These parameters include vessel pressure, DVI flow, ADS-4 flow, and collapsed liquid levels in the core, upper plenum, and downcomer. Comparisons between the calculated and measured values for these parameters are summarized below.

The calculated pressure in the upper plenum during the quasi-steady portion of the window was within the uncertainty of the measurement for each of the four validation cases described in Section 5 of WCAP-14776 and the three extended calculations described in Appendix A. Consequently, the agreement between the calculated and measured pressures is very good.

The calculated total DVI flow was within 15 percent of the measured value of the quasi-steady state during the latter portion of the window for three of the four validation calculations described in Section 5 of WCAP-14776 and all three of the extended calculations described in Appendix A of WCAP-14776. The uncertainty for the total measured DVI flow is about 15 percent, and the calculated DVI flows were generally within the uncertainty of the measurement. The deviation was larger for the validation calculation for Test SB01. As shown in Table 440.589-1 of Appendix C of WCAP-14776 the total calculated DVI flow exceeded the measured DVI flow by

26 percent during the quasi-steady state for Test SB01. However, the calculated total flow into the vessel, which includes the flow from the break as well as from the DVI lines, was within 5 percent of the measured value.

The calculated total ADS-4 flow was within 15 percent of the measured value for the quasi-steady state during the latter portion of the window for all four of the validation calculations described in Section 5 of WCAP-14776 and two of the three extended calculations described in Appendix A. For the other extended calculation described in Appendix A of WCAP-14776 Test SB01 near 10,000 seconds, the calculated ADS-4 flow exceeded the measured value by nearly 50 percent. However, the total calculated flow out of primary coolant system, which includes the flow out of the break as well as from the ADS-4 valves, was in reasonable agreement with the measured values. The primary cause of this discrepancy was that the downcomer level was high enough to cover the cold leg break with liquid in the test, leading to a relatively high-break flow and low ADS-4 flow, while the downcomer level was too low to cover the break with liquid in the calculation, leading to a relatively low-break flow and high ADS-4 flow; however, these results are conservative because the core remains covered.

The calculated collapsed liquid levels in the vessel were in reasonable agreement with the measured values. In Appendix C of WCAP-14776, Westinghouse shows that the calculated quasi-steady state collapsed liquid levels in the downcomer were less than the measured values, on average, the calculated collapsed liquid levels in the downcomer were 10 percent, or 18 cm (7 in.), less than the measured values except for SB01 as noted above. The core remained nearly full in the calculations and the tests. The calculated collapsed core liquid levels were within 10 percent (8 cm (3 in.)) of the measured values during the quasi-steady states. The calculated collapsed liquid levels in the upper plenum during the quasi-steady states were between 8 and 18 cm (3 and 7 in) less than the measured values.

On the basis of the review of WCAP-14776, the staff concludes that WCOBRA/TRAC adequately calculates the important phenomena identified in the PIRT during the LTC phase. The code is generally able to predict the important parameters to within, or near, the uncertainty of the data. The code also correctly calculates that the core remained covered with liquid and that liquid would be stored above the core in the upper plenum and hot legs during the LTC phase.

21.6.4.4.4 DSER Issues

A number of open items related to the use of WCOBRA/TRAC for the LTC evaluation of AP600 ECCS analyses were noted in the SDSER for the AP600. The following is a discussion regarding the resolution of the SDSER open items.

Open Item Discussion

Open Item	Discussion
21.6.4-1	The early part of any LOCA transient (i.e., blowdown, natural circulation, or ADS blowdown to initiation of IRWST injection) is of interest in this part of the review for the determination of the initial and boundary conditions. However, WCOBRA/TRAC LBLOCA, NOTRUMP, and WGOTHIC were still under review and, consequently, approval of WCOBRA/TRAC for LTC calculations was

contingent on the initial transient codes being approved for the AP600. Although Westinghouse stated that the LTC transient is not sensitive to variations in the initial conditions, this had not been demonstrated. The acceptability of the initial and boundary conditions for the window calculations was identified as DSER Open Item 21.6.4-1.

Westinghouse performed calculations with severe perturbations of the initial conditions and determined that the solution will converge to the same value within 300-500 seconds. This behavior was not limited to some parameters, rather it is a feature of the method suggesting that a stable and robust solution is obtained using the WCOBRA/TRAC code for the LTC phase. Westinghouse provided additional documentation on this in its letter NSD-NRC-96-4877, dated November 6, 1996. Therefore, DSER Open Item 21.6.4-1 is closed.

21.6.4-2 For the AP600 LTC analyses of a DECLG break, failure of one ADS-4 valve was assumed as the single failure. No qualifying arguments were proposed for the selection of this as the worst-case single failure. Westinghouse needs to justify the choice of single failure.

Westinghouse performed sensitivity calculations to investigate the worst single failure for a 5 cm (2 in.) cold leg break in the AP600 design as described in Section 15.6.5.4c of the SSAR. The cases analyzed considered a failure of a valve in the injection line to open and a failure of an ADS-4 valve to open. The sensitivity calculations showed that the ADS-4 failure was more significant. This was expected because the injection lines use parallel sets of valves so the injection capability is relatively unaffected by the failure of a single valve to open. Since the LTC phase is characterized by depressurization and discharge flow through the ADS-4 the most significant single failure is that of an ADS-4 valve to open. Therefore, on the basis of the calculations described in Section 15.6.5.4C of the SSAR, DSER Open Item 21.6.4-2 is closed.

21.6.4-3 A containment pressure of 172 kPa (25 psia) was assumed for the analysis of AP600 LTC analyses of a DECLG break. This pressure was calculated as the endpoint pressure in the LBLOCA analysis using WGOTHIC. No qualifying arguments were proposed to justify this pressure and why it would stay constant throughout the LTC part of the transient. Analysis of containment and vessel pressure versus time in the evolution of the LTC scenario was needed.

In Appendix B of WCAP-14776, Westinghouse describes the containment pressure calculation for the LTC phase. WGOTHIC is executed from the start of the LOCA, through the short-term phase, and long enough into the LTC phase to provide boundary conditions for the windows analyzed with WCOBRA/TRAC. A constant containment pressure can be used during a 3000 to 4000 second window because pressure variations are not significant. WGOTHIC was used to predict a conservatively low containment pressure during the LTC phase, which would decrease differential pressure between the sump and the vessel. Decreased differential pressure could lower the flow of coolant into the vessel. In response to RAI 440.646, Westinghouse performed a window calculation using

WCOBRA/TRAC in which the containment pressure was reduced even further, from 172 kPa (25 psia) to 138 kPa (20 psia). After quasi-steady state results were obtained, the calculated collapsed liquid levels in the vessel were somewhat reduced at the lower pressure. However, the core remained covered and coolable at the lower pressure. On the basis of these results, DSER Open Item 21.6.4-3 is closed.

21.6.4-4 The sump injection window of the DECLG break covered the early part of sump injection at which time the sump water was still subcooled by about 11°C (20°F). The driving head was less than 1.8 m (6 ft) of water. If the sump water were saturated, the pressure differentials would be less favorable for sump injection. A detailed account of the pressure variation in the vessel and the pressure losses because of the flow in the vessel was needed.

In response to RAI 440.645.a, Westinghouse stated that the sump injection temperature was increased from the WGOTHIC calculated value to the saturation value based on the total containment pressure for the 51-mm (2-in) cold-leg break cases presented in the SSAR. Increasing the sump injection temperature to the saturation temperature is clearly conservative. Therefore, DSER Open Item 21.6.4-4 is closed.

21.6.4-5 During the sump injection window of the DECLG break, the total injection flow was calculated at 31.8 kg/sec (70 lb/sec), while the total outflow from the vessel was calculated at 34 kg/sec (75 lb/sec). The discrepancy is not significant in the context of this window width, but it may indicate a problem for the mass and momentum solution of the transient.

In response to RAI 440.589, Westinghouse demonstrated that the vessel inflows equaled the total vessel outflows during the OSU validation calculations. Extended-time calculations were also described in Appendix A of WCAP-14776. These extended calculations showed that the quasi-steady state results were as accurate in comparison to data during long windows as they were during short windows. These results indicate that the solution is correct for the mass balance during the windows. Thus, DSER Open Item 21.6.4-5 is closed.

21.6.4-6 During the sump injection window of the DECLG break, the void fractions in the upper part of the core were calculated. The void distribution results did not seem to be realistic. For example, there was a void in the fourth-segment elevation lasting about 140 seconds, while at the same time the fifth segment shows low voiding and the sixth segment again shows high voiding. Westinghouse was requested to provide clarification of the void fraction predictions.

In response to RAI 440.646, Westinghouse showed some AP600 design sensitivity calculations in which the upper half of the core was nearly voided for up to 100 seconds while the lower half of the core was nearly full of liquid. The voiding of the upper core node was caused by a calculated redistribution of liquid from the core to the upper plenum. Westinghouse stated that the calculated redistribution of liquid was probably exaggerated by the coarse axial nodalization of the core in the LTC model. Core heat up, however, was not imminent because

of the water stored in the upper plenum. Furthermore, even if the core did begin to dry out, the low decay power during the LTC phase, coupled with the short duration of the liquid redistribution, would limit the peak cladding temperature and thus was not a concern relative to safety. The staff agreed that the potential for heat up was not significant on the basis of the results presented and that the void profiles were not of concern. Therefore, DSER Open Item 21.6.4-6 is closed.

21.6.4-7 If the water in the sump was saturated during LTC and the void fraction was higher (1) the peak cladding temperature could be higher and (2) the pressure distributions would be different. The staff requested that Westinghouse address these concerns.

DSER Open Item 21.6.4-7 is closed on the basis of the same arguments presented above for DSER Open Item 21.6.4-4.

21.6.4-8. The two windows chosen for the DECLG-break LTC analysis did not convey a sense of continuity. Coolant inventory as a function of time for the transient was not included in the analysis. Westinghouse was questioned if more windows were needed to make convincing arguments for the effectiveness of the passive cooling system.

The most limiting period during the LTC phase occurs because of the worst combination of decay power, injection flow rate, and injection temperature. The OSU validation calculations described in WCAP-14776 were performed for a variety of times, ranging from the onset of IRWST injection, until about 2500 seconds after the switch over to sump injection. The OSU experiments and the validation calculations showed that the most limiting period during the LTC phase occurred shortly after the switch over to sump injection because of the relatively high-core power, low-injection flow rate, and high-injection temperature. Thus, the windows analyzed cover the most limiting period of the LTC phase and DSER Open Item 21.6.4-8 is closed.

21.6.4-9 Westinghouse stated that there is adequate flow to cool the core and to prevent boron precipitation. No specific arguments were presented to support the statement about boron precipitation. It is apparent that during LTC there will be significant temperature gradients in the vessel plenum and in the sump. Therefore, boron precipitation will be present to some degree. However, for the time scale envisioned in the LTC analysis, such precipitation could cause either a flow obstruction or recriticality; thus, boron precipitation should be addressed.

Westinghouse demonstrated that there is continuous water flow through the ADS-4 flow paths, thus, securing renewal of the boron in the pressure vessel and not allowing boron concentration in the vessel because of continuous boiling. Water flow through the ADS-4 has been demonstrated experimentally and is predicted analytically. Therefore, DSER Open Item 21.6.4-9 is closed.

21.6.4-10 Westinghouse stated that the LTC cooling systems are designed to provide adequate cooling indefinitely without outside intervention or a power supply.

However, the following three items were not discussed: (1) the effect of water holdup or diversion in containment, (2) water loss through containment leakage, or (3) the need to replenish any water lost from containment because of leakage.

As described in Appendix B of WCAP-14776, Westinghouse performed calculations with complete leakage between compartments of the containment. These calculations represented the so-called "wall-to-wall flooding" case. This case resulted in reduced liquid levels in the containment sump, reduced injection flow rates, and reduced collapsed liquid levels in the vessel compared to the case without leakage between containment compartments. However, the core was adequately cooled in the case with leakage. These calculations close DSER Open Item 21.6.4-10 related to leakage between containment compartments.

21.6.4-11 LTC is a slowly developing calculation when performing a LOCA analysis. The staff agrees that a window approach is a suitable method to address the entire LTC period. However, the adequacy of the windows selected as representative of LTC event was not resolved.

DSER Open Item 21.6.4-11 is closed on the basis of the same arguments presented in the discussion of DSER Open Item 21.6.4-8.

21.6.4-12. The LBLOCA analysis terminates at the end of accumulator discharge. The LTC analysis initiates with IRWST flow stabilization. There is a gap in the analysis between the end of accumulator discharge and IRWST flow stabilization; this includes CMT discharge and IRWST injection initiation. The staff requested that Westinghouse analyze the portion of the transient between accumulator discharge and IRWST flow stabilization. DSER Open Item 21.6.4-12 is closed on the basis of the same arguments presented in the discussion of DSER Open Item 21.6.4-8.

21.6.4-13 Questions from the review of the experimental and calculated results were forwarded to Westinghouse in a set of RAIs. Acceptable resolution of RAI responses related to WCOBRA/TRAC-LTC was DSER Open Item 21.6.4-13. Westinghouse responded to all of the outstanding RAIs. Therefore, DSER Open Item 21.6.4-13 is closed.

21.6.4-14 Westinghouse was requested to integrate a simulation of the containment response with the AP600 design-simulated transient calculations. This was DSER Open Item 21.6.4-14.

Westinghouse presents an acceptable description of how containment pressure response is integrated with the long-term cooling analyses in Chapter 5 of WCAP-14601, "AP600 Accident Analyses - Evaluation Models." Therefore, DSER Open Item 21.6.4-14 is closed.

21.6.4-15 Westinghouse stated that there were some errors in the flow resistance and heat transfer coefficients used to analyze OSU. Westinghouse committed to reanalyze the simulated transients with corrected parameters.

DSER Open Item 21.6.4-15 is closed on the basis of the same arguments presented in the discussion of DSER Open Item 21.6.4-16 below.

21.6.4-16 Westinghouse was requested to justify the use of a window approach in analyzing the OSU transient (i.e., provide evidence that the solution converges to the correct function regardless of initial condition approximations).

In WCAP-14776, Westinghouse includes revised LTC calculations, that show no significant differences from the original calculations, and a justification of the windows analysis as applied in the OSU experimental results. Therefore, DSER Open Items 21.6.4-15 and 21.6.4-16 are closed. See Section 21.6.4.4 of this report for additional discussion.

21.6.4-17 Regarding the OSU transients used to validate WCOBRA/TRAC for LTC, Westinghouse was requested to explain how the code qualification with respect to pressure differences in the LTC transient is affected by inaccuracies in OSU instrumentation measurements.

DSER Open Item 21.6.4-17 is closed on the basis of the same arguments presented in the discussion of DSER Open Item 21.6.4-18 below.

21.6.4-18 Westinghouse should verify flow direction and the correct evolution of a calculated transient in the OSU analyses.

Westinghouse's revision to WCAP-14776, among other things, justified the calculations in view of the pressure measurement inaccuracies and the interpretation of the OSU experiments with respect to flow direction. DSER Open Items 21.6.4-17 and 21.6.4-18 are closed. See Section 21.6.4.4 of this report for additional discussion.

21.6.4.5 Summary and Limitations

WCOBRA/TRAC is judged to adequately represent the important phenomena occurring during the LTC phase. The code provided accurate predictions of system pressure, injection and ADS-4 flow rates, and collapsed liquid levels in the reactor vessel. Because of the quasi-steady state nature of the phenomena, the windows approach is also judged to be adequate for the LTC phase. Westinghouse shows that the quasi-steady state results at the end of the window are insensitive to the initial conditions assumed at the beginning of the window. Westinghouse also demonstrates that the time span of the windows is sufficient to obtain quasi-steady state results with adequate ranges of break sizes and locations to support the application of the code to the LTC phase for the AP600 design. Therefore, the staff concludes that the use of WCOBRA/TRAC as described in this evaluation using windows is acceptable for application to the AP600 design to demonstrate long-term cooling capability as required by 10 CFR 50.46(b).

Future calculations that use this code for AP600 design applications should ensure the following:

- The nodalization of the AP600 design LTC model corresponds to that used in the OSU calculations

- The window time span results in a quasi-steady state solution.
- The code is not applied outside of the corresponding parameter range from the OSU experiments. In particular, WCOBRA/TRAC is not validated for core dryout and heatup phenomena during LTC because these phenomena were not observed during the OSU experiments.

21.6.5 WGOTHIC Computer Program for Containment DBA Analysis

Summary

This section presents a technical review of the Westinghouse WGOTHIC computer program as used for the licensing analysis of AP600 containment pressure transients resulting from design-basis accidents (DBAs) such as large-break LOCAs and main steamline breaks (MSLBs). WGOTHIC is an adaptation of the GOTHIC computer program with Westinghouse additions to model the AP600 passive containment cooling system (PCS). This review is specifically limited to the use of WGOTHIC in the Westinghouse AP600 evaluation model (EM). In this model, Westinghouse represents the AP600 containment as a network of lumped parameter nodes. The WGOTHIC computer program is used with the Westinghouse AP600 EM to calculate a conservative pressure response for the AP600 passive containment, consistent with current regulatory guidance.

On the basis of this evaluation, the staff has determined that the WGOTHIC computer program, combined with the conservatively biased AP600 evaluation model, is acceptable for the evaluation of the peak containment pressure following a large-break LOCA. For a large-break LOCA, the peak pressure period extends 1200 seconds after initiation of the event. Initially, the DBA blowdown and PCS operation generate a nearly homogeneous distribution of steam and non-condensable gases. In the longer term, the actuation of the fourth stage automatic depressurization system valves (ADS-4), at approximately 1000 seconds, supports a circulation pattern which tends to sustain the homogeneity of the containment atmosphere. Under these conditions, the lumped parameter representation is acceptable for evaluating the AP600 peak containment pressure. Degradations to heat transfer caused by local inhomogeneities tend to be self-correcting; for example, a concentration of non-condensable gases on one section of the containment surface will be offset by other steam-rich sections. These considerations justify use of the WGOTHIC AP600 EM for peak pressure calculations. However, the WGOTHIC AP600 EM is not capable, or qualified, to predict the containment non-condensable distribution.

For the MSLB, the peak pressure period extends 600 seconds after the steamline break. The degree of homogenization is a strong function of break location, direction, and momentum. The MSLB blowdown creates circulation patterns that tend to homogenize the containment atmosphere above the break location sufficiently to accept the lumped-parameter representation for the evaluation of the AP600 peak containment pressure. The Westinghouse EM conservatively places the MSLB at the highest possible location.

The staff therefore concludes that the Westinghouse WGOTHIC computer program, combined with the AP600 EM methodology, can be used to demonstrate that the AP600 containment design meets the requirements of General Design Criteria (GDCs) 16 and 50. The PCS performance characterization is an integral part of this conclusion. For the first 3 hours following

initiation of the PCS, which encompasses the peak pressures, the flow is maintained at a high value and the containment shell wetted surface area is near 100 percent. In the long term (after 3 hours), the PCS flow rates drop, decreasing the shell wetted surface area. After the peak pressure period, the atmosphere may stratify in temperature or non-condensable concentration or both. These phenomena increase the uncertainty of the WGOTHIC EM PCS heat removal calculation.

Regulatory requirements for containment long-term internal pressure performance do not specify quantitative limits. GDC 38 of 10 CFR Part 50, Appendix A requires, in part, that a system to remove heat from the reactor containment be provided which reduces containment pressure rapidly following any loss-of-coolant accident and maintains the pressure at an acceptably low level.

The secondary objective for the long-term analysis is to demonstrate that the long-term pressure remains within the pressure envelope used for containment leakage calculations which support the siting evaluation. Westinghouse conservatively uses the maximum design pressure for long-term leakage calculations. Therefore a specific reduction in pressure is not necessary to determine AP600 compliance with GDC 38 or to support the siting evaluation, as long as the containment pressure rapidly diminishes from the post-event peak.

While a greater degree of uncertainty exists in the long-term WGOTHIC heat removal calculation, the staff concludes that the WGOTHIC computer program, combined with the AP600 EM methodology, is sufficient to evaluate the trend in the long-term pressure response and demonstrate that the GDC 38 requirements have been met. The staff therefore approves the use of WGOTHIC with the AP600 EM methodology, subject to the limitations and restrictions identified in Section 21.6.5.8.3.

21.6.5.1 Introduction

Westinghouse-GOTHIC (WGOTHIC) is a thermal-hydraulic computer program used for the design basis licensing analysis of the AP600 containment. The WGOTHIC computer program is used to conservatively calculate the containment thermal-hydraulic response to mass, momentum, and energy releases from postulated pipe break scenarios, e.g., design basis loss-of-coolant accidents (LOCAs) and main steamline breaks (MSLBs). Westinghouse uses WGOTHIC in a lumped-parameter fashion to evaluate the pressure and temperature response of the AP600 passive containment to design-basis accidents (DBAs). WGOTHIC is documented in the following Westinghouse topical reports:

- WCAP-14382, "WGOTHIC Code Description and Validation," May 1995
- WCAP-14407, "WGOTHIC Application to AP600," Revision 3, April 1998
- WCAP-14967, "Assessment of Effects of WGOTHIC Solver Upgrade from Version 1.2 to 4.1," September 1997

WGOTHIC is a modified version of the GOTHIC containment analysis computer program. The GOTHIC base code is documented in the following reports:

- "GOTHIC Containment Analysis Package Users Manual, Version 4.0," Numerical Applications Inc., NAI-8907-02, Revision 4, June 1994.
- "GOTHIC Containment Analysis Package Technical Manual, Version 4.0," Numerical Applications Inc., NAI-8907-06, Revision 3, September 1993.
- "GOTHIC Containment Analysis Package Qualification Report, Version 4.0," Numerical Applications Inc., NAI-8907-09, Revision 2, September 1993.

The WGOTHIC additions include a special multi-compartment heat structure component, referred to as the "clime" model, used to model the AP600 passive containment cooling system (PCS). Figure 21.6-22 is a schematic of the containment showing the essential features of the PCS. These include the steel shell, a large water storage tank, weirs for flow distribution, and an air flow path through the downcomer, riser, and chimney.

Westinghouse has submitted an application for design certification of the AP600 advanced passive reactor under 10 CFR Part 52. In support of design certification, Westinghouse elected to follow current staff guidance for conservative DBA analyses. To demonstrate that the design meets the requirements of GDCs 16 and 50, the conservatively calculated containment peak pressure is required to be below the AP600 design pressure of 411.6 kPa (45 psig) during the most limiting releases of mass and energy within the containment. The PCS acts to reduce pressure during a design-basis accident by removing energy through the containment shell. Pressure is also reduced by the compliance (i.e., the change in energy storage due to a change in pressure) of the gas within the large containment volume and by heat transfer to in-containment structures. These two mechanisms are essentially the same as in existing large dry PWR containments. However, existing containment designs also have active engineered safety features; sprays, fan coolers, and sump coolers, to remove heat to the ultimate heat sink. The AP600 design does not include active safety-grade heat removal systems. The AP600 PCS is new and unique, and therefore its performance is central to this evaluation.

The primary mechanisms for heat transfer through the containment shell are condensation on the inside of the shell, conduction through the shell, and evaporative cooling on the outside of the shell. Water is released at a controlled rate and flows down the outside of the containment shell and is heated and evaporated. The vapor formed during the evaporation process is carried away by the air flowing through the downcomer, riser, and chimney flowpath. The WGOTHIC evaluation model of the PCS component uses conservatively biased heat and mass transfer relationships for licensing analyses.

Westinghouse uses the WGOTHIC computer program to perform conservative containment licensing analyses for the AP600 design-basis accidents. Westinghouse topical report WCAP-14382 describes the WGOTHIC models and the WGOTHIC qualification and validation process. Westinghouse topical report WCAP-14407 describes the application of the WGOTHIC methodology for conducting design-basis accident (DBA) analysis for the AP600 design. Supplemental documentation has also been provided in WCAP-14967 and in Westinghouse letter DCP/NRC1247, dated February 20, 1998.

21.6.5.2 Review Process and Scope

WGOTHIC has not previously been used for licensing analyses, although its predecessor, GOTHIC, is widely used by utilities. The staff review of WGOTHIC focused on questions and concerns in areas critical to the AP600 evaluation model (EM) methodology. The AP600 EM uses the lumped-parameter modeling feature for containment analyses in support of design certification for the AP600. The WGOTHIC computer program review effort examined the following areas:

- conservation equations
- closure relationships and correlations
- numerical methods and convergence
- general modeling approach and sensitivity studies
- modeling approach for AP600-specific features
- overall applicability/conservatism for AP600 application

The staff review of the WGOTHIC computer program and AP600 specific modeling features, such as the passive containment cooling system (PCS), is documented in Section 21.6.5.4 of this report. Overall applicability of WGOTHIC and the AP600 EM for the containment design-basis accident licensing analysis is documented in Section 21.6.5.7 of this report.

Unlike existing light-water reactor containment designs, the AP600 containment design does not rely on active engineered safety features (e.g., systems such as containment sprays and fan coolers) to demonstrate that the design meets the requirements of GDCs 16 and 50. Instead, the AP600 relies on passive mechanisms, i.e., heat transfer through the containment steel shell to an evaporating water film flowing over the exterior containment surface, referred to as the passive containment cooling system. The AP600 PCS design goal is to provide a reliable first-principle-based heat removal capability. As part of the containment design process, Westinghouse prepared a PIRT in WCAP-14812, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," which identified and ranked the relevant phenomena for AP600 containment. The staff review evaluated the correctness and completeness of this report. The evaluation is documented in Section 21.6.5.5 of this report.

Demonstrating the effectiveness and performance of the PCS heat removal capability requires technical data derived from various scale testing and mathematical modeling. Previously, the need for data was less critical because engineered safety features included robust active cooling systems. However, with the reliance now being placed on passive cooling mechanisms, better analytical capability supported by experimental data is needed to confidently understand and assess the containment response to design-basis accidents. The Westinghouse large-scale test (LST) facility, which was part of the test program developed for the AP600 containment design, provided data to support the AP600 design certification. The staff examined the LST facility scaling, instrumentation uncertainties and distortions, as documented in Section 21.6.5.5 of this report. The staff evaluated the test program developed by Westinghouse, which was used both for developing model conservatisms in the AP600 EM and for WGOTHIC validation studies. These evaluations are documented in Section 21.6.5.6 of this report. WGOTHIC heat and mass transfer relationships were evaluated against small- and intermediate-scale data in WCAP-14326, "Experimental Basis for AP600 Containment Vessel Heat and Mass Transfer Correlations," and WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and

Phase 3,". In Section 21.6.5.6 of this report, these heat and mass transfer relationships are examined to document their appropriateness and conservatism over the AP600 range of applicability.

During the review of the Westinghouse AP600 evaluation model for large-break LOCA and MSLB design-basis accident analyses, the staff identified a number of implementation issues. These issues were documented in requests for additional information (RAIs). To respond to some of these RAIs, Westinghouse performed additional sensitivity studies to demonstrate the conservatism of their overall approach. Section 21.6.5.7 of this report documents the review of the EM and the sensitivity studies.

In a separate effort, the NRC staff performed independent CONTAIN (see "User's Manual for CONTAIN 1.1, A Computer Code for Severe Nuclear Reactor Accident Containment Analysis" dated July 1990) analyses of the AP600 large-break LOCA and MSLB accidents. While not directly comparable, these calculations supported the reasonableness of the Westinghouse sensitivity study results.

21.6.5.3 Applicable Requirements and Acceptance Criteria

The current guidance for demonstrating that a containment design complies with GDCs 16, 38 and 50 is delineated in Chapter 6.2 of the Standard Review Plan (SRP). The SRP addresses acceptance criteria and some specific model assumptions for design basis LOCA and MSLB analyses for all existing containment types. Westinghouse elected to evaluate the PCS performance using these current guidelines. The Westinghouse documentation for the AP600 EM is therefore intended to be consistent with the guidelines in SRP Sections 6.2.1 and 6.2.1.1.A, as well as Regulatory Guide 1.70. Westinghouse also uses approved methods for the LOCA and MSLB mass and energy releases following the guidance provided in SRP 6.2.1.3 and 6.2.1.4, respectively.

21.6.5.3.1 Peak Pressure Criteria (GDCs 16 and 50)

Acceptance criteria for existing containments include a margin between the design pressure and a conservatively calculated peak accident pressure. The margin varies from 10 percent at the construction permit (CP) stage to a peak calculated pressure "less than the containment design pressure" at the operating license (OL) stage. Thus even in instances where much data and information are known, and the staff possessed an independent, confirmatory calculational capability, a 10 percent margin is expected at the CP stage to cover uncertainties in meeting GDCs 16 and 50 after final construction, at the OL stage.

For the AP600 containment, Westinghouse proposed a criterion that the calculated peak accident pressure not exceed the design pressure (a zero-margin criterion). In meeting this criterion, Westinghouse states that it uses a conservative approach consistent with current staff guidelines. For design certification, under 10 CFR Part 52, the staff does not necessarily need the same demonstration of margin as normally expected at the CP stage. An appropriate initial test program, combined with appropriate inspections, tests, analyses, and acceptance criteria (ITAAC), is in place to assure that the assumptions about and performance characteristics of the AP600 containment and the PCS, as used in the licensing analyses, are verified before operation. The PCS ITAAC is described in the AP600 Certified Design Material. The initial test program (ITP) is described in SSAR Chapter 14.

21.6.5.3.2 Long -Term Pressure Analysis (GDC 38)

The objective of the long-term pressure analysis is to demonstrate that the containment design conforms to the objectives of GDC 38 which states;

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The guidance in SRP Item II.b of Section 6.2.1.1.A discusses compliance with GDC 38 and states that the containment pressure should be reduced to less than 50 percent of its peak value within 24 hours of the occurrence of a design-basis LOCA. This assures that the containment leak rate used for the siting evaluation is consistent with the design basis analysis assumption. In current operating reactors, this 50 percent reduction in pressure is credited in the siting evaluation. However, Westinghouse does not credit any leakage reduction due to decreased pressure. The siting evaluation is performed with a constant, design basis leak rate. Westinghouse had originally proposed that the pressure reduction be based on 50 percent of the design pressure to be consistent with current guidelines related to GDC 38. The staff found this approach acceptable, although the peak calculated pressures had been near the design value, because there was no need to demonstrate a pressure reduction for the leak rate assumption used in the siting evaluation.

Late in the review process, Westinghouse determined that it could not meet the proposed long-term objective with the original analysis approach. Westinghouse therefore revised the analytical procedure to credit the effect of two-dimensional (2-D) heat conduction (between wet and dry regions of the containment shell) when less than full coverage of the containment shell is expected. The revised procedure was first presented in Westinghouse letter DCP/NRC 1181, dated May 23,1997, and was discussed at an ACRS meeting in December 1997. Westinghouse did not identify, or at least account for, the need to consider two-dimensional (2-D) heat transfer for the long-term containment pressure response (after 3 hours when the PCS flow rate is first cut back to about one-half its initial value) in selecting the analysis methodology (GOTHIC) and in developing a model for the PCS (WGOTHIC). With the coverage area less than the initially assumed 90 percent, heat would be transferred from the hot, dry regions of the shell into the cooler, wet regions of the shell. To account for this deficiency, Westinghouse performed an ancillary calculation to credit more PCS water in the evaporation process, effectively generating a correction factor, and applied it to the limited PCS flow model (see Section 21.6.5.4.2 of this report).

During the first 3 hours of a DBA event, with the PCS flow rate maintained at 1,665.6 liters/min (440 gpm), the pressure performance envelope is similar to the envelope of existing designs which use active safety systems. When the PCS flow rate is reduced after 3 hours, the

containment tends to repressurize slightly and maintain a pressure somewhere between 218.5 and 273.7 kPa (17 to 25 psig), well below the 411.6 kPa (45 psig) design value, until 30 hours into the event, when a further reduction in the PCS flow rate occurs. When the flow is again reduced after 30 hours, the containment again repressurizes, with the resulting pressure being between 225.4 and 322.0 kPa (18 to 32 psig) for the remainder of the 72-hour design basis performance period of the PCS but continuing to decrease as the decay heat decreases. The difference between the low- and high- pressure estimates is due to effects of 2-D heat conduction when considered in the analyses. As discussed in Section 21.6.5.4.2 of this report, the staff believes that there is a real effect from 2-D heat conduction. However, as an insufficient amount of test data is available to validate this model, the staff is unable to determine how much credit should be given in evaluating the Westinghouse design performance after 24 hours. In addition, the calculated pressure is not used to demonstrate compliance with other regulatory requirements. Whether or not credit is taken for 2-D heat conduction, the staff finds the design to be in compliance with GDC 38 and the containment pressure and temperature following the limiting loss-of-coolant accident are maintained at acceptably low levels. Although the containment pressure response is different from current licensed plants, the PCS is acceptable and consistent with the passive design objectives on which the AP600 PCS is based.

21.6.5.3.3 10 CFR 52.47 Criteria

Demonstrating the effectiveness and performance of the PCS heat removal capability requires technical data derived from various scale testing and mathematical modeling. Previously the need for data was less critical because the engineered safety features included robust active cooling systems. However, with the reliance now being placed on passive cooling mechanisms, better analytical capability supported by experimental data is needed to confidently understand and assess the containment response to design-basis accidents.

The unique characteristics of a passive containment cooling system are explicitly recognized in the regulations governing the evaluation of standard plant designs. 10 CFR 52.47(b)(2)(i)(A) states that, in the absence of a prototype plant that has been tested over an appropriate range of normal, transient, and accident conditions, the following requirements must be met for a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions:"

(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;

(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;

(3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

Consistent with these requirements, the passive-plant vendor, Westinghouse, has developed and performed design certification tests of sufficient scope, including both separate-effects and integral-systems experiments, to provide data with which to assess the computer codes used to evaluate plant behavior over the range of conditions described in item 3 above.

To satisfy the requirements of 10 CFR 52.47(b)(2)(i)(A), Westinghouse has developed test programs to investigate the passive containment safety systems. These programs were designed to include both component and phenomenological (separate-effects) tests and integral-systems tests. The test programs are discussed in Section 21.6.5.6 of this report.

21.6.5.4 WGOTHIC Code Description and Assessment

In 1991, Westinghouse purchased the GOTHIC computer program, Version 3.4c, from Numerical Applications, Inc., Richland, Washington (NAI). NAI had developed the program under Electric Power Research Institute (EPRI) and international utility sponsorship. GOTHIC is considered to be a realistic containment code, and has best-estimate heat and mass transfer model options. However, Westinghouse decided to use GOTHIC in a manner consistent with the guidance in SRP 6.2.1, which is based on conservative containment pressure calculation and input models. GOTHIC did not have models which addressed the special features of AP600, such as the passive containment cooling system (PCS). Westinghouse therefore added models specific to the AP600 PCS to the EPRI version of GOTHIC for AP600 design basis analysis (DBA).

Westinghouse modified GOTHIC by adding a special component, referred to as a "clime," to model the energy removal by the PCS. Climes use new correlations and models for

- mechanistic convective heat and mass transfer
- liquid film tracking
- one-dimensional wall heat conduction
- wall-to-wall radiant heat transfer

The modified GOTHIC code is called WGOTHIC and is documented in Westinghouse topical reports, WCAP-14382 and WCAP-14407. In addition, Westinghouse developed an evaluation model (EM) approach which bounds or conservatively treats many of the code and model uncertainties in the DBA scenarios.

The historical development of the GOTHIC code, with its connections to other previous well-known computer codes, is displayed in Figure 21.6-23. The NAI version of the GOTHIC code is primarily based upon the COBRA-NC code (NUREG/CR-3262, "COBRA-NC: A Thermal-Hydraulic Code for Transient Analysis of Nuclear Reactor Components) and its features have since expanded beyond this ancestor. This figure indicates that the GOTHIC code development is the result of 20 years of continued software development along with the associated experiences, verifications, and validations.

The conclusions of an EPRI-sponsored independent review panel have been published in the GOTHIC Design Review Final Report which was provided to the staff by Westinghouse letter NTD-NRC-95-4462, dated May 15, 1995. The review panel found GOTHIC 4.0 adequate for containment evaluation because it offered the potential for more accurate and mechanistically based analyses then other available containment analysis codes. However, the panel recommended that nodal and junction methodology and the range of the verification database be justified for each intended application. This is particularly true for the first-of-a-kind situation of the AP600 PCS. To address the EPRI peer review concerns, Westinghouse purchased

GOTHIC 4.0 in 1996, and installed the AP600-related modifications. GOTHIC 4.0 is considered to be the basis for WGOTHIC 4.2.

Figure 21.6-24 displays the early Westinghouse in-house development of WGOTHIC which incorporated the clime PCS heat and mass transfer models. As a result of sensitivity studies performed as part of the WGOTHIC licensing effort, Westinghouse discovered and corrected several errors. Earlier WGOTHIC versions had been numbered 1.0, 1.1, and 1.2. Westinghouse chose to refer to the base WGOTHIC version as WGOTHIC 4.0 and the revised version as WGOTHIC 4.1, to establish a numbering system consistent with its GOTHIC parent. Version 4.1 incorporated the peer-reviewed and quality-assured changes to NAI's GOTHIC Version 4.0. Westinghouse submitted WCAP-14967 to demonstrate that the conclusions reached from verification and sensitivity studies performed using previous versions of WGOTHIC were applicable up to WGOTHIC Version 4.1. WGOTHIC 4.2 as applied to AP600 in WCAP-14407 included additional refinements and error corrections to the clime methodology. All versions of WGOTHIC were developed in accordance with the Westinghouse 10 CFR Part 50, Appendix B quality assurance program. Unless otherwise stated, this review applies to WGOTHIC version 4.2.

21.6.5.4.1 WGOTHIC Code Overview

WGOTHIC provides a two-component, three field representation of multiphase flows in multiply connected compartments of nuclear containments. WGOTHIC solves mass, momentum, and energy balances for four separate phases: steam, liquid, drops, and ice (not applicable here). The vapor phase can be a mixture of steam and non-condensable gases, and a separate mass balance is solved for each component of the mixture. The phase balance equations are coupled with mechanistic models for interface mass, energy, and momentum transfer, covering the entire flow regime from bubbly flow to film and drop flow, as well as single-phase flows (described in the following subsections). Thermal non-equilibrium effects between the phases and unequal phase velocities can be accounted for by the interfacial models.

WGOTHIC can be used for three-dimensional analysis of the thermal-hydraulic behavior of containment atmospheres and structures. Containment compartments can be modeled using a one-, two-, or three-dimensional rectangular grid (referred to as the distributed-parameter approach) or the lumped-parameter approach, or a combination thereof. For long-term containment analysis finely noded multidimensional models are deemed impractical. Lumped-parameter analysis is traditional for these problems, and Westinghouse selected this approach for its WGOTHIC evaluation model (EM) for the AP600 in WCAP-14382. The lumped-parameter volumes are connected by junctions which use a one-dimensional model for flow between compartments. WGOTHIC's models include treatment of the multidimensional momentum transport terms with an optional one-parameter turbulence model for turbulent shear, mass, and energy diffusion. However, these features are neglected when the lumped-parameter nodalization is used.

GOTHIC thermal conductors model heat transfer surfaces in the containment. In addition, the clime model, a special model developed by Westinghouse, is used to simulate the PCS. Wall heat transfer correlations are incorporated for a wide range of containment conditions, including condensation heat transfer in the presence of non-condensable gases.

WGOTHIC's overall features, as well as those of other well-known U.S. containment code packages, are listed in Table 21.6.5.4-1.

21.6.5.4.1.1 Conservation Equations

The conservation equations solved by WGOTHIC are discussed in this section. These equations are formulated for a fixed volume in space bounded by surface areas. The volume may represent the entire containment or any part of its total volume. The GOTHIC (and WGOTHIC) conservation equations are based on the following major assumptions:

(1) Compressible flow for all fluid phases.

(2) Separate mass conservation equations are solved for each fluid phase (liquid, vapor, drop), gas component, and ice phase.

(3) Separate energy conservation equations are solved for each fluid phase (liquid, vapor, drop); however, viscous dissipation and kinetic energy are neglected.

(4) The phases are at an equal pressure. Mass and energy transfer rates across phase interfaces are obtained from a set of constitutive relations.

(5) The normal component of the viscous stress is ignored.

(6) Prandtl mixing length theory is used to derive Reynolds stresses and thermal and mass diffusion. Thermal diffusion by conduction and turbulence is accounted for. Mass diffusion by turbulence is also accounted for; however, molecular mass diffusion is neglected.

(7) Turbulent and viscous stresses are only applied to the continuous phase (e.g. vapor in drop flow, liquid in bubbly flows). Turbulent stress due to interfacial interactions is neglected.

A number of assumptions were adopted by the GOTHIC development team for the convenience of the finite-volume numerical method, used in the distributed-parameter approach (one-, two-, and three-dimensional subdivision of control volumes). The WGOTHIC AP600 evaluation model (EM) is based on the lumped-parameter approach and does not use a distributed-parameter approach; therefore the following assumptions were not examined during the WGOTHIC review:

(1) The momentum equations are strictly valid only in the context of rectangular grid subdivisions.

(2) The full three-dimensional form of the momentum equation is solved for each phase, including momentum transport and viscous and turbulent stresses.

(3) When WGOTHIC lumped-parameter control volumes are connected to distributed-parameter control volumes, all conservation equation forms are reduced and momentum transport is completely eliminated in the distributed-parameter control volumes.

Application of WGOTHIC in the lumped-parameter mode introduces numerous simplifications which are discussed in the following sections.

21.6.5.4.1.1.1 Mass Conservation

Mass conservation equations are solved for four flow fields (called "phases" in the context of GOTHIC or WGOTHIC): steam, liquid, drops, and ice (not applicable for AP600), and for each of the preselected non-condensable gas components. The steam/air mixture, termed the vapor phase, can take the form of bubbles or a continuous vapor region. The liquid phase can exist as pools, films, or liquid drops. The drop phase results from three different sources: (1) break discharges, (2) de-entrainment from condensate films and (3) spray nozzles.

The integral formulation carries the common storage, convective, and diffusion terms, plus terms characterizing contributions from boundary, interface, and equipment sources. Separate mass conservation equations are used for vapor, liquid, drops, and ice. The vapor phase may consist of steam plus a number of non-condensable gases (i.e., it may have several components). The mass conservation equations for the liquid, drop, and ice phases do not account for mass diffusion.

For the lumped-parameter approach, where one computational cell is used per volume, the mass balance is maintained for a network of lumped-parameter nodes and junctions. However, the simplifying assumptions result in a reduced mass conservation equation where junctions constitute the only flow connections to a lumped-parameter node. The integrals for the convective terms are replaced by sums over all the junction connections. This introduces a set of junctions connected to the volume with respective junction areas. Junction velocities are not vector quantities in the lumped-parameter approach.

21.6.5.4.1.1.2 Energy Conservation

Energy conservation equations are solved for three fluid phases (vapor, liquid, and drops) and are considered for solid thermal conductors. In WGOTHIC the fluid energy equation is formulated in terms of enthalpy rather than internal energy. Its integral formulation carries seven terms accounting for storage, convection and flow work, thermal diffusion, mass diffusion, and contributions from boundary, interface, and equipment sources. Neglected are kinetic energy, viscous dissipation, and other energy forms not explicitly represented in the integral form. All components of the vapor phase (e.g., steam and non-condensables), are assumed to be at the same temperature.

For the lumped-parameter approach, the energy balance is maintained for a network of lumped-parameter nodes and junctions. However, the modeling assumptions result in a simplification of no computed velocity gradients and, consequently, the turbulent diffusion is set to zero. Therefore, mass and thermal diffusion terms across junctions are not considered in the lumped-parameter mass and energy conservation equations. The integrals in the convective terms are replaced by summations over all junction connections. In addition, the boundary source and interface terms are defined differently than in a full three-dimensional formulation.

Energy Conservation for Solid Conductors

Thermal conductors are used in WGOTHIC to model heat sinks such as concrete walls and floors, structural steel internals, stairwells, gratings, and primary system components such as pipes, vessels, and valves. WGOTHIC conductor geometries are limited to flat plates, solid cylinders, and hollow tubes. Therefore, only one-dimensional heat conduction models are implemented for flat plate and cylindrical geometries. The energy equation for solid conductors accounts for storage, thermal diffusion, and boundary sources. The material properties can be functions of space, the local temperature, and the local volumetric heat source. Thermal conductors may consist of multiple layers of varying materials with the assumption that the density of each type of material is constant. Other thermal surface boundary condition options include:

(a) Transient (or constant) surface temperature.

(b) Transient (or constant) surface heat flux.

(c) Convection and condensation heat transfer described by a total heat flux to or from a wall to the vapor and liquid phases, partitioned into respective heat transfer rates to the specific phases. The heat rates depend on the user-selected heat transfer options and the appropriate fluid and wall conditions. The available options are discussed in Section 21.6.5.4.1.2.1 and 21.6.5.4.1.2.2 of this report.

The WGOTHIC (and GOTHIC) energy conservation equations do not provide for a volumetric source term. The surface areas of thermal conductors may be exposed to one or more fluid phases.

21.6.5.4.1.1.3 Momentum Conservation

WGOTHIC has momentum conservation equations for three phases: vapor, liquid, and drops. The integral form of the momentum conservation equation contains the following seven terms: storage, convection, surface stress, and body, boundary, interface, and equipment sources. The general formulation, using stress tensor notation, includes contributions from the static pressure, viscous, and Reynolds stress terms. Because of the natural complexity of the momentum conservation equation, terms containing the momentum source per unit wall area (momentum source per unit interfacial area and momentum source from equipment) are especially difficult to specify for the three phases accounted for in WGOTHIC. Whereas all three phases are considered to move with different specific phase velocities, all components of the vapor phase (e.g., steam and non-condensables) are assumed to have the same velocity.

Of all of the conservation equations, the momentum conservation equation is the most affected by the change from the distributed-parameter to the lumped-parameter approach. As there is no flow field in a lumped node, no momentum equation is solved for the lumped-parameter volumes in the network. Momentum is conserved by the solution of the momentum equation in the junctions connecting the volumes constituting the lumped-parameter model for the given containment geometry under consideration.

Momentum Conservation in Junctions

One or more junctions may hydraulically connect two computational cells. They may represent doorways, pipe and cable penetrations, ductwork, vents, and stairways. Junctions are characterized by their cross-sectional area, length, and, for both ends, the lowest elevation of the volume interface and the interface height. The junction momentum equations for the three phases (vapor, liquid, and drops) are solved for each junction. Except for viscous and turbulent shear, the junction momentum equations are consistent with the momentum equations used for the sub-volume face velocities for one-, two-, and three- dimensional meshes in the distributed-parameter model. Mass residing in the junctions is not accounted for; that is, mass leaving one lumped node through a junction is instantaneously moved into the connecting volume for purposes of mass balancing. An approximate junction fluid mass is computed for use in calculating the inertia of the junction flow. Junctions allow a pool in one volume to overflow into a connecting volume with the associated displacement of the vapor phase.

The WGOTHIC junction momentum equations simulate the specific phase momentum transport between computational mesh cells. The junction momentum equation includes terms for the following: inertia, pressure gradient and local gravity head, junction gravity head, equipment source, momentum fluxes, wall shear, and interfacial drag. All of these terms require the specification of coefficients by correlations and/or input data selection. The phase density in the junction is assumed to be the arithmetic average of the phase densities of the connected volumes. Junction momentum flux terms (cited above) are not used when connecting to a lumped-parameter node, because the lumped node lacks velocity information.

To determine the donor fractions for flow out of a computational cell through a junction, the fluid in this cell is assumed to be in the form of a pool with both a vapor phase and drops above the pool surface. This model is used to calculate an effective pool height and pool vapor fraction from the cell geometric parameters and the current cell phase volume fractions. The pool vapor fraction is needed to correctly calculate the pool surface elevation height in the presence of bubbles in the pool, which raise the liquid pool height. This is estimated by using the Yeh correlation, which uses the vapor and liquid densities and the vertical vapor and liquid velocities beneath the pool surface. For the lumped-parameter approach two-phase pool dynamics are neglected.

Because of the special pool model features in WGOTHIC (and GOTHIC), two sets of phase fractions are relevant to junctions, namely, the donor cell phase and junction phase fractions. The donor cell phase fractions determine the mass flows through junctions, and the junction phase fractions control junction inertia, gravitational head, and pressure forces and are themselves dependent on the donor cell fractions.

The static pressure is determined at the vertical center of each volume. To account for buoyancy effects in the lumped-parameter approach, the variation in the static pressure within each volume must be estimated to provide the pressure at the respective junction ends. This is achieved by splitting the junction gravitational head into two contributions: the junction head (head across junction) and the local head (head from the cell center to the junction end elevation). The specific phase junction head is the product of the junction phase density and the difference in the junction end elevations.

In summary, for lumped-parameter volumes, mass and energy are conserved. However, the momentum conservation equations are neglected. Thermal-hydraulic conditions are represented by cell-averaged parameters. This limits the applicability of the lumped-parameter approach for modeling phenomena such as

- thermal stratification
- concentration gradients
- velocity profiles
- mixing flow patterns

21.6.5.4.1.2 Closure Relationships and Correlations

To solve the equations for mass, energy and momentum conservation for the three fluid fields, closure relationships, such as interfacial source terms and heat and mass transfer, are needed to describe interfacial interactions. These closure relationships are summarized and reviewed in Sections 21.6.5.4.1.2.1 and 21.6.5.4.1.2.2 of this report.

Other closure relationships are required to model the heat and mass transfer between the three fluid fields and solid structures inside the containment. These encompass the wall source terms, which include convective heat transfer, condensation and boiling (evaporation) at the structural surfaces, friction at walls, and orifice drag in junctions. Correlations are used to compute these wall source terms. These correlations are presented and discussed in Section 21.6.5.4.1.2.2 of this report.

21.6.5.4.1.2.1 Interfacial Source Terms

The WGOTHIC conservation equations do not limit the thermodynamic and mechanical characteristics of the three fields. Rather, interface source terms are specified for the computation of mass, energy, and momentum balances at the interfaces, assuming that none of the quantities is stored at the interface. WGOTHIC accounts for five field interface combinations: liquid/vapor, drops/vapor, ice/vapor (not applicable to AP600), ice/liquid (not applicable to AP600), and drops/liquid. Interfacial transport processes are induced by heat transfer due to phase change and by mechanical interactions generating interfacial mass and momentum exchanges. Correlations and/or input values have to be provided for the respective interfacial heat transfer coefficient and the associated areas of the vapor/liquid and vapor/drop interfaces.

For the conservation equations, different interface models are applied, based on the flow regime and whether the subdivided (distributed-parameter) or lumped-parameter approach is used. These interface models are described below.

Flow Regime Maps

The selection of the proper exchange coefficients (heat transfer, drag) and the related interfacial areas primarily depends upon the geometry of the phase distributions under consideration. Several flow regime maps have been proposed and implemented into current nuclear safety computer codes (RELAP5, TRAC). Many of those apply only to the special technical or safety issue for which they were developed and cannot be applied to the whole spectrum of anticipated

two-phase problems. Recognizing that an accurate prediction of flow regimes is beyond the current state of knowledge and current computational capabilities and that the selection must agree with the nodal representation of the flow field by the computational control volumes, WGOTHIC contains simple, widely applicable flow regime maps. Although the flow regime map for vertical flows is somewhat sophisticated, a more simplified selection scheme is implemented for horizontal flows. The flow regime maps incorporated into GOTHIC (and WGOTHIC) are based on previous experience over a wide spectrum of applications with COBRA-NC (NUREG/CR-3262) and COBRA/TRAC (NUREG/CR-3046).

The prevailing flow characteristics in a computational cell must be determined before the correlations for interfacial transfer can be applied. During a calculation, this information is available only for a given computational cell and its immediate neighbors. The flow regime selection is based on liquid and volume fractions. This approach requires that all volume fractions be present and that none be exactly zero, since the numerical solution procedure solves the full set of phase equations for each computational control volume for each time step. This requirement imposes the need to specify lower limits for volume fractions (see Section 12.9 of NAI's "GOTHIC Containment Analysis Package Technical Manual, Version 4.0").

Closure Relationships for Lumped-Parameter Approach

The interfacial heat transfer coefficients for the lumped-parameter approach account for the presence of pools and drops in lumped-parameter control volumes. In WGOTHIC a pool configuration is assumed to exist in a control volume if

(a) the ceiling of the control volume is closed, and

(b) the liquid volume fraction is larger than 0.2

or

(a) the ratio of liquid volume fractions of two axial neighboring control volumes is larger than 1.5, and

(b) the liquid volume fraction of the lower control volume is larger than 0.2

If the liquid volume fraction is less than 0.1, the liquid is assumed to be distributed as a film on all surface areas exposed to the atmosphere in the control volume, rather than to form a pool. For a liquid film the interfacial area is assumed to be equal to the wetted surface area. These criteria are based on obtaining reasonable agreement with data for different modeling scenarios.

The heat transfer to the pool surface from a superheated vapor is determined from the maximum of heat transfer coefficients for turbulent natural convection and forced convection based on the Reynold's analogy as applied to a flat plate and a user-specified minimum value. Increased heat transfer due to evaporation at the pool surface is accounted for by an evaporation factor, β, which considers the molecular weights of the steam and the non-condensable gas.

Drop Entrainment and Deposition

The WGOTHIC code has provisions for modeling drop entrainment and deposition phenomena. WGOTHIC includes a model (referred to as the "fog" model) which generates drops whenever the air/steam temperature is lower than the saturation temperature. The fog model is not used in the WGOTHIC EM analysis for AP600, since Westinghouse assumes that a large fraction of the blowdown liquid flow is entrained into the containment atmosphere as drops which permanently remain in the atmosphere.

Combined Interface Source Terms

The interfacial source terms for the energy conservation equations are specified with the assumption that, for any mass transfer which results from interfacial heat transfer, the mass leaves the respective phase at the phase bulk temperature and enters the other phase at the saturation temperature. The latent heat is added to or subtracted from the phase generating the phase change.

The WGOTHIC formulation computes heat transferred to the interface from the vapor and liquid phases within the context of the standard Newton's cooling law (vapor and liquid heat transfer coefficients, vapor and liquid bulk temperatures, and the interface temperature). Any excess heat at the interface is applied toward converting liquid to steam. This is achieved by calculating the rate of evaporation (or condensation) from the liquid and vapor heat transfer rates and the heat of vaporization (or condensation) with a mass transfer coefficient, an interface area, and the ratio of the difference of steam concentration at the interface and the steam concentration in the bulk to the difference between saturation and an air/steam concentration at the interface. The latter is determined by assuming saturation at the interface temperature. In this calculational step, a total set of five equations is solved iteratively for the unknown interface temperature, steam concentration, liquid and vapor side heat transfer rates, and the mass transfer rates.

The heat transfer correlations used on the vapor and liquid sides of the interface in WGOTHIC are maxima of forced and free convection heat transfer coefficients. For free convection the correlation is the same as used in CONTEMPT for turbulent free convection. The heat transfer correlations are not adjusted to account for high mass transfer rates; however, a multiplier of 1.2 is used on the vapor side to enhance agreement with experimental data.

Pool boiling initiates when the pool temperature is higher than the saturation temperature associated with the total pressure. High vaporization rates are enforced by high values for the liquid heat transfer rate and the mass transfer coefficient.

As a result of the full two-phase model in WGOTHIC, with separate conservation equations for liquid, vapor and drops, the code does not model the "temperature flash" and "pressure" processes applied in CONTEMPT for the blowdown mass and energy sources. Rather, typically the blowdown sources in WGOTHIC are specified by prescribing the enthalpy and the pressure of the injected fluid into the break subcompartment together with an average drop diameter. Steam and drops are assumed to be in thermal equilibrium at the source pressure.

For completeness, WGOTHIC can simulate both "pressure" and "temperature flash" conditions by setting the source pressure to the blowdown compartment total pressure or steam partial

pressure, respectively, together with an input specification that the liquid fraction of the blowdown fluid is deposited directly to the pool rather than as drops into the compartment atmosphere. Table 21.6.5.4-2(a) lists the interfacial heat and mass transfer models in WGOTHIC and CONTEMPT, and Table 21.6.5.4-2(b) lists the condensation and evaporation correlation used in WGOTHIC and CONTEMPT.

21.6.5.4.1.2.2 Heat Transfer Models

In this section the closure relationships and correlations in WGOTHIC are summarized and evaluated. Single- and two-phase heat transfer, condensation, and evaporation models define the wall source terms in the energy conservation equations for the vapor and the liquid phases, respectively. The energy source term accounts for heat transfer between the atmosphere and the heat sink conductors through convection, radiation, condensation, and evaporation. It also accounts for heat transfer from pools to structures. WGOTHIC provides the code user with options to specify the relevant heat transfer mode for each thermal conductor in the model under consideration:

- turbulent natural convection
- direct condensation and evaporation
- tagami blowdown (not used for AP600)
- specified revaporization
- wet wall model
- specified time-dependent heat transfer coefficient
- built-in single-phase and two-phase heat transfer package

These seven options are common to both GOTHIC 4.0 and WGOTHIC 4.2. In addition, WGOTHIC contains the correlations that were implemented by Westinghouse for the clime model used to predict the PCS behavior. The approach used in WGOTHIC for interfacial heat and mass transfer is similar to the one used in CONTEMPT. However, because of GOTHIC's two-phase modeling approach, one important difference between CONTEMPT and WGOTHIC is that in WGOTHIC the interface temperature is calculated from first principles, rather than assumed (as it is by CONTEMPT) to be equal to the saturation temperature. Therefore, WGOTHIC determines sensible heat transfer between the liquid and the pool surface and to the vapor phase even when the vapor is saturated. This heat transfer mechanism is not possible with some general containment analysis codes, such as CONTEMPT. In addition, a similar feature of the drop field model allows calculation of heat and mass transfer to or from the drops.

The source terms associated with each of these options are summarized below with an emphasis on the heat transfer modes relevant to the AP600 containment.

Turbulent Natural Convection

The McAdams correlation (McAdams, W.H., Heat Transmission, 3rd Edition, 1954) is used for turbulent natural convection from vertical plates and large cylinders. This correlation uses the liquid or vapor thermal conductivity, the viscosity, fluid density, and the Prandtl number. The length dependence is removed and the heat transfer coefficient depends only on local properties. The heat transfer coefficient actually used is the maximum of the value determined by the correlation or a user-supplied minimum value. For the latter, the code's default value is zero (no heat transfer). The documentation quotes the validity range of the McAdams

correlation to be $10^9 < GrPr < 10^{12}$ (see Table 21.6.5.6.1). The coefficient of linear thermal expansion, β, which enters the Grashof number, is equal to the reciprocal of the vapor temperature, assuming ideal gas behavior. The characteristic length cancels out of the heat transfer correlation for turbulent natural convection.

The heat transfer correlation recommended for turbulent natural convection from horizontal surfaces is the same as for vertical surfaces, except that the coefficient 0.13 changes to 0.14. However GOTHIC, and thus WGOTHIC (except for the climes), does not differentiate between the orientation of the thermal conductor surfaces. Both codes use the correlation for vertical surfaces regardless of orientation with the understanding that heat transfer from horizontal surfaces would be slightly underpredicted if this correlation were used inside the containment.

Direct condensation and evaporation

The total heat transfer from the thermal conductor surface for direct condensation includes (1) latent heat released by condensation, (2) convective heat flux between superheated vapor and saturated film, and (3) radiant heat flux from the surface to the vapor.

The condensation heat transfer is determined from the surface area for condensation, the condensation heat transfer coefficient, a multiplier for time-dependent function, and the minimum of zero and the difference between the surface and the condensate temperature, T_{sat} (the saturation temperature at steam partial pressure).

If the wall is superheated relative to the local steam, the condensation heat transfer is set to zero.

The following options are available for direct-condensation heat transfer to internal heat sinks in WGOTHIC 4.2:

- Uchida
- Gido/Koestel (not used for AP600)
- Maximum of Uchida and Gido/Koestel (not used for AP600)

The Uchida-correlation implemented in WGOTHIC (in WCAP-14407) is actually a fit to the published data as a function of the ratio of steam density to the density of the non-condensable gas mixture raised to the power of 0.8. This presentation was already contained in COBRA-NC (NUREG/CR-3262) and CONTEMPT (NUREG/CR-0255) and used for their validation. The fit has upper and lower limit values of 594 and 4.3 kJ/hr-m^2-°C (278 and 2 BTU/hr-ft^2-°F), respectively.

The review of Uchida's test setup has led to the conclusion that the data includes some effect, at least in the free natural convection regime, of velocity on the heat and mass transfer. However, the velocity effect is not explicitly available. The Uchida correlation's independence of any velocity effect makes it suitable for incorporation in lumped-parameter models (e.g., CONTEMPT), because the computational control volume atmospheres are assumed to be at rest (i.e. stagnant).

Any model improvement to account for velocity effects leads to the additional need to define an approved representative control volume velocity, which is not computed in the lumped-parameter model. For distributed-parameter control volumes, the control volume velocity is obtained by averaging the computed control volume face velocities. The volume velocity for lumped-parameter control volumes must include the junctions, as no velocities are predicted for the control volumes. Since the orientation of junctions is unspecified in the lumped-parameter model, the control volume velocities can be only roughly approximated because of the limited information available. For the vertical component of the control volume velocity, the summation on junction flows includes only those that are connected to the control volume vertically. A multiplier of either +1 or -1 is used to make the sign of the junction flow consistent with the global coordinate system in deriving the control volume velocity. For the lumped-parameter model, which does not specify the junction direction through the momentum transport input, it is always assumed that the junction is horizontally connected to the computational control volume. In general, the control volume-centered velocity for any phase is computed from the square root of the sum of the squares of the vertical and the transverse velocities (as discussed above).

Once the total condensation heat transfer rate has been computed by one of the available options, the wall condensation rate is determined. The direct condensation option includes radiation heat transfer with the assumption of grey gas and grey surrounding walls. The surface emissivity is assumed to be 0.65 for dry walls, and a value of 0.96 is used for wet surfaces (surface temperature below saturation temperature). The gas emissivity is also accounted for and complex geometries are taken into account by virtue of an effective beam length.

The convective contribution to the total heat transfer applies the convective heat transfer coefficient obtained from the built-in heat transfer package. With the individual contributions summarized above, the total wall mass and energy source terms are completely described for the direct condensation model.

Specific Revaporization

Regulatory guidelines (NUREG-0588) allow only a specified fraction of the condensation at the wall to be revaporized by the superheated atmosphere. For licensing analysis, a value of 0.08 (per NUREG-0588) is acceptable. Therefore, WGOTHIC has an option to specify a revaporization factor between 0 and 1. For non-zero values, the normal interfacial heat and the related mass transfer for superheated vapor are set to zero. They are replaced by augmented wall source terms which include revaporization.

Wet-Wall Model

This option is not recommended for the lumped-parameter model approach since its underlying assumptions are inconsistent with the pool geometry assumed for lumped-parameter control volumes.

Specified Values

The user can specify wall-to-vapor heat transfer coefficient as a function of time by providing a constant value and a time-dependent multiplier. The user can also specify the surface temperature, the surface heat flux as a function of time, or a vapor convection heat transfer

coefficient and an ambient temperature. In these cases no direct heat transfer to the fluid is computed and the wall source terms are zero.

Built-in Heat Transfer Package

GOTHIC and WGOTHIC contain a complete set of heat transfer correlations originating from COBRA-NC and COBRA-TRAC codes, which cover the entire boiling curves. However, this heat transfer correlation package is not applicable for the lumped-parameter approach. The Dittus-Boelter correlation for turbulent forced convection for either vapor or liquid properties is implemented. For vapor conditions the maximum of this and the previously discussed turbulent natural convection (McAdams) correlation is determined. For single-phase liquid the maximum of the Dittus-Boelter correlation and heat conduction through a liquid film is determined.

Momentum Source Terms

Momentum source terms account for drag effects due to wall friction and local losses due to orifices and obstructions. The total drag force is the sum of both contributions and is computed for each phase — vapor, liquid, and drop. The drag coefficient is an input quantity for each flow connection and the drag force is assumed to be proportional to the area fraction of the respective phase. Wall friction drag forces are determined only for the continuous vapor and liquid phases.

The friction factor used to calculate the wall friction drag force is specified for hydraulically smooth pipe conditions and is determined from the maximum of the analytical result for laminar flows (64/Re) and the turbulent-flow friction factor that is obtained from a curve fit of the well-known Moody data. The Reynolds number for each phase is determined by the simple approach of assuming that the flow consists of only that phase. However, the GOTHIC documentation points out that this approach, coupled with the interfacial drag models results in excellent agreement with data for two-phase pressure drop over the entire range of vapor volume fractions. No proof or reference is provided in this context to show comparisons between data and predictions. However, for the case of two-phase annular flow, the GOTHIC documentation analytically demonstrates that the selected approach corresponds closely with common empirical relationships for the two-phase multiplier for annular flow.

Floor Drag

This is an extra flow resistance applied to the liquid flow out of a computational control volume through a junction when the liquid level in the control volume is near the junction elevation. In this situation, the junction flow changes from a liquid/vapor mixture to an all-vapor flow when the liquid level drops below the junction elevation. This causes large numerical oscillations in the liquid flow through the junction and the additional resistance is used to stabilize the solution. It is called floor drag because the current model has been developed for liquid flow across the floor into a center drain. This numerical stability enhancement feature is applied to any junction where the liquid level is approaching the junction level, whether or not the junction elevation is at the floor level.

21.6.5.4.1.3 Numerical Methods

WGOTHIC uses a semi-implicit numerical solution scheme to solve the coupled set of conservation equations for mass, momentum, and energy in the fluid cells, together with the heat conduction equations for the thermal conductors. The complete set of governing equations is solved simultaneously for the state and velocity variables for each new time value. Some of the terms in the equation set are non-linear in the new time variables. Rather than iterating on the equation set to accommodate the non-linear terms, an approximate solution is obtained by applying the one-step Newton method. The resultant matrix equation can be solved by direct solution (Gaussian elimination/back substitution) methods for small problems such as those generated for most lumped-parameter models.

Solution of the Momentum Equations

The semi-implicit momentum equations for the three phases in one momentum control volume can be written in the form of three equations with coefficients A, B and C. The A coefficients include the temporal terms as well as the wall and interfacial drag coefficients. The B coefficients are the coefficients on the pressure gradient, and the coefficients C_{ij} (i = 1, 2, 3; j = 1, 2, 3) represent all the explicit terms on the flow rates of vapor, liquid and drop phases, respectively. From these flow rates, approximate velocities can be computed.

Linearization of the Mass and Energy Equations

The mass and energy equations can be solved once the approximate velocities have been obtained from the momentum equations. The mass and energy conservation equations will generally not be satisfied when the newly computed velocities are used to calculate the convective terms. Rather, residuals for mass and energy will result from the equations because of the new velocities and changes in some of the explicit terms, such as phase changes at the wall. The set of conservation equations is satisfied when all residuals for all computational control volumes in the mesh or network are simultaneously equal to zero. This can be approximately achieved by using a one-step Newton-Raphson method which considers the variation of each independent variable to bring the residuals close to zero. The variations are obtained by simultaneously solving matrix equations for all cells.

Solution of the Pressure Matrix

The solution of the pressure matrix is the center of the solution procedure. The pressure equations can be written for simultaneous solution as a matrix equation with N equations for N unknowns (N is the number of computational control volumes). Generally, the coefficient matrix is sparse and shows a banded structure (all non-zero elements occupy a certain number of columns on either side of the diagonal). The solution and storage approaches take advantage of both matrix properties.

Unfolding of Primary and Secondary Variables

When the pressure variation in each computational control volume has been obtained, the variations for the other variables are determined by using the stored reduced control volume equation. The new values for each primary variable (pressure – P, vapor fraction – α_v, droplet fraction – α_d, vapor, liquid, and droplet specific enthalpies – $\alpha_v h_v$, – $\alpha_l h_l$,– $\alpha_d h_d$, and vapor

partial pressure – $\alpha_v P_{vg}$), is obtained by adding the variation to the old values. The secondary variables liquid fraction (α_l), vapor pressure (P_{vg}), vapor, liquid and droplet enthalpies (h_v, h_l, h_d) and the new time step are calculated. The new time densities, ρ_v, ρ_l and ρ_d, are computed from the equations of state and the flows updated by the reduced momentum matrix equation. Similar formulations are used for horizontal velocities and the junction velocities.

Oscillating Flow Control

In certain situations with GOTHIC and WGOTHIC, because of the semi-implicit nature of the solution procedure, the approximate flows obtained from the momentum equation, as described above, are opposite in direction to the flows computed from the residuals of mass and energy at the end of the time step. This effect may be introduced by a number of explicitly evaluated terms. As a result of the discrepancies in the flow, more of a convected quantity may be taken out of a computational control volume than was there at the beginning of the time step. Body forces, especially buoyancy driven flows, are affected by this problem because they are explicitly evaluated. To minimize this problem, a check is made between the approximate flow and the one from the previous time step to ensure that none of the convected quantities exceeds the available quantity in the computational control volume. This precautionary logic is applied for the mass and energy conservation equations and generally maintains a stable solution for the period when the pressure gradient is inconsistent with the flow direction.

Time Step Control

The time step in GOTHIC and WGOTHIC is limited by the Courant condition in that the time step must be smaller than or equal to the ratio of the volume divided by the total volumetric flow out of a computational control volume. Compliance with this stability criterion is mandatory because of the explicit quantities in various transport terms.

Because of the explicitness of the body force terms, experience with the code over time made it necessary to limit the time step relative to the natural oscillation period of a gravity-driven system. The U-tube manometer problem is an excellent example. The natural period is given by

$$t_n = 2\pi\sqrt{L/g}$$

with L being the effective inertial length. The recommended time step limit is then $t \leq t_n/2$, which results in stable solutions for most cases. If a stable solution is not obtained, the time step size must be further reduced via input changes by the user.

Additional checks are employed in GOTHIC and WGOTHIC to ensure that variations in the new time variables are within reasonable limits compared to the old time values. This approach eliminates unphysical values, such as negative or greater-than-one phase volume fractions. For these and similar cases the time step size is automatically halved to ensure that the linearized equations are as close to being representative of the original set of non-linear equations as possible.

Variable Limits

There are two different types of variable limits in GOTHIC and WGOTHIC. This first type keeps specific variables within prespecified lower and upper bounds to ensure proper divisions and to ensure that physical parameters remain within reasonable limits.

The second type of limits relate to the model approach for control volumes and junctions and the way the vapor, liquid and drops are treated in GOTHIC and WGOTHIC. Since the numerical solution procedure requires that no phase is completely depleted, small amounts of each phase and non-condensable gas must be present in each computational control volume at all times (see Section 12.9 of NAI's "GOTHIC Containment Analysis Package Technical Manual, Version 4.0").

21.6.5.4.1.4 Technical Evaluation

21.6.5.4.1.4.1 Adequacy of the Conservation Equations

The major technical findings related to the WGOTHIC set of conservation equations are summarized below:

(1) The two-fluid, three-field set of conservation equations in WGOTHIC, together with the sets of interfacial and wall heat transfer terms, form a complete and comprehensive model.

(2) The separate sets of conservation equations for continuous liquid, continuous vapor, and droplet fields provide the code user with the ability to account for possible thermodynamic and mechanical non-equilibrium conditions between the fields.

(3) The documentation of the set of conservation equations in NAI's GOTHIC Technical Manual discusses the most general integral control volume formulation and also discussed the individual contributions of interfacial, wall, and heat flux terms. These discussions are primarily focused on features of the WGOTHIC model which are only available in a subdivided, distributed-parameter approach.

(4) The lumped-parameter approach is used in the AP600 Evaluation Model (WCAP-14407). In the lumped-parameter approach, only mass and energy conservation equations are solved for the control volumes; the momentum conservation is solved for the junctions connecting the computational control volumes.

(5) The GOTHIC documentation emphasizes the features of the subdivided, distributed-parameter approach. Limitations associated with the lumped-parameter approach are listed, but their consequences for modeling physical phenomena of interest to the AP600 containment are not consistently documented. Model assumptions inherent in the lumped-parameter approach are not fully articulated in the documentation of the set of conservation equations; rather they are spread throughout the multivolume GOTHIC manuals in terms of user guidance, modeling recommendations, and validation efforts.

(6) The lack of models for natural circulation, jet and plume entrainment and mixing and related stratification phenomena in finite geometries limits descriptions of these physical processes in the AP600 containment with the lumped-parameter EM approach.

(7) The test simulations and sensitivity studies in Westinghouse WCAPs -14407, -14382, and -14967 are sufficient to verify specific (separate effects) models and demonstrate the correct operation of the WGOTHIC code. However, as discussed in Section 21.6.5.5 of this report, there are no integral tests, at an appropriately scaled prototypic test facility, that can confirm the validity of the WGOTHIC code for the AP600 containment.

The shortcomings cited are common to all current containment analysis codes. The staff considers the WGOTHIC code (in the lumped-parameter mode) to present a more complete set of conservation equations than other lumped-parameter containment codes, such as CONTEMPT/LT, the currently accepted standard. In the lumped-parameter approach, the WGOTHIC junction model for pool and above-pool regions uses a fully integrated method of simulating buoyancy effects . The separate droplet field in control volumes and junctions offers an expanded code capability and user option which eliminates the need for liquid-carryover fractions and dropout and flashing models commonly used in similar codes. Within the above-cited limitations and constraints of the lumped-parameter approach, the WGOTHIC conservation equations are suitable for peak pressure studies of the AP600 containment when exercised by a knowledgeable user.

21.6.5.4.1.4.2 Technical Adequacy of Interfacial Source Terms

The major technical findings related to the WGOTHIC set of interfacial source terms are as follows:

(1) The GOTHIC documentation of the interfacial source terms is focused on the sophisticated features provided by the distributed-parameter model option, not on the options implemented in the AP600 EM. A number of these features, such as drop/liquid film interactions, entrainment, and deposition, do not apply for the lumped-parameter model option.

(2) Although some of the interfacial transport phenomena do not apply for the lumped-parameter model option in GOTHIC and WGOTHIC, the remaining interfacial source terms provide an acceptable mechanistic description of phase interactions.

(3) Because GOTHIC eliminates liquid film features at surfaces for the lumped-parameter model option and thus is unable to properly model the important PCS phenomena (condensate film at inside steel shell surface, coolant liquid film at outside steel shell surface), Westinghouse expanded the code by implementing the concept of climes into WGOTHIC. The clime model provides the interfacial source terms specific to the AP600 PCS.

(4) No ranges of applicability are listed for correlations and models used in the GOTHIC interfacial source term correlations and models. However, the required ranges for the interfacial source term models during and after the AP600 LOCA and MSLB are not very

different from DBA conditions in current-generation containments; which are the conditions against which GOTHIC has been well validated.

The AP600 EM does not use the sophisticated interfacial models present in WGOTHIC, favoring instead, a simple specification of the droplet mass released during LOCA events. No droplets are assumed to be released during MSLB events. Since many interfacial phenomena are not properly accounted for in the lumped-parameter approach, gravitational settling would be the only drop deposition mechanism. Westinghouse addressed this potential model deficiency by choosing drops of a sufficiently small drop diameter to remain suspended in the atmosphere for an "infinite" amount of time. The containment atmosphere is thus forced to remain saturated. Sensitivity analyses discussed in Section 5.8 of WCAP-14407 show that the EM assumptions regarding droplets are conservative. In particular, the peak pressure is reduced when no droplets are assumed to enter the atmosphere. Cases with 5 percent and 100 percent of the break flow liquid mass assumed to be in the form of droplets yield essentially identical results. While the staff believes that the modeling of droplets in the EM has a number of unphysical aspects, Westinghouse has demonstrated that the approach is conservative. The staff finds this approach acceptable for DBA pressure predictions.

21.6.5.4.2 Special Models for AP600 Analysis

The AP600 passive containment cooling system (PCS) provides post-accident containment heat removal. The PCS uses natural forces, such as gravity-driven flow, condensation, evaporation, and density-driven air circulation, to transfer heat from inside the containment to the surrounding environment, which is the ultimate heat sink for AP600.

21.6.5.4.2.1 Description of the PCS

The PCS (shown in Figure 21.6-22) consists of a water storage tank with a minimum useable capacity of 2,010 m^3 (531,000 gal) located above the containment, and three standpipes which discharge the water to a distribution bucket located over the center of the containment dome. The standpipes are configured to progressively decrease the water release rate. During the initial post-accident period, the flow rate is approximately 1,665.6 liters/min (440 gpm). After 3 hours, the flow is reduced to 454 liters/min (120 gpm) and after 30 hours the flow is reduced to 273 liters/min (72 gpm). The peak pressure for both the large-break LOCA and the MSLB occurs during the initial 3-hour period.

After the bucket is filled, the PCS flow spills over and spreads out onto the top of dome. Radial spreading vanes attached to the top of the dome keep the flow from agglomerating into one sector of the dome. The PCS flow is then redistributed into a circumferentially uniform pattern by two successive weirs located at the 6.7 m (22 ft) radius and at the 15.24 m (50 ft) radius. The AP600 steel shell (i.e., the dome and vertical sidewalls) are coated inside and out with an inorganic zinc coating to improve surface wettability and preclude the formation of fingers and rivulets.

The PCS water flows down the lower dome and sidewalls of the containment, cooling the containment by evaporating the water film. A fraction of the exterior shell is assumed to remain dry. Dry sections reject heat by radiative and convective heat transfer. However, these processes are only about 5 percent as effective in transferring heat as evaporation from the wet sections.

Heat transfer from the shell exterior is augmented by natural circulation air flow. Air enters the shield building which surrounds the AP600 containment through inlets near the top of the shield building structure. The air then flows through a U-shaped passage and exits through an elevated chimney atop the shield building. The U-shaped passage is formed by a baffle piece which separates the annulus between the shield building and the containment shell into downcomer and riser passages. The riser and downcomer are bounded by the containment shell and steel baffle, and the baffle and the shield building wall, respectively, form an integrated path for natural circulation. Heat transfer from the shell to the riser air provides the driving head for natural circulation. Unrecoverable flow resistances are based on measurements from the PCS air flow path pressure drop test, a 1/6th-scale, 14.32° wedge model of the PCS downcomer, riser, and chimney documented in WCAP-13328. The test resulted in some changes to the final design of the air annulus flow path incorporated into the AP600. The resulting data was extended, by a factor of 1.5, to account for the higher expected Reynolds number in the AP600. The form loss was later increased an additional 30 percent to account for the design of the baffle turning vane in the AP600.

In response to staff concerns with the upper annulus drains, a design change was incorporated to move the drains about 1 foot above the upper annulus drain floor. This effectively shortens the downcomer-to-riser turning region from about 1.83 m (6 ft) to 1.52 m (5 ft), once sufficient PCS water fills the region to the new drain elevation. In response to RAI 720.440F (Westinghouse letter DCP/NRC1232, dated January 28, 1998), Westinghouse determined that the geometry changes do not impact the WGOTHIC modeling of the AP600 for design-basis accident analyses and the data from the test (as modified) are still valid for the AP600 evaluation model. The existence of this pool of water will not significantly affect the already low velocity in the air entrance region. The possible effect of the cold water pool on the air density was shown to have a negligible effect on the buoyancy-driven air flow.

These PCS air flow path characteristics were reviewed by the staff and, in view of the above, found to be acceptable for WGOTHIC EM use.

21.6.5.4.2.2 PCS Flow Characterization

After a LOCA or MSLB event, a significant time will elapse following the high-2 pressure signal to the PCS discharge valves as the PCS flow fills the piping, fills the water distribution bucket, and fills and overflows the two distribution weirs, to finally establish total coverage from the second weir to the annulus drain elevation. The shell surface temperature continues to increase during the initial blowdown period while full PCS coverage is being established. PCS flow filling times were measured in the full-scale unheated water distribution tests (WDT) at a flow rate equivalent to 832.7 liters/min (220 gpm). The WDTs are documented in WCAPs -13353, -13296, and -13960. The current maximum value is now set to 1665.4 liters/min (440 gpm) to address staff concerns with the vessel wall temperature when water was first credited at 660 seconds. When the initial AP600 PCS design flow was changed, Westinghouse calculated the revised PCS delay time to be 337 seconds by scaling the measured delay times. Westinghouse conservatively takes no credit for PCS cooling prior to 337 seconds in the AP600 EM model. The enhanced delivery rate provides additional assurance that the vessel surface will not overheat and inhibit film formation. Flat plate tests (FPT) performed at the Science and Technology Center Flat Plate Facility (and documented in WCAP-12665) demonstrated the ability to wet and rewet a hot, dry, coated surface at an estimated temperature of 115.5 °C

(240 °F). As the thermal time constant of the shell is about 200 seconds, the shell exterior should not exceed this temperature during the 337-second initiation time. An analysis presented by Westinghouse in Section 7 of WCAP-14407 indicates that the containment shell would heat up from the assumed initial temperature of 48.9 °C (120 °F) to about 86.7 °C (188 °F), or a temperature rise of 37.8 °C (68 °F).

Westinghouse models the AP600 as follows: 90 percent of the surface may be wet if sufficient PCS water is available and 10 percent is always considered to be dry. These values are based on the original unheated WDT data, where the maximum PCS flow rate was 832.8 liters/min (220 gpm). Characterization of the PCS flow patterns is also based on measurements and visual observations from the unheated WDT and the heated FPT, as well as small scale tests (SST) documented in WCAP-14134, and large scale tests (LST) documented in WCAP-14135. The PCS flow is expected to form wavy film stripes centered under the notches in the water distribution weirs. The stripe widths are assumed to remain constant along the shell from the spring line to the upper annulus drain elevation, a distance of about 26 m (85 ft), until evaporation reduces the thickness of the water film to less than the minimum thickness needed for film stability. Westinghouse uses a constant value for the minimum stable film thickness for surfaces with this specific inorganic zinc coating. This value was selected to conservatively bound all available Westinghouse proprietary test data over the range of interest for AP600.

After reaching the minimum thickness for a stable film, the wetted perimeter (frontal width) of the film stripe is assumed to reduce exponentially, such that the film continues to evaporate at the stability limit. Westinghouse justified these assumptions by qualitative observations of the heated tests cited above. The staff finds this to be acceptable.

21.6.5.4.2.3 PCS Evaluation Methodology

Westinghouse developed three special models to analyze the AP600 PCS:

(1) the clime model for heat transfer to the environment

(2) PCS water coverage and film tracking

(3) two-dimensional heat conduction.

The clime model is the major Westinghouse addition to the GOTHIC computer program. The other two models are evaluated externally to WGOTHIC using spreadsheet-based calculations. WGOTHIC is limited in its ability to track the time, flow, and heat flux dependent variations in wetted and dry areas, nor does WGOTHIC calculate two-dimensional heat conduction. Instead, the Westinghouse AP600 EM uses other methods to calculate the following inputs to WGOTHIC:

- the time-dependent PCS flow which evaporates (the "evaporated flow" does not include runoff)
- a factor which increases this flow as a credit for the expected effects of two-dimensional heat conduction

The time-varying flow boundary condition supplied to WGOTHIC is the adjusted PCS flow which would evaporate if the WGOTHIC clime model did not have these deficiencies. The three models are discussed in the following sections.

21.6.5.4.2.3.1 Clime Model for Heat Transfer to the Environment

In the Westinghouse methodology, the PCS is modeled as stacks of special conductor types, called "climes". Climes model the heat transfer processes (condensation, conduction, evaporation, convection and radiation) from inside the AP600 containment to the surrounding environment. Climes solve the one-dimensional heat transfer and tightly coupled mass transfer equations for these multiregion slices. The mass and heat transferred to or from the clime are coupled to the existing GOTHIC fluid nodes through source terms. These interfaces are part of the current GOTHIC program. Stacks of clime heat structures track the thickness of evaporating and condensing water films flowing down the interior and exterior containment surfaces.

Each clime represents a horizontal slice through the AP600 containment vessel, the other PCS heat structures (the baffle and the shield building), and their interaction with adjoining GOTHIC fluid cells. A clime includes an adjacent portion of the containment internal atmosphere including the condensate film on the containment shell interior, the inorganic zinc coating on the inside of the steel shell, the vessel steel wall, the inorganic zinc coating on the outside of the steel shell, the PCS water film, the riser air region, the baffle plate, the downcomer air region, and the shield building concrete structure. PCS air flow passages are modeled in climes as embedded GOTHIC nodes. A single clime is shown in Figure 21.6-25.

WGOTHIC divides the AP600 containment shell into a grid of climes. Westinghouse uses a rather course axial grid, with three dome climes and four sidewall climes. The average sidewall clime is approximately 5.6 m (18.4 ft) tall. The tallest clime is 6.7 m (22 ft). The climes also split the containment shell into 4 azimuthal quadrants. The air riser is not azimuthally segmented. Each quadrant is further subdivided into a pair of wet and dry stacks. The stacks each have two vertical segments: from the top of the containment dome to the second weir (at the 15.24 m (50 ft) radius), and from the second weir along the sidewall surface and down to the upper annulus drain region, which is near the elevation of the operating deck.

Clime heat transfer mechanisms include condensation on the containment shell interior, and conduction through the shell wall. PCS coolant evaporates from the wet portion of the shell exterior surface, and convective heat transfer cools the dry exterior portion. Open Item 21.6.5-20 concerned condensation heat transfer on the baffle from the riser air. Westinghouse clarified the WGOTHIC documentation to indicate that heat transfer to the baffle from the riser air includes both convection and condensation, so Open Item 21.5.8-5 is closed. The baffle also receives wall-to-wall radiation from the containment shell.

Heat, which is conducted through the thin steel baffle, and convection from the baffle, warms the downcomer air flow, slightly diminishing the density head which drives the natural circulation. The baffle also dissipates heat by radiation to the shield building wall. Conduction into the shield building concrete wall is also modeled.

In WGOTHIC, the heat fluxes for each clime are calculated separately, with wet and dry climes modeled as parallel, non-communicating heat conductors. Because the PCS flow is assumed to

be uniformly redistributed at the second weir, clime stacks above the second weir and below the second weir are treated separately. The coverage fractions, expressing the fraction of the total clime surface covered by the water film, are based on measurements from the unheated WDTs. Visual observations from heated tests indicate that the coverage fraction increases as the surface temperature increases. To account for heat transfer between the wet and dry clime, Westinghouse developed a method, as discussed in Section 21.6.5.4.2.4 of this report, to modify the coverage fraction.

21.6.5.4.2.3.1.1 Clime Conservation Equations

Initially, the clime conservation equations were based on these assumptions:

- one-dimensional film flow along the wall
- one-dimensional radial heat conduction.
- constant thermal fluid properties.
- neglect of viscous dissipation.

Westinghouse chose well-known, widely used heat and mass transfer correlations for the clime heat and mass transfer methodology. Westinghouse applies additional, conservative bias multipliers for the heat and mass transfer correlations used for the climes. These multipliers were derived from both open literature and proprietary AP600 supporting test data (see Section 21.6.5.6 of this report), and address the SRP guidelines for demonstrating conservatism in the peak containment pressure calculations.

Westinghouse later modified the clime methodology to include an adjustment factor to credit two-dimensional heat transfer effects when the PCS flow is reduced and the water area coverage decreases. This modification accounts for 2-D conduction from the hot, dry region of a clime to its neighboring cold, wet region. (See Section 21.6.5.4.2.4 of this report).

Clime Heat Transfer

The clime model uses the heat transfer correlations shown in Table 21.6.5.4-3. Heat transfer to the shield building and baffle from the downcomer is considered "mixed-opposed" convection, which WGOTHIC models using the Churchill correlation. After entering the riser, the Colburn forced-convection correlation is used for the containment shell side of the riser. Scaling analysis as documented in Westinghouse letter NSD-NRC-97-5152, dated May 23, 1997, showed that forced convection is warranted with as little as a 1.1 °C (2 °F) temperature rise. The baffle side of the riser is treated as "mixed-assisted" convection. At some point along the containment cylinder and dome, the riser air may be hotter than the PCS film. Above the springline, the riser flow area expands significantly. Open Item 21.6.5-7 expressed the staff's concern regarding the use of a forced convection correlation for this region at the top of riser and above the dome. WGOTHIC documentation was clarified to indicate that the Uchida correlation is used for condensation heat transfer in this region, as well as for the very top of the chimney. Uchida is also used for miscellaneous heat structures in the air flow path. The conditions in the region are consistent with the use of the staff-approved Uchida correlation. This closes Open Item 21.6.5-7

Clime Mass Transfer

Clime mass transfer correlations are used for condensation on the inside of the containment shell, evaporation from the outer surface of the shell, and condensation on the riser baffle surface during its short initial heatup period. Convective mass transfer is a result of a concentration gradient between the liquid surface and a flowing steam-air gas mixture. To approximate the concentration gradient, the WGOTHIC model uses the following approximations:

(1) The steam concentration gradient is approximated as the difference in steam partial pressure between the bulk gas and liquid surface. The appropriateness of this approximation is discussed in Sections 21.6.5.4.1 and 21.6.5.7.4 of this report, along with mixing and circulation issues.

(2) Condensation occurs when the bulk gas steam concentration is greater than the concentration at the surface of the liquid.

(3) Evaporation occurs when the bulk gas steam concentration is less than the concentration at the surface of the liquid.

These assumptions simplify the storage, thermal diffusion, and axial transport terms of the equations. The mass transfer analogy used from the clime model is discussed in Section 21.6.5.6.1.5.

21.6.5.4.2.3.1.2 Clime Numerical Approach

The AP600 EM does not solve the initial PCS flow transient; instead, PCS flow is only applied to the dome and sidewall after coverage has been fully established. The clime flows are determined using a quasi-steady state approach. The continuity, energy, and momentum equations for the air flow in the downcomer and riser are solved in the WGOTHIC clime calculation using embedded GOTHIC lumped-parameter nodes. These nodes represent the flow network from the ambient environment through the downcomer and riser flow passages exiting back to the ambient environment at a higher elevation. Pressure boundary conditions are imposed on the two volumes representing the ambient environment. These volumes are treated outside the GOTHIC solution methodology for the field conservation equations. They only affect the source terms for the associated GOTHIC node liquid and vapor mass and energy conservation equations.

Stability and convergence problems can arise if the time steps are too large for the explicit linkage between the (implicit) clime conduction model and the (semi-implicit) GOTHIC calculation scheme. This could be a concern if the heat and mass transfer between the clime and the fluid cell were sufficiently high to cause a significant change in the temperature of the fluid in a time step, or if some condition resulting in rapid condensation occurred (such as at the initiation of PCS film flow). Numerical instability problems due to explicit coupling usually can be avoided through the use of a sufficiently small time step. If the time step used were not sufficiently small, a typical consequence would be the onset of an oscillatory instability (with a half-period of one time step). Water property routines often fail shortly after the onset of severe oscillations, so they usually do not go unnoticed. However, less severe oscillations can only be

identified by examination of key plotted results, such as wet and dry clime heat flux versus time. The staff therefore requires, as a condition for acceptance of WGOTHIC, that sufficient clime key results be examined to confirm the time step stability for each case analyzed. (See Section 21.6.5.8.3 of this report).

21.6.5.4.2.3.2 PCS Water Coverage and Film Tracking

The second special model characterizes PCS water coverage and film tracking. This is not a model in the sense that it is included in the WGOTHIC computer program. This calculation is performed external to the WGOTHIC code.

21.6.5.4.2.3.2.1 The Evaporated-Flow Model

In the WGOTHIC AP600 EM, the fractions of the clime surfaces covered by the water film are held constant throughout the transient, independent of time, axial position, or variations in heat flux. Instead of time-varying these coverage areas, the applied PCS flow rate is limited to the mass expected to evaporated. Westinghouse developed an "evaporation limited" model to account for the wetted surface area change. This model computes, outside of WGOTHIC (see Section 7 in WCAP-14407), a boundary condition in the form of an input PCS flow-versus-time table. This table specifies the PCS flow which is expected to evaporate and omits the PCS flow that is expected to run off the bottom of the vertical sidewall. This allows Westinghouse to maintain a single 90 percent wet clime model throughout a WGOTHIC analysis. Instead of time-varying areas, the WGOTHIC boundary conditions specify time-varying PCS flows. These flows are applied to the top of each wet clime stack. WGOTHIC reduces the flow entering each downstream clime in a stack by the amount evaporated upstream. When the flow rate drops below the minimum necessary to maintain a stable film, the wetted perimeter in the clime is reduced. Therefore, "wet climes" need not be completely covered by the PCS water film.

Time, axial position, and heat flux on the PCS heat removal rate enter into the WGOTHIC calculation as boundary conditions for the time-dependent PCS flow. This external calculation proceeds as follows:

(1) A constant value for the dome coverage fraction (i.e., above the second weir) is used, based on data taken from the unheated WDTs.

(2) From the second weir down, coverage fractions are shown in Table 21.6.5.4-4. These values are also based on WDT data. There are three distinct periods, or phases, of AP600 PCS flow. After 3 hours and 30 hours the PCS flow rate is reduced and the actual coverage fractions are reduced accordingly.

The modeling approach used by Westinghouse to determine coverage area and PCS flow characteristics is based on data from the WDTs. The data taken from the WDT were measurements of the wetted perimeter as a function of the flow rate. The wetted perimeter determines the coverage area, as the tests showed, the water fell uniformly along the length of the side wall of the test facility. During the initial coverage period, more than adequate supplies of water are available to preclude dryout: below the second weir, 90 percent coverage is used for the first 3 hours (Phase I), and 10 percent of the surface is always considered to be dry. These values are based on the original WDT data where the maximum test flow rate evaluated was equivalent to 832.7 liters/min (220 gpm). When the AP600 initial PCS flow rate was

increased to 1665.4 liters/min (440 gpm), Westinghouse conservatively elected not to increase the coverage fraction during the initial phase. This period encompasses the time of peak pressure, which occurs in the 1,000- to 2,000-second range. The Westinghouse response, through sensitivity studies, to RAI 480.902 showed that these PCS flow rates are adequate to remove the heat flux during the period of interest.

Based on data from the WDT, for Phase II the coverage is reduced to an equivalent 51 percent wet stripe, and during Phase III to an equivalent 30 percent wet stripe. Coverage values for Phase II and III are also considered conservative relative to the WDT measurements and include the uncertainties in the WDT coverage fraction measurements. A COL applicant will confirm these coverage fractions for the as-built AP600 containment during preoperational tests. Periodic testing, as part of the inservice testing program, will also be performed to demonstrate that the area coverage fractions are maintained over the life of the plant. The staff therefore considers these coverage fractions acceptable for use in the licensing analyses to support design certification.

Outside of WGOTHIC, a spreadsheet procedure is used to calculate the expected evaporation rate from the PCS surface. Westinghouse then revises the PCS flow-versus-time table to neglect the PCS water which would run off from the bottom clime. This revised table is used as a boundary condition for WGOTHIC. Using data from tests (e.g., the WDT, the FPT, and the LST) that utilized inorganic zinc surface coating, Westinghouse developed an application-specific film flow stability criterion. This criterion is a value for the minimum wetted perimeter to maintain a stable film, which bounds the available unheated and heated test data and the Zuber-Staub theoretical model over the AP600 range of applicability. Westinghouse adopted this empirical approach to address staff concerns (as identified in Open Item 21.6.5-18) with the initial procedure, which used the Zuber-Staub correlation normalized to the unheated WDT data. Open Item 21.6.5-18 is now closed.

This film stability criterion is used in the spreadsheet calculation performed external to WGOTHIC. Other inputs to the spreadsheet are the expected time-dependent heat flux, PCS flow rate, and coverage fraction (see Section 7 of WCAP-14407). The spreadsheet calculates the amount of water which is expected to evaporate from the containment shell surface. The adjusted time-dependent PCS flow rate is applied to a WGOTHIC AP600 calculation as a boundary condition. In WGOTHIC, Westinghouse fixes the wet clime coverage area (wetted perimeter) at 90 percent throughout an analysis. The flow entering a downstream clime is the flow which originally entered the top clime, minus the flow lost in the intervening node. It is assumed that evaporation is the only means of removing water, and a time-weighted average heat flux is used to calculate the evaporation rate. The coverage fraction for a downstream clime is assumed to be the same as its upstream neighbor, unless the flow drops below the minimum value required for film stability (as predicted by the Westinghouse film stability criterion). When the flow entering a clime decreases below this minimum, the coverage fraction used in the heat and mass transfer calculations for the clime is reduced by the ratio of the available flow to the minimum stable flow, and the wetted perimeter is assumed to exponentially decrease. If, in the actual WGOTHIC calculation, PCS water remains unevaporated at the bottom of the shell, an iteration through the external spreadsheet calculation is performed and the WGOTHIC calculation is redone. As a last step, for times after 3 hours into an event, Westinghouse increases the calculated flow, augmenting the evaporation from the shell to credit two-dimensional heat conduction effects (see Section 21.6.5.4.2.3 of this report). This is

referred to as the "evaporated-flow" model. As a result, during the critical LOCA peak pressure time frame (first 2,000 seconds), only about two-thirds of the available PCS water is used as the boundary condition. This model tends to remove heat preferentially from the upper regions of the containment. Westinghouse sensitivity studies (see Chapter 7.7 of WCAP-14407) have shown little sensitivity to vertical distribution of the applied flow, hence little sensitivity to vertical stratification in the containment atmosphere. However, concentration gradients caused by the build up of an air-rich layer near the walls may affect the PCS flow. Acceptance of the evaporated-flow model, is therefore based on the validity of the "well-mixed" containment assumption. This assumption is discussed in Chapter 9 of WCAP-14407 and Section 21.6.5.7.4 of this document.

From an analytical perspective, this approach leads to additional uncertainties in assessing heat transfer from wet climes, and circulation (mixing) and stratification within the interior upper region of the containment (the region above the operating deck). Table 21.6.5.4-5 compares the conservatisms and non-conservatisms inherent in the evaporated-flow model. Westinghouse has performed sensitivity studies to show that the calculated peak pressure results, for both LOCAs and MSLBs, are nearly equivalent, with and without use of the evaporated-flow model. These sensitivity studies, as documented in Westinghouse's response to RAI 480.873, Revision 1, showed that overall, the use of the evaporated-flow model results in a conservative peak pressure calculation.

21.6.5.4.2.3.2.2 WGOTHIC Water Film Thickness Model

As discussed in Section 21.6.5.4.2.3.2 of this report, WGOTHIC boundary conditions specify the time-dependent PCS water expected to be evaporated in each climes stack. This water is applied to the top clime and flows down the stack of climes. In each clime, a portion of the input flow evaporates; the remaining flow enters the downstream clime.

To calculate the thermal resistance across the internal and external films, Westinghouse does not use transient mass balance equations to predict the film thickness; instead, the Chun and Seban correlation is used to calculate the Nusselt number as a function of the Reynolds number (laminar flow) or Reynolds and Prandtl numbers (turbulent flow). In effect, this correlation calculates the effective average thickness for heat transfer through a wavy film. The Chun and Seban correlation was originally developed for evaporating vertical films. Using condensing data from the Wisconsin tests, Westinghouse extended this model to condensation and non-vertical surfaces. The Chun and Seban liquid film conductance model was compared to additional data from Kutateladze et al. (Kutateladze, S.S., Gogonin, I.I., Grigo'eve, N.I. and Dorohkov, A.R., "Determination of Heat Transfer Coefficient with Film Condensation of Stationary Vapour on a Vertical Surface," Thermal Engineering, Volume 24, Number 4, pp 184-186 (1980)) in WCAP-14326, Revision 3. This comparison provided additional justification for the Chun and Seban correlation in the low Reynolds number range (about 700) with the comparison showing the correlation to be a good fit to the evaporation data and conservative for the condensation data. The liquid film conductance is a low-ranked phenomenon and the use of Chun and Seban is acceptable. The Westinghouse extensions to the Chun and Seban correlation are considered acceptable by the staff. This closes Open Item 21.6.5-2.

However, use of the Westinghouse evaporated-flow methodology, together with the Chun and Seban correlation, may non-conservatively underestimate the film resistance during time phases when the evaporated-flow is much lower than the actual applied flow. During those time phases,

the Westinghouse methodology conservatively neglects the sensible heat of the runoff flow. Open Item 21.6.5-4 expressed the staff's concern regarding potential non-conservatisms in the film heat and mass transfer models as a result of using the evaporated-flow model (i.e., underprediction of the liquid film resistance and omission of runoff liquid enthalpy transport in AP600 licensing calculations). The information presented in Westinghouse's response to RAI 480.873, Revision 1, closes Open Item 21.6.5-4. As shown in the response to RAI 480.873, Revision 1, neither of these approximations is significant during the first 3 hours. The results of a sensitivity study with full PCS flow showed the evaporated-flow method is slightly conservative. After the initial period, neglecting runoff sensible heat is conservative, which offsets the non-conservatism introduced by the simultaneous use of the Chun and Seban correlation and the evaporated-flow model. As long as these two assumptions are employed together, the staff considers this model to be acceptable.

21.6.5.4.2.3.2.3 Nodalization Studies

As part of a clime sensitivity analysis performed by Westinghouse, it was demonstrated that, at least numerically in WGOTHIC, a clime could be broken up into multiple sections, either vertically or horizontally, while maintaining a single containment node and riser air node on each side of the clime. The computed results were shown to be identical. This approach appears to be inconsistent with previous Westinghouse studies in WCAP-14382. These studies used a one-to-one relationship between a clime (wet and dry pair) and a set of GOTHIC nodes (one inside containment and one in the air riser gap). However, for well-mixed containments, the staff considers this simpler approach to be an acceptable modeling practice when the nodalization of the above-deck region inside containment is demonstrated to be reasonable (converged and numerically stable). The AP600 evaluation model, as described in WCAP-14407, has been shown to be both converged (see Section 12 of WCAP-14407), and numerically stable (see Section 11 of WCAP-14407).

21.6.5.4.2.3.2.4 Sensitivity Studies

Another clime sensitivity analysis performed by Westinghouse in Section 5 of WCAP-14407 demonstrated that initial conditions, relative humidity, and fog formation in the riser had little effect on the riser air flow velocity. Fog formation was also shown to have little impact on riser-to-baffle radiation heat transfer as documented in Section 4.4.7l of WCAP-14812. The radiation heat transfer is a low-ranked phenomena. This information closes Open Item 21.6.5-9.

21.6.5.4.2.4 Adjustment Factor For Two-Dimensional Heat Conduction

After 3 hours and again after 30 hours, the flow applied to the exterior surface of the PCS shell is reduced. Data obtained from the unheated water distribution tests (WDT) showed that at the lower flow rates, only a fraction of the PCS shell is covered (see Table 21.6.5.4-4). Based on the WDT data, after 3 hours the PCS flow rate drops to 120 gpm, which results in 51 percent coverage. From 30 to 72 hours, the PCS flow rate drops to 72 gpm, resulting in 30-percent coverage.

To evaluate the containment pressure response when the water coverage fraction is reduced (after 3 hours), Westinghouse added a model to account for the effect of two-dimensional (2-D) heat transfer on the long-term containment pressure response (WCAP-14407 and

Westinghouse letter NSD-NRC-97-5152 dated May 23, 1997). This model is based on the following physical phenomenon: when the coverage area is reduced, heat will flow circumferentially from the hot, dry regions of the shell into the cooler, wet regions of the shell. To quantitatively correct for this previously unmodeled phenomenon, Westinghouse performed ancillary ANSYS 2-D heat transfer calculations to determine a credit factor as a function of coverage area. Each ANSYS calculation modeled an infinitely repeating cell which contained half of a wet and half of a dry stripe. The cell width was fixed by the width between weir V-notches, assuming that stripes have a uniform width and are each centered below each V-notch. The ratio of water evaporated in the ANSYS 2-D calculation to the water evaporated in the 1-D WGOTHIC calculation is used to generate a factor "M." The "M" factor is always greater than one, and becomes large at low coverage fractions. When the actual coverage fraction falls below 90 percent, "M" is used to multiply the applied evaporation limited PCS flow.

Concerns with the 2-D conduction model include:

- the validity of the uniform-width/notch-centered stripe assumption, particularly on the dome and on the lower portions of the PCS shell
- the validity of the tradeoff of circumferential heat flux for greater PCS flow
- the PCS shell boundary conditions used by Westinghouse for the ancillary ANSYS calculations
- the limited test data available for validation studies at low PCS flow rates

While the 2-D conduction phenomenon exists, Westinghouse has not sufficiently demonstrated that its evaluation methodology for 2-D conduction is conservative.

The staff believes that 2-D heat conduction has a real effect. However, as an insufficient amount of test data is available to validate this model, the staff is unable to determine how much credit should be given in evaluating the Westinghouse design performance after 24 hours. In addition, the calculated pressure is not used to demonstrate compliance with other regulatory requirements (see Section 21.6.5.3.2 of this report). Whether or not credit is taken for 2-D heat conduction, the staff finds the design to be in compliance with GDC 38 and the containment pressure and temperature following the limiting loss-of-coolant accident are maintained at acceptably low levels. Although the containment pressure response is different from current licensed plants, the PCS is acceptable and consistent with the passive design objectives on which the AP600 PCS is based.

21.6.5.4.2.5 Clime Model Validation

The WGOTHIC clime model was benchmarked to LST data. As discussed in Section 21.6.5.5 of this report, the LST was not appropriately scaled to be an AP600 prototypic test facility. The LST also had a plexiglass baffle and no downcomer. Because of these limitations, these tests cannot validate the correctness of WGOTHIC calculations for the AP600 containment. However, the LST did use a PCS-like evaporative cooling mechanism, and specific tests were performed with natural circulation air flow. Westinghouse used data from these tests to validate the clime calculations. This limited use of the LST data is judged to be acceptable for clime

model validation, as the analyses demonstrate that the clime model has been implemented correctly into WGOTHIC.

In addition to the LST data, Westinghouse presented separate-effects calculations and sensitivity studies to validate the heat and mass transfer models for a single clime component. These separate-effects studies helped to verify the functionality of the clime calculation as incorporated into WGOTHIC. The staff accepts the separate-effects calculations and sensitivity studies (WCAPs -14407, -14832, and -14967) as sufficient verification (i.e., demonstrating the correct operation) of the PCS models, including the clime model and the evaporated flow model. These calculations show that the models added to GOTHIC by Westinghouse to form WGOTHIC have been properly implemented and the calculated results are consistent with the data from the tests.

21.6.5.4.3 Summary of Code Changes from WGOTHIC Version 1.2 to Version 4.2

21.6.5.4.3.1 WGOTHIC 4.0 Consistency with GOTHIC Version 4.0

Many of the changes implemented by the developer of GOTHIC version 4.0 (Numerical Applications, Inc. (NAI)) had been incorporated into WGOTHIC Version 1.2 as a result of Westinghouse working closely with NAI. From this collaboration, a number of code errors and upgrades identified early in the Westinghouse development program were corrected prior to configuring WGOTHIC Version 1.2. Most of the upgrades resulting from that interaction were implemented by NAI in GOTHIC Version 4.0. NAI made these code changes to correct deficiencies, including those noted in user reports, or to provide more realistic models. NAI tested each of the changes individually and confirmed correct implementation, and the entire change set was tested as a whole in the GOTHIC Version 4.0 code release testing. This process resulted in a limited number of differences between WGOTHIC solver Version 1.2 and GOTHIC solver Version 4.0.

An assessment was performed by Westinghouse to show the effects of differences between GOTHIC Version 4.0 and WGOTHIC Version 1.2. An intermediate configuration control version of WGOTHIC, Version 1.2.2, was created. The assessment showed that the difference in calculated pressure resulted from the improved droplet model.

WGOTHIC solver was upgraded from Version 1.2 to Version 4.0 to incorporate residual changes that had been made in the upgrade to GOTHIC 4.0. WGOTHIC is therefore consistent with GOTHIC Version 4.0, which was baselined under the NAI quality assurance program.

21.6.5.4.3.2 Changes From WGOTHIC 1.2 to 4.1

The changes from WGOTHIC solver Version 1.2 to Version 4.1 did not affect the PCS heat and mass transfer correlations. The changes can be grouped as follows:

- GOTHIC 4.0 upgrades and error corrections
- GOTHIC 4.0 documentation updates
- GOTHIC 4.0 EPRI-sponsored peer review and quality assurance program

- upgrades to "ccvel" subroutine
- correction of errors in the WGOTHIC clime subroutines

The observed variation in results from WGOTHIC Version 1.2 to Version 4.1 can be attributed to a number of changes:

- The minimum value of the Uchida condensation heat transfer coefficient was changed from 15 to 2, thereby reducing the condensation rate at low steam partial pressures.
- The existing method for calculating steam saturation pressure as a function of temperature was replaced by a more accurate method.
- The wall dryout criterion used for internal heat sinks was modified to allow water to remain on walls until the wall temperature is greater than T_{sat} (total pressure), rather than T_{sat} (steam pressure).
- The droplet drag and deposition models were improved. This change affects the drop energy exchange rates.
- An error in the clime dryout model was corrected.
- Variables in the underlying GOTHIC subroutines and in the WGOTHIC clime subroutines were converted to double precision (GOTHIC versions prior to 4.0 had only used double precision where judged to be most significant, as in the matrix operations).
- Several miscellaneous modifications were made to improve consistency and reliability.

The changes made to create WGOTHIC solver Version 4.1 and its documentation have improved the code's solution, models, and reliability.

WGOTHIC 4.1 - Upgrade to "ccvel" Subroutine

The "ccvel" subroutine was upgraded to improve the cell-centered velocity calculation associated with lumped-parameter fluid nodes. The cell-centered velocity is only used in calculating heat and mass transfer in the external annulus. The resulting cell-centered velocities calculated by the two versions were verified to be similar, so it can be concluded that the differences between the velocities predicted by the two codes are not the primary reasons for the differences between calculated pressures.

WGOTHIC 4.1 - Correction to Partially Wet Clime Logic

In performing verification activities with WGOTHIC solver Version 4.0, Westinghouse discovered that the clime subroutines handled heat and mass transfer from a partially wet clime incorrectly. The error in clime logic caused the PCS heat removal to be overpredicted at the point of dryout. Changes were made to the Westinghouse clime subroutines to correct that error. The subroutines were modified to check for dryout; that is, to check if the evaporating mass flux times the clime area exceed the liquid film mass flow delivered to the top of the clime. If dryout occurs, the code divides the liquid mass flow rate by the mass flux obtained from PCS

correlations, and calculates the area which is required to completely evaporate the film. The remaining area within that clime is then assumed to have only dry heat transfer. Properties, temperatures, and heat transfer for the dry portion of the clime are calculated assuming instantaneous quasi-equilibrium, dry conditions. This approach conservatively neglects the heat capacity of the steel shell as its temperature increases from the cooler wet state to the hotter dry state.

Other minor code changes were made to correct some non-calculational problems, and WGOTHIC Version 4.1 was created.

Summary of Code Changes from Version 4.1 to Version 4.2

The following changes were made to WGOTHIC Version 4.1 in creating Version 4.2:

- created a new clime subroutine, "gvel," to provide cell-centered velocity direction for the clime calculations, allowing correct determination of assisting versus opposed convection in the downcomer
- replaced the modified GOTHIC "ccvel" subroutine, supplied by NAI, with the GOTHIC 4.0 "ccvel" subroutine and corrected the error in the effective-flow-area calculation
- replaced the single precision constants with double precision constants in subroutines mixed.f and props1.f
- increased the array dimensions for the GOTHIC conductors

Thus, known errors in the WGOTHIC clime subroutines have been corrected. In addition, known errors reported for GOTHIC Version 4.0, the basis for WGOTHIC Versions 4.0 and beyond, have been evaluated and determined not to be applicable to sections of coding exercised in the AP600 evaluation model.

Verification and validation of the code changes has been completed by Westinghouse. As part of the validation effort, a regression test was performed to confirm that the change from WGOTHIC Version 4.1 to Version 4.2 had no effect on calculated peak pressure.

Westinghouse has stated that WGOTHIC 4.2, the current licensing version, contains no known errors. The staff does not have any information to the contrary, therefore Open Item 21.6.5-15 is closed.

21.6.5.4.4 Models Not Used for AP600 Analysis

The staff review of the GOTHIC documentation that supports the WGOTHIC computer program is restricted to features and models used for the AP600 PCS DBA analyses. In Westinghouse letter NTD-NRC-95-4577, dated October 12, 1995, the staff was informed that the following

models were not used in the analysis of the AP600 containment and the supporting analyses of tests using the WGOTHIC computer program:

- Gido-Koestel condensation model
- Tagami heat transfer model
- drop entrainment and deposition
- tube and rod conductors
- emergency safety feature components (pumps and fans, valves, heat exchangers, vacuum breakers, spray nozzles, coolers, and volumetric fans)
- coupled boundary conditions
- "door" components

In reviewing the information against the GOTHIC documentation, the staff noted that the "door" component was not discussed. In response to RAI 480.470, Westinghouse informed the staff that there are no "door" components in the AP600.

As these components are not used in AP600 DBA analysis, the following models were also not considered during this review:

- distributed-parameter, finite-difference numerics
- BWR pressure suppression containment
- PWR ice condenser containment

21.6.5.5 PIRT and Scaling Analysis Methodology

The Westinghouse AP600 containment PIRT (phenomena identification and ranking table) is presented in WCAP-14812 and referred to in this document as the PIRT report. This WCAP also discusses how these phenomena are treated in the WGOTHIC evaluation model. The Westinghouse scaling analysis is documented in WCAP-14845. Both the PIRT and scaling analyses depart from the methodology described in NUREG/CR-5809. The Westinghouse PIRT and scaling approaches for the AP600 passive containment cooling system are described in Sections 21.6.5.5.1 and 21.6.5.5.3 of this report, respectively. Sections 21.6.5.5.2 and 21.6.5.5.4 of this report evaluate the PIRT and scaling analyses of the AP600 passive containment.

21.6.5.5.1 Westinghouse PIRT Methodology

The NRC staff requested that the Westinghouse PIRT include an overview of the process used to justify the conservative evaluation model for containment pressurization design analysis. Therefore, in addition to specifying accident scenarios and identifying and ranking important

phenomena, the PIRT report summarizes the WGOTHIC AP600 evaluation model treatment of all of the identified phenomena.

21.6.5.5.1.1 Accident Specification

Two limiting pressurization transients were chosen by Westinghouse as the basis for identifying and ranking phenomena to be considered in the evaluation model. The two limiting scenarios are the double-ended cold leg guillotine break from full power (Appendix K conservative value of 102 percent) and the main steamline break from 30 percent power. The main steamline break is more limiting at lower power because the liquid inventory is greater than at full power. The PIRT contains a detailed description of each scenario.

The mass and energy release and the passive cooling system (PCS) water flow rate are specified as boundary conditions for the containment design analysis. Along with the specified inside containment initial conditions, these phenomena were also identified and ranked in the PIRT.

Because the relative importance of phenomena can change as a transient progresses, Westinghouse considered four separate temporal phases for the LOCA, namely, the blowdown phase (0 to 30 seconds), the refill phase (30 to 90 seconds), the peak pressure phase (90 to approximately 1200 seconds) and the long-term phase (beyond approximately 1200 seconds). The blowdown period is the time during which the primary coolant inventory is released to the containment, resulting in a rapid pressure increase. During the refill period, the accumulators are injecting into the system and condensing steam. There is little or no mass and energy release to containment so the pressure decreases through steam condensation and heat absorption in structures. During the peak pressure phase, steam is generated by decay heat and from stored energy released from the reactor system internal heat sinks. This increased flow of energy is greater than the energy flow into containment structures and the shell so the pressure again increases. During the long-term phase, the heat removal through the shell (by the PCS) exceeds the energy release to containment, so the pressure slowly decreases. A schematic of the pressure response during the four temporal phases for the LOCA is given in Figure 21.6-26.

The mass and energy release from the main steamline break consists of superheated steam only. The inventory available for release is limited by the assumed closure of the main steam isolation valves (MSIVs). The steam release lasts for approximately 600 seconds. After the release is completed, the containment pressure decreases because there is no further addition of mass and energy. Therefore, the only temporal period considered for the main steamline break scenario is the blowdown.

21.6.5.5.1.2 Phenomena Identification and Ranking

Westinghouse used a combination of expert review, scaling analysis, testing, and sensitivity studies to identify and rank important phenomena. The identification and ranking process differed from that used in most PIRT studies in the sense that Westinghouse did not use a group of experts to identify and rank the important phenomena. Rather, a list of phenomena and rankings was developed by Westinghouse and then two groups of experts reviewed and commented on the Westinghouse PIRT.

One group of experts, referred to by Westinghouse as the internal experts, were all Westinghouse staff members knowledgeable of the AP600 project. The other group consisted of outside experts. There were four internal and four external experts. The two groups did not interact. Three of the four outside experts, referred to as the EPRI group, met and reviewed the Westinghouse PIRT. The fourth outside expert acted independently. The information in the PIRT report about the expert reviews is limited. The Westinghouse synopsis of the expert review process (Appendix A of WCAP-14812) is provided without identifying the input and opinions of individual experts.

The phenomena identified by Westinghouse are each described in some detail in Section 4.1 of the PIRT report. Phenomena are grouped by component or volume, in three categories: inside containment, containment shell, and outside containment. Phenomena associated with each component or volume are listed in the PIRT report (Table 4-1 of WCAP-14812).

Components or volumes inside containment consist of the break source, containment volume, containment solid heat sinks, initial conditions, break pool, and IRWST. From three to eight phenomena are identified and ranked for each of these components or volumes. The two components of the containment shell are the steel shell and the PCS cooling water. Fourteen phenomena are identified and ranked for the steel shell, and five phenomena are ranked for the PCS cooling water. Seven components or volumes are considered outside containment: riser annulus and chimney volume, baffle, baffle supports, chimney structure, downcomer annulus, shield building, and external atmosphere. From two to seven phenomena are identified and ranked for each of these components or volumes. A total of 81 phenomena are included in the PIRT. Rankings are provided for each of the four LOCA phases and for the main steamline break for each phenomenon. The phenomena are listed in Table 21.6-13. This table is the same as Table 4-1 of WCAP-14812.

Westinghouse provides a justification for the ranking assigned to each phenomenon. Most of the rankings are based on scaling results and expert opinion, but test results and sensitivity studies were also used to develop the ranking for some phenomena. Those phenomena that were ranked high during any phases of the LOCA or for the main steamline break have more extensive justification, generally including test results and/or sensitivity studies. Where scaling studies or test results were used to develop a ranking, Westinghouse cites specific test runs and provides numerical values of pi numbers to support the ranking.

21.6.5.5.1.3 Treatment of Phenomena in WGOTHIC Evaluation Model

Westinghouse gives the basis for ranking each of the 81 phenomena, discusses how the phenomena are implemented in the WGOTHIC evaluation model, justifies the evaluation model treatment, evaluates uncertainties, and discusses distortions in test facilities, namely the LST.

This information is a compilation of material presented in other reports, primarily the Applications Report (WCAP-14407). The inclusion of this material in the PIRT report was requested by the NRC staff. This material is necessary to show closure of the PIRT and the evaluation model. For example, sensitivity studies performed with the evaluation model for water coverage confirmed that a high ranking is appropriate. Placing the ranking basis and the evaluation model treatment of each phenomenon in the same section facilitates a comparative assessment of the rationale for the ranking and the degree of conservatism in the evaluation model.

21.6.5.5.2 AP600 Containment PIRT Evaluation

The three aspects of the PIRT report described in the Section 21.6.5.5.1 of this report, namely, the scenario identification, phenomena ranking, and treatment of phenomena in the evaluation model, are discussed below in Sections 21.6.5.5.2.1, 21.6.5.5.2.2, and 21.6.5.5.2.3, respectively. To facilitate discussion of the evaluation, Table 4-1 of the Westinghouse PIRT report is repeated here as Table 21.6-13. In the following tables, L, M, and H signify low-, medium-, and high-ranked phenomena. N/A means that the phenomenon is not applicable for that event or time phase.

21.6.5.5.2.1 Evaluation of Westinghouse Scenario Specification

The two accidents identified by Westinghouse as being limiting for the AP600 passive containment design, the double-ended cold-leg break from full power and the main steamline break from 30-percent power, are essentially the same as might be expected for a conventional PWR dry containment. The limiting events are determined primarily by the mass and energy release, so this similarity with existing designs is expected. These two events were chosen by Westinghouse as the basis for the phenomena identification and ranking and for the scaling analysis.

The actual limiting scenarios are determined by the final design calculations performed with the evaluation model. The events chosen as the basis for the PIRT and scaling are sufficiently representative of these limiting pressurization events to serve the stated purpose. The only differences are related to the actual mass and energy releases used in the SSAR analyses and minor changes to the containment layout. The staff concurs with this choice of event scenarios.

Westinghouse also described the key assumptions and initial and boundary conditions for the two scenarios. Limiting conditions for peak pressurization, such as high initial temperatures based on plant technical specification limits and maximum mass and energy releases, were chosen. The initial conditions themselves were considered by Westinghouse as part of the PIRT phenomena. Westinghouse correctly included a single-failure-to-open assumption for one of the valves in the PCS cooling water system. The staff concurs with the assumptions and the choices of initial and boundary conditions on the basis that they are sufficiently representative to provide an adequate framework for PIRT and scaling evaluations of the maximum containment pressure. The values used are typical for expected nominal conditions and the mass and energy releases are typical of those expected during DBA.

21.6.5.5.2.2 Ranking of Phenomena Evaluation

The Westinghouse approach to the use and documentation of expert opinions in the PIRT process differs markedly from that taken in most prior PIRTs. The more standard approach is to hold a meeting with all of the experts present. The phenomena are discussed among the group and an attempt is made to achieve a consensus on the ranking of phenomena. In documenting the process, the opinions of individual experts are identified. Differences of opinion are then a matter of record.

As discussed in Section 21.6.5.5.1, Westinghouse first wrote a draft PIRT report which included the report authors' rankings and discussions supporting the rankings. Separate groups of

experts reviewed the draft and then offered comments. Appendix A of the PIRT report contains a summary of the experts' comments. Westinghouse did not identify the individual experts' comments so it was not possible to identify who provided a particular comment or opinion. Because the PIRT performed by Westinghouse was not conventional, the staff put less weight on the experts' opinions and requested Westinghouse to provide a documented basis for each of the rankings. The staff requested that specific references be provided to experimental runs, sensitivity studies, and scaling analysis pi-groups used to establish the rankings. Westinghouse revised the PIRT report to include this information, which is the basis for the staff's acceptance of the PIRT results. Experts' opinions were still considered in the staff's evaluation, but less weight was given to the opinions because of the nature of the PIRT process used by Westinghouse.

The first question in evaluating the PIRT is whether the Westinghouse list of 81 phenomena is complete. While the PIRT process used by Westinghouse was non-standard in many respects, the staff concluded that it resulted in a list of phenomena which is comprehensive and complete. Even though the information provided in Appendix A of the PIRT is sparse, it does indicate that the experts were satisfied with the list of phenomena. The organization of the list was revised in response to staff comments and RAIs. The staff concurs that all relevant phenomena have been included and that the descriptions provided by Westinghouse are sufficient to clearly define each phenomenon.

Westinghouse provided rankings for each of the four LOCA phases and for the main steamline break for each phenomenon. Of the 81 ranked phenomena in the PIRT, less than half, 37, were ranked medium or high during at least one phase of the LOCA or during the main steamline break. The remaining 44 phenomena were ranked low or not applicable (N/A). Both medium- and high-ranked phenomena are treated in a conservative fashion in the evaluation model, so it is less important to distinguish between these two rankings. Low-ranked phenomena can be treated in a realistic or conservative manner, so it was important to establish that each phenomenon given a low ranking deserved that ranking.

The staff reviewed each of the low-ranked phenomena to identify any which could have been incorrectly ranked as low. Appendix A of the PIRT shows that the experts did not disagree with the ranking of any of the initially low-ranked phenomena. No experts commented that a low-ranked phenomenon should have been ranked medium or high. However, some of the rankings in the initial report reviewed by the experts were changed. For example, phenomenon 1E, Droplet/Liquid Flashing, was originally ranked high during the LOCA blowdown and refill phase. The ranking was changed from high to low in Revision 1. Experts had no opportunity to comment on this change in ranking. This is another shortcoming of the Westinghouse PIRT process. The staff notes, however, that lowering the ranking of some phenomena while establishing closure between the PIRT and the evaluation model is to be expected. Initially, a conservative approach should be used to establish the rankings. Development of additional information from scaling, experiments, and sensitivity studies may be used to revise the rankings. If the initial rankings were conservative, then most of the revisions would be expected to result in lower rankings. In the Westinghouse PIRT, lower rankings were assigned to droplet/liquid flashing and five other phenomena in Revision 1. The staff gave special attention to these six phenomena since their final rankings differed from the rankings the experts had assigned them.

In most cases, there is little question that the low-ranked phenomenon deserves its ranking. In the case of the droplets (Table 21.6.5.5-1, Phenomenon 1E), Westinghouse changed the ranking based primarily on sensitivity studies reported in WCAP-14407, which showed that there was an insignificant change in the peak pressure over the assumed range of "liquid fraction turned into drops" from 5 percent to 100 percent. For zero liquid turned to droplets, the peak pressure was lowered, so assuming that some droplets are present is conservative. The staff believes that the treatment of droplets in the Westinghouse model is non-physical. Droplets are assumed to remain in the atmosphere and to have a constant surface area throughout the LOCA scenario.

Nevertheless, Westinghouse has demonstrated that the droplets are treated in a conservative manner, just as if they were a medium- or high-ranked phenomenon. Hence, the staff finds that the treatment in the model is acceptable. The low ranking assigned in the PIRT becomes irrelevant, since treatment of drops in the model is that of a high- or medium-ranked phenomenon.

Liquid film energy transport on containment heat sinks (phenomenon 3A in Table 21.6-13) was initially ranked high during the peak pressure and long-term phases of the LOCA and medium during the LOCA refill phase and for main steamline break. In Revision 1 of the PIRT report it was ranked low for all phases of the LOCA and the main steamline break. The amount of energy carried off by the liquid film depends on the condensation rate and on the enthalpy difference between the saturated liquid and the reference enthalpy. A low level of uncertainty is associated with the enthalpy difference. The condensation rate is the main determinant of the magnitude of this term and the condensation is a high-ranked phenomenon. Westinghouse internal experts commented that this phenomenon was really a part of condensation and should be combined with that phenomenon. Since the condensation rate is the main determinant of the value of this term, the staff agrees with the Westinghouse experts. The low ranking given to this term then applies to the enthalpy difference. Scaling analysis performed by Westinghouse showed that the value of the sum of the pi-groups ($pi_{e,q,st}$ + $pi_{e,q,cc}$ + $pi_{e,f,jc}$) associated with this term was at most 0.08, so, given the high ranking assigned to condensation, treating the enthalpy difference as a low-ranked phenomenon is acceptable (e.g., the pi-group values is less than 0.1).

The PIRT experts ranked compartment filling as medium importance during the LOCA long-term phase and low otherwise. Revision 1 has this phenomenon ranked low for all phases. As for the enthalpy of fluid in the pool, the present evaluation model treatment is conservative with hot fluid modeled at the surface, as for a medium- or high-ranked phenomenon. The other aspect of filling is the closing off of flowpaths. In this regard, the uncertainty is low, volumes are estimated low (and hence will fill early), so the treatment is acceptable.

Phenomena 7A and 7B, convection and radiation heat transfer from the containment volume, were originally ranked of medium importance during the refill, peak pressure, and long-term phases of the LOCA. Westinghouse argued that a low ranking is justified because the combination of radiation and convection heat transfer is small compared to mass transfer. Since radiation is neglected and a conservative multiplier is applied to the convection heat transfer, the phenomena are conservatively treated, just as if they had been ranked medium or high. Hence, the low ranking has no consequence. The conservative treatment of these phenomena in the evaluation model is acceptable.

Convection to the downcomer (phenomenon 14A in Table 21.6-13) was ranked low in Revision 1 of the PIRT. It had initially been ranked medium during the peak pressure and long-term phases of the LOCA. The temperature differences in the downcomer region are less than those in the riser section. Westinghouse Figure 4-1, "Metals and Eckert plot," from WCAP-14845 shows that the downcomer is in the forced-convection regime. A mixed-convection heat transfer correlation recommended by Churchill is conservatively applied, although nominal multipliers are used on the correlation. In spite of the low ranking, a reasonably conservative approach was used, which the staff finds acceptable.

In one case, the staff questioned whether Westinghouse's assumptions about the phenomenon might have resulted in the low ranking and whether a higher ranking might have resulted if a different set of assumptions had been made. In the case of outside film conduction on the shell (item 7L of Table 21.6-13), the Westinghouse analysis appeared to be based originally on an assumed 5 mil film thickness. The staff asked whether assuming a thicker film would have changed the ranking. Westinghouse responded that the film thickness was not generally measured in the experiments, but that an equivalent film thickness could be backed out of the data. The film thermal conductivity was based on the Chun and Seban correlation, not on an assumed film thickness. Using the correlation, the film thermal conductivity was shown to be a small contributor to the overall thermal resistance across the shell. The staff concurs with the low ranking.

Those phenomena that were ranked medium or high during any phases of the LOCA or for the main steamline break were implemented in the evaluation model in a conservative manner. The next section summarizes the staff evaluation of the WGOTHIC treatment of the phenomena identified in the PIRT. All phenomena are included, with emphasis placed on the high- and medium-ranked phenomena.

21.6.5.5.2.3 Evaluation of Implementation in the WGOTHIC Evaluation Model

Section 4.4 of WCAP-14812 provides a roadmap of how each phenomenon is addressed in the WGOTHIC AP600 evaluation model. Details are generally given elsewhere, however, this section points to the location of the individual detailed discussions.

Table 21.6-14 summarizes the staff's assessment of the evaluation model treatment of the PIRT phenomena. In reviewing the PIRT phenomena, the staff emphasized the 37 medium- and high-ranked phenomena. Note that 18 of the 37 phenomena are ranked high and 19 are ranked medium. When a detailed discussion is needed to document the assessment, the discussion is deferred to a subsequent section in this evaluation. When the treatment by Westinghouse is clearly acceptable and no further discussion is needed (e.g. a phenomenon conservatively neglected), the assessment results are given in Table 21.6-14.

21.6.5.5.3 Westinghouse Scaling Methodology

The Westinghouse scaling evaluation in WCAP-14845 supports the PCS DBA conservative evaluation model. First, the scaling analysis is used to develop the PIRT (i.e., identify and rank physical phenomena for the containment pressurization model). Second, the scaling evaluation

identifies constitutive relations and correlations to describe the identified phenomena and then compares the range of the data used to develop the models to the range of AP600. Similarity variables are used to characterize the constitutive relations and correlations. Finally, the scaling evaluation considers the use of LST to validate conservatisms in the evaluation model at the scale of AP600. Distortions in the LST are identified and their effect on use of LST data for DBA model validation is discussed.

Containment pressure is the key variable for evaluating the performance of the passive safety systems. The evaluation model must demonstrate that the passive safety systems act to maintain the peak containment pressure below its design limit. The scaling evaluation provides support for application of the evaluation model at the scale of AP600.

One very significant element of the scaling evaluation is the pressure rate of change equation applied for two limiting scenarios, the double-ended cold-leg LOCA and the main steamline break. The dimensionless form of the pressure rate of change equation contains pi-groups whose relative magnitude is used to assist in ranking phenomena. Bottom up, component-level evaluation of the individual phenomenological models included in the pressure rate of change equation is another key element of the scaling evaluation. In Section 10 of the scaling report, Westinghouse seeks to demonstrate that the data base for the component models covers the range of similarity variables for AP600 and that the models are therefore applicable at the scale of AP600.

The integral test facility for AP600 PCS containment test data is the LST. Some of the distortions in the LST facility are identified and evaluated by contrasting the pressure rate of change equation pi-groups for that facility to the same pi-groups in the equation applied to AP600. Additional pi-groups are defined for the conservation of mass and conservation of energy equations. These pi-groups are also used in the comparison of LST to AP-600.

21.6.5.5.3.1 Component-Level Scaling - Constitutive Equations for Heat and Mass Transfer

Section 4 of the Westinghouse scaling report, WCAP-14845, describes the constitutive relations for heat and mass transfer used in the scaling equation and also, in most cases, in the evaluation model. Ranging of the variables in the correlations compared to AP600 is described in Section 10 of the Westinghouse scaling report and summarized in Table 21.6-15 of this evaluation.

Westinghouse has chosen to use established correlations from the literature to model radiation and convection heat transfer. The McAdams correlation

$$h_c = 0.13\ k/(v^2/g)^{1/3}(\Delta\rho/\rho)^{1/3}(Pr)^{1/3} \qquad \text{(Eq. 21.6.5.5-1)}$$

is chosen for turbulent free convection. The form of the McAdams correlation used depends only upon local conditions, including the ratio of the difference between the bulk and the surface density ($\Delta\rho$) and the bulk density (ρ). This correlation is used in the mass transfer model for condensation on the inside shell surface.

For turbulent forced convection, the Colburn correlation is used:

$$h_c = 0.023\ k/d_h\ (Re_d)^{0.8}(Pr)^{1/3} \qquad \text{(Eq. 21.6.5.5-2)}$$

This correlation is used in the model for evaporation mass transfer from the shell to the annulus. A standard laminar free convection heat transfer correlation from Kreith is used for the heat transfer to the liquid drops in the atmosphere.

Condensation and evaporation mass transfer (m") is modeled using a relationship from Kreith:

$$m'' = h_c\ \rho_{stm}\ (D_v/k)\ \Delta P_{stm}/P_{lm,air}\ (Sc/Pr)^{1/3} \qquad \text{(Eq. 21.6.5.5-3)}$$

In the previous equation $P_{lm,air}$ is the log-mean air pressure defined by:

$$P_{lm,air} = (P_{air,bulk} - P_{air,srf}) / \ln(P_{air,bulk} / P_{air,srf}) \qquad \text{(Eq. 21.6.5.5-4)}$$

and D_v is the air-steam diffusion coefficient. Westinghouse calculates this coefficient using a correlation given by Eckert and Drake.

Other correlations from the literature are also used. However, the above listed relationships govern the important energy transfer to and from the shell.

In Section 10 of the scaling report, Westinghouse compares the heat transfer correlations to separate-effects test data. The condensation correlation is compared to separate-effects data from the LST. The comparison is reasonable and the data are shown to cover the range of AP600. Similarly, for forced convection evaporation, the Colburn correlation is compared to data from the STC flat-plate test, and the data are also shown to cover the range of AP600. Westinghouse noted that the recommended range of the Reynolds number for the Colburn correlation includes the range for AP600.

Water coverage is scaled by Westinghouse based on three similarity groups, the Reynolds number, Marangoni number, and the Bond number for the liquid film. With one minor exception, the range of these dimensionless groups for AP600 is covered by test data from the LST and the full-scale water distribution test. The film Reynolds number on the upper sidewall of the containment exterior surface is higher in the AP600 than covered in the LST and water distribution tests. However, the Marangoni and Bond numbers show the range is adequately covered.

The Chun and Seban correlation is used to model heat transfer through the liquid film on the inside and outside of the containment shell. Westinghouse provided a comparison of this correlation to the Chun and Seban data and included data from the University of Wisconsin tests, which included surface inclinations from vertical to horizontal but covers only the laminar range. Between the Wisconsin and Chun and Seban data, the Reynolds number range of AP600 operation is covered.

Scaling of PCS air flow path resistance is also described in Section 10. Westinghouse performed a 1/6-scale air flow test, as documented in WCAP-13328, on a geometrically scaled model of an early version of the AP600 design. A Reynolds-number-based correlation was developed considering the composition of the losses as partially form and partially friction.

21.6.5.5.3.2 System Level Scaling - Pressure Rate of Change Equation

Development of the pressure rate of change equation is described in Sections 5 through 7 of the scaling report. It is applied to the LOCA and main steamline break in Section 8 of the scaling report. Coupling to the air flow equation for the downcomer and annulus is discussed in Section 9 of the scaling report. Top-down scaling of integral test data from LST is discussed in Section 10 of the scaling report.

Westinghouse derives an equation which governs the rate of pressure change (RPC) inside containment, based on their assumption of a completely mixed atmosphere. This assumption is further addresses in Section 21.6.5.5.4.2 of this report. The equation is developed from the conservation of energy equation and the equation of state in the form of a rate of change of internal energy relationship. Dimensional (WCAP-14845, Equation 63) and non-dimensional (WCAP-14845, pp. 10-14) forms of the equation are presented. The RPC equation is rendered dimensionless by dividing all terms by a reference break gas work term. The coefficients of the various terms in the dimensionless form of the equation are referred to as "pressure pi-groups."

Dimensionless forms of the conservation of energy, conservation of mass and PCS momentum equation are also developed. Coefficients in the dimensionless forms of these equations are referred to as energy, mass and momentum pi-groups, respectively. Dimensionless forms of the heat transfer relationships for the various heat sinks and the shell are also developed. Coefficients in these equations are referred to as "conductance pi-groups."

Pi-groups are evaluated for each phase of the AP600 LOCA and main steamline break in Section 8 of the scaling report. The pi-group values are then used to calculate dP/dt values. The scaling model dP/dt values are then compared to corresponding values calculated with a WGOTHIC model which is comparable to the scaling model. Some of the conservatism included in the WGOTHIC evaluation model was not included in this case to facilitate comparison to the scaling model. Values of dP/dt are presented for each phase of the AP600 LOCA. Reasonable agreement is shown between the scaling model and the WGOTHIC calculations.

21.6.5.5.3.3 Westinghouse Evaluation of Distortions in LST

Distortions in the LST data is the subject of Section 11 of the scaling report. Westinghouse states that the LST cannot be used as a direct simulation of AP600 either for the LOCA or for the main steamline break. Westinghouse presents arguments to support use of LST as a system level simulation for code validation and argued that the LST data could be used to examine the influence of lumped-parameter code biases. Westinghouse also discusses the use of LST to address component-level distortions.

The definition of distortion used by Westinghouse is based on differences in the values of pi-groups. Many of the distorted phenomena, for example, mixing and circulation, are not characterized by pi-groups. Nevertheless, all of the identified distortions, even those for which there is no associated pi-group, are addressed.

Seventeen LST distortions are listed along with descriptions of the effect of each distortion. The treatment of the distorted phenomena in the evaluation model is discussed, as is the effect of the distortion on the use of LST data. A brief summary of each distortion follows. The staff's evaluation of how each distortion affects the use of LST data is found in Section 21.6.5.5.4.3 of this evaluation.

Area to Volume Ratio

One of the more significant distortions, because it involves high ranked phenomena, is the area to volume ratio, which is eight times higher for LST than AP600. Westinghouse argues that this distortion does not affect the use of LST as a source of separate effects data.

Break Source Flow Rate

The LST was run as a steady-state test; that is, the energy addition rate from the break was approximately equal to the energy removal rate through the shell. During a transient, a considerable amount of energy in excess of the amount removed is added. This causes the pressure to increase. Since no excess energy was added in LST, the power to volume ratio was much lower than for AP600. Westinghouse argues that the break source flow rate was varied during the tests to cover a range of values typical of AP600, so for separate effects data, this is not a distortion.

Shell Thickness

Both the thermal resistance and heat capacity of the shell are affected by thickness. The LST shell is slightly over one-half the AP600 shell thickness. Westinghouse argues that this is not a distortion for separate-effects data.

Break Source Superheat

Westinghouse argues that, while the relative distortion is high, the absolute distortion is small and therefore this is not significant.

Diffuser Used for Break Source

The diffuser produces a low-velocity flow, whereas a high-velocity jet is expected during blowdown. This is a distortion during blowdown, but becomes less of a distortion for the long-term phase.

No Downcomer

LST used a fan to force flow in the riser. Pi-groups for the downcomer flow are stated to be sufficiently small that the downcomer flow does not significantly affect PCS performance. This is also a low-ranked phenomenon. Westinghouse argues that the downcomer is therefore adequately treated in the evaluation model as a flowpath with thermal and hydraulic interactions with the baffle and the shield building.

Fan Forced Riser Air Flow

Westinghouse argues that for separate-effects data, it does not matter whether the air flow is forced or induced by natural circulation.

Riser-Scaled 1/4

The LST riser was scaled 1/4 instead of 1/8 like the rest of the test. Consequently, the Reynolds number range for the separate effects data covers a range of Reynolds numbers up to about ½ of AP600. Westinghouse argues that the distortion has no effect over the range of Reynolds numbers covered and that the range of Reynold numbers above for the LST is adequately covered by other data sources.

No Circulation Below Deck

There was no connection between the steam generator compartment, where the break occurred, and the other below-deck compartments. Therefore, at the system level the circulating flow field was distorted. Westinghouse argues that since local conditions were measured, this distortion does not affect the use of LST for separate-effects data.

External Water Flow Too High

While many of the LST test cases had higher cooling water mass flow rates and higher subcooling than AP600, some tests covered the AP600 range. Therefore, Westinghouse argues that this is a ranged parameter rather than a distortion.

External Water Coverage Too High

As a consequence of the high PCS flow rates, the wetted fraction of the exterior LST surface was greater than would occur for AP600. While a given test may have a larger coverage fraction than a comparable AP600 case, the test series included a range of coverages including a dry shell. Therefore, water coverage was a ranged parameter.

External Water Flow Was Established Before Break

Two LST tests were run with water applied after the shell was heated. These tests are stated to provide separate-effects data to support validation of the external wetting and stability model.

External Water Flow Time Variations

Westinghouse argues that this is not a distortion because the reduced flow portion of the cycle was only a small fraction of the total and all but two of the tests had significantly more flow than could be evaporated. The separate-effects data was averaged over several cycles of the flow variation, the data was evaluated with both minimum and maximum flow rates and a conservative approach adopted, and data without flow fluctuations was used in the correlation comparisons. Therefore, it is argued that the flow fluctuations do not compromise the use of LST data in a separate-effects mode.

External Water Not Applied By Weirs

This is not considered to be a distortion because the method used to achieve water coverage gave distributions similar to the distribution expected for AP600.

Internal Heat Sinks Not Prototypic

On a system level, the non-prototypicality is stated to affect only the period between the blowdown and the long-term phase. Since local measurements were made, the distortion does not affect the use of LST data for separate-effects correlation validation.

Crane Rails Not The Same

Westinghouse argues that the lack of a structure in LST to simulate the crane rail is not a distortion because the quantity affected is the film conductance, which has less than a second order effect.

Condensate Drained Out

Draining the condensate to prevent filling up of the facility resulted in lack of a break pool. Since the pool was shown by the scaling analysis to be less than a second order effect, Westinghouse argues that this is not a distortion.

The above is a summary of the Westinghouse position on the LST distortions. The following section gives the staff's position on the scaling analysis and the use of LST data to support the scaling and WGOTHIC validation.

21.6.5.5.4 Evaluation of Westinghouse Scaling

In Westinghouse's use of the scaling analysis to support the PIRT rankings, it considered, among other factors, the magnitude of the RPC equation dimensionless coefficients as a measure of importance to aid in ranking phenomena. The use of scaling to support the PIRT is evaluated in Section 21.6.5.5.4.1 of this report.

In developing the RPC equation, relevant correlations and constitutive relationships were described and the validation data range was compared to that of AP600. This is essentially a bottom-up scaling approach that establishes the applicability of basic heat and mass transfer relationships used in both the scaling and evaluation models over the range of AP600. This aspect of the scaling evaluation is discussed in Section 21.6.5.5.4.2 of this report.

Use of LST as a source of separate-effects heat and mass transfer data is also addressed in the scaling study. LST data includes condensation and evaporation heat and mass transfer across the shell. It is used both to validate the combination of standard correlations chosen by Westinghouse and to establish conservative multipliers for the evaluation model. Westinghouse also argues that the LST data are useful for validating the WGOTHIC code in the short and long-term. The staff sees the essentially steady-state LST data as having a more limited role in validating the transient WGOTHIC evaluation model. The role of LST data in the scaling evaluation is discussed in Section 21.6.5.5.4.3 of this report.

21.6.5.5.4.1 Scaling Analysis to Support the PIRT

The scaling evaluation provides information to support PIRT rankings in the form of the pressure rate of change equation, which is derived from first principles and applied for two limiting scenarios, the double-ended cold-leg LOCA and the main steamline break. The dimensionless form of the pressure rate of change equation contains pi-groups whose relative magnitude is used to assist in ranking phenomena. The conservation of mass and conservation of energy equations are also placed into dimensionless form to yield mass and energy pi-groups, which are also used to assist in the development of PIRT rankings.

While the dimensionless RPC equation and the supporting dimensionless conservation of mass, conservation of energy, and heat structure conductance equations derived by Westinghouse are somewhat more complex than necessary for an adequate treatment, they contain the essential elements for ranking of phenomena. The staff does not agree with the choice of parameters used to non-dimensionalize the equations, but the choice does not, in most cases, adversely affect the value of pi-groups. In one case where the choice of non-dimensionalizing parameter resulted in a faulty pi-group value, Westinghouse identified the problem. In response to RAI 480.1017, and in a subsequent clarification of the response, Westinghouse evaluated additional pi-group values which could be affected by this same type of problem, and the evaluation did not reveal any additional anomalous pi-group values. Subsequently, Westinghouse revised the scaling report to eliminate the anomalous pi-group values.

Usually, pi-groups are the non-dimensional coefficients of the governing equations, normalized such that the largest coefficient has a magnitude equal to one. The values of the pi-groups are then in the range -1 to +1, and the magnitude can be used to judge the importance of the term for which a given pi-group is the coefficient. Westinghouse didn't follow this convention consistently, so the values of some pi-groups are slightly outside of this range. Given this, Westinghouse did not define a numerical value criterion for what constitutes the dividing line between high-, medium- and low-ranked phenomena. A subjective approach was used for ranking, which included consideration of expert opinion, test data, and sensitivity studies in addition to the scaling pi-group numerical values. While the range of numerical values for pi-groups is slightly outside of the conventional -1 to +1, this difference did not prevent use of the scaling results to support the PIRT ranking process.

Westinghouse calculated values for energy transfer conductances by normalizing all conductances with respect to the shell conductance. Numerical values of the conductances range from 0.01 to 202 (energy transfer to the drops) — see Table 8.2 in WCAP-14845. Values of the conductances can be used to determine the importance of the various heat transfer mechanisms.

Values of mass, energy and pressure pi-groups were also calculated for the four LOCA phases and the main steamline break. As noted earlier, the normalizations were not chosen to ensure that the values of all pi-groups were in the range -1 to +1. However, the largest values are on the order of 1.0. Values of the pi-groups were used by the staff to confirm closure between the scaling analysis and the PIRT. Basically, any phenomenon for which a pi-value greater than about 0.10 is calculated during any phase of the LOCA or main steamline break should be ranked as medium or high in importance, and this is indeed the case. While Westinghouse correctly used other input to assist in the ranking, the consistency of PIRT and scaling is an

essential element of the analysis. The staff accepts the scaling analysis performed by Westinghouse as an acceptable manner to provide information for use in development of the PIRT, and also notes that the information in the scaling analysis and the PIRT is consistent.

The pressure pi-group values reveal the overwhelming importance of two related phenomena, heat and mass transfer to the evaporating shell and to the structural heat sinks. An acceptable evaluation model must include these phenomena in a conservative manner and be applicable at the scale of AP600. These phenomena are thus given special attention in the staff's evaluation, especially the shell heat and mass transfer, since this aspect of the AP600 PCS design differs from present containment designs.

Pi-group values were also calculated by Westinghouse for the PCS air flow momentum equation during the LOCA phases and for the main steamline break. These pi-groups show that the downcomer has a very small effect on the buoyancy, except during the long-term LOCA phase, when the value of the pi-group is 0.16, compared to the combined riser and chimney pi-group value of 1.16. This value confirms the medium ranking given to phenomenon 13A (Table 21.6.5.5-1), downcomer annulus PCS natural circulation, during the peak pressure and long-term LOCA phases.

At the request of the staff in RAI 480.1030, Westinghouse included a comparison of the scaling model and WGOTHIC results for each phase of the LOCA scenario. The quantity compared was dP/dt as calculated by the scaling RPC equation and by WGOTHIC for a case without most of the evaluation model biases. The good comparison served as a confirmation that the scaling analysis is consistent with the WGOTHIC basic model, and vice versa.

21.6.5.5.4.2 Applicability of Correlations and Constitutive Relations

The new and unique aspect of the PCS is the use of the water-cooled shell to remove heat from inside containment. Other features, such as the structural heat sinks and the large pressure-reducing volume, are similar to existing designs for large dry containments. The AP600 containment conditions will tend to be less mixed in the long term, due to the lack of sprays and fan coolers, so stratified and spatially varying steam concentrations could be more prevalent. It is therefore essential to show that the evaluation model is conservative in its treatment of the shell and internal structures heat transfer at the scale of AP600.

The combination of correlations used to model condensation and evaporation heat and mass transfer across the shell were taken from the engineering literature. The correlations are based on local conditions and are therefore independent of scale. Similarity variables are used and the data base includes LST, where the complete phenomena across the shell were present and local variables were measured. Westinghouse developed conservative multipliers to apply to the condensation heat and mass transfer (0.73) and to the evaporation heat and mass transfer (0.84), as discussed in Section 21.6.5.6.5.3 of this evaluation.

The staff concurs that this model is appropriate for application to AP600. With the factors of conservatism applied by Westinghouse to obtain an evaluation model, the correlations for heat and mass transfer across the shell are acceptable, assuming that the local conditions at the shell wall and in the riser can be calculated in a conservative manner.

The containment atmosphere is considered to be "homogeneous" if the steam and non-condensable concentrations are uniformly distributed. "Well mixed" conditions are nearly "homogeneous," but may have small temperature gradients which drive natural circulation towards greater homogeneity. If the containment atmosphere approaches a well mixed condition, lumped-parameter models, including WGOTHIC, will provide a reasonable prediction of the bulk steam concentration. During the LOCA blowdown and MSLB blowdown the staff accepts that the break flow will be sufficient to mix the atmosphere, so that a well mixed assumption is appropriate. In this case, the shell heat and mass transfer model is acceptable at the scale of AP600.

For the LOCA, the containment will eventually stratify and will not be "well mixed." However, as long as WGOTHIC can predict the bulk steam concentration in the nodes adjoining the shell, the heat transfer model will be conservative and applicable at the scale of AP600. The lumped-parameter model characteristically gives conservative results for peak pressure because it does not include some of the mechanisms which drive the containment atmosphere to a homogeneous state, (e.g., molecular diffusion, turbulent mixing, and plume dynamics). In a lumped-parameter model a pressure gradient is needed to drive steam flow from the source to the condensing surfaces. While the mixing process is overpredicted within a lumped node, it is typically underpredicted between nodes, resulting in an overall conservative prediction of the steam concentration at condensing surfaces removed from the break. As discussed in Section 21.6.5.7.5 of this report, this conservatism is evident from lumped-parameter analysis of test data, such as HDR.

Open Item 21.5.8-5 concerned measurement of conditions in the annular region (i.e., the riser). The local-conditions correlation uses the sink temperature and the average velocity. These quantities were adequately measured in the LST to provide separate-effects data, so Open Item 21.5.8-5 is closed.

In Section 10 of the scaling report (WCAP-14845), Westinghouse shows that the database for the important heat and mass transfer correlations covers the range of AP600. Open Item 21.6.5-14 expressed the staff's concern regarding the potential use of correlations outside their range and the potential impact on AP600 licensing calculations. The information presented in Section 10 of WCAP-14845 closes Open Item 21.6.5-14.

The bottom-up, component-level evaluation of the individual phenomenological models included in the pressure rate of change equation is a key element of the scaling evaluation. A number of models, in addition to the shell heat and mass transfer model, are included. Similarity variables are used to characterize the constitutive relations and correlations. The use of similarity variables is important, since such variables include the effect of scale. That is, if the data which supports a correlation includes the range of similarity variables required for AP600, then the correlation is applicable for AP600.

Since the LST integral tests are distorted relative to AP600, arguments of applicability to AP600 based on top-down scaling are weak. Therefore, the demonstration of applicability of the evaluation model to AP600 hinges on the bottom-up scaling, and the demonstration of a conservative approach.

While scaled testing and validation of a best-estimate model using the scaled test data is a preferred approach to demonstrating that maximum pressure design criteria are met, there are other acceptable approaches. The staff accepts that an evaluation model which has been demonstrated to be conservative at the scale of AP600 is an acceptable approach to performing design basis analysis for the AP600 PCS..

The staff accepts that the heat and mass transfer correlations used in the scaling analysis are appropriate for AP600 because Westinghouse has shown that the database (see Section 21.6.5.6 of this report) covers the AP600 range of similarity variables. The staff also accepts, based on the information provided, that the WGOTHIC lumped-parameter model conservatively predicts the steam concentration conditions at the shell surface at the scale of AP600.

21.6.5.5.4.3 Use of LST Experimental Data to Support the Evaluation Model

Westinghouse has submitted documentation in WCAPs -14135 and -14326 which describe the LST tests and the use of the LST data to support the WGOTHIC evaluation model. This closes Open Item 21.3.8.1-1.

As discussed in Section 21.6.5.5.3 of this report, Westinghouse does not attempt to use scaled testing and WGOTHIC prediction of scaled tests to directly validate the WGOTHIC model. A conservative evaluation model approach is used, where Westinghouse attempts to show that the model is applicable at the scale of AP600, based upon correlations to experimental data, generally developed at smaller scales. The staff concurs that this approach is acceptable. The LST data is used in a separate-effects mode to validate the heat and mass transfer correlations used to model shell heat and mass transfer. The local-condition correlations are based on data from the literature, and have been compared to the LST data, which incorporates all of the shell heat and mass transfer phenomena (i.e., condensation of the inside and evaporation of the outside of the shell, in a single test). The range of variables was shown to adequately cover the range of AP600 (see Table 21.6-15 of this evaluation). The use of conservative correlations, where bounds were developed using LST data, assures that the important correlations in WGOTHIC will yield a conservative analysis for AP600.

The question of whether distortions in the LST can affect the usefulness of separate-effects data obtained from this facility was also addressed by the staff. Following is a discussion of each of the 17 specific distortions identified in Section 21.6.5.5.3.3 of this report and an evaluation of why the separate effects data are not significantly affected by the distortion and why LST can be used in a limited context to validate the WGOTHIC code. In virtually every case, the distortions affect the use of LST as an integral system test. The effects of distortions are significant enough to make the test not useful for this purpose. However, as a separate effects test, where only the local environment of the shell is involved, the effect of most of the system distortions is small or inconsequential.

Area to Volume Ratio

The shell surface area to containment volume ratio is eight times higher for LST than for AP600. This distortion could strongly affect the transport of steam from the source to the condensation surface. However, local conditions measured near the surface were made to obtain the separate-effects data. The correlation and the conservative multipliers developed from the data

are based only upon the measured local conditions, not on the global conditions in the containment. Similarity variables were used in the correlations. The range of the local measured variables covers the range of AP600. Whatever effect the distortions may have on the local conditions is irrelevant, since the conditions were measured and cover the range of AP600.

Break Source Flow Rate

The LST was run over a range of energy release (i.e., break source) conditions. Since the test was steady-state, the heat flux through the shell varied directly with the energy release rate. The separate-effects test conditions covered a range of shell heat fluxes which included the range for AP600. In a transient, there are times during the blowdown when the break source magnitude considerably exceeds the energy removed through the shell. While this mismatch between the break source and the shell heat flux causes the pressure to rise, it doesn't affect the nature of the heat flux through the shell. Use of steady-state separate-effects data to verify a correlation for use during a transient is an accepted practice which has proven to be widely applicable. Therefore, no distortion is involved in using the steady-state separate-effects LST data to validate and establish bounds for a correlation used during a transient.

Shell Thickness

The LST shell is slightly over one-half the thickness of the AP600 shell. Both the thermal resistance and heat capacity of the shell are affected by thickness. However, these items have a relatively low uncertainty. Furthermore, the internal heat transmission and energy storage phenomena associated with the shell metal are well-known functions of the thickness. The staff concurs that for separate-effects data, the shell thickness need not be identical.

Break Source Superheat

The use of the LST data in a separate-effects mode is independent of the break source superheat. Local steam concentration measured near the wall covered the range of AP600, in spite of differences in the break source enthalpy. The break source superheat is not included in the separate effects-data used. Therefore, this distortion is irrelevant.

Diffuser Used for Break Source

During a transient, the character of the break source changes from a jet during blowdown to a buoyant plume during the long-term phase. The steady-state LST tests simulated only the later phase of the event. On a system level this is a distortion, but it doesn't affect the separate effects-data, which was based on measured local conditions.

No Downcomer

For purposes of the separate-effects tests, the riser flow covered a range of Reynolds numbers. The range of conditions was covered without a downcomer, so the lack of a downcomer is not a distortion. When LST is used to obtain separate-effects heat and mass transfer data, the lack of a downcomer is not a factor since local conditions are used in the correlation. As a system test, the lack of a downcomer in LST is a distortion. Given the limited role of LST (i.e., as a source of

separate-effects heat and mass transfer data), the lack of a downcomer is not a significant issue. This closes Open Item 21.5.8-6.

Fan Forced Riser Air Flow

The staff agrees that for purposes of separate effects-data it does not matter whether the air flow is forced or induced by natural circulation. Given a local velocity in the riser, it cannot be determined whether the flow is driven by natural or forced convection.

Riser Scaled 1/4

Since the LST riser width was 1/4 scaled instead of 1/8 as was the rest of the test facility, the Reynolds number range for the separate-effects data covers Reynolds numbers up to about 1/2 of AP600. Over the range of Reynolds numbers covered, the distortion has no effect. Since Westinghouse uses a standard forced convection evaporation correlation, the staff accepts that the range of Reynold numbers not covered by LST is adequately covered by other data sources.

No Circulation Below Deck

The correlation validated by the separate-effects data does not depend on the circulation between the steam generator compartment, where the break occurred, and the other below-deck compartments. Therefore, this distortion does not affect the use of LST for separate-effects data.

External Water Flow Too High

The range of external water flow rates in the LST covers the range of AP600. Thus, the local conditions on the external shell surface were not distorted compared to AP600 and the data is acceptable for validating the steady-state correlation.

External Water Coverage Too High

The LST data was used to validate the total clime heat transfer package in a separate-effects mode. Local measurements were used, so water coverage fraction was not a variable used in validation of the clime package.

External Water Flow Was Established Before Break

The use of steady-state data to validate a correlation for use during a transient is an accepted practice which has proven to give accurate results. The change in local conditions on either side of the shell is slow enough for the heat and mass transfer processes at the shell surfaces to be regarded as quasi-steady-state.

External Water Flow Time Variations

While the flow fluctuations are an undesirable feature of the LST, their effect on the separate-effects data was minimized by the data analysis techniques used by Westinghouse. First, the flow did not cycle uniformly. Rather, a short reduced-flow phase recurred periodically. The reduced-flow phase of the cycle was only a small fraction of the total cycle. The

separate-effects data was averaged over several cycles of the flow variation, the data was evaluated with both minimum and maximum flow rates, and a data analysis approach adopted which minimized the measured heat and mass transfer. The flow fluctuations add some level of increased uncertainty and scatter to the data, but do not preclude the use of the data in a separate-effects mode. Westinghouse adjusted for the increased uncertainty by treating the correlation comparisons conservatively.

External Water Not Applied By Weirs

At the location where measurements were made to obtain the separate-effects data, the effect of the manner of application of the water film was dissipated. Therefore, this is not a distortion.

Internal Heat Sinks Not Prototypic

The separate-effects data was taken on the shell and not on the internal heat sinks. The non-prototypicality of the internal heat sinks was therefore not a factor from the separate-effects perspective.

Crane Rails Not The Same

The crane rail strips the liquid film from the inside shell wall, so that a new thinner film reforms below the rail. The conductance through the film is a minor component of the total thermal resistance from the inside to the outside of containment. While the crane rail can affect the local film thickness below the rail, the effect is included in the WGOTHIC model. At a system level, the lack of a crane rail is a minor distortion, but it does not affect the use of LST for separate-effects data.

Condensate Drained Out

Draining the condensate to keep from filling up of the facility does not affect the local conditions measured for the separate-effects data, since this process is physically remote from the measuring locations. Therefore, differences in the manner of accumulating and removing condensate do not distort the separate-effects data.

Westinghouse argues that the LST data can be used to qualify the WGOTHIC model in the short term and in the long term. Given the steady-state nature of the LST versus the transient nature of the AP600 pressurization, the significant number of distortions in the LST, and the lack of up-front scaling, the staff believes that the LST data is of little value in directly validating WGOTHIC from a system perspective. The prediction of LST steady-state data, however, provides evidence that the WGOTHIC coding of heat and mass transfer correlations for the shell is correct.

Resolution of the LST scaling issue was identified as Open Item 21.3.8.5-1. The treatment of non-condensables in the LST program was identified as Open Item 21.5.8-1. When the issues leading to these open items were raised by the staff, it was believed that LST would be used as a 1/8 scaled model of the AP600 for purposes of obtaining data for WGOTHIC code validation. Since that time a more limited use of the LST data has emerged. The LST is now being used to obtain separate-effects data for use in validating correlations available in the literature and for

placing conservative multipliers on those correlations. Use of LST data to validate the local-condition correlations does not require the same fidelity in all of the dimensions that would be required for a system level scaling. Therefore, issues such as the LST height of 6.1 m (20 ft) versus a scaled height of 7.2 m (23.7 ft) and differences of from 1 to 3 percent in the free volumes of various below deck compartments are no longer significant, in light of the more limited use of the LST data. Similarly, it is not necessary for the LST to produce measurements of the 3-D spatial distribution of non-condensables. The local measurement of non-condensable concentration is sufficient for the limited separate-effects data use. Therefore, Open Items 21.3.8.5-1 and 21.5.8-1 are closed. The potential effect of the resolution of these open items (and several other scaling-related open items) on the WGOTHIC code was identified as Open Item 21.6.5-25. Closing Open Items 21.3.8.5-1 and 21.5.8-1 in the manner described above restricts the use of WGOTHIC to conservative design basis analysis.

Open Item 21.5.8-3 was also related to LST measurements, in this case the containment internal velocity. Westinghouse use of LST data in a separate-effects mode does not use measurements of the local velocity field. The data is used to validate a local-condition turbulent-free convection correlation for heat and mass transfer, which does not involve velocity. Therefore, Open Item 21.5.8-3 is closed.

Open Item 21.5.8-4 concerned the measurement of condensate forming within the containment vessel. The spatial identity of the source of condensate cannot be determined with any degree of accuracy in LST. Condensate was collected into only two reservoirs from five locations. For the purpose of using LST as a system test to validate WGOTHIC this would be a major shortcoming. Given that the LST data are not being used for this purpose, lack of spatial distribution information on condensation is not a major concern. This closes Open Item 21.5.8-4.

21.6.5.5.5 Summary

Based on the foregoing, the staff has determined that the AP600 PCS PIRT identifies the significant phenomena important to the evaluation of the PCS and the AP600 containment pressure response to design-basis accidents.

The staff has determined that the AP600 PCS scaling analysis, combined with sensitivity studies presented in WCAP-14407, supports the ranking of the phenomena in the AP600 PCS PIRT. In addition, the scaling analysis verifies that the mass and heat transfer correlations used in the WGOTHIC computer program to represent the PCS are appropriate for evaluating the AP600 containment pressure response following a design-basis accident.

The scaling analysis also identified distortions in the LST that would make it difficult to use the data for direct comparison to the AP600. The LST distortions have been addressed by Westinghouse in an acceptable manner through the development of the conservative AP600 evaluation model. In view of the above, the staff has determined that the use of the LST to support the mass and heat transfer correlations used in the WGOTHIC computer program to represent the PCS is acceptable.

21.6.5.6 WGOTHIC Validation and Test Studies and Assessment

The advanced light-water reactor (ALWR) AP600 passive containment cooling system (PCS) design differs from past and current nuclear plant designs. This containment design concept represents one of the changes towards a simple, passively safe plant design as permitted by 10 CFR Part 52.

The AP600 plant design utilizes a passive containment cooling system (PCS) to transfer heat from the containment shell to the environment following an accident. The AP600 PCS is designed to remove sufficient heat from containment during the limiting design-basis accidents (DBA) to maintain containment pressure below the design limit.

The function of the PCS, in the AP600 design, is to provide a safety-grade means for transferring heat from the containment to the environment following postulated events that result in containment heatup and pressurization. The AP600 utilizes passive cooling of the free-standing steel containment vessel. Heat is transferred to the inside surface of the steel containment vessel by convection and condensation of steam and through the steel wall by conduction. Heat is then transferred from the outside containment surface by free convection to the air that enters an annular space around the steel containment shell. Cooling of the containment is enhanced by water distributed over the containment surface, heated, and then evaporated into the air stream. The heated air and water vapor rises by natural draft and exits the shield building through an outlet (chimney) located above the containment shell.

The performance of the AP600 PCS depends on the buoyant driving force of the cooling air, the air flow path pressure losses, the effective containment shell heat transfer coefficient, and the wetted PCS heat transfer area. Other factors that can influence PCS performance are wind conditions, nearby buildings and topography, inside-containment circulation patterns, water distribution patterns, and the effects of non-condensable gases inside the containment.

Key concepts and principles were evaluated by Westinghouse, and evolved into a definition of an overall matrix of testing necessary to obtain the data in support of the design. Where data was lacking, small basic research tests were conducted to demonstrate fundamental principles and feasibility of concepts. Based on these tests, larger and more sophisticated tests were designed to further evaluate the engineering and safety concepts of the design. The AP600 PCS large-scale test (LST) facility is one example of this approach. To assess the AP600 PCS heat removal capability, a testing program was prepared that included the following series of tests:

- AP600 PCS wind tunnel test (Sections 21.3.7, 21.4.7 and 21.5.7 of FSER)
- AP600 PCS water distribution test (Sections 21.3.9, 21.4.9 and 21.5.9 of FSER)
- AP600 heated-plate test (Section 21.6.5.6.2.4)
- AP600 small-scale containment cooling test (Section 21.6.5.6.6.1)

- AP600 large-scale passive containment cooling system test (Section 21.6.5.6.6.2)
- AP600 PCS air flow path pressure drop test (Section 21.6.5.6.6.3)

21.6.5.6.1 Overview of Experimental Data Base

The staff review started with preliminary studies of the heat and mass transfer correlations, documented in the following reports:

(a) "Experimental Basis for the Heat Transfer Correlations Selected for Modeling Heat Transfer from the AP600 Containment Vessel," PCS-GSR-004, Westinghouse Electric Corp., August 31, 1994.

(b) "Experimental Basis for the Mass Transfer Correlations Selected for Modeling Condensation and Evaporation on the AP600 Containment Vessel," PCS-GSR-006, Westinghouse Electric Corp., October 1994.

The initial series of requests for additional information (RAIs) was based on these reports (RAIs 480.356 to 480.368 on the heat transfer report and 480.369 to 480.375 on the mass transfer report). A WCAP report combining these preliminary reports was issued by Westinghouse:

(c) "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," WCAP-14326, Westinghouse Electric Corp., March 31, 1995.

That report addressed most of the staff concerns identified in the RAIs and brought about additional RAIs (480.404 to 480.406). A revision to WCAP-14326 was issued:

(d) "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," WCAP-14326, Revision 1, Westinghouse Electric Corp., May 1997.

Subsequent modification were required to address errors in Revision 1 (as noted in this review section) and to incorporate design changes into the report. WCAP-14326, Revision 3, April 1998, is the current version of this report.

The objective of the Westinghouse evaluation of the separate-effects tests was to validate the correlations that can be used to calculate energy transfer, by heat and mass transfer, between the containment atmosphere and the external PCS air flow path and between the PCS air flow path and the baffle, shield, and chimney. The correlations selected by Westinghouse represent the common phenomena of convective heat transfer and condensation and evaporation mass transfer.

The objectives of the study were to

- identify the appropriate correlations for the various heat and mass transfer regimes for the PCS surfaces
- compare these correlations to separate-effects tests that cover the range of dimensionless parameters expected for AP600 operation

- evaluate correlation uncertainties
- develop biases that can be applied to the correlations to bound the test data for use in AP600 licensing analyses, consistent with the conservative evaluation model

The correlations for heat and mass transfer developed for use in the AP600 evaluation model are defined consistently with the way energy transfer is modeled across the containment shell and in the PCS air flow path. The evaluation model energy transfer is calculated as follows:

- With condensation or evaporation, a liquid film is present. Energy is transported between the bulk gas and the liquid film free surface by radiation heat transfer, convection heat transfer, and mass transfer. Energy is transported by conduction through the liquid film to the solid surface.
- Dry surfaces do not have liquid films or mass transfer. Energy is transported between the bulk atmosphere and the solid surface by radiation heat transfer and convection heat transfer.
- The correlations assume the local bulk gas thermodynamic states are known both inside and outside containment. In the PIRT evaluation, WCAP-14812, phenomena that influence the distribution of bulk gas properties are separately evaluated to develop an overall conservative approach.

The AP600 PIRT, WCAP-14812, and scaling analysis, WCAP-14845, show that condensation inside containment and evaporation outside containment are the dominant high-importance transport phenomena for calculating containment pressure during DBA. Heat transfer inside and outside containment and conduction through the liquid film were identified to be of low-to-moderate importance, but require correlations since they are included in the evaluation model.

The McAdams free-convection and Colburn forced-convection heat transfer correlations are used by Westinghouse to model the PCS. The method recommended by Churchill is used to combine the free- and forced-convection correlations in the mixed-convection regime. The lower limit on the mixed-convection correlation for assisting free- and forced-flows is based on work by Eckert and Diaguila. The resulting single heat transfer correlation reduces to free-convection values at low Reynolds numbers (Re), forced-convection values at low Grashof numbers (Gr), and a combination of the two for mixed convection. The mass transfer correlation is derived from the Nusselt number (Nu) using the heat and mass transfer analogy. The Chun and Seban correlation for heat transfer by wavy laminar and turbulent conduction is used for both condensing and evaporating liquid films.

The expected operating range for the AP600 heat and mass transfer parameters are summarized in Table 21.6-15.

21.6.5.6.1.1 Heat Transfer in the AP600 PCS Air Annulus Region

The flow regime for turbulent convective heat transfer is typically qualified as either free, forced, or mixed. The combination of free and forced convection in the mixed regime is either assisting

(they work in the same direction, as in upward flow in a hot pipe) or opposed (they work against each other, as in downward flow in a hot pipe). Operating points for the AP600 air annulus region for the Grashof (Gr) and Reynolds (Re) numbers were calculated in the scaling analysis for the PCS air flow path (downcomer, riser, and chimney) and plotted on a Metals and Eckert diagram to determine the heat transfer regime. The comparisons indicate that the riser and downcomer will operate in forced convection and the chimney will operate in mixed convection. The convective heat transfer in the AP600 air annulus is expected to be turbulent rather than laminar, since the Reynolds numbers are all greater than 3,000.

Based on a review of the literature by Westinghouse, the turbulent-free-convection heat transfer correlation for gas mixtures has the form $Nu = C\,(GrPr)^N$, with the value of C varying between 0.09 and 0.15 and the value of N varying between 0.3 and 0.4. The McAdams correlation with G = 0.13 and N = ⅓ was selected by Westinghouse for calculating turbulent-free-convection heat transfer in the annulus:

$$Nu_{free} = 0.13(Gr_d Pr)^{1/3} \qquad \text{McAdams correlation (Eq. 21.6.5.6.1)}$$

This correlation is widely used to calculate turbulent-free-convection heat transfer from both vertical and inclined surfaces with either constant temperature or constant heat flux boundary conditions. The hydraulic diameter is the characteristic length in the Grashof (Gr_d) and Nusselt (Nu) numbers. Based on the experimental work of Vliet, the full gravitational acceleration is used by Westinghouse to evaluate the Grashof (Gr) number, not just the vector component parallel to the plate.

The Colburn correlation (Ref. 21.6.5.6.4) was selected by Westinghouse for calculating turbulent-forced-convection heat transfer in the annulus:

$$Nu_{force} = 0.023 Re_d^{4/5} Pr^{1/3} \qquad \text{Colburn correlation (Eq. 21.6.5.6.2)}$$

The Colburn correlation is applicable to both constant temperature and constant heat flux boundary conditions for fully developed flow in channels. The correlation is widely used to calculate turbulent-forced-convection heat transfer in long tubes and ducts. The hydraulic diameter is the characteristic length in the Reynolds (Re_d) and Nusselt numbers.

A length- or distance-dependent multiplier can be used to account for the increase in forced-convection heat transfer as the boundary layer develops at the entrance of a heated channel. This is an important consideration when modeling heat transfer in short channels. The entrance-effect multiplier is described in more detail in Section 21.6.5.6.1.2.

Based on the Metals and Eckert diagram shown in Figure 4-1 of WCAP-14845, the turbulent-opposed-mixed convection correlation, Equation 21.6.5.6.3, is used for the downcomer and chimney. Under opposed convection (downflow along a heated surface or upflow along a cooled surface), the mixed-convection correlation increases the value of the predicted Nu number over the value predicted using either the free- or forced-convection correlations alone.

The outside surface of the containment shell is expected to operate in turbulent assisted convection (upflow along a heated surface or downflow along a cooled surface) during a DBA.

A method for combining separate free- and forced-convection heat transfer correlations into a single correlation that covers free, mixed, and forced convection was recommended by Churchill and is used by Westinghouse. For turbulent opposed free and forced convection,

$$Nu_c = (Nu_{free}^3 + Nu_{force}^3)^{1/3} \qquad \text{(Eq. 21.6.5.6.3)}$$

and for turbulent assisting free and forced convection, Nu_c is the larger of the following three expressions:

$$[abs(Nu_{free}^3 - Nu_{force}^3)]^{1/3} \quad ; \quad Nu_{free} \quad ; \quad 0.75 Nu_{force} \qquad \text{(Eq. 21.6.5.6.4)}$$

Consistent with experimental data trends (laminar and turbulent) presented by Churchill, the sign between Nu_{free} and Nu_{force} is reversed in both of these equations relative to the formulations for laminar flow. The lower limit in the latter equation, which prevents the value of Nu_c from going to zero when Nu_{free} and Nu_{force} are equal, comes from Eckert and Diaguila.

As the angles of inclination approach horizontal, the assisting- and opposed-convection heat transfer coefficients should become equal. Although the correlations used for AP600 do not provide for this, it is addressed by Westinghouse in an acceptable manner for the AP600 in the evaluation model as follows:

- Only free convection is assumed inside containment, so the definition of mixed convection is not relevant inside containment.
- The downcomer and chimney have too little horizontal surface area and too little heat and mass transfer to be a concern.
- Below the first water distribution weir on the containment dome, the slope is greater than 30° so opposed and assisting convection are well defined. Above the first weir, the liquid film is subcooled. Therefore, with little or no evaporation and with the surface area less than 4 percent of the total shell areas, regions with shallow slopes are not a significant concern for the external containment shell.

21.6.5.6.1.2 Entrance Effects

The heat transfer coefficient at the entrance to a heated channel is significantly higher than the fully developed value predicted by the Colburn forced-convection heat transfer correlation. The increase in heat transfer at the entrance is attributed to the thinness of the boundary layer that develops with distance from the entrance. The entrance effect is important for modeling heat transfer in short channels, and is used by Westinghouse for the test data comparisons. Since the net entrance effect for the long AP600 riser channel is only a small increase in heat transfer,

Westinghouse concludes that entrance-effect multipliers may reasonably be neglected for AP600 licensing calculations (see Section 21.6.5.6.1.2.2).

21.6.5.6.1.2.1 Entrance Effects for Use in Separate-Effects Tests Evaluations

The entrance-effect correlation and coefficients used by Westinghouse to assess the short-channel experiments are those recommended by Boelter, Young, and Iverson. Entrance effects are not appropriate for, and are not applied to, free convection or the free-convection portion of the mixed-convection heat transfer correlations.

The heat and mass transfer correlations calculated with entrance effects are compared to results for each of the five separate-effects heat transfer tests (see Sections 21.6.5.6.2.1 to 21.6.5.6.2.5) and to the three separate-effects mass transfer tests (see Sections 21.6.5.6.3.1 to 21.6.5.6.3.3). The comparisons show that the heat or mass transfer coefficients, as represented by the Nusselt (Nu) and Sherwood (Sh) numbers, are under-predicted by 2 to 14 percent in six of the data sets, overpredicted by 3 percent in one (Eckert and Diaguila tests, Section 21.6.5.6.2.2), and overpredicted by 18 percent in one (Hugot tests, Section 21.6.5.6.2.1). The overprediction in the Hugot tests is reduced to 10 percent if the heat transfer at $x/d_h < 1.0$ is not included in the comparisons. The multipliers become large and increasingly uncertain for $x/d_h < 1.0$. These comparisons show that, overall, the entrance-effect multipliers improve the agreement between the test data and the analytical heat or mass transfer predictions for these short channel tests.

21.6.5.6.1.2.2 Entrance Effects for Use in the AP600 Evaluation Model

The AP600 riser channel differs from the test geometries because of the 1.8 m (6 ft) well, or turning region at the bottom of the baffle. For modeling simplicity Westinghouse uses a fully developed heat transfer coefficient over the full channel height. An evaluation by Westinghouse showed that the heat transfer decrease (relative to fully developed heat transfer) is more than offset by the heat transfer increase due to neglecting the entrance effect in the channel above the well, resulting in a conservative model for use in the AP600 licensing analyses.

Heat Transfer in the Well Region Below the Baffle

The annular duct created by the baffle for the AP600 starts 1.8 m (6 ft) above the bottom of an annular well. This well is about 1.4 m (4.5 ft) wide and is heated on the inside surface. In the AP600 evaluation model it is assumed, for simplicity, that the forced-convection heat transfer correlations used in the annular region can be applied within this region as well. It would be more realistic to assume a free-convection heat transfer relationship on the heated containment shell side of the well region.

In WCAP-14326, Revision 1, Westinghouse used a 0.9 m (3 ft) wide well for this evaluation, rather than 1.4 m (4.5 ft). Westinghouse confirmed that there was an error in the report and that it would be corrected; however, the overall conclusion regarding the treatment of this region in the AP600 evaluation model remains unchanged. Westinghouse has provided a revision to the report to correct this error and other errors identified by the staff (Westinghouse letter NSD-NRC-97-5338, dated September 22, 1997).

Although the upper half of the 1.8 m (6 ft) height may undergo transition to turbulent free convection, the laminar free convection correlation is used for AP600 licensing analyses. That model predicts lower heat transfer coefficients. The effect of using forced convection in the 1.8 m (6 ft) well is evaluated by comparing the total heat transfer calculated with laminar free convection in the well to the total heat transfer calculated with forced convection everywhere.

The empirical formula of McAdams was chosen for the laminar free convection mean Nusselt number, and is a function of the distance-dependent Rayleigh number (Ra_x):

$$\overline{Nu}=0.555(Ra_x)^{1/4} \qquad \text{(Eq. 21.6.5.6.5)}$$

The Nusselt number for forced flow convection used is the Colburn relationship (Eq. 21.6.5.6.2).

The evaluation was performed for both wet and dry containment surfaces at temperatures between 52 °C (125 °F) and 96 °C (205 °F) for annular flow velocities of 0.3, 2.1, and 6.1 m/sec (1, 7, and 20 ft/sec). The air temperature was set at 46 °C (115 °F), consistent with the evaluation model used for the licensing analyses. The maximum effect is expected to be a 5.6-percent reduction in the net heat transfer from the shell due to the assumed laminar heat transfer below the baffle.

Entrance Effects in the Riser Annulus

The heat transfer enhancement due to developing thermal profiles model selected by Westinghouse is based on eigenvalue solutions from Hatton and Quarmby for the developing thermal distribution within a hydrodynamically developed flow in an annulus. While the analytical solutions are quite complex, charts have been presented for enhanced heat transfer for Reynolds numbers of 7,100, 73,600, and 495,000 at Prandtl numbers of 0.1, l.0, and 10.0. The AP600 Prandtl number is very nearly unity and the riser Reynolds number ramps to 189,000. For reference, a velocity of 2.1 m/sec (7 ft/sec) yields a Reynolds number of about 70,000.

The evaluation showed that on the average, for a Reynolds number of 73,600 and a Prandtl number of 1.0, the heat transfer over the first 18.3 m (60 ft) of the annulus will exceed the fully developed value by 14.7 percent. The average heat transfer coefficient increase over the 29.3 m (96-ft) length is 7.9 percent. The same calculation for a Reynolds number of 7,096 develops a heat transfer increase of 8.7 percent, and at 495,000 a 10.8-percent increase. The reason for the increase at the higher Reynolds number is that the thermal profile does not become fully developed in the 27.5 m (90 ft) height of the annulus region in the AP600.

Conclusions

The heat transfer enhancement of 8 to 11 percent due to the entrance effect would more than offset the heat transfer degradation of approximately 6 percent due to free convection in the well.

The reduction and enhancement calculations are conservative for the following reasons:

- The presence of a turbulent eddy within the well region will disrupt the free-convection boundary layer and increase the heat transfer

- Any deviation of the velocity profile from the profile for fully developed turbulent conditions at the entrance to the annulus will also increase the heat transfer.

The calculations show it is conservative to neglect the free convection below the baffle and the entrance effects in the AP600 riser channel, and simply use a heat transfer correlation for fully developed turbulent flow over the full height from the bottom of the well to the first weir.

As discussed in Section 21.6.5.6.6.4, a design change to relocate the upper-annulus drains from the upper-annulus floor to a position about 0.3 m (1 ft) above the floor was evaluated by Westinghouse to confirm that the entrance treatment in the WGOTHIC evaluation model was unaffected.

21.6.5.6.1.3 Heat Transfer Inside Containment

Heat is transferred from the containment atmosphere to the containment inner shell surface by condensation, radiation, and convection. The AP600 containment calculations assume condensation and convective heat transfer take place at the outer surface of a thin liquid film that develops on the inside surface (the condensate film) of the containment vessel. The liquid film provides a relatively small, additional resistance to heat transfer from the containment atmosphere to the wall. Heat transfer through the liquid film is characterized by the film Reynolds and Prandtl numbers.

On the basis of scaling studies by Westinghouse (see WCAP-14845) the inside of the containment shell is expected to experience a high-velocity flow of steam and air during the MSLB event and during the blowdown phase of a large LOCA event as the break jet vigorously circulates the gas. The heat and mass transfer during this period are expected to be turbulent forced or mixed convection. After the LOCA blowdown is complete, the atmosphere is circulated less vigorously and the velocity of the steam and air flowing along the inside surface of the containment shell will be lower. This indicates that turbulent-free-convection heat and mass transfer is appropriate after blowdown. The inside of containment is conservatively modeled using turbulent-free convection throughout both the MSLB and the LOCA transients for AP600 licensing analyses.

The height-based Grashof number representing the lower limit for turbulent-free-convection heat transfer is approximately 10E+10. After the first few seconds of the transient, the height-based Grashof number is greater than 10E+10 over all but the lower 1.8 m (3 ft) (or less) of the interior shell surface. Since the turbulent-free-convection heat transfer correlation underpredicts laminar free convection heat transfer, its use is conservative over the lower 1.8 m (3 ft).

The McAdams correlation was selected for calculating turbulent-free-convection heat transfer inside containment. The correlation can be written as a function of local properties:

$$h_{free}=0.13\frac{k}{L}(Gr_L Pr)^{1/3}=0.13\frac{k}{(v^2/g)^{1/3}}[\frac{\Delta\rho}{\rho}]^{1/3}Pr^{1/3} \qquad \text{(Eq. 21.6.5.6.6)}$$

where h is the heat transfer coefficient
k is the thermal conductivity
v is the kinematic viscosity
g is the gravitational acceleration
$(v^2/g)^{1/3}$ has units of length

The term $(\Delta\rho/\rho)$ is the difference between the bulk density and the surface density, divided by the bulk density. The term $(v^2/g)^{1/3}$ has units of length and is used in the scaling analysis, WCAP-14845. Consistent with Vliet, g is not reduced by the sine of the slope from the horizontal.

21.6.5.6.1.4 Liquid Film

The AP600 containment calculations assume the liquid film is a distinct control volume with mass transfer, convection heat transfer, and radiation heat transfer into the free surface, and conduction to the solid surface. Heat is transferred through the thin films on both the inside and outside of the containment shell. The Chun and Seban correlation is used by Westinghouse to model both wavy-laminar and turbulent heat transfer across the film. For wavy laminar films,

$$Nu=0.822Re^{-0.22} \qquad \text{(Eq. 21.6.5.6.7(a))}$$

For turbulent films (with Re > 5800 $Pr^{1.06}$):

$$Nu=0.0038Re^{0.40}Pr^{0.65} \qquad \text{(Eq. 21.6.5.6.7(b))}$$

21.6.5.6.1.5 Mass Transfer Inside and Outside Containment

Convective mass transfer is a result of a concentration gradient between a flowing steam-air gas mixture and a surface. The steam concentration gradient is approximated as the difference in steam partial pressure between the bulk gas and liquid surface. Condensation occurs when the bulk gas steam concentration is greater than the concentration at the surface of the liquid. Evaporation occurs when the bulk gas steam concentration is less than the concentration at the

surface of the liquid. Westinghouse uses the definition of the steam mass flux between the surface and the bulk gas, based on work by Kreith:

$$m''_{stm} = k_g M_{stm}(p_{stm,srf} - p_{stm,bulk}) \qquad \text{(Eq. 21.6.5.6.8)}$$

where m''_{stm} is the condensing or evaporating mass flux
k_g is mass transfer coefficient
M_{stm} is the molecular weight of the steam
$p_{stm,srf}$ is the steam partial pressure at the interface
$p_{stm,bulk}$ is the steam partial pressure in the bulk gas mixture

The mass transfer coefficient, K_g, can be predicted using empirical correlations similar to those for the convective heat transfer coefficient, h_c. The Sherwood(Sh) number for mass transfer is analogous to the Nusselt number for heat transfer, and is derived from the Nusselt number using the heat and mass transfer analogy:

$$Sh = \frac{Nu}{(Pr/Sc)^{1/3}} \qquad \text{(Eq. 21.6.5.6.9)}$$

where Sc is the Schmidt number

The mass transfer coefficient for gas-phase mass transfer is defined as:

$$k_g = \frac{h_c P D_v}{R T P_{lm}} \left(\frac{Sc}{Pr}\right)^{1/3} \qquad \text{(Eq. 21.6.5.6.10)}$$

where P is the total pressure
D_v is the air-steam diffusion coefficient
R is the universal gas constant
T is the absolute boundary layer temperature, $(T_{surf} + T_{bulk})/2$
P_{lm} is the log mean partial pressure of the air,
$(p_{air,bulk} - p_{air,surf})/\ln(p_{air,bulk}/p_{air,surf})$

P_{lm} accounts for the change in heat transfer at high mass transfer rates. The Nusselt number is based on the heat transfer correlation evaluated at the boundary layer temperature. The properties in the Prandtl and Schmidt numbers are evaluated at the boundary layer temperature. Equation 21.6.5.6.10 is used to calculate both condensation and evaporation mass transfer. Boundary layer properties are evaluated at the mean of the bulk and surface conditions.

21.6.5.6.1.6 Thermal Properties

All of the thermal properties used in the heat and mass transfer correlations are represented by correlations having an estimated accuracy of 1 percent with the exception of the air-steam diffusion coefficient. The condensation and evaporation mass transfer rates calculated for the AP600 are linearly proportional to the air-steam diffusion coefficient. The diffusion coefficient correlation that is used for all AP600 and test comparison mass transfer rates, overpredicts the measured diffusion data from the literature, and hence, overpredicts the mass transfer rates by approximately 10 percent over the AP600 containment temperature range of approximately 38 °C (100 °F) to 149 °C (300 °F). However, Westinghouse uses a mass transfer bias factor for AP600 licensing analyses that offsets the diffusion correlation bias. The bias is developed from comparisons of WGOTHIC computer program predictions to test data.

The diffusion coefficient correlation used by Westinghouse is from Eckert and Drake (Table B-9, p. 787). The Eckert and Drake correlation was compared to three data sets: Kestin, Rohsenow, and Eckert and Drake data. Westinghouse concluded from these comparisons that the model overpredicts the air-steam diffusion coefficient by approximately 10 percent. A comparison of the Eckert and Drake correlation to the theoretical development presented by Bird, Stewart, and Lightfoot shows the diffusion coefficient is proportional to 1/P. Although data were not included at higher pressures to support 1/P, the references agree on the expected 1/P pressure dependence. The correlations all give the temperature dependence to be T^n, where n is greater than 1.5. The theoretical development of Bird recommends temperature exponents of 2.334 for water vapor diffusing through a non-polar gas, and 1.823 for two non-polar gasses (water is a polar gas and air is non-polar). However, the Eckert and Drake value of n = 1.81 in the correlation appears to represent the measured temperature dependence very well. Westinghouse concluded that the correlation properly represents the diffusion coefficient sensitivity to temperature change. In view of the above, the staff accepts the Eckert and Drake correlation for evaluating this diffusion coefficient.

21.6.5.6.2 Separate-Effects Heat Transfer Tests

21.6.5.6.2.1 Hugot Mixed-Convection Heat Transfer Tests

The Hugot mixed-convection heat transfer tests were conducted on a set of symmetrically heated, parallel, vertical, isothermal plates with closed sides. The channel width was 1.0 meter (3.28 ft), the channel height was 3.3 m (10.83 ft), and the plate separation distance was variable at 10 and 60 cm (3.94 to 23.62 in). The plate temperatures were varied between 40 and 160° C (104 to 320° F).

A table summarizing the test data, including values of a Grashof number, Gr_δ and a Reynolds number, Re_d, was provided in an earlier version of the report. It was not clear how these were defined, specifically, how the characteristic length was defined. Usage generally implies that it should be based on the plate spacing or hydraulic diameter, d_h, but Gr_d did not show the expected variation with L/d_h assuming L is the same in all tests (as is implied in the text). The ratio of predicted to experimental local values of the Nusselt number, Nu, was plotted against a non-dimensional height, x/d. The local Nu was considerably overpredicted for the smallest x/d values, suggesting that the entrance-effect treatment was inappropriate for the experimental configuration.

In response to RAI 480.356, Westinghouse defined the characteristic length as the hydraulic diameter, d_h. This closes SDSER Open Item 21.6.5-22a. In response to RAI 480.357, Westinghouse incorporated a model to account for entrance effects in evaluating the experimental data (see Section 21.6.5.6.1.2). The entrance-effect multipliers are not used in the AP600 containment evaluation model.

The Hugot report presented the local heat transfer coefficient, but did not report the air flow rate or velocity induced in the heated channel; therefore, it was necessary to use a computer model to calculate air flow rates as well as heat transfer. The tests were modeled by Westinghouse using the WGOTHIC code with nominal inputs. As stated in WCAP-14407, WGOTHIC calculated the buoyancy-induced air velocity, air temperature, and heat transfer coefficients. The calculations assumed a combined entrance and exit form loss of 1.5. Since the air flow rate was calculated and the channel loss coefficient was estimated to be 1.5, the heat transfer calculation includes the effect of uncertainties on the air flow rate.

The Nusselt number, Nu, is defined as $Nu= hd_h/k$, where d_h is the channel hydraulic diameter. The entrance-effect multipliers were calculated as discussed in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. Assisting-mixed-convection heat transfer for moderate Reynolds and Grashof numbers was validated by the test data. Gr_d ranges from 2.4E+09 to 4.7E+09, and Re_d ranges from 1.1E+04 to 3.54E+04. During a DBA, Westinghouse estimates that the AP600 riser values of Gr_d can be as high as 1.2E+09 and the Re_d values can be as high as 1.9E+05.

21.6.5.6.2.2 Eckert and Diaguila Mixed-Convection Heat Transfer Tests

Eckert and Diaguila conducted heat transfer tests on a vertical tube that was 4.1 m (13.5 ft) high with a 59.1 cm (23.25 in) inside diameter. Inlet and outlet air pipes and dense screens were located at each end. A 3 m (10 ft) steam jacket supplied slightly superheated steam as the heat source. Sixteen condensation chambers collected and piped condensate to a station where the flow rate was measured and the local heat flux was determined. An air flow at approximately 27 °C (80 °F), at pressures from 1 atmosphere to 680 kPa (99 psia), was forced through the test section. Tests were conducted with forced flow in both the upward (assisting mixed convection) and downward (opposed mixed convection) direction. Thermocouples at the tube center and in the tube wall provided a temperature difference from which the local heat transfer coefficient could be determined.

Westinghouse mentioned the dense screens on the inlet and outlet but provided no other information. In response to RAI 480.358, Westinghouse stated that "it is assumed that the dense screens were installed to break up the velocity profile at the entrance of the heated channel. In this way, both the temperature and velocity profiles would develop simultaneously." The response was not consistent with the Eckert and Diaguila paper. The reference paper states: "To assure that the air entered the tube properly with a constant velocity, dense screens *E* were installed at the top and bottom of the tube." Westinghouse provided a revision to the report to correct their statement and other errors identified by the staff (Westinghouse letter NSD-NRC-97-5338, dated September 22, 1997).

As in the Hugot evaluation, the entrance effects were overpredicted in the original version of the report. Westinghouse incorporated a model to account for entrance effects for the purpose of

evaluating the experimental data (see Section 21.6.5. 6.1.2). The entrance-effect multipliers are not used in the AP600 containment evaluation model.

The Nusselt number, Nu, is defined as Nu = hd_h/k, where d_h is the hydraulic diameter. Entrance-effect multipliers were calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. The mixed convection Nusselt numbers were calculated as described in Section 21.6.5.6.1.1. The test data were used to validate the mixed-convection heat transfer correlation at prototypic Reynolds and Grashof numbers. For the tests, the Gr_dPr value range is 6.9E+09 to 7.2E+10 and the Re_d range is 3.6E+04 to 3.8E+05. During a DBA, the AP600 riser values of Gr_d can be as high as 1.2E+09, and the Re_d can be as high as 1.9E+05.

21.6.5.6.2.3 Siegel and Norris Mixed Convection Heat Transfer Tests

Siegel and Norris conducted heat transfer tests on a set of symmetrically heated, parallel, vertical flat plate channels. The channel width was 1.346 m (4.417 ft), the channel height was 1.778 m (5.833 ft), and the plate separation distance ranged from 3.8 to 38 cm (0.125 to 1.25 ft). A constant uniform heat flux of approximately 3500 W/m^2 (1100 Btu/hr-ft^2) was applied.

The L/d_h ratio ranged from 3 to 24. Convection was treated as assisted mixed convection. The predicted Nu matched the experimental values fairly well at low L/d_h but increasingly underpredicted Nu as L/d_h was increased. Four tests were performed at constant L/d_h to study the effect of progressively increasing the loss coefficient from 1.5 to 35.6; however, no information was given as to how this was done. Nu was increasingly underpredicted as flow was reduced.

In response to RAI 480.359, Westinghouse explained how the loss coefficient was changed during the test by adding extensions to the bottom of the test section. This closed SDSER Open Item 21.6.5-22c. The effects of reduced air flow were also investigated by adding extensions to the bottom of the test section channel and successively decreasing the lateral area for flow into the test section. Only those tests that had the test section open at the bottom were examined for comparison.

Since the air flow rate was not given, the tests were modeled using the WGOTHIC computer program. WGOTHIC calculated the velocity, air temperature, and heat transfer coefficients. The effects of reduced air flow were analyzed by relating the flow area reduction in the test to an increase in the inlet loss coefficient.

The Nusselt number, Nu, is defined as Nu = hd_h/k, where d_h is the hydraulic diameter. Entrance-effect multipliers were calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. The predicted Nu matched the experimental values fairly well at low L/d_h but still increasingly underpredicted Nu as L/d_h was increased. The calculated Nu also increasingly underpredicted the measured values as the air flow was reduced. The mixed-convection Nusselt numbers were calculated as described in Section 21.6.5.6.1.1. The tests generated data that validated the assisting-mixed-convection heat transfer model for low Reynolds numbers and moderate Grashof numbers. For the tests, the Gr_dPr ranged from

6.43E+05 to 6.1E+08, and Re_d ranged from 1.65E+03 to 1.13E+04. During a DBA, the AP600 riser values of Gr_d can be as high as 1.2E+09, and the Re_d can be as high as 1.9E+05.

21.6.5.6.2.4 Westinghouse STC Dry Flat-Plate Tests

The Westinghouse dry flat-plate tests, WCAP-12665, which were performed at the Westinghouse Science and Technology Center, provided heat transfer data for channels with heat flux and cooling air flow rate representative of the AP600 air riser annulus during a DBA.

The test section was a vertical, 1.8 m (6 ft) long, heated flat steel plate that had been coated with the highly wettable, inorganic-zinc coating used on the AP600. A clear acrylic cover provided a channel 58 cm (23 in) wide and 10 cm (4 in) deep for the forced air flow. The plate temperature and air flow rates were varied for each test.

The Nusselt number, defined in terms of the channel hydraulic diameter, was used for the data comparison. A length-averaged, entrance-effect multiplier of 1.13 was calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. The mixed-convection Nusselt number was calculated as described in Section 21.6.5.6.1.1. The data were compared with the mixed-convection correlation and shown as a function of the Reynolds number. Since these tests were dominated by forced convection, the results correlate well with the Reynolds number. The Re_d range was 3E+04 to 1E+05.

Westinghouse evaluated the measurement uncertainties for the dry-flat-plate test facility. The uncertainty in the Nusselt number is about 20 percent with higher uncertainty for lower heat flux tests. The uncertainty reduces to about 9 percent for test with wall heat fluxes greater than 3200 W/m^2 (1000 Btu/hr-ft^2).

21.6.5.6.2.5 Westinghouse Large-Scale Dry External Heat Transfer Tests

A series of heat transfer tests was performed at the large-scale test (LST) facility at the Westinghouse Science and Technology Center. The purpose was to compile data for developing and validating the analytical heat transfer models for use in the WGOTHIC computer program. Circumferentially averaged, external heat transfer data were determined from the dry LST data. The LST facility is described in Section 21.6.5.6.6.2.

The dry heat transfer tests were performed over a range of internal test vessel pressures that bounded the AP600 containment design pressure to obtain heat transfer data at prototypic conditions and to characterize heat transfer over a range of air cooling velocities.

Data that varied with time, angular position, and elevation were collected for each test. Nusselt numbers were calculated from the data using measured surface-to-bulk gas temperature and heat fluxes that were averaged over time and circumferentially averaged at each measuring elevation. Bulk gas temperatures in the annulus were not measured at each elevation where surface temperature and heat flux were measured, so the gas temperature was interpolated from values at the next higher and lower elevations.

The steady-state, circumferentially averaged heat transfer data from 14 of the 16 dry LST tests were used to define hydraulic diameter-based Nusselt number values. Entrance-effect

multipliers were calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. The Nusselt number values were compared with predictions of the turbulent-mixed-convection correlation as described in Section 21.6.5.6.1.1. (Tests RC015 and RC016 were omitted from this comparison because the forced asymmetric annular air flow rate imposed for these tests affected the circumferential averaging.)

The wall heat fluxes for dry LST tests were typically very small and the measured ΔT's across the wall were of the same order of magnitude as the ΔT measurement uncertainty. Therefore, the uncertainty in the Nusselt number measurements is high, greater than 100 percent for the lowest heat flux tests (320 W/m^2 (100 $Btu/hr\text{-}ft^2$)).

21.6.5.6.2.6 Summary of Heat Transfer Separate-Effects Tests

Experimental comparisons were initially presented in a way that needed more explanation. All the comparisons are for channel geometry (heated vertical parallel plates or pipe geometry). The comparisons were plots of calculated and experimental local Nu values versus a dimensionless distance, x/d. The local Nu values increased approximately linearly as a function of x/d and would therefore yield heat transfer coefficients approximately independent of distance if used in a relation of the form h = kNu/x. This is the expected behavior, except near the entrance, the heat transfer coefficient should be approximately independent of distance down the channel. For channels, however, it is more usual to define the controlling non-dimensional numbers in terms of a width or hydraulic diameter (d), and evaluate h from h = kNu/d; thus defined, these Nusselt numbers should be independent of x/d, except for entrance effects. Clarification was provided by Westinghouse and the comparisons were replotted in terms of Nu_d. The new presentation clarifies that this convention is being used for the channel geometries; e.g., Reynolds and Grashof numbers, (Re_d and Gr_d) based on the channel hydraulic diameter.

Entrance effects were included in the evaluation of the separate-effects tests. However, for the AP600 licensing analyses, a simplified model is used which does not include entrance effects. Studies performed by Westinghouse, as discussed in Section 21.6.5.6.1.2.2, conclude that this simplified model is conservative. The staff accepts this simplification for the AP600 evaluation model.

The combined convection heat transfer data consists of the Hugot, Eckert and Diaguila, Siegel and Norris, Westinghouse flat plate (WCAP-12665), and Westinghouse dry large-scale tests. The predicted-to-measured Nusselt number ratio was calculated from these data and Equation 21.6.5.6.4 for opposed mixed convection as a function of the Reynolds number and as a function of the Grashof number. The mean predicted-to-measured Nusselt number ratio is 0.976 with a standard deviation of 0.278. The mean predicted-to-measured Nusselt number value near 1.0 indicates that the heat transfer correlation fits the measured data very well. The large standard deviation is believed, by Westinghouse, to result from poor fidelity in the data for the following reasons:

- The convective heat transfer correlation serves as the basis for the prediction of condensation and evaporation mass transfer. Since the mass transfer data do not show large scatter, the variation in the heat transfer data may be attributed to more uncertain data measurements.

- The deviation between predicted and measured Nusselt numbers was large in four of the Hugot tests. The entrance-effect multiplier overpredicted the Nusselt number at small distances from the channel entrance ($L/d_h < 1.0$) because of the asymptotic singularity at $x = 0$ in the entrance-effect relation.

- The LST dry heat transfer test data have an uncertainty in the measured wall heat flux (ΔT) that is as large as or larger than the value of ΔT.

- The Eckert and Diaguila data have a large variation that changes with distance because the tube centerline temperature is used to represent the bulk temperature.

- The Hugot and the Siegel and Norris tests may exhibit higher deviations due to the use of a predicted, rather than measured, test air flow rate.

During a DBA event, the riser Reynolds number can be as high as 1.9E+05 and the riser Grashof number can be as high as 1.2E+09. The convection test data covered a Reynolds number range up to 5E+05, and a Grashof number range up to 1E+11. Therefore, the test data cover the expected range of both dimensionless groups within the annulus.

Westinghouse concluded that Equation 21.6.5.6.4 provides an adequate mean prediction of the dry-assisting-mixed-convection heat transfer for the AP600 vertical wall and dome. The test data encompass the expected range of AP600 Reynolds and Grashof numbers. Since the phenomenon is not ranked high in the PIRT, it is unnecessary to bound the test results in the evaluation model. However, the multiplier developed for mass transfer is applied to convective heat transfer for use in AP600 licensing analyses.

Westinghouse uses a conservative approach for licensing analyses. Part of this conservative approach is a bias on the heat transfer correlation to account for uncertainties in the use of the separate-effects tests to qualify the heat transfer correlation. In addition, inside containment, only free convection is assumed in the licensing analyses.

Westinghouse has responded to staff concerns, as identified in RAIs 480.360 to 480.368, about the separate-effects heat transfer tests and the heat transfer correlation used for the AP600 licensing analyses. Therefore, SDSER Open Item 21.6.5-22d is closed. SDSER Open Item 21.6.5-22b is closed since Westinghouse revised the text for the Eckert and Diaguila paper in WCAP-14326, Revision 2, to properly discuss the dense screens.

21.6.5.6.3 Separate-Effects Mass Transfer Tests

21.6.5.6.3.1 Gilliland and Sherwood Evaporation Tests

Isothermal evaporation mass transfer rates were measured in a vertical pipe by Gilliland and Sherwood. A water film was applied to the inside wall of the pipe and the evaporation rate was measured for both countercurrent and concurrent flow.

These tests studied the evaporation of downward-flowing liquid films on the inside of a vertical tube 1.17 m (3.84 ft) high and 0.0267 m (1.05 in) inside diameter. Liquid and air were at approximately the same temperature (i.e., the tests were approximately isothermal, within 3 °C (5.4 °F)). Temperatures were relatively low, 25 – 56 °C (77 to 133 °F) and vapor mole fractions

were therefore low. Since the liquid and air temperatures were nearly the same (isothermal) in these tests, the evaporative mass transfer was driven by the difference in partial pressure between the liquid film surface and bulk mixture. Under these conditions, buoyancy effects were presumably minimal. Reynolds numbers were 2E+03 to 2.5E+04, with values under 1E+04 in the majority of cases.

The Nusselt number and Sherwood numbers were evaluated using the channel hydraulic diameter. Entrance-effect multipliers were calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model.

In WCAP-14326, Revision 1, the x-axis was not labeled on Figure 3.6-2. Westinghouse identified the parameter as the Reynolds number and provided a revision to the report to correct this error and other errors identified by the staff (Westinghouse letter NSD-NRC-97-5338, dated September 22, 1997).

21.6.5.6.3.2 Westinghouse STC Flat-Plate Evaporation Tests

A series of liquid film evaporation tests were performed at the Westinghouse Science and Technology Center, as documented in WCAP-12665. The purpose was to observe the behavior of a liquid film and to provide data on evaporative mass transfer. The test conditions were selected to simulate the outside of the AP600 steel containment vessel with the PCS in operation.

The test section was a vertical, 1.8 m (6 ft) long, heated flat steel plate that was coated with the AP600 inorganic-zinc coating. A clear acrylic cover provided a 58 cm (23 in) wide by 10 cm (4 in) deep channel for the forced air flow and allowed observation of the applied liquid film. The plate temperature, the applied-liquid-film temperature, and both the liquid and air flow rates were varied for each test. Six of the 23 tests were conducted with the plate sloped 15 degrees from horizontal, with all other tests conducted on a vertical surface. Reynolds numbers ranged from about 2E+04 to 1.2E+05.

Relatively high air flow rates, in comparison to the evaporation mass transfer rate, were used in these tests. Therefore, inlet and outlet average properties were used to calculate the Sherwood number for comparison with the test data.

The Sherwood number was defined using the channel hydraulic diameter. The data from the 23 Westinghouse flat plate evaporation tests were compared with predictions using the turbulent-mixed-convection correlation with an entrance multiplier of 1.13, as defined in Section 21.6.5.6.1.2, for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model.

Westinghouse evaluated the measurement uncertainties for the flat-plate test facility. The uncertainty in the Sherwood number is about 5 percent for the flat-plate evaporation tests, which were typically conducted with higher heat fluxes than the dry flat-plate tests (see Section 21.6.5.6.2.4).

The staff issued RAI 480.372 concerning the Westinghouse STC flat-plate evaporation tests. The response clarified the use of the channel hydraulic diameter for the evaluation of dimensionless numbers. This closes SDSER Open Item 21.6.5-22f.

21.6.5.6.3.3 University of Wisconsin Condensation Tests

A series of condensation tests was conducted at the University of Wisconsin. The purpose was to provide data on condensation mass transfer in the presence of a non-condensable gas at various inclination angles, velocities, and steam/air concentrations. The results are documented in WCAP-13307.

The test section was 1.9 m (6.25 ft) long, with a 0.84 m (2.75 ft) entrance length, and a 1.1 m (3.5 ft) condensing surface length. The channel cross-section was square with an area of 0.02 m^2 (0.25 ft^2). The top of the test section was a thick aluminum plate coated with the AP600 inorganic-zinc coating. Seven 0.15 m (0.5 ft) long cooling plates were attached to the back of the aluminum test plate to remove heat. Each cooling plate had both flux meters and cooling coils with thermocouples to provide redundant, diverse energy measurements. The test section could be inclined from 0 to 90 degrees from horizontal.

Relatively high air flow rates, in comparison to the mass transfer rates, were used in these tests. As a result, the change in the bulk-to-film steam partial pressure difference from inlet to outlet was small. Inlet-to-outlet average properties were used to calculate the predicted Sherwood number for comparison with the test data.

The data from the tests were converted to hydraulic-diameter-based Sherwood numbers and compared to Sherwood numbers calculated from the assisting-mixed-convection mass transfer correlation described in Section 21.6.5.6.1.1. An average entrance-effect multiplier of 1.20 was calculated as described in Section 21.6.5.6.1.2 for the purpose of evaluating the experimental data. The entrance-effect multipliers are not used in the AP600 containment evaluation model. The data covered by the tests included Re_d over the range 7E+03 to 2.5E+04, with the angle of inclination ranging from 0 to 90 degrees and with steam mole fractions ranging from about 0.12 to 0.65.

Five of the Wisconsin tests were conducted without non-condensables. These tests were also used for the liquid-film heat transfer correlation comparisons presented in Section 21.6.5.6.4.

In the tests for which comparisons were given, the experiments measured condensation rates for a steam/air mixture flowing through a channel with a cooled surface in an apparatus that could be tilted to study the effect of inclination angle upon condensation rates. The staff had three concerns with the Westinghouse description of the test:

(1) When the test section was inclined, the Westinghouse description of the test seemed to imply that the steam/air source was at the low end (implying opposed mixed convection); however, published descriptions of the experiments indicate that the steam/air mixture enters at the high end (implying assisted mixed convection) (see for example, I. K. Huhtiniemi and M. L. Corradini, "Condensation in the Presence of Noncondensable Gases," Nuclear Engineering and Design 141 ,1993, pp. 429-446).

(2) When the inclination angle is low, it is not clear that either the "assisted" or the "opposed" mixed convection treatment is appropriate.

(3) The predicted-to-measured ratios appeared to increase with increasing inclination angle, with increasing Re_d, and with decreasing steam mole fraction. These trends indicated a potential for nonconservatism in modeling the PCS channel (i.e., too much condensation predicted).

Westinghouse responded to these concerns, as identified in RAIs 480.369 to 480.371. The test facility description was revised to indicate that the air/steam source was at the high end for the inclined tests, and that the assisting-mixed-convection correlation is appropriate for the low angle comparisons. In addition, Westinghouse now uses a conservative model for the AP600 licensing calculation. The bias applied to the mass and heat transfer correlations bound the worst test data for the range of angles and steam mole fractions studied. Therefore, SDSER Open Item 21.6.5-22e is closed.

The test measurement uncertainties in the Wisconsin condensation tests result in a measured Sherwood number uncertainty of ±12 percent.

21.6.5.6.3.4 Large-Scale Test (LST) Facility - Internal Condensation

The Phase 2 (confirmatory) heat and mass transfer tests were performed on the large-scale test (LST) facility at the Westinghouse Science and Technology Center. The Phase 2 tests provided data on the transient heat transfer and distribution of non-condensable gas in a geometry similar to the AP600 containment vessel. The purpose was to provide data to develop and validate heat and mass transfer models. The results are documented in WCAP-14135.

The Sherwood numbers inside the LST are defined in terms of $(\hat{v}^2/g)^{1/3}$ for the characteristic length parameter, as described in Section 21.6.5.6.1.3. The measured Sherwood numbers were based on surface-to-bulk gas density differences and shell heat fluxes that were averaged over time and averaged circumferentially at each measuring elevation. Steam partial pressures were not measured at each elevation, so the steam partial pressures were interpolated from the next higher and lower measurement elevations.

The steady-state, circumferentially averaged mass transfer data from 7 of the 25 Phase 2 tests were converted to Sherwood numbers and compared with predictions of the free convection mass transfer correlations described in Section 21.6.5.6.1.3. The Phase 2 tests had a diffuser located below the simulated steam generator. Only tests with film coverage greater than 90 percent were included in the comparison because lower film coverage biases the circumferentially averaged test measurement. This eliminated 17 of the tests. The blind test (RC062, or 220.1) was also excluded from the data evaluation.

A compilation of the predicted-to-measured Sherwood numbers for all seven tests was presented. The mean value is 1.045 with a standard deviation of 0.167. The measured data was compared with the mass transfer correlation as a function of heat flux, steam mole concentration, and $\Delta\rho/\rho$. The correlation matches the trend in the data. The steam mole concentrations ranged from 10 to 50 percent.

Westinghouse has argued that free convection heat and mass transfer on the inside of the shell is conservative during blowdown, when a significant increase in the transfer coefficient is expected due to the blowdown-induced forced convection. LST RC064 (222.3) and RC066 (222.4) predicted-to-measured mass transfer coefficients show the effect of high internal break source kinetic energy. These two tests were conducted in a configuration that simulated an MSLB at the top of the steam generator. The steam source was a 7.6 cm (3-in) inside diameter pipe elevated to a level that simulated the top of the steam generator, rather than with a steam diffuser under the simulated steam generator as in the LOCA configuration. Each test consisted of two steady-state segments with approximately a factor of 2 difference on steam flow rate. Test RC064 had the steam source pointed horizontally at the far wall, and test RC066 had the steam source pointed vertically.

The data show that the predicted-to-measured mass transfer rations are 5 to 10 times greater than the free-convection mean value for LST tests with the diffuser below the steam generator. The location of the maximum corresponds to the elevation where the jet impinges the vessel wall: at x/L = 0.4 for the horizontal jet, and at x/L = 1.0 for the vertical jet. At all elevations the measured mass transfer coefficients were as high as or higher than the mean of the measurements for free convection, with an average value approximately twice that of the free-convection mass transfer coefficient.

Westinghouse performed an uncertainty analysis for the LST. The results indicated that the measured Sherwood number uncertainty was 26 percent for tests with measured heat fluxes greater than 8050 W/m^2 (2500 Btu/hr-ft^2). For tests with heat fluxes between 3200 W/m^2 (1000 Btu/hr-ft^2) and 8050 W/m^2 (2500 Btu/hr-ft^2), the uncertainty was about 40 percent. Tests conducted at low heat fluxes, 1600 W/m^2 (500 Btu/hr-ft^2) to 2600 W/m^2 (800 Btu/hr-ft^2) had uncertainties in the 50 to 75 percent range.

There were three outstanding RAIs (480.404, 480.405, and 480.406) concerning the LST tests used to support the heat and mass transfer correlation as presented in WCAP-14326. Westinghouse responded to these RAIs and has clarified the documentation in the revised version of WCAP-14326. Therefore, SDSER Open Item 21.6.5-22g is closed.

21.6.5.6.3.5 Summary of Mass Transfer Separate-Effects Tests

The basic observations on these results are similar to those for the heat transfer correlations. The results clearly show that the correlations give Sherwood numbers of the right order of magnitude and might be defensible as best-estimate values but not as unconditional conservative values. No "bridge" between these results and quantitative implications for AP600 analysis was provided: that is, no effort was made to evaluate a quantitative uncertainty for the correlations when applied to the AP600 or to quantitatively assess implications for the accuracy and/or conservatism of WGOTHIC results for the AP600.

The Wisconsin condensation tests exhibited some weak trends that, if extrapolated to AP600 conditions, suggested that the treatment of evaporation from the shell exterior could be somewhat non-conservative. The other test series considered exhibited no such trends. It was not clear whether a more detailed review of the test series and/or of the WGOTHIC analyses would lead to a better understanding of these differences and whether they were of any concern for AP600 analysis.

There were three outstanding RAIs (480.373, 480.374, and 480.375) concerning the use of the separate-effects mass transfer data. Westinghouse has responded to these RAIs. The concerns were related to the use of the mass transfer correlations in a best-estimate manner. Westinghouse has now developed a conservative AP600 evaluation model and the mass transfer correlation includes a conservative bias. Therefore, SDSER Open Item 21.6.5-22h is closed.

21.6.5.6.3.5.1 Evaporation

The combined evaporation test data consists of the Westinghouse flat-plate evaporation tests (WCAP-12665) and the Gilliland and Sherwood evaporation tests. The predicted-to-measured Sherwood number ratio for the Westinghouse flat-plate evaporation tests was evaluated as a function of the Reynolds number, Grashof number, and dimensionless steam concentration. The mean predicted-to-measured Sherwood number ratio is 0.936 with a standard deviation of 0.139.

The evaporation test data covered a Reynolds number (Re_d, based on the hydraulic diameter) range up to 1.2E+05 and a Grashof number (Gr_d) range up to 7.0E+10. The evaporation test data covers the expected AP600 range of both the Reynolds number (1.8E+05) and Grashof number (1.2E+09) in the riser annulus during a DBA event.

The Gilliland and Sherwood evaporation tests provided a comparison of the measured and predicted total evaporation rates at relatively low Reynolds and Grashof numbers. The heat and mass transfer correlations predicted the measured total evaporation rates with a predicted-to-measured mean value of 0.925 and a standard deviation of 0.072. Local evaporation measurements were not made and internal variations in partial pressure varied too much to represent the data as an average Sherwood number. Therefore, comparisons between the measured and predicted Sherwood numbers are not meaningful for the Gilliland and Sherwood tests.

Westinghouse concludes that Equations 21.6.5.6.4 and 21.6.5.6.8 adequately model evaporation mass transfer on the AP600 sidewall and dome. When multiplied by the factor developed for use in the AP600 licensing analyses, as discussed in Section 21.6.5.6.5.3 or this report, the evaporation correlation becomes an acceptable conservative correlation appropriate for use in the AP600 evaluation model. The range of the Reynolds and the Grashof numbers in the tests is sufficient to support the use of the correlation over the expected operating range in AP600.

21.6.5.6.3.5.2 Condensation

The combined condensation data consists of the Wisconsin condensation tests (WCAP-13307) and the internal condensation data from the Westinghouse LST (WCAP-14135). The predicted-to-measured Sherwood ratio is evaluated as a function of the Reynolds number, the ratio $\Delta\rho/\rho$, and the dimensionless steam concentration. The mean predicted-to-measured Sherwood number ratio is 0.988 with a standard deviation of 0.182.

The combined test data covered a Reynolds number (Re_d) range up to 2.6E+04 and a density ratio range, $\Delta\rho/\rho$, of 0.08 to 0.55. The Reynolds number will vary with time and position inside

the AP600 containment vessel during a DBA event. During the relatively short blowdown phase, the velocity and corresponding Reynolds number will be largest on the wall nearest the break location and decrease as the flow moves away from the break. A natural circulation flow pattern is expected to develop during the depressurization phase when the PCS is in operation. The Reynolds number along the wall will be small during natural circulation. The value of $\Delta\rho/\rho$ in the AP600 is expected to range up to 0.40, so it is bounded by the test data.

Westinghouse concludes that Equations 21.6.5.6.6 and 21.6.5.6.8 adequately model condensation mass transfer inside AP600. When multiplied by the factor developed for use in the AP600 licensing analyses, as discussed in Section 21.6.5.6.5.3 of this report, the condensation correlation becomes an acceptable conservative correlation appropriate for use in the AP600 evaluation model. The range of $\Delta\rho/\rho$ measured in the tests encompasses the range expected in the AP600.

21.6.5.6.4 Chun and Seban Liquid-Film Conductance Model

The Chun and Seban correlation is used to predict heat transfer through the condensing and evaporating liquid films. Westinghouse applies the correlation to both turbulent and wavy laminar films. Data from tests at the University of Wisconsin (WCAP-13307) were used by Westinghouse to extend the validity of the Chun and Seban correlation to condensing wavy laminar flow and to surfaces that are inclined, as in the dome region of the AP600.

The Wisconsin test facility is described in Section 21.6.5.6.3.3. Five of the 99 Wisconsin tests were conducted without a non-condensable gas present. Without a non-condensable gas, the gas-to-liquid heat transfer coefficient is so high that the gas-to-liquid temperature drop is negligible compared to the temperature drop across the liquid film. The temperature of the liquid-film surface may be assumed equal to the gas temperature and the liquid-film heat transfer coefficient can be calculated from the heat flux divided by the liquid film temperature drop. Since the heat flux, solid-surface temperature, and liquid-film surface temperature are known, the heat transfer coefficient may be derived directly from the measurements. The Wisconsin tests provided an indication of the liquid-film heat transfer coefficient for a range of surface inclinations from vertical to horizontal, covering a range of film Reynolds numbers in the wavy laminar regime.

The Wisconsin (condensing) and Chun and Seban (evaporating) data were compared to the Chun and Seban laminar and turbulent correlations (Eq. 21.6.5.6.7(a) and (b)). In the mid-Reynolds numbers (laminar to turbulent transition range), the correlation overpredicts the Wisconsin data by 25 to 35 percent. The range of film Reynolds numbers on the outside surface of the AP600 containment vessel falls within the range of the test data. Reynolds numbers on the inside surface of containment are less since film is removed at the crane rail and stiffener ring and because the inside film flow rate starts at zero at the top of the dome and increases as the film flows down. The AP600 liquid-film Prandtl number range is approximately $1.5 < Pr < 3.0$. The range of the Chun and Seban data Prandtl numbers is $1.77 < Pr < 5.9$, and adequately covers the AP600 range. Comparison of the correlation to the test data shows that the Chun and Seban correlation is a reasonable representation of the data. The large scatter in the Wisconsin liquid-film heat transfer data is believed to result from operating the tests at (or beyond) the range of operation for which the test facility was designed. The presence of even small amounts of non-condensable gases would bias the results.

The Chun and Seban liquid film conductance model was compared to additional data from Kutateladze et al. (Kutateladze, S.S., Gogonin, I.I., Grigo'eve, N.I. and Dorohkov, A.R., "Determination of Heat Transfer Coefficient with Film Condensation of Stationary Vapour on a Vertical Surface," Thermal Engineering, Volume 24, Number 4, pp 184-186 (1980)) in WCAP-14326, Revision 3, dated April 1998. This comparison provided additional justification for the Chun and Seban correlation in the low Reynolds number range (about 700) with the comparison showing the correlation to be a good fit to the evaporation data and conservative for the condensation data. The liquid film conductance is a low-ranked phenomenon and the use of Chun and Seban is acceptable.

21.6.5.6.5 Summary of Separate-Effects Tests and Mass and Heat Transfer Correlations

The Westinghouse evaluation of the separate-effects tests validates the correlations that are used to calculate energy transfer, by heat and mass transfer, between the containment gas and the external PCS air flow path, and between the PCS air flow path and the baffle, shield, and chimney. The correlations represent the common phenomena of convective heat transfer, condensation mass transfer, and evaporation mass transfer.

The objectives of the Westinghouse analysis were to

- identify appropriate correlations for the various heat and mass transfer regimes for the PCS surfaces
- compare the correlations to separate effects tests that cover the range of dimensionless parameters for AP600 operation
- evaluate correlation uncertainties
- develop biases that can be applied to the correlations to bound the test data

21.6.5.6.5.1 Energy Transfer Model

The correlations developed are used in the AP600 evaluation model and are defined consistently with the way energy transfer is modeled across the containment shell and in the PCS air flow path. The evaluation model energy transfer is calculated as follows:

- With condensation or evaporation, a liquid film is present. Energy is transported between the bulk gas and a solid through the liquid film by the following processes:
  - Between the bulk gas and the liquid-film free surface by radiation heat transfer, convection heat transfer, and mass transfer
  - By conduction through the liquid film to the solid surface
- Dry surfaces do not have liquid films or mass transfer. Energy is transported between the bulk gas and the solid surface by radiation heat transfer and convection heat transfer.

- The correlations assume the local bulk gas thermodynamic states are known both inside and outside containment. Phenomena that influence the distribution of bulk gas properties are separately evaluated to develop an overall conservative approach in the PIRT evaluation.

21.6.5.6.5.2 *Heat and Mass Transfer Correlation Validation*

Analytical correlations were selected from the literature by Westinghouse to represent heat and mass transfer to and from the AP600 containment shell and PCS air flow path surfaces. The correlations contain the physics necessary to model energy transport consistent with the energy transfer model described above. The correlations were compared to separate-effects test data and uncertainties were evaluated. The following correlations were selected for calculating heat and mass transfer and achieving the four objectives of the analysis:

- Opposed-mix-convection heat and mass transfer occur in the PCS air flow path on the downcomer side of the shield and baffle and on the chimney. Equation 21.6.5.6.3, from Churchill, is used to model turbulent-opposed-mixed-convection heat transfer in the PCS air flow path. Heat and mass transfer on the baffle and chimney are both low ranked phenomena in the PIRT, so it is sufficient to model these without additional uncertainty, consistent with the conclusion from the PIRT that only high-ranked phenomena require uncertainties (or conservative values). However, the conservative bias factor, determined for evaporation mass transfer, is applied in the evaluation model.

- Assisting-mixed-convection heat transfer occurs in the riser and chimney to portions of the PCS air flow path on the shell and baffle. Equation 21.6.5.6.4, from Churchill and Eckert and Diaguila, is used for turbulent assisting mixed convection heat transfer in the PCS air flow path. Equations 21.6.5 6.1 and 21.6.5.6.2 define the free- and forced-convection components of the mixed-convection heat transfer correlation in the PCS air flow path. Heat transfer on the shell and dome are ranked medium or low in the PIRT, so it is sufficient to model those without additional uncertainty, consistent with the conclusions from the PIRT that only high-ranked phenomena require uncertainties (or conservative values). However, the conservative bias factor, determined for evaporation mass transfer, is applied in the evaluation model. Comparisons of the assisting-mixed-convection heat transfer correlation to test data were presented. The comparisons show the correlation underpredicts the mean Nusselt number by 2.4 percent, and the test Grashof and Reynolds numbers cover the range expected for AP600 operation.

- Assisting-mixed-convection evaporation mass transfer occurs in the riser and chimney portions of the PCS air flow path on the shell and baffle. The evaporation rate is based on the mass transfer analogy to heat transfer (Equation 21.6.5.6-9). Comparison of the assisting-mixed-convection evaporation predictions and the data were presented. The comparisons show the nominal correlation underpredicts the mean data by 7.5 percent. Since this transport phenomena is ranked high in the PIRT, the data are bounded. The correlation is further biased with a multiplier of 0.84 to produce a conservative evaporation mass transfer correlation for use in AP600 licensing analyses. The comparisons show the test data encompass the expected range of AP600 operating conditions.

- Free-convection heat transfer is assumed on the inside of the shell throughout all transients. Equation 21.6.5.6.6, the modified McAdams free-convection correlation, is used to calculate heat transfer to the shell inside containment. Only free convection is assumed inside containment for AP600 licensing analyses. The assumption of free convection will underpredict the actual heat transfer coefficients inside the shell. Free convection is ranked medium or low in the PIRT, so it is sufficient to model this without additional uncertainty. The McAdams modification replaces the characteristic geometric dimension, L, with the local fluid property $(v^2/g)^{1/3}$ in the Nusselt and Grashof numbers.

- Free-convection condensation mass transfer is assumed on the inside of the shell throughout all transients. Equations 21.6.5.6.8 and 21.6.5.6.10, from Kreith, and the mass transfer analogy, Equation 21.6.5.6.9, are used to calculate mass transfer in the PCS air flow path and inside containment to the shell. Free-convection mass transfer, similar to free-convection heat transfer inside containment, replaces the characteristic geometric dimension, L, with the local fluid property $(v^2/g)^{1/3}$ in the Sherwood and Grashof numbers. Comparisons of the free-convection condensation predictions and the data was presented. The nominal correlation underpredicts the mean data by 1.2 percent. Since this transport phenomenon is ranked high in the PIRT, the data are bounded. The correlation is further biased with a multiplier of 0.73 to produce a conservative condensation mass transfer correlation for AP600 licensing analyses. The comparisons also show the range of the test data encompasses the expected range of AP600 operating conditions.

- Conduction heat transfer through the liquid film occurs on the inside and outside of the containment shell and may occur on the inside of the baffle and chimney if condensation takes place. Equations 21.6.5.6.7(a) and (b), from Chun and Seban, are used to calculate the heat transfer through the internal and external liquid films. Comparisons of predicted and measured film Nusselt numbers were presented. The comparisons show the correlation is a good nominal prediction of the film Nusselt number for both condensing and evaporating films. The comparisons also show the test data encompass the expected range of AP600 operating conditions. Since film conduction is ranked medium or low in the PIRT, it is sufficient to model this phenomenon without additional uncertainty, consistent with the conclusions from the PIRT.

- Radiation heat transfer occurs on all surfaces, but is ranked low in the PIRT on all surfaces. Consequently, it is acceptable to use a traditional T^4 model with an emissivity and beam length for opaque gases. The radiation heat transfer model is not validated through use of the separate-effects tests.

21.6.5.6.5.3 Use of the Mass and Heat Transfer Correlations in WGOTHIC

The mass transfer correlations selected for use in AP600 licensing analyses have been compared to both separate-effects tests and to integral-effects tests, including the LST. The data comparisons were presented in the form of the predicted-to-measured Sherwood number. The comparisons indicate that the correlations tend to underpredict the data with a mean value of 0.936 for evaporation and a mean value of 0.988 for condensation.

Westinghouse uses a conservative evaluation model for AP600 licensing analyses. The staff was concerned with uncertainties in the correlations and the data base, and Westinghouse has biased the correlations to account for these uncertainties. Based on comparisons of the predicted-to-measured Sherwood numbers, the bias for the evaporation mass transfer is a multiplier of 0.84 on the correlations. For condensation, the bias multiplier is 0.73 on the mass transfer correlations. The same multipliers are applied to the heat transfer correlations, based on the mass and heat transfer analogy. The multipliers were chosen to bound the comparisons and are acceptable.

Westinghouse has addressed staff concerns on the use of the heat and mass transfer correlation in WGOTHIC for AP600 licensing analyses and SDSER Open Item 21.6.5-23 is closed.

21.6.5.6.6 Integral Tests

21.6.5.6.6.1 Small-Scale Test (SST) Program

The AP600 reactor design includes a passive containment cooling system (PCS) to remove heat released to the containment following any postulated event and to transfer this heat from the containment to the environment. This system employs natural-draft air cooling and the evaporation of a water film from the outside of the steel containment shell to transfer heat from the containment vessel to the environment

The purpose of the integral small-scale test (SST) program was to demonstrate the operation of the AP600 PCS over a range of operating conditions, including postulated severe accident conditions. Tests were also run at off-nominal conditions using hot and cold film water temperatures and cold cooling air. The purpose of these tests was to avoid ice formation in the annulus region and to determine any effects cold weather could have on the passive containment cooling system design and capabilities. The results of the tests are documented in WCAP-14134.

The AP600 small-scale PCS integral containment cooling tests were conducted using the same facility originally constructed for demonstrating operation of the AP600 PCS. To permit testing over the broader range of operating conditions specified for the extension tests, the test facility was modified to test in cold weather and to simulate abnormal operating conditions. The test facility was also upgraded to improve or automate particular test measurements, such as condensate flowrate, based on experience gained from previous tests.

These tests were performed using the integral containment cooling test facility located at the Westinghouse Science & Technology Center in Churchill, PA. The integral containment cooling test facility used a 7.3 m (24 ft) tall, 0.9 m (3 ft) diameter pressure vessel to simulate the AP600 steel containment shell. The vessel could contain air or nitrogen at 1 atmosphere when cold and was supplied with steam at pressures up to 655 kPa (80 psig). A transparent acrylic cylinder installed around the vessel formed the air cooling annulus. The test vessel wall was 0.95 cm (0.375 in) thick. Water was added at the top of the pressure vessel, forming a film which flowed down over the vessel external surface. Air flow up the annulus outside the vessel cooled the vessel surface, condensing the steam inside the vessel.

Saturated steam from a boiler was throttled to a variable, but controlled, pressure and supplied to the bottom of the vessel, which initially contained 1 atmosphere of air. The steam was distributed inside the vessel by one of two steam distributor arrangements. The uniform steam distributor provided for slow radial flow, uniform along and around the central supply pipe that ran the full height of the test vessel. The uniform distributor was expected to produce the most limiting steam condensation conditions.

To establish the total heat transfer from the test vessel, measurements were recorded for steam inlet pressure, temperature, and condensate flow and temperature from the vessel. Twenty-four thermocouples were located on the outer surface of the vessel. The thermocouple measurements were weighted by the respective vessel wall areas sensed by the thermocouples and summed to obtain the average vessel outside surface temperature.

An axial fan to control the cooling-air velocity was located in the chimney region above the test vessel and formed the upper chimney for the cooling-air flowpath.

Water could be added at the top of the vessel to create a water film on the external surface. External water film flow rates onto the vessel and the flow rate of excess water from the bottom of the vessel that was not evaporated were measured.

External-cooling-air temperature, humidity, and velocity could be surveyed by traversing measurements at several elevations in the cooling annulus. Air velocity was also measured at the inlet to the air heating coil. Thermocouple measurements were sequentially sampled and converted by data acquisition equipment and recorded on paper tape or computer memory and disk.

21.6.5.6.6.1.1 Test Matrix

The test matrix included a full range of expected design basis external water flowrates, the maximum and minimum expected cooling-air velocities, and the cooling-air inlet temperature and relative humidity.

External Baffle Cooling-Air Velocity

Based on the AP600 containment transient analyses available at the time, three cooling-air velocities, 4.9 m/sec (16 ft/sec), 3.7 m/sec (12 ft/sec), and 2.4 m/sec (8 ft/sec), were examined. 4.9 m/sec (16 ft/sec) corresponded closely to the maximum calculated air velocity during wetted heat transfer shortly after PCS initiation following a postulated loss-of-coolant accident (LOCA), and to the calculated air velocity with a dry containment surface when containment internal pressure is 380 kPa (40 psig).

The 3.7 m/sec (12 ft/sec) velocity was the calculated air velocity in the baffle when the containment pressure is 240 kPa (20 psig).

The 2.4 m/sec (8 ft/sec) velocity was the air velocity calculated to generate natural circulation when the containment pressure is 172 kPa (10 psig). This pressure and air velocity defines the

condition in which containment cooling would transition from a wetted external surface to a dry surface after the stored PCS water has been used, if no operator action were taken within 3 days.

External Containment Surface Water Flow

The flow rates supplied to the top external surface of the test vessel corresponded to the prototypic water flow rates onto the AP600 containment which were used in the LOCA response containment transient analysis. These AP600 maximum and minimum PCS water supply flows were 776 L/min (205 gpm) and 204 L/min (54 gpm), respectively. Since some of the supplied water would be evaporated from the containment dome, the amount of water which reached the top of the cylindrical portion of the containment would initially be about 570 L/min (150 gpm) and would be reduced to about 136 L/min (36 gpm) at 3 days when the PCS water would be exhausted. Based on the expected design perimeter of the cylindrical portion of the AP600 containment, the maximum and minimum flow rates around the containment top, exterior, vertical surface would be 0.453 L/min/m (0.398 gpm/ft) and 0.108 L/min/m (0.095 gpm/ft), respectively. These flows were matched on the 0.9 m (3 ft) diameter PCS test vessel and an intermediate flow of 0.285 L/min/m (0.25 gpm/ft) was included in the test program.

Cooling-air Inlet Temperature / Relative Humidity

For all wetted test conditions the cooling-air inlet temperature was maintained at 54 °C (130 °F). This elevated air inlet temperature approximates the average air temperature that would occur in the full-sized AP600 cooling path, based on the maximum environmental air temperature of 46 °C (115 °F). The inlet air relative humidity was raised to 30 percent to approximate the average specific humidity that will be achieved in the full sized AP600 PCS cooling path, based on the 29 °C (85 °F) maximum wet bulb temperature for the inlet air. For comparison, identical test conditions with low relative humidity inlet air were also examined.

21.6.5.6.6.1.2 Test Validation

The following acceptance criteria were established for the tests:

- Data on forcing functions were available (i.e., steam flow rate, fan speed, water flow rates, inlet temperatures of steam, water, and air). Strict adherence to the specific absolute pressures and flow rates was not necessary but values should be nearly constant, as defined in the test matrix.

- Data were available on response variables. Condensate flow rates, excess water flow rates, air, water, and steam outlet temperatures, vessel pressure, 80 percent of the vessel and fluid temperatures, and vessel water coverage were measured.

- Unplanned excursions needed to be evaluated on a case-by-case basis. Failures that may have resulted in faulty data outputs were not acceptable.

- The vessel pressure was maintained within the specified pressure limits during the constant pressure portions of these tests.

Heat balances were performed to roughly determine the acceptability of the test data and instrument performance. Heat loads were calculated for the reported tests by three methods:

- condensate mass flow rate
- external heat loss (water and air)
- heat flux across wall

21.6.5.6.6.1.3 Matrix Tests

Base Cases with Constant Steam Flow

Seventeen base-case tests with constant steam flow were run to establish internal vessel pressures under nominal cooling conditions. The steam flow was 0.1 kg/sec (0.25 lb/sec). All other conditions were identical except for the size of the annulus. Two tests were performed using the 13 cm (5 in) annulus, the other 15 used the 38 cm (15 in) annulus. Two of the tests, one using the 38 cm (15 in) annulus (Run 38, Test 106-1SU), the other using the 13 cm (5 in) annulus (Run 70, Test 106-5U) resulted in invalid data. Test 106-1SU was aborted because of rain and test 106-SU was aborted because the annulus air velocity was incorrect.

Base Cases Repeated with Prototypic Steam Injection

Four tests were run in this category. They were all performed using the 13 cm (5 in) annulus width. These tests were performed using prototypic steam injection through a 15 cm (6 in) diameter pipe at the 1.5-1.8 m (5-6 ft) elevation offset from the center of the test pressure vessel. All these tests generated valid data.

Water Film Limits of Cooling

Three tests were run in this category. These tests determined the effects of film flow rate on coolability. The flow rates used were 1.9, 3.8, and 9.5 L/min (0.5, 1.0, and 2.5 gpm). The flow rate was changed to degrade cooling until the limits of coolability were reached. One of these tests was not completed due to a computer failure.

Water Film Distribution Limits of Cooling

Nine tests were run in this category. These tests determined the effect of water film distribution on cooling. The water film distribution was varied from 100 to 66 to 33 percent. This variable was changed to degrade cooling until the limits of coolability were reached. All of these tests generated valid data.

Vessel Air Content Effect on Heat Transfer

Three tests were run to determine the effect of vessel air content on heat transfer. The initial air content was varied from 1.0 atm to 2.0 atm. The 13 cm (5 in) annulus was used with prototypic injection. All of these tests generated valid data.

Baffle Air Flow Limits

Six tests were run to determine the limits of air flow in the baffle region. The air flow was varied from 3.7 m/sec (12 ft/sec) to natural circulation. All data generated from these tests were valid.

Water Film Temperature (Phases I and II)

There were two tests run in Phase I to determine the effects of the shell water film temperature. In both cases, the temperature was raised to 49° C (120° F), increased from the normal temperature of 27° C (80° F). These tests were performed with two steam flows: 0.05 kg/sec (0.1 lb/sec) and 0.1 kg/sec (0.25 lb/sec). Both of these tests produced valid data. Phase II tests varied both the steam flow and the water film temperatures. In the Phase II tests, the temperatures were 4.4° C (40° F) and 27° C (80° F). These tests also produced valid data.

Transient Steam Flow

These tests varied the steam flow rate. Rather than a steady state value for steam flow, a transient rate from 0.68 kg/sec (1.5 lb/sec) to 0.05 kg/sec (0.1 lb/sec) over 30 seconds was used for both tests in the category. The data generated was valid.

Ice Formation and Melt Demonstration

These tests were run to determine the onset of ice formation in the annulus region and any other effects of cold weather on the passive containment cooling capabilities of the system. Both tests produced valid data.

21.6.5.6.6.1.4 Test Summary

Westinghouse considered the SST program successful and the criteria of the program were met. Assessment of the heat balances indicated that the majority of the tests were performed in a consistent fashion and were in agreement with respect to the three methods of heat balancing. Westinghouse concluded that the tests performed as expected and their overall behavior was repeatable. The test program confirmed the expected operation of the PCS over the range of operating conditions including postulated severe accident conditions.

21.6.5.6.6.2 Large-Scale Test Facility (LST)

The AP600 large-scale containment cooling tests were performed at the large-scale test (LST) facility, located at the Westinghouse Science and Technology Center in Churchill, PA. The large-scale PCS test facility used a 6.1 m (20 ft) tall, 4.6 m (15 ft) diameter pressure vessel to simulate the steel containment shell, with a height-to-diameter ratio more typical of the actual containment shell than was available for the small-scale tests (See WCAP-14134). The larger vessel made it possible to study in-vessel phenomena, such as non-condensable mixing, steam release jetting, and condensation, as well as flow patterns inside containment. The vessel contained air at atmospheric pressure when cold and was supplied with steam at pressures up to 793 kPa (100 psig). A transparent acrylic cylinder installed around the vessel formed the air-cooling annulus. Air flow up the annulus outside the vessel cooled the vessel surface, condensing the steam inside the vessel.

Superheated steam from a boiler was throttled to a variable, test-dependent, flow rate, but at a controlled pressure, and supplied to an off-center compartment below the operating deck of the test vessel. A steam distributor provided low-velocity steam at a scaled height commensurate with that of the operating deck of the reactor plant.

To establish the total heat transfer from the test vessel, steam inlet pressure, steam flow, temperature, and condensate flow and temperature were measured. 78 thermocouples, located on both the outer and inner surfaces of the vessel's 2.2 cm (0.875 in) thick steel wall, indicated the temperature distribution over the height and circumference of the vessel. Additional thermocouples, placed throughout the inside of the pressure vessel on internal heat sinks (32), adjacent to the vessel wall (27), and throughout the internal volume (47), provided a measurement of the vessel bulk steam temperature as a function of position.

An axial fan at the top of the annular shell allowed the apparatus to be tested at higher air velocities than could be achieved during purely natural convection. The temperature of the cooling air was measured at the entrance of the annular region and upon exiting the annulus in the chimney region before the fan. The cooling-air velocity was determined by calibrating the fan controls by conducting a velocity traverse on the cooling annulus and using a heated-wire anemometer at various fan rpm control settings. A fixed-vane anemometer was also located below the fan in the air exit stream to provide a continuous output of the annulus air velocity. The heat transfer to the cooling air (i.e., its temperature rise multiplied by its specific heat and its measured flow rate) and the water evaporated provided a measurement of the total heat transfer.

21.6.5.6.6.2.1 Test Objectives

The purpose of the large-scale test (LST) passive containment cooling system (PCS) heat transfer test was to examine anticipated thermal-hydraulic phenomena on a large scale: (1) the interior natural convection and steam condensation, (2) the exterior water film evaporation, (3) air-cooling heat removal, and (4) water film behavior. This experiment was designed to induce the same sort of containment dome heat transfer processes and circulation and stratification patterns inside containment as are expected in the AP600; however, it was not meant to simulate specific AP600 accident scenarios. The large-scale test data were used to verify the WGOTHIC computer code, which is used to analyze the AP600 containment.

Baseline tests consisted of 16 steady-state tests that were performed at three constant pressure conditions to investigate the effects of various water coverage levels, various external air flow rates, and the presence of internal structures.

Phase 2 tests provided data to validate the WGOTHIC containment heat and mass transfer correlations over a range of prototypic internal conditions, including the effects of external parameters. The tests provided data on the transient heat transfer and the distribution of non-condensables. The effects of non-condensables on the containment heat transfer were observed.

Followon tests (Phase 3) examined special effects, such as the location of the steam discharge and the concentrations of non-condensables, and were not strictly necessary for code validation, but aided in the overall understanding of the containment cooling phenomena.

21.6.5.6.6.2.2 Facility Scaling

Pressure Vessel

The AP600 containment vessel is cylindrical with 2:1 elliptical heads at the top and bottom of the cylinder. The containment area above the operating deck is cooled on the outside and is considered the active heat transfer area.

Below the operating deck in the AP600 containment building, there are free volumes where equipment is installed. These volumes are normally in communication with the volume above the operating deck, so any steam released in an accident can either enter or discharge into them. If a volume has more than one path, circulation will exist between this volume and the containment atmosphere above the operating deck, and the volume is referred to as an "open volume". A volume in which the containment fluid does not readily circulate with the containment atmosphere above the operating deck is called a "dead-ended compartment".

The open volumes and dead ended volumes are approximated by the following AP600 compartments:

Open Volumes

- Refueling cavity
- Steam generator loop compartment (without break)

Dead-Ended Volumes

- Region below the 107 ft elevation of containment
- In-containment refueling water storage tank (IRWST)
- Reactor cavity
- Accumulator areas
- Chemical and volume control system module

The total free volume below the operating deck and the free volume occupied by each sub-compartment were calculated. When the large-scale test was being designed, the AP600 plant design showed that the open volumes occupied 21 percent of the total volume below the operating deck. The dead-ended volumes occupied 70 percent, and the steam generator compartment (with break) occupied 9 percent of the total free volume below the operating deck.

There are two steam generator compartments in the AP600. In the LST facility, one was considered to be part of the open volume. The other was considered the steam generator volume because steam release was simulated in it. In the AP600, the two compartments are connected and communicate with one another, in the LST there was no connection.

The Phase 2 and Phase 3 tests had representations of the open, dead ended, and steam generator volumes. The open volume provided vertical communication with the vessel volume above the operating deck. The dead-ended volume had one entrance (from the open compartment) and no exit. It did not directly communicate with the containment atmosphere above the operating deck. The percentages given above were used to determine the volume of the open, dead-ended, and steam generator compartments below the operating deck in the LST.

As built, the LST open volume occupied 21 percent of the total volume below the operating deck. The dead-ended volume occupied 70 percent, and the steam generator compartment occupied 9 percent of the total volume below the operating deck.

In the AP600 plant design, the open volumes occupy 19 percent of the total free volume below the operating deck; the dead-ended volumes occupy 73 percent of the total free volume below the operating deck; and the steam generator compartment occupies 8 percent of the total free volume below the operating deck. The volume percent occupied by each compartment in the LST was close to the volume percent occupied by each compartment in the AP600 and was found, by Westinghouse, to be sufficient to evaluate the capability of WGOTHIC to represent each type of volume in the AP600. The staff accepts these minor variations in the volumes.

Heat Sinks

The heat sinks in the AP600 are the equipment in containment and the containment structural materials. These constituents are divided into two groups: short- and long-term heat sinks. The short- term heat sinks are the materials that absorb heat quickly, whereas the long-term heat sinks continuously remove heat over the long term (e.g., the concrete transfers heat to the soil around the bottom of the containment vessel).

Short-Term Heat Sinks

The initial steam release into the containment raises containment pressure. Internal masses absorb heat, condense steam, and reduce the transient pressure. To address the effect of thermal storage on initial pressurization and evaluate non-condensable gas distributions, surface area and mass were added inside the large-scale containment model.

As steam enters the containment from the steam generator compartment, steam and air are forced into the open and dead-ended compartments below the operating deck. Relatively large areas and masses condense the steam, and the remaining air is thermodynamically stable and is expected to remain within the compartments below the operating deck. The partial density of steam above the operating deck increases together with the rate of diffusion for steam condensing on the vessel walls.

The steel in the compartment walls, deck supports, operating deck cover, and operating deck grating provides some heat storage. On the basis of the surface area available for heat transfer in the AP600, additional heat transfer area was needed in the large-scale test. Aluminum plates were added to supply additional heat removal from the containment atmosphere. The aluminum plates were designed to provide a relatively short heat transfer time constant. The time constant for effective heat removal from the steam and air containment atmosphere is approximately 100 seconds.

The required surface area was provided by aluminum mounted in banks inside the large-scale test facility. The plates were positioned in groups in: (1) the dead-ended compartment, (2) the open compartment, and (3) just above the operating deck. The three groups were representative of the additional surface area available in the AP600 in these three general locations.

Long-Term Heat Sinks

The purpose of representing long-term heat sinks in the large-scale test was to model long-term heat removal and its effect on non-condensable gas distribution. The effect of the long-term heat sinks was modeled in the LST by removing the bottom insulation surrounding the open and dead-ended compartments. Information from AP600 analyses concerning long-term heat sinks, such as heat sink surface area, heat removal rate late in the transient, and the percent of heat removed by the long-term heat sinks, were considered when evaluating long-term heat sink representation in the LST. Test numbers 218.1, 219.1, 220.1, and 221.1 in the Phase 2 test matrix included the effects of long-term heat sinks.

21.6.5.6.6.2.3 Test Matrix

The tests were conducted in three phases:

- Baseline tests
- Phase 2 tests
- Phase 3 or followon tests

Phase 1 Test Matrix

The baseline tests were reported in WCAP-13566, "AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Test Baseline Data Report," AP600 Document PCS-T2R-003, Revision 1, October 1992.

Phase 2 Test Matrix

The large-scale Phase 2 test matrix consisted of twelve tests for the AP600. The tests in the Phase 2 matrix covered a range of pressures and operating conditions that were judged by Westinghouse to be sufficient to verify the WGOTHIC code's ability to predict pressure and temperature responses to accident scenarios. All the tests were performed with water from the PCS at a temperature of 10 - 27° C (50 - 80° F). The external water flow rates were chosen to simulate the amount of coverage expected on the AP600 plant. The annulus air flow was maintained at an external air velocity of 3.7 m/sec (12 ft/sec) by adjusting the fan speed for all of the tests performed during Phase 2. It was not deemed necessary to vary the air velocity in this set of tests since the focus was on internal distributions and heat transfer.

Tests 202.3 and 203.3 were provided as repeats of the constant-pressure tests performed during the baseline test series (202.1, 202.2, 203.1 and 203.2). They were included to evaluate the effect of the addition of the steam generator model and the bottom insulation and to obtain additional test data with the enhanced instrumentation.

Tests 212.1 through 221.1 were steam-flow-rate-specified transient tests. The effects of the various parameters were investigated by changing one parameter while all others were held constant. Tests 202.3 through 217.1 addressed the effects of the short-term heat sinks installed in the test facility. Long-term heat sinks were modeled in tests 218.1, 219.1, 220.1, and 221.1 by partial removal of the insulation on the bottom of the test vessel.

For tests 212.1 and 213.1, three steam flow rates were tested as the system approached a steady-state condition after each flow adjustment. The three nominal flow rates selected were 0.1, 0.2, and 0.3 kg/sec (0.25, 0.5, and 0.75 lbm/sec). These tests were used to verify WGOTHIC's ability to predict transient behavior, and to demonstrate the effects of different PCS water coverages.

Two tests (214.1 and 215.1) were performed where air flow was allowed to develop by natural convection (before turning on the fan) to the 3.7 m/sec (12 ft/sec) air annulus flow. The steam flow rate was held at approximately 0.4 kg/sec (1 lbm/sec) throughout the tests. After steady state conditions were reached, the fan was turned on to exercise WGOTHIC's ability to model the transition between free and forced convection. Test 215.1 required that a 180-degree circumferential section be blocked off so that the annulus air flow only enters around the remaining 180-degree azimuthal section. This test examined the effect of partial blockage of the air inlet region. The 180-degree azimuthal blockage was centered around the steam generator.

Test 216.1 was a transient between two steady-state conditions: 75-percent PCS coverage and 25-percent PCS coverage in quadrants. This test can be compared to tests 212.1 and 213.1 to evaluate the difference between PCS water coverage in stripes and in quadrants.

Tests 217.1 through 221.1 addressed the effect of long-term heat sinks, helium addition to simulate hydrogen from postulated severe accidents, and steam blowdowns. The tests were performed with helium (helium, instead of hydrogen, for safety reasons) to simulate the maximum possible hydrogen concentrations.

The purpose of tests 217.1 and 218.1 was to evaluate the effect of long-term heat sinks on non-condensable distribution. Test 218.1 was similar to test 217.1, with the only difference being the inclusion of long-term heat sinks. After the system had come to steady state, helium was injected for 30 minutes, and the system was allowed to achieve a second steady state.

The purpose of test 219.1 was to evaluate how the non-condensables (specifically helium) distribute in the following scenario: achieve steady state without external water flow, inject helium, come to steady state again, then start the PCS flow. This test provided data on the effects of rapid cooling of a dry containment on non-condensable distribution.

Tests 220.1 and 221.1 addressed modeling heat transfer to heat sinks and the containment shell, as well as the effects on the flow field during the blowdown phase of a transient.

Test 220.1 modeled a blowdown of a small steamline break. The blowdown was over within a minute and was used to verify the WGOTHIC's ability to predict transient behavior. The blowdown rate for test 220.1 was based on a steamline break (SLB) at 102-percent power with a full double-ended rupture and main steamline valve failure. The test represented an AP600 limiting case with respect to containment pressure and temperature. The steamline break flow was scaled by volume (1:8) for the large-scale test to produce the steam flow. The steam flow rate was a target test condition; a slightly lower peak flow rate would not have affected the test purpose of verifying WGOTHIC, and the test was not meant to simulate the prototypical accident. This test was used as the blind test for analysis verification.

Test 221.1 modeled long-term cooling during post-accident conditions. This was accomplished by starting the test with water flow on the outside, reaching a steady state after the initial steam blowdown, injecting helium, reaching a second steady state, then shutting off the PCS flow, and reaching a final steady state. Non-condensable measurements were taken to evaluate the effect of this scenario on helium mixing. The helium injection and the steam flow rate for test number 221.1 were based on a small LOCA with an in-containment refueling water storage tank check valve failure. The test was not meant to simulate the prototypical accident, although the prototypical accident was used as guidance so that the conditions in the plant and in the large-scale test would be similar. The steam blowdown for the plant is scaled by volume (1:8) for the large-scale test. After the blowdown, the flow rate was maintained at a low constant flow of approximately 0.06 kg/sec (0.15 lbm/sec).

Phase 3 Test Matrix

Seven tests were identified as part of the Phase 3 test program. The first four were rapid pressurization transients that investigated the effects of the steam discharge location and orientation, and the last three characterized the effect of the initial internal atmosphere on condensation mass transfer.

Test Series 222

LST data from baseline and Phase 2 tests suggested to Westinghouse that non-condensable concentrations increase dramatically below the elevation of steam injection, with considerable steam mixing above the operating deck. The effect of the higher steamline elevation could be to create a larger volume of air-rich mixture, which extends above the operating deck and reduces the active heat transfer area. The higher break elevation was representative of a steamline break. This series of tests addressed the impact of the elevation and direction of the steam break on the response of the test vessel and followed the transient blowdown behavior to an ultimate steady-state condition.

The four configurations in this test series were:

222.1 Low-velocity steam flow from under the operating deck

222.2 Low-velocity steam flow at 1.8 m (5.8 ft) above the operating deck

222.3 High-velocity steam flow with horizontal discharge 1.8 m (5.8 ft) above the operating deck

222A High-velocity steam flow at 1.8 m (5.8 ft) elevation upward

Two steam injection directions were run with both pipe exits at the 1.8 m (5.8 ft) elevation, which was linearly scaled (1/8th) to the location of a steamline coming off the top of the AP600 steam generator. The tests are run with nominal 75-percent water coverage and 3.7 m/sec (12 ft/sec) air flow.

Test 223.1

This test permitted a direct measurement of the liquid-film heat transfer coefficient by reducing the non-condensable concentrations to a very low level by evacuating the test vessel. The large dome in the large-scale test vessel produced data for non-vertical surfaces that were more prototypical, providing a link to similar data from the Wisconsin condensation tests with pure steam and tested the validity of the Chun and Seban liquid-film heat transfer model used in the WGOTHIC code.

Test Series 224

These tests permitted measurement of the effect of a higher non-condensable concentration on the transient and steady-state performance of the test vessel for verifying the non-condensable partial-pressure effect in WGOTHIC models at two different steam flow rates. The vessel pressure was increased to 2 atmospheres of air before the start of steam flow.

21.6.5.6.6.2.4 Test Summary

The Westinghouse evaluation of large-scale PCS test data yielded the following information and conclusions:

- Evaporation was the primary mode of heat removal from the outside of the vessel (approximately 75-percent of the total), followed by sensible heating of the subcooled liquid film (approximately 17-percent of the total). The remainder of the heat was transferred to the environment by convection and radiation.

- The heat removal rate was proportional to the film coverage area in quadrant-coverage cases, but had a weak dependence on the coverage area in striped-coverage cases. For the same film coverage area, striped coverage provided better heat removal than quadrant coverage.

- The heat removal rate appeared to be more strongly dependent on ambient-air temperature than liquid-film temperature.

- The heat removal rate has a relatively weak dependence on annulus air velocity, which indicates that the resistance to heat transfer on the inside of the vessel is greater than on the outside.

- For all of the wetted large-scale tests (except the horizontal, high-velocity steam jet injection case), the highest heat flux occurred near the top of the dome at the elevation where the external film was applied. Although the dome represents about 30-percent of the heat transfer surface area, approximately 40-percent of the total heat removal occurred on the dome and 60-percent on the cylindrical sidewalls.

- Injection of low-velocity steam resulted in relatively good mixing above the injection location, but stratification below, causing air to be concentrated below the operating deck. The heat removal rate increased as the axial steam concentration gradient was increased (by raising the injection location).

- Injection of high-velocity steam resulted in a well-mixed vessel both above and below the operating deck).

- Injection of a light, non-condensable gas did not degrade the condensation heat transfer or affect the overall heat removal. The gas did not stratify (collect at the top of the vessel), but was well mixed above the injection location and eventually, well mixed throughout the entire vessel.

In the LST, condensate was collected from five regions: girder, inside vessel sidewall, a lower horizontal plane from which rain was collected, inside vessel (below deck), and a simulated steam generator section. Numerous attempts were made during the course of testing to measure rainout within the test vessel. In no case was any rainout measured. Westinghouse stated that the lack of rainout was due to the wetting properties of the special coating applied to the interior of the LST and AP600 shell. There may be rainout at low PCS water temperatures as a result of condensation shock. However, the staff has determined that rainout should not increase the peak containment pressure, and may even be a slight improvement as these drops would slightly enhance energy removal from the atmosphere to the sump. Therefore, Open Item 21.6.5-3, concerning the possible effects of rainout on liquid film stability, is closed.

21.6.5.6.6.3 PCS Air Flow Path Pressure Drop Test

The PCS air flow path pressure drop test was a 1/6 scale, 14.32° wedge model of the PCS downcomer, riser and chimney. The test results are documented in WCAP-13328 and, as a result of the test, some changes to the final design of the air annulus flow path were incorporated into the AP600 from the configuration initially considered. The resulting data were extended, by a factor of 1.5, to account for the higher expected Reynolds number in the AP600. The form loss was later increased an additional 30-percent to account for the design of the baffle turning vane in the AP600.

In response to staff concerns with the upper annulus drains, a design change was incorporated to move the drains about 0.3 m (1 ft) above the upper annulus drain floor. This effectively shortens the downcomer-to-riser-turning region from about 1.8 m (6 ft) to 1.5 m (5 ft), once sufficient PCS water fills the region to the new drain elevation. In response to RAI 720.440F (letter DCP/NRC1232, NSD-NRC-98-5543, January 28, 1998), Westinghouse determined that the geometry changes do not impact the WGOTHIC modeling of the AP600 for design-basis accident analyses, and the data from the test (as modified) are still valid for the AP600 evaluation model. The existence of this pool of water will not significantly affect the already low velocity in the air entrance region. The effect of the cold water pool on the air density was shown to have a negligible effect on the buoyancy-driven air flow.

21.6.5.6.7 WGOTHIC Verification Using LST Data

The lumped-parameter model has been used by Westinghouse to perform WGOTHIC analyses of selected LST tests. The purpose of the WGOTHIC model comparison to LST data was to understand how the documented biases apply to AP600 and to develop guidance for bounding the effects of those biases. The predicted and measured vessel pressures were compared for the priority and non-priority tests (see Table 21.6-16). The initial comparisons were done with WGOTHIC, Version 1.2 (WCAP-14382). As a result of changes to WGOTHIC, resulting in

Version 4.1 (WCAP-14967), selected LST tests were reanalyzed. It is also noted that data from the LST were also used for separate-effects verification of particular models, as discussed below. Additional changes to WGOTHIC were made resulting in Version 4.2, which is the current licensing version. Westinghouse determined, through a regression test as part of the validation effort, that these changes have no impact on the calculated peak pressure .

WGOTHIC 1.2 Verification Studies

The priority tests run with the lumped parameter evaluation model were tests 212.1A, 212.1B, 212.1C, 2l4.1A, 214.1B, 216.1A, 216.1B, 219.1A, 219.1B. 219.1C, 222.l, 222.4A, and 222.4A.

The steam inlet configuration for tests 212.1A, 212.1B, 212.1C, 214.1A, 214.1B, 216.1A, 216.1B, 219.1A, 219.1B, 219.1C, and 222.1 was a steam diffuser within the simulated steam generator compartment. The steam entered the containment atmosphere as a buoyant plume, as in a post-blowdown LOCA event.

The steam inlet configuration for tests 222.4A and 222.4B was a 7.6 cm (3 in) diameter pipe located 1.8 m (5.8 ft) above the operating deck. The steam entered the containment atmosphere as a high-velocity jet, as in a MSLB event.

Because the mixing and velocity magnitude inside containment differ significantly in the two configurations, they are discussed separately.

Buoyant Plume Tests

The large-scale tests conducted with the steam diffuser below deck simulated the post-blowdown portion of a LOCA. The heat and mass transfer inside the LST vessel was dominated by free convection. The low-velocity steam injection resulted in relatively good mixing and created small axial non-condensable concentration gradients above the injection location, but stratification below, causing air to be concentrated below the operating deck.

Test 222.1, which had non-condensable measurements at four locations within containment, was discussed in some detail by Westinghouse to explain the characteristics of the lumped-parameter evaluation model.

The lumped-parameter evaluation model overmixes the air (non-condensables) from below the injection location and overpredicts the velocity in the vessel. The velocity meter along the wall at A-90° (near the vessel spring line elevation) was the only meter functioning for this test. Although the measurement and prediction showed that the velocity was in the downward direction, the predicted velocity was much higher than the measured velocity.

For the lumped-parameter evaluation model, the forced-convection component of mixed-convection heat and mass transfer inside the vessel is neglected, but the vessel is also overmixed. The overmixing carries air above the operating deck. Increasing the concentration of air above the operating deck degrades the mass transfer, and thereby reduces the heat removed from the vessel.

The lumped-parameter evaluation model overpredicted vessel pressure for test 222.1, as expected.

The predicted vessel pressure for tests 212.1A, 212.1B, 212.1C, 214.1B, 216.1A, 216.1B, 219.1A, 219.1B, and 219.1C were also overpredicted. The pressure during the initial part of test 214.1A was underpredicted.

Westinghouse concluded that the initial pressure underprediction for test 214.1A was caused by the modeling of the water coverage. The water coverage fraction on the exterior vessel surface is a code input parameter. There was some question as to the water coverage fraction. The water coverage was changed from 100 percent to 78 percent at 3,000 seconds in the model, resulting in an underprediction of vessel pressure (for the time period from approximately 1500 seconds to 3,000 seconds). The water coverage in the model should have been changed continuously, starting earlier. However, since coverage was not continuously monitored, it was simply modeled as a step change.

High-Velocity Jet Tests

The large-scale tests with a 7.6 cm (3 in) steam source simulated the lower velocity portions of a MSLB.

The injection of the high-velocity steam source resulted in a well-mixed containment, based on non-condensable measurements. Based on measurements of internal velocity, the heat and mass transfer within containment had a significant forced-convection component.

The lumped-parameter evaluation model uses only free-convection heat and mass transfer models, which conservatively biases the results because the test actually had a significant forced-convection component. The vessel pressures calculated with the lumped-parameter evaluation model overpredicted the measured pressures for tests 222.4A and 222.4B.

In the evaluation model, the steam is injected at an elevated level as it was in the test. The model predicts a more stratified containment than the measurements. This is because the model entrains fluid into the jet from nodes at or above the steam injection point, resulting in a well-mixed atmosphere above the point of steam injection and an air-rich atmosphere below the steam injection point. This is contrary to the measurements, which show that the kinetic energy from the high-velocity jet mixes the entire containment.

Table 21.6.5.6.2 shows the average steady-state measured and predicted vessel pressures for all the priority tests. The lumped-parameter evaluation model overpredicts the measured steady-state vessel pressure for all the tests.

The non-priority tests run with the lumped-parameter evaluation model were tests 213.1A, 213.1B, 218.1A, 218.1B, 224.2, 221.1A, 221.1B, 202.2,224.1, 217.1A, and 217.1B.

The predicted results for the priority tests are consistent with the non-priority tests. The steady state predicted and measured pressure for the non-priority tests are given in Table 21.6-16.

Lumped-Parameter Evaluation Model Conclusions

The lumped-parameter evaluation model overpredicts the vessel pressure for steam entering the vessel as either a buoyant plume or a high-velocity jet.

The lumped-parameter evaluation model, used for AP600 design-basis accident analyses, does not resolve internal velocity and concentration fields due to its simplified momentum model and large lumped volumes. Comparisons between preliminary versions of the evaluation model and the system level LST response showed that pressure was reasonably well predicted, with a modest conservative margin. Examination of internal processes by Westinghouse identified the existence of competing internal effects. The excessive velocities predicted by the lumped-parameter model overpredicted the velocity component of mass transfer and this overmixing underpredicted the steam concentration component of mass transfer. The effect of overpredicted velocities was resolved by using only free convection for internal heat and mass transfer, thereby eliminating velocity from the condensation correlation. The overmixing issue was resolved by examining and biasing the effects of circulation and stratification in the evaluation model as discussed in Section 21.6.5.7.

WGOTHIC 4.1 Verification Studies

The results of the WGOTHIC LST integral-model comparisons to the LST data have been used to understand the biases inherent in the WGOTHIC lumped-parameter formulation, as they apply to AP600 DBA containment pressure analysis, and to guide the development of the bounding evaluation model. The LST model used nominal inputs for geometry, initial, and boundary conditions so that the effects of the code and noding used could be better isolated. The same WGOTHIC input decks that were used in the Version 1.2 studies were used as input for the WGOTHIC Version 4.1 studies. The inertial length input was modified to account for the change in the "ccvel" subroutine in WGOTHIC Version 4.1. This input change results in a similar annulus velocity for the two versions and was done to isolate the effects of other code version changes.

Since the lumped-parameter bias for a specific model is related to the noding chosen, the noding used in the AP600 containment pressure DBA evaluation model corresponds to the noding used in the LST validation work (WCAP-13328, Section 6). Therefore, Westinghouse concludes that the biases and guidance developed from the LST models are applicable to the AP600 model.

The upgrade of WGOTHIC solver from Version 1.2 to Version 4.1 does not invalidate conclusions of the LST integral-model validation. Therefore, the bounding approach to address lumped-parameter code biases in the evaluation model remains acceptable.

The following sections summarize the differences in results of the WGOTHIC solver Versions 1.2 and 4.1 for representative LST tests in both the LOCA configuration and in the MSLB configuration. The tests covered a range of boundary conditions. Three tests in the LOCA configuration were examined. Test 219.1 had both a dry and a wetted shell and had helium injection. Test 214.1 had natural and forced convection in the annulus. Water coverage

for test 216.1 varied from 75 to 25 percent (in quadrants). Test 222.4, in the MSLB configuration, had an initial steam blowdown (lasting about 60 seconds) with an elevated 7.6 cm (3 in) pipe.

Assessment of Variation in LST 219.1 Calculations

Test 219.1 was performed at a constant steam flow and with forced-air cooling. The vessel pressure came to steady state under constant steam flow without any PCS water on the vessel (219.1A). Helium was then injected and the vessel pressure allowed to come to steady state again (219.1 B). The PCS water was then turned on and the vessel pressure steadied to a third level (219.1 C).

While the vessel is dry, the measured pressure lies between the predicted pressure of the two versions. When the PCS water is turned on, the two versions predict nearly identical pressures, with both about 55 kPa (8 psi) higher than measured.

From 0 to 7600 seconds, Version 4.1 and 1.2 predict approximately the same vessel pressure. At about 7600 seconds, the two versions begin to diverge. An examination of the steam flow boundary condition by Westinghouse showed that, up to 7600 seconds, the model used as-measured steam flows which cyclically varied. From 7600 seconds to about 16,600 seconds, a constant steam flow was input. Once the input steam flow boundary condition became constant, small differences in circulation due to code version differences began to have an observable effect.

From about 7600 seconds until the water is turned on (at about 34,000 seconds), the predicted vessel pressure for Version 4.1 was lower (maximum of 7-percent difference) than both the Version 1.2 predicted pressure and the measured test pressure. After the water was turned on, Version 1.2 and 4.1 predicted approximately the same vessel pressure (Version 4.1 was slightly lower), and both versions predicted pressures about 55 kPa (8 psi) higher than measured.

The difference in calculated pressure between the two versions was consistent with the difference in air concentration distribution during the wet phase C. The calculated air concentrations for 219.1 A and B showed that the vessel was somewhat more uniform for Version 4.1 (more air above deck, less air below deck than Version 1.2). For 219.1C, the calculated air and helium concentrations for both versions were very similar. In both code versions, the trends in air and helium concentrations were similar. During the dry phase A, it was noted by Westinghouse that the pressure predicted by Version 4.1 was less than that by Version 1.2, even though Version 4.1 predicted more air above the operating deck. Although the external surface was completely dry, the effect of different annulus cell-centered velocities was more pronounced in Version 4.1. With a completely dry external surface, the resistance to vessel heat removal in the annulus was dominant relative to the condensation resistance, so the different predicted steam concentrations in the two versions had a relatively weak effect on the calculated pressure. In phase C, with water on the outside surface, the external velocity effect was nearly eliminated, and Version 4.1 predicted lower pressure consistent with relatively higher above-deck steam concentrations.

Based on results from LST 219.1, Westinghouse concluded that the net effect of the code changes was not significant for input models that are dominated by an imposed boundary

condition. Imposed boundary conditions that can be postulated to dominate the calculation, on the basis of the LST 219.1 evaluation, include time-varying steam flow rates, and actuation of the PCS.

Assessment of Variation in LST 214.1 Calculations

The initial part of test 214.1 was natural convection in the air annulus (214.1 A); the second part was forced convection in the air annulus (214.1 B). The steam flow rate was nearly constant throughout the test with some cyclic variation. Water coverage was measured at the elevation of the outside gutter where the excess water was collected. The water coverage varied in each phase. Based on test observations, Westinghouse believes that the water coverage actually changed continuously from the beginning of the test until coverage was actually measured. The input model used constant coverage values.

The pressures in the two code versions begin to diverge at the beginning of the transient. For this test the predicted vessel pressure for Version 4.1 was always higher than the predicted vessel pressure for Version 1.2, and after the step change in input water coverage, both versions predicted pressures 14 to 34 kPa (2 to 5 psi) higher than measured. The difference in relative code version results was not significantly affected by the change in external air flow when the fan was turned on (at about 9000 seconds). The pressure results were consistent with the non-condensable prediction. Version 4.1 has less axial non-condensable stratification than Version 1.2. Since the volume of the below-deck region is only about 20 percent of the total volume, a change in air content below deck results in a relatively smaller change in air content above deck.

The relatively constant input steam flow in this test may have allowed the differences in code versions to be manifested in the air distribution early in the transient and, thus, in the predicted pressure. Both code versions predicted pressures higher than measured once the input water coverage (at 3000 seconds) was set to the measured value.

Assessment of Variation In LST 216.1 Calculations

Test 216.1 had a relatively constant steam flow and forced-air cooling. The water was distributed over three quadrants for the first part of the test (216.1A) and over one quadrant for the second part of the test (216.1 B).

The predicted vessel pressures for Versions 1.2 and 4.1 were very similar, with Version 4.1 being slightly higher and both 28 kPa (4 psi) higher than measured. For 216.1 B, Version 4.1 predicted vessel pressure about 7 percent higher than Version 1.2, with 1.2 55 kPa (8 psi) higher than measured and 4.1 83 kPa (12 psi) or more higher than measured. Version 4.1 predicted more air at dome elevations and less air at lower elevations (operating deck and below-deck) than Version 1.2. Version 4.1 predicted slightly less stratification than Version 1.2.

Examination of the results for LST 216.1 again showed that with a relatively constant steam flow, the differences between the two code versions are manifested early in the calculation in a slightly lower stratification gradient, consistent with the variation in pressure.

Assessment of Variation in LST 222.4 Calculations

Test 222.4 had a steam blow down (lasting about 60 seconds), followed by a steady steam flow (222.4A). The steam flow rate was then increased and allowed to come to a second steady state (222.4B). Steam was injected upward into the vessel through a 7.6 cm (3 in) diameter nozzle 1.8 m (5.8 ft) above the operating deck. The non-condensable measurements showed this test had a relatively uniform axial air concentration, due to the high kinetic energy of the jet.

Version 4.1 predicted a lower vessel pressure than Version 1.2 throughout the transient, although the differences are small during phase A. The differences between the non-condensable predictions for Version 1.2 and 4.1 were small and had the same trend for the four elevations where measurements were taken. At 15,000 seconds, Version 4.1 began to deviate from 1.2, with Version 4.1 calculating slightly less air above the operating deck, consistent with the lower pressure in Version 4.1. Both Versions 1.2 and 4.1 predicted significantly more non-condensable axial stratification than were measured, consistent with the lumped-parameter code momentum bias.

The stratification above the assumed break node, inherent in multiple-node lumped-parameter models, was apparent in both code versions.

Conclusions

Based on a comparison of solver Version 1.2 and Version 4.1, (LST calculations for 219.1 compared to 214.1 and 216.1) the net effect may be less significant for code input models that are dominated by an imposed time-varying boundary condition. Imposed boundary conditions that can dominate the solution include changing steam flow rates and actuation of the PCS.

The lumped-parameter momentum bias in open volumes, represented by multiple lumped-parameter nodes, is apparent in both code versions. Thus, it is valid to impose stratification using lumped-parameter biases, as is done for MSLB, independent of code version.

The following remarks apply to the LST calculations presented:

- Results from LST calculations showed the calculated pressures using solver Versions 1.2 and 4.1 were both higher than the measured pressures when water was applied to the vessel shell, as is done in the AP600 containment evaluation model.

- LST calculations can be sensitive to code changes since the tests lack a flowpath into the simulated steam generator compartment, leading to non-prototypically high steam and air gradients between the above- and below-deck regions. Because of the sensitivity of condensation rate to non-condensable content, small increases in calculated circulation from the air-rich vessel heel (region below deck) affected calculated pressure by as much as 7 percent in the LST.

- The "ccvel" subroutine was upgraded, in Version 4.1, to improve the calculation of the cell-centered velocity associated with lumped-parameter fluid nodes. The cell-centered velocity is only used in calculating the PCS heat and mass transfer in the external annulus. The input decks for the evaluations of the effects of code versions discussed in this report differed only in the inertial length input for the external annulus nodes. The

resulting cell-centered velocities calculated by the two versions were verified to be similar, so it can be concluded that the ccvel changes had no significant impact.

- The LST tests were superheated, so the drop model improvement had no effect.
- The climes did not experience dryout, so the differences noted for LST results were not affected by the clime dryout error correction.

Impact of WGOTHIC 4.1 on Separate-Effects Test Calculations

The separate-effects tests used to validate the selection of heat and mass transfer correlations have been evaluated by Westinghouse using two approaches: (1) conversion of measured data to non-dimensional parameters for direct comparison to correlation results, and (2) using simple WGOTHIC once-through channel models.

For most of the tests, where the variation along the test channel (in the direction of flow) was small compared to the parameter of interest and sufficient data regarding measured boundary conditions were available, a linear variation along the channel length may be assumed. The measured data may be converted to non-dimensional parameters by hand calculations and compared directly to correlation results. Test series in this category are Eckert and Diaguila, Westinghouse dry flat plate, Westinghouse LST dry external heat transfer, Gilliland and Sherwood, Westinghouse flat plate evaporation, University of Wisconsin condensation, and Westinghouse LST internal condensation. The code changes do not affect these separate-effects test evaluations used in selecting heat transfer correlations.

For two of the separate-effects test series, boundary conditions (such as test section air flow) were not provided, or there was significant variation along the channel. For these tests, the WGOTHIC code was used to model a one-dimensional series of nodes along the section length. The simple WGOTHIC models also served to verify the correct implementation of the selected heat transfer correlations. The two series, for which data evaluation was based on a simple WGOTHIC channel model, are the Hugot and the Siegel and Norris tests.

As further verification of the correct implementation of correlations in the WGOTHIC code, simple once-through channel models were documented as part of the Westinghouse quality assurance program for the following test series: Westinghouse dry flat plate, Westinghouse flat plate evaporation, and University of Wisconsin condensation.

The code upgrades did not change the PCS heat and mass transfer correlations. Westinghouse concluded that none of the separate-effects tests used as a basis for WGOTHIC models of PCS heat and mass transfer were significantly affected by the code upgrades. Therefore, the basis for selecting PCS heat and mass transfer correlations was unaffected by the upgrade to WGOTHIC 4.1.

21.6.5.7 AP600 DBA Evaluation Methodology

21.6.5.7.1 Key Assumptions

This section presents an evaluation of the Westinghouse evaluation methodology (EM) for the analysis of DBA events (both the large break LOCA and the MSLB) in the AP600 containment. The technical adequacy of the key assumptions in the Westinghouse DBA EM is addressed. The assumptions underlying Westinghouse's WGOTHIC AP600 EM (Ref. 21.6.5.7.1) can be categorized as falling into four technical areas:

(1) containment flow and circulation characteristics
(2) mass and energy releases into the containment
(3) heat transfer to internal structures
(4) heat transfer through "clime" to ambient (environment)

The following subsections list the key assumptions made in each of these areas by the WGOTHIC EM analyses for DBA LOCA and MSLB events in the AP600 containment.

21.6.5.7.1.1 Containment Flow and Circulation Characteristics

- The WGOTHIC lumped-parameter model is applicable, which completely dissipates break momentum in the break node.

- AP600 containment non-condensable and steam distribution will be well mixed (nearly homogenized) as a result of the turbulent blowdown, and remain so throughout the peak pressure period (to around 1,200 seconds). In the long-term period (3 to 24 hours), heat transfer through the AP600 containment shell will not be significantly degraded due to the buildup of non-condensables near the condensing surface.

- Loss coefficients for intercompartment vents and junctions are constant over all phases of the transient from turbulent blowdown to relatively quiescent long-term conditions.

The justification for these assumptions is discussed in Section 21.6.5.7.2.

21.6.5.7.1.2 Mass and Energy Releases Into the Containment

- The Westinghouse mass and energy release procedure is appropriate for the AP600. This procedure neglects the reduction in containment pressure during the LOCA refill phase (30 to 90 seconds after break initiation — when break releases are negligibly small) to conservatively maximize the pressure during the LOCA peak pressure phase (90 to 1200 seconds), consistent with SRP 6.2.1.3.

- ADS stage 4 valve actuation during the most limiting LOCA events occurs at approximately 1000 seconds after initiation of the break.

21.6.5.7.1.3 Heat Transfer to Internal Structures

- Condensation and convective heat transfer in dead-ended compartments is "turned-off" after 30 seconds (i.e., after the end of the turbulent blowdown phase) to address

uncertainty in the global circulation pattern in the AP600 and modeling limitations associated with the WGOTHIC lumped-parameter model.

- All compartment floors, including the operating deck, are removed as potential heat sinks to account for both the continuous flooding of most of the below-deck compartments and the detrimental effects of stratification on condensation and heat transfer phenomena, thereby reducing the available heat sink utilization in the AP600 EM containment below and above deck.

- Fifty percent of LOCA liquid break release is arbitrarily converted into drops (although as little as 5 percent conversion of liquid is sufficient to keep the containment atmosphere at saturated conditions). Sensitivity studies performed by Westinghouse have shown that this procedure maximizes the peak containment pressure.

- The Uchida-correlation is applicable for all containment internal heat sink surfaces and covers all transient phases from blowdown to long-term. The containment shell and dome region are modeled with the AP600 specific mass and heat transfer correlation, as described in Section 21.6.5.6.

- A small value (20 mils) is assumed for the gap between steel and concrete for steel jacketed concrete heat sinks inside containment.

- The condensate film is stripped at the polar crane rail and internal stiffener ring and instantaneously transported to the sump. A similar approach is applied for condensate reaching the operating deck elevation.

- The initial conditions for pressure, temperature, and humidity inside the containment are assumed to be uniform throughout containment.

21.6.5.7.1.4 Heat Transfer Through Clime to Ambient Environment

- Initiation of PCS water is delayed by 337 seconds after break initiation to conservatively account for the time required to establish steady-state PCS film flow down the length of the containment sidewall.

- The heat and mass transfer package (refer to Section 21.6.5.4.2) embedded in the clime model covers all transient phases from blowdown to long-term.

- Free-convection heat and mass transfer at the inside steel shell surface is conservatively assumed for both the LOCA and the MSLB to eliminate the potential bias resulting from overly high predicted velocities in the lumped-parameter modeling approach.

- The "evaporation limited" flow model for the PCS film is conservative.

- A 2-D enhancement multiplier is applied. This multiplier accounts for the expected non-uniformity of thermal conditions for wet and dry outside steel shell surface fractions (striping) over the LOCA long-term phase (greater than 3 hours).

21.6.5.7.2 Containment Flow and Circulation Characteristics

Westinghouse has described the post-LOCA circulation patterns in the AP600 containment as follows:

> Following the LOCA blowdown phase, natural circulation is expected to be the dominant mechanism for containment atmosphere circulation. Once the ADS Stage 4 valves are actuated, hot steam is expected to rise from the SG compartments up to the containment dome region. As the hot steam contacts the containment inside steel shell surface it is being condensed and the cooled non-condensable gas-steam mixture will flow downward along the containment shell back towards the operating deck and CMT room floor, where the mixture may re-enter the steam generator cubicles through the open doorways. The cool mixture will be entrained with the hot steam source from the ADS valves and rises again.

Since the AP600 EM utilizes the lumped-parameter approach, predictions of flows and directions along the flow paths to match the anticipated global circulation pattern are of limited value. Open doorways in the east and west SG cubicles at the operating-deck elevation play a crucial role in Westinghouse's arguments for a sustained global circulation. Westinghouse points out that similar circulation patterns also develop over the height in the west containment quarter; however, Figures 4-130 (a) and (b) in WCAP-14407 do not readily confirm this.

Westinghouse also refers to several flowpaths between volumes above the elevation of the reactor pressure vessel upper head flange and the operating-deck elevation which supposedly support atmospheric circulation between the regions below and above the operating deck. As a result, Westinghouse expects a homogenization of the atmosphere above the operating deck such that no transverse steam/air concentration gradients exist. However, the figures in WCAP-14407 cited by Westinghouse do not show the time histories of steam/air concentrations across the operating-deck elevation. Rather, the figures show flow rates and steam concentrations, which are not representative of typical operating-deck volumes.

Additional information regarding these issues has been provided by Westinghouse in the context of Westinghouse's responses to items 480.1085F and 480.1086F in Westinghouse letter DCP/NRC1216, dated January 16, 1998. This supplemental information provides the results for horizontal temperature and concentration profiles from an analytical model, plus the additional evaluations of two LST tests with regard to horizontal temperature profiles across the LST facility at several elevations. However, none of this information concerns the natural circulation pattern discussed above.

Westinghouse's arguments and assertions that global circulation in the dome through doorways in the SG cubicles will homogenize the containment atmosphere are incomplete for the following reasons:

- The LST facility does not account for the doorway opening as a flowpath. Therefore, LST data cannot be used to confirm that circulation loops are being sustained given the prevailing sources, sinks, channel and doorway openings, and respective flow resistances along the circulation paths involving both east and west SG compartments in the AP600.

- Above-deck circulation loops require the existence of axial and transverse gradients as driving forces. However, Westinghouse's responses to the open item cited above indicate that no vertical and horizontal gradients exist (e.g., the dome is perfectly homogeneously mixed), except in the plume region and in the region close to the inside steel surface. Based on this supplemental information, the existence of such circulation patterns seems implausible, if in fact both experimental and analytical results are applicable to the AP600 conditions under consideration. A perfectly homogenized containment atmosphere would invalidate the simultaneous existence of the circulation loops.

- Westinghouse discusses circulation loops throughout WCAP-14407; for example, in Sections 4 and 9, Westinghouse asserts that atmosphere circulation through flow paths connecting the below- and above-deck regions homogenize the atmosphere mixture so that "no significant concentration gradient exists across the operating deck" (p. 4-323, bottom).

- Even if the circulation loops are sustained over time, these loops only affect the dome volume in the vicinity of both SG cubicles, which is a rather small region compared to the total dome volume. Most of the dome volume would not be affected by the type of global circulation through the SG cubicles cited by Westinghouse.

- The lumped-parameter network model for the above-deck region tends to artificially overpredict flow rates and circulations.

- The long-term steam concentration of 45 percent predicted by the WGOTHIC AP600 EM for most of the dome region appears to be high in view of the effective PCS and may be the result of

  (a) the artificially high predicted circulation
  (b) numerous conservatisms implemented into the evaluation model
  (c) or a combination of both

Because of the concerns raised during the initial review of the information provided by Westinghouse about its WGOTHIC EM approach to circulation and stratification, the NRC staff issued RAI 480.1085F. This open item questioned whether WGOTHIC treated heat transfer in a sufficiently conservative manner to accommodate temperature stratification and horizontal concentration gradients.

Westinghouse responded to the open item by developing a theoretical model for the gross circulation in the AP600 containment, supplemented by additional experimental evidence on horizontal temperature profiles from LST tests 220.1 and 217.1. Some elements of this approach were first presented at a December 11-12, 1997, ACRS meetings in Rockville, MD.

The Westinghouse theoretical model consists of three regions: region 1, the plume, including an upper region below the steel dome whose boundaries are undefined; the "recirculating stratified" Region 2; and the negatively buoyant, turbulent gas boundary layer. The region just above the

operating deck is undefined. The model, as presented, is based on first-principle elements, such as a jet/plume and a negatively buoyant boundary layer, coupled by a top feed and bottom source with some entrainment into both the plume and the boundary layer. Westinghouse did not list the thermal boundary conditions, however, during discussions held with the staff reviewers, Westinghouse indicated the boundary conditions were obtained from the measured LST results. This theoretical model assumes that a Gaussian plume with a half angle of 7.5° reaches the dome even though the plume must pass through an equipment-filled steam generator compartment. As time progresses, an air-enriched layer will move downwards from the dome and jet buoyancy will decrease. Strictly speaking, the Westinghouse model should apply only to momentum-controlled jets, although most break jets would be considered buoyant once blowdown ends and the primary system and containment pressures have equilibrated.

Although limited and oversimplified, this approach constitutes a working model with quantitative results displayed in several tables in the attachments. Westinghouse applied this boundary layer approach to two LST tests (test 220.1 and test 217.1). The comparisons presented in Subsection VIII of Attachment 3 reveal a surprisingly excellent agreement with the measured C_∞ value, with C_{BL} consistently slightly higher than the comparable measured values.

The model described in Westinghouse letter DCP/NRC1216, dated January 16, 1998, has been revised to address staff concerns with the material as presented, and has been incorporated into WCAP-14407 as Appendix D to Section 9. This closes Open Item 21.6.5-10 concerning flow field stratification and stability and jet effects in the non-condensable stratification and separation as modeled in WGOTHIC and the AP600 EM.

Several Westinghouse observations warrant further discussion. One is that for the compared conditions a substantial stratification exists low in the LST facility (between regions D and E) for both tests. Westinghouse chose to qualify these results as showing "some level of stratification" and attribute it to LST distortions (e.g., no flow connection from the simulated SG compartment). The staff believes that the elimination of global circulation throughout the LST did, in fact, lead to the accumulation of non-condensable air atop the operational deck. This phenomena is a result of continuous condensation at the inside steel shell and also applies to the AP600. Moreover, the test data presented indicate that the rising plume is unable to entrain this accumulated air layer (as claimed by Westinghouse in Section 9 of WCAP-14407). WGOTHIC, however, conservatively treats this effect by not taking credit for the heat transfer from the operating deck itself; consequently, the buildup of air on or the entrainment of air from the operating deck is not needed (see Section 21.6.5.8.3 of this report).

21.6.5.7.3 Break Scenarios

21.6.5.7.3.1 Loss-of-Coolant Accidents

The double-ended cold-leg guillotine (DECLG) rupture is the most limiting design basis loss-of-coolant accident for the AP600 containment pressure response as it postulates the break of a high-energy primary coolant pipe. A mixture of water, steam, and droplets is released into the break compartment and propagates from there to neighboring compartments below the operating deck.

Westinghouse identified the east steam generator compartment as the break compartment with the break elevation at 100 ft (i.e., beneath the steam generator). The AP600 relies on passive means to mitigate the containment pressure increase for DBAs. Westinghouse assumes that the non-safety-grade containment fan coolers do not operate.

The liquid released from the break and collected as condensate fills the reactor cavity, lower portions of the SG and the RCDT cavity during the blowdown of the primary system. As a result of the continued steam release from the break and the resulting condensation, and the ongoing operation of the passive core cooling systems (PXS), the liquid level below the operating deck continuously rises during the containment transient, reaching the CMT room elevation at about 15,000 seconds. At that point 1515 kL (53,500 ft^3) water has accumulated in the lower region to reach the CMT floor elevation. This means that the flooded compartments and their interconnecting vent flowpaths are not available as long-term heat sinks and flowpaths after 15,000 seconds.

The staff acceptance of the lumped-parameter representation in the AP600 EM is based on the condition that the containment atmosphere is relatively homogenous as a result of the break momentum. In letter DCP/NRC1314 dated March 25, 1998, Westinghouse provided additional information for large-break LOCAs, including a 7-inch CMT balance line break. The momentum, as denoted by the Froude number, for these breaks indicated that the containment atmosphere is relatively homogenous even for these smaller breaks. Therefore WGOTHIC results, based on the lumped-parameter representation in the AP600 EM, are acceptable for addressing the LOCA break spectrum to determine the limiting break.

The staff has completed its review of the methods and assumptions used to calculate the LOCA mass and energy releases. The LOCA mass and energy releases are conservatively calculated with an approved model and are consistent with the guidelines in the SRP. This source term treatment in the AP600 EM is acceptable. Therefore Open Item 21.6.5-21, concerning the models for LOCA evaluations, is closed.

The Office of Nuclear Regulatory Research (RES) presented the results of preliminary analyses for a coupled containment-to-RCS LOCA evaluation during the June 12, 1998, meeting of the ACRS Thermal-Hydraulic and Severe Accident Subcommittee. The coupled containment-to-RCS analyses were performed with the RELAP5 computer program using a model of the AP600. A special model was added to RELAP5 to simulate the AP600 PCS and containment response to a LOCA. The PCS heat transfer was characterized from NRR CONTAIN analyses. The RES analysis method differed from the staff-approved method used by Westinghouse for its DBA studies. The RES study included the containment-to-RCS interaction in determining the LOCA mass and energy releases. The RES analyses demonstrated the conservatism of the approved methodology that is used for the evaluation of the containment response to LOCAs. The RES analyses indicated that the energy released to containment, using the approved method, controls the calculated containment pressure response. The analyses are documented in RES Assessment Report RPSB-98-07, "Containment/RCS Analysis of a Large Break Loss-of-Coolant Accident in the AP600 Using RELAP5/MOD3," G. N. Lauben, August 1998.

21.6.5.7.3.2 Main Steamline Breaks

Westinghouse performed a parametric study of 16 cases (four breaks at four initial power levels) to identify the worst-case scenario for the MSLB accident. The spectrum of breaks sizes examined included the following:

- A full double-ended pipe rupture downstream of the steamline flow restrictor, with a nominal throat area of 0.129 m^2 (1.388 ft^2). The reverse flow from the intact steam generator is conservatively assumed to be controlled by the flow restrictor.
- An intermediate-size double-ended break with an area of 0.037 m^2 (0.4 ft^2)
- A small double-ended break with an area of .009 m2 (0.1 ft^2)
- A split rupture representing the largest break which does not generate a steamline or feedwater isolation signal

Westinghouse has shown that the availability of offsite power results in the most limiting case and maximizes the mass and energy releases from the break. All cases which were analyzed to determine the limiting MSLB event assume the failure of one MSIV. Westinghouse identified the full double-ended rupture at 30 percent power as the limiting MSLB break scenario for peak containment pressure. This is the result of a combination of mass and energy release and a high Froude number, indicating a momentum-driven jet.

The MSLB mass and energy release analyses were performed using an approved model and in compliance with the guidance provided in the SRP and NUREG-0588. A value of 8 percent revaporization is used for the MSLB analyses consistent with acceptable guidelines developed by the staff in NUREG-0588.

Limiting MSLB Break Location

The path of the main steamline begins at the top of the SG, where it bends 180° and follows a downward path to the CMT room. It then bends 90° to cross through the CMT room and exits through a containment penetration. A break at any position along this pathway would release very high energy steam into the containment over a short time.

Westinghouse examined possible break positions in the steamline above operating deck and possible break positions in the steamline in the CMT compartment to confirm the AP600 containment design adequacy and to develop a conservative model which, for the purpose of a conservative containment pressure calculation, accounts for the potential effects of circulation and stratification inside the dome.

Westinghouse performed analyses with the evaluation model, including the stratification heat sink biases specifically developed for the DBA LOCA, for the break positions listed above.

The same lumped-parameter nodalization developed for the LOCA evaluations is used for the MSLB evaluations, including the treatment of dead-ended compartments and heat sinks. The only difference is the selection of the node for the break source. For MSLB beaks above the operating deck the break location initially proposed by Westinghouse was a node just above the

operating deck. The below-deck MSLB break was modeled in the CMT room. As the CMT room has large heat sink surfaces and provides access to the other below-deck compartments, the WGOTHIC-computed peak pressure was 1.6 psi below the value obtained for the MSLB break position above the operating deck. Subsequent to this sensitivity study, and in response to staff concerns, Westinghouse moved the MSLB break location to an elevated node which represents the highest steamline elevation for SSAR analyses.

The MSLB mass and energy releases are calculated with an approved model and are consistent with the guidelines in the SRP. This source term treatment in the AP600 EM is acceptable. Therefore Open Item 21.6.5-21, concerning the models for MSLB evaluations, is closed.

21.6.5.7.4 Conservative Input Parameters

General modeling guidance for GOTHIC/WGOTHIC is provided in Section 16 of the NAI's Gothic User's Manual ("GOTHIC Containment Analysis Package Technical Manual, Version 4.0", NAI-8907-06, Revision 3, September 1993). These general thermal-hydraulics modeling recommendations are intended to give best-estimate model performance. Westinghouse modified selected model parameters to add conservatism to the AP600 DBA evaluation model to be consistent with the guidelines in the SRP and RG 1.70.

21.6.5.7.4.1 Break Mass and Energy Releases

Conservatively high mass and energy sources are used in the evaluation model. The LOCA refill period, during which there is no release, is conservatively eliminated. The staff accepts the use of conservatively high mass and energy release as appropriate for use in an evaluation model.

The blowdown is specified using constants or forcing-function tables to describe the transient pressure, enthalpy, and fluid flow condition. Since the boundary condition is connected to a lumped-parameter volume, it is not crucial that the boundary pressure trace the actual pressure because the source momentum is dissipated.

21.6.5.7.4.2 Break Elevation and Direction

For the LOCA, the location, elevation, and direction of the break were taken to maximize the peak pressure by limiting interactions with the below-deck compartments. For the MSLB, Westinghouse originally assumed a break location which the staff considered to be unacceptable for the design application. Westinghouse revised its methodology to use the highest possible elevation for the MSLB. The staff finds both limiting-break locations acceptable.

21.6.5.7.4.3 Break Density and Droplet Diameter

Numerous researchers have documented the effect of entrained drops in the containment atmosphere during and after DBA LOCA events in the following references:

- Gido, R.G., Koestel, A., LOCA Generated Drop Size Prediction, A Thermal Fragmentation Model, ANS 1978 Winter Meeting., Washington, D.C., November 12-17, 1978.

- Koestel, A., Gido, R.G., LOCA Drop Size Estimates, NUREG/CR-1607, LA-8449-MS, August 1980.

- Almenas, K.K., Marchello, J.M., The Physical State of Post-Loss-of-Coolant-Accident Containment Atmospheres, Nuclear Technology 44 (1979), 411-428.

- Gido, R., Lamkin, D., Koestel, A., Mechanistic Dry-Pressure-Containment LOCA Analysis, NUREG/CR-2848, January 1983.

The following observations can be made about the effect of entrained drops in the containment atmosphere during and after DBA LOCA events from the above literature:

- As a result of the drops, the atmosphere thermodynamic conditions are completely saturated.

- The drop-mechanistic approach results in an approximately 5 percent lower atmospheric pressure and slightly lower temperature than the approach used in CONTEMPT, which assumes that blowdown is instantaneously transported to the sump and atmospheric liquid rains out.

- The drop-mechanistic approach results in an approximately 2 percent lower atmospheric pressure and significantly lower atmospheric temperature in comparison with the partition approach. The partition approach considers the possibility that the break flow is partitioned into separate liquid and vapor phases. The liquid is assumed to go directly to the sump whereas the vapor phase is introduced into the containment vapor/air volume to determine pressure and temperature.

A nominal liquid-drop diameter of 0.01 cm (0.00394 in) is generally used for blowdown analysis based upon experimental test results. The influence of drop diameter on the containment pressure response should be examined over a range of 0.001 - 0.10 cm (3.9E-4 to 3.9E-2 in). The forced entrainment drop diameter in GOTHIC specifies the drop size to be used in lumped-parameter volumes when a portion of the wall condensation rate representing condensation dripping from ceilings and suspended equipment is converted to drops. The nominal, or default, value of 0.253 cm (0.1 in) is based on experimental data comparisons of GOTHIC with blowdown experiments. The value used by Westinghouse is 0.0002 cm (8.5×10^{-5} in). This is relatively small, but acts together with the large droplet fraction (50 percent of the break source) assumed in the EM. Sensitivity studies performed by Westinghouse in WCAP-14407 showed that the droplet assumptions in the AP600 EM are conservative.

When the containment atmosphere is superheated, a portion of the condensate will revaporize from passive heat sink surfaces. A maximum revaporization fraction of 0.08 (i.e., a mass condensation fraction of 0.92) is recommended in NUREG-0588 and used by Westinghouse for the MSLB.

21.6.5.7.4.4 Hydrogen Release

No hydrogen is explicitly considered in the constituents of the break flow. However, the energy equivalent to 1 percent of the zirconium in the fuel region is included in the break energy. This is consistent with the guidance provided in SRP Section 6.2.1.3. The staff therefore finds this model acceptable.

21.6.5.7.4.5 Heat and Mass Transfer

A survey of the state-of-the-art in condensing heat transfer modeling was performed by Economos et al. ("Condensation Heat Transfer Modeling for Containment Environmental Response Calculations - A Reappraisal for the Standard Review Plan," Brookhaven National Laboratory, Upton, NY, for US NRC, Office of Nuclear Reactor Regulation, Contract No. DE-AC-02-76CH00016, June 1987, Vol. I and Vol. II.) examining the applicability of these methods for containment response to a postulated MSLB. The objective of this study was to develop a technical basis for possible use in improving the CONTEMPT series of computer codes, specifically in the area of condensing heat transfer in the context of improving the SRP. This study presented an update of the investigation by Slaughterbeck for AEC in 1970. From the review, the following three correlations were selected for comparison in this study: (1) the Uchida correlation, (2) the Almenas correlation, an empirical model developed from German BMC and HDR experiments, and (3) the Gido-Koestel (G-K) correlation, a phenomenological method based on boundary layer concepts. The latter two correlations were implemented in CONTEMT4/MOD6 to allow comparative studies of the correlations with the approved licensing Uchida-model. BNL performed 30 computations (with numerous parameter variations) for CVTR (test 3), BMC (D-15), and the HDR (V21, V42, V44) experiments, as well as for a typical PWR application. For the simulations lumped-parameter models were used ranging from one to six nodes. The Uchida correlation was computed with 8 percent revaporization, unless turned off for the purpose of parameter variation.

The computational results were compared with the data and with each other. A major outcome of these comparisons was that neither of the two selected new heat transfer models provided any substantial improvement in predictive capability over the Uchida-correlation, the currently approved licensing model.

George and Singh ("Separate Effects Tests for GOTHIC Condensation and Evaporative Heat Transfer Models," Proc. 3rd Intl. Conf. Containment Design and Operation, Toronto, Canada, October 19-21, 1994, Vol. I, Session 8) evaluated the application of the GOTHIC code for two separate effect test facilities:

- The University of Wisconsin condensation tests, which are also a major part of Westinghouse's data basis for the validation of heat and mass transfer correlations (See Westinghouse WCAP-13307, "Condensation in the Presence of a Noncondensable Gas: Experimental Investigation" and Huhtiniemi, I. K., "Condensation in the Presence of

Noncondensable Gas, Effect of Surface Orientation," Ph.D. Thesis, Dept. Of Nuclear Engineering and Engineering Physics, University of Wisconsin, Madison, WI, USA, August 1991).

- Pacific Northwest Laboratories evaporation tests (Bagaasen, L. M., 106-AN Grout Pilot-Scale Test HGTP-93-0501-02, PNL-8618/UC-510, 1993) on a grout mixture used for long-term storage of hazardous waste from a heated pool to a superheated atmosphere.

University of Wisconsin Condensation Tests on Vertical and Inclined Flat Plates

The University of Wisconsin experiments reported results for the flat-plate geometry. Parameters varied in the experimental test series included the inlet temperature, steam partial pressure, flow rate, and plate inclination angle from the horizontal. An overview of these experiments was provided by Westinghouse in WCAP-13307. The heat transfer rates and coefficients were deduced from the measured temperature rise and flow rate of the oil coolant on the coolant plates at the backside of the condensing test plate and from the measured temperature profile near the condensing surface of the test plate. Local heat transfer coefficients were obtained at seven axial locations and then combined to an average heat transfer coefficient for the total test section.

The GOTHIC model for this test section used a lumped-parameter single volume (for consistency with the common lumped-parameter approach), large computational nodes, and bulk conditions to compute heat and mass transfer. As the test transients were sufficiently slow relative to the air/steam turnover rate in the test section, the computed results could be considered quasi-steady and comparable to the data. A total of nine experiments, all with vertical plate orientations, were simulated with GOTHIC, covering a velocity range from 1 to 3 m/s (3.3 to 9.8 ft/s), a steam mole fraction from 0.312 to 1.000 and temperatures from 69.8 °C (157.6 °F) to 97.2 °C (207 °F).

The Uchida correlation returns larger than measured heat transfer coefficient for velocities around 1 m/s (3.3 ft/s), while the G-K correlation provided a lower bound. Both correlations simulate the correct trend for increasing steam concentration. At higher velocities 2 to 3 m/s (6.6 to 9.8 ft/s), the Uchida correlation (although its values remain unchanged) tends to underpredict the heat transfer coefficient with increasing velocities. The G-K correlation provides an upper limit at 3 m/s and matches the data very well. The cutoff value of about 1500 W/m^2K of the Uchida-correlation at the velocity of 3 m/s (9.8 ft/s) is consistent with the upper limit set forth for licensing applications. The authors also mention that 1-D and 2-D GOTHIC models of the test section provided average heat transfer coefficients which were very close to those obtained from the lumped-parameter, single-volume model.

Pacific Northwest Laboratories (PNL) Evaporation Tests

The comparisons with the PNL evaporative pool tests confirmed GOTHIC's interfacial heat transfer logic and correlations. However, the authors point out the importance of selecting the proper heat and mass transfer correlations for the specific geometry and flow regime under consideration. In this context, it is noted that local variations are especially important under low-flow conditions and can only be accounted for with the distributed-parameter (subdivided) GOTHIC-model in the region above the evaporating surface.

Additional Test Data

The determination of heat transfer coefficients during the blowdown phase was a major objective of the early BMC tests and HDR experiments. The focus shifted toward the post-blowdown phase in the middle 80s, with added interest in severe accidents. Consequently, later HDR LOCA testing was expanded with numerous heat transfer blocks, fabricated from concrete, lead and steel and distributed at different axial positions throughout the HDR containment. Some blocks were mounted flush with concrete surfaces, others were stand-alone units (those located at the operating deck). To obtain meaningful results for medium- and long-term post-blowdown time periods, inverse multidimensional methods were applied to cope with continuously decreasing temperature differences over time.

The heat transfer coefficients were deduced from the HDR experiment T31.1 (Blowdown-Experiments in a Reactor Containment, Quick Look Report, Test Group COND, Experiments T31.1-3, (in German), Technical PHDR Report No. 57/85, Nuclear Center Karlsruhe, Germany, 1985) for three different time periods; 0 to 200 seconds, 200 to 600 seconds, and 10 to 120 minutes. The calculated heat transfer coefficients using the Uchida correlation for the same time intervals were evaluated. The measured and computed heat transfer coefficients reflect the containment behavior over time at different axial elevations. Thus, they supplement the separate-effects tests results discussed above.

The HDR experiment T31.1 heat transfer coefficients computed from the Uchida correlation were below the measured values for most of the test duration and at all elevations shown. They were especially conservative during the blowdown phase in the break compartment and the compartments in its vicinity. During the time period of 200 to 600 seconds they were either conservative by a factor of 2 or match about the data, depending upon position. During the long-term phase, 10 to 120 minutes, the Uchida correlation values were either a factor of 2 to 3 conservative over a certain time window, or non-conservative by about the same factor, depending upon the position in the HDR containment. With time, these discrepancies diminish.

21.6.5.7.4.6 Material Properties

Material properties are chosen in a conservative, manner with both heat capacity and thermal conductivity biased in the low direction. This conservative approach is acceptable.

21.6.5.7.4.7 Initial Conditions

Conservative values are used to specify the initial conditions consistent with the evaluation model approach. This is an acceptable approach. Tables 21.6-17, 21.6-18, 21.6-19 and 21.6-20 specify the conservative initial conditions selected for various physical quantities.

Westinghouse had originally proposed to use nominal conditions for the temperature distribution in the containment. This was Open Item 21.6.5-1. Since Westinghouse now uses conservative temperature values, Open Item 21.6.5-1 on the WGOTHIC AP600 EM treatment of temperature distributions in the containment atmosphere and the enclosing wall is closed.

The initial conditions have been identified and justified through sensitivity analyses to be conservative, therefore, Open Item 21.6.5-13, concerning the need to identify constrained variables and quantities and their impact on AP600 licensing calculations, is closed.

21.6.5.7.5 Geometry

21.6.5.7.5.1 Nodalization

The WGOTHIC model lumped-parameter nodalization was developed to minimize transport of steam to the containment shell, based on numerous nodalization sensitivity studies. Intercompartment flow is minimized with this modeling approach. Conservative characteristics of the model are summarized in Table 21.6-21.

The heat and mass transfer model for the shell is applicable at the scale of AP600 provided that the bulk steam concentration in the nodes adjoining the shell can be accurately or conservatively predicted. Bulk parameters are used in the heat and mass transfer correlations. When the containment is well mixed, the staff accepts the lumped parameter WGOTHIC model as adequate for calculating the peak containment pressure with the AP600 EM.

The question of how long the containment remains well mixed following blowdown is difficult to answer. For the MSLB, the break source is non-zero only during the blowdown. The peak pressure occurs near the end of the blowdown. The calculated peak pressure is the licensing criteria. Following the peak pressure, without a break source, the pressure will decrease. Therefore, the shell heat transfer in WGOTHIC is acceptable for the conservative MSLB calculations.

For the LOCA, the containment atmosphere will eventually stratify. However, at least up to the peak pressure, a buoyant plume will continue to drive mixing. In NSD-NRC-98-5526, dated January 16, 1998, Westinghouse indicates that the steam concentration in the containment will have uniform gradients in the horizontal direction except in the plume and in the boundary layer next to the wall. While the state which results is not well mixed, it is one for which lumped-parameter models are capable of calculating peak pressure in a conservative way, as discussed in Section 21.6.5.7.8.

To perform containment peak pressure design bases analyses, Westinghouse selected a lumped-parameter modeling approach with WGOTHIC for use in the AP600 containment evaluation model (EM). Chapter 4 of WCAP-14407 documents the lumped-parameter control volumes, junctions, thermal conductors types, and their associated quantities. Special modeling assumptions for each major compartment are listed in individual subsections. Cutaway views, perspectives views, and cross-sections of different planes supplement the descriptions, the tables of geometrical quantities, and input parameters listed in the respective subsections. Section 4.6 of WCAP-14407 summarizes the control volumes, describes their association with specific compartments and containment regions, and lists all junctions with their respective connected control volume nodes.

From this information, it can be discerned that there are three distinct regions in the AP600 containment EM. The first region comprises all compartments below the operating deck, the second region represents the large free-dome space and the associated infrastructure, and the third region is the dome steel shell with its neighboring regions (i.e., riser, steel baffle,

downcomer, and outer concrete (shield building) wall). The first two containment regions are modeled with WGOTHIC using the lumped-parameter representation in GOTHIC. The third region is modeled with the clime model. Section 21.6.5.4.2 provides details of the clime model, which was added to GOTHIC by Westinghouse to model the passive containment cooling system (PCS). The clime model connects the containment atmosphere to the shell by evaluating the mass and energy source term from the adjacent above-deck containment nodes to the interior shell surface (when dry) or the surface of the condensate film.

The regions below and above the operating deck are connected by a limited number of flow connections with small cross-sectional flow areas. These flow paths represent the two steam generator compartments and a few auxiliary openings, such as staircases and the elevator shaft. The flow paths allow for mass, momentum, and energy exchanges during the accident transients. The dome steel shell only transfers heat between the internal and external containment regions.

In response to RAIs 480.685 to 480.687, Westinghouse addressed the staff concern with the form loss and friction loss treatment in the AP600 EM, as identified in Open Item 21.6.5-11. The revised loss coefficients used in the AP600 EM were selected to be representative of the longer term, post-blowdown, flow rates expected, and were increased by a conservative factor. These same coefficients are used during the blowdown, and are conservative with respect to the pressure response. Sensitivity studies performed by Westinghouse show the selected loss coefficients have a weak dependency on the pressure. Open Item 21.6.5-11 is closed.

The below-operating deck compartments are modeled as single control volume nodes connected by appropriately defined junctions. No further subdivision is applied over the axial extension of these compartments, except for the SG wells to accommodate the stage 4 ADS valves. The large core makeup tank (CMT) room has been modeled as two nodes separated at the pinch line. This modeling choice provides the simplest representation possible in the lumped-parameter approach.

The large above-operating deck region has been "subdivided" into a network of lumped-parameter nodes with four axial levels and nine radial regions over the height of the vertical cylindrical section of the AP600 containment. Above that axial level, four radial nodes represent the curved dome shape for the remaining three axial levels. Each radial region represents a quarter segment of the total dome circumference. Thus, at first sight, the above-operating deck lumped-parameter network is an unusually detailed model. All containment external air flow channels are also axially subdivided into seven levels. However, the detailed azimuthal subdivision exercised inside the containment is not used for the flow channels external to the shell. One flow channel represents the total circumference of the riser and another the downcomer. This constitutes a lumped-parameter modeling approach emphasizes the axial direction and does not model for potential asymmetries in the circumferential direction.

As summarized in Chapter 4 of WCAP-14407, the nodalization of the AP600 containment is as follows:

- 90 control volumes to represent the total inner volume of the AP600 containment with all regions and compartments below and above the operating deck

- 10 control volumes to represent the PCS annular flow paths, ultimate heat sink, and the environment
- 247 junctions connecting the inside-containment control volumes
- 17 junctions connecting the PCS control volumes
- 9 junctions connecting boundary conditions
- over 200 thermal conductors categorized into about 50 different walls and conductor types with varying thickness and composition
- 60 climes to represent the containment steel shell and its associated two-sided heat and mass transfer source terms, including 4 dummy climes

Compartments below the operating deck are generally modeled as separate computational nodes. While the nodes above the operating deck are not necessarily bounded by walls, they are arranged in regular stacks and aligned based on structures and geometric changes (e.g., steam generator compartments and cylindrical section of dome). Adjacent stacks are defined such that they are aligned with the same elevations in compliance with the user guidelines provided in the NAI GOTHIC technical manuals. This modeling procedure reduces artificial flows induced by connecting nodes of different heights and elevations to a minimum and must be applied across the total model region of interest. Consequently, Westinghouse has consistently interfaced the inside containment nodalization structure to the climes and their associated elevations. In this way, a network of ordered stacks of lumped-parameter nodes inside the containment is coupled to similar stacks of climes representing PCS and associated volumes for riser and downcomer annuli.

This lumped-parameter nodalization scheme has evolved over time and jointly with the modeling of the LST experiments and the development of WGOTHIC distributed-parameter models. It has been tested for most of the LST experiments. Although originally developed for the DECLG LOCA, the same nodalization and assumptions are used for evaluation of MSLB accidents (see Table 21.6-21).

21.6.5.7.5.2 Break Source Momentum

The lumped-parameter model dissipates break source momentum in the break volume. Since increased momentum will promote mixing, this approach is generally conservative. However, momentum directing the break flow upward can potentially promote stratification, so use of the lumped-parameter model alone does not assure a conservative treatment of break source momentum effects. Since break source momentum becomes enmeshed with the entire lumped-parameter treatment, it is evaluated as a part of the lumped-parameter modeling approach.

21.6.5.7.5.3 Heat Conductors

Condensation and convective heat transfer in dead-ended compartments is "turned off" after 30 seconds (i.e., after the end of the turbulent blowdown phase) to address uncertainty in the global

circulation pattern in the AP600 and in the modeling limitations associated with the WGOTHIC lumped-parameter model.

All compartment floors, including the operating deck, are removed as potential heat sinks to account for both the continuous flooding of most of the below-deck compartments, and to account for the detrimental effects of stratification on condensation and heat transfer phenomena. This reduces the available heat sink utilization in the AP600 EM containment below and above deck.

One-dimensional heat conduction in slab geometry with a conservatively large air gap between steel and concrete is an acceptable modeling approach.

The Uchida correlation has been demonstrated to be appropriate for use on the internal heat sinks, therefore, Open Item 21.6.5-6, concerning the correlation used for conduction to internal heat sinks in the heat and mass transfer models used in WGOTHIC and in the AP600 EM, is closed.

Free-convection heat and mass transfer at the inside steel shell surface is conservatively assumed for both the LOCA and the MSLB to eliminate the potential bias resulting from the overly high predicted velocities using the lumped-parameter modeling approach. This closes Open Item 21.6.5-5, concerning the mass and heat transfer models used for the inside shell surface. The free convection mass and heat transfer models are acceptable for use in the WGOTHIC AP600 EM.

21.6.5.7.5.4 Pools and Below Operating Deck

Stratification in the break pool is included by vertical stacking of nodes. Evaporation from a pool is maximized and condensation minimized by noding to keep hotter liquid at the pool surface. Water pools are assumed to be homogeneous. Constant pool cross-sectional area is assumed. This is a conservative treatment which is acceptable to the staff.

21.6.5.7.6 Modeling Limitations

WGOTHIC automatically adjusts time step size during the transient solution, increasing the time step up to a maximum specified limit when the transient is nearly stable and decreasing the time step down to a specified lower limit when rapid changes or convergence difficulties occur. The adequacy of the time step should be determined from the mass and energy error provided in the output file. The User's Manual specifies a mass and energy error of less than 2 percent as acceptable for the determination of the maximum containment pressure response without need for a further reduction in time step size.

21.6.5.7.7 WGOTHIC Evaluation Model Validation

In Section 9.C of WCAP-14407 Westinghouse documented test results from a large set of international test facilities, The test results included comparisons with computational results from WGOTHIC. The staff review of this material reveals the following:

- The information provided by Westinghouse for the Battelle Model Containment (BMC) tests and HDR experiments is extensive. Most of the material presented shows H_2/He

concentration histories rather than the equivalent set of steam concentration histories, which were also available.

- Westinghouse participated in the International Standard Problem on the NUPEC test M-7-1 (Nuclear Power Engineering Corporation (NUPEC) System Safety Department, Specification of ISP-35, NUPEC's Hydrogen Mixing and Distribution Test - Test M-7-1, ISP35-027, Rev. 1, May 10, 1993), with a blind WGOTHIC prediction. The WGOTHIC results, with the predictions by other institutions, have been published by OECD in NEA / CSNI / R(94) 92. However, this experiment examined the mitigating effect of internal containment sprays. As the AP600 has no internal spray systems, this validation effort by Westinghouse for WGOTHIC does not apply directly to AP600-related issues.

- In preparation for test M-7-1, NUPEC also provided data from test M-4-3 for the development of proper input data for the facility geometry and for boundary conditions. Westinghouse performed a WGOTHIC prediction for this experiment, the results of which have been presented in a paper by Ofstun, R.P.; Woodcock, J.; Paulsen, D.L., Westinghouse-GOTHIC Modeling of NUPEC's Hydrogen Mixing and Distribution Test M-4-3, 3rd Intl. Conf. On Containment Design and Operation, Toronto, Canada, October 19-21, 1994, Vol. 1., and summarized in Section 9.C of WCAP-14407.

- Westinghouse also participated in the analyses of HDR test E11.2 (Narula, J.S.; Woodcock, J., Westinghouse-GOTHIC Distributed-Parameter Modeling of HDR Test E11.2, 3rd Intl. Conf. on Containment Design and Operation, Toronto, Canada, October 19-21, 1994, Vol. 1).

Non-LST Related WGOTHIC Validation Efforts

The GOTHIC code has been, and is still being, applied to a large spectrum of test data from differently scaled containment test facilities. An equally extensive effort has been, and is still being, put into direct code-to-code result comparisons, primarily by the members of the GOTHIC's User Group, consisting of 25 utilities, vendors, architect-engineers, and research organizations. This group includes a number of well-known international members. EPRI sponsors periodic meetings which have produced a wealth of GOTHIC comparisons to data and other computer codes. EPRI financed publication of the NAI GOTHIC validation report. The validation report documents comparisons between solutions of analytical problems and comparisons with data from blowdown experiments in the Carolina Virginia Test Reactor (CVTR), Battelle Model Containment (BMC), HDR, and HELD-LACE facilities. The frozen version of GOTHIC 3.4 was applied to all of the specified problems and experiments. Therefore, the validation report documents the status of GOTHIC at a specific period of its continually evolving development.

Westinghouse re-ran the validation cases from the NAI GOTHIC validation report with WGOTHIC and documented the results in the form of comparison plots showing data, GOTHIC, and WGOTHIC computational results in the same format as used in the original EPRI report. The WGOTHIC analyses were performed by Westinghouse with the same GOTHIC nodalization and initial and boundary conditions used for generating the original GOTHIC computation for the EPRI validation report. As part of their software QA program, Westinghouse re-ran a selection of these benchmarks to document that the various WGOTHIC modifications implemented by Westinghouse into GOTHIC did not impact the predictive capability of the original GOTHIC code.

21.6.5.7.8 Evaluation of AP600 Containment and the Lumped-Parameter Models with International Database

21.6.5.7.8.1 Introduction

The AP600 containment response during and after LOCA and MSLB events — without an active internal spray system — is expected to be very similar to that of a German large, dry full-pressure containment of the KWU design, at least until the PCS external liquid-film-induced cooling front reaches the inside steel shell surface. Both the AP600 and German designs have a primary steel shell containment surrounded by an air annulus and have no active internal spray systems. The AP600 has a power of 600 MW and the German BIBLIS A plant is a 1300 MW nuclear power plant (NPP).

The BIBLIS A containment pressure response after a DECLG LOCA has been published and used in the German Risk Study (German Risk Study Nuclear Power Stations, An Examination of the Risk Caused by Accidents in Nuclear Power Stations, Main Volume, (in German), Verlag Tüv Rheinland 1979). The peak pressure of 4.9 bar (71 psia) was obtained at the end of the primary system blowdown (~30 seconds) and accounted for all the appropriate conservatism in the input data and assumptions as set forth in the German risk study guidelines. As a note of interest, the guidelines require adding an additional 10 percent on top of the conservatively calculated containment pressure history. There is no second pronounced pressure peak during or after reflood; however, a change of depressurization rate was noticeable. The design limit of the KWU full-pressure containment is 5.7 bar (82.6 psia) and the failure pressure around 12-13 bar (174-188.5 psia).

The response of the large, dry containment is strongly dependent on the heat transfer to internal heat sinks during all phases of a DBA. During the long-term accident phase the subatmospheric air gap in the German-design large dry containment may also serve as a heat sink. Because of the tremendous importance of the heat transfer to heat sinks, German experimental and analytical efforts in the area of containment research focused on this issue and provide an additional database.

To validate the containment computer codes and the conservatism of the licensing approach and to generate a reliable data base, German containment research embarked on a substantial research program utilizing the Battelle Model Containment (BMC) and the HDR containment.

The staff considered this independent database and computer code analyses of BMC and HDR experiments, to support the assessment of the applicability of the WGOTHIC EM (WCAP-14407) to the AP600 containment.

The BMC facility is scaled in accordance with the power-to-volume approach. The major features and characteristics of the BIBLIS A NPP were linearly scaled at 1:4 using a cylindrical rather than spherical shape. The BMC had a concrete containment wall (with removable top cover) rather than a steel shell. All nine subcompartments were empty, and connected with uniquely defined orifices, nozzles etc., rather than with oddly shaped doors, channel, vents, as in the plant. This assumption provided a better basis for code comparisons.

The HDR constituted a real containment of a prototypical superheated steam reactor with large dimensions (volume ~11,000 m^3 (3.88E5 ft^3), dome volume:~5000 m^3 (1.76E5 ft^3), diameter 20 m (66 ft), height: 60 m (197 ft)) and about 72 subcompartments. By comparison, the AP600 containment volume is on the order of 50,000 m^3 (1.76E6 ft^3). Most compartments in the HDR were still filled with the original components during the test program. The HDR containment was not a scaled test. It was designed and licensed in accordance with the same rules and German risk study guidelines as the large BIBLIS A NPP (and all the other German reactors).

The HDR had an internal spray system installed, unlike a typical German NPP, mainly because the primary system was supposed to generate superheated steam. This originally installed internal spray system was activated during some of the later HDR containment experiments. The BMC and HDR data bases supplement each other and the combination of both should span most of the required spectrum regarding prototypicality. Therefore, computer codes whose predictions consistently match data obtained from different experiments in both facilities could be considered qualified to extrapolate from HDR size to large prototypical containments with appropriate consideration of all applicable guidelines.

21.6.5.7.8.2 Linkage Between Current PWRs and AP600 Containment Behavior

Safety assurance and licensing of current PWR containments for DBA events rely on the effectiveness of active systems in the containment to mitigate pressurization and heatup of the containment atmosphere. Internal spray systems, fan coolers, and related heat-exchanging equipment typically constitute these active systems. Knowledge about their effectiveness, availability, data, and code modeling span the current licensing space.

The AP600 containment design does not apply any of these active systems. Rather, the AP600 mitigative features are provided by passive systems which include the large containment and dome volume and surface and special provisions for heat sink efficiency, in combination with the PCS which cools the outside steel shell. In evaluating this total passive safety system for its capability to mitigate pressure and heatup, the knowledge base for current operating reactors does not suffice.

To overcome this gap in the knowledge base, Westinghouse has devised a careful roadmap which include the following elements:

- a PIRT (WCAP-14812)
- a scaling report (WCAP-14845)
- validation of the heat and mass transfer correlation package (WCAP-14326)
- separate effects (WCAP-12665)
- the SST (WCAP-14134)
- the LST integral tests (WCAP-14135)
- the lumped-parameter WGOTHIC modeling development and validation (WCAP-14382)
- the bounding AP600 evaluation model development, the analytical model development for AP600 and PCS specific features and the validation of the WGOTHIC AP600 EM through sensitivity studies presented in WCAP-14407

Experimental evidence which could establish a linkage between current and future containment designs and operations during and after DBA events would assist the evaluation of

Westinghouse's DBA methodology. The staff believes information from the HDR test program provides this link. The linkage between current design and licensing practice and the new features of the AP600 containment design can be established on the basis of data from the large-scale HDR test facility, especially test series E11 (Cron, D. Schrammel, "Investigations on Hydrogen Distribution in a Reactor Containment," Quick Look Report, Test Group E11, Experiments E11.0-6, (In German), PHDR Technical Report PHDR 111-92, 1993 and Holzbauer, H., "Parametric Open Post-Test Predictions and Analysis of the HDR-Hydrogen Distribution Experiments E11.2 and E1.4 with the Computer Code GOTHIC," (In German), Battelle Institute e.v., Final Report BleV/R67706-1, August 1992).

Evidence for this linkage and experimental proof for the equivalence of mitigative features between internal spray operation and external spray operation (similar to the AP600 PCS) is found in HDR test series E11. The staff has the following observations based on its review of two HDR tests (representative of small breaks high in the containment); test E11.2 with external sprays and test E11.1 with internal sprays:

(1) In the upper curved region of the dome, which was cooled at the outside steel shell surface during E11.2 starting at 975 minutes, no axial or transverse temperature differences existed during the cooling period.

(2) The upper curved region of the dome was completely thermally homogeneous during all other test phases when the external cooling was not applied.

(3) Additional steam injections at low positions did not lead to thermal gradients in the upper region of the dome.

(4) Over the total height of the dome from the operating deck to the dome apex, axial temperature gradients existed which change sign and size depending upon the events examined (steam release, external cooling, etc.).

(5) The same observations hold for horizontal temperature gradients and temperature differences at the operating deck level between the break (steam release) position and its opposite side.

(6) The size and sign of the temperature differences and thermal gradients in the HDR are considered maximum because only the upper part of the dome is cooled while the whole cylindrical part stays dry. This is considered an extreme condition and thus can serve as a possible upper bound.

(7) The operation of the internal spray leads to a completely homogeneous dome region. On stoppage of the spray, internally stored energy leads to a temperature increase at the upper dome.

(8) External spray operation leads to similar temperature reductions in the dome as internal spray operation.

(9) While the internal spray operation barely affects conditions near the level of the operating deck (for the mass flow rate achieved through the spray nozzles but equivalent to that of

the external spray) and did not influence below-operating-deck compartments, the cold, air-enriched layer induced by the external cooling of the upper dome region advanced to the operating deck and even propagated equally into both vertical shafts and the associated compartments below the operating deck.

All of these observations are of major relevance to the AP600 and to the WGOTHIC EM and its underlying assumptions. Specifically, observations

1, 2, and 3 confirm Westinghouse's assertion that the containment atmosphere is well mixed and its temperature is homogeneous in the dome region, which is externally cooled. This upper region is also homogeneous during periods of no external surface film cooling (i.e., dry surface, even during periods of additional steam injection).

4, 5, and 6 confirm Westinghouse's assertion that vertical and horizontal temperature differences over the total height and across the total containment diameter are about 10 °C (20 °F). The HDR data indicates that this would be the upper limit in the implausible situation that the PCS film on the whole cylindrical part of the AP600 containment shell would dry out (e.g., the conditions for which the HDR tests have been performed). The HDR data can be considered as upper limit, because its riser annulus is closed, while the AP600 riser connects to the environment through the chimney.

7 and 8 confirm the close equivalence between the temperature decreases induced by the operation of an internal spray or an external spray. This establishes a link between current and future containment designs. For the HDR, it can be speculated that the experimentally demonstrated equivalence in the containment dome cooling would have been closer, had the external cooling been initiated with 8 kg/s (17.6 lb/s), the value for the internal spray.

9 confirms Westinghouse's assertion that the PCS cooling affects below deck compartments. The delay time between external cooling initiation and subcompartment cooldown below the operating deck is long for the HDR, because the external cylindrical part of the dome is not cooled; yet the effect is very obvious approximately 130 minutes after spray initiation. Had the cylindrical part been cooled, as in the case of AP600, the impact on the below-deck subcompartments would be expected to be much stronger sooner.

In summary, the HDR data (tests E11.2 and E11.1) constitute a consistent framework for the confirmation of many of Westinghouse's assertions and positions on the basis of large-scale, fully transient experiments. An optimized GOTHIC lumped-parameter model yields conservative predictions of containment pressure and temperature for HDR test E11.2.

21.6.5.7.8.3 Application of GOTHIC and Other Codes to HDR and BMC Data

The staff has reviewed a large database consisting of experiments at BMC and HDR, as well as various computer code predications and analyses of these tests (See Section 21.6.5.9 of this report for a listing of the literature considered).

The experiments from the test series performed at the Battelle Model Containment (BMC) and the HDR facility were chosen based for the following reasons :

- the experiments were used for the GOTHIC (NAI) validation report
- the experiments were used as International Standard Problems by OECD/CSNI for the purpose of evaluating computational results from different codes using different lumped-parameter models and different correlations for heat and mass transfer
- There were experiments showing specific features important to the AP600 issues:

T31.1-3	Three large-break blowdowns with identical initial conditions but different break flow directions (orientation)
E11.1	Operation of internal spray with sump heatup and boiling; a high-elevation release position
E11.2	Similar to E11.1 but operation of external spray; stratified containment condition
E11.4	Low-elevation release position and operation of external spray; homogeneous containment condition
E11.5	Large-break LOCA at low elevation with subsequent steam releases with sump heatup and boiling; homogeneous containment condition
T31.5	Large-break LOCA at high elevation with subsequent steam and gas mixture releases
V44	Large-break LOCA into different subcompartment
V21.1	Large-break LOCA
T31.1-3 T31.5 V44	Comparison of different blowdown experiments provided upper and lower bounds of containment pressure histories in the HDR

- There were experiments with special instrumentation to measure such information as droplet velocities, droplet sizes, and local subcompartment pressures.
- There were experiments in which heat transfer coefficients were deduced from the heat transfer blocks over longer time periods (not all experiments and all heat transfer blocks have been evaluated).

Independent data from the BMC and HDR LOCA experiments for containment pressure, temperature, heat transfer, and droplet velocities and sizes have been reviewed. This data confirms Westinghouse's assertion that the blowdown peak pressure is nearly independent of break flow direction and that heat transfer to internal structures is still best described by the

Uchida correlation. In addition, comparisons between measurements and predictions based on Uchida heat transfer coefficients reveal that Uchida is very conservative during the blowdown phase and for a time period afterwards. The measured data and Uchida coefficients match the data during the long-term phase at times later than 1 hour. With respect to droplet velocities and sizes, measurements from the break subcompartment in the HDR show extremely high velocities and that at 30 seconds after blowdown initiation only drops with radii close to 5 μm were detectable.

International code benchmark exercises based on BMC and HDR experiments, demonstrate that the application of the lumped-parameter approach leads to conservative calculated blowdown peak pressures, even when these are setup to be "best estimate" predictions. Moreover, these comparisons show that when licensing and regulatory assumptions are added to the lumped-parameter model, then predicted blowdown peak pressures become highly conservative as compared to the data. This independently confirms Westinghouse's assertion that use of the lumped-parameter approach in its WGOTHIC EM results in conservative calculated blowdown peak pressure.

From comparisons of GOTHIC predictions to HDR test data, the following conclusions can be drawn:

- The steam release is more noticeable in the temperature measurements at, and above, the break elevation.
- Despite the LOCA, an axial temperature gradient exists over the height of the containment.
- GOTHIC with a lumped-parameter model simulates the axial temperature gradient and the transient temperatures over the whole time period of 1 hour extremely well.
- GOTHIC predicts the steam release effect on temperatures at different axial elevations extremely well.
- Temperature predictions are too low compared with the data over the first 5 minutes after the blowdown initiation.

Overall, the comparisons confirm GOTHIC's applicability for high-positioned large-break LOCAs. GOTHIC predicts a conservative pressure history, and matches measured temperatures at different axial elevations (stratification) very well. GOTHIC tends to predict lower (conservative) temperatures during the early phase (~5 minutes) of the overall transient.

Comparisons of HDR test E11.5 and HDR test T31.5 (Wolf, L., Mun, K., "Overview of Experimental Results for Long-Term, Large-Scale Natural Circulations in LWR-Containments After Large LOCAs," Vol. II, Assessment of HDR Experiments V21.1, V43, T31.5 and E11.5, DOE-Project, HDR Hydrogen Mixing Evaluation for Containment Safety Evaluations, Natural Global Circulation, Dept. of Materials and Nuclear Engineering, University of Maryland, College Park, MD, April 1996) provide sufficient evidence that the GOTHIC code with a lumped-parameter model has been demonstrated to reliably (and conservatively) predict containment pressure response, temperature, and even velocities with "best-estimate" input data and commonly used input parameters. Test E11.5 represented a large-break LOCA with an H_2

release in the lowest section of the HDR containment. Test T31.5 also represented a large LOCA with an H_2 release, but the release point was high in the containment.

Tests E11.5 and T31.5 cover both ends of a spectrum of break locations tested at the HDR. Test T31.5, with the associated GOTHIC prediction, is deemed to be closer to the AP600 condition.

The following are some additional observations on HDR test E11.4 (similar to E11.2, but the break position low in the containment):

- GOTHIC predicted the containment atmosphere response very well in all details for the various experimental phases (sump boiling period : 43 hours 30 minutes to 46 hours 30 minutes; external cooling period: 46 hours 10 minutes to 50 hours 12 minutes).
- Conditions in the containment during test E11.4 were in the realm of lumped-parameter models exercised in the context of current containment analysis codes.
- The GOTHIC predictions showed coherent and consistent results in all of the computed quantities as compared with the data.

21.6.5.7.9 Evaluation Conclusions

The staff has reviewed the following information:

- Steady-state separate-effects tests
- Validation of heat and mass transfer correlations
- Quasi-steady-state SST and LST tests
- WGOTHIC predictions for LST tests
- WGOTHIC application to NUPEC tests
- Development of WGOTHIC EM on the basis of selected LST tests and WGOTHIC distributed-parameter model
- Assessment of specific model assumptions, input data, and parameters
- Independent transient data from large-scale facilities covering LOCAs initiated at different axial elevations with subsequent steam releases
- Independent transient data establishing a linkage between the effects of internal spray operation and external spray operation
- Independent assessment of different versions of the GOTHIC code
- Assessment of the status of containment analysis code with lumped-parameter models

From this combination of experimental, analytical, and computational evidence, the following conclusions can be drawn:

- The major Westinghouse assumptions concerning the anticipated AP600 behavior and their treatment in the WGOTHIC EM have been confirmed through the staff's independent review of the supplemental BMC and HDR database and associated computations.

- All issues related to the blowdown time phase can be considered resolved and closed, both with respect to the anticipated AP600 containment behavior and to the WGOTHIC EM predictive capability. Specifically, it can be demonstrated that even a best-estimate lumped-parameter model leads to a conservative calculated peak pressure. That means that predictions based on conservative input values and parameters will certainly result in conservative predictions.

- All issues related to long-term cooldown are considered to be resolved because many independent experiments in different facilities and associated computations with GOTHIC confirm the validity of the model assumptions and the model input. In the majority of cases, best-estimate lumped-parameter long-term analyses with GOTHIC demonstrate slightly conservative predictions for the pressure history.

- Only when the computational model is adjusted to best fit the blowdown phase (including the peak pressure), is there a potential for systematic long-term pressure underprediction.

21.6.5.8 Conclusions

21.6.5.8.1 Introduction

The staff has reviewed the Westinghouse-GOTHIC (WGOTHIC) computer program and the AP600 evaluation model (EM) with regard to their ability to conservatively predict the pressure response of the AP600 primary containment to a DBA large-break LOCA or main steamline break (MSLB). WGOTHIC solves the conservation equations for mass, energy, and momentum for multicomponent, two-phase flow. The momentum conservation equations are written separately for each phase in the flow field (drops, liquid pools, and atmosphere vapor). The Westinghouse AP600 EM represents the AP600 containment as a network of WGOTHIC lumped-parameter nodes. A special clime component is used to model the passive containment cooling system (PCS), which provides the safety-related cooling of the AP600 containment by evaporating a water film applied to the exterior of the containment steel shell.

WGOTHIC, as documented in WCAP-14407, is based on a modified version of the GOTHIC 4.0 computer program. The GOTHIC containment analysis package was developed by Numerical Applications, Inc. (NAI), with financial support from the Electric Power Research Institute (EPRI) and a national user's group. The Westinghouse modifications to GOTHIC added the special models and features needed for AP600 licensing analysis in support of design certification. Westinghouse added analytical models to represent the unique features of the AP600 containment which included modeling the condensation heat transfer in the presence of non-condensable gases on the interior wall of the containment, one dimensional (1-D) heat conduction through the containment wall, and heat rejection on the exterior of the containment

shell via evaporative cooling, natural convection cooling, and radiative cooling. A full description of WGOTHIC 4.2, the current licensing version, and the staff's technical review are given in Section 21.6.5.4 of this report.

21.6.5.8.2 Compliance with Regulatory Requirements

The Westinghouse AP600 containment evaluation model is based on assumptions that maximize the initial stored energy within containment and minimize the rate of heat transfer from containment. The approach taken for the AP600 containment analysis has evolved from the approach used for the WGOTHIC 1.0 and WGOTHIC 1.2 analyses. To address staff concerns with some of the assumptions and modeling features employed, Westinghouse has developed a model and uses assumptions and boundary conditions that are more consistent with current practices for containment analyses for current operating reactors. The approach is consistent with the guidance provided in SRP 6.1.1.2.A, "PWR Dry Containments, Including Subatmospheric Containments." The following evaluation is based on the staff review of the WGOTHIC computer program, the AP600 EM model, and sensitivity studies (WCAP-14407), the associated PIRT report (WCAP-14812), scaling report (WCAP-14845), and validation studies (WCAP-14382 and WCAP-14967). The complete reviews can be found in Sections 21.6.5.4 through 21.6.5.7 of this report.

Compliance With 10 CFR Part 50, Appendix A

The current guidance for demonstrating that a containment design complies with GDCs 16, 38, and 50 is delineated in Chapter 6.2 of the Standard Review Plan (SRP). The SRP addresses acceptance criteria and some specific model assumptions for design basis LOCA and MSLB analyses for all existing containment types. Westinghouse elected to evaluate the PCS performance using these current guidelines. The Westinghouse documentation for the AP600 evaluation model (EM) is consistent with the guidelines in SRP Sections 6.2.1 and 6.2.1.1.A and Regulatory Guide 1.70. Westinghouse also uses approved methods for the LOCA and MSLB mass and energy releases following the guidance provided in SRP Sections 6.2.1.3 and 6.2.1.4, respectively.

Peak Pressure Criteria (GDCs 16 and 50)

Acceptance criteria for existing containments include a margin between the design pressure and a conservatively calculated peak accident pressure. The margin varies from 10 percent at the construction permit (CP) stage to a peak calculated pressure "less than the containment design pressure" at the operating license (OL) stage. Thus, even where much data and information are known, and the staff possessed an independent, confirmatory calculational capability, a 10 percent margin was expected at the CP stage to cover uncertainties in meeting GDCs 16 and 50 after final construction, at the OL stage.

For the AP600 containment, Westinghouse proposed the criterion that the calculated peak accident pressure not exceed the design pressure (a zero-margin criterion). In meeting this criterion, Westinghouse states that it uses a conservative approach, consistent with current staff guidelines. For design certification, under 10 CFR Part 52, the staff does not necessarily need the same demonstration of margin as is normally expected at the CP stage. An appropriate initial test program combined with appropriate inspections, tests, analyses, and acceptance

criteria (ITAAC), is in place to assure that the assumptions and performance characteristics of the AP600 containment and the PCS, as used in the licensing analyses, are verified before operation.

On the basis of this evaluation, the staff has determined that the WGOTHIC computer program, combined with the conservatively biased AP600 evaluation model, is acceptable for the evaluation of the peak containment pressure following a design basic accident. Although the WGOTHIC code itself is essentially a best-estimate tool, Westinghouse has taken a conservative approach in the evaluation methodology (EM) it is using to support design certification. The AP600 WGOTHIC EM uses conservative values which bound the range of most inputs, and applies conservative multipliers on the correlations used for PCS heat and mass transfer. Conservative models are used in the AP600 WGOTHIC EM to address the following areas:

- lumped-parameter network representation
- non-condensable circulation and stratification
- PCS flow and heat transfer models
- dead-ended and liquid-filled compartments

During the peak pressure period (up to 1200 seconds for LOCA, and up to 600 seconds for MSLB), these conservatisms compensate for the uncertainties introduced by the use of passive safety features, leading to an overall conservative result for the calculated peak containment pressure.

Long-Term Pressure Analysis (GDC 38)

The objective of the long-term pressure analysis is to demonstrate that the containment design conforms to the objectives of 10 CFR Part 50 Appendix A, Criterion 38, "Containment Heat Removal:"

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Item II.b of Section 6.2.1.1.A of the SRP, the staff guidance used to evaluate compliance with GDC 38, states that the containment pressure should be reduced to less than 50 percent of its peak value within 24 hours of a design basis LOCA event. This assures that the containment leak rate used for the siting evaluation is consistent with the design basis analysis assumption. In current operating reactors, credit for this 50 percent reduction in pressure is considered in the siting evaluation. Westinghouse does not credit any leakage reduction due to decreased

pressure. The siting evaluation is performed with a constant, design basis leak rate. Westinghouse had originally proposed that the pressure reduction be based on 50 percent of the design pressure for consistency with current guidelines related to GDC 38. The staff found this approach acceptable, although the peak calculated pressures are near the design value, because there was no need to demonstrate a pressure reduction for the leak rate assumption used in the siting evaluation.

Late in the review process, Westinghouse determined that it could not meet the proposed long-term objective with the original analysis approach. Westinghouse therefore revised the analytical procedure to credit the effect of two-dimensional (2-D) heat conduction (between wet and dry regions of the containment shell) when less that full coverage of the containment shell is expected. The revised procedure was first presented in Westinghouse letter DCP/NRC 0885 dated May 23, 1997, and discussed at an ACRS meeting in December 1997. In selecting the analysis methodology (GOTHIC) and in developing a model for the PCS (WGOTHIC), Westinghouse did not identify, or at least account for, the need to consider two-dimensional (2-D) heat transfer for the long-term containment pressure response (after 3 hours, at which time the PCS flow rate is first cut back to about one-half its initial value). With the coverage area less than the 90 percent initially assumed, heat will be transferred from the hot, dry regions of the shell into the cooler, wet regions of the shell. To account for this deficiency, Westinghouse has performed an ancillary calculation to credit more PCS water in the evaporation process, generating a correction factor which it has applied to the limited PCS flow model (see Section 21.6.5.4.2 of this report).

During the first 3 hours of a DBA event, with the PCS flow rate maintained at 1,665.6 l/min (440 gpm), the pressure performance envelope is similar to that of existing designs with active safety systems. When the PCS flow rate is reduced after 3 hours, the containment tends to slightly repressurize, maintaining a pressure somewhere between 218.5 and 273.7 kPa (17 to 25 psig), well below the 411.6 kPa (45 psig) design value, until, 30 hours into the event, when a further reduction in the PCS flow rate occurs. When the flow is reduced after 30 hours, the containment repressurizes further, the resulting pressure being between 225.4 and 322.0 kPa (18 to 32 psig) for the remainder of the 3-day design basis performance period of the PCS but continuing to decrease as the decay heat decreases. The difference between the low- and high-pressure estimates is based on the credit given in the analyses to the effect of 2-D heat conduction. As discussed in Section 21.6.5.4.2, the staff believes that there is a real effect from 2-D heat conduction. However, as an insufficient amount of test data is available to validate this model, the staff is unable to determine how much credit should be given in evaluating the Westinghouse design performance after 24 hours. In addition, the calculated pressure is not used to demonstrate compliance with other regulatory requirements. Whether or not credit is taken for 2-D heat conduction, the staff finds the design to be in compliance with GDC 38 and the containment pressure and temperature following the limiting loss-of-coolant accident are maintained at acceptably low levels. Although the containment pressure response is different from current licensed plants, the PCS is acceptable and consistent with the passive design objectives on which the AP600 PCS is based.

After the peak pressure period, the uncertainty in the treatment of the heat transfer processes continues to increase. These uncertainties, resulting from the EM treatment of non-condensable circulation and stratification and the effectiveness of the PCS cooling at a reduced flow rate, are difficult to quantify using the available test data. Nevertheless, the heat removal capability of the

AP600 PCS (as calculated by the WGOTHIC EM) is sufficiently greater than the decay power to conclude that the containment pressure will decrease. The staff therefore considers the design to be in general compliance with the intent of GDC 38 with respect to long-term pressure response.

Compliance with 10 CFR 52.47(b)(2)

The unique characteristics of passive containment cooling system are explicitly recognized in the regulations governing the evaluation of standard plant designs. 10 CFR 52.47(b)(2)(i)(A) states that, in the absence of a prototype plant that has been tested over an appropriate range of normal, transient, and accident conditions, the following requirements must be met for a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions":

(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;

(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;

(3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

Consistent with these requirements, the passive plant vendor, Westinghouse, has developed and performed design certification tests of sufficient scope, including both separate-effects and integral-systems experiments, to provide data with which to assess the computer programs used to analyze plant behavior over the range of conditions described in item (3) above.

To satisfy the requirements of 10 CFR 52.47(b)(2)(i)(A), Westinghouse has developed test programs to investigate the passive containment safety systems. These programs include both component and phenomenological (separate-effects) tests and integral-systems tests. The cold water distribution test (WDT) (WCAPs -13353, -13296, -13960) was a full-scale representation of the PCS flow characteristics. To comply with the requirement (3) above, additional separate-effects tests have been performed to extend the range of existing mass and heat transfer correlations used in the AP600 analysis codes (WCAP-14326).

The large-scale test (LST) (WCAP-14135) is the only integral test for the AP600 PCS. While this test exhibited a number of shortcomings in scaling and prototypicality, the LST data was not used in an integral mode. Instead, the LST data was used in a separate-effects mode to demonstrate the conservatism of portions of the evaluation model. The staff concludes that sufficient data has been provided to establish that the evaluation model is conservative at the scale of the AP600.

The staff concludes that the evaluation model contains sufficient conservatisms, including factors to compensate for shortcomings in the LST, to accept WGOTHIC in combination with the AP600 EM for DBA licensing analyses to support design certification. Specific limitations and restrictions for future analyses are given in Section 21.6.5.8.3.

21.6.5.8.3 Limitations and Restrictions

WGOTHIC, in combination with the AP600 EM, is approved for evaluating the peak containment pressure in the AP600 resulting from DBA events. It has not been qualified to predict other parameters of design interest, such as flooding levels, temperature profiles, and concentrations of non-condensable gases (e.g., air, hydrogen).

Because of the great flexibility a WGOTHIC user has in input selection, reviewers of future WGOTHIC AP600 EM analyses to support licensing actions must verify the following EM model conservatisms:

- The mass and heat transfer coefficients on the inner containment vessel surface are multiplied by a factor of 0.73. Only free convection is considered on the inner surface. The multiplier is based on an assessment of the LST and separate-effects tests, as discussed in Section 21.6.5.6.5.3 of this evaluation.

- The mass and heat transfer coefficients on the outer containment vessel surface are multiplied by a factor of 0.84. Mixed convection is considered on the outer surface. The multiplier is based on an assessment of the LST and separate-effects tests, as discussed in Section 21.6.5.6.5.3 of this evaluation.

- The vessel wall emissivity values are reduced by 10 percent to reduce the radiation heat transfer.

- The maximum PCCWST temperature allowed by the technical specifications is used as an initial condition.

- The maximum containment air temperature and maximum internal pressure allowed by the technical specifications are used as initial conditions. A zero-percent humidity initial condition is used to increase the initial stored energy inside containment.

- A single failure of one out of two valves controlling the PCS cooling water flow is assumed. This assumption provided the minimum PCS liquid film-flow rate.

- The water coverage is based on the "Evaporation Limited" flow model, as described in Section 21.6.5.4.2, and is based on the wetted surface areas provided in Table 6.2.1.1-1b of the SSAR.

- The minimum PCCWST inventory allowed in the technical specifications is used to calculate the PCS flow rate for use in the "Evaporation Limited" flow model.

- The PCS liquid-film flow is credited only following a delay period (337 seconds) necessary to establish water coverage of the shell-wetted region. This corresponds to the time needed to establish a steady liquid-film coverage pattern, on the basis on the initial flow rate.

- A 20-mil or larger air gap is assumed between the steel liner and the concrete on applicable internal heat sinks.

- The loss coefficient in the external annulus includes a 30 percent increase over the value derived from the test program.

- Condensation and convection on heat sinks in the dead-ended compartments, below the operating deck, are not credited after the blowdown period. This conservative assumption is also be employed for MSLB analyses.

- Heat transfer to horizontal, upwards facing surfaces which may become covered with a condensation film is not credited. In particular, heat transfer to the operating deck itself, which becomes covered with an air-rich layer, is not to be credited.

- Consistent with the heat sinks (structures) as identified in WCAP-14407, Revision 3, Section 4, Appendix 4B, all miscellaneous heat sinks identified by "Heat transfer coefficient = Insulated/Insulated" have been removed for licensing analyses through proper user input.

Any future AP600 licensing analyses must verify that the LOCA or MSLB characteristics are consistent with the use of the lumped-parameter representation of the AP600 with WGOTHIC, and for each calculation with significant energy transfer to the PCS through the shell, the stability of the clime heat and mass transfer solution must be examined (for example, by plotting heat transfer rates versus time for both the wet and dry climes) to confirm that the calculation has not violated the time step stability, as discussed in Section 21.6.5.4.2.3.1.2 of this report.

In the Evaporation Limited flow model, Westinghouse neglects PCS runoff sensible heat, which is conservative, and offsets the non-conservatism introduced by the simultaneous use of the Chun and Seban and the Evaporated Limited flow model. Therefore, these two assumptions must be employed together, for the staff to consider this model to be acceptable for licensing analyses, as discussed in Section 21.6.5.4.2.3.2.3 of this report.

The 2-D enhancement to the Evaporation Limited flow model, as described in Section 21.6.5.4.2.4 of this report, may not be used to credit leakage reduction for siting evaluations.

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21.7 Quality Assurance Inspections

The staff has reviewed Westinghouse's design requirements for the AP600 design, as described below.

21.7.1 QA Requirements for AP600 Design Certification Testing Activities

In Chapter 17 of the AP600 SSAR, Westinghouse describes its QA program for the design phase of the AP600 ALWR Plant Program.

In Revision 5 to SSAR Chapter 17, Westinghouse stated that effective March 31, 1996, activities affecting the quality of items and services for the AP600 Project during design, procurement, fabrication, inspection, and/or testing would be performed in accordance with the quality plan described in Westinghouse's "Energy Systems Business Unit - Quality Management System," (QMS) Revision 1. The staff's review and approval of Revision 1 to the Westinghouse QMS was documented in a letter from Suzanne Black (NRC) to N. J. Liparulo (Westinghouse), dated February 23, 1996.

Activities performed before March 31, 1996, were performed in accordance with the QA plan described in Westinghouse topical report WCAP-8370, "Energy System Business Unit - Power Generation Business Unit, Quality Assurance Plan," Revision 12a, dated April 1992. Also, activities performed before November 30, 1992, were performed in accordance with the QA plan

described in topical report WCAP-8370/7800, "Energy Systems Business Unit - Nuclear Fuel Business Unit, Quality Assurance Plan," Revision 11A/7A. Both versions of WCAP-8370 applied to all Westinghouse activities affecting the quality of items and services supplied to nuclear power plants and established Westinghouse's compliance with the provisions of Appendix B to 10 CFR Part 50.

WCAP-12600, "AP600 Quality Assurance Program Plan," currently dated January 1997, a project-specific QA plan, was developed by Westinghouse to enhance the QMS in specific areas and to establish additional commitments needed to support the AP600 Design Certification and First-Of-A-Kind (FOAKE) program. WCAP-12600 establishes the responsibility of the Nuclear Projects Division of the Energy Systems Business Unit for AP600 Design Certification and FOAKE programs and for control of the technical interface between Westinghouse and engineering groups and suppliers providing engineering services under these programs. WCAP-12600 also addresses Westinghouse's commitments to the provisions of ANSI/ASME NQA-1-1989 Edition through NQA-1b-1991 Addenda for the AP600 project.

Westinghouse also developed WCAP-12601 ("AP600 Program Operating Procedures") to establish requirements and responsibilities for developing, approving, implementing, revising, and maintaining operating procedures to meet the QA and administrative requirements of the AP600 program.

Overall, the staff conducted in-depth inspections of the five principal Westinghouse AP600 design certification test programs. The programs were inspected to determine if design and testing activities performed to support design certification of the AP600 advanced reactor were conducted under the appropriate provisions of WCAP-12600.

21.7.1.1 Core Makeup Tank Test Program

The staff conducted a QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01). During the inspection, the team assessed the Westinghouse implementation of the applicable QA criteria essential to support the AP600 design certification application, including design certification testing. Specifically, the team evaluated the effectiveness of the QA program and controls in governing the implementation of the AP600 CMT design certification testing programs.

For testing activities performed by the Westinghouse Test Engineering Group at the Waltz Mill facility, including the CMT test program, Westinghouse developed a specific QA plan or project quality plan (PQP) to implement the applicable provisions of WCAP-12601. This PQP established QA controls for the conduct of testing activities and encompassed design, construction, and configuration control criteria for the CMT test program.

The inspection team reviewed the PQP and the CMT test program specification to determine if design certification testing activities performed at the Waltz Mill facility were conducted in accordance with the appropriate provisions of Westinghouse's 10 CFR Part 50, Appendix B, QA program (WCAP-8370). Specifically, the inspection team examined the areas of performance and activities within the scope of the PQP, such as organizational responsibilities of the testing group to confirm that activities in the pertinent areas were performed under suitably controlled

conditions by properly trained personnel, and the test data collected during such activities were appropriately recorded and maintained.

The inspection team found that Westinghouse, in general, was adequately implementing the AP600 QA program plan with one exception. Westinghouse had conducted one inadequate audit at Alden Research Laboratory, Inc. (Alden Research). On October 20, 1994, the Energy System Business Unit (ESBU) Projects QA organization had established the acceptability of calibration services provided by Alden Research for AP600 design certification activities on the basis of an audit conducted on March 4, 1992. The audit, however, had not given adequate objective evidence that Alden Research was a supplier of calibration services as a "basic component" (as defined in 10 CFR Part 21), nor had it demonstrated the acceptability of Alden Research's technical and quality program capabilities with respect to the requirements in 10 CFR Part 21. This was identified as Nonconformance 99900404/95-01-03.

In Letter NTD-NRC-95-4549 (September 12, 1995), Westinghouse responded to Nonconformance 99900404/95-01-03. Westinghouse stated that Alden Research had been audited on May 25, 1995, and that although Alden Research's QA program needed improvement in several areas of its implementation, nothing had been found that would affect the calibrations performed on instrumentation for the AP600 design certification testing program. The staff found Westinghouse's reply responsive to the inspection finding and, in a letter dated December 5, 1995, told Westinghouse of its conclusion and stated that it would confirm during a future inspection the implementation of the proposed corrective and preventive actions related to the nonconformance to establish that full compliance had been achieved and maintained. This was identified as Confirmatory Item 21.7.1.1-1.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC verified that these actions had been completed and documented. On the basis of the limited scope of calibration services as a Basic Component performed by Alden Research, and regarding Westinghouse's conclusions that no audit findings identified impacted the calibration activities performed to support AP600 testing, the inspection team agreed that corrective actions taken by Westinghouse were appropriate. On these bases, Confirmatory Item 21.7.1.1-1 is closed.

21.7.1.2 Automatic Depressurization System Test Program

The staff conducted a QA implementation inspection at the Ente per le Nuove Tecnologie, L'Energia e L'Ambiente's (ENEA's) Valve and Pressurizer Operating Related Experiments (VAPORE) test facility in Cassacia, Italy, during the week of July 24, 1995 (NRC Inspection Report 99900404/95-02). ENEA implemented the pertinent provisions of WCAP-8370 at the VAPORE facility through its use of ENEA document AP600-GQ9402 ("Quality Assurance Plan Description: AP600 Test Program Conducted at the VAPORE Plant in ENEA Cassacia (Phase B)").

Under a technical cooperation agreement, Westinghouse, ENEA, and Ansaldo S.p.A. combined resources to conduct testing at the ENEA's VAPORE test facility with two major objectives (1) advance knowledge and understanding of passive safety system operations and (2) conduct testing of the AP600 ADS to provide both design information and data for computer code validation efforts needed to support AP600 design certification.

The VAPORE test specification required that testing designed to demonstrate overall ADS performance verification be conducted under a QA program that conforms to ASME standard NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," 1989 Edition through NQA-1-1991 Addenda.

During the inspection, the team reviewed the pertinent documents to determine if design certification testing activities performed at the ENEA VAPORE test facility during the ADS test program were conducted in accordance with the appropriate provisions of Westinghouse's 10 CFR Part 50, Appendix B, QA program (WCAP-8370). The team examined the performance of activities in specific areas within the scope of ENEA document AP600-GQ9402 (e.g., test control, test instrument calibration, facility and records configuration control) to confirm that activities in these areas were performed under suitably controlled conditions by properly trained personnel and that the test data collected during such activities were appropriately recorded and maintained.

On the basis of its review of these areas, the team concluded that the QA program described in AP600-GQ9402, in conjunction with Westinghouse's implementation of the pertinent criteria of WCAP-12601, gave sufficient evidence of overall QA program implementation appropriate to design certification testing, except for one finding and one unresolved item. These were identified as Nonconformance 99900404/95-02-01 and Unresolved Item 99900404/95-02-02, respectively.

Nonconformance 99900404/95-02-01

The team found that test facility as-built drawings, as required by the test specification and by ENEA document AP600-GQ9402, had not been prepared for the AP600 ADS Phase B testing at VAPORE.

In Letter NTD-NRC-95-4591 (November 9, 1995), Westinghouse responded to Nonconformance 99900404/95-02-01. Westinghouse stated that an audit of ENEA conducted in June 1995 had also uncovered an issue concerning the as-built configuration documentation of the VAPORE test facility. Ansaldo had been hired to modify the facility for AP600 testing and was responsible for preparing the appropriate documentation.

A Westinghouse review of all documentation at Ansaldo offices in Genoa, Italy in July 1995 revealed that Ansaldo had used a combination of shop drawings and field measurements to create the as-built documentation. After assessing the elements used to define the as-built configuration of the ADS test facility as well as the supporting documentation on the procurement and fabrication of the piping sections, Westinghouse concluded that the as-built documentation was in compliance with AP600 project and QA requirements. The staff reviewed Westinghouse's response and found it responsive to the concern raised in the nonconformance. In a letter dated December 5, 1995, the staff notified Westinghouse of its finding and stated that the implementation of proposed corrective and preventive actions related to the nonconformance would be reviewed during a future inspection to establish that full compliance had been achieved and maintained. This was identified as Confirmatory Item 21.7.1.2-1.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC reviewed a summary of Westinghouse's assessment activities at

Ansaldo and concluded that appropriate actions were taken to resolve this nonconformance. Accordingly, Confirmatory Item 21.7.1.2-1 is closed.

Unresolved Item 99900404/95-02-02

The team found that the ENEA QA program did not have adequate measures to effectively control the calibration status of reference instruments or standards. No provisions were in place to require their recalibration at the requisite intervals. The test specification required that the following measures be included in the detailed test procedure(s): (1) provisions for ensuring that calibration of test equipment is traceable to recognized national standards and (2) verification and documentation, to be submitted to Westinghouse, by the testing organization that the facility instruments were calibrated before testing. The ENEA QAPD document, AP600-GQ9402, provided procedures for implementing these requirements.

During the inspection, the team confirmed that all test instruments used in the ENEA VAPORE test facility had been calibrated, both before and after testing, using standards or reference instruments traceable to the Servizio Italiano di Taratura (SIT) (Italian calibration system). Also, the team reviewed the calibration records of the VAPORE test facility which provided evidence of traceability to the appropriate ENEA-controlled SIT-certified standards. This review also gave evidence of the adequacy of the calibration status of facility instrumentation during each testing phase. The team found, however, that the ENEA QA program did not contain adequate measures to effectively control the calibration status of reference instruments or standards used for instrument calibration, as no provisions were in place to require recalibration by SIT at the requisite intervals. Pending confirmation by Westinghouse that this lapse in the SIT-certified calibration interval for the ENEA reference instruments and standards did not undermine or adversely impact the VAPORE ADS test results, this issue would remain unresolved.

In Letter NTD-NRC-95-4591 (November 9, 1995) Westinghouse responded to Unresolved Item 99900404/95-02-02. Westinghouse stated that ENEA had submitted seven instruments involved in AP600 test instrument calibrations to a nationally certified laboratory. Westinghouse added that as of October 1995, five of the instruments were within expected tolerances. The remaining two instruments would be tested by the end of November 1995. The staff found Westinghouse's reply responsive to the concern raised in the unresolved item. In a letter dated December 5, 1995, the staff notified Westinghouse that the implementation of proposed corrective actions related to the unresolved item would be reviewed during a future inspection to establish their acceptability. This was identified as Confirmatory Item 21.7.1.2-2.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC verified that ENEA submitted the seven instruments involved in AP600 test instrument calibration to a nationally certified calibration laboratory in Italy (ERG/ING/PITER Division). Test results confirmed that VAPORE ADS test results had not been adversely impacted. Accordingly, Confirmatory Item 21.7.1.2-2 is closed.

21.7.1.3 Passive Residual Heat Removal Heat Exchanger Test Program

When Westinghouse conducted this test program, a configuration of straight vertical tubes represented the in-plant configuration of the PRHRHX. However, Westinghouse modified the PRHRHX design to a vertical C-shaped tube bundle. The staff asked Westinghouse to justify in

detail the applicability of the straight tube PRHRHX test data to the new C-tube configuration. This was identified as Open Item 21.3.3-1 in the DSER.

While conducting a QA implementation inspection of the CMT and PCCS LST test programs at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the staff reviewed the PRHR test program design files. Although Westinghouse did not seem to have conducted formal design reviews, all meetings, discussions, and other communications relevant to the test program had been documented in memoranda at the initiative of the responsible test engineer. As a result, a record of decisions concerning facility design and operation for this test program exists.

The team questioned the apparent lack of as-built drawings or identification of critical dimensional attributes for the PRHR test program. In discussions with the test engineer who was responsible for the program and the manager of test engineering, the team learned that there were only three critical attributes for this relatively small-scale, separate-effects test, i.e., the length, diameter, and thickness of each of the three tubes used to simulate the operation of the PRHRHX. These "critical dimensions" were measured and documented in the final test report for the program. Although these critical dimensions were not formally identified as such, the team concluded that the PRHR test report would contain sufficient information for the staff to issue its finding on the technical acceptability of the program.

The staff has completed its review of the PRHRHX test program, and has concluded that HX heat transfer models using the 3-tube test data are applicable to the C-tube design in the AP600. Discussion of data applicability and the details of the staff's review of this program can be found in Sections 21.3.3 and 21.5.3, respectively.

21.7.1.4 OSU/APEX Test Program

The staff conducted an inspection at the Oregon State University (OSU) Advanced Plant Experiment (APEX) test facility in Corvallis, Oregon, during the week of August 29, 1994 (NRC Inspection Report 99900404/94-01). Specific quality provisions applicable to the OSU facility were identified in AP600 Document Number LTCT-GAH-001 ("AP600 Long Term Cooling Test Project Quality Plan") (PQP). This plan established controls for the design and construction of the test facility as well as for the conduct of testing activities.

The inspection indicated that Westinghouse, in general, was adequately implementing the AP600 project QA program at OSU, except for a few findings in certain areas. Specifically, the team identified findings with program implementation with respect to (1) the calibration of instrumentation, and (2) the accuracy of the facility as-built drawings. These issues were identified in Nonconformances 99900404/94-01-01 and 99900404/94-01-02. Also, a concern about OSU's acceptance of test results that failed to meet established test acceptance criteria, without an evaluation and disposition being included in the test design record file, was identified as part of Unresolved Item 99900404/94-01-03.

<u>Nonconformance 99900404/94-01-01</u>

The team found that contrary to the provisions of the PQP, The Industrial Company (TIC) which calibrated thermocouples at the OSU/APEX test facility, had hired Industrial Instruments, Inc. to

calibrate the primary standards and had accepted the calibration certificates without having performed audits or a surveillance of that company. Also, OSU had failed to verify the validity of the flowmeter calibration certificates received from Foxboro Company, an unaudited commercial supplier.

In Letter NTD-NRC-95-4408, dated February 15, 1995, Westinghouse responded to Nonconformance 99900404/94-01-01. Westinghouse stated that a review, performed to identify suppliers of calibration services to the OSU test facility, revealed six suppliers for which QA oversight was needed to support test instrument calibration certifications. Because four of these suppliers were not on the Westinghouse qualified suppliers list (QSL), they were audited. Westinghouse added that these audits revealed no conditions that would invalidate the calibrations performed on any OSU test facility instruments. After the audit, Westinghouse sent all devices used to perform the post-calibration checks to a qualified supplier of calibration services for calibration. To prevent future problems, Westinghouse would revise the OSU PQP to describe the requirements for procurement of calibration services and calibration of commercially procured test instrumentation and calibration equipment.

During a subsequent QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the team reviewed the Westinghouse audit reports of the four suppliers of calibration services not on Westinghouse's QSL to confirm that the supplier's QA programs were suitable to ensure reliable instrument calibration. In general, the reports demonstrated that calibration services supplied by these companies were adequate. Some questions were raised with regard to the audit of Morris Scale, but it was determined that the instruments in question (load cells) were not used for any critical measurements or to determine key facility parameters, such as energy and mass balances. Calibration results were included as part of the final data report for the APEX test program. Westinghouse submitted this report on June 13, 1995.

The team confirmed that Westinghouse had revised the OSU APEX PQP to describe requirements for procurement of calibration services and calibration of commercially procured test instrumentation and calibration equipment as stated by Westinghouse in its letter of February 15, 1995. The team verified that all other corrective and preventive actions identified in the subject letter had been completed and documented. On this basis, Nonconformance 99900404/94-01-01 is closed.

Nonconformance 99900404/94-01-02

The team found that contrary to the provisions of the PQP (1) no procedures or instructions had been available identifying methods, accuracy, and/or the acceptance criteria to be used for determining the as-built elevations and dimensions of the OSU/APEX test facility and (2) the TIC Calibration Procedure 19, used to calibrate thermocouples for the long-term cooling tests at the OSU facility, had not noted who originated, reviewed, or approved the document.

In Letter NTD-NRC-95-4408, dated February 15, 1995, Westinghouse responded to Nonconformance 99900404/95-01-02. Westinghouse stated that a specific set of requirements were being incorporated into the OSU PQP for documenting critical dimensions, and that the two TIC procedures cited in the nonconformance had been superseded by two OSU procedures which corrected the inadequacies identified by the team. As preventive actions, Westinghouse stated that Procedure AP-3.11 in WCAP-12601 was being revised to address documentation of

critical attributes of test facilities, and that the OSU APEX PQP was being revised to include requirements for preparing instrument calibration procedures.

During a subsequent QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the team verified that Westinghouse had incorporated a set of requirements into the OSU/APEX PQP for documenting critical attributes of the facility design. The reverification of critical dimensions had been completed, and a report was being prepared.

The team also verified that two thermocouple calibration procedures used by TIC for APEX had been replaced by two OSU procedures which were incorporated into the APEX facility maintenance plan. The team verified that the two procedures properly indicated originator and management approvals and that other corrective and preventive actions identified in the February 15, 1995, letter had been completed and documented by Westinghouse. On this basis, this part of Nonconformance 99900404/94-01-02 is closed.

Unresolved Item 99900404/94-01-03

The team found that OSU had accepted test results which had failed to meet established test acceptance criteria, without placing an evaluation and disposition in the test design record file. It appeared that at least two facility characterization tests did not meet the acceptance criteria. Nevertheless, these two tests were not rerun, and no documentation was available at OSU to indicate the disposition of the apparent deviations.

Westinghouse's QA procedures require that test acceptance documentation contain evidence that any deviations occurring during a test have been evaluated and, if necessary, dispositioned. Westinghouse stated that acceptability of a test is documented in the quick-look report (QLR) for that test. Also, the FDR for each test program would contain a full listing of all tests performed in the program, would identify invalid tests, and explain why invalid tests were disqualified. Westinghouse's Test Engineering Group performs the evaluations and prepares the QLRs and the FDRs. Westinghouse added that it, not the testing organization (OSU), makes the final determination about the acceptability of test results and that documentation of this evaluation and its disposition is placed in the official design record file at Westinghouse offices in Monroeville.

During a subsequent QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the team discussed the results of the OSU inspection with Westinghouse's Test Engineering Group. The team learned that for one of the tests related to the unresolved item at OSU, the data had been determined, after the fact, as not essential for facility characterization. For the second test, the flows that were attained in the test had not met the acceptance criteria but were determined to be useful to characterize the system and were accepted on that basis.

Discussion of the reasoning used in making these decisions was to be included in the FDR and the TAR. Although the reports had not been issued at the time of the inspection and, therefore, could not be reviewed to ascertain that the disposition of these deviations had been appropriately documented, after speaking with Westinghouse's Test Engineering Group, the team concluded that the evaluations had been performed, that the reasoning behind the

decisions on data acceptability and usefulness was sound, and that the QA program requirements were met. Subsequent review of the OSU FDR confirmed that discussion on the acceptance basis had been included. Therefore, this part of Unresolved Item 99900404/94-01-03 is closed.

21.7.1.5 SPES-2 High-Pressure, Full-Height Integral Systems Test Program

The staff conducted a QA implementation inspection at the Societa' Informazioni Esperienze Termoidrauliche (SIET) SPES-2 test facility in Piacenza, Italy, during the week of October 10, 1994 (NRC Inspection Report 99900404/94-01).

The quality-related activities associated with the AP600 full-height, full-pressure integral system tests performed at SPES-2 were conducted under the auspices of a cooperative agreement between Westinghouse, ENEA, ENEL, and SOPREN-Ansaldo. The test program provided thermal-hydraulic data for computer code validation and simulated the operation of the AP600 passive safety systems. SIET described the general features of its QA system in its quality manual, 00001-QQ. Additional quality plan provisions specific to AP600 testing were also detailed in Procedure 00006-QQ-92, "Quality Plan Relative to Nuclear Area Orders," which cross referenced European quality standards to the criteria of NQA-1. In order to conduct these confirmatory tests, SIET had implemented an internal quality system which incorporates the standards of both the International Organization for Standardization (ISO) Code Series 9000 (UNI-EN 29000) and the Italian Code UNI-8450 (1983).

During the inspection, the team found that Westinghouse had established appropriate procedural controls at the SPES-2 test facility which properly incorporated the applicable provisions of WCAP-8370 except for a few findings in certain areas. Specifically, the team identified a nonconformance with program implementation with respect to the accuracy and preparation of facility as-built drawings. These issues were identified as part of Nonconformance 99900404/94-01-02. Also, the team identified an unresolved item concerning SIET's acceptance of test results that had failed to meet established test acceptance criteria, without an evaluation and disposition being placed in the test design record file. This was identified as Unresolved Item 99900404/94-01-03.

Nonconformance 99900404/94-01-02

The team found that, contrary to the provisions of WCAP-8370 and WCAP-9565, no instructions or procedures were available at the SIET SPES-2 test facility to verify critical dimensions or configuration of commercial manufacturing drawings before accepting these drawings as representing the as-built design condition and placing them under the SIET QA system. In addition, the procedures for determining system and component elevations and arrangements at the SPES-2 test facility did not prescribe the required accuracy or include any acceptance criteria for such measurements.

In Letter NTD-NRC-95-4408 (February 15, 1995), Westinghouse responded to Nonconformance 99900404/95-01-02. Westinghouse stated that SIET had prepared a procedure for AP600 test configuration control, which would list the applicable portions of SIET Document 00006-QQ-92 related to the requirements for component inspections. Also, SIET confirmed that inspections had been performed for all vessels constructed to simulate parts of the AP600 design. SIET had compiled all inspection records and would verify that the

component acceptance inspections were performed in accordance with SIET Document 00006-QQ-92. With respect to the other issue identified in the nonconformance, Westinghouse stated that SIET had prepared an AP600 test configuration control document (00011-QQ-94), which contained specific requirements related to the identification of critical facility dimensions, including applicable tolerances. Also, SIET would revise the SPES-2 test specification to identify the critical dimensions of the facility and the required tolerances. As preventive actions, Westinghouse stated that it would revise relevant procedure(s) in WCAP-12601 to address documentation of critical attributes of test facilities.

In the course of a subsequent QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the team verified that SIET had developed a procedure for inspecting components acquired from commercial contractors. In addition, SIET had reviewed records of equipment inspections for the SPES-2 test facility, and had confirmed that vendor drawings and specifications properly reflected as-built attributes of relevant components. The team verified that the SPES-2 test specification was modified to identify facility critical dimensions and associated required accuracy, and that as-built critical attributes were confirmed by SIET. On this basis, this part of Nonconformance 99900404/94-01-02 is closed.

Unresolved Item 99900404/94-01-03

The team identified an unresolved item concerning SIET's acceptance of test results that failed to meet established test acceptance criteria, without an evaluation and disposition being placed in the test design record file. Westinghouse's QA procedures require that test acceptance documentation contain evidence that any deviations occurring during a test have been evaluated and, if necessary, dispositioned. Westinghouse stated that acceptability of a test is documented in the QLR for that test. Also, the FDR for each test program would contain a full listing of all tests performed in the program, identify invalid tests, and explain why an invalid test had been disqualified. Westinghouse's Test Engineering Group performs the evaluations and prepares the QLRs and the FDRs. Westinghouse stated during the inspections that it, not the testing organization, makes the final determination about the acceptability of test results and that documentation of this evaluation and its disposition is placed in the official design record file at Westinghouse offices in Monroeville, Pennsylvania.

During a subsequent QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01), the team discussed Westinghouse's process for the evaluation and disposition of test results that do not meet the acceptance criteria, as described above. After discussing this process with Westinghouse's Test Engineering Group, the team concluded that the review procedure fulfills Westinghouse's documented AP600 QA requirements. On this basis, this part of Unresolved Item 99900404/94-01-03 is closed.

21.7.1.6 Large-Scale Passive Containment Cooling System Test Program

The staff conducted a QA implementation inspection at Westinghouse's facilities in Monroeville, Pennsylvania during the week of May 1, 1995 (NRC Inspection Report 99900404/95-01). During the inspection, the team reviewed the Westinghouse implementation of the applicable QA criteria essential to support the AP600 design certification application, including design

certification testing. Specifically, the team evaluated the effectiveness of the QA program and controls in governing the implementation of the AP600 PCCS LST design certification testing program.

Testing activities performed at Westinghouse's Science and Technology Center (STC), including the PCCS LST test program, were conducted under direct oversight by the ESBU QA organization. A separate STC-based QA plan was not identified as a requirement for STC project tasks. The quality-related activities associated with the PCCS LST tests were controlled in accordance with the test specifications and the test procedures. No specific LST quality plan was developed; however, a one-page quality plan that was prepared for the integral extension test was applied to the LST.

The results of the inspection indicate that Westinghouse, in general, was adequately implementing the AP600 Quality Assurance Program Plan with the exception of a few findings in certain areas. Specifically, the team identified nonconformances with program implementation with respect to (1) test facility configuration control, and (2) the accuracy of the facility as-built drawings. These findings were identified as part of Nonconformances 99900404/95-01-01 and 99900404/95-01-02, respectively. Also, the team identified a concern with respect to the disposition of test deviations between the Test Engineering group and the Containment and Radiological Analysis group (SDSER Confirmatory Item 21.7.1.6-1).

Nonconformance 99900404/95-01-01

The team found (NRC Inspection Report 99900404/95-01) that the PCCS LST specification did not accurately reflect the instrumentation that was procured and installed in the PCCS Large-scale Test Facility as required by the provisions of WCAP-12601. In a letter dated September 12, 1995, Westinghouse responded to Nonconformance 99900404/95-01-01. Westinghouse stated that it would revise the PCCS LST specification to reflect the final configuration of the PCCS LST. The staff found Westinghouse's reply responsive to the concern raised in the nonconformance. In a letter dated December 5, 1995, the staff notified Westinghouse of its finding and stated that the implementation of proposed corrective and preventive actions related to the nonconformance would be reviewed during a future inspection to establish that full compliance had been achieved and maintained. This was identified as Confirmatory Item 21.7.1.6-2.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC verified that PCS-T1P-002 had been revised to reflect the final configuration of the Large-Scale Test Facility. As preventive actions, Westinghouse revised test specifications for subsequent tests (including the Core Make-Up, Long-Term Cooling, and Full-Height/Full Power tests) to reflect the final configuration of the respective facilities. The inspection team verified that these actions had been completed and documented. Therefore, Confirmatory Item 21.7.1.6-2 is closed.

Nonconformance 99900404/95-01-02

The team found (NRC Inspection Report 99900404/95-01) that procedures or instructions for determining the as-built elevations and critical dimensions of the PCCS LST had not been available to or utilized by the Containment and Radiological Analysis Group as required by WCAP-9565. In Letter NTD-NRC-95-4549 (September 12, 1995), Westinghouse responded to

Nonconformance 99900404/95-01-02, stating that it would remeasure appropriate critical dimensions in accordance with a written procedure. Measurements obtained would be compared to those previously recorded and additional remeasurements would be taken if necessary. Westinghouse added that after the PCCS LST program had been completed it had revised AP600 Procedure 3.11 to require that a written procedure be used to obtain the specified critical dimensions of any safety-related AP600 test facility. The staff reviewed Westinghouse's response and found it responsive to the concern raised in the nonconformance. In a letter dated December 5, 1995, the staff notified Westinghouse of its finding and stated that the implementation of proposed corrective and preventive actions related to the nonconformance would be reviewed during a future inspection to establish that full compliance had been achieved and maintained. This was identified as Confirmatory Item 21.7.1.6-3.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC verified that proposed corrective and preventive actions had been completed and documented by Westinghouse. Confirmatory Item 21.7.1.6-3 is closed.

Confirmatory Item 21.7.1.6-1

During the inspection (NRC Inspection Report 99900404/95-01), the team reviewed Test Procedure 219.1, Revision 0, which was used during Runs 54 and 57 to verify acceptable implementation of the AP600 quality plan. The purpose of Test 219.1 was to investigate the distribution of noncondensables as a result of changes in vessel cooling. While reviewing the test data for this test, the team noticed that the rate of water flow to the top of the vessel had not been constant, as is required by the test procedure. Another target test parameter was an inlet steam flow of 0.09 ± 0.02 kg/s (0.2 ± 0.05 lb/s). This steam flow was lowered to approximately 0.054 kg/s (0.12 lb/s) to limit the vessel pressure and baffle temperatures to an acceptable range. The team considered the failure of the test conditions to meet the target test parameters to be a deviation from the test procedure.

In accordance with AP-3.11, "AP600 Testing," these deviations were recorded in the test logbook and the Test Engineering Group was notified of the deviations. The Test Engineering Group was responsible for evaluating the reported test deviations and documenting the disposition of deviations by sending an official test engineering transmittal to the test group. Representatives of the Test Engineering Group stated that it was normal practice to document deviations in the FDR. The team verified that the deviations had been documented in the FDR for Test 219.1. The engineering procedure also required the concurrence of the cognizant design group, in this case the Containment and Radiological Analysis Group, for disposition of a deviation affecting compliance with the test specification.

Representatives of the Test Engineering Group stated that the Containment and Radiological Analysis Group acknowledged the disposition of these and other deviations by their acceptance signatures on the QLR and FDR. The staff would confirm the disposition of test deviations between the Test Engineering Group and the Containment and Radiological Analysis Group during a future inspection. This was identified as Confirmatory Item 21.7.1.6-1.

During an inspection (NRC Inspection Report 99900404/97-02) on November 17 through 21, 1997, the NRC discussed the PCCS LST acceptance criteria with appropriate Westinghouse staff. The following test acceptance criteria was established for the LST:

(1) Data on forcing functions were available (e.g., steam flow rate, fan speed, water flow rates, inlet temperature of steam, water and air). Strict adherence to the specific absolute pressures and flow rates was not necessary but values should be nearly constant as defined in the test matrix.

(2) Data on response variables available, (e.g., condensate flow rates, excess water flow rates, air, water and steam outlet temperatures, vessel pressure, 80 percent of the vessel and fluid temperature, and vessel water coverage measurements) were taken.

(3) Unplanned excursions were evaluated on a case-by-case basis. Failures that could result in faulty data outputs were not acceptable.

(4) The vessel pressure was maintained within specified pressure limits during the constant pressure portions.

Variations in the PCS water coverage flow rate were not considered in the development of the test acceptance criteria. The important criterion was the target coverage area, as specified in the test matrix. Even then, the acceptance criteria did not require strict adherence to the target value, only that a nearly constant value could be determined for a test. With respect to the specific steam flow rate for test 219.1, the test acceptance criteria were followed. On these bases, Confirmatory Item 21.7.1.6-1 is closed.

21.7.2 Summary

After considering the QA implementation inspections of Westinghouse's design certification test facilities and/or programs, the staff concludes that the QA programs governing Westinghouse's AP600 design certification test programs have satisfied the requirements of 10 CFR Part 52 and the pertinent provisions of Appendix B to 10 CFR Part 50.

Table 21.6-1
Phenomena Identification and Ranking Table for AP600
Non-LOCA and Steam Generator Tube Rupture Design Basis Analyses

Component & system phenomena		(1) FW Malf	(2) ELI	(3) SLB	(4) Inad PRHR	(5) LOL	(6) Loss AC& LONF	(7) FLB	(8) Loss RCS Flow	(9) LR & BS	(10) SUIL	(11) RWAP	(12) Inad CMT or CVS	(13) RCS Dep	(14) SGTR
Critical flow		N/A	N/A	H	N/A	N/A	N/A	H	N/A	N/A	N/A	N/A	N/A	M	H
Vessel	Mixing	H	L	H	H	L	M	M	L	L	H	L	M	L	M
	Flash in upper head	N/A	N/A	M	L	N/A	L	L	N/A	N/A	N/A	N/A	L	L	L
Core	Reactivity feedback	H	M	H	H	M	L	M	M	M	H	M	L	L	L
	Reactor trip	H	L	H	H	H	H	H	H	H	H	H	H	H	H
	Decay heat	L	L	L	H	L	H	H	L	L	L	L	H	L	H
	Forced convection	H	H	H	H	H	H	H	H	H	H	H	M	H	L
	Natural circulation flow and heat transfer	M	L	H	H	L	H	H	L	L	L	L	H	L	M
RCP	Coastdown performance	L	N/A	L	L	L	L	L	H	H	N/A	N/A	L	L	L

Table 21.6-1 (continued)
Phenomena Identification and Ranking Table for AP600
Non-LOCA and Steam Generator Tube Rupture Design Basis Analyses

Component & system phenomena	(1) FW Malf	(2) ELI	(3) SLB	(4) Inad PRHR	(5) LOL	(6) Loss AC& LONF	(7) FLB	(8) Loss RCS Flow	(9) LR & BS	(10) SUIL	(11) RWAP	(12) Inad CMT or CVS	(13) RCS Dep	(14) SGTR
Pressurizer Pressurizer fluid level	L	L	M	M	L	M	L	L	L	L	L	M	L	M
Surge line pressure drop	L	L	L	L	H	L	L	M	H	L	L	L	L	L
Steam generator Heat transfer	H	H	H	L	H	H	H	L	L	L	M	L	L	M
Secondary conditions	M	L	H	L	L	M	M	L	L	L	L	L	L	H
RCS Wall stored heat	L	L	L	L	N/A	L	L	N/A	N/A	L	N/A	L	L	M
CMT Recirculation injection	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Gravity drain injection	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Vapor condensation rate	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Balance line pressure drop	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Balance line initial temperature distribution	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Accumulators Injection flow rate	N/A	N/A	M	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
PRHR Flow rate and heat transfer	N/A	N/A	L	H	N/A	H	H	N/A	N/A	N/A	N/A	H	N/A	H

Notes:

(1) FW Malf	-	Feedwater malfunction that results in a decrease in feedwater temperature or an increase in feedwater flow
(2) ELI	-	Excessive increase in secondary steam flow
(3) SLB	-	Steamline break
(4) Inadvert PRHR	-	Inadvertent operation of the PRHR system
(5) LOL	-	Loss-of-secondary-side-load events
(6) Loss AC & LONF	-	Loss of ac power and loss of normal feedwater
(7) FLB	-	Feedline break
(8) Loss of RCS Flow	-	Loss of forced RCS flow
(9) LR & BS	-	Locked RCP rotor and broken RCP shaft
(10) SUIL	-	Startup of inactive RCP at an incorrect temperature
(11) RWAP	-	Withdrawal of a rod cluster control assembly at power
(12) Inad CMT or CVS	-	Inadvertent operation of CMT or CVS
(13) RCS Dep	-	Inadvertent RCS depressurization
(14) SGTR	-	Steam generator tube rupture
N/A	-	Not applicable
H	-	High
M	-	Medium
L	-	Low

Table 21.6-2
Westinghouse Final PIRT For AP600 SBLOCA

Component Phenomenon	Blowdown	Natural Circulation	ADS Blowdown	IRWST Injection Cooling
ADS Stages 1-3 Critical Flow	H*	H*	H	L
Two-Phase Pressure Drop	N/A*	N/A*	H	L
Valve Loss Coefficients	N/A*	N/A*	H	L
Single-Phase Pressure Drop	N/A	N/A	N/A	L
ADS 1-4 Critical Flow	N/A	N/A	H	L
Subsonic Flow	N/A	N/A	L	M
Two-Phase Pressure Drop	N/A	N/A	M	M
Break Line Resistance Critical Flow (in complex Geometries)	N/A H	N/A H	L H	L N/A
Subsonic Flow	N/A	N/A	L	L
Discharge Coefficient	M	M	L	L
Accumulators Injection Flow Rate	N/A	M	H	N/A
Noncondensible Gas Entrainment	N/A	N/A	L	L
Cold Legs Flashing	N/A	N/A	L	N/A
PBL-to-Cold Leg Tee	N/A	H	H	L
Phase Separation	N/A	H	H	L
Stored Energy Release	N/A	L	N/A	L
Vessel/Core Decay Heat	H	H	H	H
Forced Convection	M	N/A	N/A	N/A
Flashing	M	N/A	M	L
Wall Stored Energy	M	N/A	M	M
Natural Circulation Flow and Heat Transfer	M	M	M	M
Mixture Level Mass Inventory	H	H	H	H
Mass Flow	M	M	L	L
Flow Resistance	L	L	L	L

Table 21.6-2
Westinghouse Final PIRT For AP600 SBLOCA

Component Phenomenon	Blowdown	Natural Circulation	ADS Blowdown	IRWST Injection Cooling
CMT Draining Effects Condensation on cold thick steel surfaces	N/A	N/A	M	N/A
Transient conduction in CMT walls	L	L	M	N/A
Interfacial condensation on CMT water surface (break size dependent)	N/A	N/A	H	N/A
Dynamic effects of steam injection and mixing with CMT liquid and condensate	N/A	N/A	H	N/A
Thermal stratification and mixing of warmer condensate with colder CMT water	L	H	H	N/A
CMT Recirculation Natural circulation of CMT and CL balance leg	H	H	N/A	N/A
Liquid mixing of CL balance leg, condensate, and CMT liquid	H	H	N/A	N/A
Flashing effects of hot CMT liquid layer	N/A	L	M	N/A
CMT wall heat transfer	L	M	M	N/A
CMT Balance Lines Pressure Drop	M	H	H	N/A
Flow Composition	M	H	H	N/A
Downcomer/Lower Plenum Flashing	N/A	N/A	L	N/A
Level	M	M	H	H
Loop Asymmetry Effects	M	M	L	L
Stored Energy Release	L	L	L	L
Hot Legs Countercurrent Flow	N/A	L	L	N/A
Entrainment	L	L	M	M
Flashing	L	L	L	N/A

Table 21.6-2
Westinghouse Final PIRT For AP600 SBLOCA

Component Phenomenon	Blowdown	Natural Circulation	ADS Blowdown	IRWST Injection Cooling
Horizontal Fluid Stratification	N/A	M	M	M
Phase Separation in Tees (Flow Region)	N/A	M	M	M
IRWST Discharge Line Flashing	N/A	N/A	N/A	N/A
Flow and Temperature Distribution in PRHR Bundle Region	L	M	L	N/A
Pool Level	N/A	M	L	H
Gravity Draining	N/A	N/A	N/A	H
Vapor Condensation	N/A	M	M	L
DVI Line Pressure Drop (Flow Resistance)	M	M	M	M
Pressurizer CCFL	N/A	N/A	M	N/A
Entrainment/De-entrainment	N/A	N/A	M	N/A
Flashing	H	N/A	M	N/A
Level (inventory)	M	M	M	N/A
Level Swell	M	L	M	N/A
Stored Energy Release	L	L	L	N/A
Vapor Space Behavior	N/A	N/A	L	N/A
Pressurizer Surge Line Pressure Drop	L	L	M	N/A
Flooding	L	L	M	N/A
Steam Generator 2ϕ - Natural Circulation	M	M	N/A	N/A
Steam Generator Heat Transfer	M	M	L	N/A
Secondary Conditions	L	M	L	N/A
U-tube Condensation	L	L	N/A	N/A
Secondary Level	L	M	N/A	N/A
Secondary Pressure	M	M	N/A	N/A

Table 21.6-2
Westinghouse Final PIRT For AP600 SBLOCA

Component Phenomenon	Blowdown	Natural Circulation	ADS Blowdown	IRWST Injection Cooling
Steam Generator Tube Draining	L	M	N/A	N/A
PRHR				
1ϕ Heat Transfer	M	M	N/A	N/A
2ϕ Heat Transfer Condensation	N/A	M	L	N/A
Non-Condensible Gas Effects	N/A	N/A	L	N/A
Recirculation Flow	M	M	L	N/A
Upper Head/Upper Plenum				
Draining Effects	L	M	M	N/A
Flashing	L	N/A	M	N/A
Mixture Level	H	H	H	H
Entrainment/Deentrainment	N/A	L	M	M
RCP				
Coast Down	L	N/A	N/A	N/A
Flow Resistance	L	L	N/A	N/A

Table 21.6-3 NOTRUMP AP600 SBLOCA Component Separate Effects Assessment Tests	
ADS Test Simulation - Assess ADS Model and Performance	
Test 340	Stage 2 open (inadvertent), 2235 psig
Test 212	Stage 1 open, 2235 psig
Test 220	Stages 1 & 2 open, 1200 psig
Test 240	Stages 1, 2, & 3 open, 1200 psig
Test 320	Stages 1, 2, & 3 open, 2235 psig
Test 242	Stages 1, 2, & 3 open, 500 psig
Test 210	Stage 1 open, 2235 psig
Test 250	Stage 2 open (inadvertent), 1200 psig
CMT Test Simulation - Assess CMT Model and Performance	
Test 307	Demo insig. change to prelim. 300 Series runs
Test 309	Demo insig. change to prelim. 300 Series runs
Test 501	Nat. circ., to 1/5 CMT heated, 6 gpm drain rate
Test 502	Nat. circ., to 1/5 CMT heated, 16 gpm drain rate
Test 503	Nat. circ., to ½ CMT heated, 6 gpm drain rate
Test 504	Nat. circ., to ½ CMT heated, 16 gpm drain rate
Test 505	Nat. circ., to Full CMT heated, 6 gpm drain rate
Test 506	Nat. circ., to Full CMT heated, 16 gpm drain rate
Test 507	Nat. circ., to 1/5 CMT heated, 16 gpm drain rate
Test 508	Nat. circ., to ½ CMT heated, 16 gpm drain rate
Test 509	Nat. circ., to Full CMT heated, 16 gpm drain rate

Table 21.6-4
NOTRUMP AP600 SBLOCA Two-Phase Level Swell Assessment

G-2 Core Uncovery Tests			
Test	Pressure (psia)	Power (Mw)	Level (in)
715	779	0.603	114
716	775	0.252	138
719	394	0.267	138
720	395	0.615	114
724	96	0.252	126
725	96	0.599	96
728	50	0.596	84
729	50	0.250	114
732	15.1	0.254	102
733	15.8	0.600	72
GE Small Vessel Blowdown Tests			
Test	Break (in)	Pressure (psia)	Level (ft)
8-21-1 (plate)	3/8	1,015	8.89
8-25-1 (plate)	½	1,020	8.82
1004-3	3/8	1,011	10.4
1004-2	7/8	1,011	10.5
Achilles Integral Systems Tests			
Test		Pressure (psia)	Power (Kw)
A1L066		17.6	80
A1L069		29.4	80

Table 21.6-5
NOTRUMP AP600 SBLOCA Integral Systems Assessment Tests

SPES-2 Test Simulations	
Test 1	1 in. cold-leg break
Test 3	2 in. cold-leg break
Test 5	2 in. DVI line break
Test 6	DEG DVI line break
Test 7	2 in. CL/CMT-B balance line break
Test 8	DEG CL/CMT-B balance line break
OSU Test Simulations	
Test SB9	2 in. CL3/CMT-1 balance line break
Test SB10	DEG CMT-1 balance line break
Test SB12	DEG DVI-1 line break
Test SB13	2 in. DVI-1 line break
Test SB14	Inadvertent ADS-1 open
Test SB18	2 in. cold-leg 3 break
Test SB23	0.5 in cold-leg 3 break

Table 21.6-6
Westinghouse AP600 LBLOCA PIRT with Comparisons to the CSAU LBLOCA PIRT and Westinghouse's Three- and Four-Loop Plant LBLOCA PIRT

	AP600 PIRT			Three- and Four-loop Plant PIRT			CSAU PIRT		
COMPONENT/PHENOMENA	B	RF	RD	B	RF	RD	B	RF	RD
FUEL ROD									
Stored Energy	9	5		9	5		9		
Oxidation			5			7			8
Decay Heat		5	8		5	8			8
Reactivity - Void	6			6					
Reactivity - Boron			5			5			
*Gap Conductance	8			8					8
CORE									
DNB	8			8					
Post-CHF Heat Transfer	8	8		8	8		7	8	
Rewet	7	5		6	5		8	7	
Reflood Heat Transfer			7			9			9
Nucleate Boiling									
Single-Phase Vapor Natural Circulation									
*3-D Flow	7		5	7		5			9
*Void Generation /Distribution	7			7					9
Entrainment/Deentrainment			6			8			
Flow Reversal/Stagnation	8			8					
Radiation Heat Transfer									
Level			8			8			

Table 21.6-6
Westinghouse AP600 LBLOCA PIRT with Comparisons to the CSAU LBLOCA PIRT and Westinghouse's Three- and Four-Loop Plant LBLOCA PIRT

	AP600 PIRT			Three- and Four-loop Plant PIRT			CSAU PIRT		
COMPONENT/PHENOMENA	B	RF	RD	B	RF	RD	B	RF	RD
UPPER HEAD									
Initial Water Temperature	8			5					
Flow Path Area	8			5					
Blowdown Flow	8			5					
UPPER PLENUM									
Hot Assembly Location	7			7					
*Entrainment/Deentrainment			6			6			9
Phase Separation			6			6			
CCFL/Fall-back									
Two-Phase Convection									
HOT LEG									
*Entrainment/Deentrainment									9
Flow Reversal	5			5					
Void Distribution									
Two-Phase Convection									
PRESSURIZER									
Early Quench/Flow	5			5			7		
Critical Flow in Surge Line									
Flashing									
Automatic Depressurization System (ADS) Interaction	NA	NA	NA						
STEAM GENERATOR									
Steam Binding			7			7			9
Delta-P, Form Losses									

Table 21.6-6
Westinghouse AP600 LBLOCA PIRT with Comparisons to the CSAU LBLOCA PIRT and Westinghouse's Three- and Four-Loop Plant LBLOCA PIRT

	AP600 PIRT			Three- and Four-loop Plant PIRT			CSAU PIRT		
COMPONENT/PHENOMENA	B	RF	RD	B	RF	RD	B	RF	RD
Passive Residual Heat Removal									
PUMP									
Two-Phase Performance	5			5			9		
Delta-P, Form Losses	8	5	7	8	5	7			8
COLD LEG/ACCUMULATOR									
Condensation		NA			8	5		9	
Noncondensible Gases			5			5			9
Discharge		8	9		8	6			
Flow Asymmetries	6			5					
High-Pressure Injection Mixing									
CMT Mixing/Interaction	NA	NA	NA						
DOWNCOMER									
Entrainment/Deentrainment		8	6		8	7		8	
Condensation		8			8			9	
CCF, Slug, Nonequilibrium		8			8				
Hot Wall (Vessel/Barrel)			7			7		5	7
Hot Wall (Radial Reflector)			5						
Two-Phase Convection									
Saturated Nucleate Boiling									
3-D Effects	5	8		5	8			9	
Flashing									
Liquid Level Oscillations			7			7			
DVI - Accumulator		8	7						

Table 21.6-6
Westinghouse AP600 LBLOCA PIRT with Comparisons to the CSAU LBLOCA PIRT and Westinghouse's Three- and Four-Loop Plant LBLOCA PIRT

	AP600 PIRT			Three- and Four-loop Plant PIRT			CSAU PIRT		
COMPONENT/PHENOMENA	B	RF	RD	B	RF	RD	B	RF	RD
IRWST Mixing	NA	NA	NA						
CMT Mixing/Interaction	NA	NA	NA						
LOWER PLENUM									
Sweep-Out		6			6			7	
Hot Wall			7			7			
Multidimensional Effects									
BREAK									
Critical Flow	9	6		9	6		9	7	
Flashing									
Containment Pressure			6			8			
ADS Flow	NA	NA	NA						
LOOP									
*Two-Phase Delta-P							7		
*Oscillations								7	9
Flow Split	8	5		8	5			7	

Table 21.6-7 Comparison of Containment Codes

Capability or Model	WGOTHIC	CONTEMPT 4 MOD 6	CONTAIN
Mass Balances	1. Steam 2. Liquid films, pools 3. Drops 4. Ice 5. Each gas	1. Steam 2. Pools 3. Each gas	1. Steam 2. Each gas
Energy Balance	1. Steam/gas 2. Water 3. Droplet	1. Steam/gas 2. Pools	1. Steam/gas 2. Water in lower part of cell
Force Balance	1. Steam/gas 2. Film/slugs 3. Drops	1. Steam/gas	1. Steam/gas
Lumped-Parameter Analysis	Yes	Yes	Yes
Multi-Dimensional Analysis	3-D Cartesian	No	No
Momentum Directions	3	0	0
Momentum Approach	Control volume or junctions	Junction	Junction
Turbulence	Yes	No	No
General Multi-phase Flow	Yes	No	No
Buoyancy Dominated Flow	Yes	Yes	Yes
Momentum Dominated Flows (Jet Entrainment)	Yes	No	No
Sprays	Mechanistic drop transport. Different coefficients used for distributed- and lumped-parameters.	No drop transport, convective heat transfer.	No drop transport convective heat transfer.
Engineered Safety Equipment (Pumps, Valves, etc)	Yes	Yes	Yes
Ice Condenser, Suppression Pool	Basic modeling	Special compartment	Special Compartment
AP600 PCS	"Clime" model for PCS convection, conduction and internal/external condensation.	Heat transfer to ambient is a boundary condition.	PCS film wetted area is a boundary condition.

Table 21.6-8
Comparison Between WGOTHIC and CONTEMPT Interfacial Heat and Mass Transfer for Lumped-parameter Modeling

WGOTHIC	*CONTEMPT*
Sensible heat transfer between the vapor and the pool surface, using turbulent free and forced convection heat transfer coefficients.	Sensible heat transfer between the vapor and the pool surface, using laminar and turbulent free convection heat transfer coefficients.
Mass transfer at the pool surface. Mass transfer coefficient is for turbulent free or forced convection by heat and mass transfer analogy. Driving potential is $(x_{si} - x_{sb})$ Where x_{si} is the interface steam concentration x_{sb} is the bulk steam concentration	Mass transfer at the pool surface. Mass transfer coefficient is for turbulent free convection by heat and mass transfer analogy. Driving potential is $\left\lvert \frac{1-x_{sb}}{1-x_{si}} \right\rvert$
Sensible heat transfer between the pool and the pool surface, using turbulent free and forced convection heat transfer coefficients. Pool surface temperature is calculated using mass and energy balances at the interface.	Pool surface temperature assumed equal to T_{sat}.
Heat and mass transfer from sprays calculated using same basic model as for the pool interface. Heat and mass transfer coefficients appropriate for water droplets are used.	Heat and mass transfer from sprays calculated using specified spray efficiency.
Blowdown water and steam components adjust to containment conditions based on fundamental models for interface heat and mass transfer.	Blowdown water and steam is forced to equilibrium at either the containment total pressure or the steam partial pressure.
Homogeneous and heterogeneous nucleation fog models.	No fog model.

Table 21.6-9
Comparison of Correlations for Heat Transfer, Condensation and Evaporation Implemented in WGOTHIC and CONTEMPT-LT/028

WGOTHIC	*CONTEMPT-LT/028*
Condensation heat transfer: • Mass Transfer Analogy (to PCS shell) McAdams - free convection (interior) Colburn - forced convection (exterior) • Uchida - internal heat sinks • Tagami - not used for AP600 • Gido Koestel - not used for AP600	Condensation heat transfer: • Uchida • Tagami
Six correlations for free convection. Correlation for forced convection.	One correlation for free convection.
Convection to the air/steam mixture, water or split between the two phases.	Convection to the air/steam mixture.
Specified heat transfer coefficient as function of time, Reynolds number, Rayleigh number or any combination of GOTHIC calculated variables.	Specified heat transfer coefficient as function of time or temperature.

Table 21.6-10 Clime Heat Transfer Correlations

Heat Transfer Mechanism		Correlation	Comment
Free convection from the containment atmosphere to the condensing film on the inner surface of the containment shell.		McAdams Ref. 21.6.5.4.12	Assumes containment is well mixed. See discussion in Sections 21.6.5.4.1,21.6.5.5.3, 21.6.5.6.1, 21.6.5.7.4
Conduction through condensate film.		Chun and Seban Ref. 21.6.5.4.13	Assumes wavy laminar flow. See discussion in Sections 21.6.5.4.3.4, 21.6.5.5.3, and 21.6.5.6.4
Conduction through the steel shell.			
Conduction through PCS film.		Chun and Seban Ref. 21.6.5.4.13	Assumes wavy laminar flow. See discussion in Sections 21.6.5.4.3.4, 21.6.5.5.3 and 21.6.5.6.4.
From Wet Sectors to Riser	Evaporation of the PCS water film to air.	Heat/mass analogy	Bias factor included*
From Dry Sectors to Riser	Forced convection to air.	Colburn	Bias factor included*
	Radiation to the baffle.		
From Riser To Baffle	Mixed convection to baffle from the riser air.	Churchill Ref. 21.6.5.4.23	Bias factor included*
	Radiation from the shell.		
	Condensation on the baffle.	Heat/mass analogy	Bias factor included*
Conduction through the baffle.			
Opposed mixed convection from the shield building and downcomer side of the baffle and upper portions of the riser including the top of the containment dome and shield building.		Churchill Ref. 21.6.5.4.23	Bias factor included*
Radiation between the baffle and the shield building.			
Conduction through the shield wall.			
Condensation/conduction to miscellaneous heat structures located in the downcomer and riser and the top section of the chimney.		Uchida Ref. 21.6.5.4.10	

*The conservative bias factors are discussed in Section 21.6.5.6.1 and 21.6.5.6.4.

Table 21.6-11
Simplified Summary of PCS Flow Rates and Coverage Area Characterization
(Data from SSAR Table 6.2.2-1)

Phase	Time Frame (hours)	Flow Rate		Heat Flux W/m^2 ($BTU/hr\text{-}ft^2$)	Coverage Below 2nd Weir (percent)
		liters/sec (gpm)	kg/sec (lbm/sec)		
I	0 to 3	27.9 (442)	27.6 (60.8)	4728.8 - 12610.1. (1500 - 4000)	90
II	3 to 30	7.8 (123.5)	7.7 (17.0)	3152.5 - 6305.1 (1000 - 2000)	51
III	30 to 72	4.6 (72.5)	4.5 (10.0)	3152.5 - 6305.1 (1000 - 2000)	30

Table 21.6-12
Evaluation of Conservatism in Evaporated-flow Model

Phenomena	Effect	Comment
Coverage area - Height - Stripe width	Westinghouse calculates the applied flow at the second weir assuming no evaporation occurs on the upper dome. The mass and energy of the flow applied at the second weir may be overpredicted. Constant 90 percent "wet" area throughout event.	These are small non-conservatisms. Significant evaporation occurs after the second weir. 90 percent flow from cold WDT tests at 220 gpm is conservative for 440 gpm (first 3 hrs.). After 3 hrs., effect is not significant if the containment above deck region is "well mixed", i.e., nearly homogeneous in temperature and non-condensable distribution.
Heat convected by water film	Latent heat of PCS runoff is neglected.	This is a small conservatism for peak pressure period. May be significant for the 1200 second to 30 hour period when the runoff fraction becomes large.
Resistance of external water film	PCS film thickness may be significantly underpredicted. Film may reach saturation temperature earlier.	This non-conservatism may be significant after the peak pressure period (>1200 secs.) when the runoff fraction is large.
Buoyancy driven air flow	Artificial water coverage profile skews evaporation heat flux towards higher elevations.	Skew expected only for post 3 hour period. Effect is small, conservative reduction in buoyant driving force.
Convection and radiation heat transfer from dry clime	Artificial water coverage profile lowers dry clime convection and radiation from upper elevations; raises convection and radiation from lower elevations.	Skew expected only for post 3 hour period. Errors may be self compensating.

Table 21.6-13
Phenomena Identification and Ranking According to
Effect on Containment Pressure

Component or Volume	Phenomenon or Parameter	LOCA Blowdown 0-30 sec	LOCA Refill 30-90 sec	LOCA Peak Pressure 90-1200 sec	LOCA Long Term >1200 sec	MSLB Blowdown 0-400 sec
Inside Containment						
1 Break Source	A. Mass and Energy	H	N/A	H	H	H
	B Direction and Elevation	H	N/A	H	L	H
	C. Momentum	H	N/A	H	L	H
	D. Density	H	N/A	H	L	H
	E. Droplet/ liquid flashing (thermal)	L	L	L	L	N/A
2 Containment Volume	A. Circulation/ Stratification	H	H	H	H	H
	B. Intercompartment flow	L	H	H	H	H
	C. Gas Compliance	H	H	H	H	H
	D. Fog (circulation)	L	H	H	H	N/A
	E. Hydrogen Release	L	L	L	L	N/A
3 Containment Solid Heat Sinks (Steel and Concrete)	A. Liquid Film Energy Transport	L	L	L	L	L
	B. Vertical Film Conduction	L	L	L	L	L
	C. Horizontal Film Conduction	L	H	H	H	H
	D. Internal Heat Sink Conduction	M	H	H	M	H
	E. Heat Capacity	M	H	H	M	H
	F. Condensation	M	H	H	M	H
	G. Convection from containment	L	M	L	L	L
	H. Radiation from containment	L	M	L	L	L
4 Initial Conditions	A. Initial temperature	M	M	M	M	M
	B. Initial humidity	M	M	M	M	M
	C. Initial pressure	M	M	M	M	M

Component or Volume	Phenomenon or Parameter	LOCA Blowdown 0-30 sec	LOCA Refill 30-90 sec	LOCA Peak Pressure 90-1200 sec	LOCA Long Term >1200 sec	MSLB Blowdown 0-400 sec
5 Break Pool	A. Circulation/ Stratification in the pool	L	L	L	M	L
	B. Condensation/ evaporation	L	L	L	M	L
	C. Convection within containment volume	L	L	L	L	L
	D. Radiation within containment volume	L	L	L	L	L
	E. Conduction in pool	L	L	L	M	L
	F. Compartment filling	L	L	L	L	L
6 IRWST	A. Mixing/ Stratification (gas & water)	L	L	L	L	L
	B. Condensation	L	L	L	L	L
	C. Convection	L	L	L	L	L
	D. Radiation	L	L	L	L	L
	E. Conduction in liquid	L	L	L	L	L
	F. Liquid level changes	L	L	L	L	L
Containment Shell						
7 Steel Shell	A. Convection from containment	L	L	L	L	L
	B. Radiation from containment	L	L	L	L	L
	C. Condensation	H	H	H	H	H
	D. Inside film conduction	L	L	L	L	L
	E. Inside film energy transport	L	L	L	M	L
	F. Conduction through shell	H	H	H	H	H
	G. Heat capacity of shell	H	H	H	L	H
	H. Convection to riser annulus	L	L	L	M	L
	I. Radiation to baffle	L	L	L	M	L
	J. Radiation to chimney	L	L	L	L	L
	K. Radiation to fog/air mixture	L	L	L	L	L
	L. Outside film conduction	N/A	N/A	L	L	L
	M. Outside film energy transport	N/A	N/A	M	M	L
	N. Evaporation to riser annulus	N/A	N/A	H	H	M

Component or Volume	Phenomenon or Parameter	LOCA Blowdown 0-30 sec	LOCA Refill 30-90 sec	LOCA Peak Pressure 90-1200 sec	LOCA Long Term >1200 sec	MSLB Blowdown 0-400 sec
8 PCS Cooling Water	A. PCCWST flow rate	N/A	N/A	H	H	L
	B. PCCWST water temperature	N/A	N/A	M	M	L
	C. Water film stability and coverage	N/A	N/A	H	H	L
	D. Film stripping	N/A	N/A	L	L	L
	E. Film drag	N/A	N/A	L	L	L
Outside Containment						
9 Riser Annulus & Chimney Volume	A. PCS Natural Circulation	L	L	M	M	M
	B. Vapor Acceleration	N/A	N/A	L	L	L
	C. Fog	N/A	N/A	L	L	N/A
	D. Flow stability	L	L	L	L	L
10 Baffle	A. Convection to riser annulus	N/A	N/A	L	M	N/A
	B. Convection to downcomer	N/A	N/A	L	M	N/A
	C. Radiation to shield building	N/A	N/A	L	L	N/A
	D. Conduction through baffle	N/A	N/A	L	M	N/A
	E. Condensation	N/A	N/A	L	L	N/A
	F. Heat capacity	N/A	N/A	L	L	N/A
	G. Leaks through baffle	N/A	N/A	M	M	N/A
11 Baffle Supports	A. Convection to riser air	L	L	L	L	L
	B. Radiation from shell	L	L	L	L	L
	C. Conduction from shell	L	L	L	L	L
	D. Heat capacity	L	L	L	L	L
12 Chimney Structure	A. Conduction through chimney	L	L	L	L	L
	B. Convection from chimney air	L	L	L	L	L
	C. Heat capacity of structure	L	L	L	L	L
	D. Condensation on chimney	L	L	L	L	L
13 Downcomer Annulus	A. PCS Natural Circulation	L	L	M	M	M
	B. Air flow stability	L	L	L	L	L

Component or Volume	Phenomenon or Parameter	LOCA Blowdown 0-30 sec	LOCA Refill 30-90 sec	LOCA Peak Pressure 90-1200 sec	LOCA Long Term >1200 sec	MSLB Blowdown 0-400 sec
14 Shield Building	A. Convection to downcomer	N/A	N/A	L	L	L
	B. Conduction through shield building	N/A	N/A	L	L	N/A
	C. Convection to environment	N/A	N/A	L	L	N/A
	D. Radiation to environment	N/A	N/A	L	L	N/A
15 External Atmosphere	A. Temperature	N/A	N/A	L	L	L
	B. Humidity	N/A	N/A	L	L	L
	C. Recirculation	N/A	N/A	L	L	L
	D. Pressure Fluctuations	N/A	N/A	L	L	L

Table 21.6-14
Summary and References for Treatment of High/Medium Ranked Phenomena

Component or Volume	Phenomenon or Parameter	Highest Ranking	Summary of Treatment (Reference Section for Treatment/Discussion)
Inside Containment			
1 Break Source	A. Mass and Energy	H	Conservatively high mass and energy source (21.6.5.7.4.1)
	B. Direction and Elevation	H	Limiting break scenarios (21.6.5.7.4.2)
	C. Momentum	H	Biased lumped-parameter noding (21.6.5.7.5)
	D. Density	H	Density of the break fluid is affected by the amount of droplets assumed to be entrained in the atmosphere. Limiting scenario selected based on sensitivity studies (21.6.5.7.4.3)
	E. Droplet/Liquid Flashing	L	One-half of liquid break flow is assumed to be in the form of drops with 8E-05 inch diameter for LOCA blowdown. This was shown to be a conservative assumption for pressurization (21.6.5.7.4.3).
2 Containment Volume	A. Circulation/ Stratification	H	Effect bounded by introducing biases and using limiting scenarios (21.6.5.7.5 & 21.6.5.7.7). Atmosphere approximates well mixed for first 1,200 seconds, i.e. up to peak pressure. Afterwards, atmosphere may not be well mixed, but only trends are important.
	B. Intercompartment Flow	H	Select scenario to minimize intercompartment flow (21.6.5.7.5)
	C. Gas Compliance	H	Standard gas constituents and properties used with conservatively low containment free volume (21.6.5.7.1 and 21.6.5.7.5)
	D. Fog (circulation)	H	Conservative drop fraction and size established by sensitivity studies (21.6.5.7.4.3)
	E. Hydrogen Release	L	Energy equal to reaction of 1 percent of zirconium in active fuel region included, but no hydrogen is included in break flow (21.6.5.7.4.4)

Component or Volume	Phenomenon or Parameter	Highest Ranking	Summary of Treatment (Reference Section for Treatment/Discussion)
3 Containment Solid Heat Sinks (Steel and Concrete)	A. Liquid Film Energy Transport	L	Energy in condensed film instantaneously transported to pool (21.6.5.7.1)
	B. Vertical Film Conductance	L	Conduction included implicitly in Uchida (21.6.5.4.1.2.2)
	C. Horizontal Film Conduction	H	Upward facing horizontal surfaces assumed insulated (21.6.5.7.5.3)
	D. Internal Heat Sink Conduction	H	One-dimensional heat conduction solution with air gap between steel and concrete (21.6.5.7.4.6)
	E. Heat Capacity	H	Conservative material properties maximize peak pressure (21.6.5.7.4.6)
	F. Condensation	H	Uchida correlation used (21.6.5.4.1.2.2)
	G. Convection from containment	M	Nominal convection correlation used (21.6.5.4.1.2.2)
	H. Radiation from containment	M	Nominal radiation correlation used (21.6.5.4.1.2.2)
4 Initial Conditions	A. Initial Temperature	M	All initial conditions are conservatively selected (21.6.5.7.4.7) Maximum Tech Spec temperature used.
	B. Initial Humidity	M	Conservative initial humidity (0 percent) used (21.6.5.7.4.7)
	C. Initial Pressure	M	Conservative initial pressure used (21.6.5.7.4.7)
5 Break Pool	A. Circulation/ Stratification in the pool	M	Stratification is included by vertical stacking of nodes (21.6.5.7.5.4)
	B. Condensation/ evaporation	M	Evaporation from pool is maximized and condensation minimized by noding to keep hotter liquid at the pool surface (21.6.5.7.5.4)
	C. Convection Within Containment Volume	L	A nominal convection correlation and pool area are used (21.6.5.7.5.4)
	D. Radiation Within Containment Volume	L	Conservatively neglected
	E. Conduction in pool	M	Water pools assumed homogeneous (21.6.5.7.5.4)
	F. Compartment Filling	L	Constant pool cross sectional area assumed (21.6.5.7.5.4)
6 IRWST	A. Mixing/Stratification (gas & water)	L	IRWST modeled as a single well mixed water volume with draining outflow as a specified function and flow calculated between IRWST and adjacent volumes. Initial level is Tech Spec minimum; transient level calculated from liquid inventory. These low ranked phenomena related to the IRWST behavior are modeled in a best estimate sense based on established conservation equation techniques. The staff finds this acceptable.
	B. Condensation	L	
	C. Convection	L	
	D. Radiation	L	
	E. Conduction in liquid	L	
	F. Liquid Level Changes	L	

Component or Volume	Phenomenon or Parameter	Highest Ranking	Summary of Treatment (Reference Section for Treatment/Discussion)
7 Containment Shell	A. Convection from containment	L	Conservative multiplier applied to McAdams free convection heat transfer coefficient (21.6.5.4.1.2.2 & 21.6.5.6.5.2)
	B. Radiation from containment	L	Conservatively neglected
	C. Condensation	H	Conservative multiplier applied. Free convection process only conservatively assumed (21.6.5.6.5.2 & 21.6.5.6.5.3)
	D. Inside film conduction	L	Chun and Seban correlation used (21.6.5.4.2.3 & 21.6.5.6.5.2)
	E. Inside film energy transport	M	Nominal model based on mass and energy conservation used to track condensate (21.6.5.4.2.1)
	F. Conduction through shell	H	1-D conduction at high PCS flow rate and 2-D conduction at lower PCS flow rates. Conservative material properties used, including degraded coating (21.6.5.4.2.3 & 21.6.5.7.4.6)
	G. Heat capacity of shell	H	Conservative material properties used (21.6.5.7.4.6)
	H. Convection to riser annulus	M	Conservative multiplier of 0.84 applied to standard correlations for heat and mass transfer (21.6.5.6.5.2)
	I. Radiation to baffle	M	Conservative model used with low emissivities and high sink temperature
	J. Radiation to chimney	L	Conservatively low emissivities used
	K. Radiation to fog/air mixture	L	Conservatively neglected
	L. Outside film conduction	L	Chun and Seban correlation used (21.6.5.4.2.3 & 21.6.5.6.5.3)
	M. Outside film energy transport	M	Conservation of mass and energy solved with low PCS flow and high initial liquid temperature (21.6.5.4.2.3)
	N. Evaporation to riser annulus	H	Conservative multiplier of 0.84 applied to standard correlations for mass transfer (21.6.5.6.5.2 & 21.6.5.6.5.3)

Component or Volume	Phenomenon or Parameter	Highest Ranking	Summary of Treatment (Reference Section for Treatment/Discussion)
8 PCS Cooling Water	A. PCCWST flow rate	H	Applied flow restricted to evaporated flow. Sensible heating of runoff flow neglected.(21.6.5.4.2.2)
	B. PCCWST water temperature	M	Conservatively high PCCWST temperature assumed (21.6.5.4.2.2)
	C. Water film stability and coverage	H	External shell water coverage area is input as a boundary condition (21.6.5.4.2.3)
	D. Film stripping	L	Neglected (21.6.5.4.2.3)
	E. Film drag	L	Neglected (21.6.5.4.2.3)
Outside Containment			
9 Riser Annulus & Chimney Volume	A. PCS Natural Circulation	M	Momentum equation is solved with 30 percent conservatism added to loss coefficients (21.6.5.6.6.3)
	B. Vapor acceleration	L	Neglected
	C. Fog	L	Neglected
	D. Flow stability	L	Potential for flow instability is negligible and is therefore neglected
10 Baffle	A. Convection to riser annulus	M	Nominal mixed convection model used (21.6.5.4.2.1 & 21.6.5. 6.5.2)
	B. Convection to downcomer	M	Nominal mixed convection model used (21.6.5.4.2.1 & 21.6.5. 6.5.2)
	C. Radiation to shield building	L	Conservative model used with low emissivities and high sink temperature
	D. Conduction through baffle	M	1-D conduction model with conservative material properties (21.6.5.4.2.1 & 21.6.5.7.4.6)
	E. Condensation	L	Nominal model including free and forced convection mass transfer (21.6.5.4.2.1 & 21.6.5. 6.5.2)
	F. Heat capacity	L	1-D conduction model with conservative material properties (21.6.5.4.2.1 & 21.6.5.7.4.6)
	G. Leaks through baffle	M	Flowpath included to model leakage
11 Baffle Supports	A. Convection to riser air	L	All baffle support related phenomena are neglected; This is acceptable to the staff for these low ranked phenomena.
	B. Radiation from shell	L	
	C. Conduction from shell	L	
	D. Heat capacity	L	

Component or Volume	Phenomenon or Parameter	Highest Ranking	Summary of Treatment (Reference Section for Treatment/Discussion)
12 Chimney Structure	A. Conduction through chimney	L	Nominal heat and mass transfer correlations used with conduction model. Conservative emissivities and material properties used. These low ranked phenomena related to the chimney structure are modeled in a best estimate sense based on established techniques. The staff finds this acceptable.
	B. Convection from chimney air	L	
	C. Heat capacity of structure	L	
	D. Condensation on chimney	L	
13 Downcomer Annulus	A. PCS Natural Circulation	M	Momentum equation is solved with 30 percent conservatism added to loss coefficients (21.6.5.6.6.3)
	B. Air flow stability	L	Potential for flow instability is negligible and is therefore neglected (21.6.5.6.6.3)
14 Shield Building	A. Convection to downcomer	L	Nominal mixed convection model used (21.6.5.6.5.2)
	B. Conduction through shield building	L	1-D conduction model with conservative material properties (21.6.5.7.4.6)
	C. Convection to environment	L	Nominal heat and mass transfer correlations used with conservative material properties (21.6.5.6.5.2)
	D. Radiation to environment	L	Conservative model used with low emissivities and high sink temperature.
15 External Atmosphere	A. Temperature	L	Maximum Tech Spec value is used (21.6.5.7.4.7)
	B. Humidity	L	Held constant at initial value (21.6.5.7.4.7)
	C. Recirculation	L	Negligible, and therefore not considered
	D. Pressure Fluctuation	L	Pressure held constant at initial value (21.6.5.7.4.7)

Table 21.6-15
Expected Operating Range for the AP600 Heat and Mass Transfer Parameters

Correlation	Parameter	AP600 range
Internal free convection Heat transfer and condensation mass transfer	$\Delta\rho/\rho$ (density ratio) $\frac{\rho_{bulk}-\rho_{surf}}{\rho_{bulk}}$	< 0.401
	Pr (Prandtl number)	0.72 to 0.90
	Sc (Schmidt number)	~ 0.52
External mixed convection Heat transfer and evaporation mass transfer	Re_d (Reynolds number)	< 189,000 (riser) < 151,000 (downcomer) < 282,000 (chimney)
	Gr_d (Grashof number)	< 1.2×10^9 (riser) < 6.2×10^9 (downcomer) < 2.1×10^{12} (chimney)
	Pr (Prandtl number)	~ 0.72
	Sc (Schmidt number)	~ 0.52
Liquid film heat transfer	Re (Reynolds number)	< 4,000
	Pr (Prandtl number)	1.5 to 3.0

Table 21.6-16
WGOTHIC Analyses of LST Using Lumped-parameter Modeling Approach

Test	Predicted/Measured Pressure (WGOTHIC V1.2)	Predicted/Measured Pressure (WGOTHIC V4.1)
Priority		
212.1A	1.15	
212.1B	1.17	
212.1C	1.20	
214.1A	1.03	Higher than 1.2 (both 14 - 34 kPa (2 - 5 psi) higher than measured)
214.1B	1.12	Higher than 1.2 (both 14 - 34 kPa (2 - 5 psi) higher than measured)
216.1A	1.11	Similar (both 28 kPa (4 psi) higher than measured)
216.1B	1.19	7 percent higher – V1.2 - 55 kPa (8 psi) higher V4.1 - 83 kPa (12 psi) higher
219.1A	1.03	About the same
219.1B	1.07	Lower (maximum 7 percent difference)
219.1C	1.31	Slightly lower (both about 55 kPa (8 psi) higher than measured)
222.1	1.18	
222.4A	1.15	V4.1 lower, small during this part
222.4B	1.28	V4.1 lower
Non-Priority		
213.1A	1.13	
213.1B	1.18	
217.1A	1.06	
217.1B	1.32	
218.1A	1.11	
218.1B	1.19	
221.1A	1.22	
221.1B	1.19	
224.1	1.25	
224.2	1.24	
202.3	1.07	

Table 21.6-17
Conservative Input Values for EM for Environmental (Outside Containment) Initial Conditions

Initial/Conservative Conditions Input Parameters	Value	Basis for Value in EM	Reference	Comments
Environmental Atmospheric Temperature	115 °F	Set to maximum safety air temperature limit by the site interface parameters	SSAR Chap. 2 WCAP-14407 Sect. 5.7 Sensitivity Study	Maximizes downcomer inlet temperature, reduces buoyancy effects
Environmental Total Pressure	14.7 psia	Standard		
Environmental Relative Humidity	22 percent at 115 °F	Limit defined by site interface parameters	SSAR Chap. 2 WCAP-14407 Sect. 5.6 Sensitivity Study	

Table 21.6-18

Conservative Input Values for EM for Inside Containment Initial Conditions

Initial/Conservative Conditions Input Parameters	Value	Basis for Value in EM	Reference	Comments
Containment Atmospheric Temperature	120 °F	Maximum Technical Specification Value. Maximizes internal heat sink temperature.	Tech. Spec. 3.6.5 (SSAR Chap. 16) WCAP-14407 Sect. 5.5 Sensitivity Study	
Containment Total Pressure	15.7 psia	Maximum Technical Specification Value. Maximizes initial pressure, amount of air, retards mass transfer	Tech. Spec. 3.6.4 (SSAR Chap. 16) WCAP-14407 Sect. 5.4 Sensitivity Study	Maximizes initial inside containment pressure prior to blowdown initiation. Maximizes initial amount of non-condensable gas content
Containment Relative Humidity	0 percent	Maximum air content	WCAP-14407 Sect. 5.3 Sensitivity Study	Maximizes initial amount of non-condensable gas content, minimizes liquid volume fraction in containment atmosphere
Initial Heat Sink Temperature	120 °F	Maximum Technical Specification Value. Maximizes initial heat sink temperatures to minimize heat transfer rate	Tech. Spec. 3.5.6 (SSAR Chap. 16) WCAP-14407 Sect. 5.5 Sensitivity Study	Maximizes all initial internal thermal conductor temperatures in order to minimize heat transfer and energy storage capacity
IRWST Liquid Volume Fraction	0.868	Minimum Technical Specification IRWST Water Volume.	Tech. Spec. 3.5.6 (SSAR Chap. 16)	Minimizes available amount of water storage for core cooling. However, maximizes non-condensable gas volume fraction from containment point of view
IRWST Water Temperature	120 °F	Maximum Technical Specification IRWST Water Temperature	Tech. Spec. 3.5.6 (SSAR Chap. 16)	Minimizes core cooling capability of replenished core coolant

Table 21.6-19
Conservative Input Values for EM for Primary System and Secondary System Conditions

Initial/Conservative Conditions Input Parameters	Basis for Value in EM	Reference	Comments
RCS Initial Conditions	Maximum operating temperature and pressure. Allowances for error and instrument dead band	SSAR Chap. 6, Section 6.2.1.3.2.1	
RCS Volume	RCS volume increased by 1.4 percent (uncertainty)	WCAP-10325-P-A	
Core Stored Energy	Core stored energy increased by 15 percent. Maximized in terms of burnup and maximum core fluid temperature	WCAP-10325-P-A	
Steam Generator Mass	Initial mass increased by 10 percent. Maximizes energy in the system	WCAP-10325-P-A	
Initial Power Level	102 percent of full power, accounting for calometric error. Maximizes energy in system	WCAP-10325-P-A	
Zircon-Water Reaction	1 percent of Zirconium reacts. Bounds guidance of SSAR for no appreciable reaction	SSAR Chap. 15	Addition of energy to maximize energy release into containment
LOCA Mass and Energy Releases	Westinghouse Primary System Safety Computer Code SATAN78	WCAP-10325-P-A	
Steam Generator Heat Release	Intact loop: 1 hour Broken loop: 0.5 hour		Substantial energy sources in SG compartments (e.g., not component) beyond containment peak pressure phase
MSLB Mass and Energy Releases	Westinghouse Primary/Secondary System Safety Computer Code	SSAR 6.2.1.4	

Table 21.6-20

Conservative Input Values for EM for Primary PCS Characteristics

Initial/Conservative Conditions Input Parameters	Basis for Value in EM	Reference	Comments
Initial Shell Temperature	Maximum Technical Specification Value for containment air temperature, which bounds initial shell temperature	Tech. Spec. 3.6.5 (SSAR Chap. 16)	
Applied External Film Flow Rate	Assumption of single failure of one of two PCS drain headers. Delivered flow is reduced by amount predicted to run off based on water coverage model. EM applies concept of flow limit by complete evaporation	WCAP-14407, Sec. 7	
External PCS Liquid Film Temperature	Set to upper bound value 120 °F. Minimizes film subcooling effect	Tech. Spec. 3.6.6 (SSAR Chap. 16)	
Film Coverage Fraction	Held constant consistent with water coverage model and flow limit specified by evaporation	WCAP-14407, Sec. 7	
PCS Coating Properties	Thermal conductivity of shell paint reduced to 25 percent of nominal value		
PCS Emissivity	Surface emissivities of shell, baffle, downcomer boundary reduced to 90 percent of nominal value		
PCS Coatings Thickness	Maximum coating thicknesses used		
Delay Time for PCS Initiation	No credit for PCS film flow prior to 337 sec. After blowdown initiation.	WCAP-14407, Sec. 7	Minimizes energy transfer to PCS, maximizes energy content inside containment
Internal Heat and Mass Transfer Correlation	Assumption of free convection only. Include multiplier of 0.73	WCAP-14326, Sec. 4.5	Neglects forced convection contribution, reduces energy transfer to PCS, maximizes energy content inside containment, eliminates potential bias of too high predicted velocities by LP-approach
External Heat and Mass Transfer Correlation	Assumption of mixed convection. Include multiplier of 0.84	WCAP-14326, Sec. 4.5	

Table 21.6-21
Conservative Input Values for EM for Geometry and Flow Characteristics

Input Parameter	Basis for Value in EM	Reference	Comments
Containment Free Volume	Nominal cold value which neglects the volumetric increase at higher temperature		Minimizes total free, inside containment volume by discarding volume increase due to thermal expansion at higher blowdown induced temperatures
Internal Heat Sinks - Area and Volume	Rev. 8 of General Arrangement Drawings. Smaller trays, piping, and miscellaneous structures are ignored		Ignoring heat sink surfaces and volumes maximizes energy content in atmosphere. No cut-off values or total volume/surface of ignored structures provided
Internal Flow Paths Area and Loss Coefficient	Rev. 8 of General Arrangement Drawings. The smaller flow paths are ignored		
External (Downcomer/Riser) Flow Paths - Loss Coefficients	Rev. 8 of General Arrangement Drawings, Loss Coefficients based on 1/6 scale annulus pressure drop test. RAI 720.440F response indicates no change is needed for new drain location	WCAP-13328	30 percent add-on values derived from experiments, increases conservatively pressure drop in bend and riser.

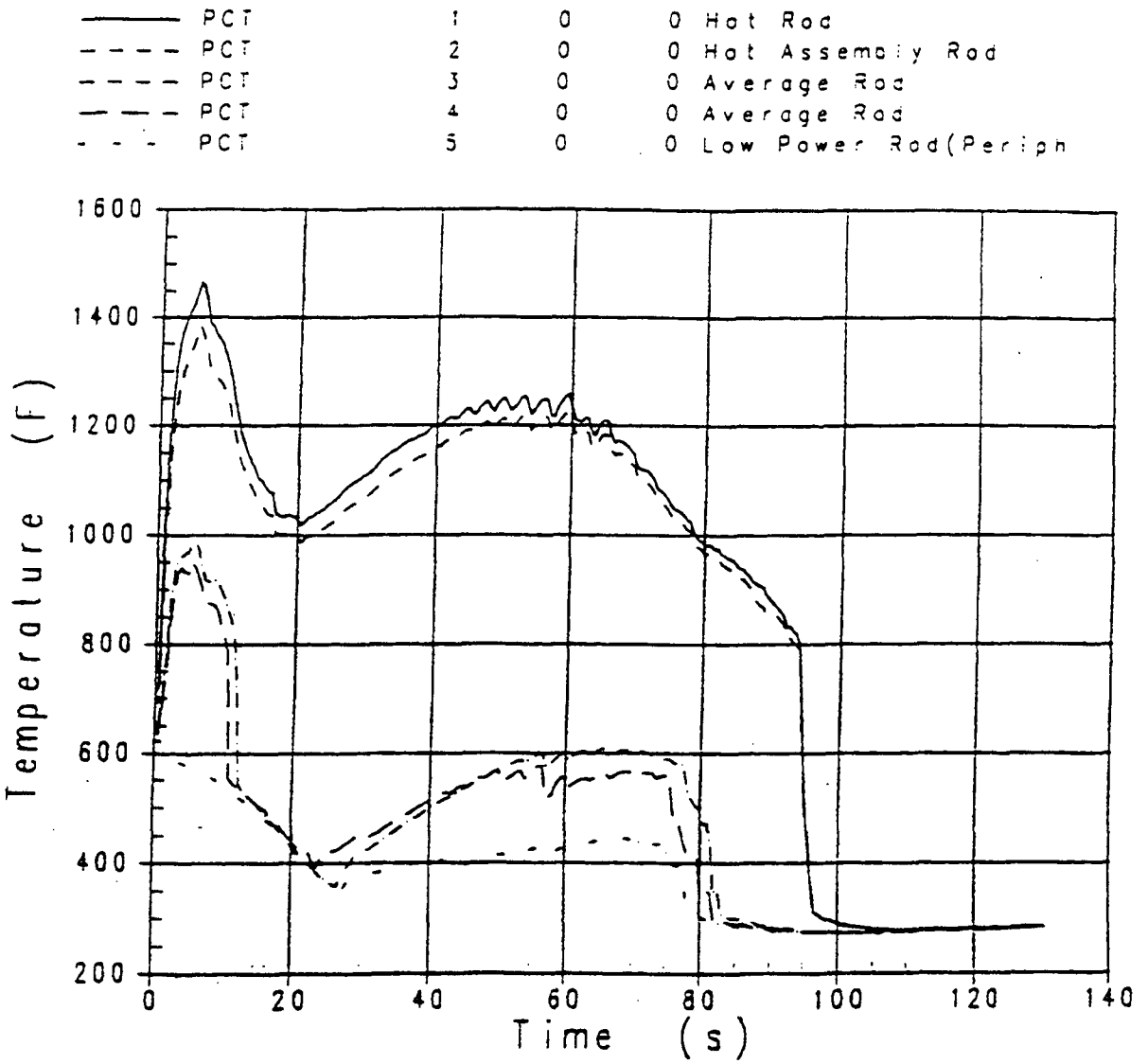


Figure 21.6-1
AP600 Peak Cladding Temperature Transient for the AP600 C_D = 0.8 DECLG Break

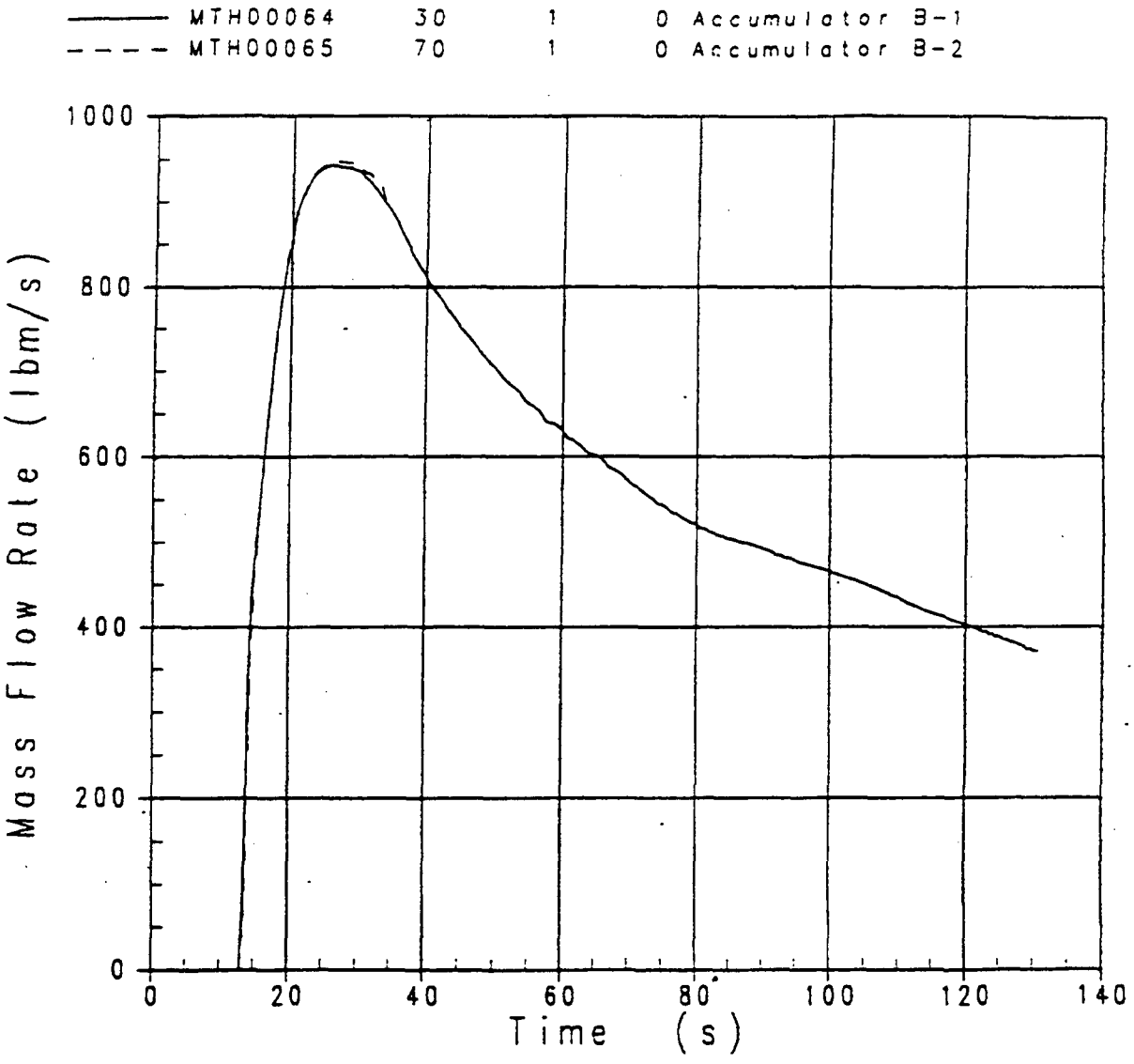


Figure 21.6-2
C_D = 0.8 DECLG Transient, Accumulator Flow Rate From One Tank

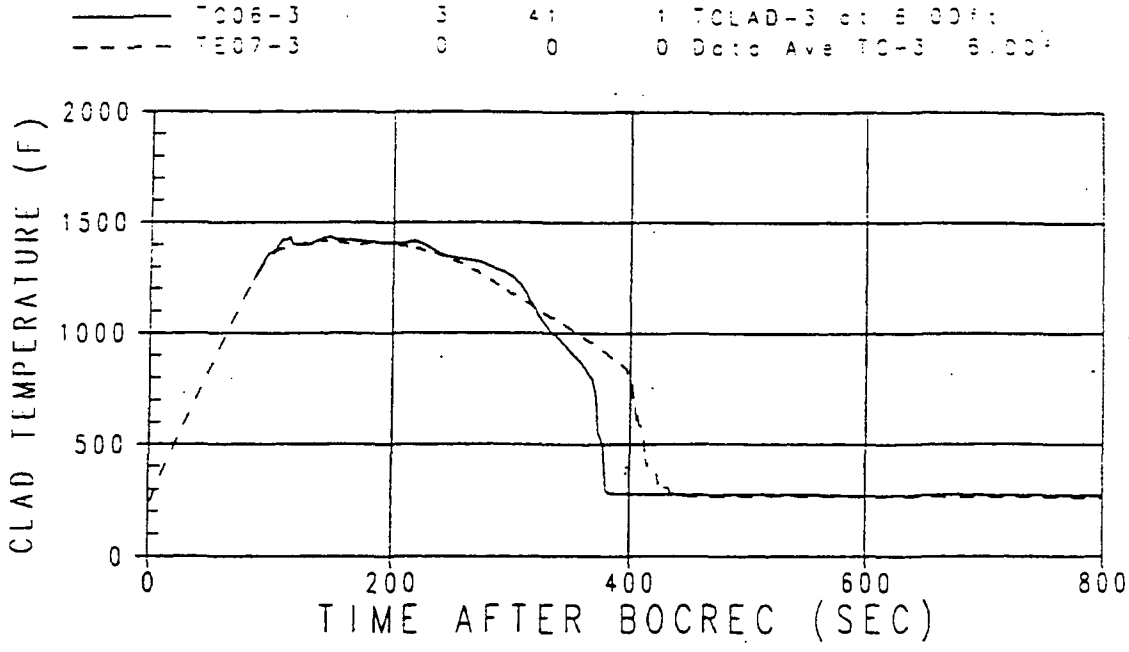


Figure 21.6-3
CCTF Run 58, Medium-Powered Rod, Clad Temperature Comparison at 6 ft

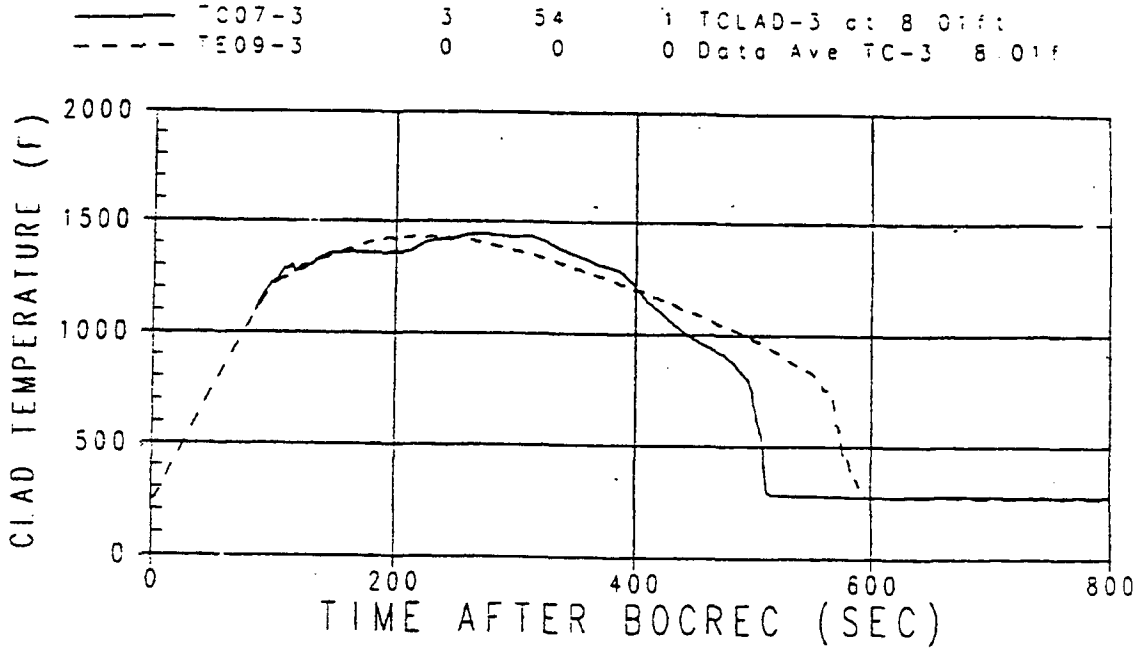


Figure 21.6-4
CCTF Run 58, Medium-Powered Rod, Clad Temperature Comparison at 8 ft

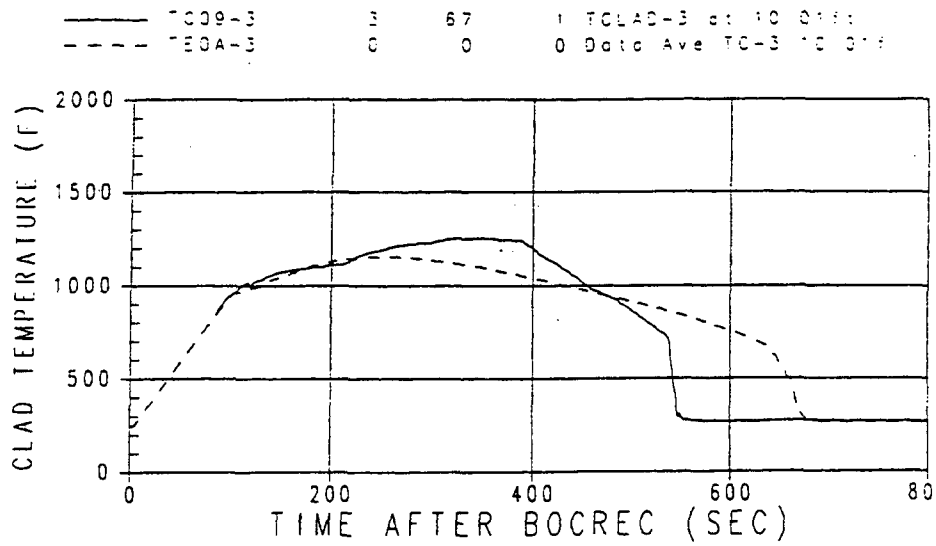


Figure 21.6-5
CCTF Run 58, Medium-Powered Rod, Clad Temperature Comparison at 10 ft

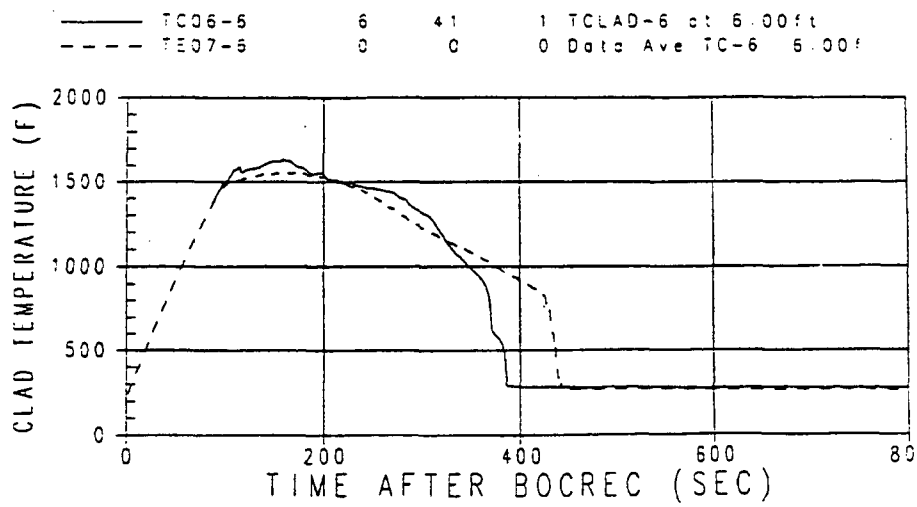


Figure 21.6-6
CCTF Run 58, High-Powered Rod, Clad Temperature Comparison at 6 ft

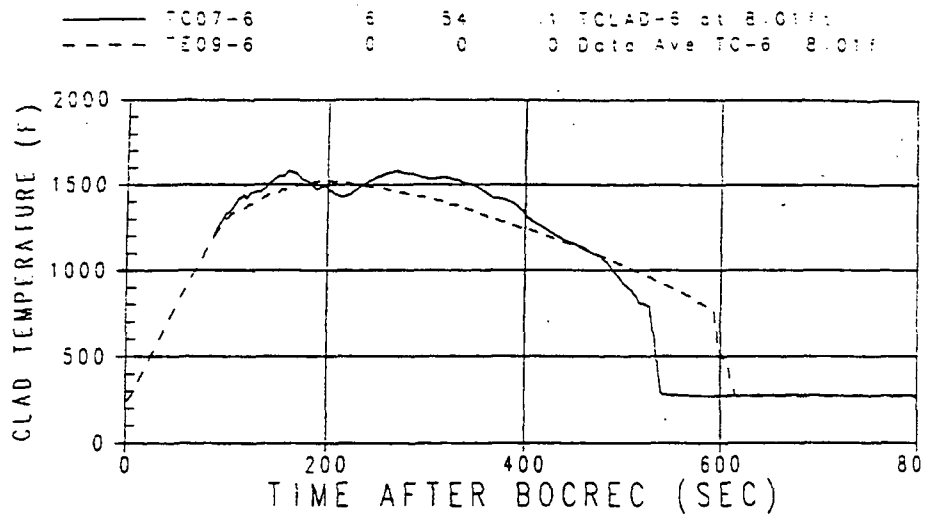


Figure 21.6-7
CCTF Run 58, High-Powered Rod, Clad Temperature Comparison at 8 ft

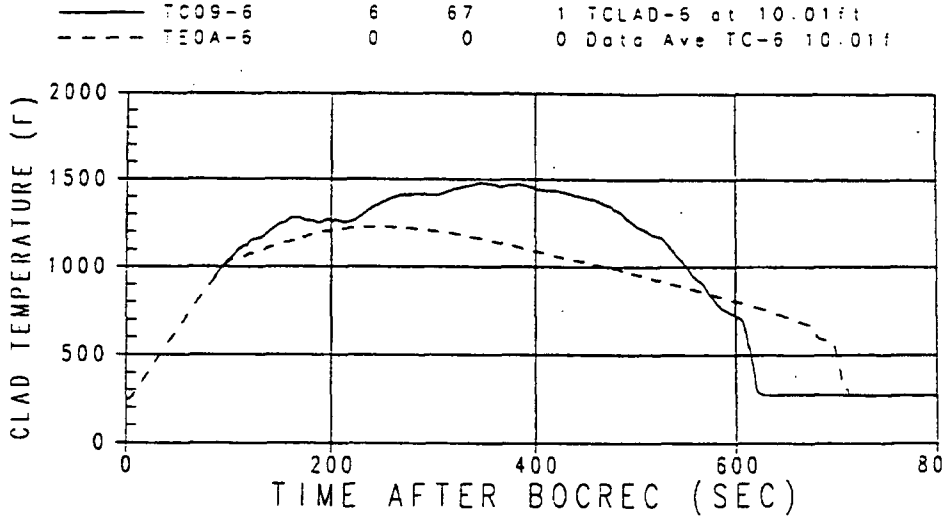


Figure 21.6-8
CCTF Run 58, High-Powered Rod, Clad Temperature Comparison at 10 ft

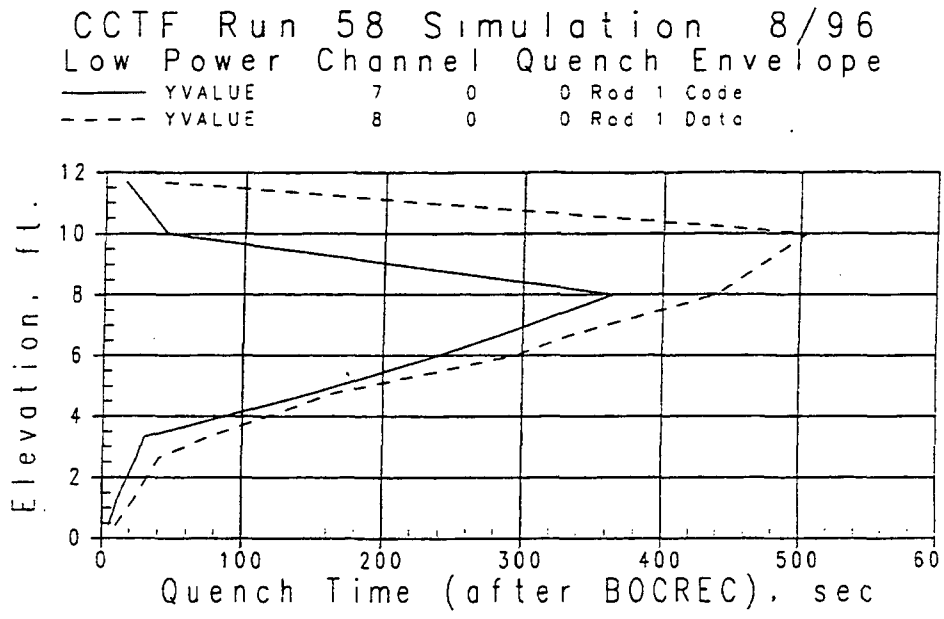


Figure 21.6-9
CCTF Run 58, Quench Envelope Comparison - Low-Powered Rod

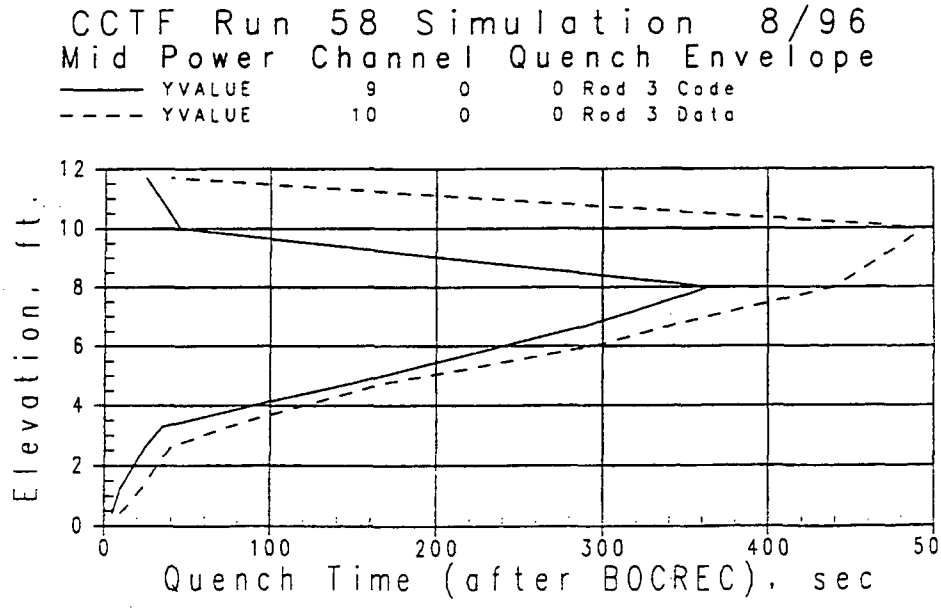


Figure 21.6-10
CCTF Run 58, Quench Envelope Comparison - Medium-Powered Rod

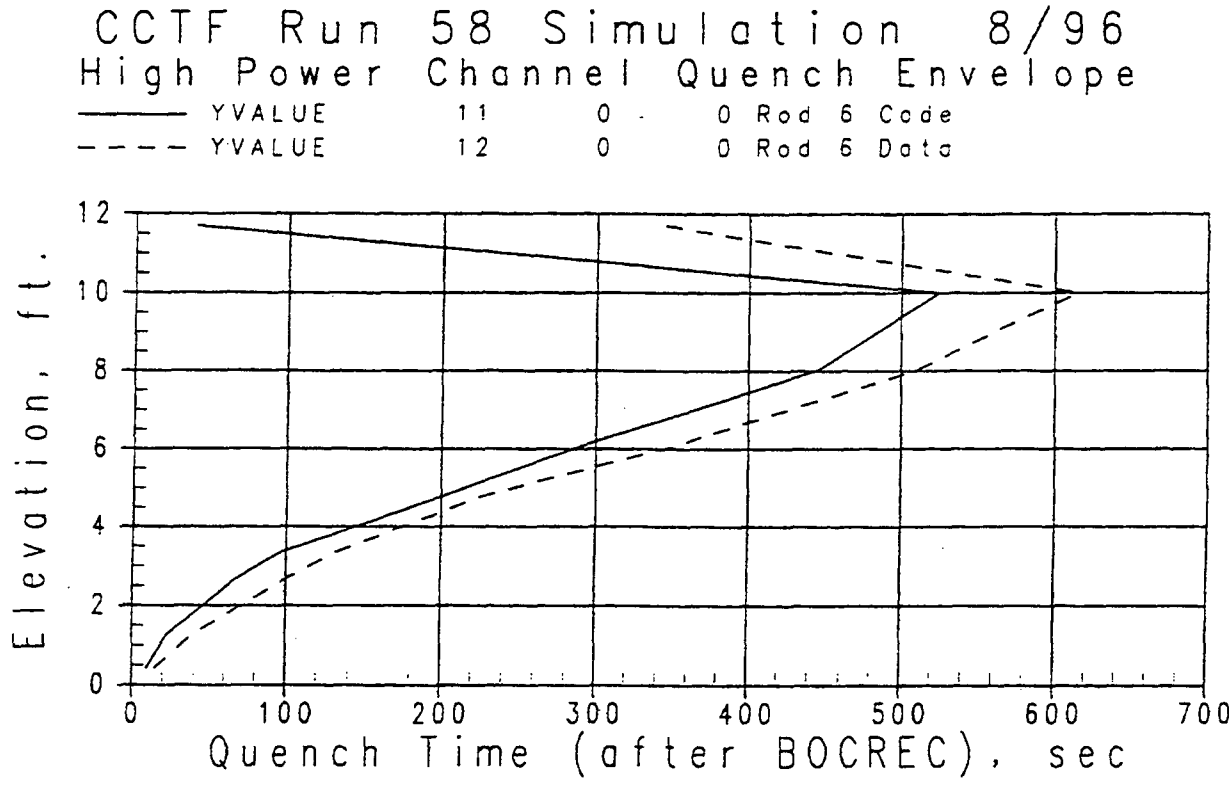
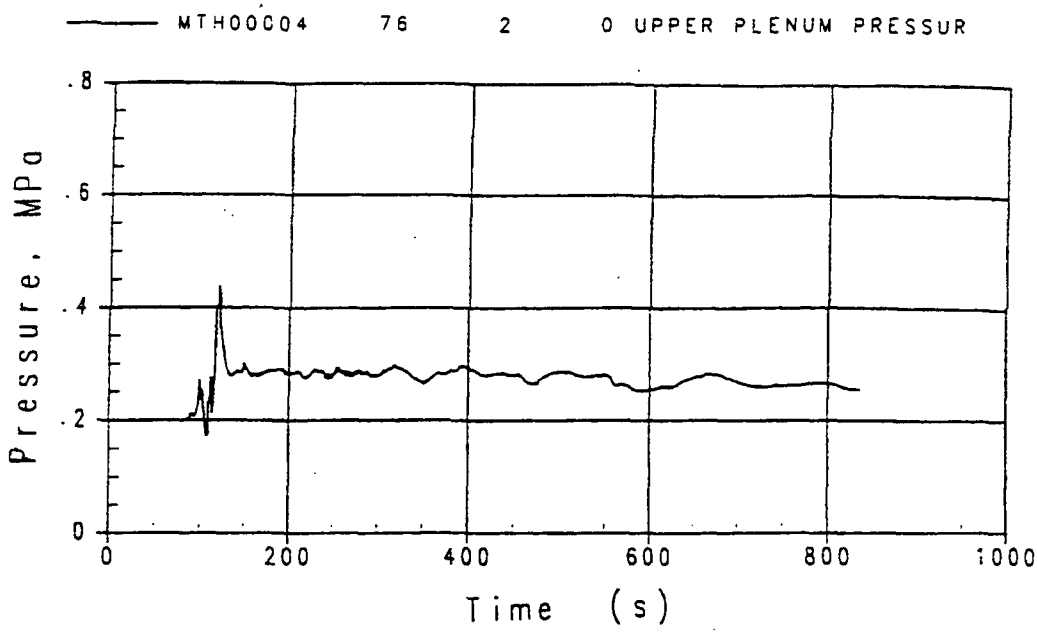


Figure 21.6-11
CCTF Run 58, Quench Envelope Comparison - High-Powered Rod



W COBRA/TRAC CCTF Run 58, Upper Plenum Pressure

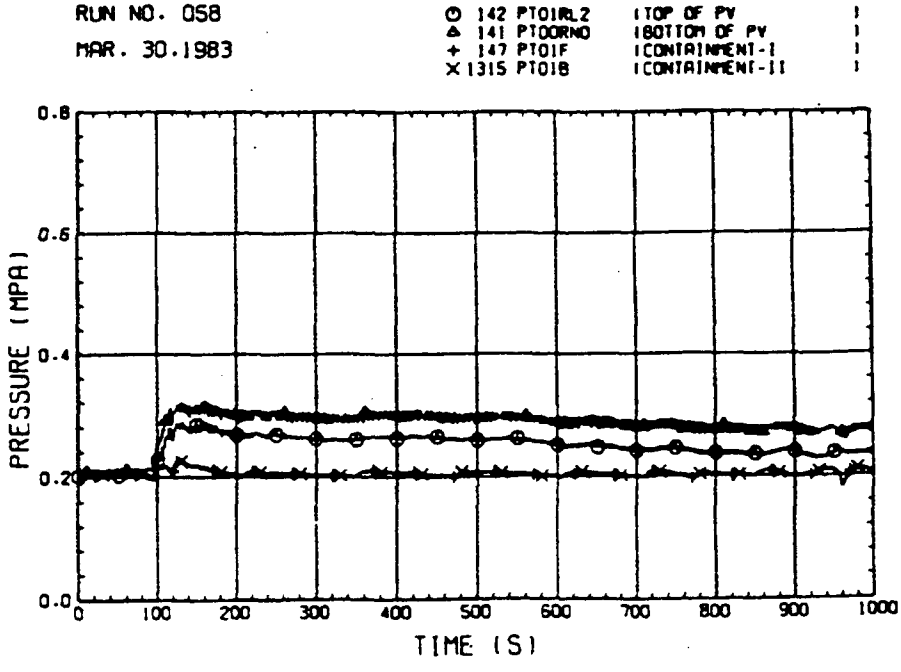
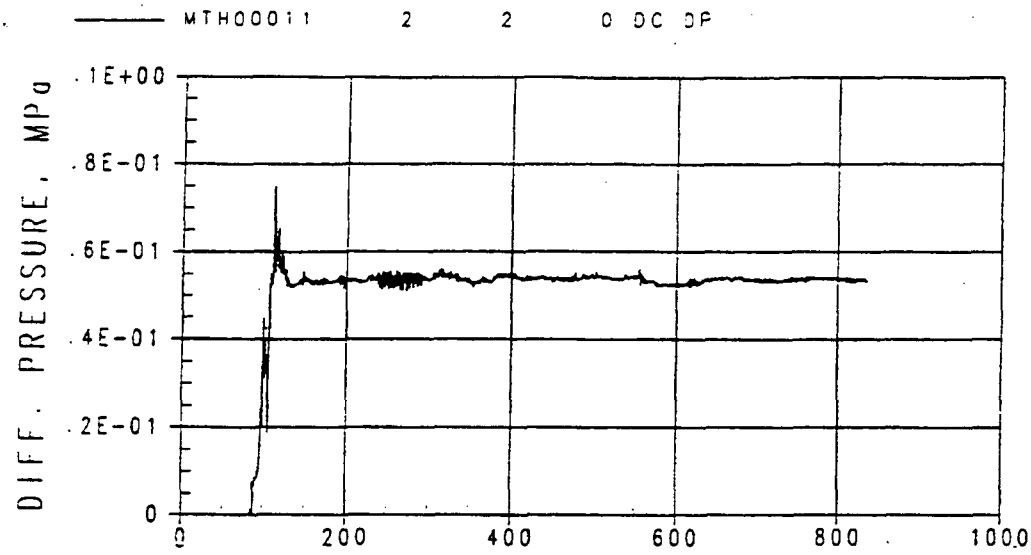


Figure C-292 from JAERI-memo 59-44b

Figure 21.6-12
CCTF Run 58, Upper Plenum Pressure Comparison



W COBRA/TRAC CCTF Run 58, Downcomer Differential Pressure

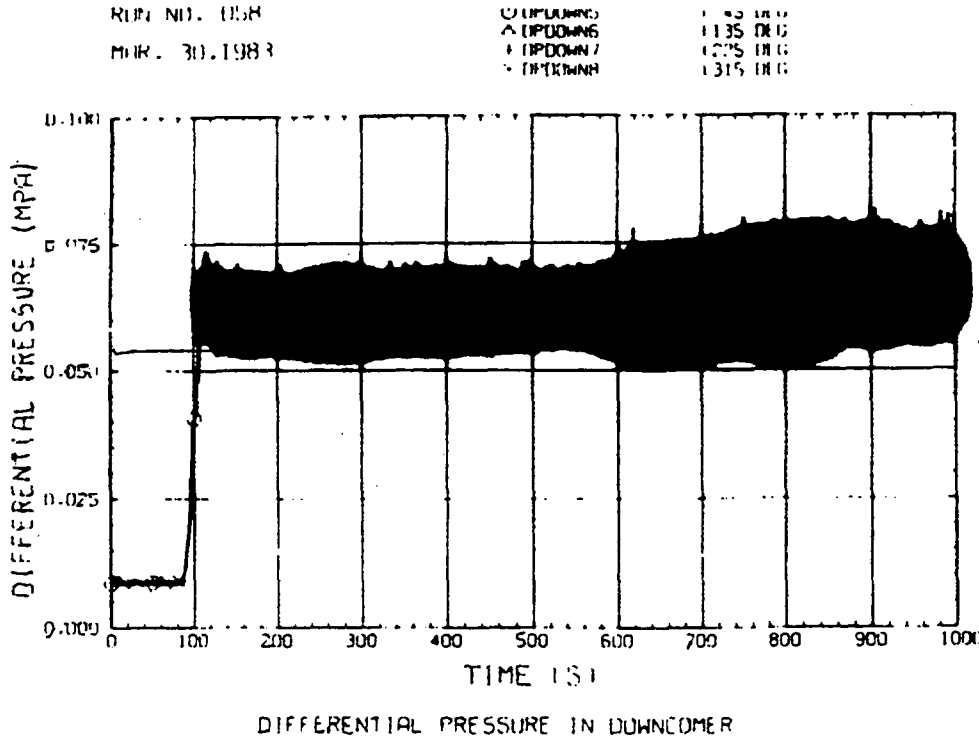
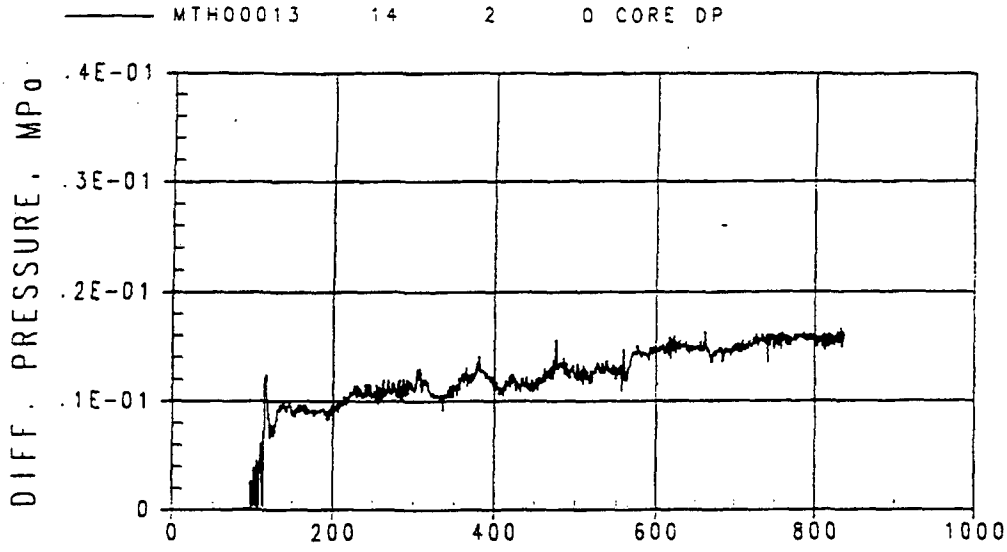


Figure D-33 from JAERI-memo 59-44b

Figure 21.6-13
CCTF Run 58, Downcomer Differential Pressure Comparison



W COBRA/TRAC CCTF Run 58, Core Differential Pressure

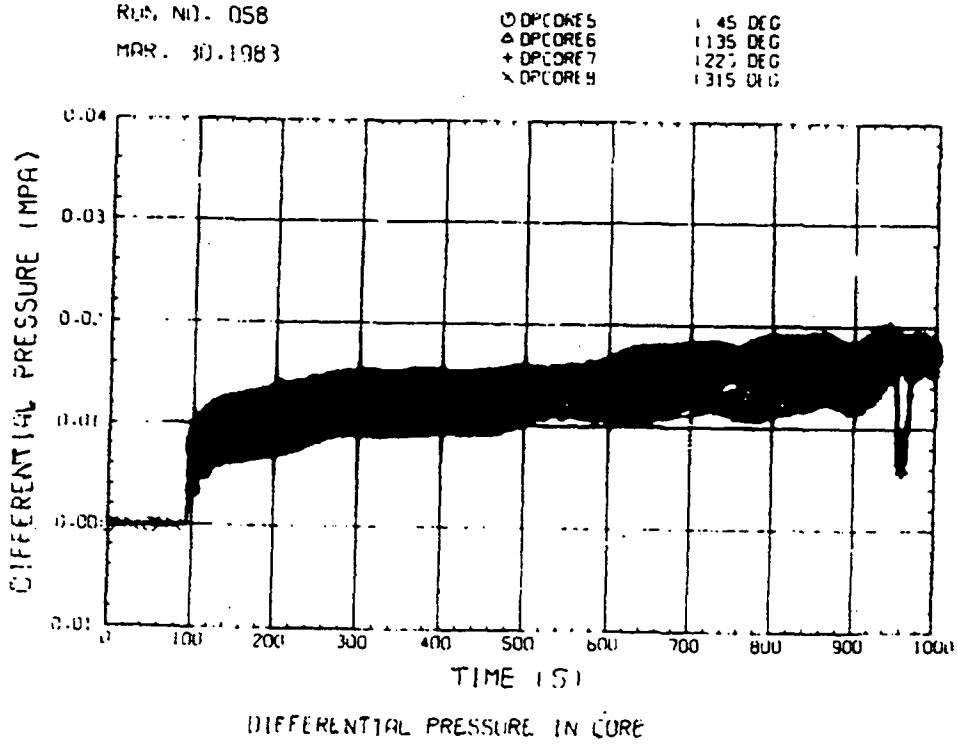
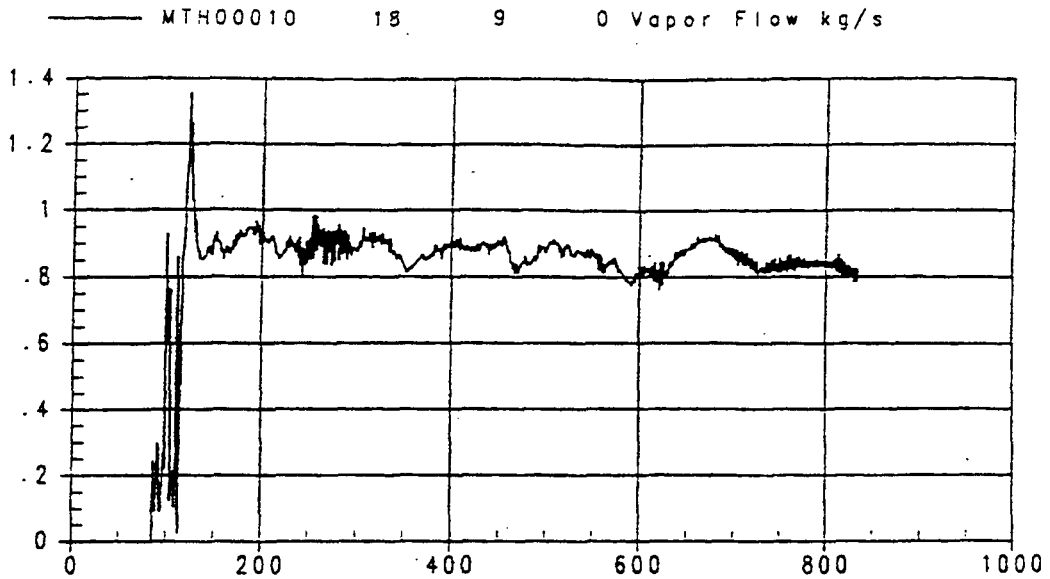


Figure D-34 from JAERI-memo 59-44b

Figure 21.6-14
CCTF Run 58, Core Differential Pressure Comparison



W COBRA/TRAC CCTF Run 58, Loop 1 Cold Leg Steam Mass Flow

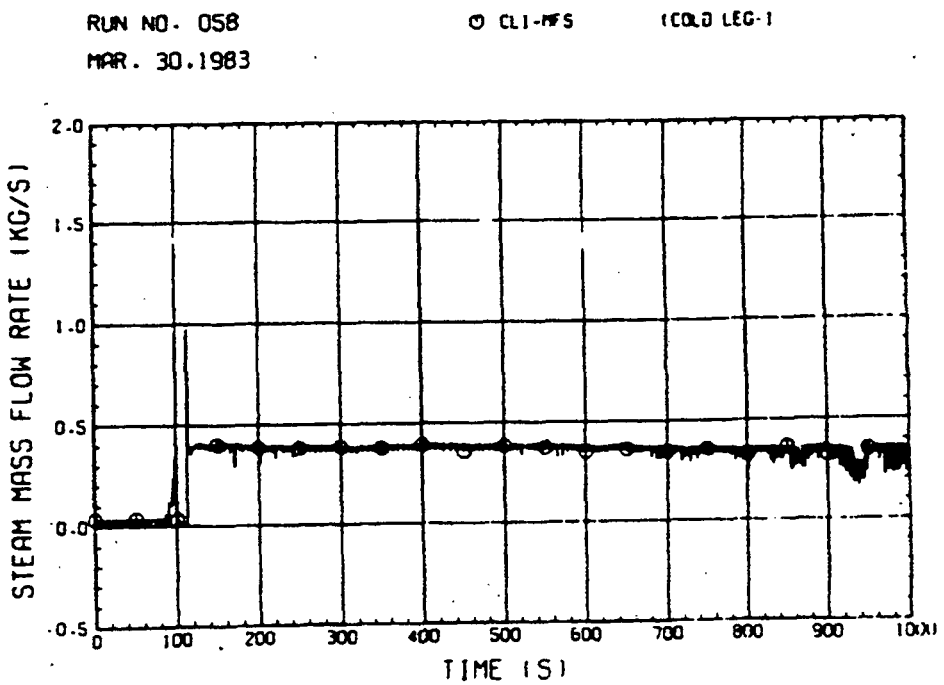
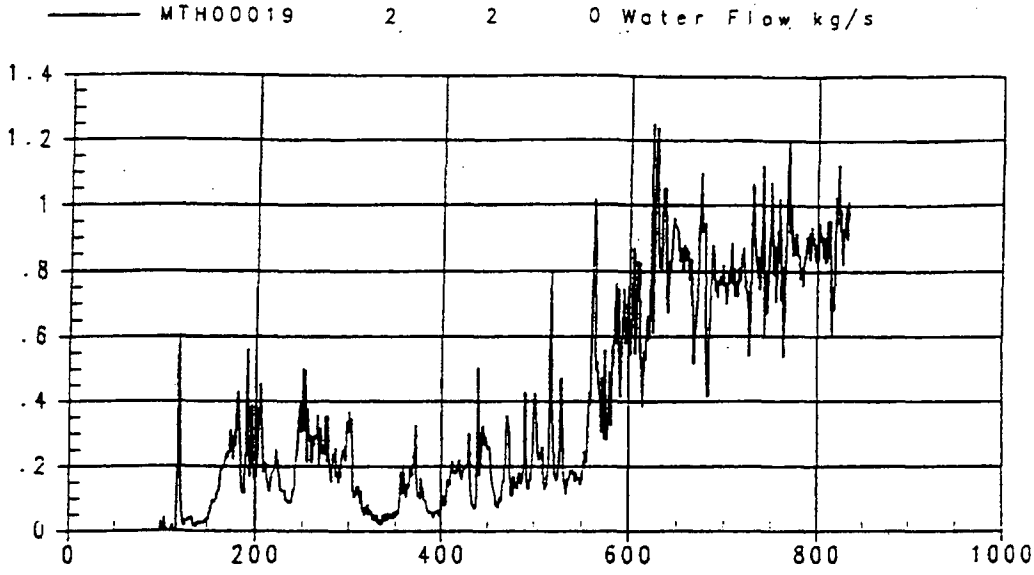


Figure E-172 from JAERI-memo 59-44b

Figure 21.6-15
CCTF Run 58, Loop 1 Cold Leg Steam Mass Flow Comparison



W COBRA/TRAC CCTF Run 58, Loop 1 Hot Leg Water Mass Flow

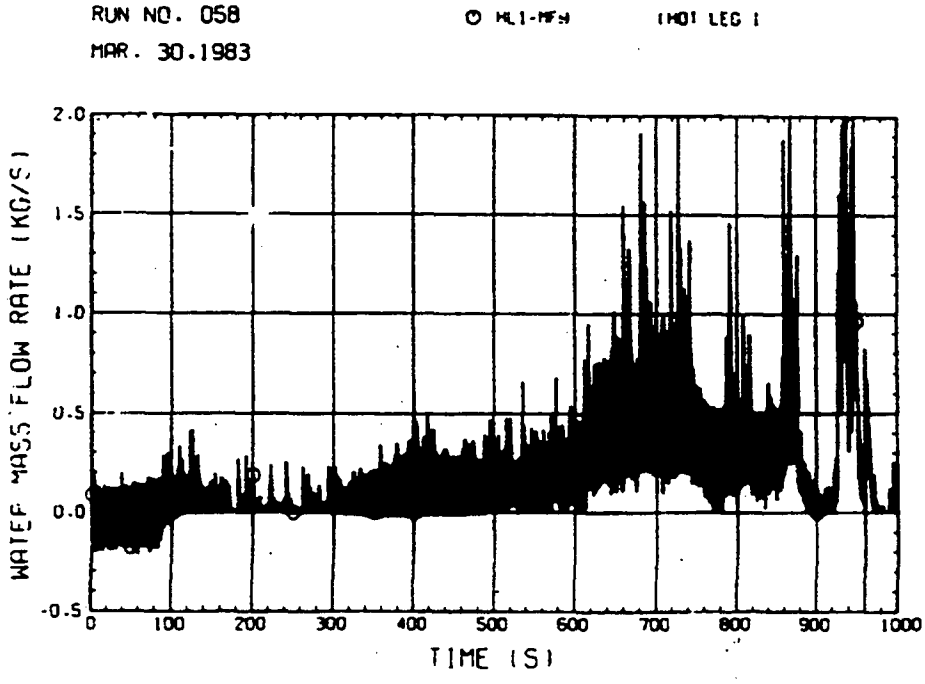
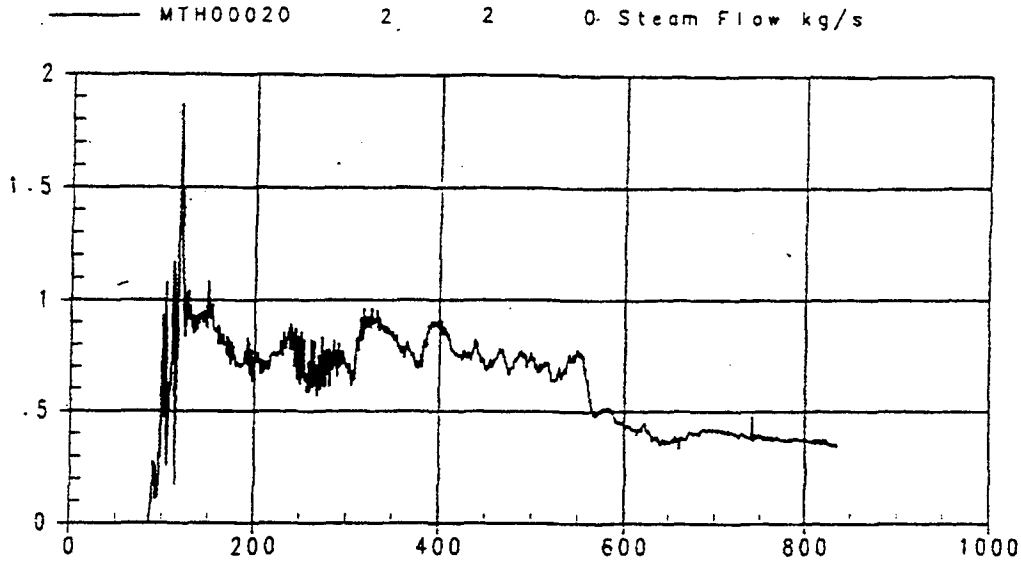


Figure E-161 from JAERI-memo 59-44b

Figure 21.6-16
CCTF Run 58, Loop 1 Hot Leg Water Mass Flow Comparison



W COBRA/TRAC CCTF Run 58, Loop 1 Hot Leg Steam Mass Flow

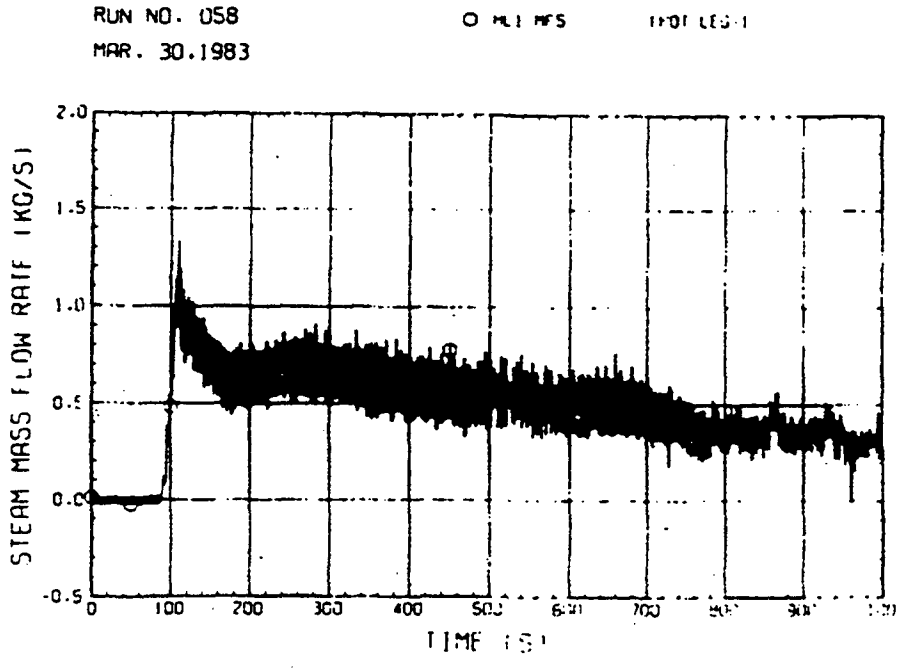
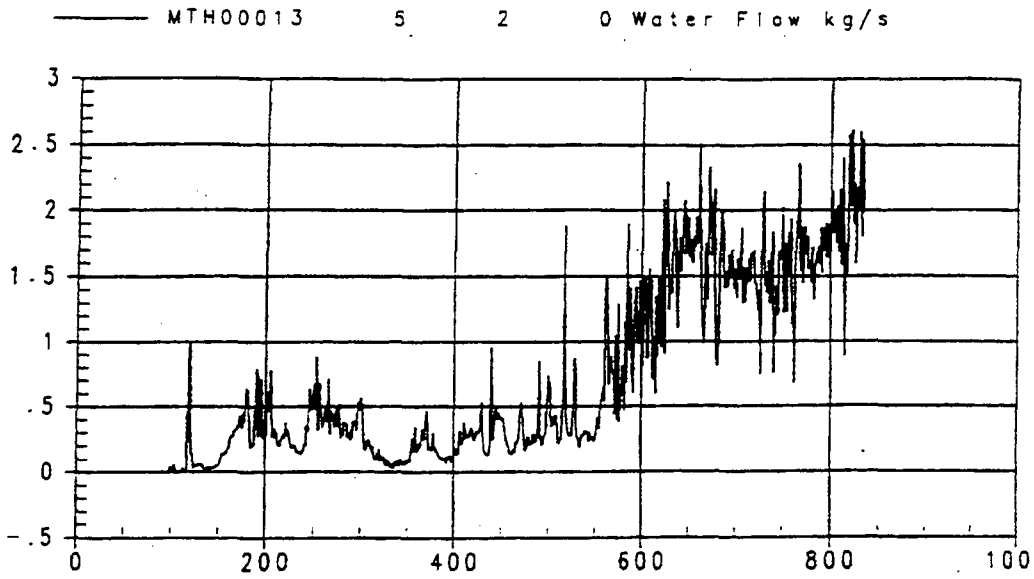


Figure E-160 from JAERI-memo 59-44b

Figure 21.6-17
CCTF Run 58, Loop 1 Hot Leg Steam Mass Flow Comparison



W COBRA/TRAC CCTF Run 58, Loop 4 Hot Leg Water Mass Flow

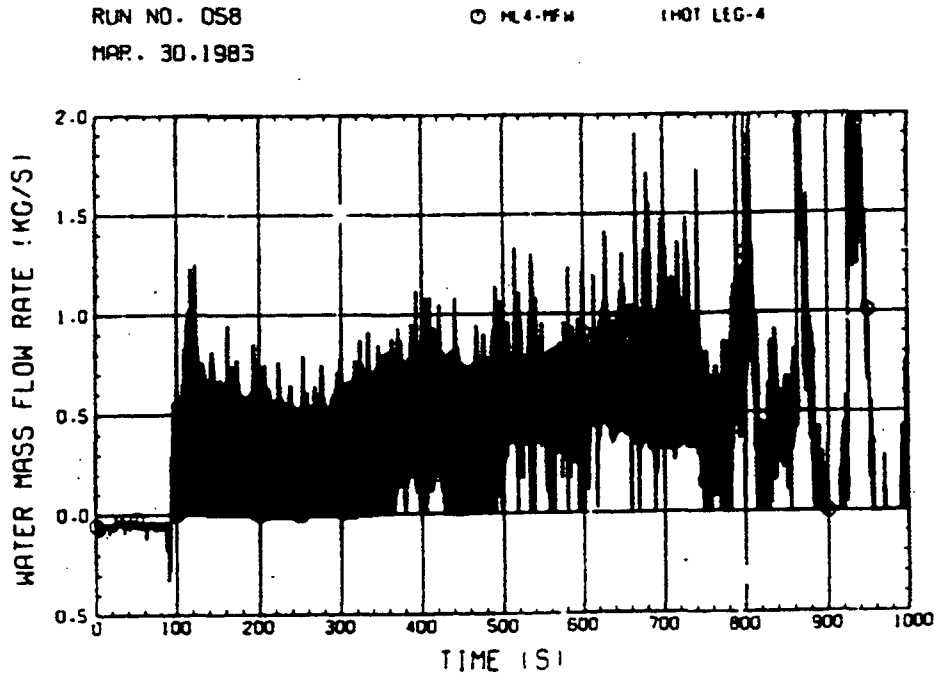
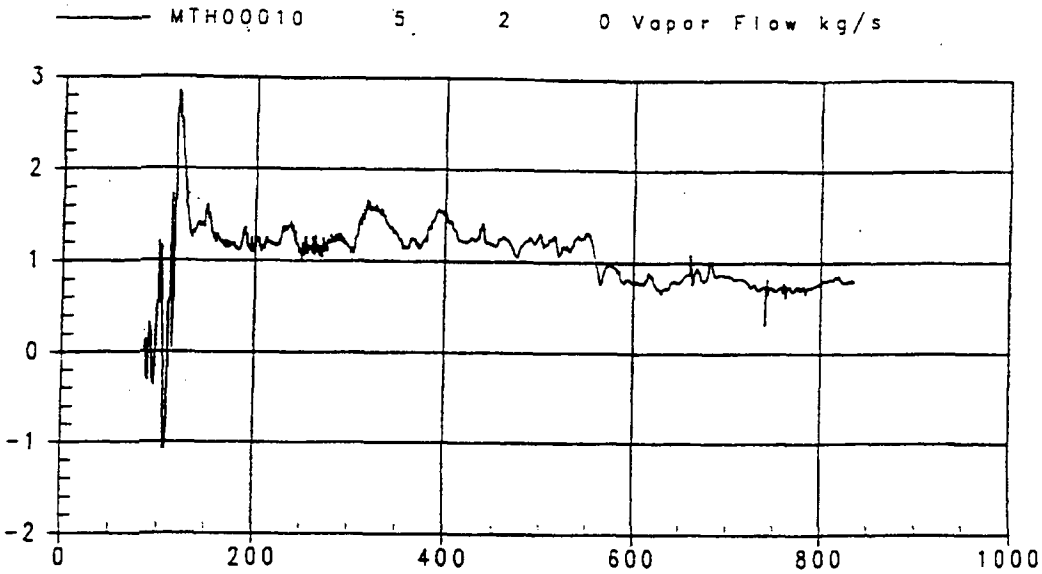


Figure E-170 from JAERI-memo 59-44b

Figure 21.6-18
CCTF Run 58, Loop 4 Hot Leg Water Mass Flow Comparison



W COBRA/TRAC CCTF Run 58, Loop 4 Hot Leg Steam Mass Flow

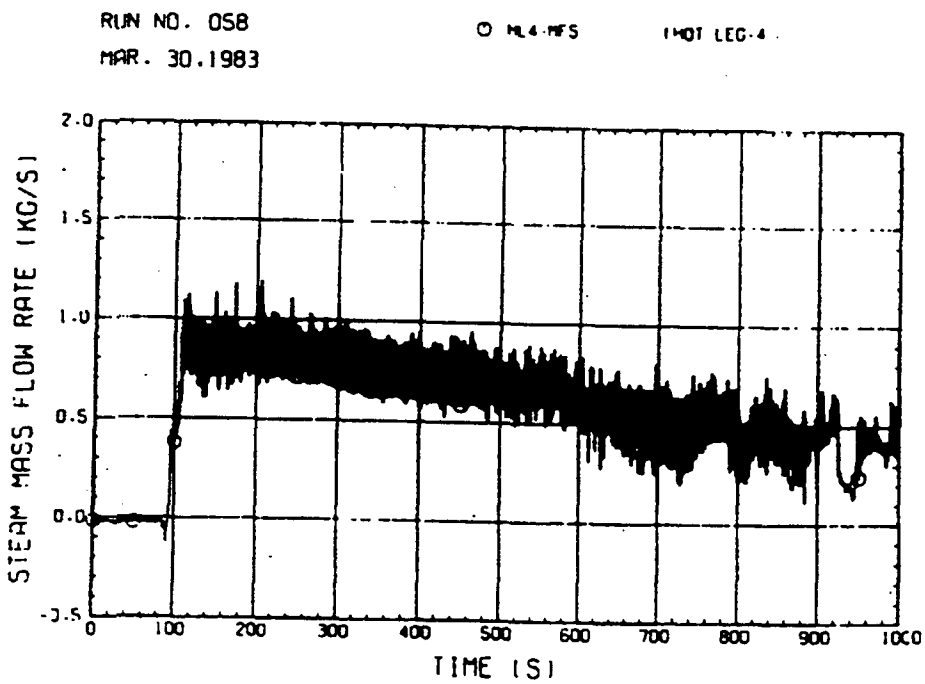


Figure E-169 from JAERI-memo 59-44b

Figure 21.6-19
CCTF Run 58, Loop 4 Hot Leg Steam Mass Flow Comparison

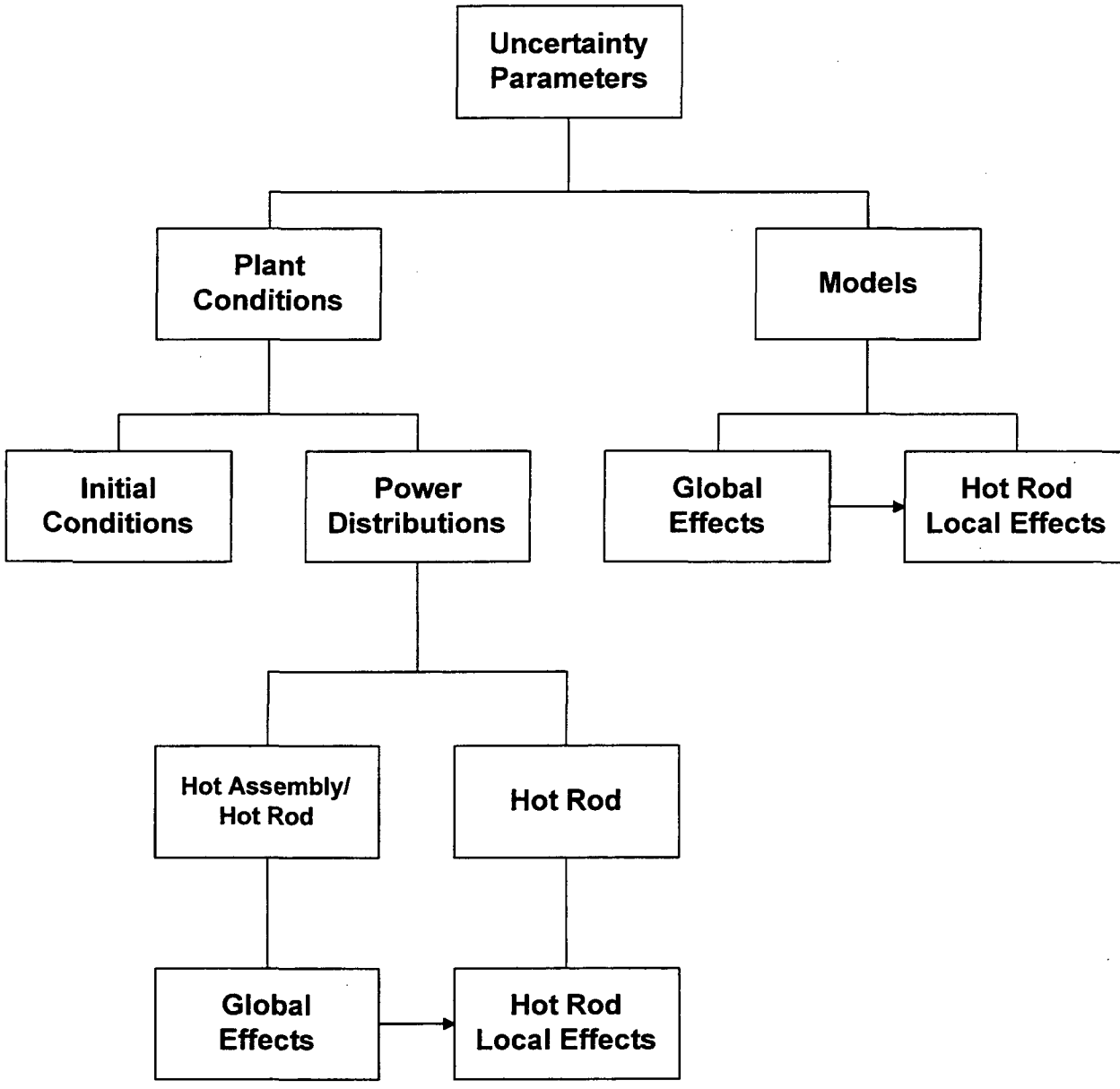


Figure 21.6-20
Breakdown of Westinghouse's Uncertainty Parameters

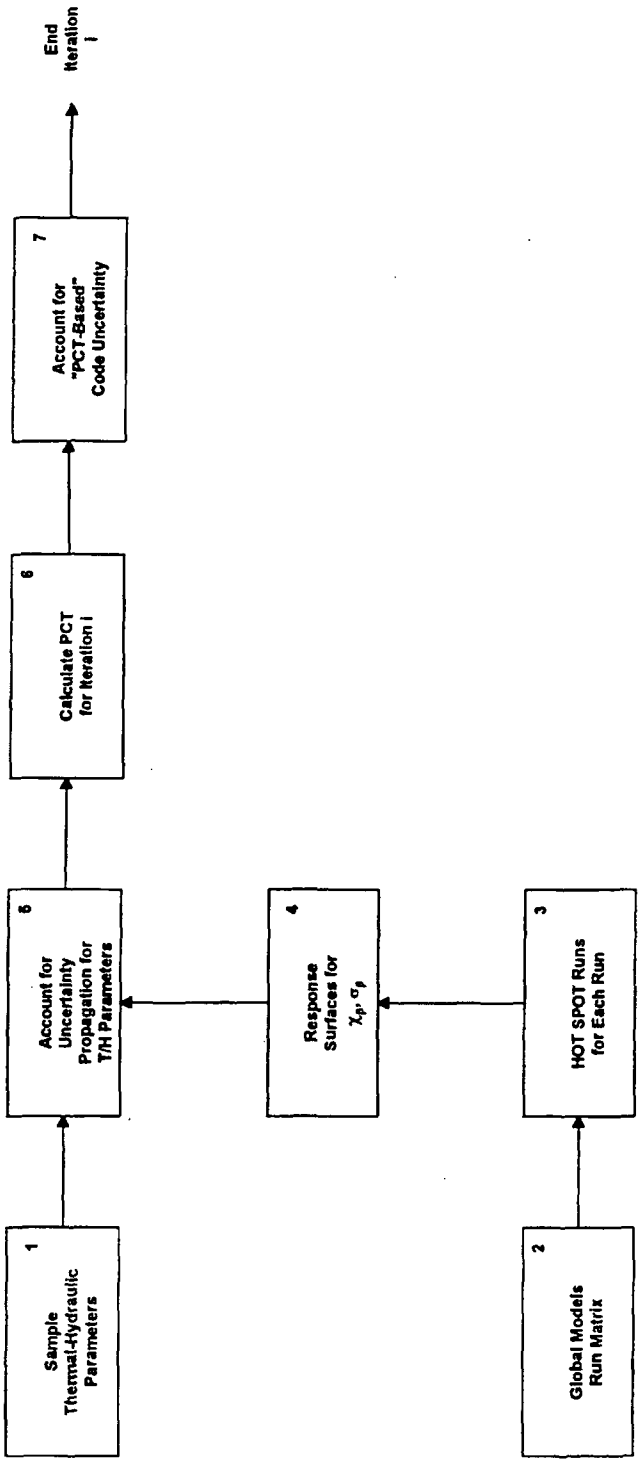
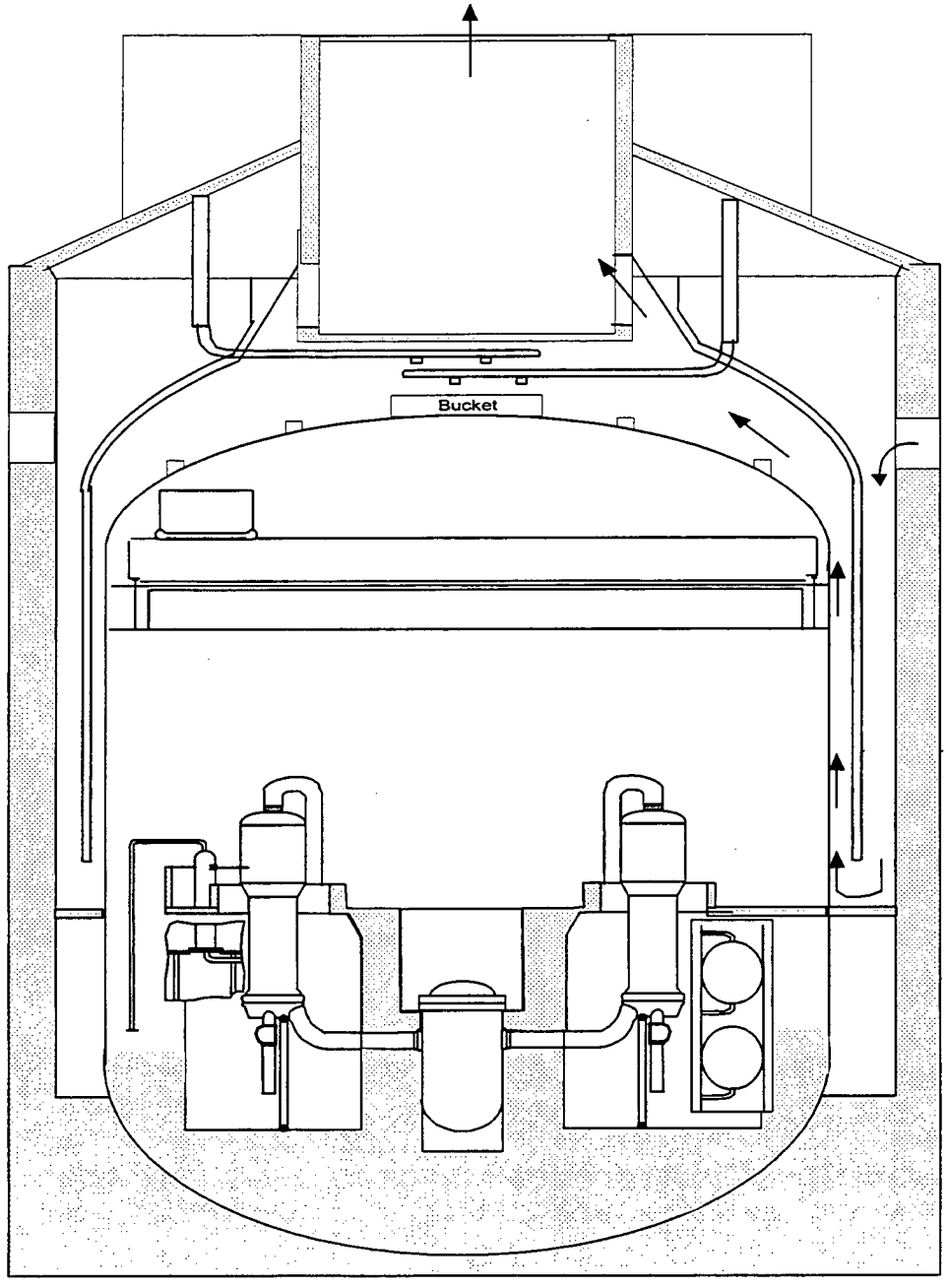


Figure 21.6-21
Flow Chart of Monte Carlo Procedure (AP600)

Figure 21.6-22 AP600 Containment



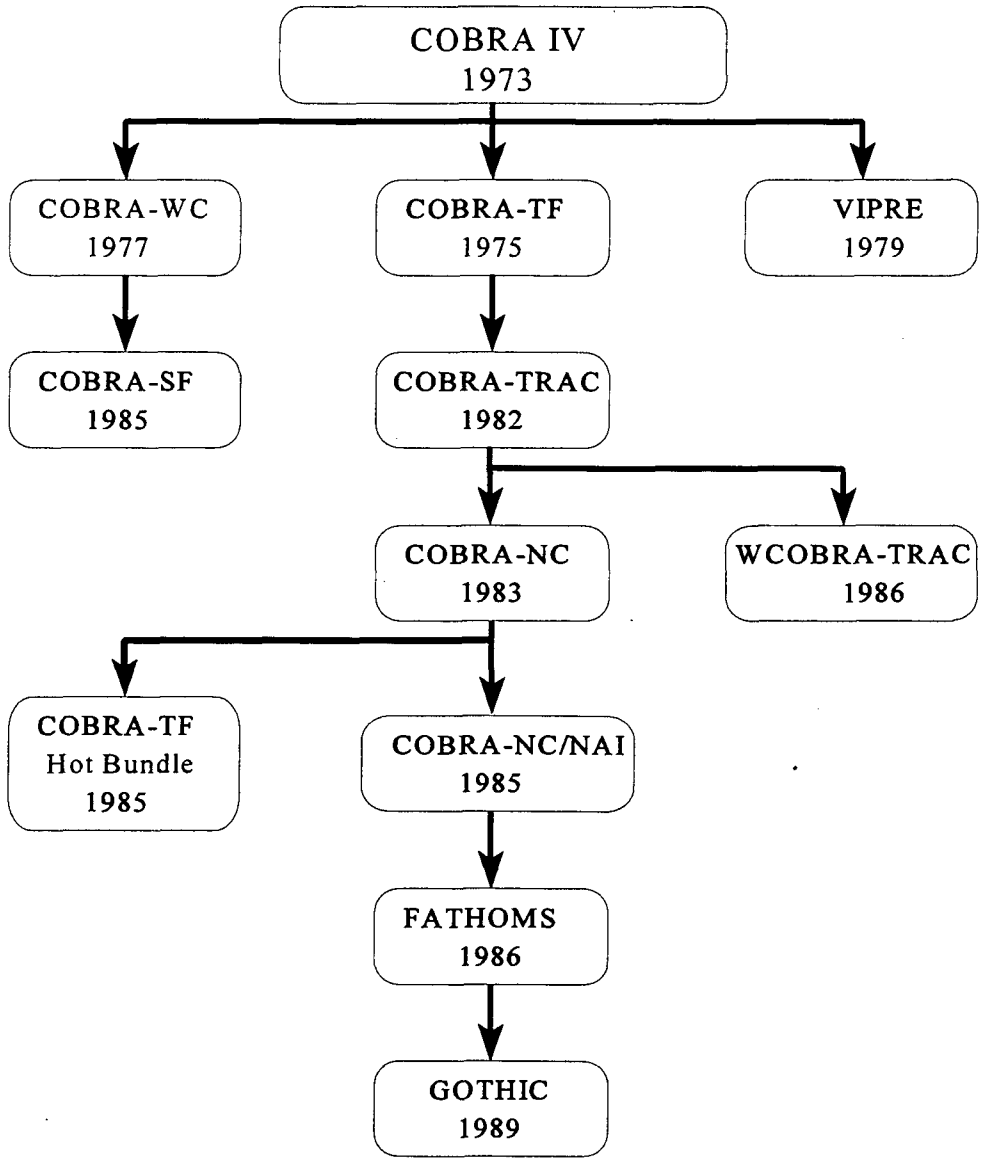


Figure 21.6-23 Historic Development of the GOTHIC Code

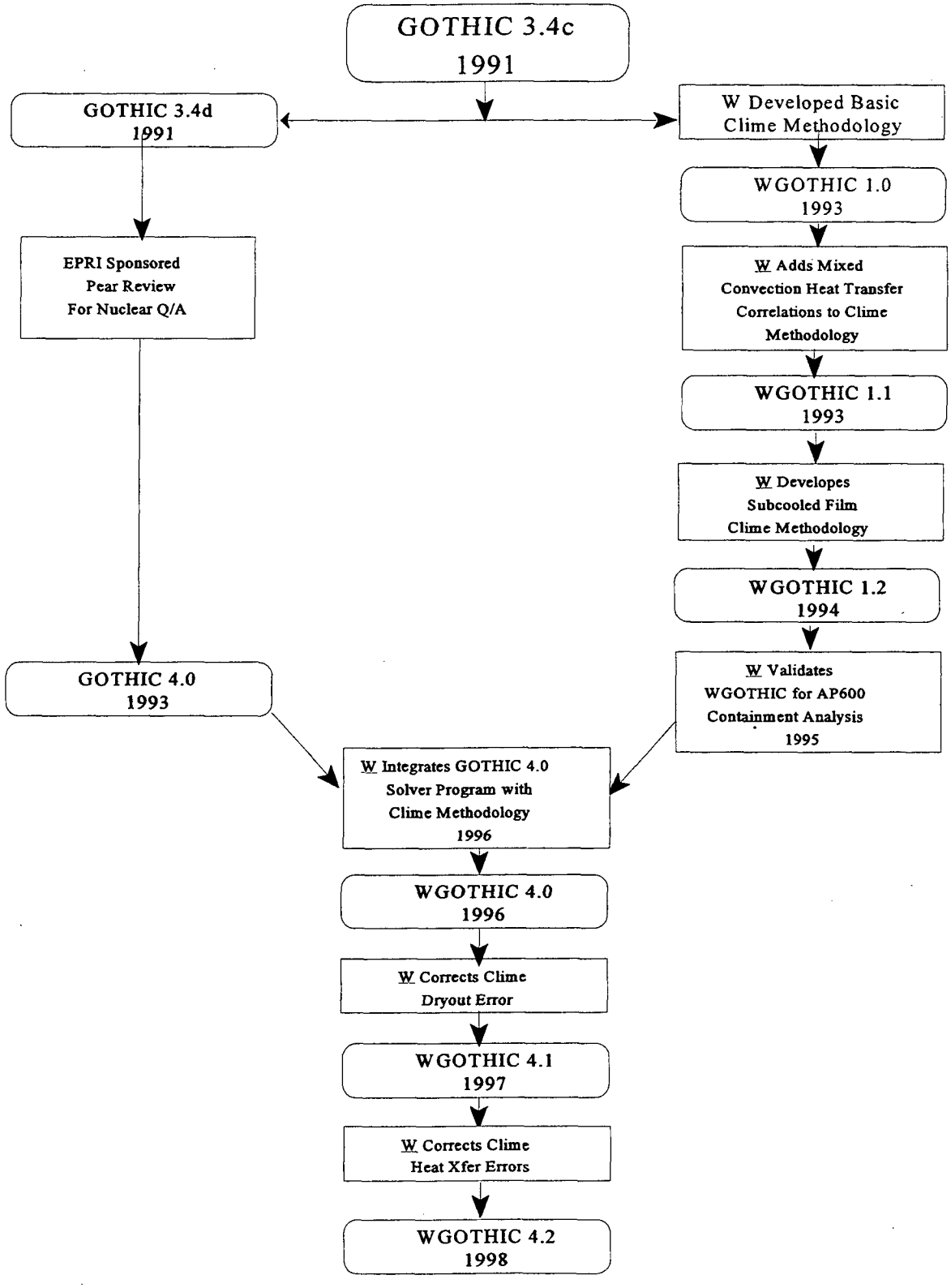


Figure 21.6-24 Development of WGOTHIC

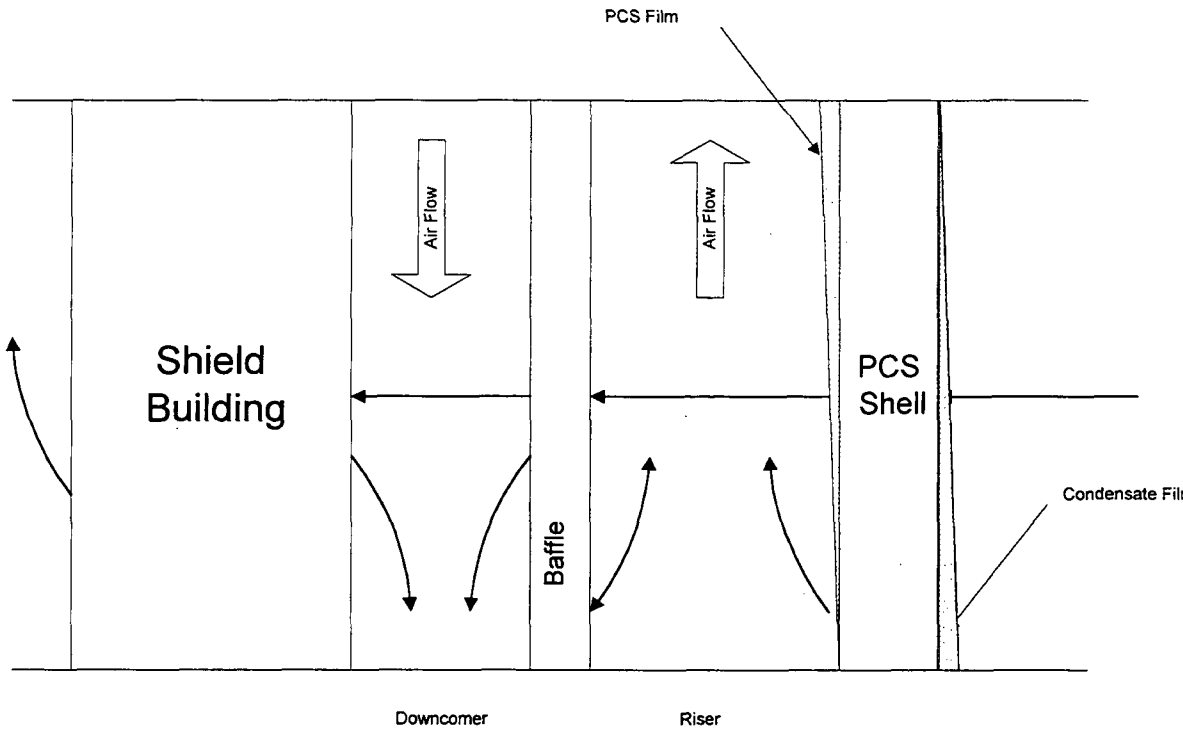


Figure 21.6-25 Simplified Representation of a Clime Heat Structure

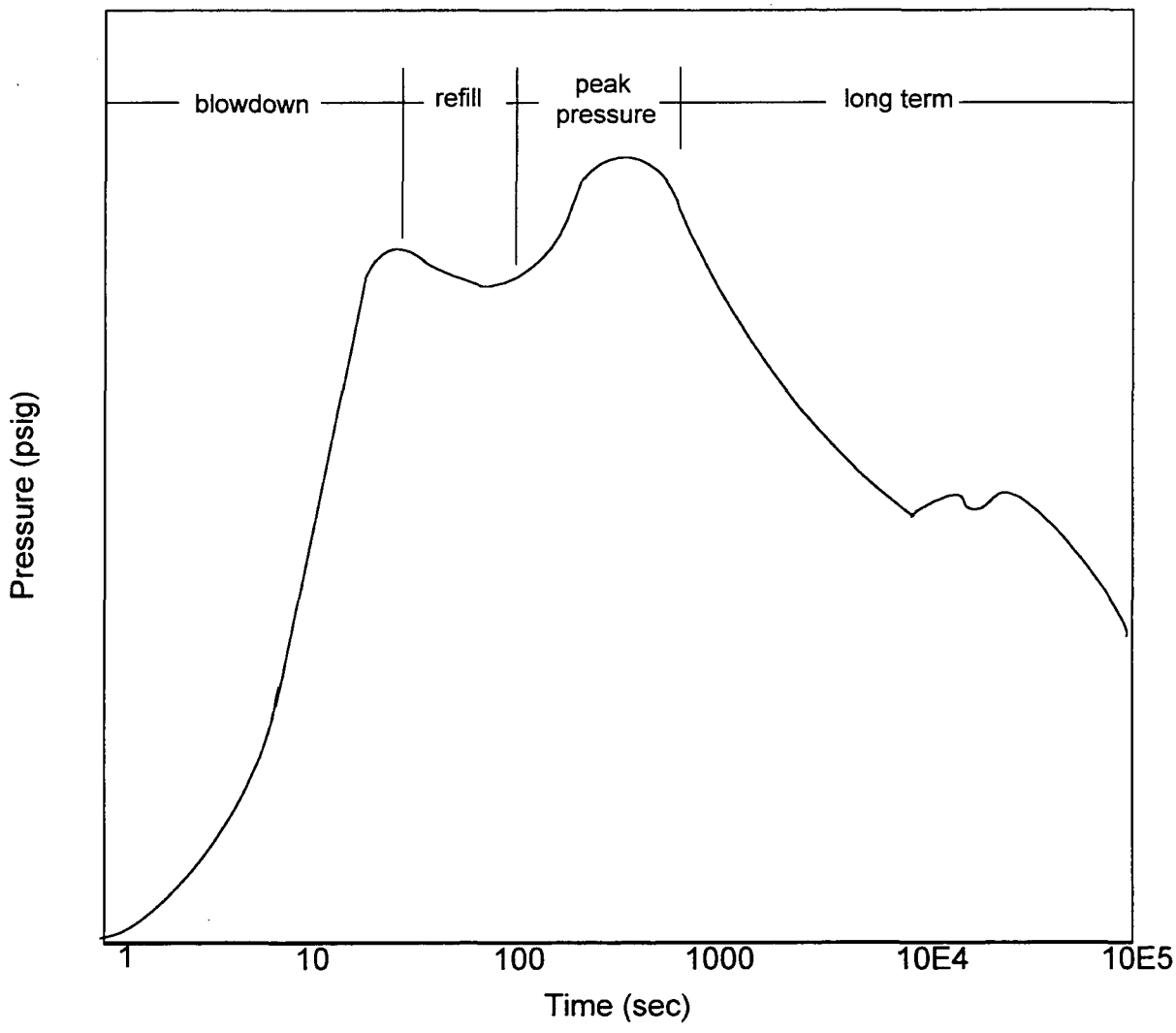


Figure 21.6-26 LOCA Time Phases

22 REGULATORY TREATMENT OF NON-SAFETY SYSTEMS

22.1 Introduction

Unlike the current generation of light-water reactors or the evolutionary advanced light-water reactors (ALWRs), the AP600 plant design uses passive safety systems that rely almost exclusively on natural forces, such as density differences, gravity, and stored energy, to supply safety injection water and provide core and containment cooling. These passive systems do not include pumps. The passive systems include some active valves, but all the safety-related active valves either require only dc safety-related electric power (supplied by means of batteries), are air operated (and fail safe on loss of air), or are check valves. The AP600 design does not include any safety-related sources of ac power for operation of passive system components. All active systems (i.e., systems requiring ac power to operate) are designated as non-safety-related, except for the instrumentation and control systems which use safety-related ac converted from safety-related dc power.

As the AP600 relies on passive safety systems to perform the design-basis safety-related functions of reactor coolant makeup and decay heat removal, different portions of the passive systems also provide certain defense-in-depth backup to primary passive features. For example, while the passive residual heat removal (PRHR) system is the primary safety-related heat removal feature in a non-loss-of-coolant transient, the automatic depressurization system (ADS) together with passive safety injection features provide a safety-related defense-in-depth backup.

The ALWR Utility Requirements Document (URD) for passive plants, promulgated by the Electric Power Research Institute (EPRI), includes standards related to the design and operation of active non-safety-related systems. The URD recommends that the plant designer specifically define the active systems relied upon for defense-in-depth that are necessary to meet passive ALWR plant safety and investment protection goals. Another important aspect of these defense-in-depth systems is long-term, post-accident plant capabilities. Passive systems should be able to perform their safety functions, independent of operator action or offsite support for 72 hours after an initiating event. After 72 hours, non-safety or active systems may be required to replenish the passive systems or to perform core and containment heat removal duties directly. The AP600 includes active systems that provide defense-in-depth (or investment protection) capabilities for reactor coolant system makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. As noted above, most active systems in the AP600 are designated as non-safety-related.

Examples of non-safety-related systems that provide defense-in-depth capabilities for the AP600 design include the chemical and volume control system (CVS), normal residual heat removal system (RNS), and the startup (backup) feedwater system (SFW). For these defense-in-depth systems to operate, the associated systems and structures to support these functions must also be operable, including non-safety-related standby diesel generators, the component cooling

water system, and the service water system. The AP600 also includes other active systems, also designated as non-safety-related, such as the heating, ventilation, and air conditioning (HVAC) system, that remove heat from the instrumentation and control (I&C) cabinet rooms and the main control room and prevent the excessive accumulation of radioactive materials in the control room to limit challenges to the passive safety capabilities for these functions.

In existing plants, and in the evolutionary ALWR designs, many of these active systems are designated as safety-related. However, by virtue of their designation in the AP600 design as non-safety-related, credit is generally not taken for the active systems in the Chapter 15 licensing design-basis accident (DBA) analyses (except in certain cases where operation of a non-safety-related system could make an accident worse). In SECY-90-406, "Quarterly Report on Emerging Technical Concerns," December 17, 1990, the staff listed the role of these active systems in passive plant designs as an emerging technical issue. In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," April 2, 1993, the staff discussed the issue of the regulatory treatment of non-safety systems (RTNSS) and stated that it would propose a process for resolution of this issue in a separate Commission paper. The staff subsequently issued SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, which discusses that process. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (SECY-94-084)," dated May 22, 1995, was essentially a revised version of SECY-94-084 issued to respond to Commission comments on that paper and to request Commission approval of certain revised positions. However, the staff's position on RTNSS as discussed in SECY-94-084 was approved by the Commission (Staff Requirements Memorandum dated June 30, 1994), and was unchanged in SECY-95-132.

In SECY-94-084, the staff cited the uncertainties inherent in the use of the passive safety systems, resulting from limited operational experience and relatively low driving forces inherent in these systems. The uncertainties relate to both system performance characteristics (e.g., the possibility that check valves could stick under low-differential-pressure conditions) and thermal-hydraulic phenomena (e.g., critical flow through ADS valves). The system performance issues were addressed in some cases by design enhancements. For example, check valve performance was improved by using biased-open check valves in the core makeup tank (CMT) discharge lines. In addition, the design for check valves in the in-containment refueling water storage tank (IRWST) injection lines and containment recirculation lines was changed to ensure that the pressure differential across these valves would be small during normal plant operation. Westinghouse also addressed, to some extent, uncertainties associated with passive system reliability, and thermal-hydraulic uncertainties by virtue of the design certification test programs discussed in Chapter 21 of this report. The NRC has also performed confirmatory integral systems testing and analyses over a broad range of conditions to help determine the thermal-hydraulic "boundaries" within which the plant responds in an acceptable manner for both design-basis events and accidents beyond the licensing design basis. These activities have reduced, but not eliminated, the thermal-hydraulic uncertainties associated with passive system performance.

The residual uncertainties associated with passive safety system performance increase the importance of active systems in providing defense-in-depth functions that back-up the passive systems. Recognizing this, the NRC and EPRI developed a process to identify important active systems and to maintain appropriate regulatory oversight of those systems. This process does

not require that the active systems brought under regulatory oversight meet all safety-related criteria, but rather that these controls provide a high level of confidence that active systems having a significant safety role are available when challenged.

The ALWR URD specifies standards concerning design and performance of active systems and equipment that perform non-safety-related, defense-in-depth functions. These standards include radiation shielding to permit access after an accident, redundancy for the more probable single active failures, availability of non-safety-related electric power, and protection against more probable hazards. The standards also address realistic safety margin analysis and testing to demonstrate the systems' capabilities to satisfy their non-safety-related, defense-in-depth functions. However, the ALWR URD does not include specific quantitative standards for the reliability of these systems.

The NRC staff met with representatives of the ALWR Program on several occasions to determine the steps needed to resolve the issue of RTNSS in passive plants, and to define the scope of requirements and acceptance criteria to ensure that systems subject to RTNSS have adequate capability and availability, when needed. In a meeting between the NRC staff and the ALWR Utility Steering Committee on January 22, 1993, the participants arrived at consensus on an overall process for determining the regulatory treatment of non-safety-related systems, and for determining the importance of passive systems and components for meeting NRC safety objectives. The following five key elements made up the process:

(1) The ALWR URD describes the process to be used by the designer for specifying the reliability/availability (R/A) missions of risk-significant structures, systems, and components (SSCs) needed to meet regulatory requirements and to allow comparisons with NRC safety goals. An R/A mission is the set of requirements related to performance, reliability, and availability for an SSC function that adequately ensure its task, as defined by the focused PRA or deterministic analysis, is accomplished. The focused PRA is discussed in further detail in Section 19.1.7 of this report.

(2) The designer applies the process to the design to establish R/A missions for the risk-significant SSCs.

(3) If active systems are determined to be risk-significant, the NRC reviews the R/A missions to determine if they are adequate, and if the operational reliability assurance process (O-RAP) or simple technical specifications and limiting conditions for operation are adequate to give reasonable assurance that the missions can be met during operation. (See Section 22.3.6 of this report for additional discussion of O-RAP and its relationship to the maintenance rule and QA)

(4) If active systems are relied upon to meet the R/A missions, the designer imposes design requirements commensurate with risk significance on those elements involved.

(5) R/A missions are not included in the design certification rule. Instead, NRC includes deterministic requirements on both safety-related and non-safety-related design features in the design certification rule.

To address these key elements, the staff and representatives of the ALWR Program prepared an appropriate process that plant designers could use to address the RTNSS issue. In a letter dated February 23, 1993, the ALWR Program submitted a proposed process for determining the appropriate regulatory treatment for active systems for passive ALWRs. In a meeting on May 20, 1993, the NRC staff and representatives of the ALWR Program reached consensus on a final process for resolving the RTNSS issue. In a letter dated May 26, 1993, EPRI described the steps in this process for determining risk-significant non-safety-related features on the basis of a Level-3 PRA. The process involves constructing a "focused PRA" to determine the importance of various active systems in ensuring that the Commission's safety goal objectives are met. Risk-significant SSCs, their R/A missions, and regulatory oversight can then be determined. The steps of this RTNSS process described by EPRI in its May 26, 1993, submittal are discussed in the following two sections.

22.2 Scope and Criteria for the RTNSS Process

The RTNSS process applies broadly to those non-safety-related SSCs that perform risk significant functions, and therefore, are candidates for regulatory oversight. The plant designer will apply the following five criteria to determine those SSC functions:

(1) SSC functions relied upon to meet deterministic NRC performance requirements such as 10 CFR 50.62 for mitigation of anticipated transients without scram (ATWS) and 10 CFR 50.63 for station blackout (SBO)

(2) SSC functions relied upon to ensure long-term safety (beyond 72 hours) and to address seismic events

(3) SSC functions relied upon under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency (CDF) of less than 1.0E-04 each reactor year and a large early release frequency (LERF) of less than 1.0E-06 each reactor year

(4) SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents. This issue was discussed in detail in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993. For the AP600, this criterion for assessing containment performance is the degree to which the design comports with the Commission's probabilistic containment performance goal of 0.1 conditional containment failure probability (CCFP) when no credit is provided for the performance of the non-safety-related "defense-in-depth" systems for which there will be no regulatory oversight. The CCFP is a containment performance measure that provides perspectives on the degree to which the design has achieved a balance between core damage prevention and core damage mitigation. CCFP was used in a qualitative manner to confirm that the design, combined with the regulatory oversight for identified SSCs, has maintained an acceptable balance between core damage prevention and mitigation, but was not used as a criterion for establishing the availability requirements for non-safety-related "defense-in-depth" systems.

(5) SSC functions relied upon to prevent significant adverse systems interactions

The staff approved the EPRI-proposed scope and criteria in its review of the ALWR URD for passive plants (NUREG-1242, dated August 1994).

22.3 Specific Steps in the RTNSS Process

To implement the approved process described above, the following specific steps were also proposed by EPRI and accepted by the staff.

22.3.1 Comprehensive Baseline Probabilistic Risk Assessment

The designer starts with a comprehensive Level-3 baseline probabilistic risk assessment (PRA). The comprehensive baseline PRA must include all appropriate internal and external events for both power and shutdown operations. Seismic events are evaluated using a margins approach. Adequate treatment of R/A uncertainties, long-term safety operation, and containment performance should be included. Containment performance should be addressed with consideration of sensitivities and uncertainties in accident progression and inclusion of severe accident phenomena, including explicit treatment of containment bypass. Mean values must be used to determine the availability of passive systems and the frequencies of core damage and large releases. Appropriate uncertainty and sensitivity analyses should be used to estimate the magnitude of potential variations in these parameters and to identify significant contributors to these variations. Results of an adverse systems interaction study should also be considered in the PRA. The AP600 baseline PRA is discussed in Chapter 19 of this report.

22.3.2 Search for Adverse Systems Interactions

The designer must systematically evaluate adverse systems interactions between the active and passive systems. The results of this analysis are to be used for design improvements to minimize adverse systems interactions and should be considered in making PRA models, as noted above.

Westinghouse evaluated adverse systems interactions in the AP600, and submitted WCAP-14477, "The AP600 Adverse System Interaction Evaluation Report" in March 1996. After staff review and comments, WCAP-14477, Revision 1, was submitted in May 1997 and Revision 2 was submitted in November 1997. The staff's review of WCAP-14477 is detailed in Section 22.5.2 of this report.

22.3.3 Focused PRA

The focused PRA includes the passive systems and only those active systems necessary to meet the safety goal guidelines approved by the Commission in SECY-94-084 (see criterion (3) in Section 22.2 of this report). The designer should consider the following in constructing focused PRAs to determine the R/A missions of non-safety-related SSCs which are risk-significant.

First, the scope of initiating events and their frequencies are maintained in the focused PRA as in the baseline PRA. As a result, non-safety-related SSCs used to prevent the occurrence of initiating events will be subject to regulatory oversight applied commensurate with their R/A missions for prevention, as discussed below.

Second, following an initiating event, the comprehensive Level-3 focused PRA event-tree logic will not include the effect of non-safety-related SSCs. As a minimum, these event trees will not include the defense-in-depth functions and their support, such as ac power, to determine if the passive safety systems, when challenged, can provide sufficient capability without non-safety-related backup to meet the NRC safety goal guidelines for a CDF of 1E-04 each year and an LERF of 1E-06 each reactor year. The designer should evaluate the containment performance, including bypass, during a severe accident. Non-safety-related SSCs which remain in the focused PRA model are subject to regulatory oversight on the basis of their risk significance.

Westinghouse has performed a focused PRA for the AP600. The staff's evaluation of the focused PRA is discussed in Section 19.1.7 of this report.

Although not discussed explicitly in these steps, an important aspect of the focused PRA is the evaluation of uncertainties, particularly those uncertainties inherent in the use of passive safety systems. Reliability/availability uncertainties are discussed in the focused PRA evaluation in Section 19.1.7 or this report. However, because of limited data and experience with the passive systems, the staff was concerned that thermal-hydraulic uncertainties could also impact the PRA results. Specifically, thermal-hydraulic uncertainties can have a direct impact on the determination of success criteria for accident sequences in the focused PRA. As noted above, this was one of the primary reasons for the development of the RTNSS process. To address the issue of thermal-hydraulic uncertainties, Westinghouse submitted WCAP-14800, "AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability." The issue of thermal-hydraulic uncertainty is treated separately in this chapter, and is discussed in Sections 22.5.4.1 and 22.5.4.4 of this report.

22.3.4 Selection of Important Non-Safety-Related Systems

The designer should determine any combinations of non-safety-related SSCs that are necessary to meet NRC regulations, safety goal guidelines, and the containment performance goal objectives. The designer should determine these combinations for criteria (1) and (5) in Section 22.2, in which NRC regulations are the bases for consideration, and for criteria (3) and (4) in Section 22.2, in which PRA methods are the bases for consideration. To address the long-term safety issue in criterion (2) of Section 22.2, the designer should use PRA insights, sensitivity studies, and deterministic methods to establish the ability of the design to maintain core cooling and containment integrity beyond 72 hours. Non-safety-related SSCs required to meet deterministic regulatory requirements (criterion (1)), to resolve the long-term safety and seismic issues (criterion (2)), and to prevent significant adverse systems interactions (criterion (5)) are subject to regulatory oversight as discussed below in Section 22.3.6.

EPRI further proposed that designers take the following steps in using the focused PRA to determine the non-safety-related SSCs important to risk:

- Determine those non-safety-related SSCs needed to maintain the initiating event frequencies at the comprehensive baseline PRA levels.

- Add the necessary success paths with non-safety-related systems and functions in the focused PRA to meet the safety goal guidelines, containment performance goal objectives, and NRC regulations. Choose the systems by considering the factors for

optimizing the design effect and benefit of particular systems. Perform PRA importance studies to assist in determining the importance of these SSCs. Recognize that the staff could require regulatory oversight for all non-safety-related SSCs in the focused PRA model needed to meet NRC requirements, the safety goal guidelines, and containment performance goals.

22.3.5 Non-Safety-Related System Reliability/Availability Missions

The designer should determine and document from the focused PRA the functional R/A missions of active systems needed to meet the safety goal guidelines, containment performance goals, and NRC performance requirements as described in Section 22.3.4. The steps described in Sections 22.3.4, 22.3.5, and 22.3.6 should be repeated to ensure that the most appropriate active systems and their R/A missions are selected.

As part of this step, the designer should establish graded safety classifications and graded requirements for I&C systems on the basis of the importance to safety of their functional R/A missions.

22.3.6 Regulatory Oversight Evaluation

Upon completing the steps detailed in the previous five sections, the designer should conduct the following activities:

- review the SSAR and the PRA, and audit plant performance calculations to determine that the design of these risk-significant, non-safety-related SSCs satisfies the performance capabilities and R/A missions.

- review the SSAR to determine that it includes the proper design information for the reliability assurance program, including the design information for implementing the maintenance rule and the operational reliability assurance process (O-RAP). (The staff notes that the development of an O-RAP is a COL action item (see Section 17.4 of this report), and that the implementation of the O-RAP is tied to existing regulatory requirements such as the establishment of reliability/availability targets for the maintenance rule and quality assurance requirements.)

- review the SSAR to determine that it includes proper short-term availability control mechanisms, if required for safety and determined by risk significance, such as simple technical specifications.

After the designer has completed these or related activities, the staff will determine the means of appropriate regulatory oversight to implement RTNSS controls.

22.3.7 NRC/Vendor Interaction

Early in the reviews (and on a continuing basis), the staff and designer will discuss the appropriateness of the focused PRA models and reliability values, R/A missions, and level of regulatory oversight for various active systems.

As noted above, Westinghouse submitted several reports documenting evaluations performed as part of the RTNSS resolution process. In addition to those addressing specific issues (e.g., baseline and focused PRAs, adverse systems interactions, thermal-hydraulic uncertainties), Westinghouse submitted WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," in September 1993. This was an initial attempt to apply the previously described process to the AP600 design and served as a starting point for the staff's interactions with Westinghouse on the issue of RTNSS. The staff's review of WCAP-13856 is discussed in further detail in Section 22.5.1 of this report.

22.4 Other Issues Related to RTNSS Resolution

In SECY-94-084, several other issues were discussed, related overall to passive plant performance or performance of specific passive safety systems. Resolution of these issues was tied by the staff to an acceptable resolution of the RTNSS issue. On the basis of the defense-in-depth equipment availability administrative controls discussed in Section 22.5.4.3 of this report, the staff was able to reach acceptable conclusions regarding the AP600 design related to (1) safe shutdown requirements as discussed in Section 6.3.1.4; (2) station blackout as discussed in Section 8.6.2.1; and (3) electrical distribution as discussed in Section 8.2.4 of this report.

22.5 NRC Review of Westinghouse's Approach to Evaluation of Systems for Inclusion in RTNSS

Westinghouse pursued a multifaceted approach to the determination of which non-safety-related systems in the AP600 should be subject to regulatory treatment, and under what conditions that treatment should apply. The approach has followed broadly the scope, criteria, and specific steps proposed by EPRI and accepted by the staff in SECY-94-084 (and approved by the Commission in SRM dated June 30, 1994), which are discussed in Sections 22.2 and 22.3 However, for the staff, the critical elements of the RTNSS process have tended to center upon the issues of the focused PRA results, the evaluation of thermal-hydraulic uncertainty in the performance of passive systems, and the ability of Westinghouse to identify success or failure of PRA sequences (i.e., core damage or no core damage) with beyond-design-basis, multiple component failures.

22.5.1 Initial Evaluation

Westinghouse submitted to the NRC WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," in September 1993. The report represented a summary of Westinghouse's evaluation of the need for regulatory oversight of non-safety-related systems. The implementation described in WCAP-13856 followed the proposed EPRI process, which had been submitted to the NRC but had not yet been accepted by the staff at the time WCAP-13856 was issued. The criteria used by Westinghouse to determine which systems required regulatory oversight were based on probabilistic assessments of passive system performance (focused PRA) and on an initiating event frequency study. In addition deterministic assessments of the AP600 design response to events such as ATWS, SBO, long-term cooling (post-72 hour) were reviewed. A summary of AP600 response to containment performance challenges, adverse systems interactions, and seismic considerations was also included in WCAP-13856.

The results of Westinghouse's probabilistic evaluations indicated that, for most at-power events, non-safety-related SSCs played minimal roles in initiating event frequencies, CDF, and LERF. Exceptions to these conclusions were found in the evaluation of non-LOCA transients, specifically turbine trip/spurious reactor trip and loss of main feedwater events. Despite determining that non-safety-related SSCs could be important in the AP600's response, Westinghouse concluded that no additional regulatory oversight was required beyond that already in place by virtue of SSAR design requirements. The rationale for this position was Westinghouse's contention that AP600 "design improvements" not fully credited in the PRA would result in increased reliability of risk-significant, non-safety-related SSCs, and that the PRA results for CDF and LERF represented very conservative bounding values.

For events at shutdown, Westinghouse's probabilistic evaluations showed that non-safety-related SSCs also could be important in several scenarios, including loss of offsite power and loss of decay heat removal, especially during reduced-inventory operations. Consequently, Westinghouse proposed "short-term availability recommendations" for the following non-safety-related SSCs:

- Offsite power system
- Main ac power system
- Onsite standby (diesel) power system
- Normal Residual Heat Removal system (RNS)
- Component Cooling Water system
- Service Water system

It is important to note that the availability controls proposed by Westinghouse applied only during reduced reactor coolant system inventory operations during cold shutdown and refueling (Modes 5 and 6).

Westinghouse's deterministic evaluations also resulted in very limited short-term availability recommendations for a few selected non-safety-related SSCs. No systems were identified for RTNSS controls in the assessments of SBO, containment performance, seismic considerations, and adverse systems interactions. Westinghouse's proposed approach with regard to post-72-hour actions was to provide safety-related connections for use with transportable equipment and supplies to support and maintain key safety functions. The transportable equipment was not to be located on site, but rather was to be available at pre-identified, off-site locations for transportation to the plant site as required. Since this equipment was not located on the plant site, Westinghouse concluded that no RTNSS controls were needed for post-72-hour support. The only systems identified for RTNSS treatment in the deterministic evaluations were those needed to meet ATWS regulatory requirements (10 CFR 50.62). For this purpose, short-term availability controls were proposed for the following systems:

- non-Class-1E dc power
- uninterruptible power supply (UPS)
- diverse actuation system (DAS), turbine trip and passive residual heat removal (PRHR) actuation functions only

(The non-Class-1E dc and UPS systems provide power to the DAS.)

The electrical systems and specified DAS functions were specified for RTNSS controls only during power operation (Modes 1 and 2).

The staff reviewed WCAP-13856 and concluded that it was inadequate to support the determination of appropriate non-safety-related SSCs for RTNSS controls. Numerous open items related to RTNSS were included in the AP600 DSER and were reflected in open items resulting from meetings and RAIs. In addition, the staff issued two Commission papers discussing technical and policy issues related to the AP600. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," was issued on June 30, 1995, and specifically identified passive system thermal-hydraulic performance reliability (Item V) and RTNSS (Item VI) as "key issues." The discussion on RTNSS in SECY-95-172 included the issues of post-72-hour actions and adverse systems interactions. SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," was issued on June 12, 1996, and provided an update to the discussion in SECY-95-172. SECY-96-128 incorporated the issue of passive system thermal-hydraulic performance reliability into the discussion of RTNSS. In addition, the staff separated the issue of post-72-hour actions from the overall RTNSS process and requested Commission approval of a proposed position regarding this issue.

To address the staff's concerns, as discussed in meetings, RAIs, SECY-95-172, and SECY-96-128, Westinghouse has provided significant additional information as follows:

- WCAP-14477, Revision 2, "The AP600 Adverse Systems Interaction Report"
- Letter reports and presentations regarding post-72-hour actions and equipment
- AP600 Focused PRA
- WCAP-14800, "AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability"
- WCAP-14869, "MAAP4/NOTRUMP Benchmarking to Support the Use of MAAP4 for AP600 PRA Success Criteria Analyses"
- WCAP-14837, Revision 2, "AP600 Shutdown Evaluation Report"

Discussion of Westinghouse's submittals and the staff's review and conclusions are contained in subsequent sections.

22.5.2 Evaluation of Adverse Systems Interactions

In response to the staff's comments on the discussion of adverse systems interactions in WCAP-13856, Westinghouse agreed to submit a comprehensive evaluation of these interactions for the AP600. WCAP-14477, Revision 0 was submitted in March 1996, and was reviewed by the staff. Numerous comments and questions were provided to Westinghouse by an NRC letter dated October 31, 1996. Westinghouse provided responses to the staff's questions and comments, and subsequently issued WCAP-14477, Revision 1 in May 1997, incorporating the information responding to those comments and questions. WCAP-14477, Revision 2, was

subsequently issued in November 1997, to be consistent with the AP600 Emergency Response Guidelines (ERGs).

WCAP-14477, Revision 2, represents a systematic compilation and assessment of potential systems interactions in the AP600. Several different types of interactions are considered, including:

- Interactions among the passive safety systems
- Interactions between passive safety systems and active non-safety-related systems
- Interactions resulting from operator errors of commission
- Spatial interactions (i.e., interactions that could occur as a result of equipment location in the plant)

After evaluating the potential adverse systems interactions that could occur in the AP600, Westinghouse concluded that there were no non-safety-related SSCs that required RTNSS treatment as a result of this specific issue.

The staff has reviewed WCAP-14477, Revision 2, to determine if Westinghouse acceptably addressed the staff's comments and questions. The staff also used the information provided in WCAP-14477, Revision 2, as part of the framework for the evaluation of Westinghouse's ERGs, in assessing both operator actions to preclude potential adverse systems interactions and interactions that could arise as a result of human commission errors. As a result of its review, the staff agrees with the Westinghouse conclusion that there are no adverse systems interactions that require the implementation of RTNSS controls. Accordingly, the staff finds Westinghouse's treatment of adverse systems interactions as described in WCAP-14477, Revision 2, to be acceptable, and does not require RTNSS treatment of any AP600 non-safety-related SSCs to address adverse systems interaction issues. The staff's evaluation of adverse systems interactions has also been reflected in the review of Unresolved Safety Issue (USI) A-17 in Chapter 20 of this report.

22.5.3 Post-72-Hour Actions and Equipment

The passive safety-related systems in the AP600 are designed to function, under design-basis conditions, for at least 72 hours without the need for action to supplement or extend their capabilities. In the URD for passive ALWRs, EPRI proposed that post-72-hour actions could be supported by the use of readily-available offsite equipment and supplies, which would be brought onto the plant site when required. The staff conditionally accepted EPRI's position in NUREG-1242, provided that a plant designer could demonstrate that these offsite resources would be able to meet R/A missions for post-72-hour support.

In WCAP-13856, Westinghouse indicated its intention to follow EPRI's proposed resolution of this issue. Accordingly, the AP600 design includes safety-related connections for equipment and supplies used to supplement or support the passive safety systems after the initial 72-hour post-accident period. However, as the staff reviewed WCAP-13856 and supporting documentation, concerns developed regarding the capability to ensure the availability of offsite

resources, including "transportable" equipment and consumables (e.g., water and fuel) to fulfill post-72-hour support functions. The staff raised several questions regarding post-72-hour actions, which were provided to Westinghouse in a letter dated May 25, 1995 (RAIs 440.400 to 440.406). In addition, the staff discussed its concerns in this area in SECY-95-172.

The staff had several discussions with Westinghouse during 1995 and early 1996 regarding the AP600 approach to post-72-hour support. In addition, Westinghouse responded to the staff's RAIs in a letter dated April 3, 1996. However, the staff concluded that Westinghouse's RAI responses did not resolve the post-72-hour issue. In SECY-96-128, the staff requested Commission approval of a position on post-72-hour actions, i.e., that post-72-hour actions related to all design-basis events be accomplished with onsite equipment and supplies. The staff recommended that replenishment of consumables could be credited after 7 days. The staff further stated that the equipment needed for post-72-hour support need not be in "automatic standby mode," but must be readily available and protected from natural phenomena, including seismic events, as required by 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2. The Commission subsequently approved the staff's position in a memorandum dated January 15, 1997. The staff further elaborated its position in a memorandum to the Commission dated June 23, 1997, in which it explicitly stated that a COL applicant would be required to have appropriate availability controls, consistent with RTNSS requirements, for non-safety-related SSCs for post-72-hour support.

The staff met with Westinghouse on February 4, 1997, at which time Westinghouse presented proposed design changes for the AP600 to respond to the staff's position regarding post-72-hour actions. The staff provided its comments on the proposed design changes by letter dated February 18, 1997. Westinghouse responded by letter to these comments on March 14, 1997. This information was supplemented by Westinghouse letters dated April 30, 1997, and May 23, 1997. The staff provided additional comments regarding its concerns about design changes and treatment of post-72-hour equipment in letters dated June 5, 1997, and July 7, 1997. Westinghouse responded to the staff's comments in a letter dated October 10, 1997, incorporating design changes related to post-72-hour actions and responses to staff concerns regarding issues such as protection from natural phenomena, long-term main control room habitability, and spent fuel pool cooling. Westinghouse has also developed RTNSS controls for the post-72-hour, non-safety-related systems that are consistent with those developed for other risk-significant, non-safety-related SSCs in accordance with the staff's position as stated in its letter to Westinghouse dated July 7, 1997.

The staff finds that Westinghouse's actions, with regard to post-72-hour actions, are acceptable and comply with the staff's approved positions as stated in SECY-94-084 and SECY-96-128. All required equipment is onsite and protected consistent with GDC 2, with consumable supplies sufficient to last at least 7 days.

22.5.4 Focused PRA and Passive System Thermal-Hydraulic Performance Reliability

The majority of the staff's effort in the RTNSS process has been concentrated on the evaluation of Westinghouse's focused PRA and on the associated issue of thermal-hydraulic uncertainty evaluation for passive system reliability. These issues were combined in SECY-96-128, reflecting the dependence of the focused PRA CDF and LERF values on the assumptions made with respect to uncertainties in thermal-hydraulic parameters. This section describes the staff's initial approach to the resolution of the issue of thermal-hydraulic uncertainty, the evaluation of

the focused PRA, and the proposed resolution of these issues by the addition of several non-safety-related systems to those previously identified for RTNSS treatment.

22.5.4.1 Evaluation of Thermal-Hydraulic Uncertainty and Passive System Performance Reliability

In WCAP-13856, Revision 0, Westinghouse concluded that its focused PRA for at-power events did not result in the identification of any systems for RTNSS treatment. According to the RTNSS process discussed in SECY-94-084, this is equivalent to determining that the Commission's safety goal objectives of a CDF no more than 1E-4 per reactor year and a LERF of no more than 1E-6 per reactor year were met without credit in the focused PRA for any non-safety-related systems. Determination of a value for CDF requires the establishment of criteria to evaluate whether a given accident sequence in the PRA results in "success" (i.e., no core damage) or in core damage. Similar criteria are needed to determine if a core damage sequence results in a large release of radioactivity outside the containment to allow calculation of the LERF. For CDF, Westinghouse based its determination on a single criterion, peak core temperature as determined using the MAAP4 computer code. Consequently, the evaluation of thermal-hydraulic uncertainty was directed primarily toward determining how passive system performance could be affected by uncertainties in thermal-hydraulic parameters that had a significant influence on the calculation of core temperatures in MAAP4, and thus on the determination of accident sequences as resulting in "success" or core damage.

The staff emphasizes that the evaluation of thermal-hydraulic uncertainty and passive system performance reliability is relevant only to the issue of events beyond the design-basis conditions used to determine the acceptability of emergency core cooling system (ECCS) performance in Chapters 6 and 15 of the AP600 SSAR. For design-basis accidents and transients, the plant must meet deterministic requirements for the performance of safety-related systems, such as the ECCS acceptance criteria in 10 CFR 50.46(b). Uncertainties are either accommodated by using conservative, bounding models and assumptions, such as those required by 10 CFR Part 50, Appendix K or are explicitly evaluated and applied to best-estimate calculations of ECCS performance.

The staff considered requiring an explicit statistical evaluation of thermal-hydraulic uncertainties associated with passive system performance in focused PRA event sequences. However, the staff determined that such an evaluation would be impractical, in view of the number of parameters and the very limited database available to permit assessment of such uncertainties under beyond-design-basis operating conditions. Instead, the staff determined that a "margins"-based approach to thermal-hydraulic uncertainty would be acceptable, as discussed in SECY-95-172 and SECY-96-128. As outlined in SECY-96-128, Westinghouse proposed to use the NOTRUMP code to perform sensitivity analyses on risk-significant accident sequences selected from the focused PRA, employing conservative, bounding inputs and assumptions, to demonstrate large margins to core damage. (The staff's evaluation of NOTRUMP for design-basis analyses of small-break LOCAs is contained in Section 21.6.2 of this report.) The NOTRUMP results would then be used to "benchmark" the MAAP4 analyses of the same sequences to demonstrate that MAAP4 results were phenomenologically similar to NOTRUMP results and to establish whether a MAAP4-predicted core temperature lower than the peak temperature criterion selected by Westinghouse was adequate to preclude core damage to a high degree of confidence.

Westinghouse documented the results of the NOTRUMP sensitivity analyses and comparisons to MAAP4 results in WCAP-14869. The parameters judged to be important in establishing bounding thermal-hydraulic inputs and models are as follows:

- decay heat
- initial conditions
- line resistances
- containment pressure
- safety system actuation logic and time delays
- break discharge coefficients

Westinghouse performed 19 analyses comparing NOTRUMP and MAAP4. Seven "base" cases covered a range of break sizes from 12.7 mm (0.5 in.) to 222 mm (8.75 in.), with both automatic and manual ADS actuation. The 12 "sensitivity" runs included variations in break location, numbers of functioning CMTs and accumulators, number of operating ADS valves, and IRWST injection conditions. The effect of partial depressurization to allow the use of the non-safety-related normal residual heat removal system (RNS) to pump water from the IRWST to the reactor vessel was also evaluated. On the basis of the SBLOCA PIRT developed for design-basis accident assessment (see Chapter 21), Westinghouse evaluated the following "high-importance" code models and thermal-hydraulic phenomena:

- break model
- interfacial condensation in the CMT
- vessel mixture level
- downcomer
- ADS-4
- ADS-1, 2, and 3
- IRWST
- accumulator

Westinghouse's overall assessment of MAAP4 was that, in most cases, MAAP4 predictions of key phenomena compared well with NOTRUMP's results, and was therefore an appropriate screening tool to evaluate PRA accident sequences to determine whether success criteria were met. Some limitations and necessity for user guidance were found, as a result of MAAP4's simplified system and thermal-hydraulic models, primarily in the calculation of phenomena associated with the ADS. In some cases, MAAP4 was found to predict non-conservatively high rates of depressurization through the ADS-1, 2, and 3 valves. However, Westinghouse downgraded the importance of this limitation with respect to assessment of PRA sequences, since ultimate "success" was more dependent on the number of ADS-4 valves that were operational. Westinghouse also determined that, prior to the benchmarking exercise, ADS-4 line resistances had not been large enough in the MAAP4 AP600 model and that non-choked flow rates were overpredicted. When this non-conservative assumption was corrected, Westinghouse found that three out of four ADS-4 valves were required to achieve the desired system depressurization response at low reactor coolant system pressures. This had an impact on PRA success criteria, since previous analyses were based on two out of four ADS-4 valves being necessary for "success." Westinghouse evaluated the impact of this change on the CDF and LERF calculated using the focused PRA, and determined that these frequencies changed slightly but were still within acceptable values. A similar evaluation was performed on the baseline PRA (as discussed in Section 50.5.6 of the Westinghouse PRA).

In conjunction with the NOTRUMP calculations, Westinghouse selected four cases in which NOTRUMP predicted core uncovery and used the LOCTA code to calculate a peak clad temperature (PCT) on the basis of the NOTRUMP thermal-hydraulic results. While the nominal criterion for "success" in a PRA sequence was defined by Westinghouse as a maximum (MAAP4) temperature of 1478 K (2200 °F), for the purposes of the MAAP4/NOTRUMP benchmarking exercise, MAAP4 "success" was based on a prediction of a duration of core uncovery of no more than 1000 seconds. The maximum duration of core uncovery predicted by NOTRUMP was 1600 seconds, with a LOCTA PCT prediction of about 1114 K (1545 °F). (This case was poorly predicted by MAAP4 because of the ADS-1/2/3 modeling limitations discussed above; MAAP4 predicted no uncovery.) A case in which both NOTRUMP and MAAP4 predicted core uncovery for approximately 1520 seconds was not evaluated with LOCTA but was estimated by Westinghouse to reach about the same PCT as the 1600-second uncovery case, on the basis of very similar predictions between the two cases of time at the inception of core uncovery and maximum core exit vapor temperature. On the basis of these comparisons, Westinghouse concluded that a MAAP4 prediction of core uncovery of less than 1000-second duration was a reasonably accurate indicator of "success."

The staff reviewed WCAP-14869 and evaluated Westinghouse's conclusions regarding the adequacy of MAAP4 for screening PRA sequences. The staff found that, in most cases, MAAP4 and NOTRUMP predicted similar trends for system behavior in the base cases and sensitivity analyses. While there was reasonable agreement between the two codes, in many cases, with respect to key thermal-hydraulic parameters, the timing of events was, in some cases, significantly different. For smaller breaks, MAAP4 tended to predict key events later than NOTRUMP, while for larger breaks the results were mixed. As noted by Westinghouse, there were cases in which the predictions of the two codes were substantially different. These differences were attributed by Westinghouse to the non-conservative MAAP4 calculations of ADS-1, 2, and 3 depressurization rates. However, the staff agrees that MAAP4 predictions of "success" are typically confirmed by corresponding NOTRUMP (and, as appropriate, LOCTA) calculations. On the basis of the benchmark study comparisons, the staff has determined that MAAP4 is an adequate screening tool for evaluating PRA success criteria for the AP600, subject to the limitations discussed by Westinghouse in WCAP-14869.

In conjunction with the MAAP4/NOTRUMP benchmarking study, Westinghouse prepared and submitted WCAP-14800, which evaluated the impact of thermal-hydraulic uncertainty on the conclusions drawn from the AP600-focused PRA. The report discussed the event sequences determined in the focused PRA and described how they were expanded and screened using the MAAP4 code. Both short-term behavior (accident initiation through IRWST injection) and long-term (IRWST injection, transitioning to sump recirculation) were evaluated. The uncertainty evaluation focused primarily on those sequences that resulted in core uncovery, and determined the impact on the focused PRA results if core uncovery sequences were considered not to result in "success." Thermal-hydraulic uncertainty with respect to containment performance was not explicitly evaluated, but the impact of the failure of containment isolation (resulting in the necessity to more completely depressurize the RCS) was considered. Westinghouse concluded that consideration of thermal-hydraulic uncertainties related to the performance of the passive safety systems slightly increased the values of CDF and LERF derived from the focused PRA, but these values were still less than the maximum acceptable values discussed in SECY-94-084. Thus, no RTNSS controls were required on non-safety-related systems beyond those identified in WCAP-13856.

The staff has performed a brief review of WCAP-14800, and has determined that Westinghouse addressed the appropriate (high-importance) contributors to thermal-hydraulic uncertainty related to performance of the passive safety systems. However, an in-depth review of this report was not performed as a result of Westinghouse actions concerning RTNSS that are not reflected in the thermal-hydraulic uncertainty assessment in WCAP-14800. The staff's evaluation of those actions and their potential impact on the uncertainty assessment in WCAP-14800 are discussed in more detail in Sections 22.5.4.3 and 22.5.4.4 of this report.

As discussed in Chapter 6 of the PRA, the staff has also reviewed Westinghouse's success criteria for the baseline PRA from the standpoint of system thermal-hydraulics and associated uncertainties reported in WCAP-14800, "AP600 Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability. On the basis of insights gained from the evaluation of Westinghouse's focused PRA (see Section 19.1.7 of this report), the most significant factor in the PRA scenarios that rely on passive systems only, particularly with regard to passive system performance, is the number of ADS-4 valves that open to complete the primary system depressurization and permit passive safety injection. The PRA success criterion that the opening of two out of four ADS-4 valves will ensure adequate depressurization of the AP600 RCS is reasonable in most cases. However, uncertainties related to pressure losses in the ADS-4 valves, as reflected in MAAP-4 modeling of the AP600 (see WCAP-14869 and WCAP-14800), required that three out of four ADS-4 valves be opened to ensure adequate depressurization. Since three out of four ADS-4 valves is equivalent to the conditions existing in many design basis sequences, and have been shown to be adequate in design basis analyses using conservative assumptions, the staff concluded that this is acceptable.

If active systems, such as the RNS, are available to provide pumped injection to the RCS, the impact of the thermal-hydraulic uncertainty in system response regarding the number of ADS-4 valves becomes relatively unimportant, since the system does not need to be fully depressurized for the RNS to provide adequate pumped injection.

Westinghouse evaluated the impact of changing the success criterion from two to three out of four ADS-4 valves in Section 50.5.6 of the PRA and determined that the impact on the CDF is small (approximately 4 percent increase in CDF). The staff estimates that this change may increase the CDF by as much 10 percent (See Section 19.1.3.5.2 of this report). In either case, this is not significant. Thus, the staff concluded that Westinghouse's success criteria are reasonable from the standpoint of system thermal-hydraulic performance.

22.5.4.2 Focused PRA

The staff's evaluation of the focused PRA can be found in Section 19.1.7 of the report.

22.5.4.3 Systems Proposed for RTNSS Controls

As discussed in Section 19.1.7, the staff concluded, as a result of its sensitivity studies, that RTNSS controls were needed, beyond those proposed by Westinghouse in WCAP-13856, to meet the safety goal objectives established for the focused PRA in SECY-94-084. According to the RTNSS process, non-safety-related SSCs that are placed under RTNSS controls can be credited in the focused PRA, which has the effect of reducing the estimated CDF and LERF. Westinghouse proposed, in a letter dated July 16, 1997, to extend RTNSS controls to several non-safety-related SSCs in addition to those originally identified in WCAP-13856 (see

Section 22.5.1), including onsite equipment needed to support post-72-hour operation (long-term shutdown) as discussed in Section 22.5.3. Additional administrative controls were proposed for the following SSCs:

- DAS Engineered Safety Features (ESF) Actuation in all modes of operation
- Injection Mode of the RNS in Modes 1, 2, and 3
- Shutdown cooling Mode of the RNS in Mode 5 (RCS pressure boundary open) and Mode 6 (with upper internals in place or refueling cavity less than full)
- Passive Containment Cooling System (PCS) Water Makeup for long-term cooling in all modes of operation
- Main Control Room (MCR) cooling for long-term shutdown in all modes of operation
- I&C Room Cooling for long-term shutdown in all modes of operation
- Hydrogen Igniters in Modes 1, 2, 5, and 6
- ac power supply (offsite and/or onsite standby diesel generators) in Modes 1, 2, 3, 4, 5 and 6
- ac power supply for long-term shutdown in all modes of operation

Westinghouse's submittal included the proposed means for the implementation of RTNSS controls in the form of short-term administrative availability controls. The administrative controls are formatted similar to Technical Specifications, with operability requirements, applicability, actions and completion times if operability requirements are not met, surveillance requirements, and bases for the availability controls. However, there are no limiting conditions for operation (i.e., there is no requirement to bring the plant to a safe-shutdown condition when operability requirements are not fulfilled) if the completion times for required actions are not met. Following discussions with the staff about the acceptability of the systems proposed for RTNSS controls and the general format of the administrative availability controls, this information was incorporated into the AP600 SSAR as Section 16.3. Westinghouse also revised WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," to update the resolution status of the RTNSS process.

The staff reviewed the systems identified for RTNSS controls and the form and content of the administrative controls proposed by Westinghouse. In a letter dated October 2, 1997, the staff informed Westinghouse that it agreed that Westinghouse had identified appropriate non-safety-related SSCs for inclusion under RTNSS controls per the criteria discussed in SECY-94-084. However, the following comments and questions were raised by the staff:

- linkage of administrative controls and specified availability and reliability goals to the Maintenance Rule (10 CFR 50.65)

- clarification of availability goals as discussed in the BASES section of the administrative controls, and addition of reliability goals
- clarification of the linkage between the Reliability Assurance Process and the regulatory requirements addressing reliability assurance, such as the Maintenance Rule and quality assurance requirements (10 CFR Part 50, Appendix B)
- inadequate basis for availability of the hydrogen igniters and clarification of the definition of operability of the igniter system
- surveillance frequency for the DAS
- consistency of guidance for calibration of resistance temperature devices (RTDs) with the Standard Review Plan

The staff discussed these issues with Westinghouse, and Westinghouse responded to the staff's concerns in a letter dated December 12, 1997. The staff reviewed Westinghouse's responses and had additional discussions with Westinghouse to resolve the specific issues regarding the content of the administrative controls as documented in Westinghouse letters dated January 30, February 10, and February 27, 1998. Westinghouse has revised the SSAR to incorporate the commitments in these letters that address the staff's concerns listed above. Therefore, the staff finds the RTNSS administrative controls acceptable. The COL applicant referencing the AP600 standard design shall develop and implement procedures consistent with the administrative availability controls in Section 16.3 of the SSAR. This is COL Action Item 22.5.4.3-1.

In addition to the administrative controls provided for the RTNSS systems in AP600 SSAR Section 16.3, the reactor vessel insulation system was identified as an additional RTNSS item. The staff concluded that including this system under D-RAP as identified in Table 17.4-1 of the SSAR is sufficient regulatory oversight as discussed in Section 19.1.7 of this report.

22.5.4.4 Impact of RTNSS Implementation on Thermal-Hydraulic Uncertainty and Passive System Performance Reliability

As discussed in Section 22.5.4.1, the staff began its review of the issue of AP600 thermal-hydraulic uncertainty before Westinghouse's proposed addition of several systems to its implementation of RTNSS (Section 22.5.4.3). The primary concerns raised by the staff in its initial review of this issue centered on the following two aspects of passive system performance:

(1) effects of uncertainties in thermal-hydraulic parameters, correlations, and models on calculation of passive system performance

(2) low driving forces inherent in passive systems (gravity drain and density-driven injection)

These two aspects are not independent of one another. For example, uncertainties in the pressure losses through a gravity-drain injection system derived from modeling the system (i.e., pipe roughnesses and friction factors; losses through expansions, contractions, valves, orifices, or bends) have a direct impact on the pressure difference available to drive flow from the injection source to the reactor coolant system. These uncertainties, in turn, can be reflected in the focused PRA in terms of the number of operational SSCs (e.g., ADS valves) required to

ensure "success." This was the case, as described in Section 22.5.4.1, where Westinghouse determined, through the MAAP4/NOTRUMP benchmarking study, that three out of four ADS-4 valves are required for the focused PRA thermal-hydraulic success criteria, rather than two out of four ADS-4 valves.

Westinghouse's proposal to include the RNS and its electrical (ac) support systems in its RTNSS implementation directly addresses the staff's concerns with regard to passive system performance. The RNS provides two functions that can be of substantial benefit in mitigating the effects of accidents:

(1) RNS can be aligned to pump water from the IRWST to the reactor vessel through the DVI lines at RCS pressures at or below the normal operating pressure of the RNS (about 1.4 MPa or 200 psia)

(2) RNS can provide decay heat removal from the RCS at low pressures (assuming that Component Cooling Water and Service Water are available)

The first function means that it is not necessary for the AP600 plant to depressurize fully to pressures slightly above containment pressure for injection from the IRWST to be initiated. The RNS operating pressure is reached by depressurization through a relatively small number of ADS-1, 2, and 3 valves. In addition, the use of pumped injection rather than gravity drain reduces the importance of uncertainties in pressure losses and other concerns, such as the ability of low-driving-force flows to force open sticking check valves. The second function provides a backup to the safety-related PRHR system, and allows decay heat removal at low RCS temperatures, which the PRHR system cannot accomplish.

In addition to the above considerations, the staff has performed extensive confirmatory testing and analysis related to AP600 system response to beyond-design-basis accidents. The tests have been performed in the ROSA/LSTF loop at the Japan Atomic Energy Research Institute (JAERI) and the APEX facility at Oregon State University (OSU). (See Chapter 21 for additional discussion of confirmatory testing). Staff analyses were performed with a version of the RELAP5/MOD3 best-estimate computer code, modified to model AP600 passive systems. The confirmatory tests investigated a wide range of initiating events, with multiple passive safety system failures, and provided data for validation of AP600 computer models. The results of the confirmatory tests and analyses indicate that the success criteria established by Westinghouse in the focused PRA are appropriate, and that the plant can respond successfully (i.e., without core damage) to accidents even with a significant number of passive safety systems failing to operate. With regard specifically to the number of ADS-4 valves required for "success" in multiple-failure scenarios, confirmatory testing and analysis has indicated that, in many instances, two out of four ADS-4 valves may be adequate to achieve full depressurization, IRWST injection, and maintain long-term cooling. For some scenarios, three out of four ADS-4 valves may be required, depending on other system conditions. This is equivalent to conditions within design-basis events (assuming failure of one ADS-4 valve as the single active failure), and has been found to be adequate within the scope of both Westinghouse's and the staff's test programs.

A final consideration on Westinghouse's RTNSS implementation proposal relates to the focused PRA. The RTNSS process, as discussed in SECY-94-084, allows credit in the focused PRA for

non-safety-related SSCs that have been placed under RTNSS controls. Westinghouse has not provided a complete, revised focused PRA that explicitly incorporates the systems now proposed for RTNSS controls. However, based on its evaluation of Westinghouse's analyses of focused PRA event sequences (see Section 22.5.4.1) and the resulting CDF and LERF values without inclusion of RTNSS-controlled systems, the staff believes that inclusion of those systems with availabilities consistent with those achieved for non-safety-related systems at currently operating plants will result in focused PRA CDF and LERF values well below the maximum acceptable values specified in SECY-94-084.

As a result of the staff's evaluation of the impact of RTNSS-controlled systems, as currently proposed by Westinghouse, the staff concludes that its concerns related to thermal-hydraulic uncertainties and passive system performance reliability are adequately addressed.

22.6 Quality Assurance

In SSAR Chapter 17, "Quality Assurance," Table 17.1, "Quality Assurance Program Requirements for RTNSS Systems, Structures, and Components," Westinghouse establishes the quality assurance requirements for RTNSS systems, structures, and components identified in the AP600 design.

23 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) has completed its review of the AP600 design. The Subcommittee on Thermal-Hydraulic Phenomena met with Westinghouse and the staff to discuss the Westinghouse testing program as well as the confirmatory testing being performed by the NRC Office of Research. The Subcommittees on Westinghouse Standard Plant Designs and on Advance Reactor Designs began its detailed review of the AP600 design in January 1995.

The ACRS has issued interim letter reports pertaining to the AP600 design dated February 19, 1998; April 9, 1998; and June 15, 1998. The staff responded to the interim letter reports in its letters dated March 23, May 20, and August 3, 1998, as well as during meetings with the ACRS on the AP600. At the 454th meeting of the ACRS Full Committee, the Committee considered the Westinghouse application for certification of the AP600 design, and issued its final report on July 23, 1998 to the Chairman of the NRC. This letter is included as Appendix G to this report. A chronology of the review by the ACRS is provided in the Attachment to the July 23, 1998 letter.

In the July 23, 1998 report, the ACRS states that Westinghouse's proposed resolutions of the issues identified in the interim letter reports are acceptable, pending staff review and approval. The ACRS concludes that, based on its

> ...review of those portions of the AP600 application which concern safety, [the ACRS] believe that acceptable bases and requirements have been established to ensure that the AP600 design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

However, the Committee identified a generic concern, stating that "[t]he present regulatory requirements for qualifying mechanical equipment are insufficient to ensure continued passive autocatalytic recombiner operation for the expected duty cycle." Environmental qualification testing of passive autocatalytic recombiners (PARs) was an item discussed in the ACRS' third interim letter dated June 15, 1998, regarding the AP600, which the staff responded to in its August 3, 1998 letter to the ACRS.

The SSAR commits the COL applicant to qualifying the PARs in accordance with the environmental qualification program for mechanical components that is discussed in Section 3.11 of this report. In addition to the normal qualification testing for mechanical components, the PARs qualification testing for the AP600 will address the possible poisoning effects from the source term constituents described in NUREG-1465, phosphates, and silicone oil. The staff has accepted the level of environmental qualification for the PARs discussed in this report, in part, because of the excess capacity provided by the two full-size PARs and two quarter-size PARs. Analyses performed by the applicant show that one PAR, the performance

of which is degraded by 90 percent, is capable of maintaining the hydrogen concentration inside containment below the lower flammability limit of 4 percent following a design-basis loss-of-coolant accident. In addition, the AP600 has been provided with 64 igniters to control hydrogen if the concentration of hydrogen inside containment exceeds the lower flammability limit of 4 percent. Furthermore, the PARs were not credited in the AP600 PRA because they were not perceived to be risk significant. Thus, their unavailability would not affect the large early release frequency.

Therefore, on the basis of the SSAR commitment to have the COL applicant environmentally qualify the PARs as specified in Section 6.2.4.1.2 of the AP600 SSAR, and the excess capability provided by the PARs' design, the staff concludes that the present regulatory requirements for qualifying mechanical equipment are sufficient to ensure continued operation of the PARs for the expected duty cycle of the AP600 standard plant.

24 CONCLUSIONS

The staff performed its review of the AP600 standard safety analysis report, probabilistic risk assessment, and Tier 1 information in accordance with the standards for review of design certification applications set forth in 10 CFR 52.48 that are applicable and technically relevant to the AP600 standard design, including the exemptions identified in Section 1.6 of this report. On the basis of its evaluation and independent analyses as discussed in this report, the staff concludes that, subject to satisfactory resolution of the confirmatory items identified in Section 1.9 of this report, Westinghouse Electric Company's application for design certification meets the requirements of 10 CFR 52.47 that are applicable and technically relevant to the AP600 standard design. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR 52.53 is provided in Appendix G of this report.

The staff also concludes that issuance of a final design approval, in accordance with Appendix O to 10 CFR Part 52, will not be inimical to the common defense and security or to the health and safety of the public. The financial qualifications of the applicable utility and the indemnity requirements of 10 CFR Part 140 will be addressed during the plant-specific licensing process for an application that references the AP600 standard design.

A final design approval, issued on the basis of this FSER, does not constitute a commitment to issue a permit, design certification, or license, or in any way affect the authority of the Commission, the Atomic Safety and Licensing Board, and other presiding officers, in any proceeding pursuant to 10 CFR Part 2.

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Project No. 676, Docket No. 52-003

11. ABSTRACT *(200 words or less)*

This final safety evaluation report (FSER) documents the technical review of the AP600 standard nuclear reactor design by the U.S. Nuclear Regulatory Commission (NRC). The application for the AP600 design was submitted on June 26, 1992 by Westinghouse Electric Corporation in accordance with Subpart B, Standard Design Certifications," of Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part 52), and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The AP600 nuclear reactor design is a pressurized water reactor with a power rating of 1933 MWt with an electrical output of at least 600 MWe. The AP600 design contains many features that are not found in current operating reactor designs. For example, a variety of engineering and operational improvements provide additional safety margins and address the Commission's severe accident, safety goal, and standardization policy statements. The most significant improvement to the design is the use of safety systems that use passive means (such as gravity, natural circulation, condensation and evaporation, and stored energy) for accident prevention and mitigation. These passive safety systems perform safety injection, residual heat removal, and containment cooling functions. Unique features of the AP600 design include an enhanced reactor core design, larger reactor vessel, larger pressurizer, an in-containment refueling water storage tank, automatic depressurization system, revised main control room design with a digital microprocessor-based instrumentation and control system, hermetically-sealed canned reactor coolant pump motors mounted to the steam generator, and increased battery capacity. In addition, the facility is designed for a 60-year life, and employs modular construction techniques in its design.

On the basis of its evaluation and independent analyses, the NRC staff concludes that Westinghouse's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the AP600 standard design.

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

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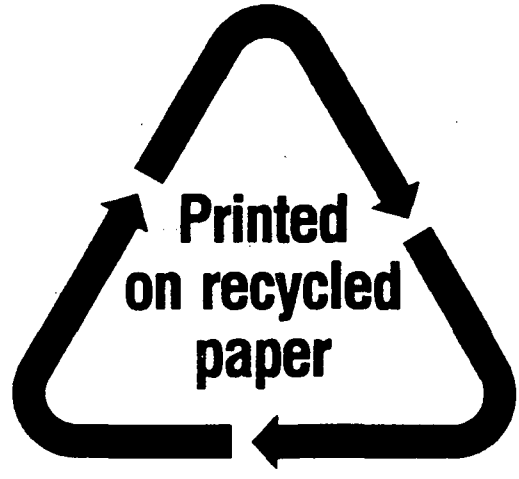

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