

21 TESTING AND COMPUTER CODE EVALUATION

21.1 Introduction

The AP1000 standard design is a two-loop, pressurized-water reactor (PWR) with an electric output of approximately 1,117 MWe, evolving from the AP600 passive plant design with an electric output of approximately 600 MWe. These advanced plant designs differ from the conventional PWRs in that passive safety systems are used for accident mitigation. Unlike the conventional active safety systems, these passive systems use only natural forces (such as gravity, natural circulation, and compressed gas). These driving forces provide means to cool the reactor core following an accident.

Since the AP1000 plant design has a thermal power of approximately 3,400 MWt (compared to 1,933 MWt for the AP600 standard plant design), the major differences from the AP600 design (as summarized in Table 1) are increased capacities of the major components to accommodate the increased thermal output. In particular, the AP1000 reactor system has a taller reactor core with a longer active fuel length, more fuel assemblies, and higher power density; a larger pressurizer; larger steam generators (SGs) with more tubes and larger heat transfer areas; and larger canned reactor coolant pumps (RCPs) with higher head, capacity, and inertia. In addition to a taller containment with a larger free volume, the capability of the AP1000 passive safety systems is increased with a larger core makeup tank (CMT) diameter; a larger in-containment refueling water storage tank (IRWST) and larger injection line diameters; a larger passive residual heat removal (PRHR) system with more tubes and longer tube length in the heat exchanger, and larger inlet and outlet line diameter; and larger valve and flow path diameters for Stage 4 of the automatic depressurization system (ADS-4). Given these differences, Westinghouse Electric Company (Westinghouse or the applicant) asserted that the AP1000 design represents an incremental change to the AP600 standard plant design, because it maintains and preserves the design configuration and arrangement, key design features, and performance characteristics of the AP600 design. Consequently, the applicant concluded that the AP600 test program and the computer codes used for safety analyses of the AP600 design-basis events also apply to the AP1000 design.

WCAP-15612, "AP1000 Plant Description & Analysis Report," December 2000, describes the AP1000 design, compares it with the AP600 design, and provides a partial, preliminary AP1000 safety analysis and margin assessment. WCAP-15613 "AP1000 PIRT and Scaling Assessment," February 2001, presents: (1) the AP1000 phenomena identification and ranking tables (PIRTs) for large-break loss-of-coolant accidents (LBLOCAs), small-break loss-of-coolant accidents (SBLOCAs), long-term cooling (LTC), non-LOCA transients, and the containment response; (2) an overview of the AP600 test program; and (3) scaling assessments of important separate-effects, integral-effects, and containment tests.

WCAP-15644, "AP1000 Code Applicability Report," April 27, 2001, documents the the applicant's assessment of the safety analysis codes that were developed and approved for the AP600 design certification to determine their applicability for use in the AP1000 design. Specifically, those safety analysis codes are (1) LOFTRAN for non-LOCA transients and SG tube rupture analyses, (2) NOTRUMP for SBLOCA analyses, (3) WCOBRA/TRAC for LBLOCA and LTC analyses, and (4) WGOthic for containment analyses.

WCAP-15833, "WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," Revision 2, December 2002, provides supplemental information intended to demonstrate that NOTRUMP provides a conservative simulation of ADS-4 venting and the onset of IRWST injection in the AP1000 design. This report includes additional validation for models in NOTRUMP and WCOBRA/TRAC for important processes during ADS-4 venting and the transition to IRWST injection.

This safety evaluation report provides the U.S. Nuclear Regulatory Commission (the NRC or staff) staff's assessment of the application to the AP1000 of the AP600 passive core cooling system test program and the LOFTRAN, NOTRUMP, and WCOBRA/TRAC analysis codes to the AP1000 standard plant design. The assessment of the AP1000 passive containment cooling system and the WGOthic code is addressed separately in this chapter. The staff's evaluation documented in this chapter concentrates on the differences between the AP1000 and the AP600 design with the understanding that the AP600 testing and computer codes were found to be acceptable for the AP600 design in accordance with the staff's evaluation documented in Chapter 21 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998. This chapter currently contains references to NUREG-1512, which provides the basis for accepting the AP600 testing and computer codes. Prior to issuing the final safety evaluation report for the AP1000, the staff will remove these references and replace the references with the basis for its conclusion that the testing and computer codes are acceptable for the AP1000. This is DSER Open Item 21.1-1.

21.2 Overview Of Advanced Passive Plant Testing Programs And Scaling Assessment

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 52.47(b)(2)(i)(A) specifies the test program requirements for certification of a standard design that utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions. These requirements essentially mandate that a passive plant vendor must develop and conduct a design certification test program that includes both separate-effects and integral-effects tests. Moreover, those tests must be of sufficient scope to provide adequate data to assess the computer codes used to analyze plant behavior over the range of conditions of normal operation, transients, and accident sequences.

For the AP600 design certification, the applicant performed both separate-effects tests and integral-effects tests to investigate the behavior of the AP600 passive core cooling systems (PXS) and to develop a database for validation of the LOFTRAN, NOTRUMP, WCOBRA/TRAC codes used to analyze the design-basis transients and accidents and the WGOthic code for containment analyses. In addition, because some of these tests were performed with scaled test facilities, the applicant also conducted scaling analyses to demonstrate the acceptability of the test database. The staff's evaluation of the AP600 test programs and scaling assessment, as well as validation of the analysis codes, are documented in Chapter 21 of NUREG-1512.

21.3 Advanced Passive Plant Testing Programs

The AP600 passive core cooling tests (summarized in Table 3.1-1 of WCAP-15613) include (1) the separate-effects tests on the passive residual heat removal heat exchanger (PRHRHX), ADS, and CMT; and (2) the integral-effects tests performed at the Advanced Plant Experiment (APEX) facility and the Simulatore per Esperienze di Sicurezza (SPES) facility.

The PRHRHX separate-effects tests were performed at Westinghouse's Science and Technology Center near Pittsburgh, Pennsylvania, with three vertical tubes submerged in a water tank to determine heat transfer characteristics of the PRHRHX and mixing characteristics

in the IRWST. Because the AP600 PRHR has a “C-tube” heat exchanger, the applicant provided justification for the applicability of the straight-tube PRHRHX test data to the “C-tube” configuration. The applicant also performed “blind” calculations of selected NRC-provided data, which were obtained from the NRC’s confirmatory test program in the Rig of Safety Assessment (ROSA)/Large-Scale Test Facility loop in Japan, which employs a simulated C-tube PRHRHX of prototypical bundle dimensions (with approximately one-thirtieth the number of heat exchanger tubes) immersed in a tank of water simulating the IRWST. The results of the Westinghouse calculations of the ROSA facility data demonstrated the adequacy of the straight-tube-based correlation for analysis of the C-tube PRHRHX.

The ADS separate-effects test program at the “VAPORE” facility of the Central Research Establishment of the Italian Energy Agency utilized full-size configuration of the piping network, exhaust pipe, and sparger for ADS Stages 1, 2, and 3 (ADS-1/2/3). The tests consisted of two phases. Phase A tests were performed for the ADS-1/2/3 with steam flow through a sparger into a larger water-filled tank to investigate the capacity of the ADS sparger in the IRWST and determine the dynamic effects on the IRWST structure. The Phase B tests were performed to investigate the thermal-hydraulic behavior of the ADS valves, piping, and sparger in the IRWST. In addition, the tests provided thermal-hydraulic performance data with test conditions that sufficiently covered the operating conditions expected in the AP600 depressurization mode.

The applicant did not perform ADS-4 separate-effects testing for the AP600 design. The applicant stated that ADS-4 was sized conservatively and tested as part of the integral-effects tests. The CMT separate-effects test facility at Westinghouse’s Waltz Mill facility in Pennsylvania is a scaled facility that adequately represents the key features of the reactor coolant system (RCS) and connecting piping that could affect CMT performance. This includes the relative elevations of the reactor vessel and CMT, as well as the flow resistance of the pressure balance line and the drain line. WCAP-13963, “Scaling Logic for the Core Makeup Tank Test,” Revision 1, 1995, documented the scaling of the AP600 CMT tests. In general, the applicant conducted the tests to characterize the CMT over the full range of thermal-hydraulic conditions that the plant will experience. The important phenomena studied included thermal stratification in the CMT, as well as the effects of recirculation, draining, and plant depressurization on CMT behavior. The tests also verified operation of the tank level instrumentation.

The SPES facility is a full-height, full-pressure, scaled full power, integral-effects test facility located at the Societa Informazioni Esperienze Termoidrauliche’s in Piacenza, Italy. The facility provided data to evaluate the operation of the PXS at high pressure, including response to an SBLOCA, SG tube rupture (SGTR), and steamline break transients.

The APEX facility is a one-fourth-scaled, low-pressure integral-effects test facility at Oregon State University (OSU). The facility provided data to evaluate the operation of the PXS at low pressure in the last part of depressurization and long-term cooling behavior in SBLOCA events. The test matrix focused on SBLOCAs for two reasons. First, within the design basis, LOCAs are the only events that cause the ADS-4 to actuate and to progress to LTC. Second, the applicant’s calculations indicated that the LBLOCA response in the AP600 is similar in many respects to that of conventional designs, and the applicant asserted that important phenomena related to LTC in an LBLOCA would be similar to SBLOCA behavior.

21.3.1 Scaling Assessment Methodology

The applicant conducted a scaling assessment to demonstrate the acceptability of the test data base for advanced passive plant design. For AP600, that assessment was documented in, WCAP-14727, "AP600 Scaling and PIRT Closure Report," Revision 2, February 1998. The scaling assessment process is based on the hierarchical, two-tiered scaling (H2TS) analysis methodology, which was first developed by N. Zuber, NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," Appendix D, "A Hierarchical Two-Tiered Scaling Analysis," November 1991. The H2TS method consists of a top-down, global, system-level scaling analysis, which considers actively participating systems and components during a transient. In addition a bottom-up component-level scaling analysis is performed, which considers important processes that occur locally within a specific region. Based on these principles, the applicant's scaling assessment methodology consists of the following steps:

- Develop a PIRT for transients and accidents to identify the important phenomena at the component level.
- Perform single-loop, system-level, top-down analyses to identify the thermal-hydraulic processes that are important to the interaction of the components.
- Compare the component-level phenomena in the PIRT with the important system-level thermal-hydraulic processes to ensure that the high-ranked component-level phenomena include those that influence the significant system-level processes.
- Perform bottom-up analyses to calculate the dimensionless time ratios (or Pi groups) of the high-ranked phenomena, identified in the PIRT for the SBLOCA and LTC, for each component.
- Compare the system-level and component-level Pi groups of the integral-effects tests (APEX and SPES) and separate-effects tests with the AP600 Pi groups to identify distortions in the tests.

The top-down scaling analysis is performed from the system-level conservation equations of mass, momentum, and energy to describe the system response to an accident. The parameters in the system equations are then normalized over their expected range to arrive at dimensionless system equations. The coefficients of the dimensionless system equations are then divided by the expected dominant term. The resulting coefficients for the terms in the normalized equations are the Pi groups, which must be preserved if a test facility is properly scaled.

The values of Pi groups within the normalized equations provide the measures of the relative importance of the parameters. Ratios of the Pi values of the facilities to the plant indicate proper scaling or distortion of the test facilities. If the Pi ratio of the test facility to the AP1000 is within the acceptance criteria (generally between 0.5 and 2), the facility is appropriately scaled for that Pi group.

The applicant's scaling analysis considered the regions of the system that were the most active, and the SBLOCA and LTC transients were divided into the following six discrete periods:

- single-phase natural circulation with active SGs;

- single-phase natural circulation with PRHR providing heat removal;
- two-phase natural circulation with PRHR providing heat removal;
- ADS blowdown;
- IRWST injection; and
- sump injection.

21.3.2 Scaling Assessment of Applicability of the Advanced Passive Plant Test Programs to the AP1000 Design

In support of its assertion that the AP600 test program is sufficient to meet the test requirements for a design certification application for the AP1000 standard plant design, the applicant submitted Topical Report WCAP-15613 which describes the AP1000 PIRTs and scaling assessment. In addition to evaluating the AP1000 PIRTs and scaling assessment performed by the applicant, the staff also performed an independent scaling analysis to address the applicability of the AP600 test program to the AP1000 standard plant design. This will be addressed in Sections 21.3.8.2 and 21.3.8.3 of this report.

21.3.3 Phenomena Identification and Ranking Tables

Section 2 of WCAP-15613 contains separate PIRTs for LBLOCAs, SBLOCAs, and non-LOCA transients for the AP1000 design, as well as those for the AP600 design. The PIRTs provide a means to identify and classify, in terms of importance, the thermal-hydraulic phenomena expected to occur in transients and accidents that must be included in the analytical models, and for which data must, therefore, be available to evaluate those analytical models. The NRC staff evaluated the AP600 PIRTs during the AP600 design certification review, and found that they capture the important phenomena, processes, and components. The staff also notes that, in general, the AP1000 PIRTs are very similar to those of the AP600 design, with only minor changes (mostly in the importance ranking of certain phenomena). In addition, although the applicant developed a separate LTC PIRT for the AP600 design, it merged the LTC PIRT into the SBLOCA PIRT for the AP1000 design. This is because the sump injection for LTC is also the final phase of the SBLOCA recovery transient. In the combined SBLOCA PIRT, the phenomena that were ranked higher in the LTC PIRT than in the SBLOCA PIRT remain higher. Also, because of the higher steam flow rate expected to result from increased core power in the AP1000 design, hot leg entrainment during the ADS-4 blowdown, IRWST injection, and sump injection phase is increased to a "high" importance ranking from "medium" in the AP600 as is the ADS-4 two-phase pressure drop during the IRWST injection phase. The entrainment/de-entrainment in the upper head and upper plenum region is also increased from "medium" to "high" ranking. In addition, some phenomena ranked as "low" importance for the AP600 design are changed to "medium" importance in the AP1000 PIRT. The applicant stated that the AP1000 PIRTs were reviewed by a group of nuclear industry experts, and some of the changes reflect their suggestions.

The staff agrees with the the applicant's positions that the AP600 and AP1000 PIRTs for LBLOCA, SGTR, and non-LOCA transients are very similar, and no new "high" ranked phenomena are expected. The minor changes in "low" and "medium" ranked processes are considered appropriate.

The staff also finds that the AP1000 SBLOCA PIRT appropriately ranks the important phenomena. The staff requested that the applicant evaluate a potential for a condensation induced water hammer (CIWH) in the direct vessel injection (DVI) line, which could occur when cold CMT or accumulator water contacts a low-velocity, stratified steam-water mixture in the DVI line during the early part of the ADS-1/2/3 blowdown in an SBLOCA. In its response, the applicant states that CIWH potential in the AP1000 DVI line is small, as substantiated by the following considerations:

- A comprehensive water hammer assessment performed by the applicant for the AP600 design concluded that water hammer potential for the DVI piping is small, given the criteria described in NUREG/CR-6519, "Screening Reactor Steam/Water Piping Systems for Water Hammer," September 1997, for piping configurations and thermal/hydraulic conditions that can result in water hammer induced by steam bubble collapse.
- Water hammer evaluations for the tests performed at the APEX and SPES facilities found no evidence of significant CIWH events in the DVI lines. There was some evidence of CIWH in the vessel downcomer at the APEX facility during recovery following simulated SBLOCAs when the accumulator injection flow rate was high. However, given the water slug velocity in the downcomer calculated by the RELAP5 code, the applicant estimated that the peak pressure in the AP600 reactor is less than 150 psi, which is significantly less than the 400 psid differential for which the affected reactor vessel components are designed.

Because the AP1000 and the AP600 DVI piping and thermal-hydraulic conditions during recovery from design-basis events are almost identical, the applicant concluded that the CIWH potential is small, with no need to add the CIWH to the SBLOCA PIRT for the AP1000 standard plant design. The staff also concludes that the CIWH is not a high-ranking phenomenon, given that the DVI line is expected to be full at the initiation of the ADS-1/2/3 blowdown, which would reduce the probability of CIWH at a time when it would be the most severe.

The staff also evaluated a need to increase the ranking of pressurizer phenomena during the ADS blowdown period. Since the AP1000 ADS-1/2/3 valve size is the same as in the AP600 design, velocities during ADS-1/2/3 blowdown are expected to be similar to those in an AP600 plant. The staff concludes that the AP1000 ranking of these processes is acceptable.

In summary, the staff finds that the AP1000 PIRTs (with the changes from the AP600 PIRTs), reflect the changes associated with the AP1000 design characteristics of higher core power density and steam flow rate.

21.3.4 Separate Effects Tests

The applicant performed separate-effects tests for the CMT, ADS valves, and PRHR heat exchanger in support of the AP600 system design. Section 4.1.1 of WCAP-15613 describes the applicant's evaluation of the applicability of these separate-effects tests to the AP1000 standard plant design. The staff's evaluations of these tests for applicability to the AP1000 design are summarized in the following subsections.

21.3.5 CMT Tests

The AP1000 CMTs have larger diameters and drain rates than do the AP600 CMTs. As described in WCAP-13963, for scaling of the CMT tests for the AP600 design, key

dimensionless Pi group parameters to scale CMT circulation and heat transfer include the Richardson number and the Friction number. In this case, the scaling assessment described in WCAP-15613 shows that the ratio of the Richardson number to the Friction number for the AP1000 CMT remains acceptably close to those in the test matrix. The applicant also showed that other CMT scaling groups (e.g., the Stanton number, the liquid heat source ratio, and the heat source ratio) that are affected by the larger CMT diameter and drain rate also remain reasonably scaled for the AP1000. Therefore, the CMT tests can be considered to be acceptable for the AP1000 design as they were for the AP600 design. The staff therefore concludes that the CMT tests remain valid for the AP1000 code validation.

21.3.6 ADS-1/2/3 Valve Tests

The ADS-1/2/3 system for the AP1000 standard plant design is identical to that of the AP600 design. Since the flow through the ADS-1/2/3 is expected to be choked during its blowdown, and since simulations have revealed that upstream pressures in the AP1000 are very similar to those in the AP600 design, thermal-hydraulic conditions affecting ADS-1/2/3 performance will be close to those in the AP600 design. Consequently, tests performed to investigate ADS-1/2/3 valve performance for the AP600 design, which included a wide range of actuation pressures and flow qualities, are considered appropriate to represent conditions in the AP1000 standard plant design.

The applicant did not perform ADS-4 separate-effects testing for the AP600 design. The applicant stated that ADS-4 was treated/sized conservatively and tested as part of the integral-effects tests, and the applicant will take the same approach for the AP1000 standard plant design. Sections 21.3.8.2 and 21.4 of this report discuss the staff's evaluation of the ADS-4 testing.

21.3.7 PRHR Heat Transfer Tests

The AP1000 PRHR maintains the same "C-tube" heat exchanger, tube diameter, spacing, and pitch ratio of the AP600 PRHR. To accommodate the higher core power, however, the AP1000 PRHR line resistances are reduced to increase the natural circulation flow, and the PRHRHX heat transfer area is increased by about 22 percent by adding tubes to the top and bottom of the tube sheet and adding length to the horizontal sections. Thus, a higher proportion of the heat transfer is expected to occur by crossflow through the horizontal section in the AP1000 design than in the AP600 design.

The PRHR tests for the AP600 standard plant design, which were performed with straight vertical tubes, did not include results for horizontal crossflow. However, the applicant conducted a subsequent analysis using data from the ROSA facility using a scaled C-tube heat exchanger to show that the heat transfer correlations developed from the vertical tube data were acceptable for heat transfer performance of the PRHR C-tube heat exchanger.

During the AP600 review, a concern was raised regarding the potential for a drastic reduction in heat transfer caused by vapor blanketing attributable to violent boiling on the outer tube surface in the top horizontal tube region of the PRHR heat exchanger. This concern was resolved on the basis of the applicant's analyses of the margin of the PRHR heat exchanger heat flux to the critical heat flux limit, and the fact that vapor blanketing was not observed in the APEX, SPES, and ROSA integral-effects test facilities. Section 4.1.1.3.2 of WCAP-15613 provides an evaluation of the AP1000 PRHR, using various heat transfer correlations on the inside and outside of the PRHR tubes to determine the heat flux on the outside of the tubes and margin to the critical heat flux limit. The results show that the expected operating conditions for the

AP1000 PRHR result in external tube heat flux values that are far below the critical heat flux limits, and are bracketed by forced flow test data from the AP600 integral-effects tests.

The applicant therefore concludes that heat transfer correlations that were developed from the AP600 test data remain valid for the AP1000 PRHR. The staff's evaluation of the effect of increasing the horizontal length on the overall heat transfer has concluded that no additional testing at the Westinghouse separate-effects test facility is required. Acceptability of the the applicant codes to predict PRHR heat transfer depends on the simulation of the integral-effects data from the ROSA facility. In Section 21.6.1 of this report, the staff discusses the PRHR HX heat transfer coorelation in the LOFTRAN code verified with the data from the ROSA and SPES facilities.

21.3.8 Scaling Analysis of Integral Effects Tests

The integral-effects tests from the APEX, SPES, and ROSA facilities provide experimental data for the AP600 code validation. However, the applicant simulated and used only the SPES and APEX facilities for the AP600 code validation and design certification. The applicability of the AP600 integral-effects tests to the AP1000 code validation and design certification is evaluated through both top-down and bottom-up scaling analyses, as described in the following subsections.

21.3.8.1 Westinghouse Scaling Analysis

Section 4 of WCAP-15613 documents the applicant's scaling evaluation to demonstrate the applicability of the AP600 test program database to the AP1000 safety analysis code validation. Specifically, that scaling evaluation provides a quantitative means to show how well important phenomena are preserved in the test facilities that were originally scaled for the AP600 plant design relative to the AP1000 design.

As in the AP600 scaling evaluation of the passive core cooling system test facilities, the top-down system-level scaling analysis of the integral-effects tests is based on the SBLOCA transients. This is because an SBLOCA transient includes broad ranges of thermal-hydraulic behavior, and all of the PXS safety features are employed during the transient. In an SBLOCA, the RCS depressurizes during initial blowdown through the break. As the safeguard ("S") signal actuates the passive safety system, the RCPs trip quickly, and the RCS passes into natural circulation. In the early stage, the RCS experiences single-phase natural circulation, with the SGs providing the dominant heat sink. This is followed by a later phase when the PRHR becomes the dominant heat sink after the SGs have drained. As the primary system drains, it passes into two-phase natural circulation, in which a mixture exists in the cold and hot legs; the CMT cold leg pressure balance line is either two-phase or steam, and the CMTs are draining. There is boiling in the core and a two-phase mixture leaves the core and flows into the hot legs. Steam or a two-phase mixture enters the PRHR with single-phase water leaving.

A similar behavior occurs in the CMTs, in which a two-phase mixture or steam enters the cold leg balance line and liquid flows from the CMT to the vessel in the DVI line. As the CMT drains to a level of 67.5 percent, ADS-1 is actuated, followed by ADS-2/3, resulting in RCS depressurization by venting the steam from the pressurizer to the IRWST. The accumulators also inject borated water into the RCS as it depressurizes below the accumulator pressure.

During the ADS-1/2/3 blowdown phase, a portion of the system (such as the DVI line, vessel downcomer, and lower plenum) remains single-phase. The remainder of the system is two-phase, including the core, upper plenum, hot legs, pressurizer, and pressurizer surge line,

which now fills in response to the activation of ADS-1/2/3. As the CMT drains to 20 percent, ADS-4 is actuated, and its blowdown further depressurizes the RCS to enable IRWST injection.

The ADS-4 blowdown transition to the inception of IRWST injection is considered critical in the AP1000 passive plant design because it is in this period that minimum inventory in the reactor vessel is expected to occur. During the IRWST injection, the RCS is an open system with the IRWST feeding the reactor vessel by gravity injection, which flows through the DVI line into the downcomer, then up and around the downcomer and out the break to the sump, or down the downcomer into the core and out the ADS-4 valves on the hot legs to the containment.

As the IRWST drains, containment sump injection (or recirculation) is initiated. The sump injection period is similar to the IRWST injection, with the exception that the system is now a closed loop with the primary system coupled to the containment, which provides for LTC.

The applicant divided the SBLOCA transient into the following six phases:

- initial blowdown
- natural circulation
- ADS-1/2/3 depressurization
- ADS-4 to IRWST transition
- IRWST injection
- sump injection

One major difference in this breakdown of the transient phases from that of the AP600 design is the addition of the ADS-4 to IRWST transition. This phase was added to facilitate its study for the AP1000 review. This is the most important phase in a SBLOCA as the minimum mixture level in the reactor vessel is expected to occur during this period. As in the AP600 scaling analysis, the Westinghouse AP1000 scaling analysis does not consider the initial blowdown phase because it is a relatively short period common to both current operating plants and the advanced passive plant design and does not involve passive safety system components.

For the top-down scaling analysis, system-level conservation equations are written to address the important processes and parameters that are involved in each specific phase. The equations are combined in a form which identifies the physical processes and key parameters of interest, such as reactor vessel inventory, pressure, quality, or void fraction. The variables in the combined equations are non-dimensionalized using reference values appropriate for the specific period of the transient, and the resulting dimensional coefficients in the equations are then normalized using the coefficient of the dominant process. The end result yields dimensionless Pi groups. The test facility/plant scaling ratios of these Pi groups are then calculated and compared to the acceptance criteria to determine if the test facility is sufficiently scaled to the full-scale plant.

For the natural circulation phase, two-phase natural circulation with PRHR providing heat removal is analyzed by combining the steady-state mass, momentum, and energy equations into a core exit quality scaling ratio expression in terms of the dominant influences (such as PRHR gravity head, PRHR flow path hydraulic resistance, and core decay power). The scaling ratios of core exit quality between the test facility and the AP1000 indicate that the SPES facility is sufficiently scaled for both the AP600 and the AP1000, whereas the APEX facility is not well-scaled for the natural circulation phase.

For the ADS-1/2/3 blowdown depressurization phase, the scaling analysis is performed with the rate of pressure change equation for the ADS depressurization process. The analysis

produces the scaling ratios of two Pi groups; one group is the ratio of core steam generated by the decay heat to RCS steam volume, and the other is the ratio of the steam venting through ADS-1/2/3 to the RCS steam volume. The resulting scaling ratios show that the SPES facility is sufficiently scaled to the AP600 and AP1000 designs; however, the APEX facility has distortion in the ratio of ADS-1/2/3 steam venting to the RCS steam volume.

The top-down scaling analysis of the ADS-4 to IRWST transition phase considers the CMT injection dominating subphase, the IRWST-injection dominating subphase, and the ADS-4 depressurization phase. For the CMT-injection and the IRWST-injection dominating subphases, the scaling analyses are derived from the transient equations of the reactor vessel inventory. For the ADS-4 blowdown, the scaling analysis is derived from the rate of RCS pressure change. The scaling analyses of these subphases generate seven Pi groups. The facility/plant scaling ratios of these Pi groups showed that the APEX and SPES facilities are sufficiently scaled to both the AP600 and the AP1000 designs, when the ADS-4 flow is critical. When the ADS-4 flow is subcritical, the SPES facility is distorted as a result of the oversized ADS-4 vent paths.

The applicant also performed a scaling analysis applicable to the NRC-sponsored test NRC-25, which was performed at the APEX facility. This test included a series of 10 "core uncover tests," in which the RCS was drained to the hot leg level and the IRWST was pressurized to simulate AP600 IRWST gravity injection. The ADS-4 vents were used to depressurize the system. The top-down analysis of the IRWST injection, where the two-phase resistance dominates, derived a Pi group for the equilibrium quality. The scaling ratio of this Pi group indicates that the APEX facility is sufficiently scaled for the AP1000 standard plant design.

The scaling analyses of the IRWST and sump injection phases are performed to determine the core exit quality, which impacts the thermodynamic state, two-phase flow regime, and pressure drop. By combining the conservation of mass, momentum, and energy, an expression is developed for the core exit quality. Expressions are then derived for the core exit quality scaling ratio, which contain a density ratio, a gravity head to resistance ratio, and a core power to enthalpy ratio. The scaling ratios show that the core exit quality of the APEX facility is sufficiently scaled to both the AP600 and the AP1000 designs; however, the SPES facility is not well-scaled to either design, as it did not simulate sump injection.

It should be noted that the top-down scaling approach used in the AP1000 review is not the same approach used in licensing the AP600 design, as documented in WCAP-14727. Unlike the AP600 design, the AP1000 top-down scaling approach combines the mass, momentum, and energy equations into a single expression for the parameter of interest and significantly reduces the number of scaling groups.

To complement the applicant's top-down scaling analysis for the AP1000 standard plant design, the staff requested (in RAIs P55 through P58, issued on August 22, 2001 during the AP1000 pre-application review) that the applicant provide the AP1000 numerical values of those Pi groups listed for the AP600 design in Tables 3.2-8 through 3.2-12 of WCAP-14727, which were derived directly from the separate momentum and energy equations. Assessments of these Pi groups would provide consistency with that accepted for the AP600 design. In response to the staff's RAIs, the applicant provided the numerical values of these Pi groups for the two-phase natural circulation, ADS blowdown, and IRWST and sump injection phases. For each Pi group in a transient phase, the applicant provided the Pi value for the SPES and APEX facilities and the AP600 and AP1000 plants, as well as the facility-to-AP1000 scaling ratios.

For the majority of the Pi groups, the scaling ratios between the facility and the AP1000 design are within the acceptance criteria. For certain Pi groups, the scaling ratios are outside of the acceptance criteria, indicating scaling distortion; however, some of the distorted Pi groups are insignificant as indicated by their small Pi values relative to other more dominating terms. These insignificant Pi groups include the inertia-to-buoyancy ratio, the phase change momentum flux-to-buoyancy ratio for the natural circulation phase, the single-phase pressure compliance-to-core power ratio, the two-phase mechanical compliance-to-core power ratio for the ADS blowdown phase, and the inertia and momentum flux terms for the IRWST and sump injection phases.

For the IRWST injection phase, the scaling ratios of the Pi group of the resistance-to-buoyancy ratio for the SPES and APEX facilities are outside of the acceptance criteria. The applicant states that the formulation of this scaling group was derived from the AP600 program on the basis of the single-phase contribution to resistance of the DVI and ADS paths, which significantly understates the two-phase resistance associated with the ADS flow path. Therefore, the scaling of this phase was reformulated for the AP1000 to account for the two-phase resistance associated with the ADS flow path in WCAP-15613. The results show that the APEX facility is well-scaled to the AP1000 design, while the SPES facility shows distorted scaling.

In the ADS blowdown phase, the scaling ratios show that the SPES facility has distortions for the Pi groups of the boiling heat to core power ratio and the single-phase mechanical compliance-to-core power ratio, while the APEX facility has distortion in the sensible heat-to-core power Pi group.

To supplement the top-down system-level scaling analyses, the bottom-up scaling analyses are performed for the important local processes or phenomena (during various phases of the transient) that are not captured in the top-down scaling analysis. For the natural circulation phase, the bottom-up scaling analyses are performed for the flow patterns and phase separation at the cold leg T-junction at the CMT balance line. The cold-leg flow pattern is analyzed on the basis of the Taitel-Dukler horizontal flow regime transition map ([AIChE Journal](#), Vol. 22, No. 1, pp. 47-55, "A Model for Predicting Flow Regime Transitions in Horizontal and Near Horizontal Gas-Liquid Flow," January 1976), and the facility/plant scaling ratio of the Froude number is calculated. The resulting scaling ratios show that the APEX and SPES facilities are sufficiently scaled to both the AP600 and the AP1000 design. The scaling analysis of phase separation at the cold leg-CMT balance line junction is performed on the basis of the correlation developed by Seeger, et al. ([Int. J. Multiple Flow](#), Vol. 12, No. 4, pp. 575-585, "Two-Phase Flow in a T-Junction with a Horizontal Inlet, Part I: Phase Separation," 1986), for a top vertical branch in a non-stratified upstream flow regime, which correlated the quality ratio to the mass flux ratio of the branch and the main pipes. The facility/plant scaling ratio of the balance line-cold leg quality ratio showed that the APEX and SPES facilities are sufficiently scaled to the AP600 and the AP1000 designs.

During the initial stage of the ADS-1/2/3 blowdown when only steam is vented, the applicant states that the APEX facility surge line length-to-diameter ratio and surge line layout are preserved relative to the AP600 standard plant design to preserve the surge line pressure drop. Because those values are unchanged in the AP1000 design, the surge line pressure drop should also be preserved for that design. However, in the later stages of ADS-1/2/3 depressurization, a two-phase mixture flows through the surge line into the pressurizer.

The applicant states that the principal investigator found that the APEX facility was probably distorted for some flow patterns (such as the slug-annular flow regime transition) with respect to

the AP600 design. Therefore, it is expected that some distortion may also exist with respect to the AP1000 design. In addition, the SPES facility length-to-diameter ratio is not scaled, and the flow pattern transition scaling analysis was not performed. The applicant contends that although there may be some distortion of flow regime in the surge line of the SPES and APEX facilities, it should only affect the later stages of the ADS depressurization when a two-phase mixture is discharged. However, since venting of the gas phase has a higher impact on RCS pressure than the discharge of the liquid phase, the surge line pressure drop should be acceptably scaled for the steam venting regime and, therefore, the data from the test facilities can be used during the ADS phase for code validation.

For the ADS-4 to IRWST transition phase, the bottom-up scaling analysis considered hot leg flow pattern, liquid entrainment from hot leg into the ADS-4, and counter-current flow in the surge line during pressurizer draining. Like the cold leg flow pattern, the hot leg flow regime transition from stratified to non-stratified flow is an important phenomenon as it influences pressure drop and entrainment in the ADS-4 flow path. Taitel-Dukler's general flow regime map for the horizontal two-phase flow is used to predict flow regime transitions. A scaling ratio expression between the test facility and plant is derived on the basis of preserving the modified Froude number used in the Taitel-Dukler flow regime map and pressure similitude. The scaling ratios show that the APEX and SPES facilities are sufficiently scaled to the AP1000 design.

The scaling analysis for ADS-4 entrainment is performed for the onset of liquid entrainment on the basis of the following correlation of the onset of liquid entrainment for a vertical offtake with stratified flow in the main pipe using an expression of the form:

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1 \left(\frac{h_b}{d}\right)^{C_2}$$

where Fr is the Froude number, U_g is the superficial velocity of steam, g is gravitational acceleration, ρ_g is the density of steam, $\Delta\rho$ is density difference, d is the off-take pipe diameter, and h_b is the distance from the top of the pipe to the stratified level. The coefficients C_1 and C_2 are discussed in WCAP-15833.

A scaling ratio relation for the entrainment onset is derived assuming pressure similitude. The APEX/AP1000 scaling ratio entrainment onset is calculated to be 0.69, which indicates that the APEX facility is sufficiently scaled; however, the SPES facility has distortion with a scaling ratio of 0.14.

The scaling analysis for the countercurrent flow in the surge line and pressurizer draining is performed on the basis of the Kutateladze flooding relation. The applicant examined the scaling of the Kutateladze number during this transition phase with the pressurizer draining. For scaling purposes, because the pressurizer is poorly vented as the ADS-1/2/3 path is plugged by a column of water above the sparger in the IRWST during this phase of a transient, the mode of pressurizer draining can be described as an equal volume replacement process so that the superficial velocities of liquid and steam in the Kutateladze number are equal. With this assumption, the scaling relationship simply states that the superficial velocity (and hence the Kutateladze number) is preserved in the test facilities and plants as pressure similitude exists.

For the IRWST injection phase, the bottom-up scaling analysis is performed for the reactor core void fraction on the basis of the Yeh correlation (Nuclear Engineering and Design 60, pp.413-429, "Mass Effluence During FLECHT Forced Reflood Experiments," 1980). By preserving the

void fraction between the test facility and the plant, the scaling ratio of the Pi group for the core exit void fraction is derived. The scaling ratio of the Pi group shows that the APEX facility is sufficiently scaled.

21.3.8.2 NRC Independent Top-Down Scaling Assessment

The staff performed an independent scaling assessment to determine whether the AP600 test program also applies to the AP1000 standard plant design. The staff's review and assessment of scaling did not address containment phenomena and was limited to those affecting the AP1000 primary system. The review and assessment of the applicability of test programs to the AP1000 design considered both the primary and containment systems. (The assessment of the containment system is addressed in Section 21.6.5 of this report.)

During its assessment, the NRC staff performed both top-down system-level and bottom-up process scaling evaluations of the SPES, ROSA, and APEX facilities for applicability to the AP1000 code validation and confirmation of safety margin. In general, at least one facility is well-scaled for the AP1000 standard plant design during the early, high-pressure blowdown periods, and later after sump injection occurs. However, the transition from ADS-1/2/3 blowdown to IRWST injection shows distortions that raise significant concerns. The staff's scaling evaluation follows the methodology developed by Idaho National Engineering Laboratory (INEL) (INEL-96/0040, "Top-Down Scaling Analysis Methodology for AP600 Integral Tests," May 1997) to evaluate scaling for the AP600 standard plant design.

The independent scaling analysis considered five separate periods:

- subcooled blowdown;
- intermediate (ADS-1/2/3 venting);
- ADS-4 blowdown;
- IRWST injection; and
- sump injection.

The intermediate and IRWST injection periods were also divided into subphases to examine additional system processes. The following paragraphs discuss the staff's conclusions regarding these scaling evaluations beginning with a summary of the top-down scaling analysis.

Subcooled Blowdown Phase

The subcooled blowdown phase is initiated by the break, and ends just after the pressurizer drains. Differences in core power and pressurizer volume between the AP1000 and the AP600 designs affect some scaling groups. However, no significant distortions were found by comparing the AP1000 Pi groups to those of the SPES and ROSA facilities. Therefore, code validation on the basis of the SPES facility is acceptable.

Intermediate (ADS-1/2/3) Blowdown Phase

The intermediate blowdown phase is considered to be composed of three subphases. Subphase I begins with pressurizer draining and extends to when the hot legs, upper head, and SG reach saturation pressure. Subphase II extends from the end of Subphase I to the initiation of net inflows to the RCS from the accumulators or CMTs. Subphase III extends from the initiation of accumulator injection or CMT draining to the opening of ADS-4. During the intermediate periods, the ADS-1/2/3 system actuates, the PRHR becomes active, and the CMTs begin to drain, as follows:

- Intermediate Subphase I: During this period the most important Pi group for the AP1000 design was found to have better agreement with the SPES facility than those for the AP600 design (There may, however, be some distortion in comparisons of Pi groups with minor importance). Code validation on the basis of the SPES facility data is therefore acceptable.
- Intermediate Subphase II: In general, scaling groups for this period were found to have good agreement between the SPES facility and both the AP1000 and AP600 designs. No significant, non-conservative distortions exist and, thus, the SPES facility is adequate for code validation. With regard to PRHR performance, the AP1000 design exhibited better agreement with the ROSA Pi groups than with the SPES Pi groups. Thus, conclusions regarding simulation of PRHR heat transfer have higher confidence if based on the ROSA facility rather than the SPES facility. However, overall code validation on the basis of the SPES facility data is considered acceptable.
- Intermediate Subphase III: The Pi groups for this period show good agreement between the SPES facility and both AP1000 and AP600 designs. Differences between scaling groups for AP1000 and the SPES facility are either small or conservative. Therefore, code validation on the basis of the SPES facility is acceptable.

ADS-4 Blowdown Phase

The staff's top-down scaling analysis shows that there may be distortions during the ADS-4 blowdown period. Early in this period when the system pressure is high, the flow is critical. Assuming critical flow, the SPES and ROSA facilities are appropriately scaled for the AP1000 standard plant design conditions during ADS-4 blowdown. The APEX facility however, was found to have non-conservative distortions. The analysis considered a 1-inch cold leg break and a double-ended DVI break, and found that the SPES facility is appropriate in both scenarios, but the APEX facility is not appropriate (By contrast, for the AP600 standard plant design, this approach found that the APEX facility is acceptable, but the SPES facility has conservative distortions). Eventually, the system pressure decreases and the ADS-4 flow becomes non-critical. Assuming non-critical flow from the system, the APEX facility becomes appropriately scaled for the AP1000 design based on scaling groups defined in the INEL scaling methodology. On that basis, code validation using the SPES facility is considered acceptable during the high-pressure phase of the ADS-4 blowdown, but the APEX facility is not considered acceptable until late in the period when the IRWST transition is about to occur.

This conclusion conflicts with the applicant's scaling analysis and the conclusion that the APEX facility is appropriately scaled while flow is critical in the ADS-4 to IRWST transition period. The scaling methodology in WCAP-15613 defines dimensionless groups and calculates values showing that the SPES and APEX facilities are correctly scaled. The response to RAI P56 (issued on August 22, 2001 during the AP1000 pre-application review), however, lists "single-loop" scaling groups for the ADS blowdown phase. The applicant cited both the SPES and APEX facilities as having distortions, yet these facilities are considered by the applicant to be acceptable for code validation. The staff concludes, however, that code accuracy and validation in the ADS-4 transition period should be based on the SPES facility simulation. The staff further discussed this issue in Section 21.4 of this report.

IRWST Injection/Drain Phase

Results of the top-down scaling analysis show that the APEX facility is appropriately scaled for the AP1000 standard plant design. Code validation on the basis of that facility is therefore acceptable.

IRWST/Sump Injection Phase

Results of the top-down scaling analysis show that the APEX facility is appropriately scaled for the AP1000 standard plant design. Code validation on the basis of that facility is therefore acceptable.

21.3.8.3 NRC Independent Bottom-Up Scaling Assessment

The staff's independent bottom-up scaling analysis conducted is as follows:

- Froude number comparisons indicate that the SPES facility appropriately scales both the hot and cold leg flow regimes for the high-pressure periods, and the APEX facility appropriately scales these regimes for the low-pressure periods of the SBLOCA and LTC.
- The experimental data in the integral-effects tests are not considered sufficient to validate code models for entrainment and carryover for the AP1000 standard plant design.
- Entrainment in the hot leg and carryover into the branch lines leading to the ADS will occur to a greater extent in the AP1000 design than in the AP600 design or the test facilities. Because of higher steam velocity during ADS-4 blowdown in the AP1000, the applicant has not yet demonstrated that the existing data are sufficient to validate hot leg entrainment models for the AP1000 design because it is basing its scaling evaluation on a correlation that may not be applicable to the AP1000 geometry.

Specifically, the AP1000 hot leg-to-branch line diameter ratio is significantly different than the ratio used in developing the entrainment onset correlation. Alternative evaluation and scaling of entrainment onset leads to the conclusion that entrainment is more prevalent and will occur at lower hot leg water levels in the AP1000 design than in the tests.

- None of the integral-effects tests appropriately scale the facilities entrainment from the pool of water in the upper plenum above the upper core plate for the AP1000 standard plant design. The staff's evaluation of entrainment from the upper plenum pool shows that the rate of entrainment in the AP1000 design will be significantly higher than shown in the integral-effects tests. In Section 21.4 below, the staff further discusses the liquid entrainment issue related to the last two findings above.

21.4 Test Program Scaling Assessment Findings

Given the evaluation discussed above, the staff finds that the AP600 test program is generally applicable for code validation of the AP1000 standard plant design. However, the staff also finds that additional validation is necessary for the liquid entrainment phenomena. The ADS-4 blowdown period to the inception of IRWST injection is important in the AP1000 passive plant design because it is during this period that minimum inventory in the reactor vessel is expected to occur. Compared to the AP600 standard plant design, the AP1000 design has 76 percent higher core power and, therefore, higher steam flow in the upper plenum, hot leg, and ADS discharge during the ADS-4 blowdown. Even though the AP1000 hot leg diameter remains the

same 78.7 cm (31 in) as in the AP600 design, the diameters of the ADS-4 valves and the off-take pipe from the hot leg are increased from 25.4 and 30.5 cm (10 and 12 in) to 35.6 and 45.7 cm (14 and 18 in), respectively. The higher steam flow and larger ADS-4 diameter will affect liquid entrainment through the ADS-4 discharge.

As described in Section 21.3.8.2, the NRC staff's top-down scaling analysis revealed that, during the early phase of the ADS-4 blowdown when the flow is critical, the APEX facility has a non-conservative distortion. The staff therefore requested that the applicant justify the basis for the acceptability of the AP600 code validation for the AP1000 design, or determine whether additional AP1000 testing is necessary for code validation of the ADS-4 blowdown. In its response, the applicant stated that it does not agree with the staff's conclusion that the APEX facility is not suitably scaled for the ADS-4 blowdown phase, and additional hot leg entrainment data for ADS-4 blowdown is not needed for AP1000 code validation. Even though it is expected that higher liquid entrainment may occur in the AP1000 design than in the AP600 design during the ADS-4 blowdown, the applicant contends that this does not render the AP600 code validation unacceptable for the AP1000 during the ADS-IRWST transition phase. Moreover, the applicant contends that the APEX test facility showed significant entrainment during the ADS-4 blowdown phase. The staff does not agree with this finding.

Westinghouse's scaling assessment for the ADS-IRWST transition phase includes both top-down and bottom-up analyses. The overall top-down scaling analysis generates several Pi groups. The facility/plant scaling ratios of these Pi groups show that the APEX and SPES facilities are sufficiently scaled to both the AP600 and the AP1000 designs for choked ADS-4 flow and with respect to core power. When ADS-4 flow is subsonic, the SPES facility is distorted as a result of its oversized ADS-4 vent paths. As discussed in Section 21.3.8.2 of this report, the staff's top-down scaling analysis showed that when the ADS-4 flow is choked, the SPES facility is appropriately scaled for the AP1000 design, but the APEX facility is distorted.

The applicant's responses to RAIs 440.151, 440.152, 440.153, 440.154, 440.155, 440.156, and 440.157 provided additional information on scaling of hot leg entrainment processes, and on the models used by the applicant to predict processes within the hot leg during a SBLOCA and LTC.

Although the applicant's bottom-up scaling analysis of entrainment onset concluded that the APEX facility is well-scaled for the AP1000 standard plant design, the staff finds that the applicant's analysis contains several shortcomings.

- The applicant's scaling analysis is based on an entrainment onset correlation in which the applicability to the AP1000 geometry has not been confirmed. This correlation is derived from experimental data with a small branch line to main pipe diameter ratio (d/D), which may not be appropriate for the AP1000 design because it has a large d/D ratio.
- Existing correlations are based on tests performed with small offtake diameter more than 10 times smaller than the main pipe diameter, as summarized by Ardron and Bryce, 1990. In the AP1000 design, the ADS-4 branch pipe diameter to hot leg diameter ratio (d/D) is large and is considerably larger than that in any supporting test data.
- The general entrainment onset correlation does not account for the effect of viscosity and liquid surface tension, which may affect the liquid entrainment. Correlations that account for these parameters suggest that significant entrainment will occur for the AP1000 design, but will not occur in the tests the applicant has used for code validation.

The staff therefore, does not agree with the applicant's approach for scaling hot leg phase separation, or that the test data in experiments used by the applicant to validate their codes adequately represent the process as it occurs in AP1000. In particular, the applicant has not demonstrated that its codes appropriately account for the high rates of hot leg entrainment observed in Oregon State's ATLATS facility (RAI 440.151), or that the codes have the ability to model all of the flow patterns that may occur in the AP1000 hot leg (RAI 440.155).

The staff, however, has performed audit calculations and sensitivity studies using RELAP5 showing that the impact of poorly modeling hot leg phase separation has only a small effect on the minimum vessel inventory for the AP1000 double-ended guillotine break of the direct vessel injection (DEDVI) line. The staff sensitivity studies took into account hot leg phase separation from the ATLATS facility at Oregon State University. The staff calculations suggest that precise modeling of hot leg phase separation is not a safety significant issue in AP1000, and that the core remains covered even under the conservative assumption of zero phase separation in the hot leg. The applicant however has not yet demonstrated that its codes show the same sensitivity and that this conclusion applies to other small break scenarios.

The staff scaling evaluation also concluded upper plenum pool entrainment to be an issue for the AP1000 design. Experiments in the APEX facility as well as in simulations of the AP600 design showed that the double-ended guillotine break of one of the DVI lines and a 10-inch cold leg break could lead to the minimum vessel inventory or core uncover. Entrainment of liquid from the upper plenum will be significant, and will be more important in the AP1000 design than in the AP600 design. The NRC staff conducted a bottom-up scaling evaluation of upper plenum entrainment. Pool entrainment is a complex process that is highly dependent on the gas velocity bubbling through the pool, and the height to which droplets and other entrained liquid must be elevated to exit the vessel. The AP1000 core power is 76 percent higher than in the AP600 design, but the upper plenum design is nearly identical. The entrainment is often defined as the ratio of the droplet upward mass flux to the gas mass flux:

$$E_{fg} = \frac{\rho_f J_{fe}}{\rho_g J_g}$$

Where ρ_f and ρ_g are liquid and gas phase densities, J_g is the gas superficial velocity, and J_{fe} is the entrained phase superficial velocity.

Expressions for E_{fg} show the functional dependence:

$$E_{fg} \propto (J_g)^n$$

The exponent n is generally 3 or higher. Assuming pressure similitude and preserving the dimensionless height ratio, the AP1000 upper plenum pool entrainment can be expected to be at least 1.76 to 5.45 times as large as that in the AP600 design. Consideration of experimental tests scaled to the AP600 design power levels leads to the conclusion that the AP1000 upper plenum entrainment is significantly higher than entrainment in the integral-effects tests.

To address upper plenum entrainment, the staff issued several RAIs including 440.162, 440.169, 440.170, 440.171, and 440.172. In these RAIs, the applicant was requested to provide additional supporting information on their modeling approach for upper plenum entrainment, and to provide justification that the correlations in their codes were validated by appropriate experimental data. Responses were provided to the RAIs and additional information was supplied in Revision 2 to WCAP-15833.

New information provided by the applicant in response to RAIs 440.164 and 440.171 to resolve the upper plenum entrainment issue raised new concerns on core modeling and the calculation of level swell.

With regards to upper plenum entrainment scaling, the staff finds that none of the integral-effects test facilities are sufficiently well-scaled to provide an acceptable database to validate thermal-hydraulic codes for the high rates of liquid entrainment that are expected to occur in the AP1000 design during ADS-4 and IRWST injection periods of an SBLOCA. The staff does not agree with Westinghouse's assertion that they have provided adequate scaling evaluation for the AP1000 upper plenum entrainment phenomena.

The applicant has also not demonstrated that the existing AP600 integral-effects tests provide data over the range of conditions necessary to validate entrainment models in the codes that the applicant intends to use. The staff concludes that the applicant must either obtain entrainment test data applicable to the AP1000 steam flow rates for code validation, or provide proper justification for the entrainment models to be used for the AP1000 applications.

21.5 Summary and Conclusions of AP1000 Testing and Scaling

Requests for additional information were formulated and sent to the applicant in September 2002, and responses were supplied by the applicant in a series of transmittals. While most of these RAIs have been answered satisfactorily or can be resolved with minor corrections, there are several important issues that require resolution. There are three issues that are of particular note: upper plenum pool entrainment; phase separation in the hot leg; and level swell in the core. Each of these processes are important during a SBLOCA.

The following discussion summarizes each of these issues:

Hot Leg Phase Separation

Hot leg phase separation refers to the thermal-hydraulic processes that occur near the ADS-4 branch line connection that act to entrain liquid in the hot leg and carry that liquid over into the ADS-4 system. As discussed in Sections 21.3.8.3 and 21.4 of this report, the staff concluded that this process was not well scaled in the tests used to validate AP1000 safety analysis codes, and that the higher steam velocities in AP1000 would result in much higher liquid carry over into the ADS-4 than in the AP600. This could cause a relatively large two-phase pressure drop in AP1000 and delay the start of IRWST injection.

Correlations currently used in the thermal-hydraulic codes, and used to scale the hot leg phase separation process for AP1000 assumes that the flow pattern is horizontal-stratified. However, recent experimental information obtained from the ATLATS facility at Oregon State University showed that the hot leg flow pattern is not horizontal-stratified when most entrainment occurs. Rather, an oscillating slug of liquid forms between the branch line and SG inlet plenum and slug behavior dominates the entrainment process.

The applicant's submittals and responses to RAIs concerning hot leg phase separation were not sufficient to demonstrate that the codes used in AP1000 safety analysis model the hot leg phase separation process correctly. However, the sensitivity studies by the NRC staff to investigate the effect of modeling this process on important AP1000 transients indicated the effect to be relatively small. This issue is considered open until the applicant confirms the sensitivity studies performed by the staff using the code(s) the applicant intends to use to model SBLOCAs in AP1000. The confirmatory analyses should range hot leg entrainment consistent

with ATLATS data and show that the uncertainty in modeling hot leg phase separation does not represent a significant safety issue in AP1000. Therefore, this is Open Item 21.5-1.

Upper Plenum Pool Entrainment

Upper plenum pool entrainment refers to the thermal-hydraulic processes that carry liquid out of the upper plenum and into the hot leg, where it is likely to be swept into the ADS-4 piping. The process is of importance in transients such as a DEDVI line or an inadvertent ADS actuation. In these transients, core uncover and cladding heat up are prevented by the liquid level above the top of the active fuel. With the higher power and steam production in AP1000, carry over from the vessel may increase and lead to core uncover.

As discussed in Section 21.4 of this report, the staff, based on its examination of scaling and test data, concluded that scaled entrainment rates in the upper plenums of the tests used to assess thermal-hydraulic codes for AP600 were too low to be used for that purpose in AP1000. In order to justify the models, correlations, and methods to predict upper plenum entrainment in AP1000, the applicant was requested to provide additional information. This was accomplished by the submittal of WCAP-15833, and by the responses to several RAIs. The applicant has committed to update WCAP-15833 to include the final RAI responses. Therefore, this is Confirmatory Item 21.5-1.

The applicant's submittals however, did not provide sufficient justification that the models and correlations in NOTRUMP or WCOBRA/TRAC have been adequately assessed to cover the ranges expected to occur in the upper plenum of the AP1000. While correlations exist to model upper plenum entrainment phenomena, the issue that remains is adequacy of the database. Existing correlations are based on relatively small diameter vessels, low gas flow rates, and for some data, air-water as opposed to steam-water. Because of the small vessel size in these data, conditions were essentially one-dimensional. Flow in the upper plenum of the AP1000 is expected to be non-uniform and three dimensional. Thus, a suitable database for assessing entrainment correlations in the upper plenum has not been established.

Given the lack of well scaled experimental data on upper plenum entrainment phenomena and the importance of predicting this process in an advanced plant SBLOCA transient, it is recommended that new experimental data be obtained to support the use of the upper plenum entrainment models in the AP1000. This data was requested by the NRC staff in letter dated March 18, 2003, from J. Lyons. Therefore, this is Open Item 21.5-2.

Core Level Swell

Level swell refers to the effect of thermal-hydraulic processes such as two-phase interfacial drag, interfacial area generation and flow pattern transitions that cause a two-phase mixture level to exceed the collapsed water level in the core. In AP1000, prediction of level swell is important in demonstrating that cladding does not undergo a significant heat up during SBLOCAs.

Information supplied by the applicant as part of the response to RAIs 440.164 and 440.171 suggests that level swell may not be adequately predicted for AP1000 and that the codes may not be predicting cladding heatup because of insufficient core nodalization and inadequate correlations used in predicting the level swell.

At a meeting of the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Thermal/Hydraulics on March 19 and 20, 2003, the subcommittee raised concerns on the high

void fractions within the core calculated by NOTRUMP, WCOBRA/TRAC-AP, and RELAP5 during recovery from SBLOCA. The applicant responded that they had also predicted high void fractions in correlating test data. The subcommittee requested that the applicant provide additional justification that the AP1000 will remain covered as predicted by the codes by comparing the collapsed liquid levels predicted by the codes to that measured in tests. This is Open Item 21.5-3.

21.6 Assessment of Applicability of the AP600 Analysis Codes to the AP1000

For the AP600 design certification, the safety analysis of the design basis transients and accidents were performed with the following computer codes:

- LOFTRAN for non-LOCA transients and SGTR analyses
- NOTRUMP for SBLOCA analyses
- WCOBRA/TRAC for LBLOCA and LTC analyses
- WGOETHIC for Containment Integrity Analysis

These safety analysis codes were validated with the test data from the AP600 test programs. WCAP-15644 documents the applicant's assessment of these safety analysis to determine their applicability and use for the AP1000 design certification. The staff's assessment of their applicability to the AP1000 is addressed in this section.

21.6.1 LOFTRAN Code Applicability

LOFTRAN simulates a multi-loop reactor system by modeling the reactor core and vessel, hot and cold leg piping, SG tube and shell sides, pressurizer, and RCPs, with up to four coolant loops. The pressurizer model includes the effects of pressurizer heaters, spray and relief and safety valve operation. The reactor core model employs a lumped fuel heat transfer model with point neutron kinetics and includes the reactivity effects of variations in moderator density, fuel temperature (Doppler), boron concentration, and control rod insertion and withdrawal. The secondary side of the model uses a homogenous, saturated mixture for thermal transients and a water level correlation for indication and control. The safety injection system including the accumulators is also modeled including the effects of pump coastdown and pump startup. Flow reversal in the reactor coolant loops is allowed except in the loop with the pressurizer, where flow reversal is not allowed. The LOFTRAN thermal-hydraulic model is best suited for use in transients in which the primary coolant system remains subcooled. LOFTRAN may be used for a main steamline break analysis where two phase conditions occur in the upper reactor vessel head. The upper head is a hydraulically stagnant region which receives only a small fraction of the main coolant flow. For accident conditions when the extent of voiding extends beyond the pressurizer and the upper head, the use of LOFTRAN would not be appropriate without additional justification.

LOFTRAN does not have a detailed core heat transfer model. An overall fuel rod to coolant heat transfer coefficient is utilized which is a parabolic fit to values specified by the user. Input values either maximize or minimize core heat transfer depending on the conservative direction for the transient of interest. The inputs are obtained from the limiting values predicted using detailed Westinghouse fuel rod design codes. For evaluations where accurate knowledge of core heat transfer or fuel temperature is important, physical conditions are transferred from LOFTRAN to more detailed thermal/hydraulic codes such as THINC, FACTRAN, and VIPRE.

The NRC staff found LOFTRAN (WCAP-7907-P-A, "LOFTRAN Code Description," issued April 1984) to be acceptable for analysis of transients and accidents at operating plants as

presented in Chapter 15 of the plant safety analysis reports. Chapter 15 safety analysis is discussed in NUREG-0800, "Standard Review Plan." This approval did not extend to LOCA or SGTR.

In order to model the SGTR, the applicant modified the LOFTRAN code to include an enhanced SG secondary-side model, a tube rupture break flow model, and improvements to allow simulation of operator actions. This version of the code is sometimes referred to as LOFTTR2. The SGTR version of LOFTRAN was reviewed and approved by the NRC staff and is discussed in Westinghouse Topical Reports WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," issued August 1985; WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1, issued March 1986; and WCAP-11002, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident," issued February 1986.

21.6.1.1 Application of LOFTRAN to Passive Plants

Additional modifications were made to LOFTRAN to model the AP600. These are described in Westinghouse Topical Reports WCAP-14234, "LOFTRAN & LOFTTR2 AP600 Code Applicability Document, Rev. 1, issued August 1997, and WCAP-14307, "AP600 LOFTRAN-AP and LOFTTR2-AP Final Verification and Validation Report," Revision 1, issued August 1997. With consideration of possible failure in the AP600 passive safeguard systems, the list of transients and accidents for which LOFTRAN has been approved is found in Table 2. This approval is discussed in the NRC staff final safety evaluation report for the AP600 (NUREG-1512).

WCAP-15612 and WCAP-15613 describe the use of the LOFTRAN code for the AP1000 evaluations. WCAP-15612 contains a general description of the AP1000 standard plant design and preliminary analyses of a subset of the transients and accidents listed in Table 2. The subset was selected by the applicant to illustrate performance of the AP1000 passive safety features and plant differences between the AP600 and the AP1000 designs. WCAP-15613 presented additional details and justifications for use of the LOFTRAN code for AP1000 analyses. WCAP-15612 gives a revised PIRT for the AP1000 transients and accidents, with a scaling assessment of the tests used to qualify the LOFTRAN code.

Since the AP1000 is similar in design to the AP600, the applicant believes modifications made to the code to model the AP600 will also address the AP1000. The differences in size between the two designs can be accounted for in the code inputs. The fuel, pressurizer, and SGs for the two passive plant designs will be similar to those used in operating plants. Unlike operating plants where the hot and cold leg nozzles are at the same elevation on the reactor vessel, for the AP600 and the AP1000, the hot and cold leg nozzles are at different elevations. The elevation difference is accounted for by modifications within LOFTRAN. The RCPs for the passive plants are of a canned rotor design with their own characteristics for developed head and torque as functions of flow rate and impeller speed. The mass of the pump flywheel was increased for the AP1000 to provide for a longer flow coastdown should an RCP inadvertently trip during operation. The pump characteristics are accounted for by inputting the proper information from the pump manufacturer. Since full pump characteristics can be input into LOFTRAN, the code is able to model the RCPs of the AP1000 when properly described in the input.

Operating Westinghouse plants use a single cold leg and RCP per coolant loop, whereas the AP600 and the AP1000 use two cold legs and two RCPs per loop. LOFTRAN is capable of evaluating the dual cold leg loop arrangement so long as the two cold legs in a loop have the

same behavior so that they can be lumped together. For conditions when the two cold legs do not behave the same, such as for a tripped RCP or locked rotor/sheared RCP shaft, the applicant inputs the net cold leg flow as a function of time. The net cold leg flow rate is calculated by external methods. The applicant presented the asymmetrical cold leg methodology to the NRC staff for review as part of the review of LOFTRAN for the AP600. The NRC staff concluded that the external flow calculational methodology is acceptable. This same methodology of calculating asymmetric cold leg flow rates outside of the LOFTRAN code will be used for analyses for the AP1000. The methodology remains acceptable.

The applicant also modified the LOFTRAN code by adding the capability to model the following additional components, which are part of the AP600 and AP1000 designs but are not present in operating plants:

- automatic depressurization system;
- core makeup tank;
- passive residual heat removal heat exchanger; and
- in-containment refueling water storage tank.

As part of the AP600 review the staff evaluated the ability of LOFTRAN to predict the performance of these components during transients and accidents without exceeding the capabilities of the code. Much of this review applies to the AP1000 design.

The applicant will not analyze events that would cause the ADS to open using LOFTRAN with the exception of inadvertent opening of a single valve. The scope of this analysis is limited to the initial few seconds so that core heat transfer can be evaluated. The analysis is terminated before significant steam voiding can occur in the reactor system.

Similarly the applicant does not analyze conditions for which two phase voiding will occur in the CMTs. The CMTs actuate during transients analyzed by LOFTRAN but steam formation in the CMT inlet lines does not occur so that the CMTs will not drain in LOFTRAN analyses. Actuation of the CMTs creates a circulation path so that the cooler borated water from the CMTs mixes with the reactor coolant. This provides for additional core cooling and boron addition to shut down the core even before the CMTs begin to drain.

Of the events listed in Table 2, steam voiding in the reactor system and CMT draining would be most likely following an MSLB. This is because the accompanying reduction in steam pressure would cause a rapid increase in the rate of heat removal from the reactor system. Cooling of the reactor system would cause a pressure reduction which might cause the CMTs to drain. A large pressure reduction in the reactor system would produce an "S" signal which would cause the RCPs to trip. The staff was initially concerned that loss of forced circulation might result in elevated local temperatures causing steam formation in the CMT pressure balance lines and intact SG tubes. The SGs for the AP1000 are significantly larger than those of the AP600 and have the potential for more pressure reduction in the reactor system than do those of the AP600. Thus voiding might be produced within the AP1000 following an MSLB while none was predicted for the AP600. Flow restrictions in the SG nozzles are the same for the AP600 and the AP1000 so the rate of reactor system pressure reduction would be approximately the same. The total water mass is greater for the AP1000 SGs so that the total pressure reduction might be greater for the AP1000. The NRC staff reserved judgment of the ability of LOFTRAN to adequately model an MSLB for AP1000 pending analyses by the applicant that significant steam would not form within the reactor system following an MSLB. These analyses were performed by the applicant as described in Section 15.1.5 of the DCD. The CMTs were

determined not to drain and the reactor coolant loops remained subcooled. Therefore, the NRC staff concludes LOFTRAN to be acceptable for MSLB analysis for AP1000.

The PRHRHX provides a passive means of decay heat removal that can be effective at all reactor system pressures. As for the AP600 the PRHRHX for the AP1000 is located within the IRWST and transfers heat from the RCS to the IRWST for conditions when the normal means to reactor heat removal might be lost. For both designs, the PRHRHX tube bundle is C-shaped and makes a single pass within the IRWST. The IRWST is described as a one lumped parameter region in LOFTRAN. Local heating of the IRWST water in the vicinity of the PRHRHX is not included in the analysis. The NRC staff originally had concerns in the AP600 review that higher local temperatures might cause the actual PRHR heat flow to be lower than that predicted by LOFTRAN. The applicant demonstrated that the PRHRHX models in LOFTRAN were adequate for the AP600 by modifying the code so that the code predications compared well with the results from scale model tests.

The PRHRHX for the AP1000 standard plant design is essentially the same design as for the AP600. The heat transfer area has been increased by 22 percent and the flow resistance in the inlet and outlet piping has been decreased so that the design heat flow rate is increased by 72 percent. The average heat flux for the AP1000 is expected to be 41 percent higher than for the AP600. The applicant calculates heat transfer through the PRHRHX in LOFTRAN using standard convective heat transfer correlations for the water flowing on the inside of the PRHRHX tubes. These correlations were found to be valid for a wide range to test conditions including those expected for the AP1000. On the secondary side of the PRHRHX, the most significant mode of heat transfer will be nucleate boiling. The applicant had previously found standard nucleate boiling correlations to over predict the heat flow from the PRHR test facility in Westinghouse Topical Report WCAP-12980, "AP600 Passive Residual Heat Removal Heat Exchanger Final Report," Revision 3, issued April 1997. The nucleate boiling heat transfer correlation used in LOFTRAN was modified to provide a best fit to the data. Further verification for the derived nucleate boiling correlation was obtained by correlation of PRHRHX data from the ROSA facility and the SPES facility experiments that were performed for the AP600. The NRC staff believes that the PRHRHX model in LOFTRAN will be valid for the AP1000. Individual analysis for the non-LOCA transients evaluated by LOFTRAN are performed taking into account the uncertainty in PRHR heat transfer. The uncertainties were determined from the scatter in the test data and are included so as to be most conservative for the transient analyzed.

21.6.1.2 LOFTRAN Code Conclusions

Based on the foregoing considerations, the NRC staff concluded that the use of LOFTRAN as described in WCAP-15612, WCAP-15613, and WCAP-15644 is acceptable for licensing calculations of the AP1000 subject to the following limitation:

- The transients and accidents that the applicant proposes to analyze with LOFTRAN are listed in Table 2 of this report, and the staff review of LOFTRAN usage by the applicant was limited to this set. Use of the code for other analytical purposes will require additional justification.

Following review of the DCD the staff concluded that the analyses which the applicant has performed for AP1000 using LOFTRAN are encompassed in Table 2, and therefore, use of LOFTRAN for AP1000 is acceptable.

21.6.2 NOTRUMP Code Applicability

NOTRUMP was first submitted for NRC review in November 1982. The code was developed to better address the thermal-hydraulic aspects of a postulated SBLOCA, which had become an issue following the accident at Three Mile Island. Following a review by the NRC staff, NOTRUMP was found to be acceptable for the analysis of SBLOCA events for Westinghouse reactor designs (WCAP-10079-P-A, "NOTRUMP A Nodal Transient Small-Break and General Network Code," issued August 1985). For NOTRUMP evaluations, an SBLOCA is considered to be a rupture in the RCP boundary with a total cross-sectional area less than 0.09 m^2 (1.0 ft^2) for which the normal charging system flow is not sufficient to maintain pressurizer level and pressure. The NRC staff has also approved the use of NOTRUMP for SBLOCA evaluation for plants designed by Combustion Engineering (WCAP-10054-P-A, "Addendum to the Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code for the Combustion Engineering NSSS," Addendum 1, issued March 1987).

NOTRUMP models one-dimensional thermal hydraulics using control volumes interconnected by flow paths (links). The spacial and time dependant solution is governed by the integral forms of the conservation equations in the control volumes and flow links. The thermal hydraulics account for non-equilibrium conditions and apply drift flux models for calculating relative velocities between the steam and liquid phases. Reactivity feedback is modeled with point kinetics neutronics. The code incorporates special models to calculate responses of the RCPs, steam separators, and the core fuel pins. A significant code feature is a node stacking capability for calculating a single mixture height in a subdivided vertical region. A two-phase horizontal stratified flow model is also included.

21.6.2.1 Summary of AP600 Evaluations of the Use of the NOTRUMP Code

With the AP600, the applicant introduced new systems and protective features for which NOTRUMP had not been previously evaluated. These include the ADS, CMTs, PRHRHX, and IRWST. The applicant investigated the capability of NOTRUMP to evaluate the AP600 systems as discussed in Westinghouse Topical Report WCAP-14206, "Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analysis," issued November 1994. The existing code was determined to be adequate regarding most features of the AP600, however, several modifications were required. A summary of these modifications follows:

- SIMARC (Simulator Advanced Real-time Code) drift flux methodology implementation
- General drift flux model modifications
- Modified Yeh drift flux correlation for use with the SIMARC drift flux method
- Inclusion of general droplet flow correlation when void fractions are between 0.95 and 1.0 when using the improved TRAC-PF1 flow regime map
- Modification of the bubbly and slug flow distribution parameter (C_0)
- Use of a net volumetric flow-based momentum equation
- Implementation of the EPRI/flooding vertical drift flux model
- Modifications to allow over-riding of the default NOTRUMP contact coefficient terms for formation of regions
- Implementation of internally calculated liquid reflux flow links
- Implementation of a mixture level overshoot model
- Modified bubble rise/droplet fall model logic
- Activation of the simplified pump model
- Implicit fluid node gravitational head model implementation
- Horizontal leveling model implementation
- Revised unchoking model implementation

- Implementation of a revised condensation heat link model
- Implementation of the Zuber critical heat flux model
- Revised two-phase friction multiplier logic
- Addition of the Henry-Fauske/HEM critical flow correlation
- Improved fluid node staking model logic
- Revised iteration method for transition boiling correlation in metal node heat links

The description of these modifications along with code verification by comparison of calculational predictions to test data is discussed in Westinghouse Topical Report WCAP-14807, "NOTRUMP Final Validation Report for AP600," Revision 5, issued August 1998. Both integral system tests utilizing simulated reactor systems and separate effects tests modeling individual components were utilized. The NRC reviewed and approved the application of NOTRUMP for the analysis of an SBLOCA for the AP600 passive reactor design in NUREG-1512. The approval was made with the following conditions. These same conditions apply to the AP1000.

- The applicant did not predict core uncover for any design bases SBLOCA event for the AP600 so that transition boiling or film boiling was not calculated to occur in the core. The staff, therefore, did not review the changes in the numerical solution techniques used in the NOTRUMP heat links to evaluate this condition. The staff concluded that this methodology may not be invoked in the application of the NOTRUMP code to the 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 noted that NOTRUMP 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 PRHRHX 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. If scenarios are found which cause non-condensable gases to reach the PRHRHX while it is actively removing heat from the primary system, NOTRUMP cannot be used to analyze those scenarios. The applicant removes consideration of PRHRHX heat flow prior to the ADS-4 actuation, which should prevent non-condensable gas from the accumulators from reaching the PRHR while it is included in the NOTRUMP model.
- NOTRUMP does not model the momentum flux terms in the conservation of momentum equation dealing with the effects of area and density change. The applicant performed an evaluation of the effect of the omitted momentum flux terms and concluded that they were of little significance with the exception of flow in the ADS-4 after reactor system pressure had decreased so that the flow velocity was no longer sonic. This deficiency and deficiencies in the ability of the code to calculate pressurizer drainage and reactor vessel downcomer level were accounted for by an imposed reduction in the IRWST level. The level reduction conservatively delays the time of IRWST injection and produces a net reduction in available volume of IRWST water. By comparison with data from the APEX facility, adjusted to account for scale, an IRWST level reduction of 0.91 m (3 ft) was determined to be appropriate. For added conservatism the applicant used an IRWST level reduction penalty of 1.83 m (6 ft).
- The NRC staff questioned the ability of the code to adequately predict liquid entrainment in branch lines. The most significant example occurs during ADS-4 operation. Flow through the ADS-4 valves exits from the reactor system hot legs from tees located at the

top of the hot leg piping. NOTRUMP assumes entrainment will occur when the mixture level in the hot legs reaches a preset elevation. That elevation is independent of the ADS-4 flow velocity. For very high ADS-4 fluid velocities, NOTRUMP may underpredict the amount of liquid entrained in the ADS lines. The resistance to vapor flow through an ADS-4 inlet line is reduced without entrained liquid. This may result in vapor flow rates through the ADS-4 that are too high resulting in an excessively high rate of reactor system depressurization. For the AP600 this effect was accounted for with the IRWST level penalty based on comparisons with the APEX facility test data.

21.6.2.2 Evaluations of the NOTRUMP Code for the AP1000

Use of NOTRUMP for the AP1000 evaluations is described in WCAP-15612 and WCAP-15644. WCAP-15612 contains a general description of the AP1000 and preliminary analyses of SBLOCAs for three postulated locations. Additional details and justifications for use of NOTRUMP for the AP1000 SBLOCA analysis are included in WCAP-15644.

In WCAP-15613 the applicant provided a PIRT for an SBLOCA at the AP1000 and an AP1000 scaling assessment of the tests that were used to qualify NOTRUMP for the AP600. The applicant concludes that NOTRUMP is qualified to perform an SBLOCA for the AP1000 with no further modifications.

The AP1000 design is essentially a larger version of the AP600 design. Changes include increasing the core power, core power density, and the capacity of the passive safety systems. The design modifications that are of primary significance for modeling an SBLOCA include increasing the size of the CMTs, IRWST, ADS-4 valves, and PRHRHX. The core length was extended to 4.27 m (14 ft) and the thermal power output was increased by approximately 76 percent. The average linear power was increased from 4.10 to 5.707 kw/ft. The design of the RCPs has been modified and the SGs are larger. The RCPs are tripped on a safety injection signal and neither the primary coolant pumps nor the SGs are expected to have a significant influence on the course of an SBLOCA event at the AP1000.

The accumulators are the same size for the AP1000 as for the AP600. The CMTs are 25 percent larger than for the AP600. Depressurization by the first 3 stages of ADS will be slower for the AP1000 since the plant is larger and the valves for the first 3 stages of ADS are the same size as for the AP600.

For the design power increase of 76 percent, more reliance will be placed on the ADS-4 to depressurize the plant so that injection from the IRWST can begin and refill the core. The ADS-4 total vent area is increased by 76 percent. The resistance to flow in the inlet lines from the hot legs to the ADS-4 valves is decreased so that the total ADS-4 relief capacity is increased by 89 percent. Instead of the IRWST level penalty that was used in the analysis of the AP600 to account for deficiencies in the NOTRUMP ADS-4 model, the applicant uses an increased resistance model in NOTRUMP for the AP1000 evaluations. Specifically the flow resistance in the ADS-4 flow paths is increased by 70 percent. The amount of increase was derived from comparisons with a stand-alone momentum flux model and comparisons with predictions from the more detailed special version of the WCOBRA/TRAC computer code which has been modified to provide for a better accounting of liquid entrainment in the reactor vessel, hot legs and ADS-4. This code version is designated as "WCOBRA/TRAC-AP." The code accounts for momentum flux so that compensating fixes in the input as is done with NOTRUMP are not required when using WCOBRA/TRAC-AP. The applicant provided comparisons of NOTRUMP predictions with those from WCOBRA/TRAC-AP in WCAP-15833-P, Rev. 2, for the period in SBLOCA analysis for AP1000 between actuation of the ADS-4 to the beginning of

IRWST injection. The applicant concludes that the comparisons demonstrate that NOTRUMP can adequately simulate the overall core cooling behavior during this period.

The applicant did not perform separate stand alone tests for ADS-4, but concludes that data taken at the APEX facility for AP600 is sufficient to benchmark the WCOBRA/TRAC-AP computer code for ADS-4 flow rates and entrained liquid fractions for AP1000. The NRC staff is concerned that higher steam velocities that would occur in the AP1000 upper plenum and into the horizontally stratified hot legs would make the applicant's scaling conclusions no longer valid. See Section 21.4 of this report. In addition the staff is concerned that the models used in WCOBRA/TRAC-AP to describe the entrained liquid fraction in the upper plenum and hot legs have not been derived from data that describes the geometry of AP1000. These concerns are also discussed in Section 21.5.

The PRHRHX is essentially the same design as for the AP600. The heat transfer area has been increased by 22 percent and the flow resistance in the inlet and outlet piping has been decreased so that the design heat flow rate is increased by 72 percent. The average heat flux for the AP1000 is expected to be 41 percent higher than for the AP600. In the AP600 review, the staff accepted the PRHRHX model because the heat transfer calculated by NOTRUMP for the SPES facility and the APEX facility experiments was lower than that measured in the experiments. PRHR heat transfer is given a medium importance in the PIRT (WCAP-15613). PRHR heat transfer is of greater importance for very small breaks since for larger breaks most of the reactor decay heat would be removed by the break.

In WCAP-14807, Rev. 5, the applicant concluded that the NOTRUMP PRHR model contains a deficiency that needs to be monitored to assure that excess PRHR heat transfer is not calculated. This is because the NOTRUMP code does not model the thermal plume in the IRWST that would occur from extended operation of the PRHRHX. For the AP1000, heat flux from the PRHRHX will be greater than for the AP600 and, therefore, more likely to produce a thermal plume in the IRWST. To account for possible nonconservatism in the PRHR-HX model, the applicant has reduced the surface area by 50 percent for all analyses using NOTRUMP for AP1000. The staff requested that the applicant provide justification that the revised PRHR-HX model is conservative. This information was provided in the response to RAI 440.054. The applicant provided a comparison of PRHR-HX heat removal for the NOTRUMP model with the model in LOFTRAN which the staff has approved for AP1000 analysis based on comparisons with test data. See Section 21.6.1.1 of this report. The results of the NOTRUMP - LOFTRAN comparison was that the NOTRUMP model was not conservative by about 6 percent for total PRHR-HX heat flow. When the 50 percent reduction factor was applied to the NOTRUMP heat transfer area, NOTRUMP was found to be conservative by about 12 percent. The staff concludes that the PRHR-HX model in NOTRUMP is acceptable for AP1000 analysis with the 50 percent area reduction.

The CMTs for the AP1000 are about 25 percent larger than for the AP600 but they are expected to perform in a similar fashion. Following a LOCA, the CMT outlet valve will open to provide makeup water to the reactor core. Opening of the CMT outlet valve will cause relatively cool borated water to circulate from the CMTs into the reactor vessel. As the reactor system becomes voided the CMTs will drain and provide cooling for the reactor core. For the AP600 review, the applicant performed comparisons of NOTRUMP predictions with data from two series of stand alone tests. The tests were designed to model CMT behavior in both the circulation and the drainage modes. The NOTRUMP code was found to predict the injected fluid to be at a higher temperature than the test data. The predicted time of the injection was usually delayed. These modeling results are conservative and the NRC staff finds the model acceptable for the AP1000.

21.6.2.3 NRC Staff Audit Calculations

The NRC staff contracted to have RELAP5 input developed for the AP1000. This work was documented in Information Systems Laboratories, Inc (ISL) ISL-NSAD-NRC-01-003, "Preliminary Results of the AP1000 RELAP5/MOD3.3 Analysis for the Two-Inch Cold Leg and Main Steamline Breaks," issued August 2001. The model was developed from an existing RELAP5 input model for the AP600 and modified to describe the AP1000 using plant data supplied by the applicant. The NRC staff uses RELAP5 as an aid in understanding and evaluating the sequences and phenomena in postulated reactor accidents. RELAP5 is not a design basis licensing tool. Conclusions on the acceptability or unacceptability of an SBLOCA for the AP1000 are based on the applicant's calculations using NOTRUMP and other Westinghouse methodology and not on results from RELAP5.

Conservative assumptions were built into the RELAP5 input so as to be consistent with those made by the applicant in running NOTRUMP:

- The analyses were initiated from 102 percent of full power.
- The failure of one of the four ADS-4 valves was assumed.
- Decay heat was set at 20 percent greater than American Nuclear Society (ANS) Standard ANS 5.1-1973, "Decay Energy Release Rates Following Shutdown of Uranium-Cooled Thermal Reactors."
- The containment pressure was set to atmospheric.

The core model in the RELAP5 analyses was somewhat more detailed than in the NOTRUMP analyses in that a hot rod was modeled with a higher heat flux than the average core. The increased heat flux of the hot rod allows for the possibility of fuel cladding heatup following departure from nucleate boiling to be assessed.

The staff performed the following analyses for comparison with the NOTRUMP results presented in Chapter 15 of the DCD:

- The inadvertent opening of both 4-inch ADS-1 valves;
- A cold leg break of 2 inches equivalent diameter in the loop without the PRHR;
- The double-ended rupture of a direct vessel injection line; and
- A cold leg break of 10 inches equivalent diameter.

None of the breaks analyzed by the staff resulted in core uncover or cladding heatup. RELAP5 calculated approximately the same minimum core water mass for all break sizes. However slightly less core water mass was predicted for the double ended DVI line break than for the other breaks.

The NRC staff audit calculation originally predicted a small amount of core uncover following a double ended DVI line rupture after actuation of ADS-4. The amount of core uncover calculated by RELAP5 was minimal and was at a time when considerable cooling of the reactor core had already occurred. This analysis was later repeated with the benefit of additional data from the applicant describing the flow limiting devices in the DVI nozzles and in the CMT discharge lines. The applicant provided other data describing the revised reactor vessel

internal structure following removal of the thermal shield which was originally to surround the core. The staff repeated its SBLOCA analyses using the revised input. The revised analysis no longer predicts that the core will become uncovered following any SBLOCA.

Table 3 shows a comparison of the sequence of events for a postulated design basis double ended DVI line break as calculated by the RELAP5 and NOTRUMP codes. The results show a faster reactor system depressurization for NOTRUMP until the time of ADS-4 actuation. After ADS-4 actuation RELAP5 and NOTRUMP predict approximately the same reactor system depressurization rate indicating that the increased loss coefficient, which the applicant adds to the ADS-4 line resistance in NOTRUMP to account for the absence of a momentum flux model is equivalent to the momentum flux model in RELAP5. The more rapid depressurization in NOTRUMP that occurs before ADS-4 actuation may result from the more conservative model for subcooled break flow in NOTRUMP. NOTRUMP predicts earlier drainage for the CMT connected to the broken DVI line which causes earlier activation of ADS 1/2/3. The staff has not determined the cause of the earlier IRWST activation in the RELAP5 analysis. As a sensitivity study the staff delayed IRWST injection until the time predicted by NOTRUMP. Even with delayed injection RELAP5 did not predict core uncover. The intact CMT continued to inject during this period.

In addition to comparisons for design basis SBLOCA, the staff also made comparisons of RELAP5 predictions with NOTRUMP as part of the AP1000 PRA review. These analyses involved the assumption of many simultaneous failures in the passive safety systems. The analyses included the simultaneous failure of both accumulators, one of the two CMTs and, all ADS 1/2/3. These failures are in addition to the single failure of one of 4 ADS-4 that is assumed for the design basis. For the multiple failure analyses both NOTRUMP and RELAP5 predicted a limited amount of core uncover. The amount of core uncover was insufficient however, to cause core heating beyond accepted limits. These results demonstrate the robust design of the AP1000 for SBLOCA mitigation since even with many multiple failures excessive core heating was not predicted to occur.

The accuracy of a computer code in predicting core uncover can be demonstrated by the code's ability to correlate experimental test data. Comparison with experimental data indicate that the interphase drag between the steam and water in the core calculated by RELAP5 is too large at low pressures resulting in overprediction of level swell and enhanced entrainment and loss of inventory from the top of the test section. For FLECHT-SEASET Boil Off Test 35658, RELAP5 was shown to predict core dryout earlier than the data and to lose coolant out of the top of the test bundle at a faster rate. Thus RELAP5 would be expected to predict less water in the core for AP1000 than would actually be the case (see NUREG/CR-5535, "RELAP5/Mod3.3 Code Manual Volume III: Development Assessment Problems," Revision 1, issued December 2001).

The applicant successfully correlated experimental test data using NOTRUMP for core uncover at low pressures and for conditions that would be present in the AP1000 following SBLOCA. These were data from the Westinghouse G2 series of tests and the ACHILLES tests in England. The G2 tests utilized a full size simulation of a Westinghouse 14 foot fuel bundle. The comparisons are in Chapter 4 of WCAP-14807, Rev. 5, and in the applicant response to RAI P69 (issued on September 17, 2001 during the AP1000 pre-application review). The predictions by NOTRUMP were conservative (more core uncover) in comparison to the test data.

As discussed in Section 21.5 of this report, the staff identified Open Item 21.5-3 which requested that the applicant provide additional justification that the AP1000 will remain covered

as predicted by the codes by comparing the collapsed liquid levels predicted by the codes to that measured in tests. Therefore, prediction of level swell is an open item in the AP1000 review.

21.6.2.4 NOTRUMP Code Conclusions

The NRC staff is continuing review of the NOTRUMP code for analysis of an SBLOCA for the AP1000. In the preceding discussions many code features have been found acceptable for the AP1000 analysis. Final code approval will be dependant on the resolution of the remaining open issues:

- The ability of NOTRUMP to adequately predict liquid entrainment from the upper plenum to the hot legs and to the ADS-4 is a concern to the staff since the amount of entrainment will affect the ability of the ADS-4 to depressurize the reactor and affect reactor vessel liquid inventory. The applicant has not yet provided an adequate basis for the staff to conclude that the entrainment predictions of NOTRUMP are conservative for SBLOCA analyses. Additional details of the staff's review of the entrainment issue are in Section 21.4 of this report.
- The ability of NOTRUMP and WCOBRA/TRAC-AP to conservatively predict level swell in the core during recovery from an SBLOCA was questioned by the ACRS Thermal/Hydraulic Subcommittee. The applicant was required to provide additional justification that the amount of water predicted by the codes to cover the core with a two phase level is consistent with that measured in tests. The applicant is currently evaluating available test data to address the subcommittee's concern.

The staff continues to investigate differences in RELAP5 and NOTRUMP predictions for SBLOCA. Both codes show that the core will remain covered. The staff has performed sufficient analyses to bound the differences in code predictions. To ensure that the phenomena are well understood, however, the staff and the applicant will continue to investigate code differences.

21.6.3 WCOBRA/TRAC Description and Qualification for LBLOCA Analyses in AP1000

WCOBRA/TRAC is a realistic (best estimate) analysis code designed to simulate LBLOCA accidents. The results of the analysis can be compared to the acceptance criteria in 10 CFR 50.46, based on the guidelines described in RG 1.157 and the code scaling, applicability and uncertainty methodology. The code has been reviewed and accepted in NUREG-1512 for application to AP600. The review criteria are defined in 10 CFR 50.46(a)(1)(i) which requires: (1) the analytical technique realistically describe the LOCA behavior; (2) that results be compared to applicable experimental data; (3) that analytical uncertainties be estimated; and (4) uncertainties be accounted in the calculated ECCS performance.

The staff found that WCOBRA/TRAC realistically describes the AP600 behavior during a LBLOCA based on the review of the models, correlations, and code assessment results. The staff also reviewed all uncertainty distributions, response surface generation and their application to determining the 95th percentile of peak clad temperature. Finally, the applicant provided in DCD Tier 2 Section 15.6.5.4A an assessment of the DVI line response during a LBLOCA and concluded that the WCOBRA/TRAC assessment is also applicable to AP1000.

In Section 21.3.3 of this report "Phenomena Identification and Ranking Tables (PIRT)" the staff found that the AP1000 LBLOCA transients are very similar to AP600 and no new high ranked

phenomena were identified. The minor changes in low and medium ranked processes will have minor impact on the calculations and therefore modifications to the code are unnecessary. Therefore, the staff concludes that WCOBRA/TRAC is acceptable for use for the AP1000 LBLOCA analysis.

21.6.4 Long Term Cooling Analysis Using WCOBRA/TRAC

The LBLOCA performance of AP1000 depends critically on the performance of the ADS-4, which has a dual role. ADS-4 completes the depressurization phase for the initiation of the IRWST injection, and ejects a steam-water mixture during the LTC phase to prevent boron accumulation and precipitation in the vessel. The performance of the ADS-4 valves during the LTC phase is the subject of this Section.

The AP1000 flow path inside pipe diameter is 14 inches (18 inches nominal) off the hot leg in a vertical direction. The hot leg diameter is 31 inches. The corresponding values for the AP600 are 10 inches (12 inches nominal) for the flow path pipe and 31 inches for the hot leg. Water entrainment in the ADS pipe depends on steam velocity, hot leg water level and flow regime, while possible de-entrainment in the pipe depends on the pipe diameter and flow regime. Regarding the LTC phase, Section 2.3.3 of WCAP-15644, "AP1000 Code Applicability Report," issued April 27, 2001, states that "The simulation in WCAP-14776 predicting the APEX [facility] tests validate and justify the application of WCOBRA/TRAC to the AP1000 design certification long term cooling ECCS performance analysis." However, it was not clear to the staff how WCAP-14776, "WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report," Revision 3, issued May 1997, "validates and justifies" the application of the code to the AP1000 standard plant design. The AP1000 pipe diameter that feeds the ADS-4 has been increased from 10 inches in the AP600 design to 14 inches in the AP1000 design, and water entrainment in the 14-inch pipe has not been discussed in relation to the steam velocity and water level in the hot leg.

The applicant provided a response to RAI P019 (which was issued on June 28, 2001, during the AP1000 pre-application review) which stated, "Inasmuch as the scaling basis of the OSU facility remain valid for the AP1000 design, this validation basis, which was approved for AP600, remains adequate for AP1000." The "AP1000 PIRT and Scaling Assessment" in WCAP-15613 for the sump injection phase of the LTC treats only the discharge function of the ADS-4, and did not address liquid entrainment through the ADS-4 venting. In response to RAI 440.091R1, the applicant presented data and calculations to demonstrate that steam velocity through ADS-4 in AP600, AP1000 and the OSU experiments are all in the same flow regime. The OSU results demonstrated liquid entrainment. In addition the applicant mapped these points on a flow-regime map for a vertical pipe to demonstrate that all of the above data are similar and reinforce the conclusion that droplet entrainment will take place. (Wallis, G.B., "Phenomena of Liquid Transfer in Two-Phase Annular Flow" International Journal of Heat and Mass Transfer, Volume 11, pp.783-785, 1968). The total mass flow rate at 5,000 seconds into the transient is about 90.7 kg/sec (200 lbm/sec), while the vapor mass flow rate is about 11.3 kg/sec (25 lbm/sec). The staff concluded that sufficient liquid will be expelled from the vessel through the ADS-4, and that boron will not concentrate nor precipitate in the vessel during LTC.

21.6.5 WGOTHIC Computer Program

Westinghouse-GOTHIC (WGOTHIC) is a thermal-hydraulic computer program used for the design-basis licensing analysis of the AP600 and AP1000 passive containment designs. 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 LOCAs and main steamline breaks (MSLBs)). The applicant uses WGOTHIC in a lumped parameter fashion to evaluate the pressure and temperature response of the passive containment to design-basis accidents (DBAs). WGOTHIC is documented in a series of Westinghouse topical reports:

- Westinghouse Electric Company, LLC, WCAP-14382, "WGOTHIC Code Description and Validation," by M. Kennedy, et al., May 1995.
- Westinghouse Electric Company, LLC, WCAP-14967, "Assessment of Effects of WGOTHIC Solver Upgrade from Version 1.2 to 4.1," September 1997.
- Westinghouse Electric Company, LLC, WCAP-14407, "WGOTHIC Application to AP600," Revision 3, by J. Woodcock, et al., April 1998.
- Westinghouse Electric Company, LLC, WCAP-15846, "WGOTHIC Application to AP600 and AP1000", Revision 0, April 2002.

The applicant devised a program plan which included a number of elements to address the passive containment cooling system (PCS) concept. A series of studies, including a PIRT, a scaling analysis, and both separate and integral tests to obtain the information needed to develop models for use in WGOTHIC to evaluate the containment performance during design-basis accidents (DBAs), were carried out, as follows:

- A PIRT was prepared to identify the phenomena important to understanding the PCS performance: WCAP-14812, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," Revision 2, April 1998.
- A scaling report was prepared to evaluate test data against the prototypical design: WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design-basis accidents," dated March 1998.
- To better understand the effects of adverse weather conditions, severe terrain, adjacent structures, and building design variations on the air flow within the PCS annulus, the applicant conducted a series of wind tunnel tests on scale models of the AP600 containment: 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."
- Water distribution tests were conducted to determine the PCS water coverage fraction on the containment dome and cylindrical shell as a function of the PCS water flow rate: WCAP-13353, "Passive Containment Cooling System Water Distribution, Phase 1 Test Data Report;" WCAP-13296, "PCS Water Distribution Test Phase II Report;" and WCAP-13960, "PCS Water Distribution Phase 3 Test Data Report."
- Separate effects studies were performed: WCAP-12665, "Tests of Heat Transfer and Water Film Evaporation on a Heated Plate Simulating Cooling of the AP600 Reactor Containment," April 1992, Westinghouse Electric Corporation,
- Small scale integral tests were performed: WCAP-14134, "AP600 Passive Containment Cooling System Integral Small-Scale Tests Final Report," August 1994.

- Large scale integral tests (LST) were performed: WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," Revision 1, April 1997.
- The heat and mass transfer correlation package, used in WGOTHIC to evaluate the PCS performance, was validated: WCAP-14326, "Experimental Basis for AP600 Containment Vessel Heat and Mass Transfer Correlations," Revision 2, April 1998.
- The development and validation of the lumped-parameter model for use in WGOTHIC: WCAP-14382, "WGOTHIC Code Description and Validation," May 1995.
- The bounding AP600 evaluation model (EM) development, the analytical model development for the AP600 and the PCS specific features, and the validation of the WGOTHIC AP600 EM through sensitivity studies were presented in WCAP-14407, "WGOTHIC Application to AP600," Revision 3, April 1998.

Based on the staff's review of the applicant's program, the staff determined that the WGOTHIC computer program, combined with the conservatively biased evaluation model, is acceptable for the evaluation of the peak containment pressure following a DBA. Although the WGOTHIC code itself is essentially a best-estimate tool, the applicant has taken a conservative approach in the evaluation methodology it is using to support design certification. The 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.

The staff's evaluation concerning the application of WGOTHIC to the AP600 passive containment design is provided in NUREG-1512. On the basis of that evaluation, the staff determined that the WGOTHIC computer program, combined with the conservatively biased AP600 evaluation model, was acceptable for the evaluation of the AP600 peak containment pressure following a DBA.

21.6.5.1 WGOTHIC Application for the AP1000

The applicant requested a pre-certification review for the AP1000 design to evaluate the applicability of the use of the AP600 computer programs and test databases used to support these programs for the AP600 standard plant design. The following documents were provided to the staff:

- Westinghouse Electric Company, LLC, WCAP-15612, "AP1000 Plant Description & Analysis Report," December 2000.
- Westinghouse Electric Company, LLC, WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.
- Westinghouse Electric Company, LLC, WCAP-15644, "AP1000 Code Applicability Report," April 27, 2001.

In its original submittal (WCAP-15612), the applicant provided scoping calculations that were not consistent with the approved models and methodology developed by Westinghouse for use of the WGOTHIC computer program and approved by the staff in NUREG-1512 for licensing evaluations. The applicant provided the results of unverified studies using the approved modeling approach in response to the staff's requests for additional information (RAIs) (M.M. Corletti, "Westinghouse Responses to Requests for Additional Information Related to Pre-Certification Review of the AP1000," (Proprietary and Non-Proprietary), September 12, 2001).

The AP1000 WGOTHIC model used for the design certification review is consistent with the approved models and methodology developed by Westinghouse for use of the WGOTHIC computer program and approved by the staff for licensing evaluations.

21.6.5.1.1 Assessment of the AP1000 PIRT

The staff reviewed the PIRT and scaling assessment report (WCAP-15613) and found that it did not sufficiently describe the expert review process of the PIRT. The staff requested that the applicant provide a summary of the experts' reasoning behind no changes at the "component or volume" level as used in Table 2.6-1 of WCAP-15613. The applicant provided (Letter from M. M. Corletti, "AP1000 Pre-Application Review – Acceptance Review of Codes Submission and Responses to Requests for Additional Information Pertaining to the AP1000 Pre-Certification Review," July 31, 2001) two letters, one from Professor Per Peterson and one from S. G. Bankoff, as an indication of the considerations given to the PIRT process. The letters provide some insight into the process used by the experts in the PIRT process. The overall conclusion was that the differences between the AP600 and the AP1000 plants are modest to small and can be addressed in the analysis.

In support of its determination that there is no need to account for new phenomena for the AP1000, the applicant provided information regarding the need to rewet a surface that has been heated above the saturation temperature. For the AP600, the PCS film temperature was calculated to increase to over 93.3 °C (200 °F), but was not predicted to reach the boiling point. For the AP1000, there was a question as to whether the surface temperature would reach the saturation temperature before the PCS achieved full coverage for the LOCA. Given the results of the analyses presented by the applicant, the staff could not conclude that containment shell temperatures would not exceed 100 °C (212°F) prior to full water coverage, at 337-seconds into the LOCA. In the letter from M. M. Corletti, "Westinghouse Responses to Requests for Additional Information Related to Pre-Certification Review of the AP1000," September 12, 2001, the applicant provided analyses which demonstrated that full water coverage would be achieved for the LOCA prior to the exterior shell temperature reaching 100 °C (212°F). The 337-second time period used for the AP1000 calculation was based on the AP600 design. The applicant has provided an analysis which shows that the delay time for the AP1000 is less than 337-seconds (M. M. Corletti, "Westinghouse Responses to Requests for Additional Information Related to Pre-Certification Review of the AP1000," September 12,2001).

The shell heatup evaluation has been incorporated into WCAP-15846, Section 7, as part of the design certification. This evaluation also justifies the use of the 337-second delay time for the AP1000.

The larger height of the AP1000 (compared to the AP600) could cause more complex recirculation patterns, thereby influencing mixing. Less homogeneity of the containment atmosphere above the operating deck could result, with higher temperatures in the upper dome. The multi-node (lumped parameter) model used by the applicant should conservatively predict any changes in the homogeneity and the resulting temperatures in the upper dome region.

The staff agrees with the Westinghouse PIRT conclusions that the difference between the AP600 and the AP1000 do not change the ranking of the phenomena, that no new phenomena have been identified, and that the models developed to address the high and medium ranked phenomena for the AP600 remain applicable for the AP1000.

21.6.5.1.2 Assessment of the AP1000 Scaling Evaluation

The applicant used the AP600 scaling study to support the AP1000 review. However, the staff and the applicant agreed during the AP600 review that the LST was not properly scaled for transient situations. The LST is only valid for steady-state conditions, as acknowledged by the evaluation of PIRT. In response to a staff request (Letter from M. M. Corletti, "AP1000 Pre-Application Review – Acceptance Review of Codes Submission and Responses to Requests for Additional Information Pertaining to the AP1000 Pre-Certification Review," July 31, 2001), the applicant clarified that the LST is not well-scaled for either AP600 or AP1000. However, the LST does support the mass and heat transfer correlations used in the WGOTHIC code for the AP600 and AP1000 designs.

The staff noted that the test data for the chimney did not cover the range of Grashof and Reynolds numbers for the AP1000 standard plant design. The applicant noted (Letter from M. M. Corletti, "AP1000 Pre-Application Review – Acceptance Review of Codes Submission and Responses to Requests for Additional Information Pertaining to the AP1000 Pre-Certification Review," July 31, 2001), that the model conservatively does not use the "clime" heat and mass transfer correlations in the chimney region, so the range of test data is, therefore, not relevant. The staff reviewed the expected AP1000 range for these correlations as compared to the test data range for the PCS riser/downcomer region, as provided by the applicant in a letter dated July 31, 2001, entitled, "AP1000 Pre-Application Review Acceptance Review of Codes Submission and Responses to Requests for Additional Information Pertaining to the AP1000 Pre-Certification Review," (DCP/NRC1481) and a letter provided by the applicant dated April 16, 2001, entitled, "Response to NRC Letter Requests for Additional Information Pertaining to the AP1000 Pre-Application Review Dated January 30, 2001," (DCP/NRC1476).

The staff agrees with the applicant's conclusion that the mass and heat transfer correlations are acceptable for the evaluation of the AP1000 and that the AP600 test program adequately covers the expected ranges for which these correlations are used.

21.6.5.1.3 Assessment of Potential Changes or Errors in GOTHIC

In response to the staff's concerns regarding errors in the GOTHIC manuals and also improvements and error corrections in recent versions of GOTHIC, the applicant provided additional information (Letter from M. M. Corletti, "AP1000 Pre-Application Review – Acceptance Review of Codes Submission and Responses to Requests for Additional Information Pertaining to the AP1000 Pre-Certification Review," July 31, 2001) to address these issues.

As part of the AP600 review and certification, the applicant provided clarifications regarding discrepancies in the GOTHIC manuals, and the issues regarding these discrepancies were resolved as a part of the AP600 certification. The WGOTHIC version used by the applicant for the AP1000 design certification is based on the same GOTHIC code used for the AP600 review.

Regarding the correction of a number of errors and deficiencies in the GOTHIC code, the applicant stated that its procedures require evaluation of identified errors. The potential impact of any errors that could affect the results of safety analyses would be covered in a revision to the WGOTHIC documentation (WCAP-15846). In regard to improved physical models in newer versions of GOTHIC, the applicant has demonstrated the conservatism of the models that are presently used in WGOTHIC, so use of newer models is unnecessary.

21.6.5.2 WGOTHIC Code Description

WGOTHIC is a modified version of the GOTHIC containment analysis computer program:

- Westinghouse Electric Company, LLC, NAI-8907-02, "GOTHIC Containment Analysis Package Users Manual," Version 4.0, by T. George, et al., Revision 4, Numerical Applications Inc., June 1994.
- Westinghouse Electric Company, LLC, NAI-8907-06, "GOTHIC Containment Analysis Package Technical Manual," Version 4.0, by T. George, et al., Revision 3, Numerical Applications Inc., September 1993.
- Westinghouse Electric Company, LLC, NAI-8907-09, "GOTHIC Containment Analysis Package Qualification Report, Version 4.0, by L. Wiles, et al., Revision 2, Numerical Applications Inc., September 1993.

The WGOTHIC additions include a special multi-compartment heat structure component, referred to as the "clime" model, used to model the PCS. The essential features of the PCS include the containment steel shell, the large PCS water storage tank, the weirs on the upper containment dome for flow distribution, and the air flow path consisting of a downcomer, riser, and chimney region. The PCS acts to reduce pressure during a DBA by removing energy through the containment shell. It also reduces pressure through the compliance (i.e., the change in energy storage resulting from a change in pressure) of the gas within the large containment volume and through 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 passive containment design does not include active safety-grade heat removal systems. The PCS is unique and, therefore, its performance is central to the evaluation of containment performance during DBAs.

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, where it is heated and evaporated. The vapor formed during the evaporation process is carried away in the air flow through the downcomer, riser, and chimney flowpath.

Although the WGOTHIC code itself is essentially a "best-estimate" tool, the EM used by the applicant in support of design certification is a conservative approach. The 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. In particular, the WGOTHIC EM uses conservative models to address the following areas:

- lumped-parameter network representation;
- non-condensable circulation and stratification;
- PCS flow and heat transfer models; and
- dead-ended and liquid-filled compartments.

During the peak pressure period, 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.

In combination with the EM, the staff has approved WGOTHIC for evaluating the peak containment pressure resulting from DBA events (NUREG-1512). However, the program has

not been qualified to predict other parameters of design interest, such as flooding levels, temperature profiles, and non-condensable concentrations (e.g., air, hydrogen).

21.6.5.3 Limitations and Restrictions for Licensing Analyses

On the basis of its evaluation, the NRC staff found that the WGOTHIC computer gives users great flexibility when selecting inputs. As a result, NRC staff reviewers of future WGOTHIC EM analyses to support licensing actions must verify the following conservatisms in the EM model:

- 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.
- 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.
- The vessel wall emissivity values are reduced by 10 percent to reduce the radiation heat transfer.
- The maximum passive containment cooling water storage tank (PCCWST) temperature allowed by the technical specifications (TSs) is used as an initial condition.
- The maximum containment air temperature and maximum internal pressure allowed by the TSs 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 three 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 and the wetted surface areas.
- The minimum PCCWST inventory, as allowed in the TSs, is used to calculate the PCS flow rate for use in the “Evaporation Limited” flow model.
- The PCS liquid film flow is credited following a delay period necessary to establish water coverage of the shell-wetted region. This corresponds to the time needed to establish a steady liquid film coverage pattern in the liquid film based 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 value assumed for the air gap must be justified.
- The loss coefficient in the external annulus should include 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, should not be credited after the blowdown period. This conservative assumption should also be employed for MSLB analyses.

- Heat transfer to horizontal, upward-facing surfaces which may become covered with a condensation film is not credited. In particular, the operating deck itself, which becomes covered with an air-rich layer, should not be credited.
- For each calculation with significant energy transfer to the PCS through the shell, the stability of the "clime" heat and mass transfer solution should 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.
- In the "Evaporation Limited" flow model, the applicant neglects PCS runoff sensible heat, which is conservative and offsets the non-conservatism introduced by the simultaneous use of the Chun and Seban and "Evaporation Limited" flow models. Therefore, these two assumptions must be employed together for the staff to consider this model to be acceptable for licensing analyses.
- The two-dimensional enhancement to the "Evaporation Limited" flow model may not be used to credit leakage reduction for siting evaluations.

21.6.5.4 Conclusion

The staff agrees with the applicant's PIRT conclusions that the difference between the AP600 and the AP1000 do not change the ranking of the phenomena, that no new phenomena have been identified, and that the models developed to address the high and medium ranked phenomena for the AP600 remain applicable for the AP1000.

The staff agrees with the applicant's conclusion that the mass and heat transfer correlations are acceptable for the evaluation of the AP1000 and that the AP600 test program adequately covers the expected ranges for which these correlations are used.

On the basis of its review of the materials submitted by the applicant, the staff has concluded that the WGOTHIC computer program, when applied using the methodology adopted for the AP600 review, is applicable to the AP1000 standard plant design.

21.7 Summary and Conclusions

The staff has evaluated the applicability of the AP600 test programs and analysis codes to the AP1000. Based on its review, the staff finds that except for those items discussed below, the experimental data produced by the AP600 separate-effects and integral-system test programs are appropriate for verification of the processes expected in the AP1000, and the analysis codes validated for the AP600 design are applicable to the AP1000. Therefore, the requirements specified in 10 CFR 52.47(b)(2)(i)(A) are met subject to the resolution of these items.

- The NRC staff has concluded that the scaled entrainment rates in the upper plenums of the tests used to assess thermal-hydraulic codes for AP1000 are too low to be used for that purpose. The process of upper plenum entrainment is importance during recovery from SBLOCA since core uncover and cladding heat up are prevented by the liquid level above the top of the active fuel. The staff has examined the data used by the applicant to qualify the codes for AP600 and determined that non of these tests are sufficiently well scaled to model upper plenum entrainment for AP1000. New experimental data should be obtained and used by the applicant to support the use of the upper plenum entrainment models in the safety analysis of SBLOCA for AP1000.

- The ability of NOTRUMP and WCOBRA/TRAC-AP to conservatively predict level swell in the core during recovery from an SBLOCA was questioned by the ACRS Thermal/Hydraulic Subcommittee. The applicant is required to provide additional justification that the amount of water predicted by the codes to cover the core with a two phase level is consistent with that measured in tests.
- The adequacy of the two node core model used in the WCOBRA/TRAC analyses of long term cooling was questioned by the ACRS Thermal/Hydraulic Subcommittee. the applicant agreed to investigate the effect of core noding for long term cooling analysis by performing analyses with additional noding detail.

A confirmatory issue is the effect of hot leg phase separation on minimum reactor vessel inventory during recovery from a SBLOCA. The NRC staff has made calculations using RELAP5 showing that precise modeling of hot leg phase separation is not a safety significant issue for AP1000, and that the core remains covered even under the conservative assumption of zero phase separation in the hot leg. The applicant has agreed to provide demonstrations that its codes show the same sensitivity and that this conclusion applies to other small break scenarios This information has not yet been received by the staff. Therefore, this is Confirmatory Item 21.7-1.

The staff continues to investigate differences in RELAP5 and NOTRUMP predictions for SBLOCA. The results of this investigation will be reported in the final safety evaluation report.

Table 1

Major Differences between AP1000 and AP600 Design

SYSTEMS/COMPONENTS	AP1000	AP600
Overall Plant		
Net Electric Output, MWe	1117	600
Hot Leg Temperature, °F	610	600
Core		
Core Power, MWt	3400	1933
Number of Fuel Assemblies	157	145
Active Fuel Length, ft	14	12
Average Linear Power, kW/ft	5.707	4.10
Steam Generators		
Model	Delta-125	Delta-75
Heat Transfer Area/SG, ft ²	123,538	75,180
Number of Tubes/SG	10,025	6,307
Reactor Coolant Pumps		
Rated HP/pump, hp	6,000	3,500
Rated Head, ft	365	240
Pump Inertia, lb-ft ²	16,500	4,956
Rated Flow/pump, gpm	78,750	51,000
Pressurizer		
Total Volume, ft ³	2,100	1,600
Volume/MWt, ft ³ /MWt	0.615	0.825
Containment		
Free Volume, ft ³	2.06E+6	1.71E+6
Safety Injection		
Core Makeup Tanks		
Volume/CMT, ft ³	2500	2000
In-Containment Refueling Water Storage Tank		
Minimum Water Volume, ft ³	78,900	74,500
Minimum Water height, ft	28.79	27
Available driving pressure, psi	9.79	9.04

SYSTEMS/COMPONENTS	AP1000	AP600
Injection Line Size, inches	8	6
Injection Line to Sump Tee Size, inches	10	6
Injection Line Resistance, %	32	100
Passive Residual Heat Removal System		
Heat changer Number of Tubes	689	671
Heat exchanger heat transfer area, ft ²	5278	4326
PRHR inlet/outlet line diameter, inches	14	10
PRHR Flow Path Resistance, %	33	100
Automatic Depressurization System		
ADS-4 squib valve diameter, inches	14	10
ADS-4 hot leg off-take pipe diameter, inches	18	12

Table 2

Non-LOCA Transients to be Analyzed Using LOFTRAN

Feedwater system malfunctions
Excessive increase in steam flow
Inadvertent opening of a steam generator relief or safety valve
Steamline break
Inadvertent operation of PRHRHX
Loss of external load/turbine trip/MSIV closure
Loss of offsite power
Loss of normal feedwater flow
Feedwater line rupture
Loss of forced reactor coolant flow
Locked reactor coolant pump rotor/sheared shaft
Control rod cluster withdrawal at power
Dropped control rod cluster/dropped control bank
Startup of an inactive reactor coolant pump
Inadvertent actuation of the CMTs during power operation
Inadvertent increase in coolant inventory
Inadvertent opening of a pressurizer safety valve or ADS valve
Steam generator tube rupture

Table 3

Double Ended DVI Line Break Comparison Chart

Event	Time (seconds)	
	RELAP5	NOTRUMP
Break Initiates	0.0	0.0
Reactor Trip Signal	30.6	13.1
“S” Signal	35.3	18.5
Reactor Coolant Pump Trip	50.3	24.5
Intact CMT begins to drain	258.	260.
ADS-1 Actuates	321.1	182.7
ADS-2 Actuates	401.1	252.7
Accumulator Injection Begins	403	251
ADS-3 Actuates	521.1	372.7
ADS-4 Actuates	641.1	492.7
Intact Accumulator Empties	830.7	598.4
IRWST Injection Begins	1306	2076
Intact CMT Empties	2500	2006