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APPLICABILITY OF THE NOTRUMP COMPUTER CODE TO AP600 SSAR SMALL-BREAK LOCA ANALYSES

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by

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

SUMMARY

This document provides a general discussion of the features of the Westinghouse NOTRUMP small-break loss-of-coolant accident (LOCA) emergency core cooling system (ECCS) Evaluation Model and of modifications being made to the calculational methodology of the model in order for it to be applicable to the AP600. The NOTRUMP small-break LOCA ECCS Evaluation Model is applicable to AP600 when the modifications described herein are implemented and the code is validated against the AP600 integral and component tests. The existing small-break LOCA ECCS Evaluation Model nodalization for the reactor coolant system is applicable to AP600 with minor changes, such as those to represent the AP600 passive safeguards systems.

Information is provided regarding the historical background and capabilities of the NOTRUMP computer code and the current Westinghouse small-break LOCA ECCS Evaluation Model. A brief overview of the important transient phenomena observed in design-basis and AP600 small-break LOCA analysis calculations is provided. The key design features of the AP600 are compared to those of standard two-loop PWRs, which have previously been analyzed and licensed using the NOTRUMP small-break LOCA ECCS Evaluation Model. The previous applications of the NOTRUMP small-break LOCA ECCS Evaluation Model are discussed in relation to the AP600 key design features. Enhancements to NOTRUMP to facilitate modeling of the AP600 are indicated, and the validation program for AP600 is described.

1.0 INTRODUCTION

Westinghouse Electric Corporation, in conjunction with the United States Department of Energy and the Electric Power Research Institute (EPRI), has developed an advanced light water reactor design, known as AP600. AP600 is a 1940 MWt, 600 MWe two-loop pressurized water reactor (PWR) that utilizes passive safety systems.

This report outlines the applicability of the NOTRUMP computer code for analyses of the AP600. A detailed description of the NOTRUMP code is provided in a separate report,[1] and a description of the code applicability to standard-design Westinghouse plants can be found in Reference 2. NOTRUMP was originally developed in response to specific NRC concerns regarding the small-break modeling methods, as presented in reports NUREG-0611 and NUREG-0623. The Westinghouse development of NOTRUMP addressed these concerns and complies with the requirements of Section II.K.3.30 of NUREG-0737 Enclosure 3.

The Westinghouse small-break LOCA analysis NOTRUMP model, currently approved by the NRC for the evaluation of the emergency core cooling system (ECCS) of a Westinghouse-designed pressurized water reactor (PWR) nuclear steam supply system (NSSS), is conservative and in conformance with Appendix K of 10 CFR 50. In the interest of more realistic analysis, the small-break LOCA analytical capability of NOTRUMP is being confirmed by its application to the AP600 test facilities. Modifications to the presently approved code for Appendix K small-break LOCA analyses are being implemented to enable NOTRUMP to predict AP600 thermal-hydraulic small-break LOCA transient behavior with reduced uncertainty.

NOTRUMP is a general, one-dimensional network code. The spatial detail of the reactor coolant system (RCS) is modeled by control volumes appropriately interconnected by flow paths. The spatial-temporal solution is governed by the integral forms of the conservation equations in the control volumes and flow paths. Special models that represent important components, such as reactor coolant pumps, steam generators, and the core, are included. NOTRUMP is extremely flexible, allowing for choices among various two-phase fluid and drift flux models. Another significant feature is node-stacking capability with a single mixture elevation, which eliminates the possibility of unrealistic layers of steam and mixture in adjacent vertical control volumes. A noding configuration and appropriate two-phase flow models for small-break LOCA analyses are developed for the AP600 to take advantage of these model characteristics. Both integral and separate-effects test data are utilized for verification of the individual AP600-related models, where necessary.

The NOTRUMP model of AP600 remains a 10 CFR 50 Appendix K Evaluation Model. Appendix K requires that the Moody correlation be used for the two-phase break flow calculations. Therefore, the AP600 small-break analysis model incorporates the Moody correlation to comply with the Appendix K requirement, but it is applied only to the calculation of break flow through a postulated pipe rupture. The critical flow through the ADS valves, which are regarded as part of the AP600 safeguards system, is calculated as described in this report.

The AP600 NOTRUMP small-break LOCA ECCS Evaluation Model represents the core fuel rods as an average fuel assembly. Decay heat, as stipulated in Appendix K of 10 CFR 50,[3] is applied during the transients; this introduces a large amount of conservatism into the calculation.

1.1 Historical Background

In 1974, the Final Acceptance Criteria (FAC),[3] set forth in 10 CFR 50.46 as the Acceptance Criteria for ECCSs for light water nuclear power reactors, specified the ECCS analysis requirements for plants fueled with uranium oxide pellets within cylindrical Zircaloy cladding. With the advent of the Final Acceptance Criteria in 1974, Westinghouse developed a small-break LOCA ECCS Evaluation Model incorporating the WFLASH computer code. It was used extensively for analysis of the ECCS response to small-break LOCAs in Westinghouse plants with Zircaloy-clad fuel.

Following the accident at Three Mile Island Unit 2, the NRC focused additional attention on the small-break LOCA and the analyses performed to demonstrate that the ECCS can meet the requirements of 10 CFR 50.46. In NUREG-0611,[4] the NRC outlined technical issues regarding the capability of certain models in the WFLASH computer program to simulate the reactor coolant system (RCS) response to a small-break LOCA. While specific models in WFLASH, such as the thermal equilibrium assumption relative to accumulator injection flow, were not able to predict the exact response of the physical phenomena, Westinghouse maintained that the overall ECCS Evaluation Model using the WFLASH computer program was suitably conservative.

Section II.K.3.30 of Enclosure 3 to NUREG-0737[5] clarified the Post-TMI requirements of the NRC regarding small-break LOCA modeling. Section II.K.3.30 of NUREG-0737 required that the licensees revise the small-break LOCA ECCS models along the guidelines specified in NUREG-0611 or justify the continued acceptance of the model. Furthermore, in Section II.K.3.31 of Enclosure 3 to NUREG-0737, the NRC required that each licensee submit a new small-break LOCA analysis using an NRC-approved small-break LOCA Evaluation Model that satisfied the requirements of NUREG-0737 Section II.K.3.30.

In response, the Westinghouse Owners Group (WOG) enlisted Westinghouse to develop the NOTRUMP[1] computer program for reference in new small-break LOCA ECCS Evaluation Model[2] calculations, based on the desire of the WOG to perform licensing evaluations with a computer program specifically designed to calculate small-break LOCAs with phenomenological accuracy. A small-break LOCA is a rupture of the RCS pressure boundary with a total cross-sectional area less than 1.0 $ft.^2$, in which the normal operating charging system flow is not sufficient to sustain pressurizer level and pressure; it is an American National Standards Institute (ANSI) Condition III incident. The purpose of the small-break LOCA analysis is to demonstrate the capability of the ECCS to effectively maintain the reactor in a safe condition following the loss of primary inventory due to a rupture in the RCS boundary having a total break area on the order of 1.0 $ft.^2$ or less.

The use of NOTRUMP for application to small-break LOCA ECCS Evaluation Model analyses was approved by the NRC in May 1985. The NRC concluded that incorporating the NOTRUMP computer program in the Westinghouse small-break LOCA ECCS Evaluation Model was acceptable for performing licensing calculations in compliance with Section II.K.3.30 of Enclosure 3 to NUREG-0737 for all Westinghouse-designed nuclear steam supply systems. Subsequently, Westinghouse also received NRC approval for application of the NOTRUMP small-break LOCA ECCS Evaluation Model to Combustion-Engineering-designed nuclear steam supply system.[6]

Westinghouse and the WOG demonstrated in generic studies that the results obtained from calculations with the WFLASH small-break LOCA Evaluation Model were, in general, conservative relative to those obtained with the NOTRUMP small-break LOCA Evaluation Model.[7] Licensees could then demonstrate compliance with Section II.K.3.31 of Enclosure 3 to NUREG-0737 by referencing the generic studies and providing some plant-specific information.

1.2 NOTRUMP Modeling Capabilities

NOTRUMP is a one-dimensional thermal-hydraulic computer code that is capable of analyzing the thermal-hydraulic behavior of LOCAs with sizes up to a break area on the order of 1.0 ft.^{2} NOTRUMP employs a thermal non-equilibrium model with a two-field representation of the fluid. The two fields are a mixture field (or mixture region) and a vapor field (or vapor region). NOTRUMP is not intended to be applied in the analysis of large double-ended guillotine ruptures of the RCS primary loop piping.

Five field equations and a drift flux model are used to calculate separate liquid and vapor flows and conditions. These equations are solved by a semi-implicit integration technique. Thus, mass and energy equations are solved at volume centers referred to as fluid nodes, while the momentum equation is solved between the fluid nodes via flow links. The equation of state is solved using the Newton-Raphson technique, whereby the nodal pressure is obtained for the known mass and energy of each region of a node.

Metal structures and heat transfer between the structures and fluid are represented by metal nodes and heat links explicitly. Boundary conditions can be applied using special nodes and links referred to as boundary nodes (either fluid nodes or metal nodes) and critical links (either flow links or heat links). Core nodes are used to represent the reactor core as an average channel for which average vessel values of pressure, mixture level, core decay heat, and core vapor flow are calculated.

NOTRUMP includes detailed fluid flow and heat transfer models to accurately represent the thermal-hydraulic phenomena related to two-phase mass and energy convection. Special models are available to accurately represent the effects of two-phase flow, interfacial heat and mass transport, phase separation, and counter-current flow limitations for various configurations. Extensive heat transfer correlations represent regimes from liquid convection, through nucleate and transition boiling, to stable film boiling, forced-convection vaporization, steam forced convection, and condensation.

The spatial detail of the fluid system is modeled by the fluid nodes, interconnected by flow links. The spatial and temporal solution is governed by the integral forms of the conservation equations in the nodes and links. The flexible noding capability in NOTRUMP permits a full nodal treatment of both the primary and secondary sides of a nuclear power plant. NOTRUMP utilizes a momentum balance that permits an accurate calculation of inventory distribution among the fluid nodes and flow links. The drift flux and bubble rise models in NOTRUMP permit modeling of vertical slip flow, including countercurrent flow using flow regime maps. The treatment of phase separation (both natural and forced) permits an accurate calculation of the two-phase mixture level response within the primary and secondary reactor coolant systems (RCSs) in a pressurized water reactor.

The capability of realistically calculating the complex thermal-hydraulic response of single- and two-phase fluid flow under various conditions has permitted the application of NOTRUMP to a broad spectrum of problems, ranging from design-basis small-break LOCAs to natural circulation flow in steam generators. Examples of some of the applications of the NOTRUMP computer code are provided below:

- Severe primary-side accident scenarios to examine various recovery actions to mitigate the consequences of inadequate core cooling scenarios when auxiliary feedwater is available in Westinghouse operating plants[8]

- Transient response to small-break LOCAs in support of studies of reactor vessel integrity issues for Westinghouse operating plants[9]

- Studies of two-phase natural circulation to address concerns related to the phenomena and to recovery processes[10,11]

NOTRUMP contains specific models to calculate the behavior of significant pressurized water reactor components:

- Reactor coolant pump (RCP) behavior is calculated using single-phase homologous curves and an equivalent density model to account for two-phase behavior.

- The accumulator model calculates the nitrogen cover-gas behavior based on polytropic expansion of the gas.

- The core node and related fuel rod model calculates heat generation in the fuel and heat conduction from the fuel through the gap and cladding into the coolant. Fission product decay heat is calculated using decay heat standards. Radial heat conduction in the rod allows for gap expansion or contraction due to thermal expansion of the fuel and thermal and elastic expansion of the cladding. Heat transfer from the cladding to coolant is calculated by a heat transfer correlation appropriate to the fluid conditions and cladding temperatures.

- Models and correlations are available to calculate the following phenomena over a range of conditions encompassing expected small-break LOCA conditions:

  - critical break flow
  - countercurrent flow

- Interfacial phase separation and heat transfer models calculate mass and energy transfer between mixture and vapor regions in a fluid node.

- Metal heat releases from structures are calculated using metal nodes and heat links that calculate heat transfer from structures to coolant using a complete range of heat transfer correlations dependent on fluid conditions in each region of a fluid node.

The NOTRUMP computer code and small-break LOCA analysis methodology have been evaluated and approved by the NRC for use in calculating the performance of the ECCS for design-basis small-break LOCAs for both Westinghouse and Combustion Engineering NSSS designs in compliance with the requirements of Appendix K to 10 CFR 50. With its inherent two-phase thermal hydraulic capability, NOTRUMP can be used to evaluate the system response for several RCS configurations under a wide range of accident analysis conditions. Therefore, it has been selected for use as the Appendix K Evaluation Model computer code for the AP600 small-break LOCA analysis.

1.3 Application of NOTRUMP to Small-Break LOCA ECCS Analyses

In the Westinghouse small-break LOCA ECCS Evaluation Model, the NOTRUMP computer code is applied in the calculation of the RCS transient response to the small-break LOCA. The hot assembly fuel rod transient thermal performance is calculated with the small-break version of the LOCTA-IV[35] computer code.

In NOTRUMP small-break LOCA ECCS Evaluation Model analyses, the primary and secondary RCS fluid volume spatial detail is represented by a network of fluid nodes interconnected by flow links. The structural metal mass is represented by metal nodes that are interconnected by heat links to represent various heat transfer paths between metal structures and surrounding fluid. The NRC-approved noding scheme for the NOTRUMP small-break LOCA ECCS Evaluation Model analyses is shown in Figure 1-1. The existing NOTRUMP small-break nodalization of the RCS will be the basis for the analysis of AP600, with the inclusion of fluid volumes and connections representing the passive safeguards systems.

NOTRUMP utilizes drift velocity bubble rise models to calculate phase separation within the fluid nodes. Various flow regime dependent drift velocity models are employed to allow NOTRUMP to accurately calculate the phase separation throughout the RCS. NOTRUMP's phase separation capabilities are enhanced by the node stacking and mixture-level tracking capability. Multiple fluid nodes may be vertically stacked to accurately represent void fraction gradients, while allowing a single

mixture level to be tracked. This eliminates the unrealistic calculation of distinct mixture levels within each fluid volume in a set of vertically stacked non-homogenous fluid volumes. This feature allows NOTRUMP to utilize a nodalization of the RCS that accurately tracks the various density gradients and mixture levels within the RCS that are characteristic of a small-break LOCA.

The convective transport of fluid mass around the reactor system is modeled in NOTRUMP with flow links that interconnect the fluid volumes. A single momentum equation is used to calculate flow through a single flow link. For two-phase flow situations, NOTRUMP models the two-phase composition and relative slip between the phases in vertical flow paths with flow regime dependent drift flux models. The drift flux models are consistent with, and complement, the fluid node phase separation capabilities described earlier.

The drift flux models are able to treat both cocurrent and countercurrent flow. For horizontal flow paths, cocurrent and countercurrent two-phase flow and slip may be modeled. The relative slip between the phases is modeled through the use of a horizontal interfacial shear correlation. This provides NOTRUMP with the capability to realistically represent phase separation and mass and energy transport throughout the RCS. For instance, liquid that accumulates in the hot legs and upflow side of the steam generators is accurately calculated to drain back into the reactor vessel upper plenum once two-phase cocurrent natural circulation has stopped.

Thermal non-equilibrium is permitted within each fluid node between the upper, predominantly vapor, region and the lower, predominantly liquid, region where the heat transfer at the interface is calculated with an interfacial heat transfer relationship. This feature is essential in order to realistically model the non-equilibrium behavior that occurs in the reactor system during a small-break LOCA.

In the AP600, thermal non-equilibrium is particularly significant in the steam generator plenum, which receives the PRHR outlet flow. There, subcooled liquid and saturated and superheated steam will coexist as a result of the local injection of relatively cold water. The magnitude of the non-equilibrium interaction that occurs in this plenum will influence the system transient response to a small-break LOCA event. The thermal non-equilibrium capability in NOTRUMP is complemented by mechanistic interfacial heat transfer models that calculate the heat transfer between the mixture and vapor regions within a node, should a thermal non-equilibrium condition exist.

The thermal non-equilibrium capability also contributes to the realistic modeling of energy transport between the primary and secondary sides of the steam generators. Energy transport between the primary and secondary sides is important during a small-break LOCA transient since it affects the pressure and mass distribution calculations. The NOTRUMP thermal non-equilibrium model results in a realistic calculation of the heat transfer processes in the steam generators.

Early in a small-break LOCA transient, the steam generators act as heat sinks and are a significant mechanism for decay heat removal. When the AP600 PRHR is active and effective, the steam generators tend to act as heat sources because the primary RCS depressurizes below the steam

generator secondary-side pressure. During this accident, various modes of heat transfer are encountered on both the primary and secondary sides of the steam generator. Each of these heat transfer regimes has its own heat transport characteristics, which must be individually modeled in order to provide a realistic representation of the influence of the heat transfer process between the primary and secondary on the rest of the RCS. NOTRUMP utilizes mechanistic heat transfer models for each of these modes. Furthermore, metal heat transfer in the steam generator tubes, pressurizer, and reactor vessel are modeled.

The RCS mass inventory is to a large degree determined by the break flow models. For two-phase break flow, the Moody model is used in Evaluation Model calculations, as required by Appendix K to 10 CFR 50. For subcooled stagnation conditions, the NOTRUMP small-break LOCA ECCS Evaluation Model uses the modified Zaloudek model for break flow. These models are not changed for application of the NOTRUMP small-break model to the AP600 plant calculations; these calculations are consistent in philosophy and basis with the approved Appendix K NOTRUMP Evaluation Model, including the initial and boundary conditions assumed.

The small-break LOCA is typically characterized by the relatively slow draining of the RCS, with mixture levels in the different components. Gravity effects are very significant in small-break LOCA transients, and they are included in NOTRUMP's momentum balance. Gravitational terms in NOTRUMP account for the elevations of fluid nodes and flow links and for the effects of phase distribution in stratified fluid nodes. The NOTRUMP treatment of gravitational terms allows for modeling of regions where these terms are important, such as the AP600 passive systems. Draining is accompanied by the formation of distinct mixture levels throughout the RCS. These mixture levels vary with time and with the actuation of the AP600 passive safeguards system and are dependent upon the transient two-phase transport of mass and energy within the RCS. Consequently, the degree of accuracy with which a system model is capable of simulating the transient response is dependent upon the capability of the model to accurately represent the transient mass and energy distribution. The NOTRUMP code can calculate the transient mass and energy distribution and is therefore appropriate for application to small-break LOCA ECCS Evaluation Model analysis calculations for the AP600.

1.4 Important Small-Break LOCA Transient Phenomena

For any plant, the small-break LOCA transient response is a function of the design of the facility, the size and location of the break, the assumptions regarding the availability of the various auxiliary systems, the ECCS-engineered safeguards characteristics, and the core power level. For design-basis small-break LOCAs in standard Westinghouse-designed NSSS, the limiting break location is in the cold leg. Additional potential scenarios for the AP600 include the inadvertent actuation of the ADS and pipe ruptures within the passive safeguards systems.

Following a small rupture of piping in the RCS in which the primary fluid inventory loss exceeds the charging fluid makeup capability, depressurization will result in a reactor trip on low pressurizer pressure. Insertion of rod control cluster assemblies complements possible void formation in the core

and results in a decrease of the heat generation rate to residual decay heat levels. The AP600 passive safeguards systems are actuated when the appropriate setpoint, such as low pressurizer pressure, is reached; the "S" signal also causes the reactor coolant pumps (RCPs) to automatically trip following a brief delay.

During the period when the break flow is all liquid, the RCS will depressurize to a pressure near the steam generator secondary-side pressure. Consequently, the secondary-side conditions during the small-break LOCA have an important influence on the transient response. The secondary side may pressurize to the main steam safety valve setpoints. The secondary energy removal capability is then a function of the secondary-side safety valve setpoint and the secondary feedwater flow rate and temperature of that system, if it is assumed to operate.

After sufficient mass depletion, flow through the break will make a transition to two-phase or all-vapor flow. For breaks in the cold leg of a standard Westinghouse PWR, vapor generated by the core decay heat must vent through the pump suction leg loop seal section to exit through a cold leg break. The loop seal steam venting process does not apply to AP600 because it contains no RCP loop seals. The amount and distribution of mass in the RCS of AP600 is influenced by the design of the passive safeguards systems and the actuation of ADS. These systems are designed to retain sufficient mass inventory in the primary RCS of the AP600 to minimize any core uncovery during the entire duration of small-break LOCA transients.

Table 1-1 presents a phenomena identification ranking table (PIRT) for AP600 small-break LOCA events. For different phases of the event (initial blowdown, saturation natural circulation, ADS blowdown, in-containment refueling water storage tank cooling), the importance of various phenomena in different components are ranked. Phenomena that show an "H" are identified as highly important, those with an "M" are of moderate importance, and those items marked with an "L" are identified as being of low importance for AP600 small-break LOCA events.

Examination of the phenomena identified in Table 1-1 indicates that many are the same as those for a conventional plant small-break LOCA event for which NOTRUMP is already qualified. NOTRUMP is fully capable of predicting the highly important break critical flow and core decay heat for a 10 CFR 50 Appendix K Evaluation Model small-break LOCA analysis. It is also capable of predicting the highly important core mixture level and mass inventory, as well as natural circulation flow through and heat transfer in the core.

A brief description follows of the PIRT phenomena during the several phases of a small cold-leg break transient in the AP600. The reactor is assumed to operate at normal full-power, steady-state conditions at the start of the blowdown. The break opens at time zero, and the pressurizer pressure begins to fall as mass is lost out the break. This depressurization is largely defined by critical flow through the break. With the break located at the bottom of the cold leg, a mixture flow exits the break for the majority of the transient, since the mixture level stays high in the reactor vessel. The pressurizer pressure falls below the safety signal setpoint, causing the reactor to trip. The safeguards ("S") signal

follows and results in opening the core makeup tank (CMT) isolation valves. Once the residual fissions die away, core power is defined by the decay heat model. The RCPs trip after a short delay, after which the primary RCS is cooled by natural circulation, with the steam generators removing the energy through their safety valves (as well as by the break). Among the medium importance items during the blowdown, the NOTRUMP pump model will predict RCP performance both before and after trip according to pump characteristic curves. Also, NOTRUMP metal nodes and heat links model the stored energy in and heat transfer from thick and thin metal in the reactor vessel and pressurizer. Note that all of these phenomena are essentially the same for AP600 as for conventional PWRs.

Blowdown phase phenomena unique to the AP600 are those associated with the CMT delivery. Once the CMT isolation valves open on a "S" actuation signal, the CMT injects borated water due to gravity-driven recirculation into the RCS through the DVI lines. The CMT volume injected is replaced with cold leg liquid, which circulates through the cold leg balance line; this hot liquid collects at the top of the CMT.

The primary system exists in a quasi-steady-state condition with the steam generators during a short interval in which the secondary steam generator removes core decay heat.

The steam generator in AP600 plays a much more limited role in the natural circulation cooling phase than is true for conventional plants. Because the PRHR is activated on an "S" signal during a small-break LOCA, the in-containment refueling water storage tank (IRWST) becomes the primary heat sink for the RCS early in the transient, and the steam generator secondary side becomes a heat source. Therefore, any condensation in the steam generator tubes ceases early on during AP600 small-break LOCA transients. Even though NOTRUMP has detailed models for condensation heat transfer in the steam generator tubes,[2] they are not of as great importance for AP600 as for a conventional plant. The more easily predicted heat transfer reverse path of the secondary heating the RCS primary, which occurs also in conventional plants, is long-lived and is well predicted by NOTRUMP. The CMT continues to deliver in the recirculation mode for a while, but eventually a vapor region forms at the top of the CMT volume. As the CMT drains while injecting, its level in time falls to the ADS actuation setpoint, which initiates the third phase of the AP600 small-break LOCA transient, "ADS blowdown."

Table 1-1 relates AP600-specific components, events, and phenomena that occur during the automatic depressurization of the RCS to achieve water injection by gravity from an IRWST. Since the first stage of ADS creates an opening atop the pressurizer, the pressurizer two-phase fluid level increases markedly. Pressurizer tank level and surge line phenomena are significant factors in the depressurization behavior following ADS actuation. Flashing of fluid in the RCS occurs once again due to the depressurization.

Following actuation of first-stage ADS, the second and third stages of ADS activate via timers. Accumulator injection reduces the flow delivered from the CMT, and CMT flow may even be stopped temporarily due to pressurization of the DVI line by the accumulator. The CMT drain rate, DVI line,

and cold leg balance line behaviors are highly significant because the ADS fourth-stage actuation is based upon the CMT liquid level decreasing below a low-low setpoint value. Of somewhat less importance is the condensation of vapor on the CMT walls, since the recirculation has resulted in heating of the CMT.

Critical flow through the ADS stages is the major factor in determining when the RCS has depressurized to the extent that the gravity injection of water from the IRWST can begin. Fourth-stage ADS performance will be particularly affected by the nature of flow in the hot legs. Successful operation of the ADS leads into the IRWST injection cooling phase of the AP600 small-break LOCA event. The IRWST injection then continues into the long-term cooling phase of the accident. By the time of IRWST injection, the CMT is either completely empty, or very nearly so; therefore, CMT phenomena have become relatively unimportant, while the IRWST gravity drain rate through the DVI is highly important. ADS flow, especially through the fourth stage, is highly important, together with hot leg flow phenomena. Moreover, the break critical flow behavior is now less important than before because all the ADS flow paths are open, providing a large area through which to vent steam. Maintaining the core covered with liquid is a function of the decay heat level and the IRWST flow.

The impact of noncondensible gas released with the accumulators empty of liquid during AP600 small-break LOCAs is identified as being of low importance in the PIRT. Because nitrogen gas liberated when the accumulators empty is of interest for AP600, a noncondensible gas model is being considered for the AP600 LOCA analyses.

The AP600 test program has been established to address phenomena identified in the PIRT for the blowdown and natural circulation, with the ADS blowdown and the IRWST injection phases. Test facility results show CMT, DVI line, and PRHR behaviors via the integral test programs. Component tests also investigate the phenomena that occur in the CMTs and ADS system piping. The NOTRUMP code version applied in the AP600 plant analysis will be validated against data from the AP600 separate effects and integral systems test facilities to ensure its prediction capability of the phenomena identified in the PIRT.

a,c

Figure 1-1 NOTRUMP Evaluation Model Noding of Conventional Westinghouse Plants

TABLE 1-1
PHENOMENA IDENTIFICATION RANKING TABLE FOR AP600 SMALL-BREAK LOCA

COMPONENT PHENOMENON	BLOWDOWN	NATURAL CIRCULATION	ADS BLOWDOWN	IRWST INJECTION COOLING
BREAK Critical Flow	H	H	H	M
Subsonic Flow	N/A	N/A	N/A	M
VESSEL/CORE Decay Heat	H	H	H	H
Forced Convection	M	N/A	N/A	N/A
Flashing	M	N/A	M	L
Wall Stored Energy	M	N/A	M	M
Natural Circulation Flow and Heat Transfer	M	M	M	M
Mixture Level Mass Inventory	H	H	H	H
RCP RCP Performance	M	N/A	N/A	N/A
PRESSURIZER PZR Fluid Level	M	M	H	L
Wall Stored Heat	M	M	M	L
PRESSURIZER SURGE LINE Pressure Drop/Flow Regime	L	L	M	L
STEAM GENERATOR 2ϕ - Natural Circulation	L	M	L	L
SG Heat Transfer	L	M	L	L
Secondary Conditions	L	M	L	L

TABLE 1-1 (Cont.) PHENOMENA IDENTIFICATION RANKING TABLE FOR AP600 SMALL-BREAK LOCA				
COMPONENT PHENOMENON	BLOWDOWN	NATURAL CIRCULATION	ADS BLOWDOWN	IRWST INJECTION COOLING
HOT LEG Flow Pattern Transition	L	H	H	H
ADS 1-4 Critical Flow	N/A	N/A	H	H
Subsonic Flow	N/A	N/A	L	H
CMT Recirculation Injection	M	M	L	L
Gravity Draining Injection	N/A	M	H	L
Vapor Condensation Rate	N/A	M	M	L
CMT BALANCE LINES Pressure Drop	M	H	H	L
Flow Composition	M	H	H	L
ACCUMULATORS Injection Flow Rate	N/A	M	H	N/A
Noncondensible Gas Entrainment	N/A	N/A	L	L
IRWST Gravity Draining Injection	N/A	N/A	N/A	H
Vapor Condensation Rate	N/A	N/A	M	L
DVI LINE Pressure Drop	M	M	M	M
PRHR Natural Circulation Flow and Heat Transfer	L	H	M	L

2.0 RANGE OF APPLICATIONS OF NOTRUMP SMALL-BREAK LOCA ANALYSES

The NOTRUMP small-break LOCA model has been applied to various transient simulations of design-basis emergency core cooling system (ECCS) analyses to support the licensing of conventional Westinghouse nuclear power reactors. It is now being applied to calculate the small-break LOCA transient response of the AP600.

2.1 Comparison of Key Plant Parameters

The design of the AP600 was examined to identify differences relative to the typical two-loop PWRs that could affect the ability of the NOTRUMP small-break LOCA ECCS Evaluation Model to accurately calculate the AP600 transient response. Key small-break LOCA ECCS analysis parameters that have an important influence on the transient response were examined. Parameters compared for AP600 and a typical Westinghouse-designed two-loop plant to identify areas in the NOTRUMP small-break LOCA ECCS Evaluation Model that might require additional consideration are shown in Table 2-1. These parameters are discussed below, and only the passive safety systems require significant added consideration.

The AP600 primary system utilizes a four-cold-leg, two-hot-leg configuration with canned-motor primary reactor coolant pumps (RCPs). The pressurizer used in the AP600 design has a volume of 1300 $ft.^3$ (36.81 m^3), which is 30 percent larger than any operating two-loop PWR. The larger pressurizer allows the unit to tolerate operational transients with increased margin. There are 145 fuel assemblies in the AP600 core that, for the plant rating, reduce the average power density relative to conventional plants. The lower power density provides additional critical heat flux margin for operational transients, as well as margin for postulated design basis accidents, such as the loss-of-coolant accident (LOCA). The two-hot-leg and four-cold-leg design also results in smaller cold legs for AP600 compared to a current PWR—22-in. (0.559-m) in diameter for AP600 versus 27-in. (0.686-m) in diameter for current PWRs. [illegible], injection flow is directed into the reactor vessel downcomer from the passive safety systems, which is a unique feature of the AP600 to minimize the safety injection lost during RCS loop breaks.

The steam generator type influences the small-break LOCA transient response by affecting the primary to secondary heat transfer capability and the propensity for countercurrent reflux liquid flow into the reactor vessel upper plenum through the hot legs. The decay heat load is small relative to the available steam generator heat transfer surface area in all steam generator designs, and the design differences of the AP600 unit are insignificant as regards LOCA transient response relative to conventional Westinghouse plant steam generators. Therefore, the difference in steam generator type will not adversely affect the ability to apply NOTRUMP to the AP600.

The main steam safety valve (MSSV) setpoint affects primary-side pressure by affecting the heat transfer between the primary and secondary sides. A lower MSSV setpoint pressure could have a beneficial effect on the small-break LOCA transient response. In the AP600, the PRHR actuates on an

safeguards ("S") signal and thereafter acts as the heat sink for the primary RCS. The steam generator heat transfer characteristics remain as in the conventional PWRs, and for the large majority of the small-break LOCA transient, the secondary acts as a heat source to the RCS primary. The heat transfer methodology may remain the same as in the approved Evaluation Model. Moreover, the core and RCP input models apply equally as well to AP600 and conventional plants and need not be altered. Note that the AP600 linear heat rate of 4.2 Kw/ft. (13.8 Kw/m) is far below the typical two-loop PWR value of 6.2 Kw/ft. (20.4 Kw/m), so additional margin exists.

While the AP600 RCPs show some design differences when compared to the Model-93A RCPs in typical two-loop plants, the homologous curves are very similar. Since the homologous pump curves are used as input in the NOTRUMP small-break LOCA ECCS Evaluation Model calculations, the difference in RCP design is considered to be insignificant to the analysis calculations and will not adversely affect the ability to apply NOTRUMP to the AP600. A more significant change than the type of pump is the automatic RCP trip that occurs when an "S" signal is generated during an AP600 transient. As a result, the RCPs quickly become idle and act only as a hydraulic resistance.

The effect of the safety system design differences is discussed in greater detail in Section 2.2.

2.2 AP600 Design Differences Affecting Small-Break LOCA Analyses

In the AP600 design, the ECCS is comprised of the safety-grade passive core cooling system (PXS) and a passive containment cooling system (PCS) to mitigate the consequences of postulated accidents. The AP600 PXS is shown in Figure 2-1 and includes:

- Two large 2000 $ft.^3$ (56.63 m^3), full-system-pressure core makeup tanks (CMTs) that provide gravity-fed makeup water to the primary system and emergency core cooling injection in the event of a LOCA, following generation of an "S" signal.

- An automatic depressurization system (ADS), with stages located on the pressurizer and the RCS hot legs, that acts to depressurize the primary system in a controlled manner when the CMTs have injected a significant portion of their inventory. Two independent flowpaths comprise the first three stages of ADS; they are attached to the top of the pressurizer, each consisting of one 4-in. (0.102-m) and two 8-in. (0.203-m) lines, with each line containing two normally closed valves that discharge into the in-containment refueling water storage tank (IRWST) through individual spargers. The 12-in. (0.305-m) ADS lines are attached to the top of each of the hot legs of the primary loop. These fourth-stage ADS lines discharge directly to the containment and are sized to ensure that the primary system pressure can be reduced to near the containment pressure.

- A passive residual heat removal heat exchanger (PRHR), which is a C-tube heat exchanger located in the IRWST, that provides decay heat removal in the event of any loss-of-steam generator heat removal function. The PRHR also provides additional depressurization

capability for small LOCA events in the AP600, since it provides a direct method of heat transfer from the primary system to the IRWST following an "S" signal.

- An IRWST that collects the discharge from the ADS and, once the primary system is depressurized, provides long-term, gravity-fed cooling water to the reactor vessel. Condensed steam on the inside of the containment shell is directed back to the IRWST so that a reflux cooling loop is established within the containment.

- Two large 2000-ft.^{3} (56.63-m^3) gas-driven accumulators connected to the direct vessel injection lines that provide high flow for rapid core recovery for postulated large-break LOCAs.

- A specially constructed lower containment volume into which the liquid break flow collects, as well as any residual condensate from the containment that does not collect into the IRWST. At low IRWST levels, the containment floods above the reactor loops, creating a gravity head of water that is sufficient to continue core cooling flow into the reactor downcomer through the direct vessel injection lines.

The AP600 passive systems accomplish the same safety functions as the active systems used in current reactor designs by using natural gravitational forces instead of active components, such as pumps, heat exchangers, fan coolers, and sprays and their supporting electrical, HVAC, and cooling water support systems.

The safety strategy of AP600 is to provide makeup of borated water and heat removal functions using the CMTs and the PRHR in the event that normally available active systems are not available. Additionally, the ADS provides a controlled, automatic primary system depressurization in the event of a significant loss-of-primary-side water inventory. Primary system injection at high pressure is provided by the CMTs, and as the primary system depressurizes to intermediate pressure, injection is provided by the accumulators. To facilitate the depressurization, the ADS first-stage flow paths are activated based on a low CMT level signal. The second and third stages of ADS are then actuated automatically with appropriate time delays. The fourth-stage ADS flow paths are activated when a low-low CMT level is reached. When the primary system is depressurized to near the containment pressure, core cooling is provided by the gravity flow from the IRWST to the reactor vessel. The long-term cooling water for the core is provided from the IRWST and/or the containment when/if it becomes flooded.

In addition to the PXS, AP600 uses a passive containment cooling system. The PCS is a safety-grade system capable of transmitting heat directly from the containment structure to the environment. The PCS removes enough heat from the containment structure to prevent the containment from exceeding its design pressure and temperature and to reduce the pressure significantly in the longer term following any postulated design basis events.

The PCS utilizes the containment shell and the concrete shield building surrounding it to create a natural circulation air flowpath up along the steel containment shell. Following an event where the containment pressure is increased, water is gravity-fed onto the center of the containment dome and wets a large portion of the containment shell. As the water flows down along the outside of the steel containment shell, it is heated and evaporated into the naturally circulating air. The water is gravity-fed from an annular tank built into the roof of the shield building.

This combined use of air and water results in an extremely effective means of heat removal at containment shell temperatures below 212°F (100°C)—the boiling point of water. Once actuated, the system serves as the reactor plant's safety-grade ultimate heat sink with no reliance on electrical power, moving mechanical components, or support systems. Also, air cooling alone can provide continued heat removal for an unlimited time if the three-day supply of water is not replenished.

The functions of the AP600 passive safety system components are compared with those of current PWRs in Table 2-2.

The ECCS is the most significant safety system in mitigating small LOCAs. The ECCS design for AP600 differs from typical two-loop plants in the following ways:

- The pumped injection flow is not a safety-related system for AP600. In design basis analyses, only the passive safeguards systems are presumed to be available, and they operate based on gravity forces.

- The AP600 CMTs provide substantially higher flow at high RCS pressures than is the case for conventional two-loop plants. This not only maintains mass inventory within the RCS during a small LOCA but also, together with the PRHR, provides a significant depressurization of the RCS.

The design differences between the AP600 and a typical two-loop plant affect the applicability of the NOTRUMP small-break LOCA ECCS Evaluation Model to AP600. As depicted in Figure 2-1, the ECCS design involves accumulators, CMT and IRWST injection, the PRHR, and the ADS. Note that no single active failure can prevent any of the safety-injection water tanks from delivering during a LOCA event. The safety-injection water is supplied to the reactor downcomer via two direct vessel injection lines.

Chapter 15.6 of the AP600 SSAR[12] presents a detailed discussion of the performance of these systems during a spectrum of small-break LOCA events. Furthermore, the small-break LOCA cases presented in References 12 and 13 also are accompanied by a detailed discussion of predicted behavior. The discussions in these references will not be repeated in this document. In general terms, the CMT draining activates the ADS during small-break LOCA events, and the ADS depressurizes the RCS to near containment pressure and permits injection from the IRWST. These interactive phenomena in delivery of ECCS during a small-break LOCA are markedly different from a conventional PWR.

where the pumped delivery of ECCS water is independent of the system mass distribution. No primary (ADS) depressurization valves exist in conventional PWRs. In those plants, the core is recovered in a small-break LOCA event at RCS pressure well above containment pressure. Because injection of water from the IRWST is crucial to the ECCS performance of the AP600, the Appendix-K-required limiting single active failure (as employed in the 1992 SSAR small-break LOCA analysis) is identified as the failure of one fourth-stage ADS valve to open.

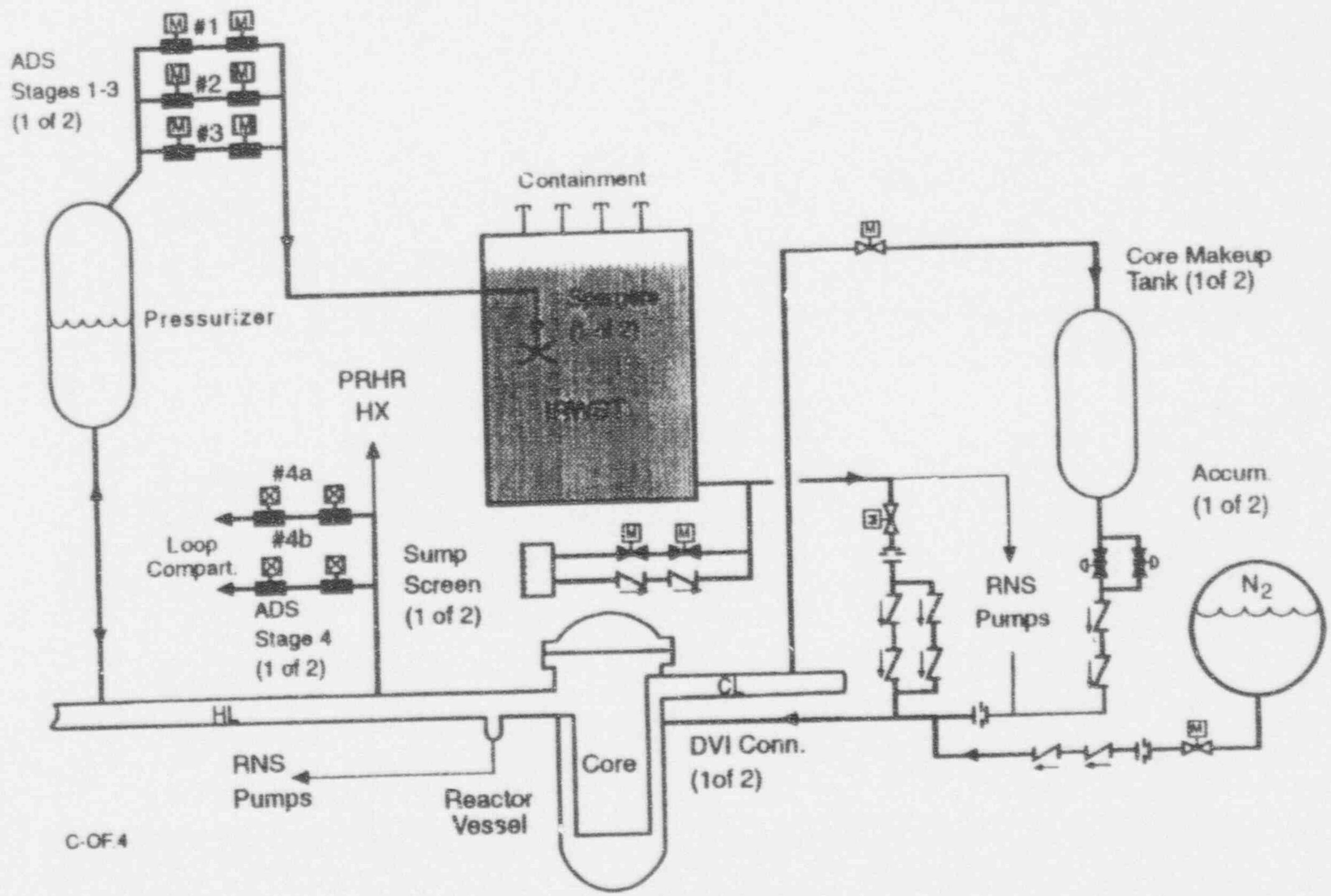

Figure 2-1 AP600 Passive Safety Injection System

TABLE 2-1 COMPARISON OF KEY PLANT PARAMETERS			
Key Plant Parameters	**AP600**	**Typical 2-Loop PWR**	**Addressed in NOTRUMP via**
Steam Generator	Delta-75	51	Input
Reactor Coolant Pump	Canned-Motor	93A	AP600 has no loop seals: RCP region renoded
MSSV Setpoint (psia)	1100	1100	Input
Number of Fuel Assemblies	145	121	Input
Fuel Rods/Assembly	264	179	Input
Core Power Level (MWth)	1933	1615	Input
FQT	2.60	2.40	Power shape input
F-delta-H	1.65	1.75	Power shape input
Accumulator Pressure (psia)	700	700	Input
Safety Injection	Passive Systems	HHSI, LHSI	Nodal models of AP600 PXS are derived

TABLE 2-2
AP600 PASSIVE SAFETY SYSTEM COMPONENTS

Function	Current PWRs	AP600
Reactor Shutdown	- control rods - rideout (negative power coefficient, auxiliary feedwater, chemical, and volume control)	- control rods - rideout (more negative power, PRHR, CMT)
RCS Overpressure	- pressurizer relief - high-pressure trip - pressurizer safety valves	- larger pressurizer - high-pressure trip - pressurizer safety valves
RCS Heat Removal	- main feedwater - auxiliary feedwater - manual feed/bleed (PZR, PORV, HHSI)	- PRHR HX - auto feed/bleed (CMT/IRWST, ADS) - manual feed/bleed (accumulators/RNS, ADS)
High-Pressure Injection	- charging pumps - high-head pumps	- CMT - accumulator/IRWST (ADS) - accumulator/residual heat removal (ADS)
Low-Pressure Injection	- accumulators - low-head pumps	- accumulators - IRWST (ADS)
Long-Term Recirculation	- low-head pumps feeding high-head pumps	- containment sump (ADS)
Containment Heat Removal	- fan coolers - containment spray pumps/heat exchanger	- external air + water drain - external air only cooling

3.0 NOTRUMP MODELING OF AP600

3.1 Introduction

The NOTRUMP code has been validated for use as a small-break LOCA analytical tool via the qualification work performed during its development by Westinghouse. This work included separate effects tests, system effects experiments, and full-scale plant sensitivity analyses. The details of the work are found in References 1 and 2.

The separate effects tests include blowdown vessel tests, core uncovery tests, and loop seal tests. Several blowdown tests were performed and simulated with the NOTRUMP code. The agreement between the code and the test results was good for all of the valid test cases. The core uncovery tests were conducted for pressures between 2.07 and 6.89 MPa with prototypical heat fluxes and mass flow rates as expected during small-break LOCAs in PWRs. The core uncovery test comparisons showed that the NOTRUMP core models predict the mixture level well for steady-state and transient conditions. Figures 3-1 and 3-2 are taken from Reference 2 to illustrate NOTRUMP's capability to predict core thermal-hydraulic behaviors and mass inventory observed in tests[3,4] modeling reflood transients in small-break LOCA cases; this is one of the high importance items of the PIRT in Section 1.4. The test comparisons are for the bounding high pressure/low power and low pressure/high power test cases of those reported in Reference 2. The NOTRUMP simulation corresponds well with the test data.

The system effects experiments included bench marks with the LOFT L3-1, LOFT L3-7, and Semiscale S-UT-08 experiments. The NRC concluded that the benchmarks with the LOFT experiments were acceptable.[2] For the S-UT-08 experiment, it was shown that the data could be simulated using NOTRUMP with a detailed steam generator model. Westinghouse then proposed a simplified steam generator model that resulted in conservative peak cladding temperatures. The NRC found the simplified model acceptable for licensing calculations.

A variety of sensitivity analyses were performed and documented in Reference 2. Additionally, the NRC did independent audit analyses that are discussed in their Safety Evaluation Report.[2] That Safety Evaluation Report found NOTRUMP to be acceptable for small-break LOCA licensing calculations.

The approach adopted for analyzing the small-break LOCA ECCS performance of the AP600 is to utilize the NOTRUMP computer code in compliance with Appendix K. It is recognized that NOTRUMP may need additional verification to ensure that modeling of the temperature/density differences and low pressure conditions associated with a postulated small-break LOCA event in the AP600 can be achieved with the same confidence as the modeling of conventional plants. Model validation through simulation of the AP600 test facilities will demonstrate the ability of NOTRUMP to predict interaction effects of the passive safety systems and the primary loop. Overall, the conservative features of Appendix K are combined with NOTRUMP thermal-hydraulic capabilities

benchmarked against AP600 test data to create a code version with which to perform an analysis that demonstrates the ability of AP600 to meet the 10 CFR 50.46 Final Acceptance Criteria for small-break LOCA events.

3.2 NOTRUMP Code Possible Enhancements for AP600

The unique nature of the AP600 depressurization relative to conventional plants requires enhancements in some of NOTRUMP's modeling capabilities. Specific enhancements to the AP600 NOTRUMP code that enable it to predict pertinent AP600 small-break LOCA phenomena at the low pressures that occur will be validated via comparisons with the AP600 small-break LOCA test data.

3.3 NOTRUMP Code Externals for AP600

Boundary conditions, controls, and properties are provided via a separate library of FORTRAN routines to the NOTRUMP base code in the NOTRUMP user externals. Reference 2 describes in detail the externals for the Westinghouse small-break LOCA ECCS Evaluation Model. The AP600 design is described in detail in the SSAR.[12] Design modifications subsequent to the SSAR submittal are found, together with further small-break LOCA analysis, in the February 15, 1994 and June 30, 1994 Design Change Reports.[13, 14]

Differences in design between AP600 and the Westinghouse standard plants require changes to the existing NOTRUMP user externals. Following is a review of the NOTRUMP User External library functions described in Section 4 of Reference 2 and their applicability to AP600 small-break LOCA analyses:

FEMIXFN, FVMIXFN: these functions remain the same as described in Reference 2. The external FEMIXFN returns the mixture level fraction for each stratified variable cross-sectional area fluid node as a function of the mixture volume fraction, FVMIXFN.

FAMVFN: this function remains the same as described in Reference 2. It returns a multiplier on the mixture interfacial surface area as a function of mixture elevation for variable area stratified fluid nodes.

WFLOW: this function supplies the total mass flow rate for pumped ECCS safety injection, steam generator secondary feedwater and steam flow. THOT upper head temperature initialization, UHI (upper head injection) accumulator flow initialization and vessel reflux flow are calculated by this function. The control functions of reactor trip, steam and feedwater isolation, "S" signal generation, and RCP trip are also generated by this function. Most of the capabilities in this function remain unchanged. They are: pumped safety injection flow and steam generator steam flows as a function of node pressure, signal setpoint calculations for reactor trip, safeguards actuation, and feedwater flow.

The inner reactor vessel reflux flow links are retained. This functional capability is extended to other locations within the AP600 as well.

UHI reflux type flow links are not needed to model any UHI flow in the AP600 design. However, the direct vessel injection (DVI) flow into the downcomer represents a similar physical situation, as regards heat transfer in a volume to an injected, highly subcooled liquid stream. The model originally developed for UHI heat transfer is applied to interfacial heat transfer where the DVI flow paths enter the downcomer.

The initial upper head temperature in the AP600 vessel is between the RCS THOT and TCOLD values. The existing THOT model[2] is being modified to enable the proper AP600 vessel upper head temperature to be specified in steady-state computations utilizing the actual AP600 flowrates in the upper head flow paths.

WRLOCTA / WHLOCTA / EHLOCTA: these functions remain as described.[2] They are used to generate an input file for the LOCTA computer code rod heatup computation.

VOLHEAT: this function remains as described in Reference 2 and performs core power computations. Volumetric heat generation rate is computed as a function of power shape and point kinetics during transients before reactor trip occurs; after residual fissions cease, the Appendix K decay heat model defines core power.

A user-specified power versus time option is used for the test simulations.

UMIFN and UVIFN: the external functions UMIFN(I) and UVIFN(I) provide the overall heat transfer coefficient in Btu/ft^2-sec-°F from the mixture to the interface and the vapor to the interface, respectively, for all stratified interior fluid nodes. In the NOTRUMP small-break LOCA ECCS Evaluation Model user externals, nonequilibrium interfacial heat transfer is provided in all stratified interior fluid nodes by assuming interfacial heat transfer from both the mixture to the interface and from the vapor to the interface. The interfacial heat transfer is by conduction alone unless special modeling, such as is indicated above for the AP600 DVI flow, is introduced. Adjustments and/or additions to the UMIFN/UVIFN models may prove to be necessary based on test facility simulations. If so, the new model(s) will be described in the individual test facility NOTRUMP preliminary validation reports and the overall NOTRUMP validation report.

FAFL(K): this user external function provides a flow area multiplier for each non-critical flow link. In the NOTRUMP small-break LOCA ECCS Evaluation Model externals, this function serves three purposes, 1) steady-state flow convergence, 2) accumulator check valve model, and 3) assurance of Appendix K conservatism during the loop seal steam venting process.

The steady-state flow convergence function of FAFL remains as described in Reference 2, with the variable dimensions increased as necessary; it adjusts loop resistance to match a specified input flow rate.

Since the AP600 has no loop seals, the Appendix K loop seal steam venting logic is irrelevant to AP600 calculations.

Check valve performance is very important to the AP600 design. The need for improved check valve models was identified as follows: the check valve model shall allow for an accurate representation of the AP600 valve types and characteristics. Input variables used by the generalized check valve model added for AP600 include:

- the pressure difference required to begin forward flow through the check valve
- the pressure difference required to fully open the check valve
- the pressure difference at which a fully open check valve begins to close
- the pressure difference at which a check valve fully closes

Limitations on the input to the check valve model are that the pressure difference to fully open cannot be less than that to begin forward flow, and the pressure difference to fully close cannot be greater than that to begin closure, which in turn cannot be greater than that required to fully open the valve. Also, the pressure difference needed to begin forward flow cannot be less than that at which the valve fully closes.

These four input variables can be used to define a hysteresis loop for a check valve. As an example, a hypothetical check valve with the following characteristics can be modeled:

- Starting from the fully closed position, it requires 1.0 psid to begin forward flow through it
- At 2.0 psid it is fully open
- Once fully open, it doesn't begin to close until the pressure across it drops to 1.5 psid
- At 0.5 psid, it is fully closed

The hysteresis effects are not modeled unless the check valve opens or closes fully. For example, suppose the above check valve begins fully closed, differential pressure across it increases to 1.5 psid, and then pressure across it decreases to 0.9 psid; it will have fully closed again when the pressure differential across it decreased back down to 1.0 psid.

These input variables can be given negative values. For example, if the check valve described above was to remain fully open until the pressure across it decreases to -1.0 psid and then slam shut instantly, both valve closure inputs are set equal to -1.0.

RPMPUMP / CSTPUMP: these functions remain the same as described.[2] RPMPUMP provides RCP speed prior to trip, and CSTPUMP is a logical variable set based on the occurrence of trip.

CMETAL / CPMETAL / DCMETAL: these functions remain as described in Reference 2, except that stainless steel properties are added as follows: thermal conductivity CMETAL is given by the equation [ ][(a,c)] where T = °F and K = Btu/hr-ft-°F. This equation is considered an adequate fit of several data sets; DCMETAL is the derivative of this expression with respect to temperature. For CPMETAL, the specific heat, based upon data from several references [15, 31, 32, 33], the following expressions were derived and are used:

C_p = [ ][(a,c)] for 0°F ≤ T ≤ 250°F

and:

C_p = [ ][(a,c)] for 250°F ≤ T ≤ 1400°F

where C_p = Btu/lb - °F
T = °F

DWFLHMI / DWFLHVI / DWFLHMJ / DWFLHVJ: these functions remain the same as described in Reference 2. They provide derivatives of total mass flow rate with respect to mixture region and vapor region enthalpy for critical flow links.

DWFLPI / DWFLPJ: these functions remain the same as previously described.[2] They provide derivatives of total flow with respect to the change in upstream/downstream pressure for critical flow links.

EMFLUID: this function remains the same as previously described.[2] It provides the mixture elevation for each boundary fluid node.

FAMHL / FAVHL / FAMIXHL: these functions remain as described in Reference 2. They provide heat transfer area fractions in stratified fluid nodes.

HMCONDN / HVCONDN: accurate prediction of condensation heat transfer is necessary to provide an accurate prediction of AP600 test facility and plant phenomena. The condensation correlations available[2] are extended as follows: the two-phase coefficient is based on the empirical correlation of Shah[16], a Nusselt-type coefficient is provided for horizontal and vertical PRHR tube heat transfer.[22] Further review of condensation that occurs in the test facilities may indicate other correlations are also necessary.

HMFLUID / HVFLUID: these functions remain as described.[2] They provide mixture and vapor region enthalpies, respectively, in boundary fluid nodes.

PUTINUS: this function, which reads input, must be extended to read added input variables associated with the added capability that the AP600 NOTRUMP externals possess. Refer to Table 3-1 for a listing of the additional input variables.

PFLUID: this function returns fluid node pressure for boundary nodes. Its capability is extended to permit the specification of an input table of pressure versus transient time. In this way, a predicted containment response can be provided directly to the code as a boundary condition.

Following is a discussion of the unique systems presented in the AP600 and a functional description of the modeling necessary to address them.

Automatic Depressurization System (ADS):

The ADS is designed to depressurize the primary RCS to very near the prevailing containment pressure, such that gravity injection of water from the IRWST is achieved. Three stages of the ADS are located atop the pressurizer, while the fourth-stage ADS paths are connected to the two hot legs. In order to model the behavior of this AP600-specific system, the following pertinent parameters and coding that model the plant logic are provided in the AP600 NOTRUMP user externals:

- Setpoints for actuation of ADS stages based on current CMT mixture levels

- The lag between the achievement of an ADS actuation level signal setpoint and the initiation of flow through an ADS stage

- The ADS valve opening time. Flow area through a valve during this time interval is linearly interpolated from full area

- The minimum time delay permissible between ADS stage openings

- The ADS valve full open effective critical flow area

- Linking of valve signals to specific valves to NOTRUMP links

The ADS valves will be modeled using the NOTRUMP input option, which states critical flow shall be computed using the Henry-Fauske correlation combined with the homogeneous equilibrium model.

The Henry-Fauske/homogeneous equilibrium critical flow model is implemented into the NOTRUMP base code, not the user externals. It was taken directly from the RELAP4/MOD5 code, as documented in ANCR-NUREG-1335, September 1976, Volume 1. The Henry-Fauske/homogeneous equilibrium critical flow model in NOTRUMP consists of three subroutines, HFSUB, HFSATR, and HEMFLOW. The HFSUB subroutine uses the Henry-Fauske subcooled correlation[17] to calculate the critical flow through a junction whose donor node is subcooled. The HFSATUR subroutine uses the Henry-Fauske

saturated correlation to calculate the critical flow through a junction whose donor node is saturated. The HEMFLOW subroutine uses the homogeneous equilibrium model (HEM) correlation[18] to calculate the critical flow through a junction.

To calculate the critical junction flow, a test is made to determine the state of the fluid in the junction's donor node. If a junction's donor node is superheated, its critical flow is calculated in subroutine HEMFLOW. If a junction's donor node is subcooled, its critical flow is calculated in subroutine HFSUB. If a junction's donor node is saturated, the calculation of its critical flow depends on the static quality in the donor node. If the donor node's static quality is above the transition quality, the critical flow is calculated in the HEMFLOW subroutine. If the static quality is below the transition quality, the critical flow from the HEMFLOW subroutine is multiplied by the square root of (transition quality/donor node static quality), and the junction's critical flow is set to the minimum of this value and the critical flow calculated in subroutine HFSATUR. The transition quality is set at 0.10.

Validation of the critical flow model will be established via comparison to test facility results. The externals coding opens the ADS valve(s) in the pertinent time sequence and thereafter facilitates calculation of flow through the open ADS paths.

Core Makeup Tanks (CMT):

As described in Reference 19, the CMTs are modeled using standard fluid nodes, metal nodes, heat links, and flow links. Identification of the CMT nodes and links is necessary in the externals in order to activate the tanks at the proper time. The CMTs are isolated during normal operation; actuation on an "S" signal with a signal delay time and valve opening time is required. Nodalization of the CMT is validated against the CMT component test data.

Passive Heat Exchanger (PRHR):

PRHR heat transfer is modeled by applying the standard NOTRUMP heat transfer correlations together with appropriate condensation model(s), as described above in HMCONDN. Identification of the PRHR nodes and links is therefore needed, together with coding to access the PRHR-desired condensation correlation. The PRHR is isolated during normal operation; it should possess its own logic for actuation on an "S" signal with a signal delay time and a valve opening time. PRHR nodalization is validated against integral facility test data.

Table 3-1 shows the additional variable names (additional to the Evaluation Model small-break externals) that are required for the AP600 NOTRUMP externals. Note that the FORTRAN subroutines that comprise the externals are frozen for the final test and SSAR analyses. They are then loaded with the NOTRUMP base code to create a configured, controlled executable code version for those AP600-related applications.

Signal Setpoint Calculations:

The safety signal setpoints are calculated in the user external WFLOW in the NOTRUMP AP600 small-break LOCA ECCS Evaluation Model user externals. The safety signal setpoints calculated include the reactor trip/steam generator shutdown time, S-signal initiation time, and reactor coolant pump (RCP) trip signal times.

The reactor trip and steam generator shutdown times are assumed to be the same, since turbine isolation is initiated on a reactor trip. This time is set when the pressure in the pressurizer fluid node falls below the pressure setpoint designated in the input. The S-signal initiation time is set when the pressure in the pressurizer falls below the pressure setpoint designated in the input. The reactor coolant pump trip signal times are set when the pressurizer pressure falls below the pressure setpoint that is input. Coastdown of the RCP begins after an input time delay has elapsed.

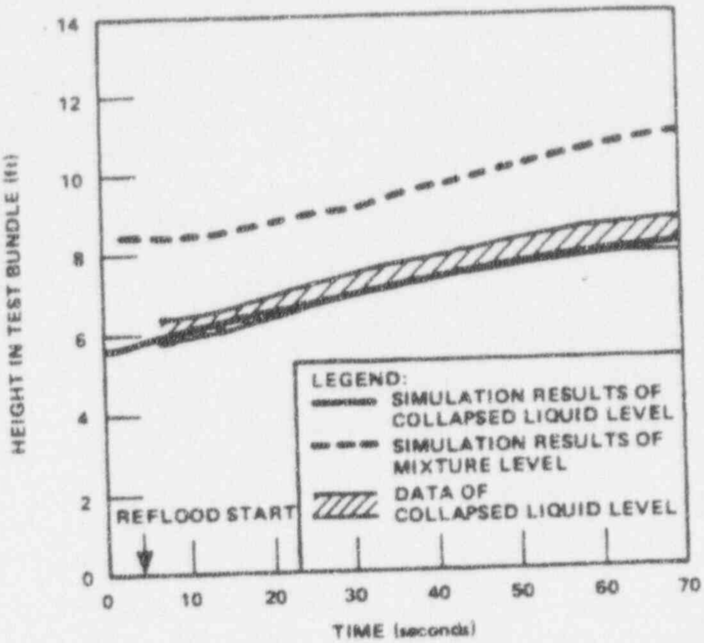

Figure 3-1 NOTRUMP Simulation Results of ORNL Reflood Test 3.02.10F

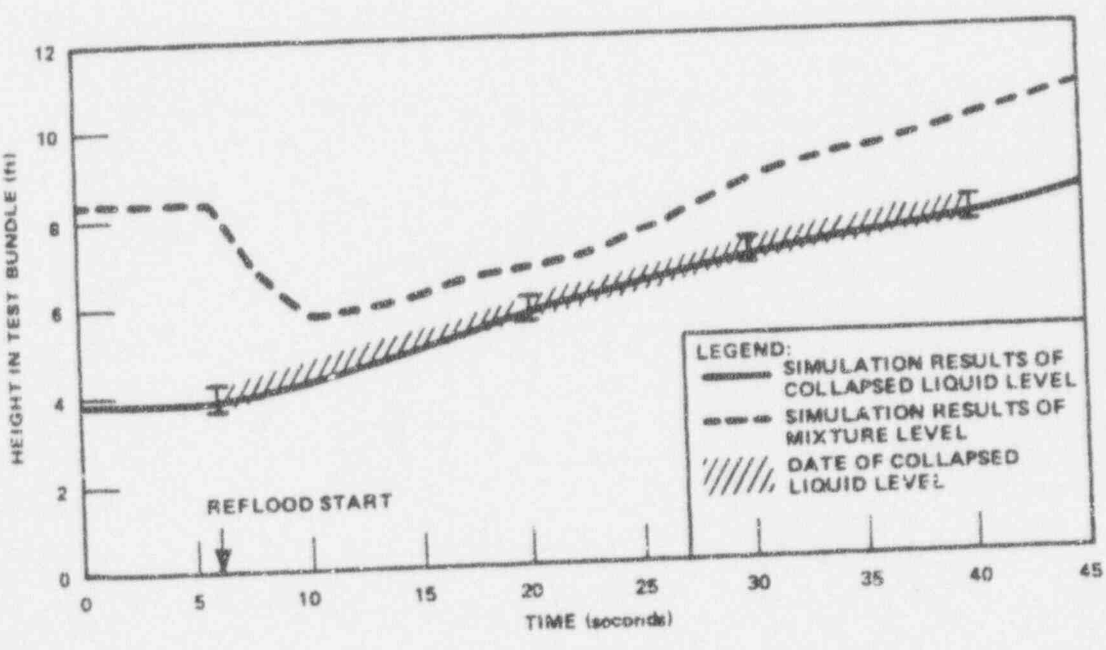

Figure 3-2 NOTRUMP Simulation Results of ORNL Reflood Test 3.02.10H

TABLE 3-1 VARIABLES ADDED TO NOTRUMP CODE EXTERNALS FOR AP600	
New Variable	Description

(a,c)

TABLE 3-1 (Cont.) VARIABLES ADDED TO NOTRUMP CODE EXTERNALS FOR AP600	
New Variable	**Description**

(a,c)

TABLE 3-1 (Cont.)
VARIABLES ADDED TO NOTRUMP CODE EXTERNALS FOR AP600

New Variable	Description

(a,c)

4.0 VALIDATION OF THE NOTRUMP AP600 MODELING

The SSAR NOTRUMP noding diagrams for the AP600 application are shown in Figures 15D-1 through 15D-4 of Reference 12 for the spectrum of small LOCAs. Only passive, safety-related systems are modeled in the design-basis SSAR analyses. The NOTRUMP AP600 input model used for the SSAR analyses has been defined to comply with the standard Westinghouse small-break LOCA Evaluation Model methodology.[1,2] Features of the NRC-approved Westinghouse small-break ECCS Evaluation Model using the NOTRUMP Code, as included in Figures 15D-1 through 15D-4 of Reference 12, are discussed in this section. In the AP600, all loops of the plant are explicitly represented. The model was developed consistent with the two-hot-leg, four-cold-leg plant loop configuration. The vessel downcomer is represented, as is the lower plenum. The NOTRUMP core model for AP600 uses axial core nodes of equal height. The upper plenum and the upper head regions of the vessel are represented, with individual flow links representing the guide tube flow path and the flow holes in the upper support plate. The pressurizer and the helical pressurizer surge line down to its connection with the loop 1 hot leg are included.

The following material describes the noding for one of the two loops modeled. This loop is referred to as the pressurizer loop since it contains that component. The noding for the remaining loop is the same as described below, except that the node numbers are different. The horizontal portion of the pressurizer loop hot leg is represented. The steam generator inlet plenum and the inclined hot leg vertical pipe entering the inlet plenum are modeled together.

For the AP600 steam generator, primary fluid nodes equal in size represent the fluid volume of the heat transfer tubes. The top elevation of the nodes is equal to the inside top of the highest heat transfer tube, and the bottom elevation of the nodes is equal to the bottom of the steam generator tube sheet. The nodes are connected to each other. The core and steam generator noding studies[2] remain the basis for the NOTRUMP nodings of these components.

Each primary-side fluid node in the steam generator is connected to one metal node by a heat link for modeling of heat transfer from the primary to the steam generator tube metal. Heat links connect the tube metal nodes to the secondary fluid nodes for tube metal-to-secondary heat transfer. The NOTRUMP heat transfer model for parallel flow on the outside of tubes is appropriate for the AP600 and is used on the secondary side. The steam generator outlet plenum receives the effluent from the PRHR heat exchanger. Note that because the AP600 design does not contain loop seals, the loop seal noding is removed from the AP600 NOTRUMP nodalization. The two RCPs and the cold legs complete the loop and connect into the downcomer.

Flow links are used to transmit mass and energy between the fluid nodes. Figures 15D-1 through 15D-4[12] shows the modeled flow links. Many of the connections between fluid nodes are made with single flow links. However, in a significant number of locations, pairs of flow links are used. These pairs are referred to as horizontal stratified flow link pairs. They are modeled as such to represent

stratified flow and to allow for the possibility of stratified cocurrent or countercurrent vapor-liquid flow.

The steam line and feedwater flows on the pressurizer loop are modeled via flow links. The corresponding flow links exist in loop 2 as well.

Passive safety injection is modeled with several flow links. The pressure balance lines connect loop 2 (the non-pressurizer loop) cold legs with the core makeup tanks (CMTs). In turn, the CMTs are connected with the vessel downcomer by the direct vessel injection (DVI) lines. Note that the pressurizer to CMT balance lines have been deleted from the AP600 since the SSAR analysis; the corresponding flow links are removed from the NOTRUMP model for the Reference 14 LOCA analysis. The PRHR is included and is connected to the pressurizer loop hot leg and steam generator outlet plenum.

To obtain a better representation of the AP600 small-break LOCA phenomena, the standard SSAR plant nodalization will be adjusted as appropriate based on evaluations of test facility data. Comments on the PXS modeling follow:

- A [ ][a,c] representation of the PRHR heat exchanger loop is being used. PRHR actuation occurs on a safeguards ("S") signal. This early actuation of the PRHR makes it far more significant for small-break LOCA events, not only for subcooled heat transfer from the primary but also as a steam condenser prior to ADS actuation. Condensation heat transfer correlations are applied when primary-side steam condenses in the PRHR; for other heat transfer regimes, the standard NOTRUMP models and correlations are used.

- The CMT is modeled as described in Reference 19. Additional fluid nodes represent the cold leg pressure balance line and the direct vessel injection line, which are connected to each CMT.

- The ADS piping into the IRWST is being modeled, as shown in Figure 15D-1, subject to verification in modeling the ADS Phase B tests.[36]

- The IRWST is modeled [ ][a,c]

The CMT recirculation and delivery behavior has been demonstrated by the CMT component tests and also by integral test facility results. The behavior of the PRHR heat exchanger has also been observed in the integral facility results. These test simulations will validate the small-break LOCA NOTRUMP model capability to predict pertinent PRHR and CMT phenomena and their impact on the AP600

system response. The NOTRUMP code is only applied to the initial portion of the total draining time of the AP600 IRWST. Shortly after the initiation of the IRWST injection, the AP600 enters long-term cooling, and the long-term cooling code will be applied to consider the long term draining of the IRWST. Thus, the fact NOTRUMP is only being applied for a time period in which the initial cold layer at the tank bottom persists in the IRWST means that only a one-dimensional model of this tank is necessary. Likewise, condensate formed in the containment building and its effect on the IRWST (and thereafter, the RCS) will be examined in the long-term cooling code analysis of AP600. Use of a simple pressure boundary condition is sufficient for the time span of the NOTRUMP analysis.

4.1 Test Simulations

A comprehensive test and analysis program has been developed to confirm the passive safety features of the AP600 design. The program includes large-scale separate effects tests on the major components, such as the CMT and the ADS piping and sparger. These experiments are being used to validate the AP600 NOTRUMP modeling of the respective components. The NOTRUMP model of the CMT test facility was submitted,[19] and the ADS facility model NOTRUMP preliminary validation report will be submitted in March 1995. Reference 19 addresses the prediction of CMT phenomena, including the condensation of steam on the tank walls and liquid surface, the liquid phase temperature distribution, and phase separation. It demonstrates that NOTRUMP can acceptably predict the pertinent thermal-hydraulic phenomena observed in the CMT component tests. The ADS test facility NOTRUMP report will address the prediction of such phenomena as flow rate through and fluid conditions within the ADS piping and condensation at the sparger.

In addition to the separate effects tests that examine the thermal-hydraulic behavior of a particular component for computer code validation, there are two integral systems tests that examine systems interaction in the AP600 design. The integral systems effects tests are being used to verify the capability of NOTRUMP to predict on a best-estimate basis integrated passive safety system behavior and passive safety system interactions during postulated small-break LOCA events.

The two integral systems tests are the full-height, full-pressure SPES-2[37] (Simulazione PWR per Esperienze di Sicurezza) test and the one-quarter scale, low-pressure Oregon State University (OSU) integral systems test.[38] They concentrate on small-break LOCA transients in which the passive safety systems determine the plant response. For each of these tests, a separate preliminary report documenting the NOTRUMP simulations and validation will be submitted in the second quarter of 1995.

The SPES-2 test facility[20] is located in Piacenza, Italy and is operated by SIET (Societa Informazioni Esperienze Termoidrauliche) for ENEL (Ente Nazionale l'Energia Elettrica) and ENEA (Ente per le Nuove Tecnologie, l'Energia e l'Ambiente). The original SPES test facility was constructed in 1985 as a full-height, full-pressure model of a Westinghouse three-loop PWR. The original tests examined operational transients, such as station black-out, single- and two-phase natural circulation, and

small-break LOCAs. In 1992, Westinghouse, in cooperation with ENEL, ENEA, SIET, and Ansaldo, modified the SPES facility (SPES-2) for a program to test the integrated behavior of the AP600 passive safety systems.

The SPES-2 facility is a full-height, full-pressure simulation of the AP600 primary system and passive core cooling system (PXS). The SPES-2 experiments examine the AP600 PXS response for a range of small-break LOCAs at different locations on the primary system using either the PXS or the passive and active systems. The SPES-2 test simulations with NOTRUMP will be documented in a preliminary report to be submitted in April 1995.

The OSU low-pressure AP600 integral systems test facility[21] is located at the Corvallis campus. The OSU experiments examine the passive safety system response for the small-break and large-break LOCA transition into long-term cooling. A range of small-break LOCAs are simulated at different locations on the primary system. Different single failure cases are examined to confirm that the worst case scenario was used in the AP600 SSAR analysis. Selected tests continue into the long-term cooling, post-accident mode in which the passive safety injection is from the reactor sump, as well as the IRWST. The validation of NOTRUMP against the OSU test results will be documented in a May 1995 preliminary report.

Note that all of the NOTRUMP documents listed above are preliminary validation reports. The final NOTRUMP verification and validation report for the AP600 application will be submitted in September 1995.

4.2 Compliance with Appendix K

To demonstrate compliance of the NOTRUMP model, the text of Appendix K, " ECCS Evaluation Models," will be subdivided into its individual sections, beginning on the following page. Commentary is then provided on a section-by-section basis.

The AP600 small-break LOCA SSAR analysis has been performed to the 1974 10 CFR 50 Appendix K requirements using the NOTRUMP computer code. NOTRUMP has been approved by the NRC for use in the small-break LOCA analysis of Westinghouse[2] and Combustion Engineering[6] PWRs. The AP600 reactor vessel is very similar to the standard Westinghouse three-loop plant vessel, and the AP600 loop layout is very similar to that of a Combustion Engineering PWR with one hot leg and two cold legs per loop, except that there is no RCP loop seal. Thus, NOTRUMP has been approved for use in the 10 CFR 50 Appendix K small-break LOCA analyses of reactor geometries that very closely resemble the AP600 design.

Passive safety systems are the unique features of the AP600. These systems are modeled using the fluid control volume/flow link approach already verified as indicated above for NOTRUMP PWR applications. The SSAR analysis has assumed the limiting potential single failure, the failure of one fourth-stage ADS vent path to open on demand. This failure limits the depressurization capability of

the AP600 reactor coolant system (RCS) in such a way that more time is required to reach the RCS pressure at which water may be delivered from the IRWST. Reducing fourth-stage ADS venting capability increases the potential for core uncovery during the system depressurization before IRWST injection is achieved. Other required, conservative features of Appendix K, which are intended to ensure an overall level of conservatism in the Evaluation Models, include the ANS-1971 decay heat with 20 percent uncertainty added, the use of Moody for saturated and two-phase fluid critical break flow, and the use of the limiting axial power distribution and time in core life. Applying Appendix K to AP600 increases the severity of small-break LOCA conditions and provides a consistent level of overall conservatism with current PWRs. The AP600 small-break LOCA analysis also includes conservatisms not specifically required by Appendix K. Review the AP600 NOTRUMP Evaluation Model relative to the 10 CFR 50 Appendix K criteria, on a section-by-section basis as in Reference 3:

I. Required and Acceptable Features of the Evaluation Models

A. Sources of Heat During the LOCA.

For the heat sources listed in paragraphs 1 to 4 below it shall be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for such uncertainties as instrumentation error), with the maximum peaking factor allowed by the technical specifications.

AP600 Compliance

As is the case for current-generation LWRs, the use of 102 percent power, as stipulated in Appendix K, bounds the AP600 operational uncertainty in power measurement. The use of the maximum peaking factors permitted in the technical specifications provides a conservative basis for the AP600 analysis, consistent with the intent of Appendix K.

A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences, for the spectrum of postulated breaks and single failures analyzed.

AP600 Compliance

Westinghouse has compiled a comprehensive database of axial power distributions for existing Westinghouse plant core designs using the NRC-approved RAOC methodology (WCAP-10216-P-A). RAOC methodology generates an exhaustive set of power distributions without definition of a specific operating strategy. This database provides power shapes for ECCS Evaluation Model calculations that comply with the Appendix K regulatory requirements. The database, as formulated, bounds the range of possible AP600 axial power distributions. The routine use of the gray rods during AP600 core power maneuvers will minimize the xenon

oscillations that produce highly top-skewed power shapes that are limiting in rod heatup for small-break LOCA events that uncover the core. The power shape used in NOTRUMP (and also for the LOCTA-IV rod heatup computation of any AP600 small-break LOCA case that predicts uncovering of the core) is a top-skewed, limiting shape from the Westinghouse power distribution database that corresponds to the AP600 core design peaking factors. This approach is consistent with the currently approved small-break LOCA methodology for existing Westinghouse PWRs and will predict the limiting rod heatup consistent with the intent of Appendix K.

1. The Initial Stored Energy in the Fuel. The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy). To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

AP600 Compliance

The PAD computer code version 3.4 (WCAP-10851-P-A) considers all Appendix K modeling requirements noted above and has been licensed by the NRC for use in the Westinghouse ECCS Evaluation Models. The AP600 is equipped with 17 x 17 Vantage Plus fuel, an existing product that can be analyzed by PAD. If an AP600 small-break LOCA case uncovers the core, a LOCTA-IV burnup study will be performed for that case to identify the calculated peak cladding temperature at the limiting time in fuel life. This approach will predict the limiting rod heatup for the AP600 consistent with the intent of Appendix K.

2. Fission Heat. Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

AP600 Compliance

NOTRUMP code kinetics have been licensed by the NRC as applicable to existing core designs. Since the AP600 is equipped with Vantage Plus fuel (a current product) in a low-power density design, the existing kinetics modeling in NOTRUMP remains appropriate for AP600 LOCA analyses. Furthermore, the AP600 control rod design functional capability remains the same as in existing PWR plant designs. The rods fall into the core by gravity upon receipt of a reactor trip signal and terminate the core fission process. The shutdown reactivities, including allowance for uncertainties and rod insertion, are given values to maximize fission heat generation (and thus the calculated rod heatup) consistent with the intent of Appendix K.

3. Decay of Actinides. The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

AP600 Compliance

Use of the highly conservative Appendix K decay heat value, as approved in the Westinghouse small-break LOCA (NOTRUMP, LOCTA-IV) Evaluation Model[2], will bound core decay power in the AP600 plant.

4. Fission Product Decay. The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard.[24] The fraction of the locally generated gamma energy deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

AP600 Compliance

The NOTRUMP and LOCTA-IV computer codes for the Westinghouse Evaluation Model for small-break LOCA events model fission product decay heat as stipulated above. The use of 1.2 times the 1971 standard significantly overstates actual decay heat behavior, so the use of this algorithm is conservative in any small-break LOCA application, including the AP600. Not only will Appendix K decay heat predict a conservative clad heatup during any core uncovery transient, but its use will also maximize the core boil-off predicted. Overpredicting the decay heat-induced core steam generation will maximize the challenge to the capability of the AP600 systems to depressurize the plant to achieve the gravity injection of IRWST water. The overprediction by approximately 20 percent of the true decay heat/core boil off rate is one of the main sources of margin in the AP600 SSAR analysis. The use of Appendix K decay heat and the other required Appendix K features provides significant margin in the AP600 small-break LOCA SSAR calculations.

5. Metal-Water Reaction Rate. The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation.[25] The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall also be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less than 1.5 in. each way from the location of the rupture, with the reaction assumed to be steam limited.

AP600 Compliance

The AP600 contains ZIRLO™-clad fuel, so the use of the Baker-Just Appendix K metal-water reaction function is conservative, just as it is for conventional plants.

6. Reactor Internals Heat Transfer. Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

AP600 Compliance

Current small-break LOCA Evaluation Model computations model metal heat from the pressurizer, the reactor vessel walls, etc., as indicated above. NOTRUMP is fully capable in this area. The AP600 vessel and loop metal heat is modeled in accordance with established techniques to properly represent its effect on the small-break LOCA transients. In addition, the metal in the core radial reflector structure is modeled in NOTRUMP.

7. Pressurized Water Reactor Primary-to-Secondary Heat Transfer. Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

AP600 Compliance

The AP600 plant is equipped with Westinghouse Delta-75 steam generators. The Delta-75 steam generators are U-tube units, similar in design to other Westinghouse-manufactured units. Since NOTRUMP modeling has been demonstrated to be applicable to other Westinghouse steam generator units, it is also applicable to the AP600 analysis. The AP600 does not possess a safety-related auxiliary feedwater system, so no credit is taken for any startup feedwater delivery to the steam generators during the design-basis AP600 small-break LOCA events.

B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Each Evaluation Model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-

dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

AP600 Compliance

As previously noted, the NRC-approved PAD computer code can properly represent all fuel performance characteristics of the AP600 Vantage Plus fuel design during normal steady-state operation. The NRC-approved LOCTA-IV computer code is used to calculate fuel rod heatup in any predicted core uncovery during the small-break LOCA transient; LOCTA-IV contains applicable fuel cladding swelling and burst models. If the core thermal-hydraulic conditions are severe and core cooling is degraded, LOCTA-IV will predict cladding swell, rupture, double-sided metal-water reaction, and a flow blockage penalty consistent with the conservative requirements of Appendix K.

C. Blowdown Phenomena

1. *Break Characteristics and Flow.*

a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

AP600 Compliance

A spectrum of small-break LOCAs, from a 1-in.-diameter cold leg break to a full double-ended rupture of a CMT pressure balance line, is reported in the AP600 SSAR. Further small-break LOCA analysis results are reported in References 13 and 14. This spectrum includes cold leg, DVI line, and pressurizer vapor space (the inadvertent ADS actuation case) breaks and demonstrates that the AP600 complies with the 10 CFR 50.46 Acceptance Criteria for postulated small LOCA events. Large-break LOCA events are analyzed separately using the WCOBRA/TRAC computer code, following the best-estimate analysis approach permitted by the 1988 modification to 10 CFR 50 Appendix K.

b. Discharge Model. For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model.[26] *The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperature calculated by this variation has been achieved.*

AP600 Compliance

The NRC has approved NOTRUMP for small-break LOCA Appendix K ECCS Evaluation Model applications. The NOTRUMP Evaluation Model uses the modified Zaloudek critical flow correlation for the subcooled stagnation state and the Moody critical flow correlation when saturated or two-phase conditions exist at the break.[2] These models are applied to the postulated break in the AP600 small-break LOCA analyses, so conformance with the Appendix K rule is maintained. During the nearly two decades since the adoption of Appendix K, it has been established that the Moody model tends to overpredict actual critical break flow mass flow rates. The SSAR spectrum of small-break LOCAs includes an adequately wide range to obviate any need for further analysis with a different critical break flow correlation. In addition to the effect of any postulated break, the AP600 reactor coolant system will be depressurized automatically in a staged manner through use of a series of valves. To preclude overprediction of the capability of the automatic depressurization system (ADS) valves, a more realistic critical flow model (Henry-Fauske merged into homogeneous equilibrium) has been applied to the ADS valves, as described in the SSAR. Calculating the AP600 system response in this fashion ensures that the vent capability of the ADS is represented more accurately than would be the case by simply applying Zaloudek and Moody as stipulated by Appendix K for the break flow.

c. End of Blowdown. (Applies Only to Pressurized Water Reactors.) For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining "end of bypass" include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

AP600 Compliance

This is not relevant to small-break LOCA transient analyses.

d. Noding Near the Break and the ECCS Injection Points. The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

AP600 Compliance

Good engineering practice has been applied to define the break region noding in NOTRUMP. The methodology used is based on development studies[1] that were approved by the NRC for NOTRUMP Appendix K small-break LOCA analyses. A similar methodology is being applied in the SPES and Oregon State facility simulations.

2. Frictional Pressure Drops. The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation[27] or a combination of the Thom correlation[28] for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation[29] for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

AP600 Compliance

The NOTRUMP modeling of two-phase frictional pressure losses has been approved by the NRC as being in compliance with Appendix K. The appropriate friction and form loss resistances for the AP600 design geometry are input to NOTRUMP for the small-break LOCA analysis.

3. Momentum Equation. The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

AP600 Compliance

This section calls for proper physical equations to be used in the Evaluation Model. The NOTRUMP code[1,2] momentum equation formulation has been approved as acceptable for use in a 10 CFR 50 Appendix K Evaluation Model.

4. Critical Heat Flux.

a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical

parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following correlations.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following correlations.

AP600 Compliance

The MacBeth correlation is used in NOTRUMP to judge the occurrence of critical heat flux and post-CHF heat transfer. The correlation has been reviewed by the NRC and determined to be in compliance with Appendix K. It remains applicable to the AP600 application.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

AP600 Compliance

This section is pertinent to the large-break LOCA core reflood phase and is not applicable to small-break LOCA computations.

5. Post-CHF Heat Transfer Correlations. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer co-efficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

AP600 Compliance

Post-CHF heat transfer correlations are included in the NRC-approved NOTRUMP and LOCTA models for core heat transfer. They remain valid for the AP600 application because the AP600 fuel is no different from fuel used in many other Westinghouse plants.

6. Pump Modeling. The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

AP600 Compliance

The AP600 is equipped with a safety-related automatic reactor coolant pump (RCP) trip that is activated by a safeguards actuation signal. The automatic RCP trip feature of the AP600 is unique among Westinghouse plant designs; appropriate flow resistance for the pumps during coastdown and during the post-coastdown is modeled in NOTRUMP. The pump modeling in NOTRUMP was reviewed by the NRC and found to be in compliance with Appendix K.

7. Core Flow Distribution During Blowdown. (Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations, the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

AP600 Compliance

The NOTRUMP/LOCTA Evaluation Model has been reviewed and approved for determining the core flow distribution during a small-break LOCA event. There is nothing unique about the flow distribution through the AP600 core, so the approved NOTRUMP and LOCTA small-

break LOCA Evaluation Model nodings and techniques are applied to the reactor vessel lower plenum/core/upper plenum region.

D. Post-Blowdown Phenomena; Heat Removal by the ECCS

1. Single Failure Criterion. An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation, the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

AP600 Compliance

The General Design Criteria also require consideration of the limiting single failure. Each break in the AP600 small-break LOCA SSAR analysis spectrum has assumed the identified limiting single failure that inhibits the reactor coolant system (RCS) depressurization to the pressure at which the IRWST can inject. This limiting failure for depressurization of the RCS is the failure of one of the fourth-stage ADS vent paths to open.

2. Containment Pressure. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

AP600 Compliance

As implied in this section of Appendix K, containment pressurization due to the postulated small-break LOCA event would aid the AP600 in achieving IRWST injection. Prediction of a higher containment pressure increases steam density in the AP600 RCS and thus, facilitates the venting of steam through the ADS.[23] It is conservative to assume that no pressurization above the initial 14.7 psia containment pressure value occurs in the AP600 small-break LOCA analyses. This requires the RCS to depressurize to a lower value before the IRWST can inject.

3. Calculation of Reflood Rate for Pressurized Water Reactors. The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise, the pump rotor shall be assumed to be running free. The ratio of the total liquid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data.

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

AP600 Compliance

This sentence is directed to the reflood phase of a postulated large-break LOCA event. The NOTRUMP accumulator discharge model in the AP600 SSAR calculations, which is identical to the model approved for current PWR applications, remains applicable to the AP600.

4. Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.

AP600 Compliance

NOTRUMP currently models the steam-water mixing associated with pumped safety injection. For the application of NOTRUMP to the AP600 (with its passive safety systems and direct vessel injection), the code will be additionally benchmarked against the AP600 integral systems tests. In this way, steam-water mixing condensation effects are appropriately considered in the AP600 SSAR small-break LOCA design analyses.

5. Refill and Reflood Heat Transfer for Pressurized Water Reactors. For reflood rates of one in. per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results.

AP600 Compliance

This section is not applicable to small-break LOCA transients.

II. Required Documentation

1. a. A description of each Evaluation Model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the Evaluation Model, must be furnished to the Nuclear Regulatory Commission upon request.

AP600 Compliance

References 1 and 2 provide a detailed description of the existing small-break LOCA Evaluation Model NOTRUMP computer code. A description of the model changes and comparisons with AP600 component and integral test data will be submitted for NRC review, including this document.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

AP600 Compliance

Solution convergence of the NOTRUMP computer code has been established for the current Evaluation Model. Since no change has been made to the numerical solution scheme, there is no need to repeat that effort for the AP600 NOTRUMP code.

3. Appropriate sensitivity studies shall be performed for each Evaluation Model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

AP600 Compliance

The NOTRUMP nodalization of the AP600 reactor coolant system (RCS) is very similar to that approved by the NRC for use in Appendix K small-break LOCA analyses of Westinghouse and Combustion Engineering PWRs. The ample NOTRUMP noding sensitivity studies performed to establish the vessel, steam generator, and coolant loop modeling are documented in References 1 and 2. The nodalizations of the unique AP600 passive components (the automatic depressurization system (ADS), and the core makeup tanks (CMTs) and their balance line connections) are established by modeling the respective tests. The final noding employed in the additional May 1995 AP600 small-break LOCA analysis provides sufficient thermal-hydraulic detail while at the same time maintaining computational efficiency. The AP600 SSAR noding scheme is further validated via the NOTRUMP predictions of the SPES and Oregon State University integral systems tests.

4. To the extent practicable, predictions of the Evaluation Model, or portions thereof, shall be compared with applicable experimental information.

AP600 Compliance

NOTRUMP modeling of the AP600 will be validated by benchmarking against the ADS and CMT component test results and against the small-break LOCA tests in the SPES and Oregon State University integral test facilities.

5. General Standards for Acceptability - Elements of Evaluation Models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this Appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

Sensitivity cases pertinent to the event being analyzed will be included in the SSAR preliminary small-break LOCA additional analysis to be submitted in May 1995. The specific cases are described below:

- RCS Loop Break Location Study – Loop 1 versus Loop 2 Cold Leg: The purpose of the break location sensitivity case is to evaluate the effect of the cold leg break location on the calculated performance of the AP600 following a postulated small-break LOCA transient. For Westinghouse PWR designs, the cold leg location has been found to be the limiting location for small-break LOCAs as a result of its reduced ability to vent vapor to the break. Vapor can vent to the break very quickly for hot leg and pressurizer vapor space breaks, which reduces the system inventory loss compared to equivalent size cold leg breaks. In addition, for AP600 the pressurizer vapor space break location is considered via the inadvertent ADS actuation case.

  For the cold leg location, somewhat different system behavior can occur when the postulated break is located in the passive residual heat removal (PRHR) loop rather than the CMT loop.

- Delayed Injection for the DEDVI Break: The DEDVI break is the limiting case with regard to available safety injection flow capability. As such, achieving effective water injection from the one available (intact) accumulator is the critical factor in maintaining the RCS mass inventory for this event. In order to minimize the accumulator delivery the CMT level setpoint at which the ADS actuation signal is generated is assigned a low value. Moreover, the single failure assumed is that of one first stage (and third stage) ADS path in order to delay a fully effective accumulator delivery.

A sensitivity case has already been performed in Reference 30:

- Reduced ADS Fourth-Stage Valve Effectiveness: The inadvertent ADS actuation case is totally dependent on the ADS to depressurize the RCS primary to achieve IRWST injection. Therefore, this event was chosen for a study in which the fourth stage ADS valve effective

critical flow area is reduced to assess the possible impact of ADS valve performance uncertainty. The results demonstrated the margin present in the AP600 ADS design.

These cases are performed to comply with the 10 CFR 50 Appendix K requirement for appropriate sensitivity cases. Applying the approved Appendix K Evaluation Model produces a conservative assessment of the AP600 emergency core cooling system (ECCS) performance for small-break LOCA events.

5.0 CONCLUSIONS

The NOTRUMP computer code's modeling capabilities are appropriate for analysis of small-break LOCA events in AP600. The nodalization of the AP600 primary system in NOTRUMP is based on the Westinghouse ECCS Evaluation Model methodology, which is adapted slightly to model the AP600 configuration. Possible enhancements to NOTRUMP are being investigated for the prediction of the unique AP600 small-break LOCA depressurization.

Simulation of the AP600 test facilities with NOTRUMP will validate that the small-break LOCA analysis methodology developed for AP600 is appropriate for design certification analysis applications. A spectrum of cases will be analyzed in the SSAR preliminary small-break LOCA analysis in May 1995.

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