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LOFTRAN & LOFTTR2
AP600 CODE APPLICABILITY DOCUMENT

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by
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LOFTRAN[1] is a digital computer code that simulates the transient behavior of a multi-loop pressurized water reactor system. The code simulates a multi-loop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps, with up to four reactor coolant loops. The code has an extensive history of use in performing design- and licensing-basis non-loss-of-coolant accident (non-LOCA) analyses and has been reviewed and approved for use in non-LOCA analyses by the NRC.

The LOFTTR2 code is a specialized version of the LOFTRAN code, modified for the analysis of steam generator tube rupture events. LOFTTR2 includes an enhanced steam generator secondary-side model, a tube rupture break flow model, and improvements to allow simulation of operator actions. The code is documented in References 24, 25, and 26 and has been reviewed and approved by the NRC for steam generator tube rupture analyses.

The AP600 is a two-loop pressurized water reactor with passive (natural) emergency safeguards features. The significant features of the AP600 that have been added to LOFTRAN and LOFTTR2 for design basis analysis are:

- Passive residual heat removal (PRHR) system
- Core makeup tanks (CMT)
- Reactor vessel head vent

The LOFTRAN/LOFTTR2 pressurizer safety valve model was enhanced to simplify the simulation of inadvertent reactor coolant system (RCS) depressurization events due to opening of an automatic depressurization system (ADS) train. The new versions of LOFTRAN and LOFTTR2 are referred to respectively as LOFTRAN-AP and LOFTTR2-AP. Figure 1 illustrates the relationship between LOFTRAN, LOFTTR2, LOFTRAN-AP, and LOFTTR2-AP.

This report will:

- Summarize existing models in LOFTRAN and LOFTTR2 that continue to be used in LOFTRAN-AP and LOFTTR2-AP (Section 1.0)

- Review AP600 features and identify new model requirements for inclusion in LOFTRAN and LOFTTR2 (Section 2.1)

- Review non-LOCA transients and explain applicable code versions (Section 2.2)

- Summarize application assumptions and methods used in LOFTRAN and LOFTTR2 for design basis analyses (Section 2.3)

- Explain in detail new models added to LOFTRAN to create LOFTRAN-AP and LOFTTR2-AP (Section 3.0)

- Summarize the verification plans for new models added to LOFTRAN and LOFTTR2 (Section 4.0)

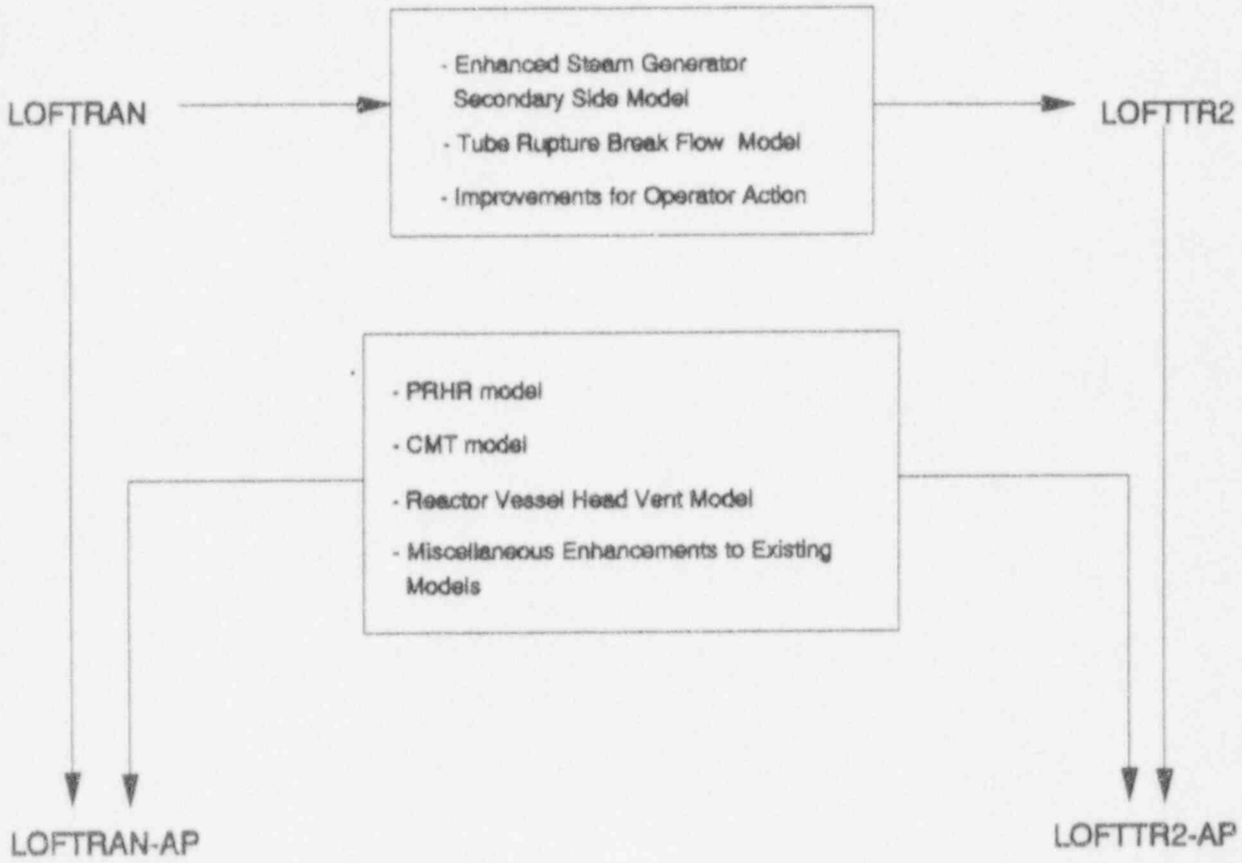

Figure 1 Relationship of LOFTRAN Code Versions

1.0 INTRODUCTION AND OVERVIEW

1.1 Introduction

LOFTRAN[1] is a digital computer code developed to simulate behavior in a multi-loop pressurized water reactor system. The code simulates a multi-loop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps (RCPs), with up to four reactor coolant loops. The code has an extensive history of use in performing design- and licensing-basis non-loss-of-coolant accident (non-LOCA) analyses and has been reviewed and approved for use in non-LOCA analyses by the NRC.[20]

The LOFTTR2 code is a specialized version of the LOFTRAN code, modified for the analysis of steam generator tube rupture events. LOFTTR2 includes an enhanced steam generator secondary-side model, a tube rupture break flow model, and improvements to allow simulation of operator actions. The LOFTTR2 code version is documented in References 24, 25 and 26 and has been reviewed and approved by the NRC for steam generator tube rupture analyses.

The AP600 is an advanced two-loop pressurized water reactor design, which includes many features that differ from previous PWR designs that use the LOFTRAN code family for design-basis transient analyses. Among the new AP600 features are passive safeguards systems, canned RCPs, twin reactor coolant system (RCS) cold legs per RCS loop, and an automatic RCS depressurization system.

This report confirms the applicability of LOFTRAN-based codes to non-LOCA and steam generator tube rupture analyses for the AP600. Reviews of the LOFTRAN code, AP600 features, and applicable design basis analyses are performed, and differences and limitations are identified. In some cases, the AP600 differences are accommodated by the adaptation of the current LOFTRAN models with appropriate conservatism. In other cases, new models or enhancements to existing models have been added to LOFTRAN to create an advanced plant version. The upgraded advanced plant version is designated as LOFTRAN-AP.

As described above, the LOFTTR2 code is a derivative of the LOFTRAN code. The main RCS models of LOFTTR2 remain the same as those of LOFTRAN. The LOFTTR2 code version contains an enhanced steam generator secondary-side model, tube rupture break flow model, and improvements for operator action simulation. The same new models and enhancements that have been added to LOFTRAN to create an advanced plant version for AP600 analyses are also incorporated into LOFTTR2. The advanced plant version of LOFTTR2 is designated as LOFTTR2-AP.

Due to the major commonalities of LOFTRAN and LOFTTR2, whenever LOFTRAN is referred to in this report, it will also be applicable to LOFTTR2, however, any exceptions or differences will be noted. Similarly, whenever LOFTRAN-AP is mentioned, it is meant to include LOFTTR2-AP unless there is an exception, and this will be noted in detail.

1.2 LOFTRAN Overview

The following subsections provide a brief summary of the major models contained in the LOFTRAN code. A more detailed description of these models and other models in LOFTRAN can be found in Reference 1.

1.2.1 Reactor Core Model

The LOFTRAN core kinetics model consists of a lumped fuel heat transfer model, a point neutron kinetics model, and a decay heat model. In addition, the code can calculate departure from nucleate boiling (DNBR) during a transient.

Fuel Heat Transfer Model

The fuel heat transfer model uses up to 40 axial nodes (user-specified) and up to four radial nodes in the fuel (one per loop), with a fixed parabolic axial power distribution of 1.5 peak to average value.

Fuel specific heat varies with fuel temperature, and a fixed 2.6 percent of the heat is assumed to be generated directly in the coolant. The overall fuel-to-coolant heat transfer coefficient (UA) is a parabolic fit to values input as a function of radially averaged fuel temperature by the user. The values that are input are usually either maximum or minimum heat transfer values, depending on the conservative direction for the transient of interest, and are obtained from values predicted from more detailed Westinghouse fuel rod design codes.

This method accounts for the effect of fuel-clad gap width variation with fuel temperature. The model is adequate for predicting average core power response for all except the most rapid core power transients, such as the rod cluster control assembly (RCCA) ejection and RCCA bank withdrawal from subcritical faults where the LOFTRAN code is not used. In addition, for transients where specific values of the heat flux or fuel temperatures are important, the LOFTRAN nuclear power versus time is transferred to a more detailed transient fuel heat transfer model (the FACTRAN[5] code) for calculation of hot and average channel heat flux.

The point neutron kinetics model in LOFTRAN uses six delayed neutron groups and employs an implicit finite difference solution technique for stability. A source term and the prompt neutron lifetime are included in the equation. The model takes into account reactivity changes due to changes in moderator temperature, the Doppler effect, boron concentration, control rod position, and input values of reactivity versus time. Moderator density and boron-worth coefficients, variable rod worth versus position, and an integral Doppler defect versus power with a correction for water temperature change are input by the code user as well as a trip reactivity versus time curve.

In addition, the code contains the capability for using core quadrant weighted density, water temperature, and boron concentration to determine the reactivity feedback in order to conservatively predict the course of transients with large loop temperature and core power distribution asymmetries, such as in the stream line break accident. The weighing factors must be supplied by the code user.

The accuracy or conservatism of the point kinetics model employed by LOFTRAN is dependent upon how representative the reactivity and feedback coefficients are, as given by the user.

Decay Heat Model

Decay heat in LOFTRAN is calculated from a five-group precursor model in a manner similar to the delayed neutron precursors. The default value closely follows the ANS (1971) + 20 percent curve for finite irradiation (end of equilibrium cycle) plus the actinide contribution. The total value used can be scaled up or down by the user, or different constants can be input. For some transients (e.g., steam line break), decay heat is a benefit and may be conservatively set to zero for the analysis.

DNBR Evaluation Model

LOFTRAN has the capability for calculating the value of the DNBR during a transient using a simple calculational model. The model employs user-input values of the change in DNBR with respect to changes from nominal in the core average power, average coolant temperature, flow, and pressure. Experience has shown that this model is sufficiently accurate over the range from the nominal to the limiting DNBR; nevertheless limiting minimum DNBRs are checked with a more detailed sub-channel analysis calculation of the type performed by a code, such as THINC.[28,29]

The LOFTRAN model is not used to calculate DNBR for loss of RCS flow faults or for faults where asymmetric power distributions are important, such as the steam line break or dropped assembly RCCA faults. LOFTRAN-calculated boundary conditions may be used with FACTRAN and THINC for prediction of minimum DNBR for loss of RCS flow faults and with single or multi-channel sub-channel THINC models for steam line break DNBR evaluations.

1.2.2 Reactor Coolant Loop Model

Reactor Coolant Loops

The reactor coolant loop model employs a nodal technique with the number of nodes (actually control volumes) specified by the user. The code can handle up to 160 core sections, 10 hot leg sections per loop, 16 steam generator tube sections per loop, and 8 cold leg sections per loop. The loop model reproduces the layout of standard Westinghouse PWR plants.

Generally, a typical analysis employs about one-half of the number of allowable sections in each component. The pressurizer can be located in any loop, the only restriction being that reverse reactor coolant flow is not allowed in the loop with the pressurizer. A homogeneous-equilibrium slug flow model is used, thus the code will handle void generation; but the steam and water phase are always in equilibrium, and there is no slip. This model is entirely adequate for cases with moderate void generation and under-pumped flow conditions.

Although the code calculates pressure drops around the loops based on flow rate and input loss coefficients, an averaged coolant loop pressure is used in computing fluid properties. The fluid equations solved are those for conservation of mass and energy; the momentum equation is only solved to determine overall loop flow rate, with the change in flow versus time assumed to be uniform around the loop.

The code can initialize with reverse flow in one or more loops, although the flow in the core and the loop with the pressurizer must be positive. Boron transport is handled with the transport time delay, which is correct only if the volumes and flow rates are such that an exact volume replacement occurs in one time step. This is not likely to be important except for a fast transient with safety-injection under a low flow condition where the boron concentration could vary significantly around the loop.

Reactor Coolant Flow

The basic equation of motion is solved for flow, including effects of friction pressure losses, elevation (density) heads, pump head, and fluid momentum. RCP homologous curves are input by the user and are used by the code to compute pump head and torque.

The pump speed equation includes the effect of pump motor torque, hydraulic torque on the impeller, pump windage and friction, and pump rotating inertia. The flow model is used for pump coastdowns, locked rotor, and natural circulation flow calculations. The code can calculate transient flow reversals due to a RCP start-up or shutdown in one or more loops. The equations solved by this model are straightforward, and the results have been found to be adequate in comparison with actual flow coastdown measurements.

Reactor Vessel Mixing

Reactor vessel mixing in the inlet and outlet plena is simulated by the code based on user input. Only a few transients result in large inlet temperature asymmetries (for example the steam line break accident) and are sensitive to this input.

1.2.3 Pressurizer Model

The pressurizer model computes the mass and energy balances in a two-region (water and steam) pressurizer. Since the water level may change during a transient, a variable control volume model is used. Each region is assumed to be uniform (perfect mixing).

Condensation or superheating is allowed in the steam region, and evaporation or subcooling in the water region. Water drops are assumed uniformly distributed in the steam region and fall at a constant rate, while steam bubbles are uniformly distributed in the water region and rise with a constant velocity. The model includes the effects of heaters, spray, and relief and safety valves, with their appropriate control systems.

Safety analysis calculations are usually performed conservatively, assuming no pressure control if such control would improve the results or with full control if this is the conservative direction.

Relief valve flow rates for steam relief as a function of pressure are input to the LOFTRAN code. For water relief, the valve area may be input, and the homogenous-equilibrium model (HEM) from ANS Standard N661 is used to calculate flow.

1.2.4 Steam Generator Model

On the primary side, the steam generator model contains multiple (up to 16) tube sections. In LOFTRAN, the secondary side is represented by a single volume model with a saturated mixture of steam and water. In LOFTTR2, the secondary side is also represented by a single node. However, in LOFTTR2, the secondary side has a two-region node, where the lower region may be saturated or subcooled and the upper region may be saturated or super heated. Multiple tube sections are used on the primary side in order to simulate a log-mean temperature difference (LMTD) type response. The overall UA is initialized by the code to match the nominal input conditions provided by the user. The nominal conditions are obtained from the plant design thermal-hydraulic conditions. The code uses the primary mass flow rate, heat flux, and secondary-side pressure to compute changes in the heat transfer coefficient due to changes that can affect the primary- and secondary-side film resistance.

A steam generator water level correlation is provided for simulating steam generator level indication. Safety system actuations on steam generator water level are not based on the level correlation; instead, it is based on a user-input value of an equivalent secondary-side mass. This value is conservatively chosen by the user based on output from a more detailed steam generator model that computes steam generator water mass at the level setpoint.

Steam and feedwater flow are determined, based on the user option selected. Although the steam turbine is not explicitly modeled in LOFTRAN, the effect of the turbine control system is simulated by assuming constant steam demand prior to turbine trip. As steam pressure rises, steam flow will remain constant due to closure of the turbine throttle control valves. As steam pressure falls, the opening of the turbine throttle control valves is simulated by an input-specified, design excess, valve capacity.

Once the turbine throttle control valves are fully open, steam flow decreases proportionally with steam pressure. Changes in steam demand may be simulated by inputting steam flow versus time. In addition to steam flow demanded by the turbine, LOFTRAN calculates steam relief through safety valves, the steam dump system, and through pipe breaks in various locations.

Feed flow versus time can be input as tabular data, set equal to steam flow within the code, or feedline breaks can be simulated. The Moody correlation[21] with f(L/D) = 0 is used to compute break flow. Break quality versus mass can be input by the user. Steam and feed line isolation are simulated, and steam line check valves can also be specified.

Auxiliary feedwater flow is simulated as a constant flow versus time after actuation and is assumed to be injected in a slug flow model through a user-specified purge volume. The user may control the fraction injected to each steam generator.

Several options are provided for the code to account for the effect of the degradation in heat transfer surface area caused by uncovering the steam generator tubes due to loss-of-water inventory. In one option, the user may input a secondary-side water volume, below which the tubes start to uncover, and the code will reduce heat transfer area linearly with the further reduction in water volume. The user determines the appropriate input (high or low value), depending upon which value is conservative for the transient of interest.

LOFTRAN also has available a built-in correlation that calculates steam generator riser quality and reduces heat transfer with water volume once a user-input value of quality is exceeded. This value is obtained by the user from a much more detailed steam generator model. Alternatively, the UA calculated with the detailed model may be input directly as a function of steam generator water mass. Except for a few faults involving long-term effects of a loss-of-feedwater fault, a feed line break or a steam line break, this model is not important since the reactor trips on the low steam generator level trip before uncovering of the tubes occur.

1.2.5 Control and Protection System Simulation

Control systems that are simulated include automatic rod control, steam dump control, and pressurizer pressure control via pressurizer heaters, spray, and relief valves. A complete digital simulation of each control system is provided, including linear and non-linear gain units, auctioneering, lead-lag compensation units, filters, PID controllers, dead bands, and simulation of the time responses of the sensor inputs. The protection systems that are simulated include reactor and turbine trips, safety injection actuation, and steam line and main feedline isolation.

Failure of one or more protection channels may be simulated. Protection system inaccuracies and time responses are simulated by inputting protection setpoints plus appropriate error allowances and actuation delay times.

2.0 APPLICABILITY TO THE AP600 PLANT

2.1 Review of AP600 Features and Identification of Needed Code Modifications

The LOFTRAN code has been used to simulate licensing basis events, parameter sizing, and control studies for operating pressurized water reactors (PWRs) for several years. Its principal application has been for non-loss-of-coolant-accident (non-LOCA) transients and steam generator tube rupture, however, LOFTRAN is very versatile and has been used for a wide range of transient types. The code includes user options and boundary conditions that allow the simulation of reactivity perturbations, loss-of-forced-reactor coolant system (RCS) flow, secondary-side increases or decreases in steam and feed flow including those caused by secondary-side breaks in the main steam or feedwater system, and depressurization of the RCS through the pressurizer steam space region.

LOFTRAN simulates a multi-loop PWR system by a model containing the reactor vessel, hot and cold leg piping, reactor coolant pumps (RCPs), steam generator (tube and shell sides), and pressurizer. The code includes a core model that simulates the thermal hydraulics and kinetics of the core Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and control rods are included. The code includes simulations of various safety features and the automatic actuation of the safety features by the protection system.

Table 2-1 summarizes plant components and features affecting the analysis methods and models of non-LOCA and steam generator tube rupture events. Table 2-1 identifies the differences and similarities between the AP600 and current PWRs. As can be seen, there are some significant differences, but overall the plant designs are similar. The differences that impact transient simulations with LOFTRAN are handled in several ways:

- Addition of new models to LOFTRAN
- Minor changes or enhancements to existing LOFTRAN models
- Adaption of the current LOFTRAN models with code input adjusted to provide conservative simulations

It should also be noted that the significance of the AP600-specific features varies, depending upon the transient. The passive features of the AP600 do not play a role in ensuring that acceptance criteria are met for all design basis events. For example, some events are mitigated by tripping the reactor. The relationship between new features and specific events is addressed in Section 2.2.

The AP600 fuel, pressurizer, and steam generators are similar to that used in operating PWRs that have used LOFTRAN for design basis analyses. The AP600 reactor vessel is functionally and dimensionally similar to operating plants, with the exception of an elevation offset between the inlet and outlet nozzles. In the LOFTRAN code, the elevation of the inlet and outlet nozzles relative to other RCS components is set to a single value. Accuracy of these elevations is important in calculating elevation heads for determining passive residual heat removal (PRHR) and core makeup tank (CMT) flow rates. Therefore, a modification was made to LOFTRAN to allow the code user to define inlet and outlet nozzle elevations.

Canned Reactor Coolant Pumps

The AP600 uses canned motor RCPs, as compared to the shaft seal pumps used on current PWRs. Like the shaft seal pump, the AP600 RCP is a vertical single-stage centrifugal pump. The AP600 pump uses hermetically sealed canned motors, which obviates the need for a seal around the motor shaft. During certain non-LOCA transients the most important phenomena to be predicted is the RCP flow coast down.

LOFTRAN contains an RCP model that uses head-speed-flow and torque-speed-flow homologous curves. Transient pump speed is calculated by the momentum equation that considers RCP rotating inertia and friction losses. While the AP600 RCP characteristic parameters (head, flow, inertia, etc.) differ from those of the shaft seal pumps, the basic principles governing the transient behavior of the pumps remain unchanged. The LOFTRAN RCP model is applicable and adequate for simulating AP600 transient analyses.

Twin Cold Legs

The current generation of Westinghouse PWRs use a single cold leg and RCP per RCS loop, whereas the AP600 uses two cold legs and two RCPs per RCS loop. The LOFTRAN code simulates a single cold leg and RCP per RCS loop. No changes have been made to the codes to simulate the twin cold leg arrangement. In all the AP600 simulations performed with LOFTRAN, the twin cold leg arrangement is simulated by lumping together the twin cold legs into one.

LOFTRAN contains several RCS flow model options. These options can be described in two groups:

- Code internal flow calculation. RCS flow is calculated based on a pressure drop and fluid momentum balance. Pump kinetics are calculated based on pump homologous curves.

- RCS flow input as a boundary condition as a function of time.

Both flow options require the lumping of the twin cold legs together. When the first RCS flow option is used, LOFTRAN directly performs the RCS flow calculations. Because of the lumped cold leg assumption, uniform flow will be predicted for the twin cold legs on each loop. Thus, this model can only be used for cases where flow conditions in a RCS loop are symmetric (e.g., full RCS flow is present, and all four RCPs are simultaneously coasting down, or when the two RCPs on the same loop are simultaneously coasting down). Cases where asymmetric cold leg flows would occur in a loop, such as a single RCP coasting down or a locked rotor, cannot be performed using the internal flow calculation option in LOFTRAN.

In the second RCS flow model option, flow is input by the user as a function of time. With this model, only net cold leg flow is simulated, however, non-uniform cold leg flow rates in a loop can be accounted for. For example, this flow model option can be used for the asymmetric flow case, where on a single loop, one RCP is operating while the other RCP is not operating. In this example, the AP600 steady-state flow through the cold leg with the operating RCP would be on the order of 68,000 gpm. A reverse flow on the order of 22,000 gpm could be expected in the cold leg where the RCP is not operating. The net flow for the loop would be 68,000 gpm - 22,000 gpm or 46,000 gpm. This net flow can be input as a boundary condition using the second LOFTRAN RCS flow model option.

Net RCS loop flow rates can be calculated (by hand or other codes) and input as boundary conditions to LOFTRAN. There are a limited number of design basis transients where asymmetric flow could occur in the cold legs on a given loop:

- Partial loss-of-forced-reactor-coolant flow events
- Locked or broken RCP shaft events
- Startup of an inactive RCP

Section 2.2 discusses the methodology used to derive the RCS flow rates used as input boundary conditions for these events.

Core Makeup Tanks (CMTs)

The AP600 uses gravity-driven CMTs for reactor coolant makeup and emergency boration, as compared to pumped safety injection, which is used on current generation PWRs. A CMT model has been developed and included in LOFTRAN-AP and LOFTTR2-AP. Validation of the CMT model is discussed in Section 4.0.

The CMT is connected to the RCS through a discharge injection line to the reactor vessel downcomer and an inlet pressure balance line connected to the cold leg. There are two operating modes for the CMTs: water recirculation between the RCS and the CMT, and CMT draindown with steam coming up the cold leg balance line to the CMT.

During water recirculation, hot water from the cold leg enters the CMTs, and the cold borated water in the tank is discharged to the RCS. This results in RCS boration and a net increase in RCS mass.

During CMT draindown, steam is supplied to the CMTs to displace the water that is injected into the RCS. This steam is provided through the cold leg pressure balance line. The CMT draindown mode does not occur during design-basis non-LOCA and steam generator tube rupture events, therefore, only the water recirculation mode of injection must be simulated by the CMT model. A detailed description of the model is provided in Section 3.1.

The AP600 uses two CMTs that inject directly into the reactor vessel downcomer. The two CMT cold leg pressure balance lines are connected to the cold legs on the loop opposite the pressurizer. With this arrangement, the performance characteristics of both CMTs will be the same except for events where asymmetric cold leg conditions are occurring. As discussed previously, the transients where asymmetric cold leg conditions occur are limited to RCP fault initiated events. The CMT model of the LOFTRAN-AP code version simulates the dynamics of a single CMT and assumes the performance of both CMTs is identical.

Anytime CMT operation is initiated, an automatic RCP trip is initiated by the protection system. The RCP trip is single failure proof. Whenever CMT operation is initiated, all forced RCP flow will be terminated, and uniform flow in the twin cold legs of an RCS loop will occur.

The original AP600 CMT design, as described in Revision 0 of the SSAR, also included a pressure balance line from the pressurizer to the inlet of the CMT. Non-LOCA and steam generator tube rupture safety analyses presented in Revision 0 of the SSAR were performed with this pressurizer connection line. With the pressurizer connection line, the CMT draindown injection mode occurred during non-LOCA and steam generator tube rupture events. The design change to remove the balance line from the pressurizer simplified the CMT modeling requirements for transient events by removing the potential for the CMT draindown injection mode.

Direct Vessel Injection

The AP600 CMTs inject directly into the downcomer of the reactor vessel. In the LOFTRAN-AP and LOFTTR2-AP code versions, the CMT injection fluid is added into the cold legs. However, the local pressure in the downcomer is used in the model for calculating CMT flow rates. The code model will therefore calculate the correct flow rates, but there will be a conservative time delay until fluid and boron from the CMT is transported to the core.

Reactor Vessel Head Vent

The AP600 has redundant safety-related remotely operated head vent paths that connect to the top of the reactor vessel and discharge to the in-containment refueling water storage tank (IRWST). A head vent model has been incorporated into the LOFTRAN-AP and LOFTTR2-AP version of the codes that allows the code user to control vent flow as a function of time and simulate manual operator opening and closing of the valve. A detailed description of the model is provided in Section 3.3.

PRHR & IRWST

Unlike current generation PWRs that use a pumped emergency feedwater system for decay heat removal, the AP600 uses PRHR heat exchangers, which consist of two banks of C-tubes immersed in the in-containment refueling water storage tank IRWST. Inlet piping to the PRHR heat exchangers is connected to the hot leg of the RCS, and outlet piping of the PRHR heat exchangers is connected to the steam generator outlet plenum. The PRHR system is located on the same loop as the pressurizer. A PRHR model was included in the LOFTRAN-AP and LOFTTR2-AP code versions, which allow the user to input the PRHR geometric data. The PRHR model is a multi-node model with flow and heat transfer calculated by the code during the transient. The model contains several heat transfer correlation options that allow the code user to under predict or over predict heat transfer, depending upon the conservative direction for the transient being analyzed. A detailed description of the model is contained in Section 3.2.

Startup Feedwater

While the AP600 plant does not contain an emergency feedwater system, it does contain a comparable system known as the startup feedwater system. The startup feedwater system is a non-safety related (control grade) pump system that provides feedwater to the steam generators for decay heat removal. The system is not credited to mitigate the consequences of design basis transients, however, the plant control system could actuate startup feedwater during a transient. The consequences of transients, such as steam line break, could be more severe or the duration of the transient could be extended if the startup feedwater system provided additional inventory to the steam generators. The existing LOFTRAN emergency feedwater model is a general and flexible model that allows the code user to supply conservative feedwater flow rates. The functional capability of the startup feedwater system is similar to the emergency feedwater system used on current PWR plants. The existing LOFTRAN emergency feedwater system model is used on the AP600 for simulation of the startup feedwater during transients where additional feedwater flow to the steam generators would increase the severity of the event.

Reactor Protection System

The AP600 uses digital-based technology for actuation of the protection system functions and safety monitoring, as compared to analog-based systems used on previous PWRs. With respect to the performance of design-basis accident analyses, the digital and analog systems are functionally the same and no modifications to LOFTRAN are required. The AP600 protection system does include several new protection functions that are not found on current PWRs, such as signals for automatic actuation of the PRHRs and CMTs, RCP trip on safeguards ("S") signal, low Tcold "S" signal, etc. Modifications were made to LOFTRAN to allow automatic simulation of some of the new AP600 protection system functions.

Automatic Depressurization System (ADS)

The ADS is not required for mitigation of design-basis non-LOCA and steam generator tube rupture events. Thus, detailed simulation of this system is not required for these events. However, an inadvertent opening of an ADS stage is addressed as a design basis event using LOFTRAN.

The ADS valves are designed to open slowly. To simulate the slow opening, the existing pressurizer relief valve model was modified, such that a slowly opening valve could be simulated for use in analysis of an inadvertent opening of an ADS train.

The modified pressurizer relief valve model in LOFTRAN-AP includes an option such that the code user can specify pressurizer relief valve flow area as a function of time. Flow rate through the valve is calculated in LOFTRAN-AP using the Moody correlation.[21]

This modified pressurizer relief model in LOFTRAN-AP is used to simulate an inadvertent opening of a pressurizer safety valve or any of the first three stages of ADS. It should be noted that this model for ADS is only used in Section 15.6.1 for short-term inadvertent depressurization analyses to demonstrate that the protection system can detect the fault and trip the reactor before the departure from nucleate boiling ratio (DNBR) limit is exceeded. For this type of analysis, maximizing the depressurization rate is conservative. Assuming that the ADS flow is choked at the valves and neglecting interactions with the IRWST results in a conservatively high depressurization rate. Also, this model is not applicable for simulation of the fourth stage of ADS, which is connected to the hot leg. The long-term effects of this event are addressed using a code designed to calculate LOCA events.

Summary

As previously noted, the AP600 specific features are addressed in LOFTRAN in one of three ways. These are through the addition of a new model, modifications to an existing model, or through conservative treatment via code input. Three new models have been added to LOFTRAN to address AP600 features. These are:

- Core makeup tank model
- Passive residual heat removal model
- Reactor vessel head vent model

The following modifications to existing LOFTRAN models were made:

- Addition of new AP600 protection system actuations
- Addition of user input to model elevation difference of reactor vessel inlet and outlet nozzles
- Modification to pressurizer safety valve model to allow simulation of slower ADS valve opening

The AP600 twin cold legs per RCS loop are addressed by conservative code input treatment.

2.2 LOFTRAN Code Versions Used for Design Basis Transient Analyses

Section 2.1 reviews the AP600 components and systems design and identifies code changes for accident analyses in general. The non-LOCA and steam generator tube rupture transients cover a wide range of initiating events and phenomena. In this section-each of the design basis events using LOFTRAN-based codes will be reviewed.

The LOFTRAN-AP version need only be used for design basis analysis if the new safety system models are expected to be actuated and are used to mitigate the event or may interact with other systems, such that the characteristic behavior of an event is different. If the new safety system models are not actuated during a transient, then the unchanged original version of LOFTRAN should continue to be adequate. A review of the design basis transients will show that the mitigation of many non-LOCA events is not accomplished by these new features. Many events are terminated simply by features that exist in the original unchanged version of LOFTRAN, such as reactor trip, turbine trip, feedline isolation, steam line isolation, or opening of safety relief valves. Some transients require no actions, since a safe equilibrium state may be reached. The following subsections present a review of the design basis analyses and will identify the code version requirements needed for each event.

Of the various non-LOCA events, the LOFTRAN family of codes is not used for all AP600 analyses nor was it used for all analyses of current generation of PWRs. Transients for which LOFTRAN is not used are:

- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

- Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from subcritical or low power start-up conditions

- Inadvertent loading and operation of a fuel assembly in an improper position

- Spectrum of RCCA ejection accidents

The analyses of these events use methodologies based on codes other than LOFTRAN or its derivatives.

Table 2-3 presents a phenomena identification ranking table (PIRT) for the AP600 non-LOCA events that can be analyzed by LOFTRAN-based code. Table 2-3 ranks the importance of various component or system phenomena to specific events; phenomena that show an "H" indicate high importance, those with an "M" are of moderate importance, and those marked with an "L" are of low importance. In some cases, a phenomena may not be applicable to a transient, and this is indicated as N/A.

It should be noted that the importance rankings of Table 2-3 were based on the analysis time frame as presented in Chapter 15 of the SSAR. The design-basis analyses results of Chapter 15 generally cover the transient until a safe state is reached. For many Condition II events, the initiating fault may be quickly terminated by an automatic protection system action. For example, a fault that causes inadvertent RCCA withdrawal at a power event will cause reactor power to increase until an overpower type reactor trip occurs. The reactor trip causes an immediate reduction in power and also terminates the inadvertent RCCA withdrawal. Immediately following reactor trip the plant will be in a safe state and the plant may be maintained in a safe stable state or cooled down further using normal plant shutdown procedures. The analysis results presented in Chapter 15 for this event only cover the event from initiation till shortly after reactor trip and the phenomena considered applicable in the PIRT will only be considered over this time frame.

Inspection of the AP600 phenomena identified in Table 2-3 indicates that many are the same as those for conventional PWRs, however, a key difference between AP600 and conventional PWRs is the increased importance of natural circulation flow and related phenomena.

Following is a review of the non-LOCA and steam generator tube rupture events. The review will summarize the consequences of each event and how it is mitigated, which will identify the code version needed for analysis of the event.

Feedwater Malfunction that Results in a Decrease in Feedwater Temperature or an Increase in Feedwater Flow

Analyses of feedwater system malfunctions that result in a decrease in feedwater temperature or an increase in feedwater flow are presented in Sections 15.1.1 and 15.1.2 of the SSAR. These faults, result in a cool down of the RCS. If a negative moderator temperature coefficient exists, a power increase excursion may occur. Similar to previous PWR plant designs, these events are mitigated by reactor trip and isolation of the feedwater system. The response of the AP600 to this event is very similar to that for current Westinghouse PWRs, and the AP600 passive features do not contribute to the plant response. Either LOFTRAN or LOFTRAN-AP can be used in the analysis of these events.

Excessive Increase in Secondary Steam Flow

Analyses of an excessive increase in steam flow is presented in Section 15.1.3 of the SSAR. An excessive increase in steam flow results in a power mismatch between the reactor core and the steam generator load. Protection against this type of event is provided by reactor trip (e.g., overpower ΔT, overtemperature ΔT, or power range high neutron flux). Reactor trip may not be encountered due to error allowances in reactor trip setpoints, and in this case, a stable safe equilibrium condition will be reached. The AP600 response to an excessive increase in steam flow is similar to that of current PWRs, and the modifications included in LOFTRAN-AP for the AP600 do not contribute to the plant response. Thus, either LOFTRAN or LOFTRAN-AP can be used in the analysis of these events.

Steam Line Break

An inadvertent opening of a steam generator relief or safety valve is analyzed in Section 15.1.4 of the SSAR. Steam system piping failures are analyzed in Section 15.1.5 of SSAR. These events result in a depressurization of the main steam system. The steam releases as a consequence of these events result in an initial increase in steam flow, which decreases during the event as the steam system pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cool down results in an insertion of positive reactivity.

The most severe cool down induced reactivity transient is caused by a double-ended steam line rupture. On the AP600, a double-ended rupture will cause a rapid depressurization of both steam generators until a low steam line pressure setpoint is reached. Exceeding the low steam line pressure setpoint generates a steam line isolation signal, a safeguards ("S") actuation signal, and a reactor trip signal. If the break is between the steam generator and the main steam line isolation valve, closure of the main steam isolation valves will only isolate a single steam generator. One steam generator will continue to depressurize. The "S" signal also isolates the main feedwater system, which will terminate the addition of inventory to the faulted steam generator. The startup feedwater system that could be actuated during the event is automatically isolated when cold leg temperature decreases below the low Tcold setpoint. The CMTs are started on an "S" signal to provide boration to prevent a return to criticality or to attenuate the power excursion if recriticality occurs. The PRHR is also actuated on an "S" signal, and it will increase the RCS cooling, thereby causing the addition of positive reactivity. The double-ended steam line rupture will also depressurize the RCS to the point at which the accumulators inject.

The sequence of events during a steam line break will depend on the break size and the initial power level. For the AP600, this sequence is very similar to that of previous PWRs except for the use of CMTs and the PRHR instead of pumped safety injection and emergency feedwater. On previous PWRs, the emergency feedwater would continue to add fluid to the faulted steam generator until it was manually isolated by the operator, whereas on the AP600, control-grade start up is automatically isolated by the protection system. The LOFTRAN-AP version is used to model the operation of the CMTs and PRHR.

Inadvertent Operation of the PRHR

The inadvertent actuation of the PRHR system (Section 15.1.6 of the SSAR) causes an increase in core reactivity by decreasing reactor coolant temperature. Depending upon the initial power level and the reactivity feedback, a reactor trip on an overpower function may occur. Following reactor trip, continued operation of the PRHR cools and depressurizes the plant. The primary pressure, pressurizer level, or cold leg temperature may exceed protection system setpoints, and the CMTs will be initiated. Under certain conditions, the CMTs may increase the pressurizer water level to the point that the pressurizer could become filled with water unless the reactor vessel head vent is opened. Analysis for this event uses LOFTRAN-AP version of the code because of the need to simulate the PRHR, CMTs, and the reactor vessel head vent.

Loss-of-Secondary-Side Load Events

Loss-of-load events, presented in Sections 15.2.2 through 15.2.5 of the AP600 SSAR, are characterized by a rapid reduction in steam flow. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure and a consequential heat up of the primary side. The transients are mitigated by opening of the steam generator and pressurizer safety valves and by tripping the reactor on high pressurizer pressure, high pressurizer water level, or overtemperature ΔT trip signals. The AP600 passive features and other modifications included in LOFTRAN-AP do not contribute to the plant response during the portion of the transient where plant parameters may approach acceptance criteria. Either LOFTRAN or LOFTRAN-AP can be used in the analysis of loss-of-load events.

Loss-of-ac Power and Loss of Normal Feedwater Events

Events such as the loss-of-ac power to station auxiliaries (SSAR Section 15.2.6) and the loss of normal feedwater (SSAR Section 15.2.7) result in a reduction in secondary-side heat sink. Protection in the initial portion of the transients is provided by tripping of the reactor on low steam generator water level. Steam relief through the pressurizer and steam generator safety valves also mitigates the events. Following reactor trip, stored and core decay heat is removed by the PRHR. During the transients, conditions may be reached such that a "S" signal is reached, and the CMTs will be actuated. Under certain conditions, the CMTs may increase the pressurizer water level to the point that the pressurizer could become filled with water unless the reactor vessel head vent is opened. Analyses for these events use the LOFTRAN-AP version of the code.

Feed Line Break

A feed line break (SSAR Section 15.2.9) will also cause a reduction in secondary-side heat sink. Protection in the initial portion of the transients is provided by reactor trip on low steam generator water level. Following reactor trip, the steam generator level will continue to decrease, and the PRHR will be actuated on a low wide-range steam generator level signal. The steam generators will continue to blow down until a low steam line pressure setpoint is reached. On a low steam line pressure signal, the main steam line isolation valves will be closed and an "S" signal will be generated that will actuate the CMTs. The LOFTRAN-AP version of the code with the simulation of the CMTs and the PRHR is used for analysis of this event.

Loss-of-Forced RCS Flow, Locked or Broken RCP Shaft Events

Loss-of-forced RCS flow events (SSAR Section 15.3) may be caused by electrical faults or mechanical RCP faults. The electrical faults may cause all or a subset of the RCPs to coast down. The breaking or locking of an RCP shaft results in a rapid loss of forced RCS flow in one cold leg. These faults are characterized by increasing RCS temperatures and pressures. If prompt protective action is not taken, core thermal limits and/or RCS pressure boundary limits may be approached. Mitigation of these events is accomplished by tripping the reactor and opening of pressurizer safety valves. The response of the AP600 to these events is very similar to that for current Westinghouse PWRs, and the AP600 passive features do not contribute to the plant response. Either LOFTRAN or LOFTRAN-AP can be used in the analysis of these events.

A complete loss of power to the RCPs will result in a uniform coast down of all RCPs. As discussed in Section 2.1, if the flow coast down is expected to be uniform in the cold legs of a loop, then the use of lumped cold legs and the LOFTRAN internal RCS flow calculation model is adequate.

The AP600 has two electrical busses to supply power to the RCPs. Each of the two busses supplies electrical power to two RCPs. The RCPs are connected to the busses, such that the two pumps sharing an electrical buss are from opposing RCS loops. With this electrical arrangement, the following partial loss of forced reactor coolant flow events can be postulated:

- Two out of four RCPs coast down due to a buss fault. The two RCPs coasting down are on opposing reactor coolant loops.

- One out of four RCPs coast down due to a RCP fault or a breaker fault.

These partial loss-of-flow events are Condition II events and are analyzed in Section 15.3.1 of the SSAR. If the reactor is at power when these type of events occur, a rapid increase in coolant temperature and pressure occurs. The events are mitigated by tripping of the reactor and opening of the pressurizer safety valves.

The instantaneous seizure or breaking of a RCP shaft results in transients that are similar but more severe than those caused by RCPs coasting down. RCP shaft seizure or break events are Condition IV events and the analyses are presented in 15.3.3 and 15.3.4 of the SSAR. These events are also evaluated assuming ac power is lost at reactor trip, which causes the unfaulted RCPs to coast down.

The partial loss of forced reactor coolant flow events due to RCPs coasting down or RCP shaft break or seizure events can be conservatively analyzed using the original version of LOFTRAN. While these events result in asymmetrical cold leg flow rates, only the net loop flow delivered to the vessel is important in calculating whether core thermal limits are exceeded. Thus, the lumping of the two cold legs on each RCS loop together is acceptable if net loop flow rates are available from an alternate source.

Net RCS loop flow during partial loss-of-flow events are calculated using the following general procedure.

Given the following data:

- RCP homologous curves for head and hydraulic torque
- RCP friction and windage losses
- RCP inertia
- RCS pressure drop loss coefficients
- Pressure drop loss coefficients for an inoperable RCP (free spinning and locked)

The RCP speed can be calculated from:

$$dS/dt = (T_m - T_h - WIND*S^2 - FRICT\ S^{.5}) / (PUMPI/g_c) \qquad \text{(Eq. 2.2-1)}$$

where T_m = torque supplied by the motor
= 0 ft.-lb for pumps coasting down

T_h = hydraulic torque on impeller, ft.-lb

WIND = RCP motor windage loss, ft.-lb-sec^2

FRICT = Pump friction loss term, ft.-lb-sec$^{0.5}$

PUMPI = RCP rotating inertia

g_c = 32.174 lbm-ft./lbf-sec^2

S = RCP speed, radians/sec

dS/dt = transient change in pump speed

Using Equation 2.2-1 and the above data, transient RCP speed can be calculated for partial loss-of-flow events where the RCPs are coasting down. After pump speed is calculated, an iterative approach can be used. With an initial flow estimate, RCS pressure drops can be calculated from the RCS pressure losses. Given an estimate for RCS flow and the calculated RCP speed, pump head characteristics can be found from the RCP homologous curves. If the RCS flow estimate is appropriate, the RCS pressure losses will be equal to the developed pump head. This procedure is continued until a match is found between the pressure losses and the pump head. A variation of this method is used for locked rotor and broken shaft transients. For these events, the appropriate RCP pressure loss coefficient (free spinning or locked) is used for the faulted RCP instead of the homologous curves. The flow is assumed to decrease to the locked rotor or broken shaft condition over one LOFTRAN time step. This conservatively ignores fluid momentum and will underpredict flow in the faulted RCP.

Startup of an Inactive RCP at an Incorrect Temperature

Startup of an RCP when the cold leg temperatures differ significantly (AP600 SSAR Section 15.4.4) may result in excessive cooling of the core. If a negative moderator temperature coefficient exists, a power increase excursion may occur. The transient is mitigated by high nuclear flux reactor trips. This event is initiated from an at-power condition where the cold leg flows are initially asymmetric. When the RCP is started, RCS flow in the cold legs and loops approaches a symmetrical condition.

Net RCS loop flows are input to LOFTRAN as forcing functions. The initial steady-state RCS net loop flow rates with a RCP out of service are calculated by hand. Flow through the initially inactive RCP is ramped to full flow using a ramp rate that is conservative with respect to possible RCP startup times. The analysis can be performed with either LOFTRAN-AP or LOFTRAN since the modifications included in LOFTRAN-AP do not influence the response of this event.

RCCA Withdrawal at Power

An uncontrolled RCCA bank withdrawal at power (AP600 SSAR Section 15.4.2) results in an almost immediate increase in core power. The heat extraction rate from the steam generators lags behind the core power generation until the steam generator safety valve setpoint is reached. As a result, there will be an increase in reactor coolant temperature, and an increase in RCS pressure that may lift the pressurizer safety valves. Core thermal limits may be exceeded if the RCCA bank withdrawal is not terminated. The event is terminated when a reactor trip setpoint is reached (high neutron flux, overtemperature ΔT), and all RCCA are reinserted. The course of the transient is the same as for conventional PWRs; the unique AP600 features do not affect the consequences of this transient. The analysis of this transient can be performed with either LOFTRAN or LOFTRAN-AP.

Inadvertent Operation of the CMT or Chemical and Volume Control System

An inadvertent "S" signal (AP600 SSAR Section 15.5.1) will actuate the CMTs and trip the reactor and the RCPs. The PRHR is automatically actuated during the event to remove decay heat. Depending on the degree of conservatism used in the analysis, long-term inventory injection by the CMTs may increase the pressurizer level until the pressurizer is water solid. If pressurizer level becomes high, the operator will open the reactor vessel head vent valves and relief excess RCS inventory.

The inadvertent operation of the chemical and volume control system (CVS) (AP600 SSAR Section 15.5.2) can result in a similar transient. Initially, the uncontrolled borated CVS flow will cool the RCS and reduce core power until an "S" signal occurs. The CMTs will be actuated with results similar to the inadvertent operation of the CMTs. The initiation of these events is predicated on the undesired operation of the CMTs, and the events are mitigated by operation of the PRHR and possibly the reactor vessel head vent valve. The analysis of these events is performed with LOFTRAN-AP.

Inadvertent RCS Depressurization

Inadvertent depressurization analyses are presented in Section 15.6.1 of the SSAR. On previously licensed PWRs, inadvertent RCS depressurizations were postulated to occur due to inadvertent opening of pressurizer relief or safety valves. While the reactor is at power, margin to departure from nucleate boiling (DNB) limits will be reduced as the RCS pressure decreases. Violation of DNB limits is precluded by tripping the reactor on low pressurizer pressure or overtemperature ΔT. The LOFTRAN code contains a pressurizer safety and relief valve model that is used for short-term depressurization a ses to demonstrate that DNB limits will not be violated prior to reactor trip. The AP600 does not h pressurizer power-operated relief valves, however, depressurization events could be postulated to occur on the AP600 due to failures or errors affecting the pressurizer safety valves or the first three stages of ADS valves. The pressurizer safety and relief valve model in the original LOFTRAN code version could be used to analyze an inadvertent opening of the pressurizer safety valve or the any of the first three stages of the ADS system, however, the LOFTRAN model simulates an instantaneous opening of the valves. The ADS valves have opening time of 20 seconds or more, therefore, the modified pressurizer relief valve model of LOFTRAN-AP, which contains a time varying opening of valves, is used to simulate the slow opening of the ADS valves.

Steam Generator Tube Rupture

Analysis of the steam generator tube rupture event is presented in Section 15.6.3 of the SSAR. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. The AP600 design incorporates several protection systems and passive design features that automatically terminate a steam generator tube leak and stabilize the RCS. These include reactor trip, CMT actuation, PRHR actuation, pressurizer heater shutoff, CVS isolation, and startup feedwater isolation. The plant response following a steam generator tube rupture until primary to secondary break flow is terminated is analyzed with LOFTTR2-AP because of the modeling of CMTs and PRHR.

Summary

Only a limited number of the design basis accidents that are normally analyzed using LOFTRAN require the use of any of the new features that were added to the new LOFTRAN-AP and LOFTTR2-AP code version. Some events can use the original version of LOFTRAN or the special advanced plant version. Table 2-2 summaries which code version, LOFTRAN, LOFTRAN-AP or LOFTTR2-AP, can be used for a specific transient.

2.3 Analysis Methods Used With LOFTRAN

Many assumptions must be considered when applying a code to the analysis of a specific event. Many parameters important to safety analyses may have a range of possible values. These variations may be due to:

- Approximations used to calculate the input
- Approximations in the analysis code calculations
- Simplifying assumptions in the analysis
- Plant hardware uncertainties
- Cycle life effects (fuel cycle - burnup, instrument drift)

The AP600 design basis analyses performed with LOFTRAN address these variations by using a conservative bounding approach. In general, conservative ranges are used on affected input parameters. In some instances, conservatisms may also be applied to output parameters, such as the acceptance criteria. Following is a summary of the more important parameters and how they are treated in the safety analyses. Also included are discussions on other important analysis assumptions that affect the conservatism of the analyses.

Initial Conditions

The treatment of uncertainties on initial conditions used in the AP600 analyses is the same as has been used in analysis of previously licensed PWRs. For events that are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the departure from nucleate boiling ratio (DNBR) safety analysis limit values. This procedure is known as the revised thermal design procedure (RTDP) and is discussed in Reference 22.

For accidents that are not DNB limited or for which the RTDP is not employed, the initial conditions are obtained by adding the maximum steady-state errors to rated values. Initial values for core power, average RCS temperature, and pressurizer pressure are selected to minimize margin to the acceptance criteria of concern in the analysis.

Reactivity Coefficients Assumed in the Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. The use of bounding values for these parameters on the AP600 is consistent with the application in other PWR analysis submittals. In some analyses, bounding conservative values for the time in core life are assumed. In other cases, conservative combinations of parameters from different times in core life are used, although these combinations may not represent possible realistic situations.

RCCA Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the RCCA's position as a function of time and rod worth as a function of rod position. For accident analyses, the critical parameter is the time of insertion up to the dashpot entry, or approximately 85 percent of the rod cluster travel.

The simulation of negative reactivity insertion due to reactor trip is accomplished by a user input curve of reactivity versus time after release of the RCCAs.

Figure 2-1 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core, which can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to the point kinetics core models used by LOFTRAN.

There is inherent conservatism in the use of Figure 2-1, in that it is based on a skewed flux distribution, which would exist infrequently. For cases other than those associated with unbalanced xenon distributions, significantly more negative reactivity is inserted than that shown in the curve, due to the more favorable axial distribution existing prior to trip. This methodology, used for the AP600 negative reactivity insertion as a function of position, is the same as that used on previously licensed PWRs.

The time required to insert the RCCAs is a function of RCS flow. The coast down of the AP600 RCPs is faster than a shaft seal RCP used in other PWRs. Therefore, two methods are used to calculate RCCA position versus time characteristics for AP600 analyses:

a) RCCA position versus time calculated assuming full RCS flow

b) RCCA position versus time calculated assuming all RCPs are coasting down simultaneously with the insertion of the RCCAs

Method "a" is the same as is used on current PWRs for determining RCCA position versus time analysis input. On the AP600, this method is used for any analysis where some or all of the RCPs are operating at the time of RCCA insertion.

Method "b" takes credit for increased RCCA velocities resulting from decreasing RCS flow. The insertion versus time is calculated based on the transient core flow, assuming that all RCPs are coasting down simultaneously with RCCA insertion.

Position versus time curves representative of the AP600 for methods "a" and "b" are shown in Figure 2-2. Method "b" results in ~ 0.6 second reduction in the time to insert the RCCAs.

Negative reactivity insertion as a function of time for input to LOFTRAN is obtained by combining the data from Figures 2-1 and 2-2. The position versus time curve appropriate for the RCS flow conditions is used. Negative reactivity versus time curves representative of the AP600 are shown in Figure 2-3.

Protection System Setpoints and Actuation Delays

A reactor trip signal acts to open two trip breaker sets connected in series, feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the RCCAs, which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. These time delays are incorporated into the AP600 analyses.

Limiting trip setpoints are assumed in AP600 accident analyses. The difference between the limiting trip point assumed for the analysis and the nominal trip point as specified in the plant technical specifications represents an allowance for instrumentation channel error and setpoint error. This setpoint philosophy is the same as has been used on previously licensed PWRs.

Safeguards Systems Performance Data

Safety analyses are performed with bounding performance parameters for emergency safeguards systems, such as the passive residual heat removal (PRHR) heat exchangers or the core makeup tanks (CMTs). The PRHR data are used that minimize or maximize the heat removal capability of the PRHR. Similarly for the CMTs, data are used that would minimize or maximize the makeup and boration capability of the CMTs. Tables 2-5 and 2-6 summarize the assumptions used in these minimum and maximum safeguards data sets.

The selection of the minimum or maximum safeguards data sets is established on an event-by-event basis. For example, the steam line break analyses for the evaluation of core thermal margin use the CMT minimum safeguards data set to minimize boron injection but use the PRHR maximum safeguards data set to maximize the RCS cool down transient. Conversely, the analysis of the inadvertent actuation of the CMT use CMT maximum safeguards data set to maximize the overfill transient but use the PRHR minimum safeguards data set to minimize the coolant shrinkage ability of the PRHR.

Single Failures

SECY-77-439[23] provides a description of active failures, which result in the inability of a component to perform its intended function. An active failure is defined differently for different components. For valves, an active failure is the failure of a component to mechanically complete the movement required to perform its function. This includes the failure of a remotely operated valve to change position on demand. The spurious, unintended movement of the valve is also considered as an active failure. Failure of a manual valve to change position under local operator action is included.

Spring-loaded safety or relief valves that are designed for and operate under single-phase fluid conditions are not considered for active failures to close when pressure is reduced below the valve set point. However, when valves designed for single-phase flow are challenged with two-phase flow, such as a steam generator or pressurizer safety valve, the failure to reseat is considered as an active failure.

For other active equipment, such as pumps, fans, and rotating mechanical components, an active failure is the failure of the component to start or to remain operating. For electrical equipment, the loss of power, such as the loss of off-site power or the loss of a diesel-generator, is considered as a single failure. In addition, the failure to generate an actuation signal, either for a single-component actuation or for a system-level actuation, is also considered as an active failure.

Spurious actuation of an active component is considered as an active failure for active components in safety-related passive systems. An exception is made for active components if specific design features or operating restrictions are provided that can preclude such failures (such as power lockout, confirmatory open signals, or continuous position alarms).

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety-related equipment and does not include through-process errors or similar errors that could potentially lead to common cause or multiple errors.

The AP600 design basis analyses include the most limiting single active failure where one exists. In some instances, because of redundancy in protection equipment, no single failure can adversely affect the consequences of the transient. The protection system uses four independent divisions where two-out-of-four logic performs the desired protective action. A failure of a protection system division or a sensor will not prevent the desired function.

The actuation valves for the PRHR are arranged in parallel paths, such that a single failure of one of the valves to open will not cause the PRHR to be inoperable, however, this failure will result in higher pressure drops in the line. The CMT system has a similar arrangement. The pressure losses used in the minimum safeguards data sets assume that one of the parallel valves fail to open. Thus, the minimum safeguards data sets have an inherent single failure assumed in them.

Loss of ac Power

Non-LOCA and steam generator tube rupture analyses for previously licensed PWRs considered the availability of ac power to station auxiliaries. In general, ac power is assumed to be lost at the time of the fault or at the time of reactor trip and turbine trip in the transient analyses. This loss of ac power is not a single failure but is considered as a potential consequence of the event. This assumption impacts the analyses of previous PWRs in two ways: first, when ac power is lost the RCPs begin coasting down; and second, it impacts the performance of pumped safeguards systems.

Current generation PWRs rely on active (pumped) systems for emergency RCS makeup and boration and for emergency decay heat removal. These active systems require ac power to operate. If ac power to the station auxiliaries is lost, these active systems require the use of emergency diesel generators. This dependency on diesel generator power resulted in an actuation delay of active safeguards system until after the diesel generators could be brought up to speed. The dependency on diesel generators also created another possible single failure to be considered in the analysis: the loss of safeguards trains simultaneously in several systems (.i.e. a safety injection train, an emergency feedwater train, and a containment cooling train). The AP600 uses passive safeguards systems that are not dependent on emergency diesel generators in the event of a loss of ac power to station auxiliaries.

Thus, if ac power is lost there is no additional actuation delay for the AP600 passive safeguards systems, and the single failure of a diesel generator with resultant loss of multiple safeguards systems is not applicable to the AP600.

The loss of ac power to station auxiliaries also results in a coast down of the RCPs, which is also true for the AP600. However, on the AP600, an automatic RCP trip occurs whenever the CMTs are started, thus the loss of ac power tends to become unimportant in cases where the RCPs are automatically tripped.

In the AP600 non-LOCA and steam generator tube rupture analyses, the effects of the loss of ac power to station auxiliaries continues to be examined. However, the AP600 passive systems and the automatic RCP trip by the protection system tend to result in identical or very similar analyses, whether ac power is available or not.

TABLE 2-1
COMPONENTS AND FUNCTIONS OF IMPORTANCE TO NON-LOCA AND STEAM GENERATOR TUBE RUPTURE ANALYSES

Components or Functions		AP600	Current PWRs
Reactor Vessel		Cylindrical vessel with hemispherical heads; coolant flow enters through inlet nozzles, flows down through core barrel-vessel wall annulus, turns at the bottom, and flows up through the core	Same
		Inlet nozzle elevation above the outlet nozzle elevation	Inlet and outlet nozzles at same elevation
Core (fuel assemblies)		17x17 12' active length V5H fuel assembly	Same
RCS Piping		Two loops	Two, three, and four loops
		Two cold legs per loop	Single cold leg per loop
		RCP is connected directly to SG outlet plenum	RCP suction is connected to SG outlet plenum using loop seal piping configuration
Reactor Coolant Pumps		Canned motor RCPs	Shaft seal type RCPs
		Two RCPs per RCS loop (one per cold leg)	One RCP per RCS loop
Pressurizer		1300 ft.^{3} tank with cylindrical heads	Similar
		Electrical heaters	
Steam Generator		Vertical shell and U-tube evaporator with integral moisture separator; also has integral flow restrictor at steam outlet nozzle.	Similar
Emergency Safety Features	Safety feature actuation system	Digital	Analog
	Emergency reactivity insertion	Rod cluster control assemblies	Same
	Emergency makeup & boration	Core makeup tanks	Pumped safety injection
		Accumulators	Accumulators
		Direct vessel injection	Loop or direct vessel injection

TABLE 2-1 (Cont.) COMPONENTS AND FUNCTIONS OF IMPORTANCE TO NON-LOCA AND STEAM GENERATOR TUBE RUPTURE ANALYSES			
Components or Functions		**AP600**	**Current PWRs**
Emergency Safety Features	Emergency decay heat removal	Passive residual heat removal (PRHR) heat exchangers and IRWST Control-grade startup feedwater system	Pumped emergency feedwater
	Automatic RCS depressurization	Automatic fourth stage system. stages 1 through 3 connected to pressurizer; fourth stage connected to hot leg	No comparable system
	RCS overpressure protection	Spring-loaded pressurizer safety valves	Same
	SG overpressure protection	Spring-loaded safety valves	Same
	Secondary side isolation	Closure of main steam isolation valve in each steamline with backup provided by closure of turbine stop valves Closure of main feedwater isolation valve in each feedline with backup provided by closure of main feed control valve	Same Same
	RCS overfill protection	Manual reactor vessel head vent	No comparable system

TABLE 2-2
APPLICABLE CODE VERSIONS FOR AP600 DESIGN BASIS ANALYSES

Events	SSAR Section	Code Version		
		LOFTRAN	LOFTRAN-AP	LOFTTR2-AP
Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature or an Increase in Feedwater Flow	15.1.1 15.1.2	X	X	
Excessive Increase in Secondary Steam Flow	15.1.3	X	X	
Inadvertent Opening of a Steam Generator Relief or Safety Valve Steam System Piping Failure	15.1.4 15.1.5		X	
Inadvertent Operation of the Passive Residual Heat Removal System	15.1.6		X	
Loss of External Electrical Load Turbine Trip Inadvertent Closure of Main Steam Isolation Valves Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	15.2.2 15.2.3 15.2.4 15.2.5	X	X	
Loss of ac Power to the Plant Auxiliaries Loss of Normal Feedwater Flow	15.2.6 15.2.7		X	
Feedwater System Pipe Breaks	15.2.8		X	
Partial Loss of Reactor Coolant System Flow Complete Loss of Reactor Coolant System Flow	15.3.1 15.3.2	X	X	
Reactor Coolant Pump Shaft Seizure Reactor Coolant Pump Shaft Break	15.3.3 15.3.4	X	X	
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	15.4.2	X	X	
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	15.4.4	X	X	
Inadvertent Operation of the Core Makeup Tanks During Power Operation	15.5.1		X	
Chemical and Volume Control system Malfunction that Increases Reactor Coolant Inventory	15.5.2		X	
Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Opening of an ADS Valve	15.6.1		X	
Steam Generator Tube Rupture	15.6.3			X

TABLE 2-3
PHENOMENA IDENTIFICATION RANKING TABLE FOR AP600 NON-LOCA AND STEAM GENERATOR TUBE RUPTURE DESIGN BASIS ANALYSES

Component & System Phenomenon	(1) FW Malf	(2) ELI	(3) SLB	(4) Inad-vertent PRHR	(5) LOL	(6) Loss ac & LONF	(7) FLB	(8) LOSS of RCS Flow	(9) LR & BS	(10) SUIL	(11) RWAP	(12) Inad-vertent CMT or CVS	(13) RCS Dep.	(14) SGTR
Critical Flow	N/A	N/A	H	N/A	N/A	N/A	H	N/A	N/A	N/A	N/A	N/A	M	H
Vessel Mixing	H	L	H	H	L	M	M	L	L	H	L	M	L	M
Flashing in Upper Head	N/A	N/A	M	L	N/A	L	L	N/A	N/A	N/A	N/A	L	L	L
Core Reactivity Feedback	H	M	H	H	M	L	M	M	M	H	M	L	L	L
Re.‸tor Trip	H	L	H	H	H	H	H	H	H	H	H	H	H	H
Decay ‸leat	L	L	L	H	L	H	H	L	L	L	L	H	L	H
Forced Co‸vection	H	H	H	H	H	H	H	H	H	H	H	M	H	L
Natural circulation Flow and Heat Transfer	M	L	H	H	L	H	H	L	L	L	L	H	L	M
RCP Coastdown Performance	L	N/A	L	L	L	L	L	H	H	N/A	N/A	L	L	L
Pressurizer Pressurizer Fluid Level	L	L	M	M	L	M	L	L	L	L	L	M	L	M
Surge Line Pressure Drop	L	L	L	L		L	L	M	H	L	L	L	L	L
Steam Generator (SG) Heat Transfer	H	H	H	L	H	H	H	L	L	L	M	L	L	M
Secondary Conditions	M	L	H	L	L	M	M	L	L	L	L	L	L	H

TABLE 2-3 (Cont.)
PHENOMENA IDENTIFICATION RANKING TABLE FOR AP600 NON-LOCA AND STEAM GENERATOR TUBE RUPTURE DESIGN BASIS ANALYSES

Component & System Phenomenon	(1) FW Malf	(2) ELI	(3) SLB	(4) Inad-vertent PRHR	(5) LOL	(6) Loss ac & LONF	(7) FLB	(8) LOSS of RCS Flow	(9) LR & BS	(10) SUIL	(11) RWAP	(12) Inad-vertent CMT or CVS	(13) RCS Dep.	(14) SGTR
RCS Wall Stored Heat	L	L	L	L	N/A	L	L	N/A	N/A	L	N/A	L	L	M
CMT Recirculation Injection	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Gravity Draining Injection	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Vapor Condensation Rate	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Balance Line Pressure Drop	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Balance Line Initial Temperature Dist.	N/A	N/A	H	H	N/A	H	M	N/A	N/A	N/A	N/A	H	N/A	L
Accumulators Injection Flow Rate	N/A	N/A	M	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
PRHR Flow Rate and Heat Transfer	N/A	N/A	L	H	N/A	H	H	N/A	N/A	N/A	N/A	H	N/A	H

(1) FW Malf - Feedwater Malfunction that Results in a Decrease in Feedwater Temperature or an Increase in Feedwater Flow
(2) ELI - Excessive Increase in Secondary Steam Flow
(3) SLB - Steamline Break
(4) Inadvertent PRHR - Inadvertent Operation of the PRHR
(5) LOL - Loss of Secondary Side Load Events
(6) Loss ac & LONF - Loss of ac Power and Loss of Normal Feedwater
(7) FLB - Feed Line Break
(8) Loss of RCS Flow - Loss of Forced RCS Flow
(9) LR & BS - Locked RCP Rotor and Broken RCP Shaft
(10) SUIL - Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature
(11) RWAP - RCCA Withdrawal at Power
(12) Inad-vertent CMT or CVS - Inadvertent Operation of the CMT or Chemical and Volume Control System
(13) RCS Dep. - Inadvertent RCS Depressurization
(14) SGTR - Steam Generator Tube Rupture

H - High Importance M - Moderate Importance L - Low Importance N/A - Not Applicable

TABLE 2-4 PRHR PERFORMANCE DATA ANALYSIS ASSUMPTIONS	
Minimum Safeguards Data (Minimum heat transfer capability)	**Maximum Safeguards Data** (Maximum heat transfer capability)
One PRHR heat exchanger assumed[1]	Two PRHR heat exchangers assumed[1]
Maximum valve stroke times assumed	Minimum valve stroke times assumed
Maximum initial IRWST temperature	Minimum initial IRWST temperature
Minimum initial IRWST fluid inventory	Maximum initial IRWST fluid inventory
Maximum pressure loss coefficients[2]	Minimum pressure loss coefficients
Minimum experimental heat transfer coefficient	Maximum experimental heat transfer coefficient

NOTES:

1 The technical specifications allow one PRHR heat exchanger to be out of service during operation. Therefore, the minimum heat transfer case will only assume one heat exchanger. The maximum heat removal case will assume both heat exchangers are available for use.

2 The actuation valves for the PRHR are arranged in parallel paths such that a single failure of one of the valves to open will not cause the PRHR to be inoperable. However, the failure of one of the valves to open will result in higher pressure drops in the line. The CMT system has a similar arrangement. The pressure losses used in the minimum safeguards data sets assume that one of the parallel valves fail to open. Thus, the minimum safeguards data sets have an inherent single failure assumed in them.

TABLE 2-5 CMT PERFORMANCE DATA ANALYSIS ASSUMPTIONS	
Minimum Safeguards Data (Minimum CMT makeup and boration capability)	**Maximum Safeguards Data** (Maximum CMT makeup and boration capability)
Maximum valve stroke times	Minimum valve stroke times
Maximum line pressure losses[1]	Minimum line pressure losses
Maximum initial CMT temperature	Minimum initial CMT temperature
Minimum initial boron concentration	Maximum initial boron concentration

NOTES:

1 The actuation valves for the PRHR are arranged in parallel paths such that a single failure of one of the valves to open will not cause the PRHR to be inoperable. However, the failure of one of the valves to open will result in higher pressure drops in the line. The CMT system has a similar arrangement. The pressure losses used in the minimum safeguards data sets assume that one of the parallel valves fail to open. Thus, the minimum safeguards data sets have an inherent single failure assumed in them.

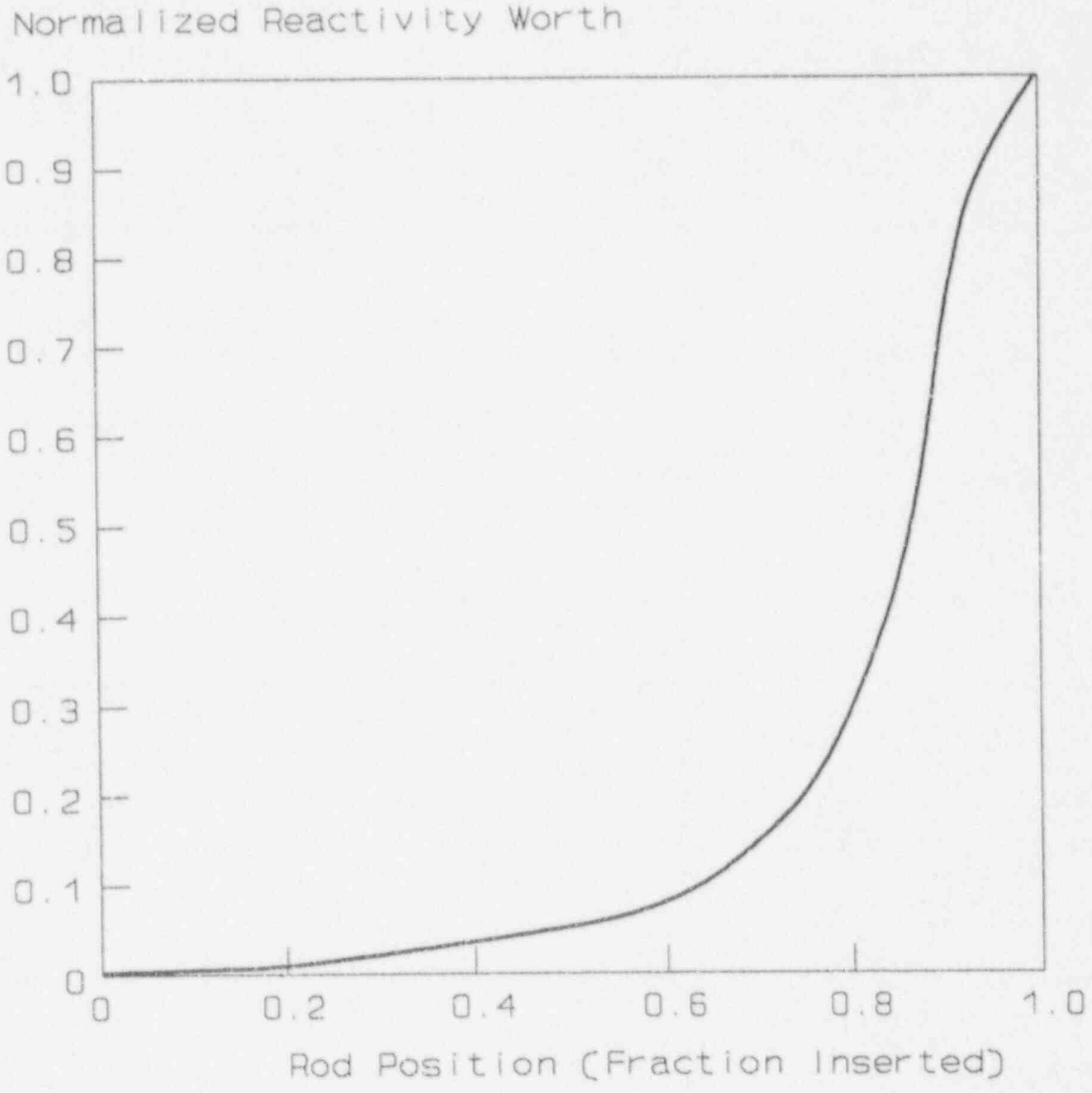

Figure 2-1 Normalized RCCA Reactivity Worth Versus Position

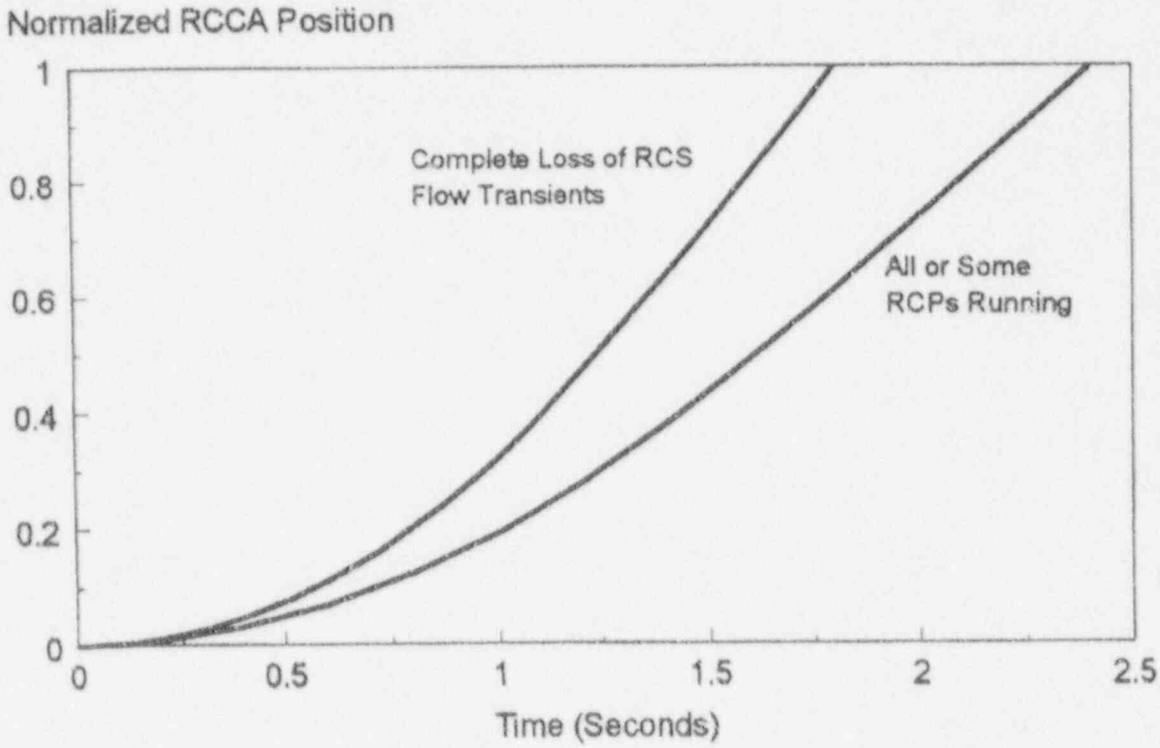

Figure 2-2 RCCA Position Versus Time

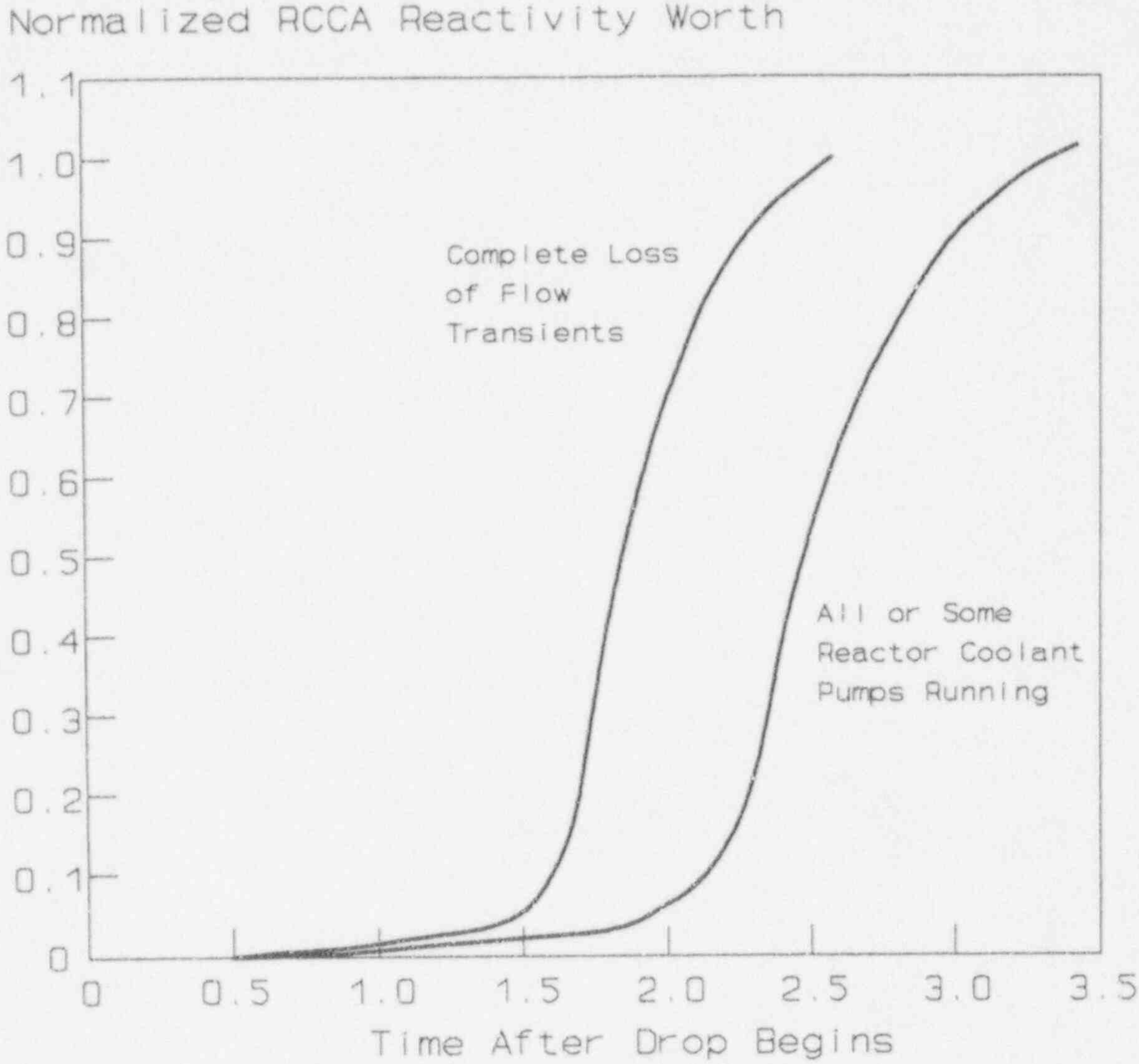

Figure 2-3 Normalized RCCA Reactivity Worth Versus Time

3.0 SPECIFIC AP600 COMPONENT MODELS

3.1 Core Makeup Tank

The core makeup tank (CMT) model is a multi-node model that simulates the tank, the balance line connecting the reactor coolant system (RCS) cold leg with the top of the CMT, and the injection line connecting the bottom of the CMT with the reactor vessel. The thermal-hydraulics model simulates the flow in the CMT lines and tracks mass, energy, and boron concentration in the CMT. The CMT model calculations are performed explicitly from the RCS thermal-hydraulic calculations. A single CMT is simulated; to simulate multiple CMTs, flow rates into and out of the CMT model are doubled.

Fluid noding in the CMT model is as follows:

- Fifteen liquid nodes in the CMT tank
- Eight nodes in the injection line
- Three nodes in the balance line between the cold leg and the CMT

Heat transfer from the tank fluid through the walls of the tank is simulated and 15 metal nodes (1 for each fluid node) are used.

Boron concentration is tracked on a node basis in the cold leg balance line and the injection line. In the CMT, boron is tracked on a tank average basis, which effectively assumes perfect mixing of the boron within the tank with fluid entering from the cold leg balance line. This assumption conservatively underpredicts the boron concentration of CMT injection.

The same slug flow model used in the main RCS loop calculations of LOFTRAN are used to transfer mass and energy from node to node in the CMT model.

Flow Calculations

Flow calculations in the CMT injection line and the cold leg connection are made expressly from the main RCS calculation done in LOFTRAN. As boundary conditions to the CMT model, the following pressures from the main RCS loop are used to calculate the CMT line flow rates:

P_{VESSEL} = pressure at CMT injection point in reactor vessel downcomer
P_{CL} = pressure in cold leg where the balance line connects

In calculating the flow rates in the injection line and the cold leg balance line, the following general momentum equation is used:

$$\frac{dw}{dt} = g_c \frac{\Delta P_{Driving} - K \frac{w^2}{\rho}}{L/A}$$

where:

dw/dt	=	rate of change in mass flow rate, lbm/sec^2
$\Delta P_{Driving}$	=	driving pressure, lbm/ft.^{2}
g_c	=	gravity acceleration, lbm-ft./lbf-sec^2
L/A	=	inertial length (length/area derived from user input volumes and flow areas), 1/ft.
K	=	pressure loss coeıııcient (user input values), (lbf/ft.^{2})/((lbm/sec) (ft.^{3}/sec))
w	=	mass flow rate, lbm/sec
ρ	=	fluid density, lbm/ft.^{3}

Driving pressure ($\Delta P_{Driving}$) is calculated using:

$$\Delta P_{Driving} = P_{CL} - BH_{BL} + BH_{TANK} + BH_{IL} - P_{Vessel}$$

In calculating the CMT line flow rates, the driving pressure (buoyancy head) in several regions (injection line, cold leg balance line, and the CMT) is used. As discussed previously, each of these regions is divided into several nodes. The buoyancy head in each region is calculated as:

$$BH_{region} = \sum_{i=1}^{n} \rho_i \, h_i$$

where:

BH_{region}	=	Buoyancy (elevation) head difference in the region		
region	=	BL	-	pressure balance line between cold leg and CMT
		TANK	-	CMT
		IL	-	injection line between CMT and vessel
ρ_i	=	fluid density in node i		
h_i	=	height of node i		
n	=	number of nodes in the region		

During non-LOCA transients, the CMT will operate in water recirculation mode. The draindown injection mode of operation does not occur during non-LOCA events. However, af.er long-term operation of the CMT, the CMT temperature will be elevated. If the RCS is depressurized to the saturation temperature of fluid in the CMT, then flashing may occur in the CMT line. The CMT model uses homogeneous nodes and stratification of the steam in the upper region of the cold leg line will not be simulated. The effect of any potential stratification is accounted for by applying a penalty to the buoyancy head of the cold leg balance line.

Boiling is detected if the water subcooling in the balance line or CMT is smaller than a prescribed user input value. Flashing is assumed to occur if the following is true:

$$Tsat - T_{node\ i} < D_{cmt\ sat}$$

where:

Tsat:	water saturation temperature at the CMT pressure
$T_{node\ i}$:	water temperature in the node i
$D_{cmt\ sat}$:	subcooling limit (input parameter)

If boiling is detected or if the subcooling limit is exceeded, the potential steam accumulation at the CMT pipe top is taken into account by a penalty on the cold leg to CMT balance line buoyancy (BH_{BL}) calculation. Assuming that there is only steam in the vertical pipe portion at the CMT top, the buoyancy is increased by the following quantity:

$$Penalty = Hbub\ (\rho_{bal} - \rho_{steam})/144$$

where:

Hbub:	equivalent height of the stratified zone. A realistic calculation may be done with the descending length of the inlet CMT pipe. A very conservative calculation may be done with a bigger value that stops the natural circulation as soon as boiling is detected.
ρ_{bal}:	mixture density in the cold leg to CMT line top node.
ρ_{steam}:	saturation steam density at the CMT pressure.

Heat Transfer Calculations

Heat transfer from the tank fluid to the tank metal wall and from the tank metal wall to the containment air is simulated. There are 15 metal nodes used for the tank wall (one metal node for each fluid node). Axial conduction between the tank metal nodes is neglected. Heat transfer is calcula... sing the following equations.

The heat transfer over a time step from the fluid CMT node to the CMT metal node is calculated using:

$$Q_{inner_i} = UA_{inner_i} (T_{Metal_i} - T_{water_i}) dt$$

Where:

Q_{Inner_i} = total energy transferred from the ith fluid node to the ith metal node over the code time step, Btu.

UA_{Inner_i} = heat transfer coefficient times the surface area between the CMT ith fluid node and ith metal node, Btu/sec-°F

T_{Metal_i} = temperature of the ith CMT metal node, °F

T_{Water_i} = temperature of the ith CMT fluid node, °F

dt = time step size, seconds

The heat transfer between the CMT metal node and the containment atmosphere is similarly calculated using:

$$Q_{Outer_i} = UA_{Outer_i} (T_{Metal_i} - T_{air}) dt$$

Where:

Q_{Outer_i} = total energy transferred from the ith metal node to the containment atmosphere over the code time step, Btu

UA_{Outer_i} = heat transfer coefficient times the surface area between the CMT ith metal node and the containment atmosphere, Btu/sec-°F

T_{air} = containment temperature, °F

The metal node temperature is computed using:

$$T_{Metal_i}(t) = T_{Metal_i}(t-dt) - \frac{(Q_{Inner_i} + Q_{Outer_i})}{MCp_i}$$

$T_{Metal_i}(t)$ = metal node temperature at time equal to t, °F

$T_{Metal_i}(t-dt)$ = metal node temperature from the previous time step at time equal to t - dt, °F

MCp_i = metal node heat capacity, Btu/°F

The metal node inside and outside heat transfer coefficient time surface areas and metal node heat capacity are code input parameters. Values may be entered for each of the 15 metal nodes. The containment atmosphere temperature is also a code input parameter.

3.2 PRHR and IRWST Model

3.2.1 General

The passive residual heat removal (PRHR) heat exchanger model is divided into the regions shown in Table 3-1. These regions include all of the PRHR inlet and outlet piping, the headers, channel heads and the heat exchangers, and up to 45 nodes can be used in these 5 regions. The inlet and outlet piping regions are simulated as vertical nodes, and the inlet and outlet header and channel head regions are simulated as horizontal nodes. The heat exchanger region is set up to model either vertical or C-tube type heat exchangers. User input allows specification of whether a heat exchanger node is vertical or horizontal, but horizontal nodes are not used in the calculation of buoyancy head in the PRHR model. Depending upon the orientation of the PRHR nodes, different heat transfer correlations are used. No heat transfer is simulated in the inlet and outlet regions. Table 3-1 summarizes the noding used in the SSAR analyses.

Heat removed from the PRHR is transferred to the in-containment refueling water storage tank (IRWST) which is IRWST is modeled as a single region. Initial IRWST conditions, such as temperature and fluid mass, are input to the model, as well as pressure as a function of time. Energy and mass are tracked in the IRWST node. Fluid in IRWST node is assumed to be a homogeneous mixture (i.e., perfect mixing is assumed in the IRWST tank). If saturation temperature is reached in the IRWST, steaming from the pool is accounted for.

3.2.2 PRHR Flow Calculations

The PRHR flow calculations are made expressly from the main RCS flow calculations. Based on a converged RCS solution, the following RCS pressures are defined:

P_{IN} = pressure at the inlet to the PRHR system where it connects to the RCS hot leg, psia (see Point A on Figure 3-1)

P_{OUT} = pressure at the outlet to the PRHR system where it connects to the steam generator outlet plenum, psia (see Point B on Figure 3-1)

These two pressures are used as boundary conditions to the PRHR model. The change in fluid flow is assumed to be uniform throughout the PRHR loop. The change in PRHR fluid flow is calculated by resolving the following momentum equation:

$$\frac{dw}{dt} = \frac{g_c}{V/A^2} \left[(P_{IN} - P_{OUT}) - \Delta P_{ff} + \Delta P_{BH} \right]$$

where:

dw/dt = rate of change of PRHR flow, lbm/sec^2

ΔP_{ff} = The overall friction and form pressure loss throughout the PRHR loop. ΔP_{ff} is calculated by summing the individual node pressure losses. ΔP_{ff} is calculated using:

$$\Delta P_{ff} = \sum_{i=1}^{n} k_i \, W \, Q$$

n is the total number of nodes in the PRHR loop. k_i is the pressure loss coefficient (psf/[gpm lbm/sec]) for node i. W and Q represent the PRHR loop mass and volumetric flow and are based on the values from the previous time step.

V/A^2 = overall inertial length (L/A) of the PRHR loop. Calculated from the input volumes and flow areas.

g_c = conversion factor=32.2 lbm ft. / lbf sec^2

ΔP_{BH} = net buoyancy driving head in the PRHR loop.

Given:

Z_{PHL}	=	elevation of the top of the PRHR heat exchanger above the hot leg (ft.)
Z_{BOT}	=	elevation of the bottom of the PRHR heat exchanger above the hot leg (ft.)
Z_{HLCL}	=	elevation of the cold leg above the hot leg (ft.)
Z_{CL2SG}	=	elevation of the PRHR outlet piping connection to the steam generator outlet plenum above the cold leg (ft.)
ρ_i	=	average fluid density in the vertical inlet piping to the PRHR (lbm/ft.^{3})
ρ_{hx}	=	average fluid density in the vertical portion of the PRHR heat exchanger (lbm/ft.^{3})
ρ_o	=	average fluid density in the vertical outlet piping to the PRHR (lbm/ft.^{3})

The pressure difference in the PRHR loop due to buoyancy head is calculated using:

$$\Delta P_{BH} = -(Z_{BOT}\ \rho_i) - (Z_{PHL}-Z_{BOT})\rho_i + (Z_{PHL}-Z_{BOT})\rho_{hx} + (Z_{BOT} - Z_{HLCL} - Z_{CL2SGC})\rho_o$$

3.2.3 Heat Transfer Models

The PRHR heat transfer model accounts for the primary-side and secondary-side heat transfer coefficients, tube metal, and deposit build-up on the primary and secondary sides. Because PRHR model uses cylindrical tubes, cylindrical geometry is used in the solution of heat transfer. The overall heat transfer coefficient is of the following form:

$$h_t = \frac{1}{\dfrac{1}{h_p\ r_i\ /\ r_o} + \dfrac{r_o\ \log(r_o\ /\ r_i)}{K} + \dfrac{1}{h_s} + FF_p + FF_s}$$

where:

h_t	=	overall heat transfer coefficient, Btu/hr-ft.^{2}-°F
h_p	=	primary-side heat transfer coefficient, Btu/hr-ft.^{2}-°F
h_s	=	secondary-side heat transfer coefficient, Btu/hr-ft.^{2}-°F
K	=	PRHR tube metal conductivity, Btu/ft.-hr-°F
r_i	=	inner radius of the PRHR tube, ft.
r_o	=	outer radius of the PRHR tube, ft.
FF_p & FF_s	=	user input primary-side and secondary-side tube fouling factors, hr-ft.^{2}-°F/Btu

The user has the ability to select various correlations for the primary and secondary side of the tubes. For the primary-side heat transfer coefficient, the user has the option to select the Dittus-Boelter or the Petukov-Popov correlations (see Appendix A).

The secondary-side heat transfer coefficient is modeled by means of a boiling curve, as shown in Figure 3-2, with the following regimes:

- Natural convection heat transfer
- Pool boiling heat transfer
- Post-critical heat flux heat transfer (transition boiling)
- Stable film boiling heat transfer

The heat transfer models used for the secondary-side of the PRHR heat exchanger are summarized in Table 3-2. Detailed equations for the secondary-side models are summarized in Appendix A.

The following steps are performed at each time step and for each section of the PRHR heat exchanger to evaluate the heat transfer:

1. Calculate primary-fluid temperature in the tube node and the local outside pressure and temperature (pool temperature)

2. Calculate primary-side heat transfer coefficient

3. Calculate secondary-side heat transfer coefficient using natural convection heat transfer correlation

4. Calculate secondary-side heat transfer coefficient and heat flux using pool boiling heat transfer correlation

5. Assume the heat transfer mode related to the higher heat flux of steps 3 or 4

6. Calculate the total heat transfer coefficient and evaluate the heat flux by applying the related ΔT

   ($\Delta T = T_{PRIMARY} - T_{POOL}$ for free convection, $\Delta T = T_{PRIMARY} - T_{SAT}$ for pool boiling)

7. Calculate critical heat flux

8. If the calculated heat flux (from step 6) is less than the critical heat flux, then the calculated heat flux is accepted and the calculation is finished; otherwise, the following steps (9 through 12)are performed

9. Evaluate the minimum temperature for stable film boiling (T_{MSFB})

10. Evaluate the outer tube wall temperature (T_{OUT})

11. If $T_{MSFB} < T_{OUT}$, then the secondary heat transfer coefficient is assumed to be the Bromley heat transfer coefficient, otherwise

12. Evaluate the secondary heat transfer coefficient by interpolating between the critical heat flux and the Bromley heat flux

Note that the above steps will produce the boiling curve shown in Figure 3-2. The film boiling heat transfer coefficient is not evaluated, but is assumed to be equal to the Bromley coefficient.

3.3 Head Vent

The reactor vessel head vent model has two available flow options:

Option a) User input head vent flow versus time
Option b) Head vent flow internally calculated based on line resistances and/or Fauske/HEM Critical Flow Model[4]

Option a

In this option, a table of head vent line mass flow rate as a function of time is input to the code. Transient head vent flow is then linearly interpolated from the table.

Option b

This model simulates the reactor vessel head vent line using three segments and is illustrated in Figure 3-3. The first segment is the line piping between the reactor vessel head and the flow orifice, the second segment is the flow control orifice, and the third segment is from the orifice to the IRWST. For each segment the flow area, pressure loss resistance and elevation are code input parameters. Constant back pressure is assumed in the IRWST and is also input to the model.

The model calculates choked flow through the orifice using the Fauske/HEM critical flow model. Orifice pressure is assumed to be at the saturation pressure corresponding to the upper head temperature plus 1°F.

The model then calculates flow using an orifice equation, based on the input line resistances, flow areas, and the pressure difference between the vessel head and the IRWST. The elevation head difference for each of the three segments is also considered.

Flow through the head vent line is then assumed to be the minimum of the predicted orifice critical flow rate or the flow rate calculated using the orifice equation.

A time table is also used to allow simulation of manual plant operator actions for opening and closing the relief valves.

TABLE 3-1
AP600 SSAR PRHR MODEL

Region	Number of Nodes Used
PRHR inlet piping from connection on hot leg to inlet header	2
PRHR inlet header piping and inlet channel heads	1
PRHR heat exchanger tubes	5 horizontal nodes at the top of the heat exchanger 12 vertical nodes 5 horizontal nodes at the bottom of the heat exchanger __ 22 total
PRHR outlet channel heads and outlet header piping	1
PRHR outlet piping from the outlet header to connection on the steam generator outlet plenum.	2

TABLE 3-2
PRHR HEAT EXCHANGER SECONDARY SIDE HEAT TRANSFER CORRELATIONS

Heat Transfer Regime	Vertical Tube Sections	Horizontal Tube Sections
Pool Convection Heat Transfer	McAdams (Section 1.3 of Appendix A) or Eckert-Jackson* (Section 1.4 of Appendix A)	McAdams correlation for horizontal tubes (Section 1.3 of Appendix A)
Pool Boiling Heat Transfer	Rohsenow (Section 1.6 of Appendix A) or PRHR experimental* (Section 1.5 of Appendix A)	Same as the vertical correlations, however correlation coefficients appropriate for horizontal tubes are used
Critical Pool Boiling Heat Flux	Griffith Correlation* (Section 1.7a of Appendix A) or Berenson correlation w/ & w/o Zuber correction for subcooling (Sections 1.7b and c of Appendix A)	Same as for vertical
Post CHF Heat Transfer (Transition boiling)	Interpolated between CHF and minimum stable film boiling (Section 1.10 of Appendix A)	Same as for vertical
Minimum Stable Film Boiling	See Section 1.8 of Appendix A	Same as for vertical
Stable Film boiling Heat Transfer	Bromley-Pomeranz correlation (Section 1.9 of Appendix A)	Same as for vertical

* Correlation options used in SSAR analyses

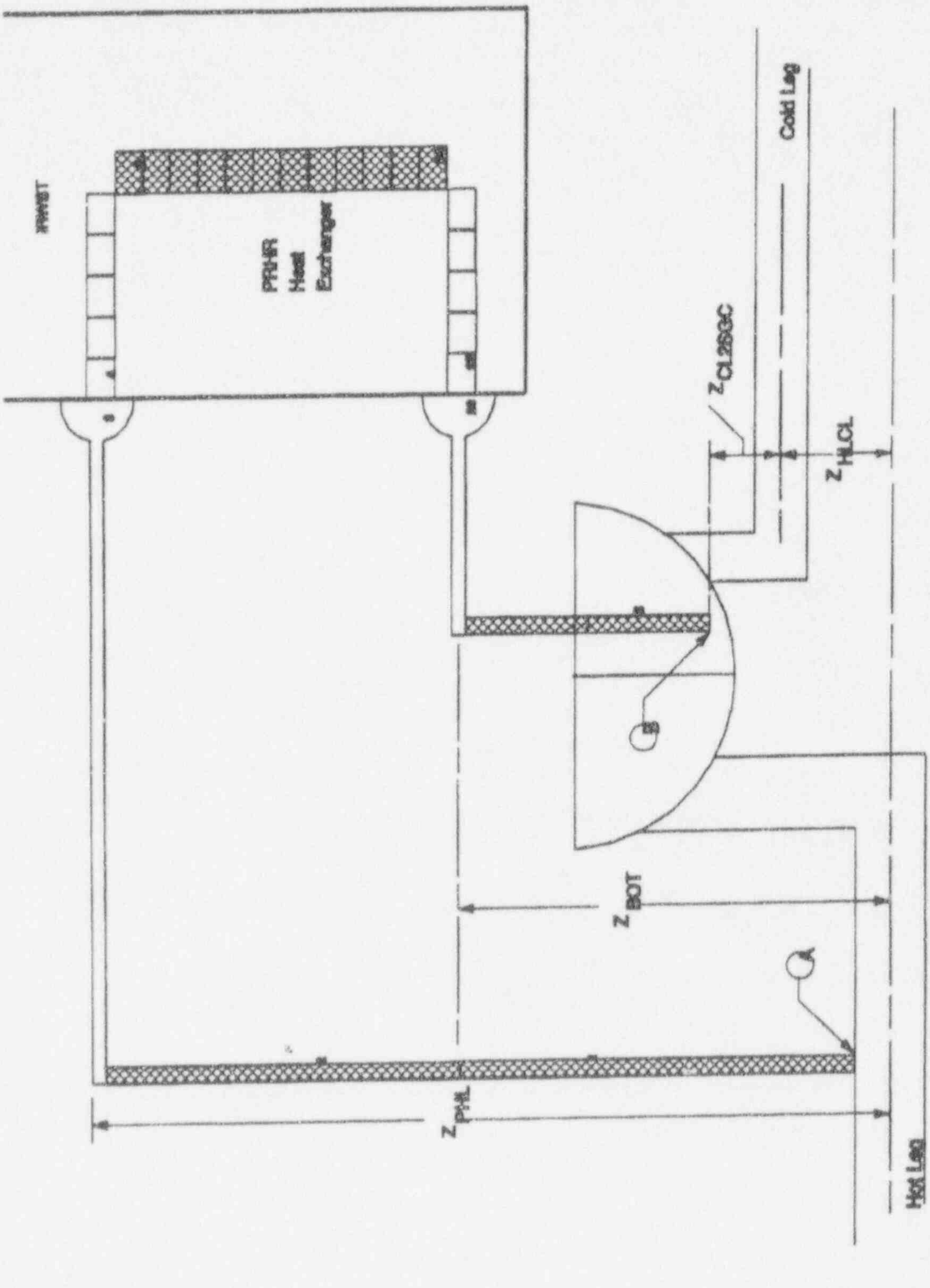

Figure 3-1 PRHR Model

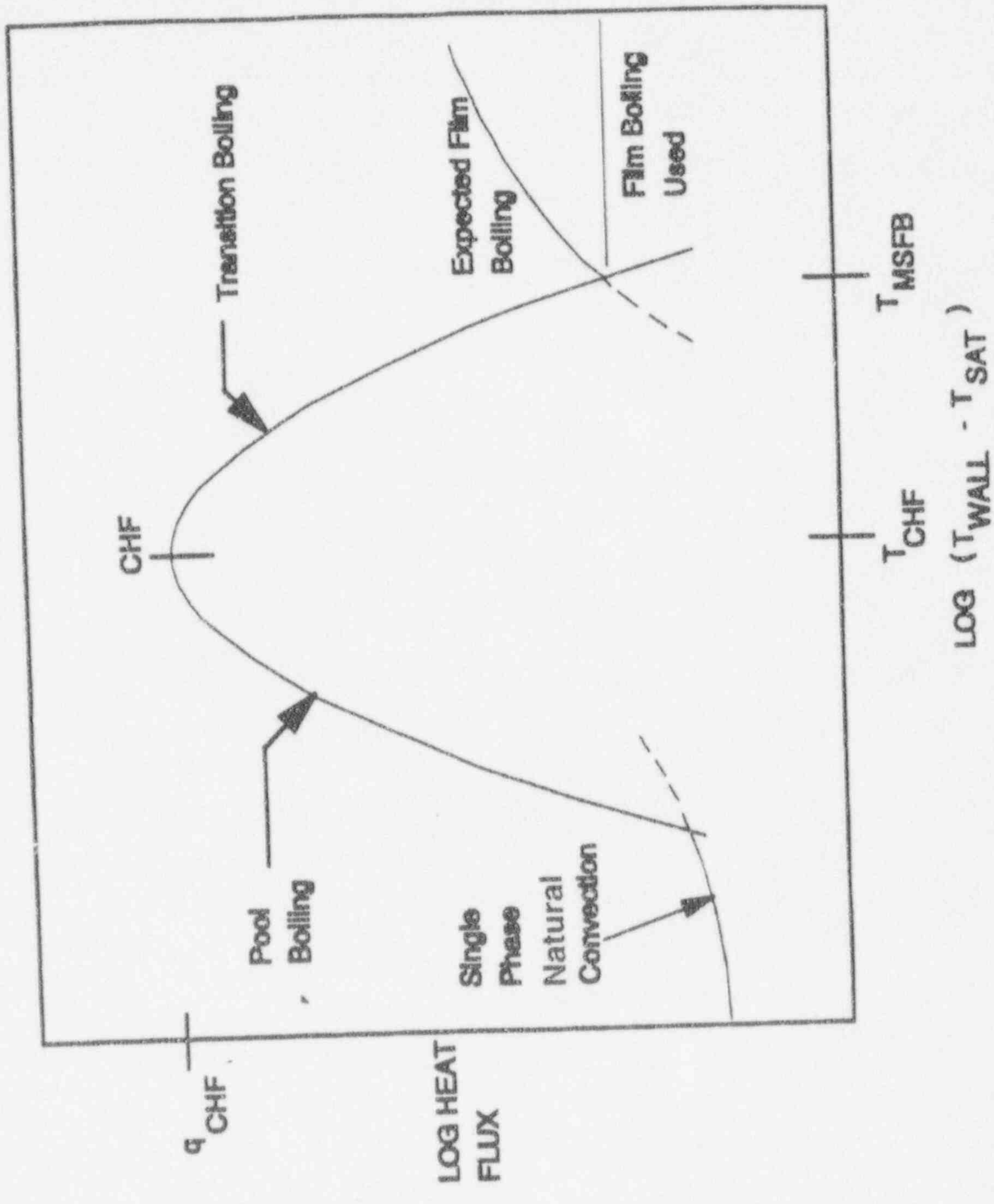

Figure 3-2 PRHR Model Heat Transfer Regions

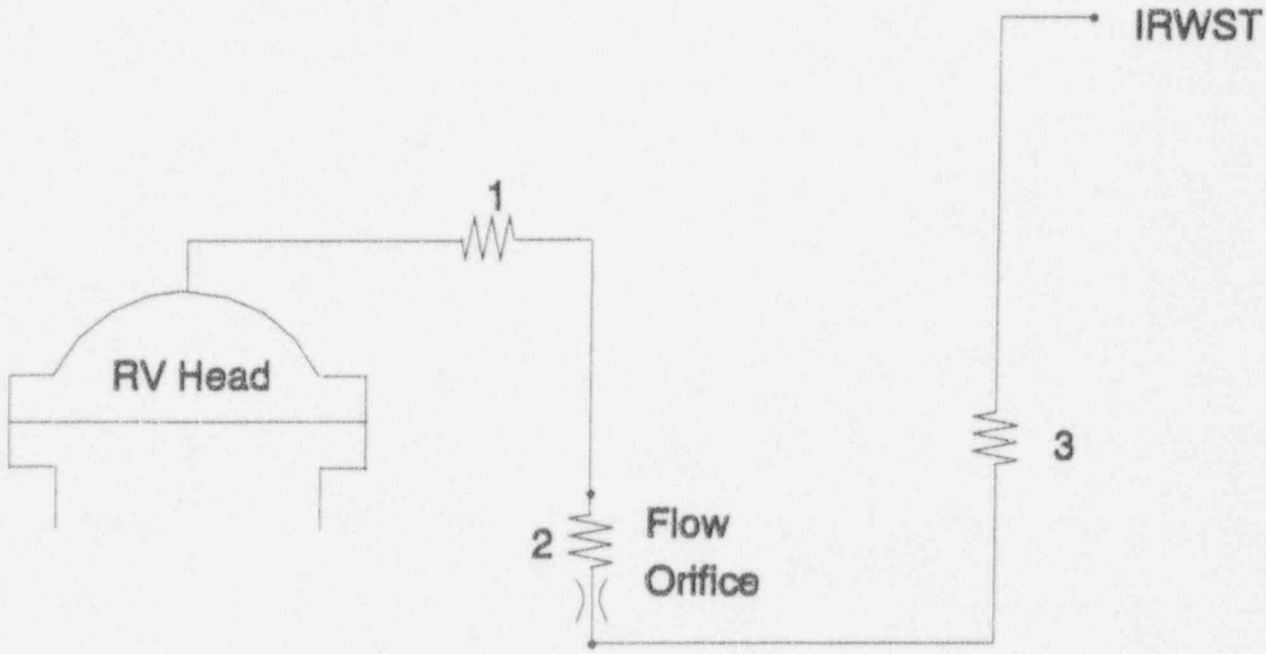

Figure 3-3 Reactor Vessel Head Vent Model

4.0 LOFTRAN VERIFICATION ACTIVITIES

The LOFTRAN-AP code version was developed by adding a PRHR model, a CMT model, and other minor changes and enhancements to the LOFTRAN code, as discussed in Section 2.1. The original LOFTRAN models, such as the core, the reactor coolant loop, RCPs, pressurizer, and steam generators remain unchanged. Validation of the original version of LOFTRAN is summarized in Reference 7 and consists of comparisons of LOFTRAN to plant data and to other thermal-hydraulic programs.

Further validation for the new LOFTRAN-AP PRHR model, CMT model, and the integral AP600 plant response with these passive safeguards systems will be based on the following tests:

- SPES-1 natural circulation tests
- PRHR component tests
- CMT component tests
- SPES-2 steam generator tube rupture and steam line break tests

SPES-1 Natural Circulation Tests

Further verification of the LOFTRAN-AP RCS natural circulation capability is performed by comparison of tests performed at the SPES-1 facility. The SPES-1 facility is a three-loop full-height facility scaled in the ratio of 1/427 with respect to a standard Westinghouse PWR three-loop plant. Scaling criteria are aimed toward natural circulation and small-break LOCA.

The AP600 LOFTRAN verification is based on test # SPNC-01, which focuses on single-phase natural circulation. The test is reported in Reference 3. This comparison has already been completed and is summarized in Appendix 15B of the SSAR.

PRHR Component Tests

The heat transfer mechanisms used in the LOFTRAN-AP PRHR model were verified by comparisons with tests done at the Westinghouse PRHR test facility. A LOFTRAN-AP model is setup to simulate the three-tube arrangement of the test facility. To simulate test conditions, the pressure losses in the PRHR are adjusted to match the required flow rate and the hot leg temperature is also carefully matched with the required boundary test conditions. This comparison has been completed, and the results are presented in Appendix 15B of the SSAR.

CMT Component Tests

Verification of the LOFTRAN-AP CMT model is performed by comparison to CMT component tests. For verification purposes, a stand-alone version of the LOFTRAN-AP CMT model has been set up that allows input of CMT test boundary conditions. During design-basis non-LOCA and steam generator tube rupture events, the CMT will exhibit the recirculation mode of injection instead of draindown mode of injection. The verification will use the CMT 500 test series, which are natural circulation tests followed by draindown and depressurization.

Results of these comparisons are presented in a Preliminary Validation Report for CMT Tests[30] and will be further documented in the Final Verification and Validation Report which will be completed in April 1995.

SPES-2 Steam Generator Tube Rupture and Steam line Break Tests

The SPES-2 test facility is a 1/395-scale full-height, high-pressure test facility. The SPES-2 test facility includes the reactor vessel loops, pressurizer, steam generators, PRHR heat exchanger, and CMTs. A detailed description of the SPES-2 facility is provided in Reference 27. For LOFTRAN-AP verification purposes, simulation of the following tests will be performed:

Test No. 9	-	Design-basis steam generator tube rupture with non-safety systems on and operator action to isolate steam generator
Test No. 10	-	Design-basis steam generator tube rupture with non-safety systems on and no operator action
Test No. 11	-	Design-basis steam generator tube rupture with manual ADS (blind test)
Test No. 12	-	Large steam line break (blind test)

Simulation of these tests will be used to further validate the PRHR and CMT models of LOFTRAN-AP. These test are full system transient tests that will validate the integrated LOFTRAN-AP reactor coolant loop models and the new passive safeguards system models.

Results of these comparisons will be presented in a LOFTRAN Preliminary Validation Report for SPES-2 Tests and will be further documented in the Final Verification and Validation Report.

5.0 CONCLUSION

The AP600 design includes differences from previously licensed PWRs that impact some of the analyses of design basis events. To address these differences a specialized AP600 version of LOFTRAN was developed called LOFTRAN-AP. LOFTRAN-AP includes new models and modifications to existing models that make the code applicable for AP600 analyses. In particular, new models for the PRHR, the CMT and the reactor vessel head vent are included in LOFTRAN-AP.

The significance of the differences between the AP600 and previous PWRs depends upon the transient being analyzed. Some design basis analyses are not impacted by AP600 unique features. For these design basis analyses, LOFTRAN or LOFTRAN-AP can be used.

For AP600 steam generator tube rupture design basis analyses, a specialized version of LOFTTR2 called LOFTTR2-AP was developed. LOFTTR2-AP includes the same modifications and new models as LOFTRAN-AP.

Validation of LOFTRAN-AP and LOFTTR2-AP code versions for AP600 non-LOCA and steam generator tube rupture design basis analyses will be performed by simulations of AP600 test facilities. The test simulations include CMT and PRHR component tests and SPES semi-scale integral system tests.

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APPENDIX A
HEAT TRANSFER CORRELATIONS USED IN THE PRHR

1.1 Dittus-Boelter Heat Transfer Correlation

The Dittus-Boelter correlation[8] is used for the PRHR primary side heat transfer coefficient. The correlation is expressed as:

$$\overline{Nu_D} = .023\ Re^{0.8}\ Pr^{0.4}$$

1.2 Petukov-Popov Heat Transfer Correlation

The Petukhov-Popov heat transfer correlation[9] is used on the primary side of the PRHR heat exchanger. The correlation is written as:

$$\overline{Nu_D} = \frac{\frac{f}{8}\ Re\ Pr}{K_1 + K_2\ (\frac{f}{8})^{1/2}\ (Pr^{2/3} - 1)}$$

where :

f	=	friction factor = $(1.82\ Log_{10}Re - 1.64)^{-2}$
K_1	=	$1. + 3.4\ f$
K_2	=	$11.7 + 1.8\ /\ Pr^{1/3}$

1.3 McAdams Heat Transfer Correlation

For free-convection heat transfer, the McAdams correlation[10] is used. Two different forms are used depending upon whether the PRHR tube section is vertical or horizontal.

Vertical

$$\overline{Nu_L} = \frac{\overline{h_c}\ L}{K} = 0.13\ (Gr_L\ Pr)^{1/3}$$

Rearranging terms, the correlation can be written as:

$$\overline{h_c} = K\ 0.13\ (g\ \beta\ Pr/\nu^2\ )^{1/3}\ \Delta T^{1/3}$$

Horizontal

$$\overline{Nu_D} = 0.53\ (Gr_D\ Pr)^{1/4}$$

where:

ΔT = outer tube wall temperature (T_{wall}) minus water pool temperature (T_{pool}), °F

The water properties are calculated at an average film temperature defined as:

$T_{avg} = (T_{wall} + T_{pool}) / 2.$

1.4 Eckert-Jackson Heat Transfer Correlation

The Eckert-Jackson Heat Transfer Correlation[11,12] is used for free convection and is written in a format similar to the McAdams correlation:

$$\overline{Nu} = \frac{\overline{h_c}\ L}{K} = 0.021\ (Gr_L\ Pr)^{2/5}$$

or:

$$\overline{h_c} = 0.021\ \frac{K}{L}\ (Gr_L\ Pr)^{2/5}$$

This heat transfer correlation is dependent on the distance from the bottom of the heat exchanger. The local heat transfer coefficient becomes:

$$h_c(x) = \frac{6}{5}\ 0.021\ \frac{K}{x}\ (Gr_x\ Pr)^{2/5}$$

In evaluating the heat transfer coefficient, the temperature difference (ΔT) is set equal to the outer wall temperature (T_{wall}) minus the IRWST bulk temperature (T_{pool}).

1.5 PRHR Experimental

A boiling correlation was developed based on Westinghouse PRHR tests which are given in Reference 13. The correlation has the following form:

q	=	$a\,\Delta T^{b}$
q	=	heat flux (BTU/hr-ft.^{2})
a	=	User input multiplier. Two different user input values for horizontal and vertical PRHR tube sections are available. Different values are used for best estimate heat transfer or to conservatively minimize or maximize heat transfer. Table 1-1 summarizes the values used.
b	=	User input exponent. Two different user input values for horizontal and vertical PRHR tube sections are available. Different values are used for best estimate heat transfer or to conservatively minimize or maximize heat transfer. Table 1-1 summarizes the values used.
ΔT	=	$T_{wall} - T_{sat}$

a,c

TABLE 1-1
COEFFICIENTS USED IN PRHR EXPERIMENTAL HEAT TRANSFER CORRELATION

	a		b	
	Vertical	Horizontal	Vertical	Horizontal

1.6 Rohsenow

The Rohsenow correlation[14] is of a similar form as the experimental PRHR correlation of 1.5 except the multiplier is a function of the pool pressure. The correlation is implemented with the following form:

$q = a\,\Delta T^{b}$ where a is a function of the pool pressure and b = 3.

$$a = f(p) = \mu h_{fg} \sqrt{\frac{g \rho_{fg}}{g_c \sigma}} \left[\frac{C_{pf}}{h_{fg} Pr^{1.7} C_{sf}}\right]^3$$

or:

$$q = \mu h_{fg} \sqrt{\frac{g \rho_{fg}}{g_c \sigma}} \left[\frac{C_{pf}}{h_{fg} Pr^{1.7} C_{sf}}\right]^3 \Delta T^3$$

where:

q	=	heat flux, Btu/hr-ft.^{2}
Pr	=	Prandtl number for saturated liquid=$C_{pf}\ \mu_f$ / K_f
C_{sf}	=	Empirical constant which depends on the nature of the heating surface/fluid combination, 0.014 for steel/water.

1.7 Critical Pool Boiling Heat Flux

The critical heat flux on the secondary side of the PRHR is calculated with one of the following two correlations:

a) Griffith Correlation[15,16]

b) Berensen CHF formula

User input is available to select the desired correlation. The Griffith Correlation is described in Section 1.7a and the Berensen correlation is described in Section 1.7b. The correlations shown are derived for saturated conditions. Section 1.7c describes the Zuber correction factor for subcooling which the user may optionally apply to the CHF correlations.

1.7a Griffith CHF Correlation

$$CHF_G = 0.9\ (1 - \alpha)\ \frac{\pi}{24}\ h_{fg}\ \rho_g^{0.5}\ (\ g\ g_c\ \sigma\ (\rho_f - \rho_g)\ )^{0.25}$$

where:

CHF_G	=	Griffith critical heat flux, [BTU / hr ft.^{2}]
α	=	IRWST void fraction

1.7b Berensen CHF formula

The Berensen CHF formula is:

$$CHF_B = \frac{CHF_G}{0.9}\left(\frac{\rho_f}{\rho_f - \rho_g}\right)^{0.5}$$

where:

CHF_B = Berensen critical heat flux, BTU / hr ft.^{2}
CHF_G = Griffith CHF, see 3.4.3.8a

1.7c Zuber Correction for Subcooling

To extend the validity of the CHF correlations to pool subcooled conditions an option is provided to add Zuber's correction[17] to the Griffith & Berensen critical heat flux correction. The correction is as follows:

$$\frac{q_{crit\text{-}subcooled}}{q_{crit\text{-}saturated}} = 1. + \frac{5.3}{h_{fg}\,\rho_{fg}}\sqrt{K_l\,\rho_l\,C_p}\;\left[\frac{g(\rho_f - \rho_g)}{g_c\,\sigma}\right]^{1/4}\left[\frac{\sigma\,g\,g_c\,(\rho_f - \rho_g)}{\rho_g^2}\right]^{-1/8}(T_{sat} - T_{pool})$$

If this equation is evaluated assuming the IRWST is at atmospheric pressure and a pool temperature of 100. °F where:

T_{sat}, h_{fg}, ρ_{fg}, ρ_f & ρ_g are the saturation properties evaluated at the IRWST pressure of 14.7 psia

K_l, ρ_l & C_p are evaluated at 14.7 psia and 100. °F

σ is evaluated at 100. °F

then the equation becomes:

$$\frac{q_{crit\text{-}subcooled}}{q_{crit\text{-}saturated}} = 1. + 0.0263\,(T_{sat} - T_{pool})$$

In LOFTRAN and LOFTTR2 this equation has been implemented as:

$$\frac{q_{crit-subcooled}}{q_{crit-saturated}} = 1. + (QSUBCR)\ (0.0263)\ (T_{sat} - T_{pool})$$

where QSUBCR is a user input adjustment factor. For SSAR Analyses, QSUBCR was set to a value of 1.0.

1.8 Minimum Stable Film Boiling

The minimum stable film boiling point (T_{MSFB}) [15] is calculated as the minimum of $T_{min}(1)$ and $T_{min}(2)$ where:

$$T_{min}(1) = T_{HN} + (\ T_{HN} - T_l\ )\ \sqrt{\frac{(K\ \rho\ C_p)_l}{(K\ \rho\ C_p)_{wall}}}$$

and

$$T_{min}(2) = T_B + 0.42\ (T_B - T_l)\ [\ \sqrt{\frac{(K\ \rho\ C_p)_l}{(K\ \rho\ C_p)_w}}\ (\frac{h_{fg}}{C_{p\ wall}\ (\ T_B - T_l\ )})\ ]^{0.6}$$

and

$$T_B = T_f + 0.127\ \frac{\rho_g\ h_{fg}}{K_g}\ [\ \frac{g\ (\rho_f - \rho_g)}{\rho_f + \rho_g}\ ]^{2/3}\ [\ \frac{g_c\ \sigma}{g\ (\rho_f - \rho_g)}\ ]^{1/2}\ [\ \frac{\mu_g}{g\ (\rho_f - \rho_g)}\ ]^{1/3}$$

where:

P	=	IRWST local pressure, psia
DP	=	3203.6 - P
T_{HN}	=	homogeneous nucleation temperature
	=	$705.44 - 4.722\times10^{-2}\ DP + 2.3907\times10^{-5}\ DP^2 - 5.8193\times10^{-9}\ DP^3$
T_l & T_f	=	set equal to the saturation temperature at local IRWST pressure, °F
$(K\ \rho\ C_p)_{wall}$	=	product of thermal conductivity (Btu/hr-ft.-°F), density (lbm/ft.^{3}) and specific heat of PRHR tube wall (Btu/lbm-°F)

$(K \rho C_p)_l$	=	product of thermal conductivity (Btu/hr-ft.-°F), density (lbm/ft.^{3}) and specific heat of saturated liquid (Btu/lbm-° F) at the local IRWST pressure
$C_{p\ wall}$	=	specific heat of PRHR tube wall, Btu/lbm-°F

h_{fg}, ρ_f, ρ_g, K_g and μ_g are evaluated at the local IRWST pressure, Btu/lbm

1.9 Film Boiling Heat Flux

The film boiling heat flux is calculated from the modified Bromley-Pomeranz correlation[15,18]

$$q = 0.62 \left[\frac{D_H}{\lambda_c}\right]^{0.172} \left[\frac{K_g^3\ \rho_g\ (\rho_f - \rho_g)\ h'_{fg}\ g}{D_h\ \mu_g\ (T_w - T_{sat})}\right]^{1/4} (T_w - T_{sat})$$

where:

$$\lambda_c = 2\pi \left[\frac{g_c \sigma}{g\ (\rho_f - \rho_g)}\right]^{1/2}$$

$$h'_{fg} = h_{fg} \left[1. + 0.4\ C_{pg} \frac{(T_w - T_{sat})}{h_{fg}}\right]$$

The saturation properties T_{sat}, h_{fg}, ρ_f, ρ_g, μ_g, and C_{pg} are evaluated at the local IRWST pressure. Surface tension (σ) is evaluated at T_{sat}. The wall temperature (T_w) is set equal to the minimum stable film boiling temperature (T_{MSFB}) which was calculated in Section 1.8.

1.10 Transition Boiling Heat Flux

The transition boiling heat flux is calculated by interpolating between the CHF heat flux and the heat flux at minimum stable film boiling . The transition boiling heat flux is found using:

$$Q_{TB} = \delta\ Q''_{CHF} + (1 - \delta)\ Q''_{msfb}$$

where:

$$\delta = \left[\frac{T_{wall} - Tmsfb}{T_{CHF} - T_{msfb}}\right]^2$$

1.11 Symbols

C_p	=	fluid heat capacity (Btu/lbm-°F)
C_{pf}	=	specific heat capacity of saturated water, Btu/lbm-°F
C_{pg}	=	specific heat capacity of saturated steam, Btu/lbm-°F
D	=	Equivalent tube diameter (ft.)
$\overline{h}_c$	=	average heat transfer coefficient (Btu/hr-ft.^{2}-°F)
g	=	gravity acceleration=4.17 x 10^8 ft./hr^2
g_c	=	conversion factor=4.17 x 10^8 (lbm-ft.)/(lbf-hr^2)
h_{fg}	=	heat of vaporization, Btu/lbm
Gr	=	Grashof number

$$Gr_L = (g\,\beta / \upsilon^2)\,L^3\,\Delta T$$
$$Gr_D = (g\,\beta / \upsilon^2)\,D^3\,\Delta T$$
$$Gr_x = (g\,\beta / \upsilon^2)\,x^3\,\Delta T$$

K	=	Fluid thermal conductivity (Btu/hr-ft.-°F)
K_f	=	thermal conductivity of saturated water, Btu/hr-ft.-°F
K_g	=	thermal conductivity of saturated steam, Btu/-hr-ft.-°F
L	=	Length of heat transfer surface (ft.)
Nu	=	Nusselt Number

$$\overline{Nu}_D = \overline{h}_c\,D / K$$
$$\overline{Nu}_L = \overline{h}_c\,L / K$$

Pr	=	Prandlt number = $C_p\,\mu / K$
Re	=	Reynolds number = $D\,v\,\rho / \mu$

T_{pool}	=	average pool temperature in the IRWST, °F
T_{sat}	=	saturation temperature, °F
v	=	water velocity in tubes (ft./hr)
x	=	distance from bottom of heat exchanger, ft.
β	=	thermal expansion factor = $-(1/\rho)(\delta\rho/\delta T)_p$
μ	=	fluid viscosity, lbm/ft.-hr
μ_f	=	viscosity of saturated water, lbm/ft.-hr
μ_g	=	viscosity of saturated steam, lbm/ft.-hr
ρ	=	fluid density, lbm/ft.^{3}
ρ_f	=	density of saturated water, lbm/ft.^{3}
ρ_g	=	density of saturated steam, lbm/ft.^{3}
ρ_{fg}	=	density difference between saturated water and steam, lbm/ft.^{3}
σ	=	surface tension of liquid to vapor interface, lbf/ft.
υ	=	μ/ρ = kinematic viscosity, ft.^{2}/hr