ENCLOSURE

SAFETY EVALUATION REPORT FOR NOTRUMP, A WESTINGHOUSE EVALUATION MODEL FOR ANALYZING SMALL BREAK LOCAs (WCAP - 10079 and WCAP-10054)

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(WCAP-10079 and WCAP-10054)

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I. BACKGROUND

NUREG-0737 is a report transmitted by a letter from D. G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors and applicants for operating reactor licenses forwarding TMI Action Plan requirements which have been approved by the Commission for implementation. Section II.K.3.30 of Enclosure 3 to NUREG-0737 outlines the Commission requirements for the industry to demonstrate its small break loss of coolant accident (SBLOCA) methods continue to comply with the requirements of Appendix K to 10 CFR Part 50.

The technical issues to be addressed were outlined in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants." In addition to the concerns listed in NUREG-0611, the staff requested licensees with U-tube steam generators to assess their computer codes with the Semiscale S-UT-08 experimental results. This request was made to validate the code's ability to calculate core coolant level depression as influenced by the steam generators prior to loop seal clearing.

In response to TMI Action Item II.K.3.30, the Westinghouse Owners Group has elected to reference the Westinghouse NOTRUMP code as their new licensing small break LOCA model. Referencing the new computer code did not imply deficiences in WFLASH to meet the Appendix K requirements. The decision was based on desires of the industry to perform licensing evaluations with a computer program specifically designed to calculate small break LOCAs with greater phenomenological accuracy than capable by WFLASH.

This SER documents the staff review of the NOTRUMP computer program for calculating small break loss of coolant accidents (LOCA). Our review concludes that NOTRUMP is acceptable for calculating small break LOCA events.

The following is our evaluation of the Westinghouse small break LOCA model using NOTRUMP.

II. SUMMARY OF NOTRUMP AND THE TOPICAL REPORTS

NOTRUMP was submitted to the NRC in a letter (NS-EPR-2681) dated November 12, 1982, from E. P. Rahe (W) to C. O. Thomas (NRC). NOTRUMP is a thermal-hydraulic computer program developed for analysis of FSAR Chapter 15 transient and accident events, as identified in NUREG-0800, the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (SRP).

The code models one-dimensional thermal hydraulics using control volumes interconnected by flow paths (junctions). The spatial and time dependent solution is governed by the integral forms of the conservation equations in the control volumes and flow paths. The thermal hydraulics account for nonequilibrium thermodymanics and apply drift flux models for calculating relative velocities between the steam and liquid phases. Reactivity feedback is modeled with point kinetic neutronics. The code incorporates special models to calculate responses of the reactor coolant pumps, steam separators, and the core fuel pins. Another significant code feature includes a node stacking capability for calculating a single mixture elevation. This eliminates unrealistic layering of steam and liquid mixture in adjacent vertical control volumes (known as pancaking effects). A two-phase horizontal stratified flow model is also included.

The Westinghouse small break evaluation model was submitted as two topical reports. The topical report WCAP 10079 describes the governing equations, their numerical solution, and addresses the code's modeling capabilities. It documents the input requirements and output capabilities. Some developmental qualification calculations with experimental data from separate-effects tests are also presented.

Separate effects and integral test comparisons for SBLOCA qualification are described in the companion report, WCAP 10054. This topical report details the Westinghouse small break LOCA model as applied to licensing calculations. WCAP 10054 describes the nodalization, the input options and the user external models applied in the licensing evaluation of a SBLOCA. In addition to the separate-effects and integral test comparisons, the topical report also addresses modeling sensitivity studies and the II.K.3.30 concerns documented in NUREG-0611.

The following describes the use of NOTRUMP, its analytical models, code validation, analytical staff audits, audit of the quality assurance procedures applied in the development of NOTRUMP, and the staff's conclusions.

III. EVALUATION OF THE NOTRUMP ANALYTICAL MODELS

NOTRUMP is a general (variable) nodalization code. Plant models are constructed from generalized control volumes (fluid and metal nodes), flow links, heat sources and heat sinks. The nonequilibrium thermodynamics and hydraulics include several drift flux options to calculate relative vapor/liquid velocities (slip). Fission heat is calculated using reactivity and reactor kinetics. The code uses the same thermodynamic water properties as used in WFLASH. The code has an extensive number of forced and natural convection heat transfer correlations covering the spectrum of the boiling curve. The flow regime maps are based on models developed by Taitel, Dukler, and Bornea and by Westinghouse. The critical flow correlations available are the Moody model, a modified Zaloudek model, and the Murdock and Baumann model. Special purpose models include flooding, bubble rise, mixture level tracking, externals which provide the user flexibility to "program" user specific modifications, a continuous contact flow link, variable area flow links, and a horizintal stratified flow model. Component models include an accumulator, a centrifugal pump, steam separators, and a fuel rod model. The user has available control volumes, flow paths and heat slabs which can be used to control pressure, enthalpies, mixture levels, mass flows and heat fluxes as a function of time. Simple valves are simulated as input flow loss coefficients.

The following summarizes some of the specific models used in NOTRUMP.

1. Field Equations

(a) Thermal Hydraulics

The thermal hydraulic models in NOTRUMP are similar to the two region nonequilibrium pressurizer models used in many of the advanced thermal-hydraulic computer programs. Each node (volume) can simulate stratified flow and two regions in thermodynamic nonequilibrium. The nodal pressure is common to the two regions. Each region is homogeneous and in equilibrium. The region containing liquid can be single or two phase. Bulk condensation and flashing within each region are equilibrium processes. Steam transfer out of the liquid phase (region) is calculated using a drift flux bubble rise model. Heat transfer between the two regions is modeled as a user specified heat transfer coefficient, while heat transfer to boundaries are modeled as a function of the stratified level within the node. The model separately calculates a local heat transfer coefficient for both the top node region ("vapor") and the bottom node region ("mixture"). Nonequilibrium effects present in homogenized situations, such as subcooled boiling, are not simulated. The fluid links (flowpaths) can either be stratified or homogeneous. The special purpose horizontal stratified flow model is documented in §7.d. Liquid entering and exiting a node through the links is deposited into the mixture region of the respective node. However, there are special exceptions to this generalized model. The distribution of the convected vapor within the node is governed by the geometry of the situation; the relative positions of the mixture level and the link. Special modifications are made in the case of the mixture level tracking model which is documented in §7.e. Either region can disappear completely as the node undergoes a transition to full of liquid or steam.

The one dimensional nonequilibrium drift flux thermal hydraulic equations consist of separate mass and internal energy balances for each region, a mixture momentum equation, plus an algebraic drift flux correlation for each of the links. The derivation of these node/link balance equations begins with the exact local instantaneous one dimensional thermal hydraulic equations and are transformed into the form of node/link equations (that is spatially discrete) by integrating over the staggered cells. The staggered mesh techniques solve the mass and energy balances at centers of each "node" cells. The momentum balance is evaluated on a "link" cell, which is displaced over by a half nodal cell.

The code applies standard node mass equations which account for time dependent (temporal) derivatives, convection and interregion mass transport.

The node internal energy equations contain terms for temporal derivatives, enthalpy transport, moving boundary, interregion energy transport and boundary heat fluxes. It neglects internal viscous dissipation which should have negligible impact on the results. All nuclear generated heat is directly deposited in the fuel pin (no gamma heating is generated in the coolant). This results in a conservative fuel pin temperature. The interregion transport terms are given by the interfacial mass and energy transfer model.

The interface mass/energy transfer model calculates the mass and energy transfer between each region of an interior fluid node and the interface between the two regions. The macro-balance of mass and heat transfer from the lower mixture region through the interface to the upper vapor region are developed for various fluid conditions such as subcooled, saturated or superheated states. For calculating heat transfer, the bulk temperature of each region and the temperature of the interface are used. At the interface, the heat transfer and mass transfer due to evaporation and condensation are also considered. If only a single region exists or if both regions are saturated, the interface mass and energy transfer rates and their derivatives are all set equal to zero. If both regions of an interior fluid node are subcooled or superheated, the interface condition is no longer saturated, and a different mass and energy transfer model is used; the mass transfer is set to zero and the heat flux is continuous across the interface.

The heat transfer rates from the region to the mixture-vapor interface can be controlled by the user. The user-supplied externals (see § 7.f) UMIFN and UVIFN are used to define the overall heat transfer coefficients from the mixture region to the interface and for the heat transfer from the vapor region to the interface, respectively.

There appears to be no general consensus and insufficient applicable data for modeling the interregion mass and energy transport terms for many of the nonequilibrium fluid states. It is very much state of the art and depends upon the nonequilibrim phenomena being considered. NOTRUMP, by formulating these terms as input heat transfer coefficients and areas which are adjustable through input, has considerable flexibility. Where significant unknowns exist, Westinghouse selects conservative inputs.

The momentum equations used to solve the individual phase junction flows consist of a mixture momentum equation for the total flow and a drift flux correlation to partition the flow between the separate phases. The equations used are one dimensional and neglect vector momentum effects. See §7.d for discussions of a horizontal stratified flow link.

The analytical point balance equations used for the drift flux thermal hydraulics are all local volume averaged equations (i.e., averaged over phase boundaries). They are macroscopically averaged over the channel cross sectional area, which means that they are "one dimensional" and that the channel profile effects have to be accounted for in "extra" terms (i.e., covariance terms), where further approximations are made. With the current state of the art, there does not appear to be a consensus about the importance of terms brought about by the local volume averaging versus the channel cross-section averaging processes. The use of a single momentum equation implies that the inertias of the separate phases cannot be treated. The model therefore would not be appropriate for situations when separate inertial effects are significant. For the small break transients, these effects are not significant.

The following terms are included in the mixture momentum balance; inertia, momentum flux, pressure differentials, gravity, wall friction and terms for expansion/contraction losses (for[illegible]). Viscous transport is disregarded as it is normally negligible. A refi[illegible] is made to the hydrostatic pressure differential drop through the node since the nodal pressures are computed at the top of the node. The terms for form losses are derived on the assumption of steady state frictionless isentropic flow through the area change. The sonic velocity is assumed constant across the area change and the homogeneous sonic velocity is used. A number of hardwired friction loss correlations are available to the user. The momentum flux term requires user input weighting factors. This is discussed in §8 on numerical techniques.

Terms relating to the various covariances are all neglected. (Covariance = the arithmetic mean of the products of the deviations of corresponding values of two quantitative variables from their respective means.) This means that the cross sectional profile effects are assumed to be unimportant relative to flat profiles for velocity, density, void fraction and pressure. Similar assumptions are made of the profile effects for the mixture energy balance equation. With this set of approximations it can be shown that the mixture momentum equation derived by Westinghouse is identical to the formulation of Ishii's one dimensional drift flux model, which is generally viewed as the current state of the art.

The final equation, the equation for the relative velocity, is the standard drift flux relationship. The drift flux model used in NOTRUMP is quite general and consists of correlations for drift velocity and distribution parameters for a number of fluid conditions. There are sixteen available options for the drift flux correlations, but only five of the options are used for the Westinghouse small break licensing model as described in WCAP-10054 (the Westinghouse Evaluation Model methodology report). This review will therefore limit itself to the options used for licensing analysis. These are, in the terminology of the user input description, IDRFTFN equal to #1, #3, #12, #13, and #15. All options are checked for flooding conditions and the flowrates are concurrently limited by the selected flooding correlation, which is user input. The following addresses the drift flux options in greater detail.

(i) Zuber Correlation Limited by Flooding. (IDRFTFN=1)

The Zuber correlation for churn-turbulent bubbly flow, limited by the Wallis flooding correlation, is used with the flux-weight void fraction approach for the closure relationship. Closure relationships are discussed in §7 on numerical technique. The distribution parameter C_o, as applied in options #1 and #3, is justified only for conditions where the void fraction profile is flat; such as in the case of a large vessel. However, Westinghouse based its justification for using this model with proprietary loop seal separate effects tests (described in WCAP- 10054 and reviewed in §V on code qualification). This is the strongest possible basis, since clearance of the loop seal plays a major role in SBLOCA analysis. Therefore, the assumption is well justified vis-a-vis the loop seal data.

(ii) General Bubble Flow (IDRFTFN=3)

This model was introduced by Zuber and Findley and is based on the method which accounts for the void fraction effects using the terminal rise velocity. The flux-weight void fraction approach is used. A table representation method is used as a means of avoiding an iterative numerical solution procedure. The table is, however, recalculated each time step since V_{gc} is a dynamically changing quantity. This option is used only on the steam generator secondary side in the small break analysis. To assure the validity of this application, the bubble diameter should be on the order of 10^{-1} to 2 cm. As long as steam generator tube uncovery (concurrent with a severe depressurization rate) does not occur, this option is acceptable. Should such conditions occur (e.g., large steam line breaks), additional justification of the model (in a best-estimate mode) would be required.

(iii) Improved Version of TRAC Vertical Flow Model (IDRFTFN=12)

Based on the TRAC-PF1 flow regime map, this model utilizes three different drift-flux correlations and uses the void propagation approach for the closure relationship. The bubble/churn turbulent flow regime and the annular flow regime correlations were derived and based on the work of Ishii. The correlations compare well with data from a wide range of tests. For slug flow, the velocity correlation was derived theoretically by Davis and Taylor/ Dumitresan (for zero viscosity slug flow). Westinghouse does not propose to use this option for small break analysis. However, as a countercurrent flow regime map does not yet exist in the NOTRUMP code, for countercurrent flow conditions, the code is programmed to automatically switch to this model even if the fourteenth and the fifteenth models were originally chosen by the user.

(iv) Simplified Yeh Correlation (IDRFTFN=13)

This is based on the Yeh void fraction correlation which was developed for predicting the void fraction in a vessel. The data base for the development is the Westinghouse small break LOCA G-2 tests and is restricted to co-current vertical flow conditions. Westinghouse uses this correlation for the core model in the small break analysis. The correlation was also compared with ORNL small break tests which are described in WCAP-10054 and reviewed in §V. The void propagation approach is chosen for the closure relationship. Application of the Yeh correlation to flow conditions other than for co-current vertical flow is questionable as the database used for the development of the Yeh correlation does not include such data. However, there is, to our knowledge, no other data available for core boildown. Thus the Westinghouse approach defines the current state of the art. Based upon validation calculations with integral tests, we find the model acceptable.

(v) Flow Regime Dependent TRAC-P1 Vertical Flow Model (IDRFTFN=15)

This correlation is similar to the twelfth model except that the vertical flow regime map is based on the work of Dukler, et al., instead of the TRAC developers. This model is limited to co-current vertical flow and will automatically switch to the twelfth model if countercurrent flow occurs. This correlation is used in the small break analysis with the void propagation approach. We find this model acceptable.

This correlation, with a special flooding model is also applied at the hot leg to steam generator inlet plenum connection. The model is based on Westinghouse proprietary separate effects tests. We find this model acceptable.

The staff finds the model of the system hydraulics acceptable. This conclusion is based upon a review of the models, upon the code verification submitted by Westinghouse and upon audit analyses, to be described later.

(b) Thermal Conduction Model

NOTRUMP employs the following conduction models: a simple heat slab and a detailed fuel pin. The fuel pin model is reviewed in §5.d. Heat conduction in heat slabs is simulated in one dimension using a one node approximation to the exact Fourier equation with temperature dependent thermal properties and no heat sources. Heat transfer is to the fluid only. The conduction equation is simulated by the use of steady state resistances for cylindrical geometries. This is therefore a quasistatic approximation. UAs for the two sides of the heat slabs are proportioned according to the fluid node mixture level. The effective flow area for heat transfer and the equivalent diameter are recalculated at each step of the transient rather than assumed fixed. There is a special boundary heat slab option where the slab temperature is an input.

The staff concludes that the thermal conduction model used in NOTRUMP is acceptable. Our conclusions are based upon the technical derivation of the governing equations and upon independent audit analyses described later in this report.

2. Material Properties

Heat slab thermal properties are user input to NOTRUMP. The fuel rod thermal properties are programmed into the code. These include density, conductivity, heat capacity and emissivity for UO_2, ZrO_2 and Zircaloy-4. The properties are taken directly from the LOCTA-IV fuel pin model, which was previously approved. The mechanical properties, such as the Young's modulus and Poisson's ratio for clad and the linear thermal expansion coefficients for both fuel and clad are taken from the Westinghouse SBLOCA code, WFLASH.

The thermodynamic properties of water are from WFLASH. The water transport properties are from SATAN. Both SATAN and WFLASH have been previously approved for LOCA analysis. The material properties are therefore acceptable.

3. Heat Transfer

This section only addresses the heat transfer correlations for the heat slab conduction model. The correlations used in conjunction with the fuel pin model are evaluated in §6.d as part of that model. Nodal interregion heat transfer coefficients are external input. The heat transfer from the slab to the fluid is based upon the boiling curve. The correlations used are shown in Table 1 Proceeding down Table 1 is functionally equivalent to going up the boiling curve. When T_{wall} is used to switch heat transfer regimes the code selects the correlation which gives the lowest temperature at the slab-fluid boundary (wall temperature). This switching criterion is equivalent to selecting the largest heat transfer coefficient and is logically consistent with the form of the boiling curve. It has been effectively applied in other system transient codes. The selection between forced convection and natural convection is also based on this criterion and not on mass flux rates. Although most of the correlations are extensively used in reactor system transient analysis, a number of them are Westinghouse specific correlations. The Sandberg correlation and the Westinghouse transition boiling correlation are employed in the previously approved Westinghouse LOCTA code for LOCA calculations.

The transition heat transfer correlation, Dougall/Rohsenow, and the MacBeth CHF correlation were approved in Appendix K to 10 CFR 50. As observed from the Oak Ridge experimental data, the Dougall/Rohsenow correlation was found to be nonconservative for calculating peak clad temperatures. When modeling the heat transfer across the fuel rod, the "Westinghouse" transition boiling heat transfer correlation is used in its place (See §6d). Westinghouse provided justification for the use of the Thom correlation for low flow conditions expected during a SBLOCA. The Heineman correlation was derived from ANL data. In the SBLOCA range of interest it gives values which are essentially identical to the Dittus-Boelter correlation. Separate correlations for the saturated and superheated regimes can be input through the use of the user externals.

Correlations for condensation heat transfer during steam and two-phase conditions are also modeled in NOTRUMP. For two-phase condensation, an empirical heat transfer correlation, developed by Shah, is applied. This correlation is based upon a large number of tests under a wide range of conditions expected during a SBLOCA. The steam only condensation heat transfer coefficient was based upon Westinghouse proprietary experiments on a 16-tube PWR steam generator model. This correlation was based upon 98 tests.

During this review Westinghouse revised and improved the heat flux derivatives required for Porsching's scheme for the natural convection regime. These derivatives account for certain previously neglected pressure and temperature dependencies.

The above heat transfer correlations are acceptable as implemented.

4. Friction Correlation

This section reviews the constitutive relationships for the momentum balance. The correlations used to model the mixture equation are evaluated first. Afterwhich, the correlations for the interphase coupling are addressed.

The wall friction factor used in NOTRUMP is the standard D'Arcy factor with a multiplier to account for two phase effects. The code models eight different single phase correlations to handle flow inside a tube, normal cross flow, parallel flow outside the tubes, cross flow at the steam generator tube U-bend region, and valve flow (requiring user-supplied flow coefficients). The code does not apply laminar flow correlations. Most of these correlations were obtained from open literature and have been commonly used in industry. Others are proprietary Westinghouse correlations.

All form losses are input as constant coefficients which are selected to match steady-state pressure drop data. The coefficients remain constant throughout the calculated transient. The variable loss coefficient option model is not used for licensing calculations. Westinghouse stated that it does not intend to use the hardwired U-bend crossflow loss correlation for SBLOCA analyses. If in future applications friction loss models 1, 2, 5, 6 and 8 are applied, further justification will be required.

Table 1. Slab Heat Transfer Map

Boiling Regime		Forced Convection		Natural Convection	
		Subcooled	Saturated	Subcooled	Saturated
Switches on T_{wall}	Single Phase	Dittus-Boelter (W correlation for cross flow)		McAdams	
	Nucleate Boiling	Thom* or Jen-Lottes	Shrock-Grossman forced convecation vaporization for $\alpha >$ VFCFCV		
CHF†		MacBeth	MacBeth		
Switches on T_{wall}	Transition	W Correlation			
	Film	Sandberg	Dougall & Rohsenow		
Switches on T_{wall}	Superheat	Heineman		McAdams	
Condensation Regime		Shah		Westinghouse	

* ≡ Used in licensing calculations.

†CHF ≡ Critical Heat Flux.

The two phase multiplier used is the Thom modification of the Martinelli-Nelson correlation. This model is acceptable per 10 CFR Part 50 Appendix K for LOCA analysis at pressures above 250 psia.

Interphase constitutive relationships are required in only one special case. This is when the drift flux thermal hydraulics (the drift flux correlations are reviewed in §1.a) are not used and the horizontal stratified flow model is specified. For this particular model, the code applies the Wallis interphase friction factor correlation for annular flow assuming a smooth interface.

Finally a number of flow regime maps are also modeled. These are the cocurrent horizontal flow map of Taitel and Dukler (1976) and the cocurrent vertical upward flow map of Taitel, Bornea and Dukler (1980). These maps are current state of the art and compare reasonably against data. Westinghouse has, however, extended the upflow concepts of Taitel, Bornea and Dukler to derive a flow regime map for vertical cocurrent downflow. The data base for this map is limited, as are those for all flow maps, but this extension appears appropriate.

Westinghouse stated that their SBLOCA licensing model does not use a horizontal flow regime map. The onset of cold leg coolant stratification is determined by the loop seal clearance criterion. This criterion is based upon scale model testing performed by Westinghouse and calculates onset of loop seal clearing when the mixture level in the upstream vertical section drops to the appropriate elevation of the horizontal section of the U bend. This criterion is not hardwired but is an automatic consequence of the specific nodalization and models Westinghouse applies for the loop seal. For the hot leg, stratification is assumed to occur upon appropriate vapor and flow conditions. This is justified on grounds that the time between hot leg vapor formation and hot leg inlet uncovery is short during pumps-off conditions.

The staff finds the flow regime maps and the friction correlations acceptable.

5. Critical Flow

The break flow models are based on donor stagnation pressure, recipient stagnation pressure, and donor stagnation enthalpy. The Moody choked-flow correlation is used for a saturated donor stagnation state, and a modified Zaloudek critical-flow correlation or the orifice equation is used for a subcooled donor stagnation state. The Murdock and Baumann correlation is used for a superheated donor stagnation state. A weighting procedure is used to provide a smooth transition between choked and unchoked flow. The code user has the flexibility to input break flow multiplier coefficients.

The critical flow models used in NOTRUMP are acceptable.

6. Component Models

(a) Accumulator

The accumulators are modeled as special type of fluid nodes. An accumulator fluid node consists of an upper "nitrogen" region and a lower "mixture" region of water. The behavior of the nitrogen is assumed to obey a polytropic expansion law. A realistic polytropic coefficient (based on full scale accumulator tests) is specified by the user. No mass and energy transport is calculated at the interface between the two regions. The "Newton-Raphson" method is used to determine a system pressure with the node volume held constant. The "pressure search" is based on the two regions sharing a common system pressure with the constraint that the sum of the volumes of both regions is equal to the total volume of the accumulator. When two-regions are present, the thermodynamic properties for the water are determined by solving a single equation, given the pressure, the internal energy, the fluid mass of the lower region, the initial reference pressure, the volume of the node, and the volume of the upper nitrogen region. For the one region case, as would occur when the accumulator is empty, the pressure is calculated by the polytropic expansion law.

We find the NOTRUMP accumulator model acceptable. This approval of the thermal-hydraulic model for system pressure and thermodynamic states is based on technically acceptable analytical equations and applicable experimental validation.

(b) Separator

NOTRUMP has two swirl vane models and a Peerless vane model available for simulating the steam generator. One swirl vane model is used only when the flow direction for the vane inlet, outlet, and drain are the same, as occurs during normal power operation. Westinghouse, however, has stated that the separator models are not used in their SBLOCA analyses. If in future SBLOCA applications the separator model is used, we will require further justification for the model.

(c) Pump

The pump model is 'ar to the SATAN VI model, which has been previously approved for licensing application. The model applies the standard four quandrant homologous torque/head curves normalized to rated conditions. Under two-phase flow conditions, an empirical "equivalent density" is used with the single phase homologous curves to determine the pump head. This model is applicable during subcooled donor volume conditions through superheated conditions. The hydraulic torque and friction losses are calculated as a second order polynomial in the pump model. Electrical torque is taken to be zero during coastdown. The equivalent density formulation allows analysis of locked rotor and zero flow situations. The calculated head is included in the mixture momentum balance equations; the hydraulic torque is corrected for density changes.

The staff finds the pump model in NOTRUMP acceptable.

(d) Fuel Rod Model

The fuel rod is modeled using multinode radial heat conduction equations through the fuel rod with an explicit gap conductance. Axial heat conduction is not modeled. The materials properties used are reviewed in § 2.

Heat generation in the clad, due to metal-water reaction, is based on the Baker-Just equation, with no limitations for steam availability.

The fuel gap gas pressure is calculated using the perfect gas law. The analysis considers not only the plenum and gap volume but also four sources of voids within the fuel pellets. The gap pressure, as well as differential fuel-cladding thermal expansion, are used to calculate the gap width. From this, together with the gas thermal conductivity (as affected by the input gas composition), the gap conductance is calculated. Thermal radiation effects are also included in the gap conductance equation. A separate fuel code that accounts for densification, fuel swelling, and cladding creep, provides the volume and gas pressure input at the beginning of a transient. Rod-to-fluid heat transfer coefficients are calculated using the following correlations: subcooled fluid, Dittus-Boelter (forced convection) or McAdams (natural convection); nucleate boiling, Thom; critical heat flux, W-2, W-3, or MacBeth, or GE transient CHF (the W-2 and W-3 correlations are used for licensing evaluations); transition boiling, "Westinghouse", or combination of subcooled film boiling, Tong, Bishop and Sandberg with saturated film boiling, Dougall and Rohsenow, or stable film boiling, Groeneveld; and steam cooling (including radiation), Yeh, et al. Westinghouse stated that in the post CHF regime the Dougall-Rohsenow correlation is not used. The Westinghouse correlation is used in its place. This correlation was previously approved for application in the LOCTA code. The staff therefore finds the "Westinghouse" post-CHF correlation acceptable in NOTRUMP.

Table 2 tabulates the heat transfer correlations used with the fuel rod heat slab model. The fuel pin model is not identical to the heat slab model (Table 1). With the exception of the natural convection and steam cooling heat transfer correlations and the deletion of clad axial heat conduction, the fuel pin model (thermal conduction, mechanical stress calculation, thermal properties, etc.) was abstracted from the LOCTA-IV code. The transition heat transfer correlation is the same as the one programmed in the LOCTA-IV code. LOCTA-IV was previously approved for loss of coolant transient analysis. The mechanical properties were obtained from WFLASH, which was also approved for LOCA analysis. Deletion of clad axial heat conduction maximizes the peak clad temperature. This is therefore acceptable.

Table 2. Fuel Pin Heat Transfer Map

Boiling Regime		Forced Convection		Natural Convection	
		Subcooled	Saturated	Subcooled	Saturated
	Single Phase	Dittus-Boelter		McAdams	
Switches on T_{wall}	Nucleate Boiling	Thom			
CHF		W-2/W-3* or	MacBeth or GE		
	Post CHF	Sandberg, Dougall & Rohsenow or Groeneveld or W Transition Correlation*			
	Steam	Yeh, et al.			

*Used in licensing calculations

The McAdams correlation is applied to natural convection. The Yeh correlation for the steam cooling region is based on experiements using bundle geometry, prototypic thermal hydraulic conditions and corrects for spacer effects.

The staff finds the NOTRUMP model for the fuel rod acceptable for small break analysis.

7. Special Purpose Models

(a) Bubble Rise

NOTRUMP applies a drift-flux model to calculate phase separation within a node. This bubble rise model assumes that the liquid velocity is small at the interface. The same correlations used in the drift-flux model are also applied in the bubble rise model to obtain C_o and $\langle\langle V_g \rangle\rangle$. If the velocity of the interface is small relative to $\langle\langle V_g \rangle\rangle$, it can be further assumed that the liquid velocity at the interface is relatively small and that it is equal to the velocity of the interface between the regions which are spatially homogeneous. The slip velocity at the interface (used to compute the interface bubble mass flux rate) can then be approximated by the local relative velocity (i.e., $\langle\langle V_f \rangle\rangle - \langle\langle V_g \rangle\rangle$ at the interface between the lower and upper region). The NOTRUMP bubble-rise model is well suited for a large vessel geometry. In fact, since the term of $\langle\langle V_{gj} \rangle\rangle$ is evaluated at the known void fraction α_{mix} for the bubble rise model, the application of the drift flux correlations to the bubble rise model is simpler than the corresponding application to the flow links. The interface mixture mass flow is approximated as the algebraic sum of the vertical flow into the mixture region.

Based upon validation with applicable experimental data and previously approved models (WFLASH), we find the bubble rise model in NOTRUMP acceptable.

(b) Flooding Model

The NOTRUMP flooding correlations are written in terms of the Kutateladze numbers. Two correlations are mentioned in WCAP-10079. The first one is a general correlation with the form of the Wallis correlation. The constants, k_1, m and n were determined using Aerojet Nuclear Corporation flooding test results. The value for k, which fits the test data best, is 4.17. However, a significantly different value is applied based on the Pushkina and Sorokin results. This value gives results which are more realistic. The second flooding correlation (PWS 2.3) is based on Westinghouse proprietary experimental data from a large scaled test facility with PWR geometry for the hot leg and steam generator inlet plenum. The experiments were conducted by Commissariat a L'energie Atomique, Electricite de France, FRAMATOME and Westinghouse. The tests were conducted since traditional correlations for flooding in vertical tubes would not necessarily be applicable for hot leg and steam generator inlet plenum geometries. For example, the Richter flooding correlation (a Wallis type of flooding correlation) underpredicts the French experimental data.

For the small break model described in WCAP-10054, Westinghouse applies both the non-geometry-dependent Kutateladze flooding criterion and the geometry specific Wallis correlation in selected locations throughout the reactor coolant system. Westinghouse justified the use of the Kutateladze model through comparisons against a series of test data for a diameter size range larger than 0.4 inches. Taitel and Dukler have independently derived the same relation theoretically. The use of Kutateladze model, therefore, has its theoretical base. The Wallis correlation has a tendency to predict smaller values of gas velocity as the film thickness increases at the flooding point. The flooding velocity is affected not only by tube diameter and the physical properties of liquids and gases, but also by the geometries of tube ends. In addition, the effect of tube length become significant in tubes 1.5-3 m in length as the film flow rate is increased. It is not significant in tubes shorter than 1.5 m. Neither correlation accounts for these factors. The Semiscale test S-UT-08 indicated significant liquid hangup in the steam generators. This is strongly influenced by the flooding and heat transfer criteria. Westinghouse has validated its model to the Semiscale S-UT-08 experimental data. This comparison (against the steam generator inventory hangup) demonstrated the adequacy of the NOTRUMP flooding correlation for steam generator modeling.

(c) Point and Continuous Contact Flow Link

The point and continuous contact flow link models (junctions, in RELAP terminology) determine the flow composition and the relative position of the appropriate nodal mixture level and the flow link elevation at the ends of the node. The point contact model assumes a point contact between flow link and node. In the case of the continuous contact flow link model, the flow geometry is assumed to be a circular pipe which contacts the side of a fluid node. The point contact model leads to discontinuity in the void fraction and static quality as the mixture elevation crosses the flow link elevation. Numerical oscillatory behavior could occur, but often can be eliminated or minimized by reducing the time step. The point connection model is more appropriate for vertically oriented links such as those connecting the nodes of a stack. The model could be used for links through which a mixture elevation is not expected to pass or is expected to pass through quickly.

The continuous contact models were developed for horizontal links representing large diameter pipes or flow channels. The standard continuous contact model is not appropriate for vertical flow, particularly in the case of a break flow link. Sophisticated models are available in NOTRUMP to account for vapor pull through and liquid entrainment at the break. The vapor pull-through and liquid entrainment models, used in conjunction with the continuous contact flow links, are based on recommendations made in the study of Crowley and Rothe (CREARE, Inc.). Crowley and Rothe compared existing models against low pressure small break pipe data obtained by CREARE. Data in this area is very limited. The following correlations were recommended by CREARE and were incorporated into NOTRUMP.

For vapor pull-through during draining from the node bottom or the node side, the correlation of Lubin and Hurwitz for vortex flow is used. For the case of liquid entrainment in a side break, a model derived analytically by Craya and also verified experimentally by Gariel is used. Finally, for the onset of entrainment from a top break, the correlation of Rouse is used. Based on these correlations, the elevation for vapor pull-through, or liquid entrainment can be obtained for each case.

There appears to be very little data to validate these models. The CREARE data was obtained at low pressures ($\sim$ 40 psi) for small diameter pipes ($\sim$ 3") and subcooled fluid conditions. The effects of flashing and scaling are uncertain. Westinghouse stated that the model is only used for a very short period during a SBLOCA analysis. This is because the draining of the horizontal pipe section occurs very quickly. These models were incorporated into NOTRUMP to more realistically characterize the behavior of the fluid conditions exiting the break. Based on sensitivity studies, the impact of this model had negligable influence on the outcome of the analysis. We find this model acceptable for licensing applications.

(d) Horizontal Stratified Flow

NOTRUMP's horizontal stratified flow regime model consists of two separate flow components: a lower liquid component and an upper vapor component. To model this condition, a pair of flow links (or junctions) are applied, one link represents the lower (liquid) component and the other link the upper (vapor) component. The two links share the same upstream node, downstream node, upstream elevation, downstream elevation, non-zero continuous contact flow diameter and length to assure consistency with the physical conditions. The geometry used for the model is a horizontal circular pipe.

Collapsed liquid levels are calculated from the liquid mass balance at both the upstream and downstream ends of the pipe. The upstream and downstream void fractions for a link can be determined from the collapsed liquid level to have either the value of zero or one. The variable flow areas, inertial lengths, and equivalent diameters of a link can then be computed accordingly.

The model can calculate co-current flow as well as countercurrent flow. With one exception, the momentum balance in both links is treated identically to the momentum balance in a regular flow link. The upstream and downstream static pressures are calculated using quasistatics in the transverse direction and includes the hydraulic head. The one exception occurs for the momentum equation for stratified flow. This accounts for both wall friction and interfacial shear pressure losses. The wall friction term remains the same as that for a regular flow link. The interfacial shear term is evaluated using the Wallis correlation for annular flow with an assumption of a smooth interface.

We find the NOTRUMP stratified flow link (junction) model acceptable.

(e) Node Stacking and Mixture Level Tracking

The node stacking and mixture level tracking model determines a unique mixture level in vertically stacked nodes. This is accomplished by a special logic (in addition to the standard logic) of the drift flux and the bubble rise model. A stack is specified by the user. The special logic applied in this model assumes that at any given time a stack has only one node which is stratified. This option modifies the flow link conditions between two nodes when the mixture elevation moves from one node to another.

To obtain the mixture level in a node, an overall nodal volume balance is calculated. The volume of the mixture region is determined from the local mixture mass and local void fraction. This approach is acceptable.

(f) User Supplied Externals

A wide ranging set of user supplied subroutines and functions are available which permits the "programming" of user specific modifications. Some of the more important parameters which can be adjusted and controlled in this manner are: node mixture level fraction, node mixture volume fraction, link critical flow, condensation heat transfer coefficient, pump speed, multipliers for flow and heat transfer areas, heat source, interphase heat transfer coefficients and heat transfer rates. Monitored variables which can be used to determine the values of the "controlled" variables comprise an extensive list of what Westinghouse terms trace variables. The list ranges from heat transfer areas to flow regimes to enthalpies and time. The user has signficant flexibility to program user specific models. System trips are implemented with the use of these user externals.

8. Numerical Technique

NOTRUMP evaluates conservation of mass, momentum and energy with five balance equations. These are solved "separately" from the slip velocity equation.

The mass, mixture momentum and energy balance equations are separated into a node/link form and integrated over a staggered mesh; mass/energy cells (nodes) and momentum cells (links). Flows are solved at momentum cell centers while densities and enthalpies are solved at volume centers (volume averaged properties). This is similar to the method used in the RELAP4 code developed under USNRC sponsorship. The thermodynamic state (specific volume and pressure) at the momentum cell center is derived by "donor" cell algorithms for the evaluation of the specific volume and an approximate steady state mechanical energy (Bernoulli) equation which neglects momentum flux effect, friction and form losses, but retains the gravity dependence to calculate the pressure. The volume "averaged" pressure computed with the balance equations is assumed to be the pressure at the top of the mass/energy cells. The "donor" cell algorithms used for the evaluation of the specific volume are reviewed below in the discussion of the slip velocity equation (the same algorithm is used for the mixture equation and the slip equation).

The flow at the volume center also requires a closure relationship as it is used to compute the momentum flux at the momentum cell edges. It is computed as a weighted average of the upstream and downstream link flows. The user has the option of inputting the desired weighting factors. For applications where momentum flux contributes to the results of the analysis, Westinghouse must justify the selection of the weighing factors. For SBLOCA analyses, Westinghouse, does not intend to calculate the momentum fluxes. For SBLOCA conditions the momentum fluxes are insignificant.

All staggered mesh techniques require extra closure relationships (assumptions or boundary conditions) for calculating flows. Those used in NOTRUMP are typical. These include evaluating the friction and form loss at the junction; to approximate the inertia term flow with the junction flow; to evaluate the momentum flux change due to area change at the junction; and to set volume average properties to volume center quantities.

The above mentioned boundary condition balance equations are solved as a function of time by the well known FLASH-4 technique, developed by Porsching, and treats the convected energy explicitly. There is a limited time step selector which attempts to adjust step size based on nodal mass, energy and pressure changes. The code has the capability to recalculate time steps when the accuracy criteria are not met. It also handles water packing problems similar to RELAP4. The general numerical approach is acceptable.

Slip terms in the balance equations are calculated implicitly. In the context of Porsching's scheme, this means that the slip velocities are part of the linearized variables. The slip velocities are solved by evaluating the drift flux correlation at the links. However in order to do so, the specific volumes and void fractions are required at the links. Two approaches for obtaining the closure relationships are provided. These are the flux weighted void fraction option and the void propagation option. The first option weights the donor nodes with the respective liquid and gas volumetric fluxes to determine the void fraction. The void fraction is obtained by averaging of the specific volumes of the connecting cells if it proves impossible to predict which volume(s) is the donor. The second option calculates void propagation which uses the volumetric fluxes, the void sonic velocity and the void propagation velocities to determine the donor node. Donor node quantities are then used (not flux weight averaged).

The flux weighted void fraction approach is non-physical because the void fraction is not directly related to $\langle j_f \rangle$ and $\langle j_g \rangle$. Thus, the flux-weighed void fraction may not be representative of the actual void fraction at a flow link. However, this approach contributes to the smoothing of the numerical problems to assure that the solution is well-behaved. The second approach is the void fraction propagation option which is basically an approach where the donor void fraction, based on the net mass flow rate direction, is used for the link void fraction. This approach is physically sound. For exceptional cases, the void shock is used when the directions of $<j_g>_{top}$, $<j_f>_{top}$, $<j_g>_{bot}$ and $<j_f>_{bot}$ are not the same. A special model for the void shock treatment was therefore developed.

As a result of using the flux weighted void closure relationships, the set of equations for the phase velocities (more directly the volumetric fluxes), given the mixture velocity and the drift flux correlation, become nonlinear. Instead of using an iterative solution procedure, the equations are reduced to a single polynomial equation in the gas volumetric flux for some of the simpler drift flux correlations. The equation is solved analytically and the correct root is selected on the basis of physical reasoning. When the drift flux correlation is complicated, the void propagation approach is mandatory. This use of the donor void fraction eliminates the void fraction as an unknown and the set of algebraic equations for the phase can once again be solved analytically.

In the special case of the carryover and carryunder models, the top and bottom node qualities are set to one or zero, respectively, in the solution of the drift flux equations. This prevents the "back flow" of liquid in the carryover model and the "back flow" of bubbles in the carryunder model, thereby numerically smoothing the transition to loss of natural circulation in the steam generator.

The final equation in the thermal hydraulic set, the equation of state, is solved for nonequilibrium situations using the specific volume equation and the Newton-Raphson technique, given the known mass and internal energies of each of the two regions. This is identical to the approach taken in a number of current best estimate nonequilibrium pressurizer models.

For the fuel pin model, the thermal conduction equation is spatially discrete by using the standard box integration technique and a first order finite difference approximation for the gradients. The temporal part is solved using the fully implicit first order finitie difference scheme. Standard tri-diagonal factorization schemes are used to invert the resulting matrices. The coupling to the coolant energy equations through the boundary fluxes is treated in tandem. In the context of Porsching's scheme, the equations are not linearized with respect to fuel pin temperatures. While there is no iteration on the boundary fluxes, the hardwired choice of one fuel pin time step per coolant time step should result in sufficient accuracy.

Finally, as noted in §1.b the thermal conduction equation is spatially integrated over a simple heat slab for a one node approximation. For the temporal solution, the slab is treated explicitly in the context of Porsching's scheme; the slab equation is linearized with respect to the thermal hydraulic variables but the thermal hydraulic equations are not linearized with respect to the slab temperature. As described previously, NOTRUMP does not model a node for the slab/coolant interface temperature (wall temperature). Since the wall temperature is applied in heat transfer regime switching criteria, the wall temperature is solved simultaneously and in a consistent manner with the slab temperature using the continuity of heat flux at the interface.

We find the numerical schemes in NOTRUMP acceptable.

IV. REVIEW OF THE WESTINGHOUSE NOTRUMP SBLOCA APPLICATION

This section reviews the specific application of NOTRUMP to SBLOCA licensing calculations, as described in WCAP-10054. This section reviews the nodalization options selected and the particular user external models ("programmed"). The design dependent user input parameters are not reviewed in this report. The specific input nondefault/default values reviewed are those corresponding to the system representation as illustrated in Fig. 3-8-1 in WCAP-10054 (proprietary nodalization).

1. NOTRUMP Default Options

The default input variables for NOTRUMP include: neglect of momentum flux; selection of the Thom correlation over the Jen-Lottes for the nucleate boiling region; the standard coefficients for the critical flow correlations, with a discharge coefficient of unity; the Wallis recommended exponent for the third drift flux model; the widely used value of a void fraction of 0.9 for transition to the forced convection vaporization heat transfer regime; the

W-2/W-3 correlation for CHF and the Westinghouse transition boiling correlation after CHF; the standard coefficients for the first drift flux model; and the implicit NOTRUMP numerical method. We find these default values acceptable.

The non-default variables can be classified as either, (1) variables which describe each component (steam generator, etc.) in terms of the NOTRUMP models (flow links, drift flux models, etc.); and (2) variables used in the hardwired correlations.

The hardwired correlations include: the Thoma cavitation constant for the two phase pump model; pump specific homologous curves, user specified friction factors which match input pressure drop data and vary with two-phase conditions during the transient; adjustment of the bubble velocity constant in the third drift flux model to fit the steam generator level and steaming rate; and the initialization of the fuel pin conditions which are obtained by performing a steady state transient calculation with input inlet boundary conditions.

The staff finds the above options of default input selection acceptable. The justification for and acceptability of the input variables are based upon separate effects tests/systems effects experiments submitted by Westinghouse, Appendix K prescribed models, and staff audit comparisons of NOTRUMP using RELAP5 and TRAC.

2. <u>User External Models</u>

NOTRUMP has a wide ranging set of user supplied subroutines and functions which would allow the "programming" of user specific modifications. The current Westinghouse SBLOCA model does apply user externals. This section reviews a number of the more important user external models.

(a) <u>Area/Level Models</u>

For the core/vessel region the user externals model the variable area T (lower plenum-core inlet) and the inverted T-node (upper head) to evaluate the core mixture level given the correct mixture volume. A user externals model also calculates the corresponding mixture interfacial area. This option models a complex area volume as one node. We find this model acceptable.

(b) Heat Transfer Coefficients

The heat transfer coefficients between the vapor and mixture regions in each node (which govern the nonequilibrium behavior) are input through the use of the user externals. In general, prototypic data is difficult to obtain for the interphase heat transfer, particularly since the NOTRUMP model is expressed in terms of an interfacial temperature.

In the upper plenum, where reflux can occur, NOTRUMP models the heat transfer on the vapor side of the reflux for both a high and low Reynolds number. For the plenum mixture the heat transfer mechanism is assumed to be conduction only. On the liquid side, the reflux is assumed to be saturated. While there is uncertainty regarding these proposed mechanisms, the nonequilibrium effects are diminished by the hot leg conditions where a large surface area exists.

In the cold legs, where safety injection occurs, NOTRUMP models the heat transferred by the liquid side for stratified pipe flow conditions. In the absence of injection, a different condensing correlation is used. A heat transfer model is also applied on the vapor side. The code conservatively models the liquid jet created by the safety and accumulator injection, and applies applicable heat transfer correlations on both the liquid and vapor sides. The surface area of the stratified pipe flow dominates the heat transfer between the liquid and steam regions. The nonequilibrium in the legs is therefore governed by the stratified pipe flow correlations. The condensing correlations used for the stratified pipe flow are current state of the art.

For upper head injection, the heat transfer correlations used are the same as applied during upper plenum reflux, except for the liquid correlations where conduction only is assumed. Since none of these correlations are condensation correlations the heat transfer should be underestimated and the nonequilibrium should be maximized.

(c) Flooding Correlation

Westinghouse implements the Kutateladze flooding correlation through the user externals. The acceptability of this correlation is reviewed in Sections III, V, and VI.

(d) Volumetric Heat Generation

The user external VOLHEAT determines the fission and decay heat generation rate for each core node. The decay heat model is 1.2 times the proposed 1971 ANS infinite operation curve and includes the contribution of the actinides. This model is in compliance with Appendix K to 10 CFR 50.

V. CODE QUALIFICATION

Westinghouse presented qualification work in the form of three categories: separate effects tests, system effects experiments and full scale plant sensitivity analyses. The separate effects tests and comparisons against plant scale systems experiments such as LOFT are examined in this section. The details of the full scale plant sensitivity analyses and comparisons against staff audit calculations with RELAP5 and TRAC are discussed in Section VI. This evaluation is divided into two parts: separate effects tests and systems effects experiments.

1. Separate Effects

There are a number of distinct physical phenomena which could occur during a small break loss of coolant accident. These phenomena are addressed by the following NOTRUMP models:

- Core uncovery and mixture level calculation, applying drift flux correlations
- Loop seal clearing and the horizontal stratified flow models
- Steam generator inventory hangup and the flooding correlations

- Nonequilibrium ECCS injection and the nonequilibrium mass/energy transfer terms

- Break flow and the vapor pull through/liquid entrainment models.

- SG natural circulation heat transfer and the condensation/reflux drift flux models with flooding limits.

Westinghouse justified its analytical approximations and assumptions in its models by performing a number of comparisons against selected separate effects data, both on the lab scale as well a "scaled up" component tests. The following describe these comparisons.

(a) Blowdown Vessel Tests

Three blowdown tests of an 11.2 m tall vessel (at Battelle Institute, Frankfort, Maine) were analyzed with NOTRUMP. A multiplier of four on the verical-flow-link drift velocity was required to give the best overall results for the three tests. In the first test, the vessel was about half full of water and the break was located in the steam space. The calculated flow rate oscillated about the measured flow rate (vs. time). This oscillation was attributed to the use of a constant drift-velocity multiplier for all vertical flow links (junctions).

The second test was initiated with the break covered by water. Results for this comparison were similar to those for the first test. In the third test, the vessel was nearly filled with water and the break size was decreased to 1.97 in. dia. (vs 5.5 in. for the first and second tests). Agreement between the calculations and the tests were very good. Break discharge coefficients of 0.60, 0.60, and 0.65, respectively, were used for the three calculations.

NOTRUMP was also benchmarked with a depressurization test of a 190.5 in. tall vessel conducted at Battelle Northwest Laboratories. While applying a discharge coefficient of 0.80 to the orifice diameter of 1.687 inch, the code showed good agreement with the data.

Additional benchmarks were performed on three tests of a 5.19 m tall vessel (at the Betulla plant of C.C.R. Euratom at Ispra). The first test had a 5 inch diameter break located 8.14 ft. above the initial water level. The second test had a 4 inch diameter break 3.87 ft. below the initial water level, and the third test had a 2 inch diameter break with the initial water level at the lower edge of the break nozzle. Discharge coefficients of 0.55, 1.00 and 0.85, were used in the respective calculations. The NOTRUMP calculations showed good agreement with the test data except for the second test. For the second test, the calculated pressure dropped substantially faster than the measured pressure.

The Ispra tests were performed with an earlier versions of the code and applied options which do not exactly correspond to the current Westinghouse SBLOCA model (different drift flux correlations have been added to the code, etc.). The data was limited (mixture level, for example, was not measured) and inconsistent (staff's limited review of the data indicate non physical bubble separation assumption is required to match the inventory and depressurization history). To match the system inventory history Westinghouse needed to increase the drift flux correlation by a factor of four. These tests were discarded as part of the NOTRUMP validation package, due to the problems described above.

(b) Core Uncovery Tests

Core uncovery tests were conducted at the Oak Ridge National Laboratory (ORNL) using an 8x8 bundle with a configuration typical of a Westinghouse 17x17 fuel assembly. The power profile of the bundle was uniform and 12 ft in height. The tests were conducted for pressures between 300 and 1000 psia with prototypic heat fluxes and mass flow rates as expected during SBLOCAs. Westinghouse compared NOTRUMP with two series of tests; steady state uncovery tests and bottom reflood tests starting from a steady state uncovered condition. Inlet coolant temperatures ranged between 200°F and 400°F. Typical core uncovery was between three and four feet. These conditions are similar to the Westinghouse SBLOCA calculations reported in WCAP 10054.

For the steady state uncovery tests, the comparison of collapsed liquid levels and mixture levels showed good agreement (one data point appears to be spurious). The "measured" mixture levels were inferred from the thermocouple readings. The good agreement confirms the adequacy of the drift flux model used for the core thermal hydraulics. There are no mixture level comparisons for the transient reflood tests but the collapsed liquid level is within the scatter of the data. The clad temperatures are well predicted. The fluid temperature predictions were higher than the test data. This is attributed to the NOTRUMP logic which conservatively models the reflux fallback and prevents quench at core uncovery locations. Finally, it is important to recognize that the ORNL tests and the Westinghouse data supporting the Cunningham and Yeh model are the only data available for these conditions. The Westinghouse model for core mixture level is the current state of the art. The test comparisons can be interpreted as showing that the core thermal hydraulics predict the mixture level well for steady state and transient conditions. We find the NOTRUMP core model acceptable.

(c) Loop Seal Tests

Tests on a one third linear scale air/water model of a Westinghouse PWR loop seal were performed in a quasistatic mode at Cadarache, France. Data was collected of gas flow, pressure drop, residual liquid mass, and void fraction. In addition to data acquisition, the experiments were designed to provide visual studies of the various flow regimes. As liquid inventory distribution is important during a SBLOCA, Westinghouse correlated loop seal clearing with the liquid mass inventory as a function of steam flow. A number of entrainment correlations were used to calculate the loop seal clearing phenomena observed in the tests.

Implementing the various entrainment models available in open literature, Westinghouse determined which correlations were applicable to the loop seal geometry. Upon benchmark analyses with the applicable correlations, Westinghouse showed good agreement with the experimental data.

The method used in the benchmark analyses were labeled the "detailed loop seal model." For economic considerations, Westinghouse developed a simplified noding scheme and justified the "simplified model" for use in calculating small break LOCAs. We find the Westinghouse loop seal model acceptable.

2. Integral Systems Effects Experiments

(a) LOFT Experiments

NUREG-0611 required licensees of Westinghouse designed NSSS to benchmark their SBLOCA thermal-hydraulic computer codes with the LOFT L3-1 and LOFT L3-7 experiments. The following describes the benchmark analyses of NOTRUMP with the above mentioned LOFT experimental data.

The LOFT L3-1 test simulated a 4-inch equivalent diameter break. The calculated steady conditions used for the transient analysis agreed well with the measured conditions. The only noticeable exception was the inactive loop hot leg temperature which was slightly higher than the data. This discrepancy was also observed in the L3-7 simulation and could be attributed to the stagnant conditions of the broken loop.

Two sets of transient calculations were performed for L3-1; one using the Zaloudek/Moody break flow models and the other forcing the break flow to match the experimental data. For the forced break flow case, the upper plenum pressure behavior was well predicted. The nonequilibrium model did a reasonable job predicting ECCS injection.

As expected, the calculation with the Zaloudek/Moody break flow models did not compare as well with the experimental data. The Moody break flow model is recognized as over estimating the mass flow during two-phase conditions. The calculated integral flow showed that by transient end, the mass remaining in the system is underestimated by 25%. The thermal-hydraulic trends, however, were predicted by the NOTRUMP code. We find the benchmark analyses of NOTRUMP with the LOFT L3-1 and L3-7 experimental data acceptable.

(b) Semiscale S-UT-08 Experiment

The staff made a specific request that Westinghouse validate its NOTRUMP computer program with the Semiscale S-UT-08 experimental data. This request was made for two reasons. First, the S-UT-08 data of a simulated 5% cold leg break uncovered a thermal-hydraulic phenomenon which has previously not been observed to the extent seen in this experiment. In specific, it appeared that total core uncovery occurred (assuming collapsed level) prior to clearing of the loop seals at the suction of the reactor coolant pumps. This uncovery was attributed to complex hydraulic and heat transfer regimes within the steam generators which depressed the core coolant by developing significant resistance to steam venting.

The second reason for requesting benchmark analysis with S-UT-08 data resulted from staff independent plant audits with RELAP5/Mod 1.5, which calculated the pre-loop-seal-clearing core level depression phenomena and resulted in a calculated peak clad temperature in excess of expectation (1800°F). It should be noted that the staff does not consider the RELAP5/Mod 1.5 results as representative of actual plant responses to a SBLOCA (see section VI, "Analytical Staff Audits," for additional detail).

In response to the above mentioned staff request, Westinghouse proceeded to benchmark NOTRUMP with the S-UT-08 data. Westinghouse demonstrated that NOTRUMP is capable of simulating the S-UT-08 data. As the noding of the steam generator increased, the fidelity of the code to model the core coolant depression phenomenon (prior to loop seal clearing) increased accordingly. However, Westinghouse demonstrated that even though the proposed simplified licensing nodalization does not calculate a conservative pre-loop-seal-clearing core level depression for very small break sizes, it nevertheless does result in a conservative post-loop-seal-clearing PCT.

We find the NOTRUMP model acceptable for licensing application.

VI. ANALYTICAL STAFF AUDITS

Staff audit analyses were performed using both the RELAP5 and the TRAC computer programs. These programs, or codes, were developed by NRC's Office of Nuclear Regulatory Research for staff independent evaluations of PWR thermal-hydraulic transient and accident responses. This section describes the results and conclusions of the audit analyses.

To demonstrate a licensing application with NOTRUMP, Westinghouse provided a mini-break spectrum of a RESAR-3S nuclear steam supply system. The staff performed independent audit analyses on a similar plant with peaking factors, ECCS and EM boundary conditions similar to the Westinghouse assumptions (Table VI-1 describes some of these assumptions). The staff analyses included the use of the Moody critical flow model, as required by Appendix K to 10 CFR 50, but followup calculations with a best-estimate break flow model showed no significant difference in the calculated peak clad temperature (PCT).

The staff audit had to be repeated twice. The initial analysis was performed with the RELAP5/Mod 1.5 computer program. The results were not typical of previous SBLOCA analyses results, in that the PCT occurred prior to clearing of the reactor coolant pump suction loop seals. Previous analyses, including integral systems tests, have shown the PCT to occur following loop seal clearing. The phenomenon of core level depression prior to loop seal clearing was observed in the Semiscale S-UT-08 experiment, conducted by the Idaho National Engineering Laboratory (INEL). But, even for S-UT-08, the PCT occurred following loop seal clearing. The staff, therefore, was not convinced of the acceptability of the RELAP5/Mod 1.5 calculations.

The S-UT-08 data showed a significant depression of the coolant level within the core for an extended period of time prior to clearing of the pump suction leg loop seals. The duration of core uncovery was in excess of that previously observed by experiments or calculations. The core level depression phenomenon

TABLE VI-1

SBLOCA AUDIT ANALYSIS ASSUMPTIONS

	Item	Value
1.	Break opens at the bottom of one of the cold legs	instantaneously
2.	Pressurizer low pressure (reactor trip)	1860 psia
3.	Reactor trip signal delay	2.0 sec.
4.	Time for inserting control rods	2.4 sec.
5.	Turbine stop valve starts to close	at reactor trip signal
6.	Turbine stop valve closure time	0.5 sec.
7.	RCS pump trip	at reactor trip signal due to loss of offsite power
8.	Main feedwater trip	at reactor trip signal due to loss of offsite power
9.	Main feedwater pump coastdown	linear coastdown over 5 sec
10.	Pressure low-low pressure (safety injection)	1760 psia
11.	Safety injection delay time	25.0 sec.
12.	Auxiliary feedwater on time	60.0 sec after loss of offsite power
13.	Auxiliary feedwater flowrate	20.7 lb/sec/SG
14.	Accumulator pressure	600 psia
15.	Accumulator water volume	1050 ft^3/unit
16.	Loop seals in the intact loops are not permitted to clear prior to clearing of the loop seal in the broken loop	Westinghouse conservative assumption

was attributed to the rate of downcomer to upper plenum coolant flow (part of the core bypass flow) and steam generator thermal-hydraulics which experienced complex interactions between flow regimes, steam condensation rates and flooding conditions (see Figure VI-1).

The staff calculations with RELAP5/Mod 1.5 resulted in a calculated PCT of 1800°F for the postulated 2 inch diameter break. This calculation did not include heating contributions from Metal/Water reaction, which would be expected to occur in this temperature range. Figure VI.2 illustrates the variation between the NOTRUMP licensing evaluation and that of RELAP5/Mod 1.5.

Upon further review of the RELAP5/Mod 1.5 calculations, the staff requested NRC's Office of Nuclear Regulatory Research (RES) to benchmark RELAP5/Mod 1.5 with the Semiscale S-UT-08 data. While RELAP5/Mod 1.5 was able to predict the observed core level depression, the remaining system parameters were not in good agreement with the data. It was therefore concluded that the core level depression was not calculated for the right reasons. The staff then reanalyzed the S-UT-08 experiment with RELAP5/Mod2 and TRAC-PF1/Mod1 computer codes. It was determined that both computer codes required updating to model the S-UT-08 experimental results. The RELAP5 analysis required atypical nodalization to model the flow path connecting the upper plenum to the hot legs. When applying the same nodalization technique to the RESAR-3S plant model, non-realistic liquid inventory collected within the hot legs. The analysts, thereby, reverted back to the previous "standard" nodalization for the RESAR-3S calculation.

The results of the RESAR-3S NSSS audit break spectrum calculations (calculated twice, once with a Model-F (high elevated feedwater location) and then with a Model-D (low elevated feedwater location) steam generator), are tabulated on Table VI-2 and plotted in Figure VI-3. As shown on Table VI-2, the PCT calculated by RELAP5 occurred prior to loop seal clearing for break sizes less than or equal to 3 inches in diameter. For breaks greater than 3 inches in equivalent diameter, the PCT occurred following the clearing of the loop seals. Figure VI-4 separates the PCT overlay results into the pre-loop-seal clearing and post-loop-seal clearing components. By reanalyzing the break spectrum with RELAP5/Mod-2, the calculated PCT decreased 600°F from the Mod 1.5 results. This difference

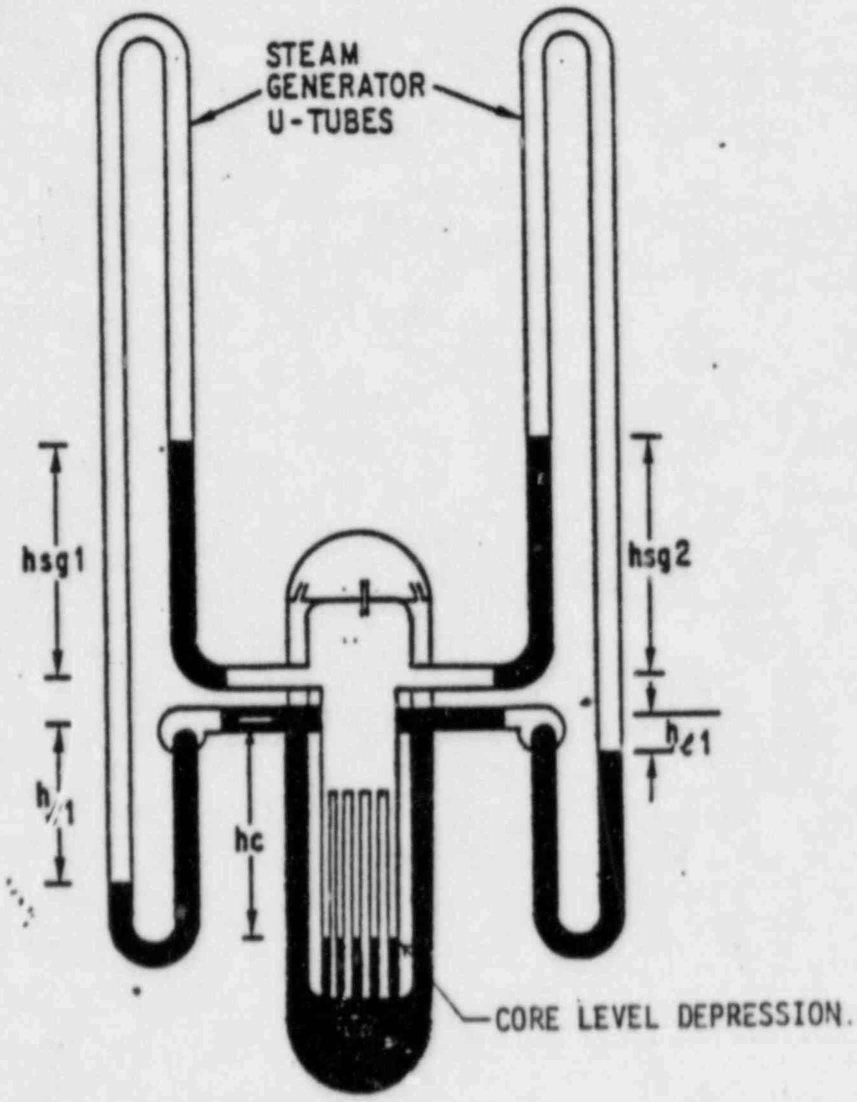

FIGURE VI-1

Primary Inventory Distribution Before Loop Seal Clearing

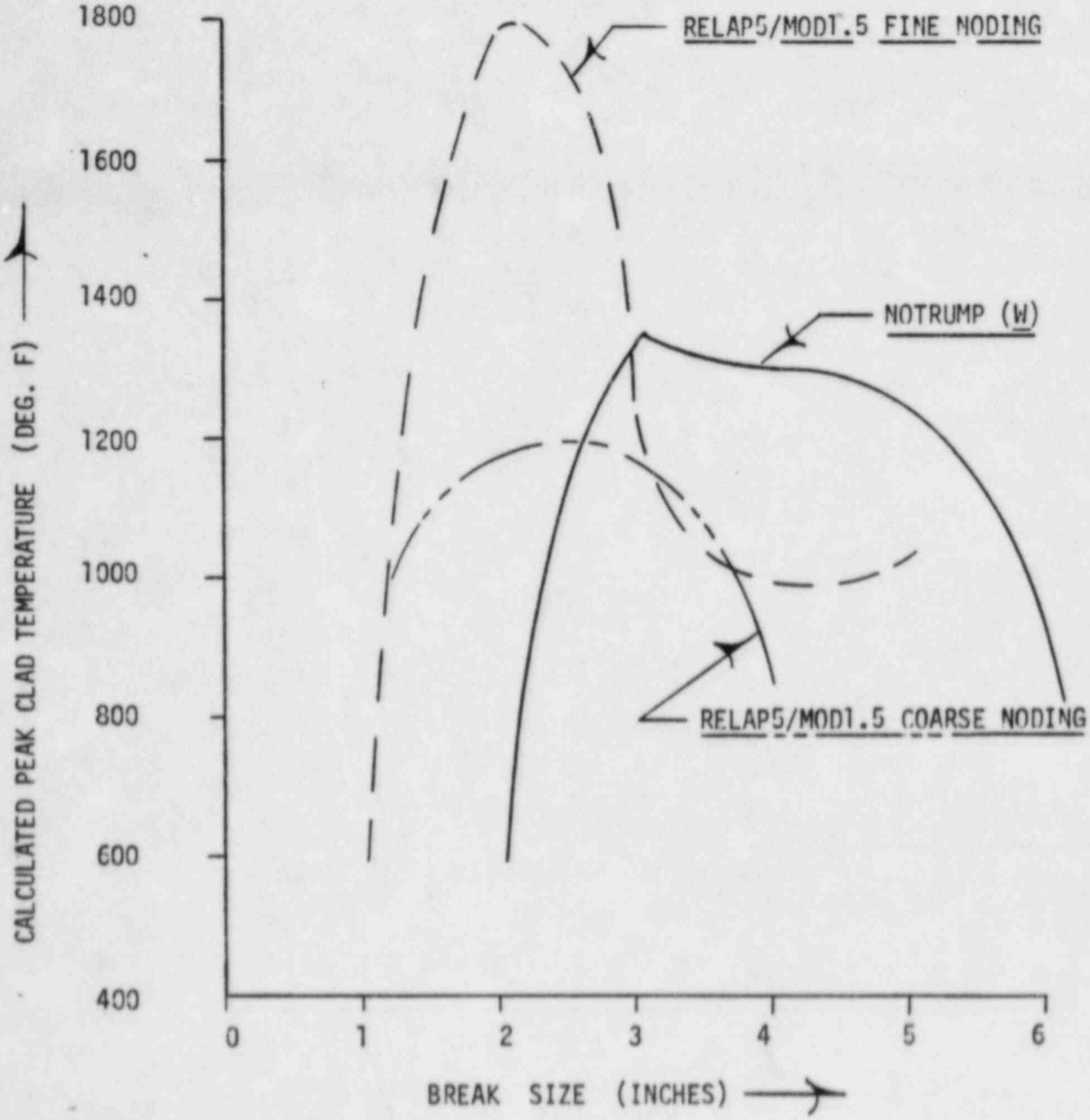

Figure VI-2

RELAP5/MOD 1.5 Calculated Results Versus NOTRUMP Calculated Results is attributed to the new flow regime maps and conservation equations incorporated into Mod2. The TRAC-PF1/Mod1 calculations for the 2, 3, and 4 inch breaks showed no fuel temperature excursion.

The NOTRUMP calculations predicted the PCT to occur following loop seal clearing for all break sizes. Based upon the audit results outlined above, the staff requested Westinghouse to benchmark NOTRUMP with the S-UT-08 experiment and to peform steam generator nodalization studies (see section V.2). With increased steam generator noding, NOTRUMP was shown to reasonably simulate the S-UT-08 data. However, Westinghouse demonstrated that while for very small breaks NOTRUMP does not appear to accurately predict the pre-loop-seal-clearing phenomenon with its Evaluation Model (EM) nodalization, the calculated post-loop-seal-clearing PCT was conservatively calculated. When pre-loop-seal-clearing core level depression is calculated, the calculated PCT following loop seal clearing decreased. Due to the significant costs in calculating SBLOCAs, Westinghouse elected to minimize its nodalization. As a consequence, the pre-loop-seal-clearing/core level depression phenomenon will not be accurately modeled for very small break sizes, but the resulting PCT will be conservative.

After a detailed review of the results calculated by TRAC and RELAP5, the staff concluded that additional benchmarking of both TRAC and RELAP5 to calculate the pre-loop-seal-clearing core level depression phenomenon is required. Validation of the NRC codes will occur following acquisition of additional data from ROSA-IV (in Japan) and Semiscale (in the USA). Data from these facilities are anticipated by January 1, 1986.

Through detailed reviews of the Westinghouse NOTRUMP code and audits, the staff concluded that:

(1) With adequate nodalization, NOTRUMP is capable of predicting the S-UT-08 experimental data.

(2) Westinghouse demonstrated that the simplified EM nodalization will not conservatively calculate a pre-loop-seal-clearing core level depression phenomenon for very small break sizes, as observed in the S-UT-08 experimental data, but will result in a conservative peak clad temperature.

TABLE VI-2

BREAK SPECTRUM PCT AUDIT ANALYSIS RESULTS
(DEG. F)

Break Size (inches)	NSSS With Model-F Steam Generator			NSSS With Model-D Steam Generator	
	NOTRUMP	RELAP5/MOD2	TRAC-PF1 MOD1	RELAP5/MOD1.5	RELAP5/MOD2
2 (PRE)*	No Heatup	959°F	No Heatup	1800°F	1199°F
2 (POST)**	No Heatup	No Heatup	No Heatup	No Heatup	No Heatup
3 (PRE)*	No Heatup	790°F	No Heatup	1220°F	768°F
3 (POST)**	1342°F	No Heatup	No Heatup	960°F	No Heatup
4 (PRE)*	No Heatup	657°F	No Heatup	930°F	860°F
4 (POST)**	1287°F	860°F	No Heatup	990°F	912°F
5 (PRE)*	No Heatup	760°F	--	840°F	642°F
5 (POST)**	1249°F	No Heatup	--	1060°F	969°F
6 (PRE)*	--	1005°F	--	--	975°F
6 (POST)**	828°F	1148°F	--	--	1026°F
7 (PRE)*	--	883°F	--	--	770°F
7 (POST)**	--	962°F	--	--	681°F

* = PRE-LOOP-SEAL CLEARING PCT

** = POST-LOOP-SEAL CLEARING PCT

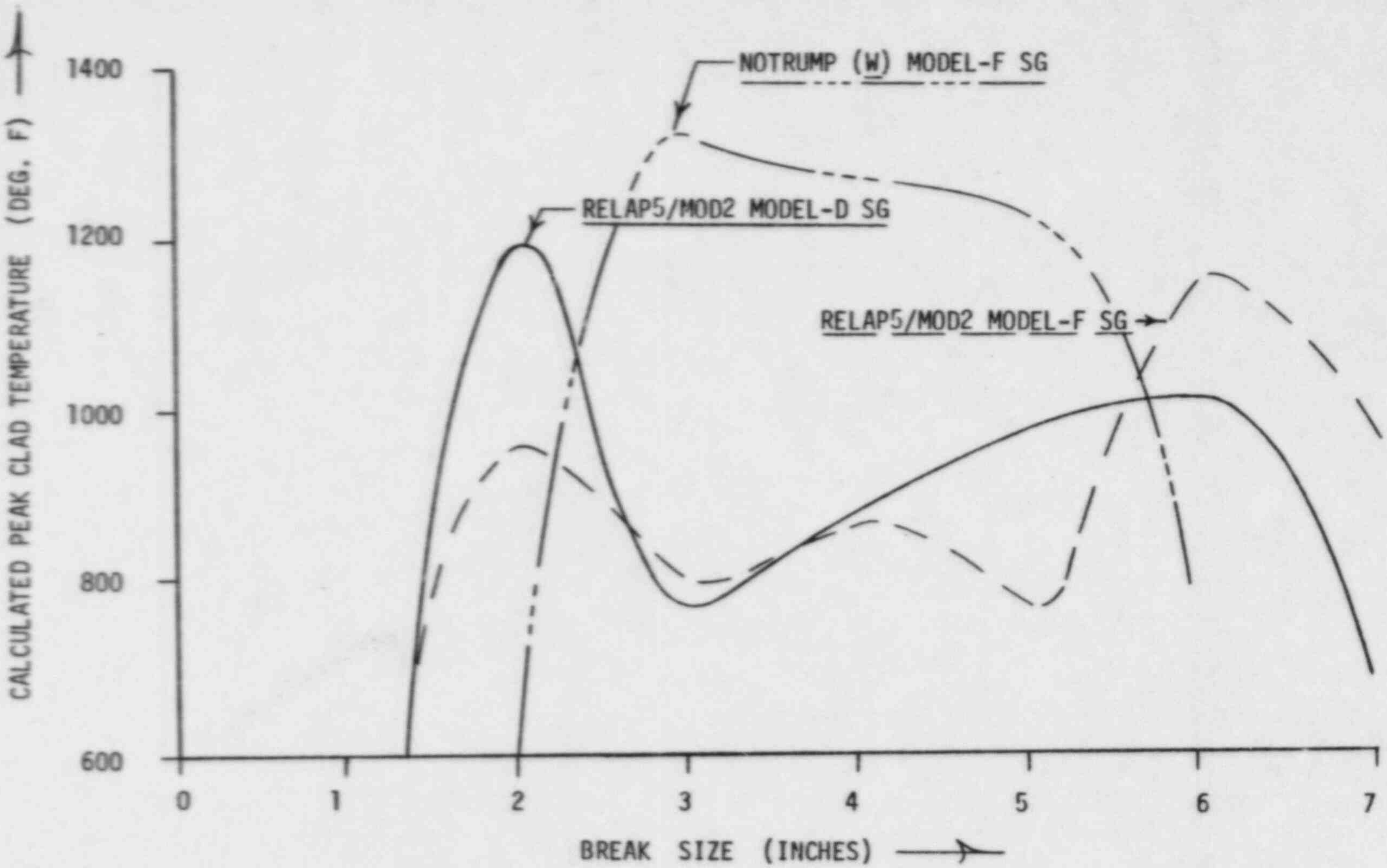

FIGURE VI-3

AUDIT ANALYSIS RESULTS

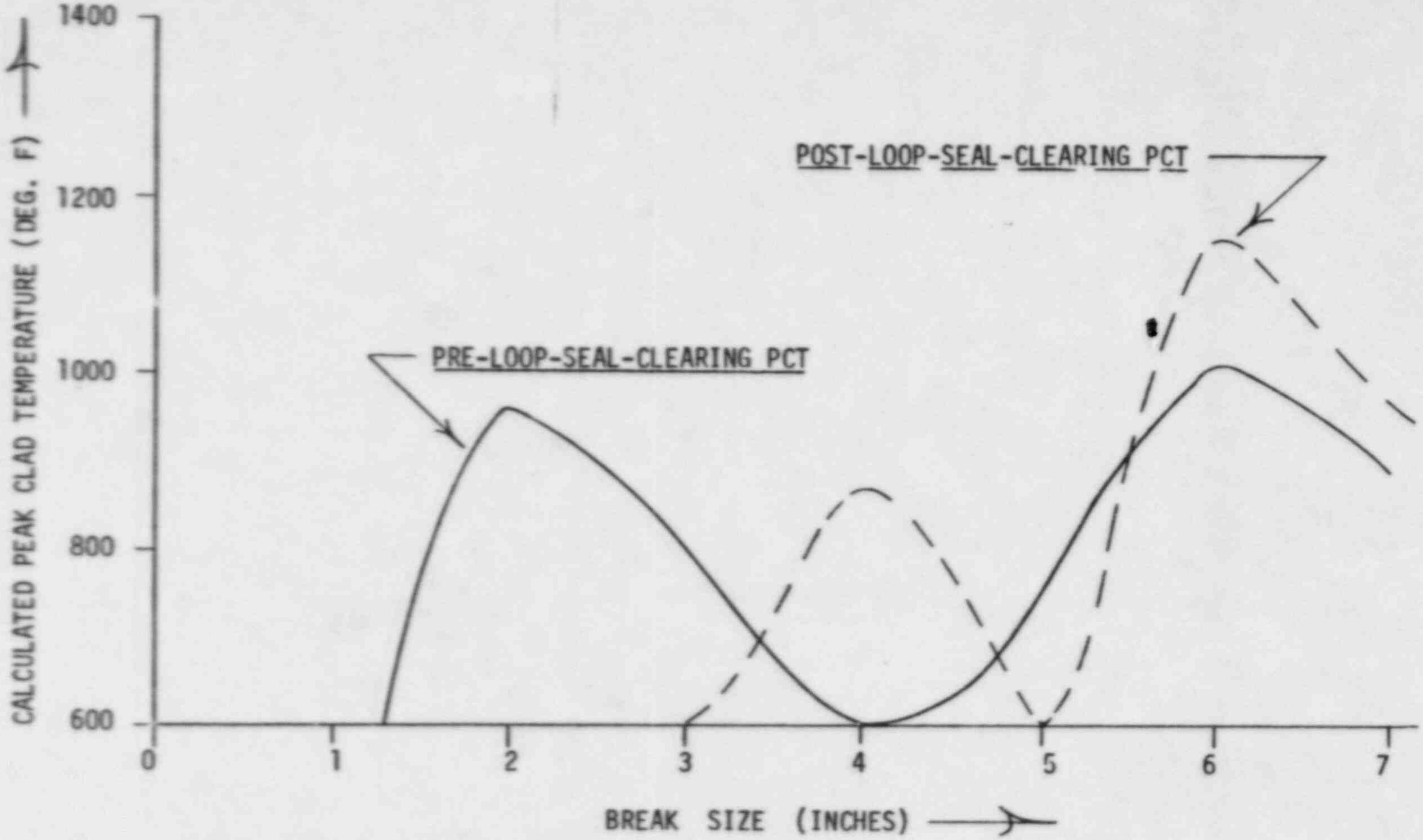

FIGURE VI-4

RELAP5/MOD2 AUDIT RESULTS

RESAR-3S With MODEL-F-SG

(3) While the NRC audit tools predict different thermal-hydraulic system responses than NOTRUMP, the staff has insufficient data to conclude the adequacy of RELAP5/Mod2 and TRAC-PF1/Mod1 to accurately model the pre-loop-seal-clearing core level depression phenomenon observed in the calculations. A detailed program has been initiated to obtain additional data for staff's validation of its computer codes.

(4) Based upon the more mechanistic and validated models in NOTRUMP, versus RELAP5 or TRAC, and the sensitivity studies performed by Westinghouse, the staff finds the NOTRUMP computer program aceptable for licensing application.

(5) Based upon all available experimental data, the peak clad temperature has always occurred following clearing of the reactor coolant pump loop seals. The staff therefore concludes the Westinghouse NOTRUMP code acceptable for licensing application.

VII. Quality Assurance Audit

On October 1 through 5, 1984, the NRC conducted an audit of the Westinghouse quality assurance procedures used to develop the NOTRUMP computer program. The audit examined the user's manuals, the theoretical manuals, and the verification analyses applied to NOTRUMP. Conclusions of this inspection were:

(a) Applicable separate effects data were not available for verifying most of the NOTRUMP component models. It was concluded that these models could be verified indirectly through integral experimental verification.

Upon staff review of the separate effects and integral experiment benchmarks, the staff finds the component models selected for licensing calculations acceptable.

(b) The steam generator noding study does not appear to be sufficient to justify the current noding scheme.

The above finding was based upon information obtained from staff audits with RELAP5/Mod 1.5. Since then, new analyses with RELAP5/Mod 2 and TRAC-PF1/Mod 1 have demonstrated the NOTRUMP results as conservative. As outlined in Section V.2 and VI of this SER, the staff is continuing to assess its analytical audit tools and obtain additional data for code validation and a better understanding of the pre-loop-seal-clearing core level depression phenomenon. All available experimental data have shown the core level depression phenomenon prior to loop seal clearing as not resulting in limiting challenges to the peak clad temperature, as defined in 10 CFR Part 50.46. Should future data indicate a potential that the pre-loop-seal-clearing PCT could be limiting, the staff will require Westinghouse to provide further justification of the NOTRUMP model.

(c) One case was detected where an error in the computer program was not corrected in all verification calculation notes. This does not imply an error in the code, but an error in the documentation which could, in the future, be referenced for other application.

(d) Two cases were found which did not contain a verifier's signature.

The above noncompliances ((c) and (d)) were exceptions to the rule rather than any observed pattern in the QA of the NOTRUMP development. No error has been uncovered which would have significant impact on the results calculated by NOTRUMP.

Further details of this inpsection are documented in the inspection report number 99900404/84-03, dated January 7, 1985. The staff finds the quality assurance procedures applied in the development of NOTRUMP acceptable.

VIII. CONCLUSIONS

The NOTRUMP computer program was developed by Westinghouse to more accurately and efficiently assess the consequences of a small break loss of coolant accident (SBLOCA). The Westinghouse Owners Group referenced NOTRUMP as its response to NRC concerns documented within Section II.K.3.30 of Enclosure 3 to NUREG-0737. The staff concludes that NOTRUMP is capable of assessing best-estimate NSSS responses to postulated SBLOCAs. This review only addressed the application of NOTRUMP for licensing evaluations and compliance with Appendix K to 10 CFR Part 50. The code options available to the user but not applied in licensing evaluations were not reviewed.

The staff reviewed NOTRUMP's compliance with Appendix K to 10 CFR Part-50, and find it in full compliance. Table VIII-1 outlines the NOTRUMP conformance with Appendix K.

NOTRUMP was demonstrated to calculate the thermal-hydraulic phenomena expected during postulated SBLOCAs. This included core coolant level depression prior to clearing of the loop seals. This level depression phenomenon was observed in the Semiscale S-UT-08 experimental data. However, upon simplifying the steam generator nodalization, the code would no longer calculate a conservative pre-loop-seal-clearing core uncovery for very small break sizes. The post-loop-seal-clearing peak clad temperature (PCT), however, was increased. Westinghouse thereby concluded that by not modeling the steam generators in detail, the Evaluation Model calculated a conservative PCT. The staff is further reviewing the pre-loop-seal-clearing core level depression phenomenon and scaling effects to operating plants through the Semiscale and ROSA-IV experimental facilities. Results from these facilities are expected by December 31, 1985.

The staff concludes that NOTRUMP is an acceptable computer program for use in performing licensing calculations of small break loss of coolant accidents for Westinghouse designed nuclear steam supply systems (e.g., 2, 3, and 4 loop plants, including those with upper head injection design).

Upon receipt of this SER by Westinghouse, NOTRUMP, as documented in WCAP-10079 and WCAP-10054, is designated as the new Westinghouse licensing tool for SBLOCA evaluations and thereby replaces the WFLASH code for SBLOCA application. We require that Westinghouse resubmit both WCAP-10079 and WCAP-10054 and incorporate this SER, modifications resulting from this review, and documentation of the questions and answers generated during this review. This review also fulfills the requirements in TMI Action Item II.K.3.30 for computer code validation. We therefore find the NOTRUMP code acceptable for TMI Action Item II.K.3.31 application.

TABLE VIII-1

NOTRUMP CONFORMANCE WITH APPENDIX K TO 10 CFR 50

This table details the conformance of the NOTRUMP computer models with the requirements of Appendix K to 10 CFR Part 50.

Appendix K.I Section	NOTRUMP Conformance
A. Sources of Heat During the LOCA	
1. Initial Stored Energy in the Fuel	
a. Fuel Thermal Conductivity, Function of	
1) Burnup	
2) Temperature	Negligible effects
3) Initial density	Accounted for in Eqs. (T-101 a and b) See Q 440.65.1†
b. Gap Thermal Conductance, Function of	
1) Burnup (Fuel densification and expansion	Compliance as in LOCTA (per p. T-106)
2) Gas composition and pressure	Accounted for in Eqs. (T-55) and (T-68)-(T-78)
3) Initial cold dimensions	Accounted for in Eqs. (T-59) - (T-62)
4) Cladding creep	Compliance as in LOCTA (per p. T-16)

*Unless otherwise noted the report referenced is WCAP-10079.
†Q = first round question number.

Appendix K.I Section	NOTRUMP Conformance
2. Fission Heat	Standard Point Kinetics
3. Decay of Actinides	11 Fission Product Decay Energy Groups and 6 Delayed Neutron Precursor Groups
4. Fission Product Decay	1.2 x the 1971 Proposed ANS Standard
5. Metal Water Reaction Rate (Baker-Just eqn.)	Accounted for by Eq. (T-36)
6. Reactor Internals Heat Transfer (piping, walls, etc.)	Complied with. Refer to Sects. 3-1-5 and 5-2-5*
7. Primary-to-secondary Heat Transfer	Complied with. Refer to Sect. 6 (and Sects. 3-4 and 5-3*)
B. Swelling and Rupture of Cladding	
1. Swelling and Rupture, Function of	
a. Temperature, Function of	
1) Gap conductance, function of	
a) Temperature	Accounted for in Eqs. (T-59) - (T-62)
b) Swelling and rupture	Compliance as in LOCTA (per p. T-16)
b. Pressure Differential	Accounted for in Eq. (T-63)
2. Effect on Cladding Oxidation and Embrittlement, and Hydrogen Generation	Accounted for in Eqs. (T-36) - (T-54)
C. Blowdown Phenomena	
1. Break Characteristics and Flow	
a. Spectrum of Breaks	Code application
b. Discharge Model	

*WCAP-10054 (otherwise WCAP-10079).

Appendix K.I Section	NOTRUMP Conformance
1) Moody model	Moody saturated break flow model and Modified Zaloudek for subcooled break flow. Refer to Sect. M-2-1
2) Spectrum of discharge	Code application
c. End of Blowdown	
1) End of bypass	Not Applicable to SBLOCAs.
d. Noding Near Break and ECCS Injection Points	Not a SBLOCA concern due to stagnation conditions and mechanistic models.
2. Frictional Pressure Drops	
a. Reynolds No. Effect	Modeled in NOTRUMP. See Eqs. (5-35) - (5-53)
b. Two-phase Multipliers	Accounted for in Eq. (5-34) (Thom-Martinelli-Nelson)
3. Momentum Equation	
a. Temporal Change of Momentum	
b. Momentum Convection	
c. Area Change Momentum Flux	
d. Compressiblity Effect	
e. Pressures Loss due to Wall Friction	Accounted for in Eq. (2-33)
f. Pressure Loss from Area Change	
g. Gravitational Acceleration	
4. Critical Heat Flux	
a. Steady State Correlation	Complied with. Refer to p. T-25 (W-2 & W-3)
b. Transient Correlation	Complied with. Refer to p. T-25 (Westinghouse Transition Correlation)
c. Re-establishment of Nucleate Boiling	Not Applicable to SBLOCAs

Appendix K.I Section	NOTRUMP Conformance
5. Post-CHF Heat Transfer Correlation	
a. Model Used	Westinghouse Transition Heat Transfer Corr. Previously approved for LOCTA.
b. The Groeneveld correlation shall not be used Near its Low-pressure Singularity	Not used for small breaks (Q 440.65.6)
c. Heat transfer after Saturated fluid-cladding ΔT exceeds 300°F	Complied with. See p. T-26.
d. Reestablishment of Transition Boiling Heat Transfer should not occur during blowdown, except during reflood.	Not Applicable to SBLOCAs.
6. Pump Modeling	
a. Momentum Transfer Between Fluid and Impeller	Homologous curves and a dynamic model used.
b. Pump Resistance Justification	SATAN-VI
c. Two-phase Region Justification	SATAN-VI
7. Core Flow Distribution During Blowdown	
a. Cross-flow and Blockage Accountability	Not important for SBLOCAs. Conservative to assume average core mixture level.
b. Hot Channel Inlet Enthalpy and Determination	Calculated in LOCTA with boundary conditions from NOTRUMP.
D. Post-blowdown Phenomena: Heat Removal by the ECCS	
1. Single Failure Criterion	Code application
2. Containment Pressure	Code application assumes Atmospheric

Appendix K.I Section	NOTRUMP Conformance
3. Reflood Rate	
a. Carryover Fraction Determination	Not applicable to SBLOCAs
b. Accumulator Gas Effect	Code does not model noncondensibles. SBLOCAs equilibrate above the pressure at which the accumulators empty (G. T. 125 psi) outside the accumulators. Within the accumulator an ideal gas law is assumed.
4. Steam Interaction with ECCS Water	
a. Zero Steam Flow in the Intact Loops While Accumulators Discharge Water	The code is capable of preventing steam flow through the intact loops during accumulator injection.
5. Refill and Reflood Heat Transfer	
a. Conservative with Respect to FLECHT Data	Not applicable for SBLOCAs.
b. Steam Cooling Only at Low Reflood Rates	Mechanistic models benchmarked with applicable data.
c. Blockage Accountability	Accounted for in LOCTA

Appendix K.II Section	NOTRUMP Conformance
1. a. Description of Model	
1) Equations Used	Fully documented in WCAP-10079.
2) Finite-difference Approximations	Detailed in Sects. 2 and 8
3) Assumptions Made	Documented in WCAPs-10079 & 10054
4) Parameter Values	
b. Adequacy of Detail	Documented in WCAPs-10079 & 10054
c. Computer Program Listing	Will be locked in NRC's safe, consistent with previous practices.
2. Solution Convergence Demonstration	WCAP 10054 sensitivity studies.
3. Sensitivity Studies	Provided in Sect. 5*
4. Comparison with Experiments	Provided in Sect. 11 and Sect. 6*

*WCAP-10054 (Otherwise WCAP-10079).

ENCLOSURE 2

MULTIPLANT ACTION ITEM F-57 SER FOR WESTINGHOUSE PLANTS WITH ZIRCALOY FUEL ASSEMBLIES

SAFETY EVALUATION REPORT
TMI ACTION ITEM II.K.3.30 FOR
WESTINGHOUSE PLANTS

I. BACKGROUND

NUREG-0737 is a report transmitted by a letter from D. G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors and applicants for operating reactor licenses forwarding TMI Action Plan requirements which have been approved by the Commission for implementation. Section II.K.3.30 of Enclosure 3 to NUREG-0737 outlines the Commission requirements for the industry to demonstrate its small break loss of coolant accident (SBLOCA) methods continue to comply with the requirements of Appendix K to 10 CFR Part 50.

The technical issues to be addressed were outlines in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants." In addition to the concerns listed in NUREG-0611, the staff requested licensees with U-tube steam generators to assess their computer codes with the Semiscale S-UT-08 experimental results. This request was made to validate the code's ability to calculate the core coolant level depression as influenced by the steam generators prior to loop seal clearing.

In response to TMI Action Item II.K.3.30, the Westinghouse Owners Group (WOG) has elected to reference the Westinghoue NOTRUMP code as their new licensing small break LOCA model. Referencing the new computer code did not imply deficiencies in WFLASH to meet the Appendix K requirements. The decision was based on desires of the industry to perform licensing evaluations with a computer program specifically designed to calculate small break LOCAs with greater phenomenological accuracy than capable by WFLASH.

The following documents our evaluation of the WOG response to TMI Action Item II.K.3.30 confirmatory items.

II. SUMMARY OF REQUIREMENTS

NUREG-0611 required licensees and applicants with Westinghouse NSSS designs to address the following concerns:

A. Provide confirmatory validation of the small break LOCA model to adequately calculate the core heat transfer and two-phase coolant level during core uncovery conditions.

B. Validate the adequacy of modeling the primary side of the steam generators as a homogeneous mixture.

C. Validate the condensation heat transfer model and affects of non-condensible gases.

D. Demonstrate, through noding studies, the adequacy of the SBLOCA model to calculate flashing during system depressurization.

E. Validate the polytropic expansion coefficient applied in the accumulator model, and

F. Validate the SBLOCA model with LOFT tests L3-1 and L3-7. In addition, validate the model with the Semiscale S-UT-08 experimental data.

Detailed responses to the above items are documented in WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

III. EVALUATION

The following is the staff's evaluation of the TMI Action Item requirements outlined above.

A. Core Heat Transfer Models

The Westinghouse Owners Group (WOG) referenced the NOTRUMP computer code as their new computer program for small break loss of coolant accident (SBLOCA) evaluation. NOTRUMP was benchmarked against core uncovery experiments conducted at the Oak Ridge National Laboratory (ORNL). These tests were performed under NRC sponsorship. The good agreement between the calculations and the data confirmed the adequacy of the drift flux model used for core hydraulics as well as the core heat transfer models of clad temperature predictions. Further details of the core model is documented in the staff's SER for NOTRUMP.

The staff finds the core thermal-hydraulic models in NOTRUMP acceptable. This item is resolved.

B. Steam Generator Mixture Level Model

NUREG-0611 requested licensees and applicants with Westinghouse designed NSSSs to justify the adequacy of modeling the primary system of the steam generators as a homogeneous mixture. This question was directed to the WFLASH code. NOTRUMP, the new SBLOCA licensing code models phase separation and incorporates flow regime maps within the steam generator tubes. The adequacy of this model was demonstrated through benchmark analyses with integral experiments, in particular with Semiscale test S-UT-08. Further details of the steam generator model and its validation are documented in the staff's SER for NOTRUMP.

The staff finds the steam generator model in NOTRUMP acceptable. This item is resolved.

C. Noncondensible Affects On Condensation Heat Transfer

NUREG-0611 requested validation of the condensation heat transfer correlations in the Westinghouse SBLOCA model and an assessment of the consequences of noncondensible gases in the primary coolant. The condensation heat transfer model used in NOTRUMP is based on steam experiments performed by Westinghouse on a 16-tube PWR steam generator model. For two-phase conditions, an empirical correlation developed by Shah is applied.

The staff finds the condensation heat transfer correlation in NOTRUMP acceptable.

The influences of noncondensible gases on the condensation heat transfer was demonstrated by degrading the heat transfer coefficient in the steam generators. The heat transfer degradation was calculated using a boundary layer approach. For this calculation, the noncondensible gases generated within the primary coolant system were collected and deposited on the surface of the steam generator tubes. The sources of noncondensibles considered were:

(i) Air dissolved in the RWST.
(ii) Hydrogen dissolved in the primary system.
(iii) Hydrogen in the pressurizer vapor space.
(iv) Radiolytic decomposition of water.

With a degradation factor on the heat transfer coefficient, the limiting SBLOCA was reanalyzed for a typical PWR. The WOG, thereby, concluded that formation of noncondensible gases in quantities that may reasonably be expected for a 4-inch cold leg break LOCA presents no serious detriment on the PWR system response in terms of core uncovery or system pressure. What perturbation was observed was minor in nature.

The staff finds acceptable the Westinghouse submittal on the influences of noncondensible gases on design bases SBLOCA events. Our conclusion is based on the limited amount of noncondensible gases available during a design basis SBLOCA event, as well as results obtained from Semiscale experiments which reached similar conclusions while injecting noncondensible gases in excess amount expected during a SBLOCA design basis event. This item is resolved.

D. Nodalization Studies For Flashing During Depressurization

As a consequence of the staff's experience with modeling SBLOCA events with NRC developed computer codes (in particular the TMI-2 accident), the staff questioned the adequacy of the nodalization in the licensing model to calculate the depressurization of the primary system. The staff therefore requested validation of the Westinghouse Evaluation Model to properly calculate the depressurization expected during a SBLOCA event.

Through nodalization studies and validation of the NOTRUMP licensing model with integral experiments (e.g., LOFT and Semiscale), Westinghouse demonstrated the acceptability of the nodalization and nonequilibrium models. Additional details of these studies are documented in the staff's SER for NOTRUMP.

The staff finds the Westinghouse model acceptable for calculating depressurization during SBLOCA events. This item is resolved.

E. Accumulator Model

WFLASH, the previous Westinghouse small break loss of coolant accident (SBLOCA) analysis code, applied a polytropic gas expansion coefficient of 1.4 to the nitrogen in the accumulators. The WOG was requested to validate this accumulator model in light of data obtained through the LOFT experimental programs for SBLOCAs. Westinghouse reviewed the applicable LOFT data and determined the need to perform full scale accumulator tests. Based upon these tests, Westinghouse modified the polytropic expansion coefficient to a more realistic value. Of interest is Westinghouse's conclusion that the selection of either a high or low expansion coefficient had negligible effect on the calculated peak clad temperature (PCT). This insensitivity is only appropriate to NOTRUMP, with its nonequilibrium assumptions.

The staff finds acceptable the polytropic expansion coefficient in the NOTRUMP code. This item is resolved.

F. Code Validation

Following the Three Mile Island event of 1979, staff analyses with NRC developed computer codes led to concerns that detailed nodalization was required to simulate realistic systems responses to postulated SBLOCAs. As a consequence, licensees and applicants with Westinghouse plants were requested to validate their licensing tools with integral experiments. In specific, the NRC requested that the computer codes be validated with the LOFT L3-1 and L3-7 experimental data. In addition, the staff also requested that the code be benchmarked with the Semiscale S-UT-08 experimental data.

Westinghouse performed the above benchmark analyses. For the LOFT tests, Westinghouse showed good agreement between the NOTRUMP calculations and the experimental data. For the S-UT-08 test, Westinghouse demonstrated that NOTRUMP did a reasonable job calculating the experimental data. However, this required a more detailed nodalization of the steam generators then used in the licensing model. With the less detailed licensing nodalization, the pre-loop-seal-clearing core level depression phenomenon, as observed in the S-UT-08 data, was not conservatively calculated for very small breaks. However, the calculated peak clad temperature was demonstrated to be higher (more conservative) with the coarse nodalization. The staff, therefore, finds acceptable the NOTRUMP computer code and the associated nodalization for SBLOCA design basis evaluation.

This item is resolved.

IV. CONCLUSION

The Westinghouse Owners Group (WOG), by referencing WCAP-10079 and WCAP-10054, have identified NOTRUMP as their new thermal-hydraulic computer program for calculating small break loss of coolant accidents (SBLOCAs). The staff finds acceptable the use of NOTRUMP as the new Westinghouse licensing tool for calculating SBLOCAs for Westinghouse NSSS designs.

The responses to NUREG-0611 concerns, as evaluated within this SER, have also been found acceptable.

This SER completes the requirements of TMI Action Item II.K.3.30 for licensees and applicants with Westinghouse NSSS designs who were members of the WOG and referenced WCAP-10079 and WCAP-10054 as their response to this item.

Within one year of receiving this SER, the licensees and applicants with Westinghouse NSSS designs are required to submit plant specific analyses with NOTRUMP, as required by TMI Action Item II.K.3.31. Per generic letter 83-35, compliance with Action Item II.K.3.31 may be submitted generically. We require that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs.