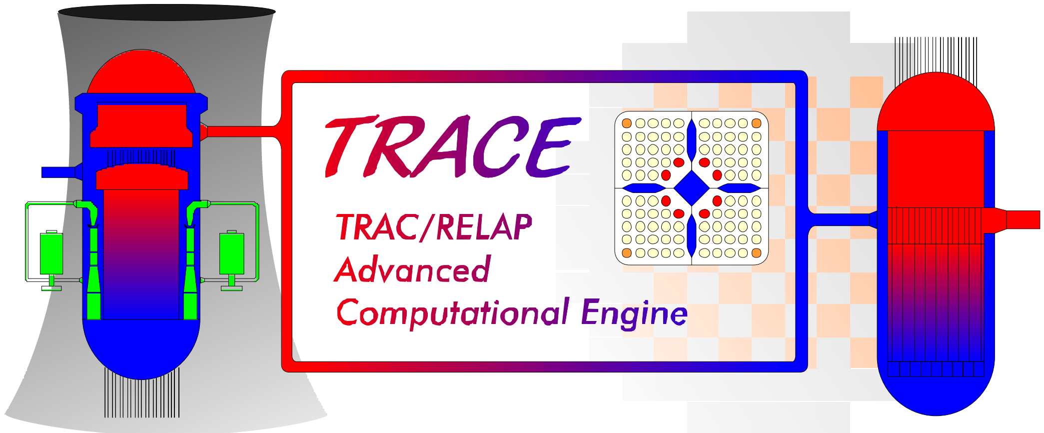

TRACE V5.0
ASSESSMENT MANUAL

Main Report

Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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Many individuals have contributed to the NRC code consolidation effort and to this manual, in particular. In a project of this magnitude and complexity, and given the long histories of the NRC predecessor codes and their associated manuals (from which this manual has evolved), it is rather difficult to sort out and keep track of each and every individual contribution of authorship. Rather than attempt to cite individual contributors to this particular manual (and run the risk of excluding somebody, either past or present), we simply acknowledge all known contributors to the TRACE code development and assessment project, in general.

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Executive Summary

TRACE Version 5.0 Assessment Report

The TRAC/RELAP Advanced Computational Engine (TRACE - formerly called TRAC-M) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in light water reactors. It is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. TRACE Version 5.0 represents the initial release of this analysis code.

TRACE is designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in pressurized light-water reactors (PWRs) and boiling light-water reactors (BWRs). It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. The models and correlations in TRACE provide the capability to model multidimensional two-phase flow, nonequilibrium thermodynamics, core and system heat transfer and reactor kinetics.

This report is one of four documents that comprise the basic TRACE documentation set. The other three are the Theory Manual, the Theory Manual Supplement, and the User's Guide. The Theory Manual, and the Theory Manual Supplement provide a detailed description of the models and correlations used by the code. The User's Guide provides user of TRACE a description of the input variables and instructions on how to execute the code. The purpose of this report is to document the performance of TRACE Version 5.0. This is accomplished through the simulation of numerous experimental tests and comparison of predicted and measure results. Because TRACE is intended to have a wide range of applicability, a large number of assessment cases are necessary. The simulations were selected based on phenomena identified as being important in Phenomena Identification and Ranking Tables (PIRTs) that have been proposed for large and small break loss of coolant accidents. For each phenomena, assessments are performed to evaluate TRACE's ability to model and simulate the phenomena. Large scale systems tests are also included as part of TRACE assessment to demonstrate the code's ability to simulate integral systems performance. The assessments demonstrate that TRACE is capable of simulating the phenomena expected in these types of hypothetical accidents.

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Foreword

Advanced computing plays a critical role in the design, licensing and operation of nuclear power plants. The modern nuclear reactor system operates at a level of sophistication whereby human reasoning and simple theoretical models are simply not capable of bringing to light full understanding of a system's response to some proposed perturbation, and yet, there is an inherent need to acquire such understanding. Over the last 30 years or so, there has been a concerted effort on the part of the power utilities, the U. S. Nuclear Regulatory Commission (USNRC), and foreign organizations to develop advanced computational tools for simulating reactor system behavior during real and hypothetical transient scenarios. The lessons learned from simulations carried out with these tools help form the basis for decisions made concerning plant design, operation, and safety.

Confidence in the computational tools, and establishment of their validity for a given application depends on assessment. TRACE, like other two-fluid codes, is composed of numerous models and correlations. When applied to full scale nuclear power plant conditions, many of these models and correlations can be applied outside of their original database. By assessing the code against well-scaled thermal-hydraulic tests, it is possible to show that the code and its constituent model packages can be extended to conditions beyond those for which many of the individual correlations were originally intended. The assessment process however, can also indicate potential deficiencies in the code and its constituent models and correlations. This provides motivation and guidance for eventual code improvement. The assessment reported in this document thus has two objectives; to identify the phenomena and range of conditions over which TRACE Version 5.0 can be used, and to identify those phenomena and processes in which the code may have shortcomings.

The assessment matrix for TRACE Version 5.0 is large, and considers all of the major thermal-hydraulic processes encountered in large and small break LOCAs in conventional light water reactors. Models have been added to TRACE and assessed for conditions expected in advanced light water reactors. However, because of the unique and often proprietary nature of test data for these advanced plants, that assessment is documented in supplements to this report.

This document is organized into four overall segments. Sections 1 through 6 summarize the performance of TRACE, and describe the processes for which assessment was performed. Appendix A contains a set of basic test cases designed to demonstrate that TRACE can calculate some of the most fundamental physical processes such as two-dimensional conduction in solids and two-phase pressure drop in a simple pipe. Appendix B documents TRACE assessment against a wide range of separate effects tests. These tests examine the capability of TRACE to simulate specific reactor thermal-hydraulic processes and events such as reflood heat transfer and emergency core coolant (ECC) bypass. Simulation of separate effects test data help to assess specific model packages within TRACE under a range of thermal-hydraulic conditions. Appendix C contains the results of TRACE assessment against integral effect test data. The integral effects test facilities include facilities for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), and include a range of physical scale.

1

Introduction

The TRAC/RELAP Advanced Computational Engine (TRACE) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state thermal-hydraulic behavior in light water reactors. It is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. Previously, TRACE was referred to as TRAC-M. The TRACE name was adopted in order to differentiate the present code with earlier versions. Through the assessment process, many of the thermal-hydraulic models adopted into TRAC-M were found to be deficient and new models and corrections were implemented. Because of the large number of thermal-hydraulic model revisions, TRACE should be considered to be a new code as opposed to a simple revision of one of its predecessor codes.

The report documents assessment performed for TRACE Version 5.0. All simulations reported in this document were performed with a "frozen" version of the code so that differences between the various cases can not be attributed to changes in models and correlations. This manual is accompanied by three other reports; a Theory Manual that describes the models and correlations in TRACE Version 5.0, a Theory Manual Supplement which provides details on the various correlations and helps to explain why certain correlations were selected for us in the code, and a User's Guide that provides a detailed description of the input specifications and modeling recommendations. Future major releases of the TRACE code will go out as a set, so that as improvements to the code are made, the effect of those changes on the simulations and code performance can be made readily apparent.

TRACE Version 5.0 is intended to be used for large and small break loss of coolant accident analysis in conventional light water reactors. This version of TRACE is also capable of simulating the ESBWR, although specific assessment relating to that plant is documented in a separate report. While development has been performed to make TRACE capable of simulating events in Advanced CANDU reactors as well as the AP1000, appropriate assessment for those plant types has not yet been conducted. TRACE is also capable of performing transient analysis for events such as steam generator tube ruptures, and main steam line breaks. For transients in which the analyst expects significant feedback between the thermal-hydraulics and nuclear kinetics, TRACE can and should be coupled with the PARCS kinetics package.

The assessment process may be viewed as consisting of the following tasks:

- Formulate assessment matrices for each class of transient studied, e.g., large-break LOCA, small-break LOCA.
- Select key parameters for these classes.
- Perform assessment calculations. Compare the test data with the results of calculations for the key parameters.
- Perform uncertainty analyses for the capability of the codes to predict test data and plant transients.

This report addresses the first three steps of the assessment process. While the basis for uncertainty analysis comes from comparison of predicted and measured results, the effect of model uncertainties are plant and model dependent. This step will be addressed in other documents as uncertainty methodolgies for TRACE application are developed.

The assessment that is documented in this report is directed primarily at the phenomena that are expected to occur during transients and large and small break LOCAs. These processes are discussed in the following sections, and in general are consistent with the highly ranked thermal-hydraulic processes identified in several Phenomena Identification and Ranking Tables (PIRTs) that have been published in the open literature. The assessment matrix to accomplish this is large as a result.

This report consists of four main parts; a main body and three appendices. Sections 1 through 6 discuss the phenomena and processes examined, and summarize the overall results of the assessment and comparisons to experimental data. Where possible, the assessment results are characterized with a bias and uncertainty in order to provide guidance to analysts when applying the code and to code developers for future model improvements. In particular, Section 8 describes the code deficiencies and large uncertainties that exist in the TRACE Version 5.0 thermal-hydraulic models.

Individual assessments are documented in Appendices A, B, and C. Appendix A contains a set of fundamental assessment cases. These include analytical test problems in which there are known solutions and for which behavior is well known. Appendix A also includes comparisons of TRACE predictions to data from fundamental tests involving two-phase flow. These simulations help to insure the numerics and most basic models and correlations are correct.

Appendix B documents comparisons of TRACE predictions of a wide variety of separate effects tests. Simulation of these tests examine various model packages in TRACE and their ability to predict phenomena such as two-phase critical flow, reflood thermal-hydraulics, and condensation. Individual tests included in these assessments were selected so as to assess TRACE over a broad range of thermal-hydraulic conditions.

Appendix C documents comparisons of TRACE predictions against data for several integral test facilities. These assessments demonstrate the ability of TRACE to simulate transient events in large scale facilities representing PWRs and BWRs. These simulations are useful in understanding the code’s ability to predict system-wide effects.

2

Overview of the TRACE Code

Description of TRACE

The TRAC/RELAP Advanced Computational Engine (TRACE - formerly called TRAC-M) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in light water reactors. It is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool.

TRACE has been designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in pressurized light-water reactors (PWRs) and boiling light-water reactors (BWRs). It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. Models used include multidimensional two-phase flow, nonequilibrium thermodynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Automatic steady-state and dump/restart capabilities are also provided.

The partial differential equations that describe two-phase flow and heat transfer are solved using finite volume numerical methods. The heat-transfer equations are evaluated using a semi-implicit time-differencing technique. The fluid-dynamics equations in the spatial one-dimensional (1D), two-dimensional (2D), and three-dimensional (3D) components use, by default, a multi-step time differencing procedure that allows the material Courant-limit condition to be exceeded. A more straightforward semi-implicit time-differencing method is also available, should the user demand it. The finite-difference equations for hydrodynamic phenomena form a system of coupled, nonlinear equations that are solved by the Newton-Raphson iteration method. The resulting linearized equations are solved by direct matrix inversion. For the 1D network matrix, this is done by a direct full-matrix solver; for the multiple-vessel matrix, this is done by the capacitance matrix method using a direct banded-matrix solver.

TRACE takes a component-based approach to modeling a reactor system. Each physical piece of equipment in a flow loop can be represented as some type of component, and each component can be further nodalized into some number of physical volumes (also called cells) over which the fluid, conduction, and kinetics equations are averaged. The number of reactor components in the problem and the manner in which they are coupled are arbitrary. There is no built-in limit for the number of components or volumes that can be modeled; the size of a problem is theoretically only limited by the available computer memory. Reactor hydraulic components in TRACE include PIPEs, PLENUMs, PRIZERs (pressurizers), CHANs (BWR fuel channels), PUMPs, JETPs (jet pumps), SEPDs (separators), TEEs, TURBs (turbines), HEATRs (feedwater heaters), CONTANs (containment), VALVEs, and VESSELs (with associated internals). HTSTR (heat structure) and REPEAT-HTSTR components modeling fuel elements or heated walls in the reactor system are available to compute 2D conduction and surface-convection heat transfer in Cartesian or cylindrical geometries. POWER components are available as a means for delivering energy to the fluid via the HTSTR or hydraulic component walls. FLPOWER (fluid power) components are capable of delivering energy directly to the fluid (such as might happen in waste transmutation facilities). RADENC (radiation enclosures) components may be used to simulate radiation heat transfer between multiple arbitrary surfaces. FILL and BREAK components are used to apply the desired coolant-flow and pressure boundary conditions, respectively, in the reactor system to perform steady-state and transient calculations. EXTERIOR components are available to facilitate the development of input models designed to exploit TRACE's parallel execution features.

The code's computer execution time is highly problem dependent and is a function of the total number of mesh cells, the maximum allowable timestep size, and the rate of change of the neutronic and thermal-hydraulic phenomena being evaluated. The stability-enhancing two-step (SETS) numerics in hydraulic components allows the material Courant limit to be exceeded. This allows very large time steps to be used in slow transients. This, in turn, can lead to significant speedups in simulations (one or two orders of magnitude) of slow-developing accidents and operational transients.

Historical Background

TRACE Version 5.0 is the latest in a series of TRAC codes produced by the NRC. Previous versions include TRAC-M, TRAC-PD2/MOD1, TRAC-PF1, TRAC-PD2, TRAC-P1A, and TRAC-P1. Each successive version benefited from advances in computing speed and numerical solution techniques so that the most recent versions are capable of tracking multiple components (water, steam, liquid solute and non-condensable gas) in a three-dimensional flow field. Nodalization of reactor systems have also become more detailed and complex.

The first publicly released version, TRAC-P1 (Ref. 2-1), was completed in December 1977, following a preliminary version completed in 1976 that consisted of only 1D components. It was in this initial TRAC-P1 development that 1D loop component models were first used.

TRAC-P1 was designed primarily for analysis of large break LOCAs in PWRs. However, because of its generality, TRAC-P1 could be applied to many types of analyses ranging from blowdowns in simple pipes to integral tests in multiloop facilities. A refined version, TRAC-P1A (Ref. 2-2), was released in March 1979. TRAC-P1A was more efficient than TRAC-P1 and incorporated improved hydrodynamic and heat transfer models. It was also easier to implement on various computers. TRAC-PD2 (Ref. 2-3) contained additional improvements in reflood and heat transfer models, along with an improved numerical solution method. While designed as a LBLOCA code, TRAC-PD2 was applied to small break LOCA problems and to the Three Mile Island accident.

TRACE-PF1 (Ref. 2-4) was designed to improve the ability of TRAC-PD2 to handle SBLOCAs and other transients. TRAC-PF1 had all the major improvements of TRAC-PD2. But in addition, it used a two-fluid model with stability-enhancing two-step (SETS) numerics in the 1D components. The two-step numerics allowed for large time steps during slow transients. A non-condensable gas field was added to the 1D and 3D hydrodynamics, and significant improvements were made to the trip logic and input. TRAC-PF1 was released publicly in July 1981.

The development of TRAC-PF1/MOD1 (Ref. 2-5) maintained the models (Ref. 2-6) necessary for applying the code to LBLOCAs and added or modified models as necessary to enhance the application of the core to SBLOCAs and operational transients. In particular, a number of user conveniences to promote application of the code to transients involving complex control of the plant were added. TRAC-PF1 contained generalized reactivity-feedback, trip, and control system modeling and additional components to model the balance of plant.

TRAC-PF1/MOD2 Version 5.4 (Ref. 2-7) improved upon the numerical solution schemes and closure relationships in previous code versions. Version 5.4 was used in a number of developmental assessment calculations however, the results of those calculations were never officially published.

TRAC-M/F77, Version 5.5 was released in December 1999 (Ref. 2-8). This version implemented standard F77 throughout the code, and new reflood model was added. Reference 2-9 reported the assessment performed to validate the code. TRAC-PF1/MOD2 Version 5.4.25 served as the base code version for TRAC-M development.

TRAC-M/F90 (Ref. 2-10) was developed starting with TRAC-M/F77 Version 1.10 as the starting point. TRAC-M/F90 included a complete rewrite of the databases in Fortran 90 which improved portability and allowed a more modular programming structure. This enabled the incorporation of boiling water reactor modeling capabilities similar to those in TRAC-B. These enhancements inlcuded addition of CHAN, JETP Components and replacement of the MOD2 versions of the TURB and SEPD Components. Other modifications included a single junction component to facilitate RELAP style modeling. Near the completion of these enhancements to TRAC-M, the code was renamed TRACE in order to distinguish it from its predecessors.

TRACE Characteristics

Some distinguishing characteristics of the TRACE are summarized below.

Variable-Dimensional Fluid Dynamics

A 3D (x, y, z) Cartesian- and/or (r, θ, z) cylindrical-geometry flow calculation can be simulated within the reactor vessel or other other reactor components where 3D phenomena take place. All 3D components, such as Reactor Water Storage Tank, where 3D phenomena are modeled, are named VESSEL although they may not have any relationship with the reactor vessel. Flows within a coolant loop are usually modeled in one dimension using PIPE and TEE components. The combination of 1D and 3D components allows an accurate modeling of complex flow networks as well as local multidimensional flows. This is important in determining emergency core coolant (ECC) downcomer penetration during blowdown, refill, and reflood periods of a LOCA. The mathematical framework exists to directly treat multidimensional plenum- and core flow effects, and upper-plenum pool formation and core penetration during reflood.

Non-homogeneous, Non-equilibrium Modeling

A full two-fluid (six-equation) hydrodynamic model evaluates gas-liquid flow, thereby allowing important phenomena such as countercurrent flow to be simulated explicitly. A stratified-flow regime has been added to the 1D hydrodynamics; a seventh field equation (mass balance) describes a noncondensable gas and/or steam field; and an eighth field equation tracks dissolved solute in the liquid field that can plated out on surfaces when solubility in the liquid is exceeded.

Flow-Regime-Dependent Constitutive Equation Package

The thermal-hydraulic equations describe the transfer of mass, energy, and momentum between the steam-liquid phases and the interaction of these phases with heat flow from the modeled structures. Because these interactions are dependent on the flow topology, a flow-regimedependent constitutive-equation package has been incorporated into the code. Assessment calculations performed to date indicate that many flow conditions can be calculated accurately with this package.

Comprehensive Heat Transfer Capability

TRACE can perform detailed heat-transfer analyses of the vessel and the loop components. Included is a 2D (r,z) treatment of fuel HTSTR component. Heat conduction with dynamic finemesh rezoning during reflood simulates the heat transfer characteristics of quench fronts. Heat transfer from the fuel rods and other structures is calculated using flow-regime-dependent HTCs obtained from a generalized boiling curve based on a combination of local conditions and history effects. Inner- and/or outer-surface convection heat-transfer and a tabular or point-reactor kinetics with reactivity feedback volumetric power source can be modeled.

Component and Functional Modularity

The TRACE code is completely modular by component. The components in a calculation are specified through input data; available components allow the user to model virtually any PWR or BWR design or experimental configuration. Thus, TRACE has great versatility in its range of applications. This feature also allows component modules to be improved, modified, or added without disturbing the remainder of the code. TRACE component modules currently include BREAKs, FILLs, CHANs, CONTANs, EXTERIORs, FLPOWERs, HEATRs, HTSTRs, JETPs, POWERs, PIPEs, PLENUMs, PRIZERs, PUMPs, RADENCs, REPEAT-HTSTRs, SEPDs, TEEs, TURBs, VALVEs, and VESSELs with associated internals (downcomer, lower plenum, reactor core, and upper plenum).

The TRACE program is also modular by function; that is, the major aspects of the calculations are performed in separate modules. For example, the basic 1D hydrodynamics solution algorithm, the wall-temperature field solution algorithm, heat transfer coefficient (HTC) selection, and other functions are performed in separate sets of routines that can be accessed by all component modules. This modularity allows the code to be upgraded readily with minimal effort and minimal potential for error as improved correlations and test information become available.

Physical Phenomena Considered

As part of the detailed modeling in TRACE, the code can simulate physical phenomena that are important in large-break and small-break LOCA analyses, such as:

- 1) ECC downcomer penetration and bypass, including the effects of countercurrent flow and hot walls;
- 2) lower-plenum refill with entrainment and phase-separation effects;
- 3) bottom-reflood and falling-film quench fronts;
- 4) multidimensional flow patterns in the reactor-core and plenum regions;
- 5) pool formation and countercurrent flow at the upper-core support-plate (UCSP) region;
- 6) pool formation in the upper plenum;
- 7) steam binding;
- 8) average-rod and hot-rod cladding-temperature histories;
- 9) alternate ECC injection systems, including hot-leg and upper-head injection;
- 10) direct injection of subcooled ECC water, without artificial mixing zones;
- 11) critical flow (choking);
- 12) liquid carryover during reflood;
- 13) metal-water reaction;

- 14) water-hammer pack and stretch effects;
- 15) wall friction losses;
- 16) horizontally stratified flow, including reflux cooling,
- 17) gas or liquid separator modeling;
- 18) spacer grids in fuel-rod assemblies;
- 19) noncondensable-gas effect on evaporation and condensation;
- 20) dissolved-solute tracking in liquid flow;
- 21) reactivity-feedback effects on reactor-core power kinetics;
- 22) two-phase bottom, side, and top offtake flow of a tee side channel; and reversible and irreversible form-loss flow effects on the pressure distribution

Restrictions on Use

The TRACE code is not appropriate for transients in which there are large changing asymmetries in the reactor-core power such as would occur in a control-rod-ejection transient unless it is used in conjunction with the PARCS spatial kinetics module. In TRACE, neutronics are evaluated on a core-wide basis by a point-reactor kinetics model with reactivity feedback, and the spatially local neutronic response associated with the ejection of a single control rod cannot be modeled.

The typical system model cannot be applied directly to those transients in which one expects to observe thermal stratification of the liquid phase in the 1D components. The VESSEL component can resolve the thermal stratification of liquid only within the modeling of its multidimensional noding when horizontal stratification is not perfect.

TRACE does not evaluate the stress/strain effect of temperature gradients in structures. The effect of fuel-rod gas-gap closure due to thermal expansion or material swelling is not modeled explicitly. TRACE can be useful as a support to other, more detailed, analysis tools in resolving questions such as pressurized thermal shock.

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3

Thermal-Hydraulic Processes

Introduction

One of the most important tasks associated with the development of a best estimate thermal-hydraulics code is identification of the processes and phenomena that have the most dominant influence on transients of interest. TRACE is intended for wide application. It is intended for use with both PWRs and BWRs for accident analyses that include large break LOCA, small break LOCA, steam generator tube ruptures, and other typical Chapter 15 scenarios. These transients and plant types require TRACE to be able to perform adequately for a large number of physical processes. To identify the physical processes and thermal-hydraulic phenomena that are most important for TRACE assessment, Phenomena Identification and Ranking Tables (PIRTs) were reviewed for these intended applications. The following sections discuss available PIRTs, and the phenomena that are commonly considered to be among the most important. The assessment matrix for TRACE is then presented.

Phenomena Identification and Ranking Tables

The primary purpose of a Phenomena Identification and Ranking Table (PIRT) is to rank, relative to a figure of merit, the physical processes and phenomena that affect plant behavior during a particular transient event. PIRTs are also valuable in setting the objectives for separate and integral effects programs, code development and improvement programs, and in code uncertainty quantification programs. The PIRT process is described by Wilson and Boyack (Ref. 3-1), and the general approach described by them has been used by several developers of best-estimate thermal-hydraulic codes. Development of a PIRT is one of the cornerstones of the Code Scaling, Applicability and Uncertainty (CSAU) (Ref. 3-2) method for best-estimate thermal-hydraulic code applications. With the CSAU method, the PIRT is used to identify the most important processes for a particular scenario and help insure that those processes are be accounted for in the thermal-hydraulics code and are addressed in an uncertainty evaluation. As such, a PIRT is generic and not specific to any code.

In this report PIRTs are used to insure that the TRACE code is assessed for and adequately predicts the phenomena to LOCA and plant transients. The phenomena and assessments provide the basis by which model uncertainty ranges can be established so that TRACE can be incorporated with a statistical uncertainty methodology.

Evaluation Criteria and Ranking Scales

Evaluation criteria for a PIRT depend on the transient of interest, and the regulatory intent for the proposed application. Frequently, PIRTs developed for large break LOCA consider the figure of merit to be the effect of a process on peak cladding temperature or cladding oxidation. PIRTs for small break LOCA or passively cooled reactor systems are concerned with the two-phase level in the reactor vessel, which directly impacts the PCT and cladding oxidation. In the identification of the processes and phenomena for assessment of TRACE, peak cladding temperature and cladding oxidation are the regulatory figures of merit.

Several different techniques have been used by the expert panels to reach a consensus in the relative ranking of the phenomena. A formal ranking technique, the Analytical Hierarchy Process (AHP) was used by one of the groups in the CSAU study. Both panels in the CSAU study used a numerical scale from 1 to 9 to rank the processes, with 9 representing a process considered very important. Most PIRT panel since then have adopted a simpler scale in which a "high" ranking means the phenomena significantly impacts the figure of merit, a "medium" ranking means the phenomena has a moderate impact, and a "low" ranking means the phenomena has a minor impact. Phenomena that are not expected to occur or physically can not impact the evaluation criteria are assigned an "inactive" or "not applicable" ranking.

In this report, the simpler "high", "medium" and "low" ranking terminology will be used in discussing the relative importance of various phenomena.

LOCA Transient Behavior

PIRTs and the relative rank of the processes involved are plant and scenario dependent. Therefore, it is useful to discuss in general terms the plants and transients assumed in the PIRTs.

PWR Plant and Large LOCA Description

The plant type of primary consideration is a Westinghouse 4-loop PWR. The primary coolant system of a Westinghouse PWR consists of the multi-loop arrangement as shown in **Figure 3-1**. In a typical 4-loop configuration, each loop has a vertically oriented steam generator and a coolant pump. The coolant flows through the steam generator within an array of U-tubes that connect inlet and outlet plenums. The system includes a pressurizer, which is connected to one of the hot legs.

During normal operation, flow enters the reactor vessel through inlet nozzles that connect the cold legs to the downcomer annulus that is formed between the inside of the reactor vessel and the outside of the core barrel. Coolant flows downward through this annulus to the inlet plenum formed by the lower head of the vessel. From the lower plenum the coolant flows upward through the reactor core into the upper plenum. From the upper plenum, the coolant flows through the hot legs to the steam generator.

The design basis accident is a double-ended guillotine break in a cold leg between the reactor coolant pump and the reactor vessel.

The blowdown period (0-30 seconds) is the result of a break in the coolant system through which the primary coolant is expelled. Within a fraction of a second following the break, the core begins to void and the fuel rod go through departure from nucleate boiling. The negative void reactivity rapidly shuts down fission heat in the core. With diminished cooling, the cladding heat up. Interaction between the pump and break cause intermittent flow reversals. The primary system pressure decreases rapidly, and injection of coolant from the cold leg accumulators begins. Much of the flow early in blowdown however is swept around the downcomer and out the break due to high steam flows in the downcomer. As the blowdown progresses, some of the accumulator water collects in the downcomer and penetrates to the lower plenum. The peak cladding temperature during the blowdown phase can be 1089 K (1500 F) or more assuming a loss of offsite power and the worst single failure in the emergency core cooling system.

The refill period occurs between about 30 and 40 seconds following the start of the LOCA. The primary pressure has decrease to a level so that the low pressure safety injection system activates and begins to inject. The lower plenum begins to fill with water from the accumulator as coolant bypass around the downcomer ends. While the low plenum is refilling, the cladding heats up at a near adiabatic rate due to decay heat and stagnant flow conditions in the core. Some rods swell and burst during this period.

The reflood period begins at approximately 40 seconds as the low plenum fills and coolant begins to re-enter the core. Water fills the downcomer and provides the driving head for refilling the core. The bottom of the core quenches, generating a two-phase mixture that provides cooling to the upper elevations of the core. The fuel rods continue to heat however until the quench front moves upward through the core. During the reflood period, additional rods may swell and burst. Zirconium-water reaction can occur for high temperature regions of the core. As the quench front continues to advance, upper elevations are cooled by a dispersed, non-equilibrium two-phase mixture of superheated steam and entrained droplets. Eventually, cladding temperatures reach a maximum and begin to decrease until the entire core is quenched.

PWR Small Break LOCA Description

Breaks with flow area between approximately 7.125×10^{-5} and $0.7854\ \mathrm{m}^2$ (7.67×10^{-4} and 1.0 ft^2 are typically considered small break LOCAs. Breaks in this range are sufficiently large enough to depressurize the primary system to the high pressure safety injection set point and cause the

HPSI system to activate. Breaks smaller than 7.125×10^{-5} m^2 are considered "leaks" and the reactor charging system is usually sufficient to make up lost inventory and prevent rapid depressurization. (This area corresponds to a 3/8-inch diameter hole.)

The limiting small break size depends on the core power level, performance of the high head safety system, and the accumulator backpressure. The limiting break is one that is large enough so that the break flow exceeds the HPSI system make up capacity, but is small enough so that the primary system does not quickly depressurize to the accumulator setpoint. During the time it takes to depressurize to the accumulator setpoint, core uncovery can occur. For most plants, the limiting break hole size is 0.0508 to 0.1016 m (2 to 4 inches) in diameter.

A small break transient is characterized by five periods; blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. While the duration of each period is break size dependent, the small LOCA transient can be characterized as follows:

Blowdown: On initiation of the break, there is a rapid depressurization of the primary side of the RCS. Reactor trip is initiated on a low pressurizer pressure setpoint. Pump trip occurs either automatically at reactor trip (if the assumption is made that off-site power is lost coincident with reactor trip), or by the operators approximately 15-45 seconds following reactor trip if off-site power is available, based on plant Emergency Operating Procedures. Loss of condenser steam dump effectively isolates the steam generator secondary side causing it to pressurize to the safety valve setpoints, and release steam through the safety valves. A safety injection signal occurs when the primary pressure decreases below the pressurizer low-low pressure setpoint and SI begins after a signal delay time. The RCS remains liquid solid for most of the blowdown period, with phase separation starting to occur in the upper head, upper plenum and hot legs near the end of this period. During the blowdown period, the break flow is single phase liquid only. Eventually, the rapid depressurization ends and the RCS reaches a pressure just above the steam generator secondary side pressure.

Natural Circulation: At the end of the blowdown period, the RCS reaches a quasi-equilibrium condition which can last for several hundred seconds depending on break size. During this period, the loop seals remain plugged and the system drains from the top down with voids beginning to form at the top of the steam generator tubes and continuing to form in the upper head and top of the upper plenum region. Decay heat is removed by the steam generators during this time. Vapor generated in the core is trapped within the RCS by liquid plugs in the loop seals, and a low quality flow exits the break. This period is referred to as the natural circulation period.

Loop Seal Clearance: The third period is the loop seal clearance period. When the liquid level in the downhill side of the steam generator is depressed to the elevation of the loop seal, steam previously trapped in the RCS can be vented to the break. The break flow, previously a low quality mixture, transitions to primarily steam. Prior to loop seal venting, the inner vessel mixture level can be drop rapidly, resulting in a deep but short core uncovery. Following loop seal venting, the core level recovers to about the cold leg elevation, as pressure imbalances throughout the RCS are relieved.

Boil-Off: Following loop seal venting, the vessel mixture level will decrease. In this period, the decrease is due to the gradual boil-off of the liquid inventory in the reactor vessel. The mixture level will reach a minimum, in some cases resulting in a deep core uncovery. The boil-off period ends when the core collapsed liquid level reaches a minimum. At this time, the RCS has depressurized to the accumulator setpoint, and the core boil-off rate matches the delivery of safety injection to the vessel.

Core Recovery: The core recovery period extends from the time at which the inner vessel mixture level reaches a minimum in the boil-off period, until all parts of the core quench and are covered by a low quality mixture. The small break LOCA is over and the long-term cooling period begins when once the entire core is quenched and the safety injection flow exceeds the break flow.

BWR Plant and LOCA Description

Steam and recirculation water flow paths in BWR are shown in **Figure 3-2**. The steam-water mixture first enters steam separators after exiting the core. After subsequent passage through steam dryers located in the upper portion of the reactor vessel, the steam flows directly to the feedwater system. The water, which is separated from the steam, flows downward in the periphery of the reactor vessel and mixes with the incoming main feed flow from the turbine. This combined flow stream is pumped into the lower plenum through jet pumps mounted around the inside periphery of the reactor vessel. The jet pumps are driven by flow from recirculation pumps located in relatively small-diameter external recirculation loops, which draw flow from the plenum just above the jet pump discharge location.

The design basis accident for a BWR/6 is a double-ended break in the suction-side of the recirculation line.

Shortly after the break, the reactor scrams, typically on drive flow pressure. Because of the large flow reductions immediately following the LOCA caused by the depressurization, there is a rapid increase in the core average void fraction. The negative void reactivity rapidly shuts down the core. The flow reverses in the broken loop jet pump. With the flow reversal all the drive flow to that jet pump is lost and one-half the drive flow that is supporting the core flow is lost.

A loss of offsite power is also assumed. Thus, there is no power to the recirculation pump, which means that the intact loop pump also starts to coast down. The coastdown time of the pump is on the order of 10-15 seconds. With the loss of pumped flow, there is an almost instantaneous and large reduction in the core flow, which causes an early boiling transition in the core, typically within one second after the break. At this time the cladding temperature rapidly increases. The resulting blowdown peak cladding temperature is dominated by the stored energy in the fuel.

Valves are closed to isolate the system, typically within four seconds after the LOCA. System depressurization and loss of liquid inventory continue. As a result of the loss of inventory, the water level in the downcomer decreases and as the water level eventually drops down to the top of the jet pump. This opens a flow path through which steam can flow to the break. The rate of depressurization increases following jet pump uncovery.

During normal operation, the inlet subcooling at the bottom of the core is 11 K (20 F). With the rapid depressurization, there is a large amount of flashing of the fluid in the lower plenum, this occurring at approximately 10 s. This causes a large increase in the coolant flow through the core, quenching the fuel, and returning the cladding temperature to the saturation temperature. As the LOCA and depressurization continue, the level inside the core region decreases, as well as forming a level in the lower plenum region. The flow into the core is limited and the core uncovery leads to a second boiling transition. That typically happens at approximately 20 seconds into the transient. Within 35-40 s following the LOCA, the high pressure core spray system begins to deliver coolant to the top of the core, the time being determined by the time to start the diesel generator that drives the high pressure core spray system. The low pressure injection begins when the system pressure drops below the shutoff head for the pumps, typically on the order of about 200 psi. A second transition and core heatup begins in the period 20-35 s. This heatup is terminated by the operation of the BWR-6 safety systems. The BWR-6 has one high-pressure coolant system, one low-pressure core spray system, and three low-pressure coolant injection (LPCI) systems injecting into the bypass region. The worst single failure for the BWR-6 is the failure of one of the diesel generators that will drive two of the LPCI systems. The outcome of this failure is that the system behavior is based on the availability of the high-pressure core spray, the low-pressure core spray and one LPCI system that injects into the bypass region. Given the operation of these systems, the core refills before the lower plenum. The refilling and reflooding processes restores the liquid inventory in the core and quenches the core in the period 100-150 s following the LOCA. Throughout the transient, the best-estimate peak cladding temperature for nominal conditions is approximately 700 K (800 F). The upper bound estimate for a 95%/95% upper bound is approximately 922 - 978 K (1200-1300 F).

For the BWR-4, the ECC configuration is slightly different. However, The early part of the transient is very similar to the BWR-6. These differences cause the core reflood during the refilling and reflooding phase of the LOCA to take somewhat longer than in a BWR-6. This results in a somewhat higher peak cladding temperature for the BWR-4, with the peak cladding temperature for nominal conditions being approximately 1000 'F and the upper bound estimate approximately 1033 - 1089 K (1400-1500 F). The BWR-2 is the older-generation BWR without jet pumps. The core cannot be reflooded. The peak cladding temperature is controlled by a balance between decay heat and the core spray heat transfer. Typically, the peak cladding temperature occurs late in the transient, perhaps 600-800 seconds following the LOCA. Quenching of the fuel rods is also very slow. The upper bound peak cladding temperature for the BWR-2 is approximately 1200 K (1700 F). For these plants, cladding oxidation, rather than PCT, may be limiting.

BWRs are designed to automatically convert postulated small-breaks that would uncover the core into a large-break through the activation of an Automatic Depressurization System (ADS). The ADS opens several of the standard safety relief valves, causing a controlled depressurization with system response quite similar to that for a postulated large break in the reactor steam line. Thus, both large and small break LOCAs in a BWR can be addressed through a single PIRT.

Applicable PIRTs

This section describes several PIRTs used to help determine the assessment matrix for TRACE. While PIRTs for advanced plants such as AP1000 and ESBWR were examined, the phenomena of interest in this assessment report are for conventional light water reactors. In many cases however, the phenomena important in conventional LWR LOCAs are also important in the advanced plants so the assessments performed in this report also have value in understanding the performance of TRACE with passively cooled designs. In order to insure that TRACE is fully assessed for new and advanced plants, the Code Applicability Report for that design should be consulted. As of this writing, TRACE Code Applicability Reports are in preparation or planned for ESBWR and the EPR. The following sections discuss publicly available PIRTs for large and small break LOCA in conventional plants.

Large Break PIRTs

The original CSAU application was performed for a Westinghouse 4-loop PWR assuming a large break LOCA scenario. The PIRT that was developed for CSAU is reported by Shaw et al. (Ref. 3-3). Phenomena were ranked by two independent committees; a group of experts and a group of thermal-hydraulic analysts from Idaho National Engineering Laboratory (INEL). The findings of the two committees were in good agreement. The important phenomena in each LBLOCA phase are summarized as follows. For blowdown, the fuel rod stored energy, break flow, and RCP degradation have the highest rank. During refill, the highest-ranked phenomena are cold leg and downcomer condensation (ECCS bypass related) and downcomer multi-dimensional flow. For reflood, the core reflood heat transfer, void generation, three-dimensional flow, entrainment and de entrainment in the upper plenum and hot legs, steam binding, and the effect of noncondensible gases in the cold legs received the highest rating. Table 3-1 lists a summary of the highest ranked phenomena addressed in the CSAU study.

Table 3-1. Summary of Highest Ranked Large Break LOCA Phenomena (Ref. 3-2)

Phenomena	Component(s)	Blowdown	Refill	Reflood
Stored energy	Fuel rod	●		
Oxidation	Fuel rod			⊖
Decay heat	Fuel rod			⊖
Gap conductance	Fuel rod			⊖
Post-CHF heat transfer	Core		⊖	
Reflood heat transfer	Core			●
Minimum film boiling temperature	Core	⊖	⊖	

Table 3-1. Summary of Highest Ranked Large Break LOCA Phenomena (Ref. 3-2)

Phenomena	Component(s)	Blowdown	Refill	Reflood
Stored energy	Fuel rod	●		
Entrainment / De-entrainment	Upper plenum, Hot Leg			●
Steam binding	Steam generator			●
Critical flow	Break	●	⊖	
Three-dimensional flow	Core		●	●
Voiding	Core			●
Pressurizer early quench	Pressurizer	⊖		
Condensation	Downcomer, Cold Leg		●	
Hot wall heat transfer	Downcomer			⊖
Two-phase performance	Pump	●		
ΔP, form losses	Pump			⊖
Noncondensible gas partial pressure	Cold leg/Accumulator			●
Two-phase ΔP	Loop	⊖		
Oscillations	Loop		⊖	●
Flow split	Loop		⊖	

In this table, the symbol ● denotes a ranking by at least one of the two panels of 9, and ⊖ denotes a ranking of 7 or 8. In the ranking criteria used by the CSAU panels, values from 7 to 9 were all considered to be of "high" importance. A complete summary of the rankness of the two panels is provided in Table 1 of Reference 3-2.

Reference 3-4 also considered large break LOCAs in both PWRs and BWRs. While high burned fuel was one of the main interests in the PIRT development, the thermal-hydraulic phenomena identified and ranked provide additional information on processes that a code must be assessed for. The PIRT for PWR and BWR LOCA was divided into seven subcategories: "Initial conditions," "Transient power distribution," "Steady state and transient cladding to coolant heat transfer (blowdown, refill, reflood) and core spray heat transfer," "Transient coolant conditions as a function of elevation and time," "Fuel rod response," "Multiple rod mechanical effects," and "Multiple rod thermal effects."

The categories "steady state and transient cladding to coolant heat transfer (blowdown, refill, reflood) and core spray heat transfer" and "transient coolant conditions as a function of elevation and time" dealt with thermal-hydraulic processes during LOCAs. Within the steady state and transient cladding to coolant heat transfer and core spray heat transfer subcategory, each phenomenon, with the exception of radiation heat transfer to coolant, was judged as being of high importance by the panel. Film boiling over a wide void fraction, rewet, rod-to-spacer grid thermal-hydraulic interaction, and spacer grid rewetting and droplet breakup were consider to be important processes. The panel recognized that the complex processes of dryout, film boiling and rewetting are at once of great importance in determining cladding temperature and fairly unknown because fundamental models do not exist and there is large scatter of data. The influence of the spacer grids on heat transfer and rewetting was also highlighted by the panel's importance vote. In addition, within the "Transient coolant conditions as a function of elevation and time" subcategory, temperature, flow rate and direction, quality, void fraction, and cross flow effects due to flow blockage were assigned a high ranking. The "fuel rod response" category considered phenomena associated with the fuel rods also considered by the CSAU expert panels. In Reference 3-4, within the "Fuel rod response" subcategory, burst criteria, location of burst and time-dependent gap-size heat transfer consider to have high importance.

While not the result of a PIRT process, Reference 3-5 provides a list of important phenomena for BWR LOCA. The list contains twenty-four phenomena that occur at some point during the LOCA. Table below lists the phenomena identified in Reference 3-5.

Table 3-2. BWR LOCA Phenomena

Break flow
Channel and Axial Bypass Flow and Void Distribution
Corewide Radial Void Distribution
Parallel Channel Effects - Instabilities
ECC Bypass
CCFL at UCSP and Channel Inlet Orifice
Core heat transfer incl. DNB, dryout, rewet, surface to surface radiation
Quench front propagation for both fuel rods and channel walls
Entrainment and deentrainment in core and upper plenum
Separator behavior incl. flooding, steam penetration and carryover
Spray cooling
Spray distribution
Steam dryer - hydraulic behavior
1- and 2-phase pump recirc. behavior including jet pumps

Table 3-2. BWR LOCA Phenomena

Break flow
Phase separation and mixture level behavior
Guide tube and lower plenum flashing
Natural circulation - core and downcomer
Natural circulation - core bypass, hot and cold bundles
Mixture level in core
Mixture level in downcomer
ECC mixing and condensation
Pool formation in upper plenum
Structural heat and heat losses
Phase separation in T - junction and effect on break flow

Small Break PIRT for a Westinghouse PWR

Two PIRTs have been developed for small break LOCAs in conventional PWRs. Reference 3-6 considered a Westinghouse 4-loop PWR, and Reference 3-7 considered a Babcock & Wilcox lowered loop PWR.

For the Westinghouse PWR, Table lists the processes that were assigned a "high" ranking in at least one period of the transient. These highly ranked processes were summarized by grouping them into several overall categories as follows:

Break Flow, which includes the effects of Upstream Flow Regime (i.e. vapor pull-through, etc.) in addition to the break location and geometry of break (i.e. orifice or crack type geometry).

Mixture Level, which is a process accounted for by models and correlations for vertical interfacial drag and bubble rise. Mixture level also refers to the void distribution and phase separation processes that are important in several components of the RCS.

Horizontal Flow Regime, which refers to processes that effect lateral flow. These processes are accounted for by models and input assumptions related to flow resistance in the horizontal components, flow regime predictions of horizontal stratification and CCFL, and the transition between horizontal flow regimes (especially stratified and slug).

Loop Seal Clearance refers to those processes that occur in the loop seal piping during the brief period in which steam first passes through the loop seal(s) to the broken cold leg. While brief, the period has important effect on the remainder of the transient. Highly ranked processes include several interfacial processes in the horizontal segment of the loop seal such as stratification and drop entrainment, as well as the entrainment and CCFL that may occur in the vertical pump suction pipe.

Fuel Rod Model, covers a number of processes that affect the fuel and cladding, including clad to fluid heat transfer. Clad oxidation is included in this group because of the possibility for deep core uncovery and metal-water reaction. (Preliminary expectations are that small break peak cladding temperatures will be well below the threshold for a significant metal-water reaction rate.) "Mixture Level" is included in the Fuel Rod Model category due to its effect of void fraction on local heat transfer coefficients. The Fuel Rod Model category also accounts for power related parameters such as uncertainties that affect local peaking factors on the hot rod, and variability in core-wide axial and radial power shapes.

Steam Generator Hydraulics refers to several processes, the most important of which are counter-current flow and CCFL in the steam generator tubes, condensation in the tubes during the periods when heat flows from the primary to the secondary, and the overall resistance to primary side flow through the steam generator. These processes in turn effect reflux and core depression during the loop seal clearance period.

Condensation refers to the interfacial heat and mass transfer at surfaces between subcooled liquid and saturated steam. This process is important in the cold leg and possibly the downcomer when stratification occurs. The low ranking early in the transient is due to the lack of significant liquid subcooling and stratification. Also of significant interest (but not highly ranked) is condensation to the safety injection liquid jet at the entry to the cold leg.

Table 3-3. Small Break Processes with at Least One High Ranking

Period:	**Blowdown**	**Natural Circulation**	**Loop Seal Clearance**	**Boil Off**	**Recovery**
Process					
FUEL ROD					
Oxidation				●	●
Decay Heat	●	●	●	●	●
Local Power (Local Peaking)				●	●
CORE					
3-D Power Distribution				●	●
Post-CHF Heat Transfer				●	●
Rewet / Tmin				●	●
Mixture Level			●	●	●
UPPER PLENUM					

Table 3-3. Small Break Processes with at Least One High Ranking

Period:	Blowdown	Natural Circulation	Loop Seal Clearance	Boil Off	Recovery
Process					
Hot Leg-Downcomer Gap Flow			●		
Counter-current Flow & CCFL (at Hot Leg Nozzle)		●	●		
STEAM GENERATOR					
Primary side heat transfer (U-tube condensation)	●		●		
CCFL / Tube Voiding			●		
Primary Flow Resistance (Two-Phase ΔP)			●		
PUMP SUCTION PIPING / LOOP SEAL					
Entrainment/Flow Regime/Interfacial Drag			●		
Horizontal Stratification			●		
COLD LEG					
Condensation (stratified)				●	●
Horizontal Stratification/Flow Regime			●	●	●
DOWNCOMER / LOWER PLENUM					
Mixture Level/Flashing/Void Fraction			●	●	●
BREAK					
Critical Flow In Complex Geometries	●	●	●	●	●
Upstream Flow Regime	●	●	●	●	●

Small Break PIRT for a Babcock & Wilcox PWR

Reference 3-7 reported a PIRT for small break LOCA in a Babcock and Wilcox "lowered loop" PWR. A unique feature of the B&W vessel internals is the reactor vessel vent valves (RVVVs). These valves are hinged at the top, and are held by gravity in the closed position during normal operation. The core barrel contains eight of these vent valves, which are situated around the perimeter of the core barrel in the upper part of the downcomer. During a small break LOCA, in which the break is assumed to be in the cold leg, these vent valves provide a path for steam flow from the upper plenum to the downcomer and then directly to the break.

Phenomena were ranked according to their effect of peak cladding temperature and liquid inventory in the core. The small break LOCA transient was be divided into four distinct phases; blowdown phase, natural circulation phase, loss-of natural circulation phase, and the boiler-condenser phase. The most important phenomena during the blowdown phase are the break flow and the decay heat of the core. The most important phenomena during the natural circulation phase are the natural circulation in the vessel and the steam generator, the decay heat of the core, the break flow, the phase separation in the U-bend, the ECCS flow, and the performance of the RVVVs. During the loss of natural circulation phase, the important phenomena are the performance of the RCPs, which may be run momentarily in this phase, the ECCS flow, and the break flow. Finally, during the boiler-condenser mode phase, the ECCS flow, the break flow, the heat transfer in the steam generator.

Table 3-4 lists the eight phenomena which were considered to be important in the transient.

Table 3-4. B&W Plant Small Break Phenomena

Phenomena	Component(s)
Break flow	Break
Natural circulation	Vessel and Steam Generator
Decay heat	Core
RCP Performance	RCP
ECCS flow	High pressure safety injection
Steam generator heat transfer	Steam Generator
Phase separation in U-bend	Hot leg U-bend
RVVV performance	Vessel

Related PIRTs

Three additional PIRTs that are of interest to possible future development and assessment with TRACE are documented in References 3-8 through 3-10. Reference 3-8 presents a PIRT for large break LOCA in an AP600. While the AP600 design is passively cooled, numerous phenomena important to that design are included in the PIRTs for conventional reactors. References 3-9 and 3-10 pertain to BWR core stability and rod ejection accidents respectively. These PIRTs may be useful in extensions TRACE or coupled versions of TRACE/PARCS to those types of events.

References

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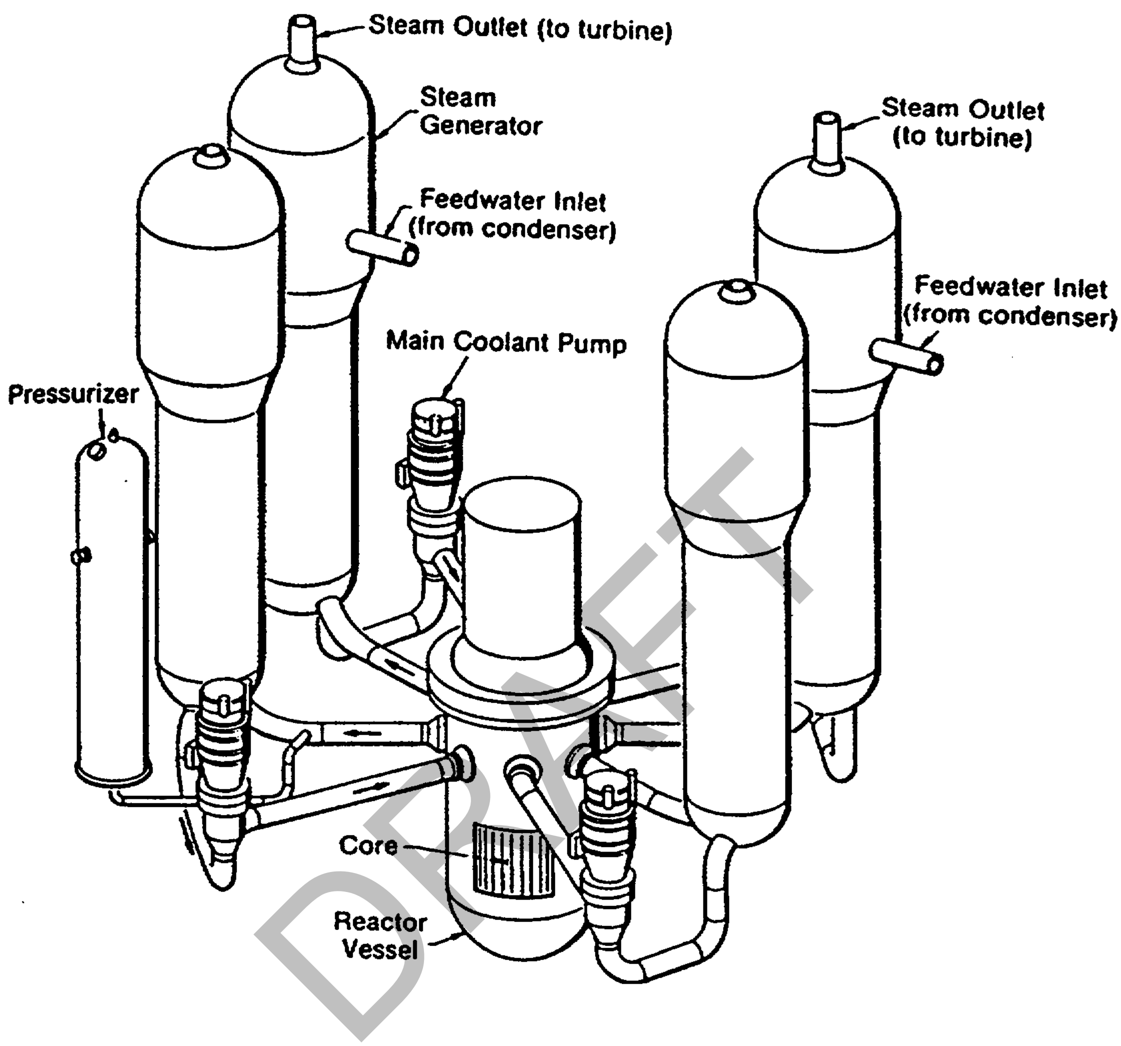

Figure. 3-1. PWR Primary Coolant System.

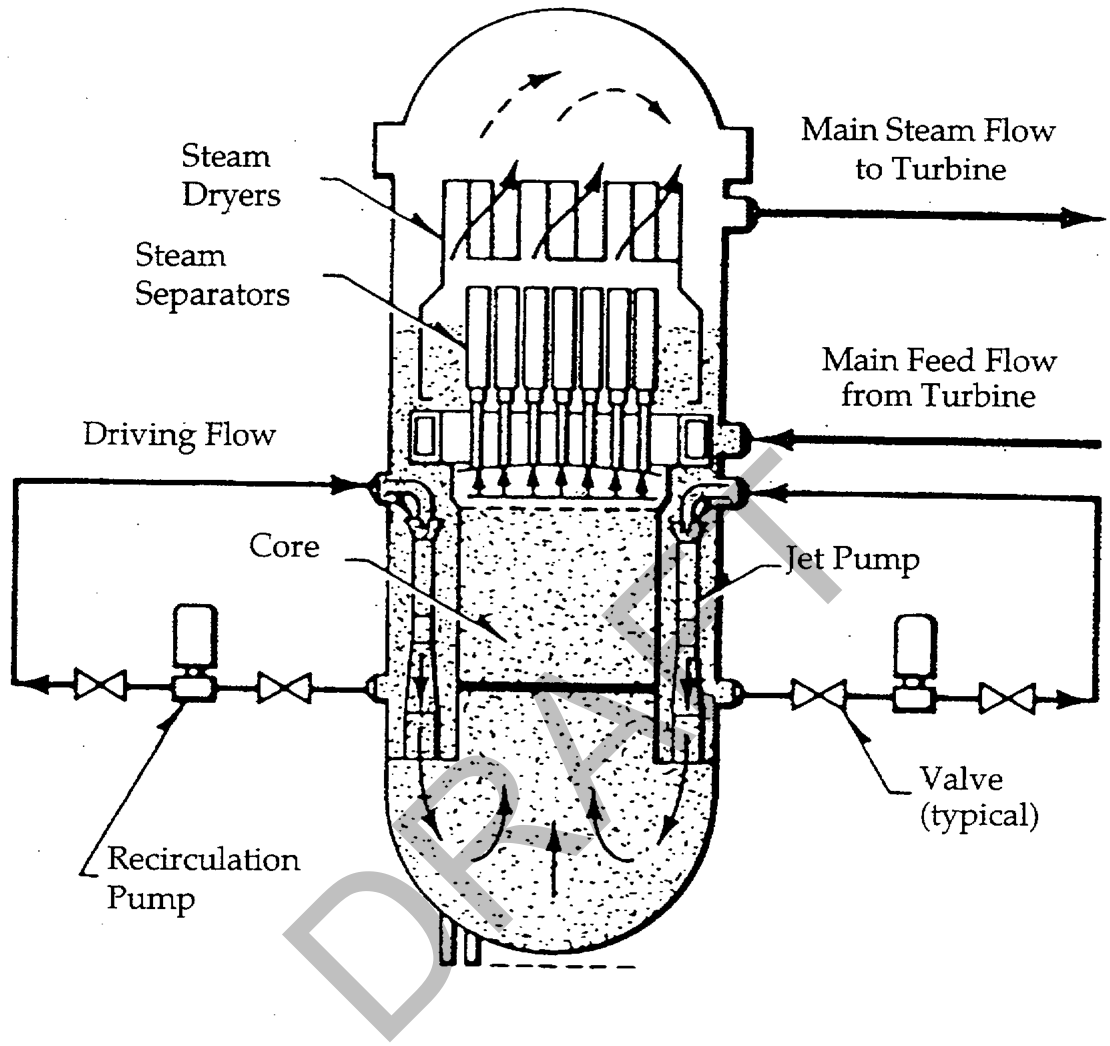

Figure. 3-2. Steam and Recirculation Water Flow Paths in a BWR.

4

TRACE Assessment Matrix

Introduction

Best estimate thermal-hydraulics codes are designed to provide realistic, rather than conservative, predictions of plant behavior for a variety of hypothesized accidents and transients. To accomplish the objective of performing realistic predictions for a broad range of plants and accident scenarios, a thermal-hydraulics code must simulate a large number of complex physical processes. Because many of the models and correlations used to represent these physical processes are not based on first principles and may not always provide an accurate prediction of a given process, these codes must be assessed against suitable experimental data. As described in Reference 4-1, the final goal of code assessment is to arrive at a qualified judgement of the accuracy with which the code, in this case TRACE, can predict accidents and transients in a full scale light water reactor.

The assessment process consists of the following tasks:

1. Establish the accuracy in predicting the best available integral system data from sub-scale test facilities and when possible, full scale light water reactors.

2. Determine whether relevant thermal-hydraulic phenomena are modeled well enough to justify application of the TRACE to light water reactor conditions for which full-scale data are unavailable. This step involves application of the code to a series of separate effects tests.

The accident scenarios for which TRACE is intended to provide reliable predictions are practically unlimited. This is particularly true for small break LOCAs and transients in which operator actions may be involved. Thus, it is not possible to obtain sufficient integral effects test results such that it would cover all the intended applications of the code. In addition, the integral effects facilities contain atypicalities and scaling distortions relative to full scale plants.

It follows then that without sufficient integral effects data to completely assess a code, performance must be tested against separate effects and other small scale test data. Separate effects data presents the opportunity to examine simulation of phenomena in detail and provides assurance that the code is not a black box tuned to a particular set of data.

Tests for assessment should also cover the modeling requirements for the intended application of the code. Modeling requirements are defined in the next section.

Modeling Requirements

Reference 4-2 considered the types of accidents in LWRs, and the modeling capabilities for realistic analysis of those accidents. The objective of realistic analysis is to a the best estimate prediction of the vital system variables such as pressure and mixture level in the reactor vessel, and the maximums for cladding temperature and local oxidation.

The best estimate code modeling requirements for initial steady-state calculations in both BWRs and PWRs are:

a. Complete geometrical simulation of the reactor system with realistic modeling of all the important flow paths, material masses, and system components.

b. Steady-state or unperturbed transient modeling of mass, momentum and energy distribution for the coolant, including flow velocities and temperature, for single-phase and two-phase flow in all reactor components.

c. Single-phase pressure drop in pipes, bends, fuel bundles, area changes, and in all special reactor components.

d. Single-phase heat convection for water and steam and boiling heat transfer.

e. Two-phase density (or void fraction) and velocity distribution in the boiling channels, including subcooled boiling voids.

f. Two-phase pressure drop in boiling channels and in other reactor components such as pipes, separators, etc.

g. Steady-state heat conduction and temperature distribution in solids.

h. Heat conduction in the gap between fuel and cladding.

i. Realistic modeling of the characteristics of specific system components such as pumps, steam separators, jet pumps, etc.

j. Reliable approximation of the thermodynamic and transport properties of the reactor materials such as fuel, cladding, vessel, piping and the coolant (liquid and vapor).

Two different scenarios of large and small breaks in a PWR and a large break LOCA in a BWR are reviewed. Typical events, physical processes and modeling requirements for many of those various events in each scenario were noted. It is then observed from the PIRT processes identified that, in spite of the very wide variation of events in different scenarios, there is only a limited number of basic computational capabilities that are needed to model the dominant effects in all scenarios.

Following are two separate listings of the identified best estimate modeling capabilities for LOCA transients in BWRs and PWRs, respectively (Ref. 4-2).

BWR Modeling Requirements

a. Time dependent distributions of mass, momentum and energy for the coolant material in system components.

b. Time dependent velocities and local flow densities in all one-dimensional and three-dimensional components including the following items:

- One-dimensional flow through fuel channels, pipes, valves, pumps, etc.
- Multidimensional flow through downcomer, lower plenum, upper plenum, bypass, and steam dome.
- Flow through jet pumps in forward and reverse directions, with proper pressure loss coefficients.
- Flow through steam separators and dryers in forward and reverse directions, with proper loss coefficients.

c. Critical flow calculation at the break and at any internal junction that may experience very steep pressure gradients.

d. Interfacial exchanges of mass, momentum and energy between vapor and liquid, including the effects of various two-phase flow patterns.

e. Circulation pump characteristics, including all four quadrants.

f. Safety and relief valve component modeling capability.

g. Transient heat conduction and temperature distribution for fuel, cladding, and other solid structures.

h. Single-phase heat transfer to vapor and liquid in different flow geometries.

i. Gap heat conductance.

j. Boiling heat transfer including nucleate, transition, and film boiling at all pressures.

k. Critical heat flux (or dryout) prediction relevant to BWR fuel geometries.

l. Nonequilibrium temperature distribution between vapor and liquid with individual heat transfer between either phase and the channel walls.

m. Liquid entrainment in vapor and de-entrainment.

n. Countercurrent flow and CCFL effects at the side entry orifice and at the upper tie plate geometries.

o. Radiation heat transfer between any fuel rod and other rods, surrounding steam and droplets, and the channel walls.

p. Minimum film boiling temperature and rewet heat transfer.

q. Power calculation with time-dependent neutron kinetics model, including at lease six groups of delayed neutrons.

r. Decay heat of fission products with contribution from transuranic elements.

s. Realistic trips with appropriate delay actions and parameter dependencies.

t. Control system models with universal simulation capabilities.

u. Containment simulation capability, including dry and wet wells, heat transfer, pool boiling, and condensation on the walls.

PWR Modeling Requirements

PWR modeling requirements are largely the same as for BWRs, as listed in the previous section. The only differences are in the areas of multidimensional flow in the core, pressurizer modeling with heating and water injection, heat exchange and boiling in steam generators, and consideration of cocurrent and countercurrent two-phase flows inside horizontal connections between the reactor vessel and steam generator.

For completeness, the entire list of modeling requirements for PWRs is given below:

a. Time-dependent distributions of mass, momentum and energy for the coolant material in all system components.

b. Time dependent velocities and local flow densities in all one-dimensional and three-dimensional components including the following items:

- One-dimensional flow through pipes, valves, pumps etc.
- Multidimensional flow through downcomer, lower plenum, the reactor core, and upper plenum.

- Flow through the primary side and secondary sides of steam generators, with proper loss coefficients for forward and reversed flow directions.

c. Critical flow calculation at the break and at any internal junction that may experience very steep pressure gradients.

d. Interfacial exchanges of mass, momentum and energy between vapor and liquid, including the effects of various two-phase flow patterns.

e. Circulation pump characteristics, including all four quadrants.

f. Safety and relief valve component modeling capability.

g. Transient heat conduction and temperature distribution for fuel, cladding, and other solid structures.

h. Single-phase heat transfer to vapor and liquid in different flow geometries.

i. Gap heat conductance.

j. Boiling heat transfer including nucleate, transition, and film boiling at all pressures.

k. Critical heat flux (or dryout) prediction relevant to PWR fuel geometry and to both sides of the steam generator tubes.

l. Nonequilibrium temperature distribution between vapor and liquid with individual heat transfer between either phase and the channel walls.

m. Liquid entrainment in vapor and de-entrainment.

n. Countercurrent flow and CCFL effects in vertical pipes, horizontal pipes, and in steam generator tubes.

o. Pressurizer simulation with liquid injection and level prediction capability.

p. Minimum film boiling temperature and rewet heat transfer.

q. Power calculation with time-dependent neutron kinetics model, including at least six groups of delayed neutrons.

r. Decay heat of fission products with contribution from transuranic elements.

s. Realistic trips with appropriate delay actions and parameter dependencies.

t. Control system models with universal simulation capabilities.

Selection of Test Data

Cases for code assessment must cover the broad ranges dictated by expected modeling requirements, and the physical phenomena that occur during the transients of interest. The assessment cases should also provide the means to examine specific model and correlation packages so that model bias and uncertainties can be developed, and include cases that approximate as best possible an actual light water reactor. This sections describes the various tests used to assess TRACE. They are categorized as fundamental test cases, and separate and integral effects tests. The selected cases for code assessment involve each of these categories. The following sections briefly describe the tests used for TRACE assessment and their contribution to satisfying the various modeling and PIRT requirements.

Fundamental Tests

The fundamental tests, sometimes referred to as "basic tests" and compliment the "model development tests" that are used to study thermal-hydraulic interactions on a very idealized level. These problems generally involve very simple geometries and in some instances have an analytical solution. While these tests by themselves do not qualify the code for a LOCA analysis, they do provide unique information on specific models and correlations and numerics used by the code. The fundamental tests sometimes help to identify discontinuities between correlation ranges and in how the code transitions from one correlation to another.

The factors influencing the selection of fundamental tests for TRACE assessment included:

- Availability of a well known, publicly accessible database. It was desirable that data and information for these fundamental tests be found in the technical literature.
- If the candidate test were part of the developmental assessment base or it had been used in development of correlations or other empiricism in the code.
- Tests utilizing fluids other than air, water or steam were excluded from the database because they would require equations of state other than those of interest in a LOCA analysis. Use of such fluids would result in additional uncertainty due to the equation of state.

The fundamental tests documented in Appendix A of this report are:

<u>Radial and Axial Heat Conduction Tests</u>: Both of these examine the solution of steady-state conduction in a geometry similar to a fuel rod. The TRACE solution to problems involving one-dimensional and two-dimensional conduction in a rod are compared to exact analytical solutions. This comparison helps to validate the TRACE conduction solution and adequacy of radial nodalization used in most rod models.

<u>Drain - Fill Problem</u>: The ability of TRACE to predict the motion of a water level can be assessed by the 1D drain and fill test problem. The purpose of this test problem is to verify that TRACE

has the ability to track liquid level across node boundaries and to determine the accuracy in calculating the gravity head as the cells slowly drain and fill.

Oscillating Manometer: The capability of TRACE to predict motion of the interface between liquid and gas is assessed by the oscillating manometer case. Of particular interest is the ability to track liquid level across node boundaries. An analytical solution for liquid motion in a frictionless U-tube manometer can be obtained from the governing equation of motion for the liquid interface motion, and this is compared to the TRACE solution.

Vertical Two-Phase Flow Tests: The objective of this test problem was to validate TRACE for predicting adiabatic two-phase upflow in a simple vertical pipe. TRACE predictions are compared to experimental data from experiments carried out in the Argonne National Laboratory (ANL) loop facility. The experiments of interest were adiabatic vertical upflow of air-water mixtures at atmospheric pressure.

TPTF Horizontal Flow Tests: The objective of this test problem is to validate TRACE for predicting horizontal two-phase flow in a relatively large-diameter pipe. Data from the Two-Phase flow Test Facility (TPTF) was used in comparisons with TRACE predictions. The TPTF separate-effect test data used in these comparisons are for horizontal co-current steam-water flow at high pressures (3.0 MPa to 8.0 MPa) in a horizontal test section of a 0.18 m (7.09 in) internal diameter.

Single and Two-Phase Wall Friction: The ability of TRACE to correctly predict the pressure drop due to wall friction in single and two-phase flows was examined. Analytical values calculated using the Churchill correlation for wall friction factor were compared against TRACE predictions for single and two-phase flow. Reynolds number versus dimensionless film thickness data from several free falling film experiments were also used to assess the wall friction model in TRACE.

Single Tube Flooding: The ability of TRACE to predict flooding and counter-current flow in vertical pipes is examined by comparing TRACE predictions of CCFL to well-known correlations. The validation provides guidance on modeling CCFL with TRACE for vertical pipes with diameters similar to those in steam generator tubes and for pipes with relatively large diameters.

CISE Adiabatic Tube: The TRACE interfacial shear model was assessed by comparing predictions versus measurements for adiabatic steam-water two-phase flows in a vertical pipe. Steady-state data from CISE Test R-291, which was previously to assess the TRAC-BD1/MOD1 code, was used to examine TRACE for flows with void fractions ranging from 0.16 to 0.94.

Separate Effects Tests

The separate effects tests are designed to produce detailed information on the behavior of individual system components or parts of the overall system. Each separate effects test is subject to an imposed set of initial and boundary conditions which are parametrically varied to cover the ranges expected during postulated accidents and transients. These types of tests feature a wide range of geometric scales, although many are full scale, and have a wide range of parametric variations.

The factors influencing the selection of separate effects tests for TRACE assessment included:

- Coverage of system components except for those such as centrifugal pumps that can be described purely empirically. Performance of those empirically based components can be determined from the integral effects tests in which they are used.
- Design: The more prototypical the design, the greater the capability to reproduce the processes in that particular component. So, test facilities that more faithfully modeled actual hardware were more likely to be used in assessment.
- Scale: While tests at varying scale are useful, if only one facility could be modeled and simulated as part of assessment, facilities closest to full scale were selected.
- Quality, quantity, and availability of the measurements. Test programs with well documented and high quality measurements were selected over some tests in which data has become difficult to obtain or for which facility descriptions are inadequate for detailed modeling.
- Diversity of the initial and boundary conditions.

The separate effect tests used for assessment of TRACE, documented in Appendix B of this report are:

<u>Marviken</u>: The calculation of critical flow is an important consideration in all of the PIRTs for large and small break LOCA. The Marviken full-scale critical-flow experiments were designed to simulate pressure-vessel blowdown. These experiments provide experimental data for subcooled and two-phase water critical-flows exiting a simulated break from a tank pressurized to about 5-MPa. Flow exited from a pipe test section attached to the bottom of the tank. The TRACE computer code was used to predict critical flow conditions for six tests at the Marviken facility.

<u>Moby Dick</u>: The Moby Dick Flow Experiments were performed at the Centre d'Etudes Nucleaires de Grenoble with the objective to study steady-state, two-phase, two-component critical flow in a vertical, divergent nozzle at low pressure. During testing, a low quality water and nitrogen mixture flowed at high velocity through a vertical test section with a divergent nozzle. Flashing was observed downstream of the nozzle, and pressures and void fractions were measured at various points along the test section. TRACE was used to simulate a total of 10 tests; 5 with critical flow and 5 with non-critical flow.

<u>Super Moby Dick</u>: The Super Moby Dick Experiments, which were also performed at the Centre d'Etudes Nucleaires de Grenoble in France, were designed to study steady-state critical flow in nozzles at medium to high pressure for various thermal-hydraulic conditions. TRACE was used to predict steady-state flow conditions for eight tests. Four of the tests simulated flow through a long divergent nozzle. The other four tests simulated flow through an abrupt expansion.

<u>THTF Blowdown Tests</u>: The Blowdown Heat Transfer (BDHT) program conducted by Oak Ridge National Laboratory (ORNL) studied heat transfer phenomena in PWRs during the

blowdown period of a loss of coolant accident. TRACE was used to simulate four steady state tests, and three transient tests as a means to validate modeling and prediction of post-CHF heat transfer, rewet and minimum film boiling temperatures.

THTF Mixture Level Swell and Core Uncovery Tests: The THTF bundle was also used to conduct a series of tests intended to produce rod bundle conditions expected during a small break LOCA. These tests provided detailed axial void fraction profiles at high pressure, and included steady-state tests with significant bundle uncovery. TRACE was used to simulate 12 of these THTF mixture level and uncovery tests to help validate models for mixture level swell, interfacial drag, and rod bundle heat transfer.

FLECHT-SEASET Forced Reflood: The FLECHT-SEASET facility used to conduct the unblocked bundle forced and gravity reflood experiments is a separate-effect test facility in which the bundle is isolated from the system and the thermal-hydraulic conditions are prescribed at the bundle entrance and exit. Within the 161 rod unblocked bundle, the dimensions are full scale compared to a PWR except for the overall radial dimension. TRACE was used to simulate eight forced reflood tests. These simulations help to validate TRACE for reflood thermal-hydraulics, including post-CHF and steam cooling heat transfer, interfacial drag and heat transfer, and entrainment and de-entrainment in rod bundles.

GOTA: A series of experiments were performed on the GOTA separate effects test facility to investigate the effectiveness of the Emergency Core Cooling System (ECCS) used in BWRs. From this series, two experiments were selected for simulation with TRACE; Run 27 to investigate radiation heat transfer in channeled rod bundles and Run 42 to examine thermal-hydraulic behavior for a combination of top spray and bottom flooding bottom-up and top-down reflood including rod quenching in channeled rod bundles.

RBHT Reflood: The RBHT facility is designed to simulate a full-length portion of a Pressurized Water Reactor (PWR) fuel assembly. The facility consists of a 7x7 rod bundle with 45 electrically heated rods and a low mass housing that allowed visualization of droplet breakup at several of the bundle’s mixing vane grids. Tests were run at constant power so that transient periods were much more prolonged than in other reflood test programs. The six reflood tests simulated by TRACE help in examination of models for reflood thermal-hydraulics, including post-CHF and steam cooling heat transfer, interfacial drag and heat transfer, and entrainment and de-entrainment in rod bundles, and spacer grid effects.

RBHT Mixture Level Tests: The RBHT bundle was also used for a series of mixture level swell and core uncovery tests to provide detailed axial void fraction profiles. Seventy-three steady-state were simulated to examine the ability of TRACE to predict axial void fraction over a range of power, subcooling, pressure, and injection flow rate. One transient test was conducted to provide data for small break boil-off and core recovery. This test was also simulated to examine TRACE for small break LOCA convective heat transfer.

RBHT Steam Cooling: The RBHT facility was also used to conduct a series of steam cooling heat transfer experiments. Seven of these steady-state tests, with bundle inlet Reynolds numbers ranging from 2000 to 20000, were simulated with TRACE. These assessments help to validate

TRACE convective heat transfer models for large break LOCA reflood thermal-hydraulics and for small break LOCA boil-off and recovery period heat transfer.

FRIGG: FRIGG is a Swedish test loop facility used to study the thermal-hydraulic performance of a simulated fuel bundle for the then-proposed Marviken boiling-water reactor (BWR). The power supply for the FRIGG loop was sufficient to power an electrically heated rod bundle at near prototypical plant conditions. The FRIGG loop was used to investigate single- and two-phase pressure drops, axial and radial void distributions, burnout in natural and forced circulation, and natural-circulation mass velocity, as well as the stability limit and some limited transient conditions. TRACE was used to simulate 29 FRIGG tests that measured axial void profiles in order to validate models for predicting mixture level swell.

GE Vessel Blowdown: General Electric (GE) performed a series of experiments to investigate thermal-hydraulic phenomena such as critical flow, void distribution, and liquid-vapor mixture swell during blowdown conditions. Two of the GE level swell tests were simulated with the TRACE.

Wilson Bubble Rise Tests: A series of high pressure steam-water tests to determine the void fractions in a bubbling two-phase mixture.tests were conducted in a 36-inch (0.91 m) diameter by 25-ft (7.62 m) tall pressure vessel located at an electrical power generating station of the then Wisconsin Electric Power Company. Data from these tests were used as the basis of the Wilson bubble rise model. Simulations of 60 tests are used in this report to assess TRACE for prediction of void fraction at high pressure conditions and validate models for interfacial drag.

Bankoff CCFL Tests: Tests were conducted by Bankoff using air/water and steam/water in countercurrent flow in a small scale test apparatus. The countercurrent flow occurred through perforated plates of various geometries. TRACE was used to simulate these tests to validate its ability to predict CCFL at an upper core plate.

UCB-Kuhn Condensation Tests: A series of condensation experiments was performed in the early 1990's at the University of California at Berkeley in support of the PCCS design of the GE Simplified Boiling Water Reactor (SBWR). The tests provided data on film condensation for the downward flow of mixtures of steam and noncondensable gases inside vertical pipes. A total of 30 tests were selected for TRACE simulation. The assessment matrix includes condensation tests in laminar film mode at different pressures and for varying non-condensable (air) qualities, and pure steam cases.

Dehbi-MIT Condensation Tests: An experimental investigation was conducted at MIT to determine the effects of noncondensable gases on steam condensation under free convection conditions. The test section used in the Dehbi-MIT condensation experiments consisted of a 3.5 m long, 3.8 cm diameter, copper cylinder located centrally inside a 4.5 m long, 0.45 m diameter stainless steel vessel. Twenty-one TRACE simulations were run to assess models used for steam condensation on external walls in the presence of a noncondensable gas.

Univ. of Wisconsin Condensation Tests: An experimental investigation to examine the effect of surface orientation and forced convection on the condensation of steam in the presence of a noncondensable gas was performed by researchers at the University of Wisconsin. Six tests were simulated with TRACE to assess condensation models under conditions of forced flow.

FLECHT-SEASET Steam Generator Tests: In this program, a series of heat transfer tests were run on a model steam generator operating under simulated LOCA conditions. The model steam generator contained 32 full-length U-tubes instrumented with thermocouples to measure secondary fluid, tube wall, and primary steam temperatures. The purpose of these tests was to measure and characterize the steam generator secondary side to primary side heat release under postulated inlet fluid conditions for a hypothetical PWR LOCA. Four tests were simulated with TRACE to examine the code's ability to predict steam binding.

Model Boiler #2 (MB-2) Tests: The MB-2 was a 0.8% power-scaled model that was designed to be geometrically and thermal-hydraulically similar to the Model F steam generator. In 1986, an MB2 test program was conducted to provide test data on PWR steam generator response to certain accident transients. TRACE was used to simulate four tests that approximated a main steam line break in order to validate models for primary to secondary heat transfer.

MIT Pressurizer Tests: A research program conducted at MIT investigated pressurizer performance. Several types of tests were performed including insurge of subcooled water into a pressurizer partially full of saturated water, insurge of subcooled water into a hot empty pressurizer, insurge of subcooled water into a partially full pressurizer followed by outsurge, and outsurge when the pressurizer is partially full. Two tests were simulated with TRACE to verify the code's ability to predict pressure response where to wall and interfacial condensation and flashing effects and thermal stratification in the water are important.

Integral Effects Tests

The integral systems tests are designed to produce as closely as possible the overall reactor coolant system thermal-hydraulic behavior under conditions simulating various postulated accidents and transients. These tests often consider the effects of parametric variations and are useful in examining the sensitivity of a code to initial and boundary conditions.

Factors that influenced the selection of integral tests for assessment were as follows:

- Coverage of accident and transient scenarios relevant to code objectives
- Facility design (PWR or BWR). Both plant types were required.
- Facility scale. Facilities with larger geometric scale are preferred. Scale considerations included not only volumetric scale but also number of active loops, core height and lateral scale, and steam generator height.
- Quality, quantity, and availability of the measurements. Test programs with well documented and high quality measurements were selected over some tests in which data has become difficult to obtain or for which facility descriptions are inadequate for detailed modeling.

The integral effect tests used for assessment of TRACE, documented in Appendix C of this report are:

Loss-of-Fluid Test (LOFT): The LOFT facility was located at the Idaho National Engineering Laboratory. This facility includes a 50-MW(t),volumetrically scaled, pressurized water reactor (PWR) system. With recognition of the differences in commercial PWR designs and inherent distortions in reduced scale systems, the design objective for the LOFT facility was to produce the significant thermal-hydraulic phenomena that would occur in commercial PWR systems in the same sequence and with approximately the same time frames and magnitudes. TRACE was used to simulate three large break LOCA tests and two small break LOCA tests run in the LOFT facility.

Upper Plenum Test Facility (UPTF): UPTF was part of the international 2D/3D Program, and was used to provide experimental data on thermal hydraulic steam/water behavior in the upper plenum, loops, and downcomer during end-of-blowdown, refill and reflood phases after LOCA. As part of the UPTF test matrix, Test 5, 6, 7 and 21 series were run as downcomer separate effects tests (SET), simulating cold leg breaks with either cold leg ECC injection (Test 5, 6, and 7) or downcomer ECC injection (Test 21).

Cylindrical Core Test Facility (CCTF): The CCTF, also part of the 2D/3D Program, was designed to simulate the thermal-hydraulic behavior of a commercial 1100 MWe PWR during the refill and reflood phases of a large-break LOCA with the break in the cold leg. Data from several tests were used to assess TRACE reflood thermal-hydraulics models. The tests simulated were C2-4 (Run 62), C2-5 (Run 63), C2-6 (Run 64), C2-8 (Run 67), C2-1 (Run 55), C2-AA2 (Run 58), and C2-12 (Run 71). These tests include variations in total power, radial power profile, system pressure, and ECC injection location.

Slab Core Test Facility (SCTF): The SCTF experimental program was operated by the Japan Atomic Energy Research Institute (JAERI) during the 1980s as part of the 2D/3D Program. The SCTF had a full-height, full-radius, one-bundle-width, electrically heated core. The core consists of eight 16 x 16 fuel bundles arranged in a row to produce a slab core configuration, each bundle containing both heated and unheated rods The objectives of the SCTF testing were to study two-dimensional hydrodynamics and heat transfer in the core and the performance of the ECCS during the end of blowdown and during the refill and reflood phases of a LOCA in a PWR. TRACE simulations of seven SCTF reflood tests, Runs 604, 605, 606, 607, 611, 621, and 622, were made to assess the reflood thermal-hydraulics models.

BETHSY: BETHSY is a scaled model of a 2775 MWt, 900 MWe (three loops, 17 x 17 fuel bundles) FRAMATOME pressurized water reactor (PWR) specifically designed for the study of PWR accident transients. The BETHSY primary system has three identical loops, each one is equipped with a main coolant pump capable of delivering up to the nominal flow rate and an active steam generator. Two small break tests were simulated with TRACE, Test 9.1b and Test 6.2TC. BETHSY Test 9.1b (International Standard Problem No. 27 or ISP-27) was a 0.5% (5.08 cm or 2-inch) cold leg break without available high pressure injection system (HPIS). BETHSY Test 6.2TC was for a 5.0% (15.24 cm or 6 inch) cold leg break without available HPIS. Both simulations contribute to assessment of TRACE for small break LOCA phenomena.

ROSA: The ROSA-IV Large Scale Test Facility (LSTF) is a large integral effects test facility used to simulate LOCA and operational transient behavior in PWRs. It is a 1/48 volumetrically scaled, full height, full pressure model of a reference 3423 MWt Westinghouse four-loop Trojan class PWR. The LSTF facility operated at the temperature and pressure of the reference PWR. It had two loops (such that the volume scaling ratio in the loop piping was 1/24), each containing all the components of the reference plant. Six small break LOCA tests were simulated with TRACE; SB-CL-01 ,SB-CL-05 , SB-CL-14 , SB-CL-15 , SB-CL-16 , and SB-CL-18. These tests covered a range of equivalent break sizes from 0.5% to 10% of the cold leg flow area, as well as breaks at the top, bottom, and middle of the cold leg.

Semiscale: The Semiscale facility, located at the Idaho National Engineering Laboratory, has been an important part of the NRC light water reactor safety program since the early 1970's. Originally intended to study large-break loss-of-coolant accidents, subsequent facility modifications have provided data for LOCAs over a broad spectrum of sizes with additional experiments conducted to investigate reflood heat transfer, steam generator tube rupture, station blackout events, natural circulation, secondary induced transients, and power loss transients and has been operated to examine proposed recovery procedures for various _ transients. Natural circulation experiments were performed in the Semiscale Mod-2A test facility. Of those tests, TRACE simulations were made for two natural circulation tests (S-NC-2 and S-NC-3). The Semiscale facility hardware configuration for the NH test series was labeled as Mod-2C. Tests S-NH-1 and S-NH-2 were simulated with TRACE. Test S-NH-1 simulated a 0.5% cold leg pipe break. Test S-NH-2 simulated a 2.1% cold leg pipe break. Both assumed with complete loss of all high pressure safety injection.

Two Loop Test Apparatus (TLTA): TLTA is a single electrically heated bundle facility, a 1/624 volume scaled version of a standard BWR/6-218 in and was a predecessor to the better scaled FIST facility. reactor. The system models major regions and components in a BWR, such as the lower plenum, core, core bypass, upper plenum, steam separator, steam dome, downcomer, jet pumps and recirculation loops. TRACE was used to simulate TLTA Test 6425 Run 2 and Test 6424 Run 1, to validate the code for BWR LOCA thermal-hydraulics. Test 6425 Run 2 was a double-ended rupture of a recirculation line at plant nominal condition, while Test 6424 Run 1 was run at a higher power. The key events and governing phenomena of interest in these tests include critical flow, counter current flow limitation (CCFL), core heatup, and ECCS injection.

Full Integral Simulation Test (FIST): The FIST facility was scaled to a GE BWR/6-218 standard plant. The FIST was full scale in height, but its fluid volumes and flow areas were scaled 1/624 which corresponds to a single bundle in FIST. The full-size bundle with electrically heated rods simulated the reactor fuel assembly. A centrifugal separator with a dryer provided steam-water separation with pressure drop characteristics of a BWR/6. The FIST facility included two recirculation loops, each driving its own jet pump housed in the downcomer pipe, external to the main vessel. Two tests were simulated with TRACE; Test 6SB2C a0.05 ft^2 recirculation line break without High Pressure Core Spray (HPCS), and Test 6SB1 which was similar to Test 6SB2C, but included a stuck open Safety Relief Valve (SRV).

Steam Sector Test Facility (SSTF): SSTF was a full-scale mock-up of a 30^o sector of a GE BWR/6-218 (624 bundles) design. It simulates major regions of the reference reactor design such as the steam dome, steam separator, reactor core, core bypass, downcomer, lower plenum, control rod guide tubes and jet pump volumes, etc. A series of tests were conducted in the SSTF to study thermal-hydraulic phenomena during the refill/reflood phase of a boiling water reactor LOCA. TRACE code calculations are compared with experimental data from SSTF Test EA 3.1 Run 111 and Test EA 3.3-1 Run 119. Test EA3.1 Run 111 was a system transient response test and was conducted to study controlling phenomena related to a BWR/4. The main purpose of Test EA3.3-1/119 was to investigate the sensitivity of the system response to the break size. The phenomena of interest in these two tests include counter current flow limit (CCFL), ECCS injection mixing in the upper plenum and lower plenum, and parallel channel phenomena,

Formulation of Assessment Matrices

For a complete and thorough assessment, TRACE must be exercised to test its ability to model different phenomena for the spectrum of transients occurring in each type of transient. Different transients within each type may occur because of different initial and boundary conditions, assumed equipment failures, or break sizes. LWR transients can be classified as large break-LOCAs, intermediate- and small-break LOCAs, and various other transients. Large-break LOCAs are characterized by strong turbulence during blowdown and reflood; for small-break LOCAs, gravity effects, stratification, and natural circulation are important phenomena to be simulated. These phenomena have been identified and ranked for importance in the PIRTs discussed earlier.

Individual experiments are selected to cover the range of important phenomena for the different types of transients. Because it is necessary to extrapolate the results of code assessment to full-size LWRs, it is desirable to select facilities at different scale to help insure modeling of some processes by the code is not scale dependent.

Assessment Matrix for Large Break LOCAs in PWRs

Table 4-1 presents a cross-reference of major phenomena for large break LOCAs in PWRs and the facilities used for TRACE assessment that address those phenomena. The phenomena in this table are those from the PIRT developed for CSAU (Ref. 4-3). Four large scale test facilities are used to address PWR large break phenomena; LOFT, CCTF, SCTF, and UPTF. Coverage of the various periods of a large break LOCA by these facilities is shown in Table 4-2.

All physical processes have at least one test in which the process was simulated or partly simulated except "oxidation" and "decay heat" as they are associated with fuel rod modeling. Both however are addressed by specific correlations in TRACE, and the uncertainty associated with those correlations can be determined from the developmental database. Therefore, specific assessment is not considered necessary.

"Stored energy" and "gap conductance" are partially simulated by LOFT. While LOFT did include a nuclear core, the fuel rods were unpressurized unlike those in commercial plants. Uncertainties in these processes however, can be addressed by comparing TRACE initial fuel temperatures to predictions by qualified fuel performance codes such as FRAPCON to insure that reasonable values are obtained in steady-state plant calculations.

Two processes related to the pump component, "two-phase performance" and "ΔP, form losses" also have relatively weak coverage in the assessment matrix. In a large break LOCA however the loops void quickly during blowdown and as a result "two-phase performance" by a reactor coolant pump can be important only over a few seconds. Following voiding of the loops, the pump acts as a resistance to steam flow and form loss can be important. Uncertainty in the pump resistance should be considered in applications with TRACE.

Assessment Matrix for Small Break LOCAs in PWRs

Table 4-3 presents a cross reference of phenomena for small break LOCA in PWRs with the assessment cases selected for TRACE. The phenomena are those identified for small break LOCA in Reference 4-4. Four integral test facilities are used for small break LOCA assessment; LOFT, Semiscale, ROSA-IV, and BETHSY. Table 4-4 compares the facilities with the various periods of a small break LOCA.

All highly ranked processes are simulated by at least one facility with the exception of fuel rod processes "oxidation", "decay heat" and local power." As discussed for large break LOCA, both oxidation and decay heat are modelled in TRACE by specific correlations and uncertainty can be determined from the developmental database. Additional validation for these processes is not considered necessary.

Assessment Matrix for LOCAs in BWRs

Table 4-5 presents a cross reference of phenomena for large and small break LOCAs in BWR with assessment cases selected for TRACE. The phenomena are those listed in Table 5-4 of Reference 4-2. Two integral test facilities meant specifically for BWRs included in the assessment matrix are FIST and TLTA. SSTF includes many features of an integral facility as well.

All processes are simulated or partially simulated by at least one test facility except for "Phase separation in T - junction and effect on break flow." This process is partly addressed by the ROSA-IV small break simulations in which the break orientation was varied in the assessments. In addition, break location, discharge coefficient, and orientation should considered in a plant analysis using TRACE in order to determine the most conservative set of conditions. Thus, additional specific assessment for this process is not consider essential.

Comparison to CSNI Assessment Matrices for Code Validation

The OECD / CSNI Principal Working Group 2 (Ref. 4-5) also developed a set of validation cases for thermal-hydraulic code development. Table 4-6 lists the phenomena and test facilities for large break LOCA, Table 4-7 and the phenomena and facilities for small break LOCA. The physical processes needed to be validated were not identified by a formal PIRT panel, but a comparison between Table 4-1 and Table 4-6 or between Table 4-3 and Table 4-7 shows very good agreement.

Table 4-1. Assessment Matrix for Large Break LOCA in PWRs

Key:

● = simulated
⊖ = partially simulated
- = not simulated or measured

	Blowdown	Refill	Reflood	LOFT	UPTF	CCTF	SCTF	Marviken	Moby Dick	Super MobyDick	THTF Blowdown	FLECHT-SEASET	RBHT Reflood	GOTA Reflood	FLECHT-SS SG Tests	MB-2 SG Tests	MIT Pressurizer Test	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	Bankoff CCFL Tests
Critical flow (break)	●	⊖	-	⊖	⊖	⊖	⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Stored energy (fuel)	●	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Oxidation (fuel)	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Decay heat (fuel)	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Gap conductance (fuel)	-	-	⊖	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Post-CHF heat transfer (core)	-	⊖	-	⊖	-	●	●	-	-	-	●	⊖	⊖	-	-	-	-	-	-	-	-
Reflood heat transfer (core)	-	-	●	⊖	-	●	●	-	-	-	-	●	●	●	-	-	-	-	-	-	-
Minimum film boiling temperature (core)	⊖	⊖	-	-	-	●	●	-	-	-	●	●	●	●	-	-	-	-	-	-	-
Three-dimensional flow (core)	-	●	●	⊖	-	●	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Voiding (core)	-	-	●	⊖	-	●	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Entrainment / De-entrainment (UP, HL)	-	-	●	-	-	⊖	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Steam binding (SG)	-	-	●	⊖	-	●	⊖	-	-	-	-	-	-	-	●	●	-	-	-	-	-
Pressurizer early quench (pressurizer)	⊖	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	⊖	-	-	-	-
Condensation (downcomer, CL)	-	●		●	●	●	⊖	-	-	-	-	-	-	-	-	-	-	⊖	⊖	⊖	-
Hot wall heat transfer (downcomer)	-	-	⊖	●	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Two-phase performance (pump)	●	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table 4-1. Assessment Matrix for Large Break LOCA in PWRs

Key: ● = simulated; ⊖ = partially simulated; - = not simulated or measured	Blowdown	Refill	Reflood		LOFT	UPTF	CCTF	SCTF	Marviken	Moby Dick	Super MobyDick	THTF Blowdown	FLECHT-SEASET	RBHT Reflood	GOTA Reflood	FLECHT-SS SG Tests	MB-2 SG Tests	MIT Pressurizer Test	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	Bankoff CCFL Tests
ΔP, form losses (pump)	-	-	⊖		⊖	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Noncondensible gas part. press.(CL,accum.)	-	-	●		-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	⊖	⊖	⊖	-
Two-phase ΔP (loop)	⊖	-	-		●	-	●	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Oscillations (loop)	-	⊖	●		●	-	●	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Flow split (loop)	-	⊖	-		●	-	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table 4-2. Integral Tests for Large Break LOCA in PWRs

Key: ● = simulated; ⊖ = partially simulated; - = not simulated or measured	Blowdown	Refill	Reflood
LOFT	●	●	●
UPTF	●	●	-
CCTF	-	●	●
SCTF	-	-	●

Table 4-3. Assessment Matrix for Small Break LOCA in PWRs

Key:
● = simulated
⊖ = partially simulated
- = not simulated or measured

Process	Blowdown	Natural Circulation	Loop Seal Clearance	Boil Off	Recovery		LOFT	ROSA-IV	BETHSY	Semiscale	SCTF	UPTF	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	Bankoff CCFL Tests	Single Tube Flooding
Oxidation (fuel rod)	-	-	-	●	●		-	-	-	-	-	-	-	-	-	-	-	-	-	-	
Decay Heat (fuel rod)	●	●	●	●	●		-	-	-	-	-	-	-	-	-	-	-	-	-	-	
Local Power (fuel rod)	-	-	-	●	●		-	-	-	-	-	-	-	-	-	-	-	-	-	-	
3-D Power Distribution (core)	-	-	-	●	●		-	⊖	⊖	-	●	-	-	-	-	-	-	-	-	-	
Post-CHF Heat Transfer (core)	-	-	-	●	●		⊖	●	●	●	●	-	⊖	-	●	-	-	-	-	-	
Rewet / Tmin (core)	-	-	-	●	●		⊖	●	●	●	●	-	-	-	●	-	-	-	-	-	
Mixture Level (core)	-	-	●	●	●		-	●	●	●	⊖	-	●	●	●	●	-	-	-	-	
Hot Leg-Downcomer Gap Flow (UP)	-	-	●	-	-		-	●	●	●	-	-	-	-	-	-	-	-	-	-	
Counter-current Flow & CCFL (UP /Hot Leg Nozzle)		●	●	-	-		-	●	●	●	-	-	-	-	-	-	-	-	-	●	
Primary side heat transfer / U-tube condensation (SG)	●	-	●	-	-		●	●	●	●	-	-	-	-	-	-	●	●	●	-	-
CCFL / Tube Voiding (SG)	-	-	●	-	-		⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	●
Primary Flow Resistance / Two-Phase ΔP (SG)	-	-	●	-	-		●	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Entrainment/Flow Regime/Interfacial Drag (loop seal)	-	-	●	-	-		-	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Horizontal Stratification (loop seal)	-	-	●	-	-		-	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Condensation to stratified layer (cold leg)	-	-	-	●	●		-	●	●	●	-	⊖	-	-	-	-	-	-	-	-	-
Horizontal Stratification/Flow Regime (cold leg)			●	●	●		-	●	●	●	-	⊖	-	-	-	-	-	-	-	-	-

Table 4-3. Assessment Matrix for Small Break LOCA in PWRs

Key: ● = simulated ⊖ = partially simulated - = not simulated or measured	**Blowdown**	**Natural Circulation**	**Loop Seal Clearance**	**Boil Off**	**Recovery**		LOFT	ROSA-IV	BETHSY	Semiscale	SCTF	UPTF	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	Bankoff CCFL Tests	Single Tube Flooding
Process																					
Mixture Level/Flashing/Void Fraction (downcomer /LP)			●	●	●		●	●	●	-	●	●	-	-	-	-	-	-	-	-	-
Critical Flow In Complex Geometries (break)	●	●	●	●	●		-	●	●	-	-	-	-	-	-	-	-	-	-	-	-
Upstream Flow Regime (break)	●	●	●	●	●		●	●	●	●	-	-	-	-	-	-	-	-	-	-	-

Table 4-4. Integral Tests for Small Break LOCA in PWRs

	Blowdown	Natural Circulation	Loop Seal Clearance	Core Boiloff	Core Recovery
SEMISCALE	●	●	●	●	●
BETHSY	●	●	●	●	●
ROSA-IV	-	●	●	●	●
LOFT	●	⊖	-	-	-

Table 4-5. Assessment Matrix for LOCA in BWRs

Key:

● = simulated
⊖ = partially simulated
- = not simulated or measured

	FIST	SSTF	TLTA	UPTF	Marviken	GE Vessel Blowdown	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	GOTA Reflood	GOTA Radiation	FLECHT-SEASET Reflood	RBHT Reflood	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	SCTF	CCTF
Break flow	⊖	-	⊖		●	-	-	-	-	-	-		-	-	-	-	-	-	-
Channel and Axial Bypass Flow and Void Distribution	●	-	●	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Corewide Radial Void Distribution	⊖	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	⊖	-
Parallel Channel Effects - Instabilities	-	●	-	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
ECC Bypass	⊖	●	⊖	●	-	-	-	-	-	-	-		-	-	-	-	-	-	-
CCFL at UCSP and Channel Inlet Orifice	●	●	●	-	-	-	●	●	-	⊖	-		-	-	-	-	-	-	-
Core heat transfer incl. DNB, dryout, rewet, surf.to surf. rad.	●	-	●	-	-	-	-	-	-	●	●	●	●	●	-	-	-	●	●
Quench front propagation for both fuel rods and channel walls	●	-	●	-	-	-	-	-	-	●	-	●	●	●	-	-	-	●	●
Entrainment and deentrainment in core and upper plenum	⊖	⊖	⊖	-	-	-	-	-	-	●	-	●	●	●	-	-	-	●	●
Separator behavior incl. flooding, steam penetration and carryover	●	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Spray cooling	⊖	-	⊖	-	-	-	-	-	-	●	-		-	-	-	-	-	-	-
Spray distribution	-	●	-	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-

Table 4-5. Assessment Matrix for LOCA in BWRs

Key:

● = simulated
⊖ = partially simulated
\- = not simulated or measured

	FIST	SSTF	TLTA	UPTF	Marviken	GE Vessel Blowdown	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	GOTA Reflood	GOTA Radiation	FLECHT-SEASET Reflood	RBHT Reflood	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	SCTF	CCTF
Steam dryer - hydraulic behavior	⊖	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
1- and 2-phase pump recirc. behavior including jet pumps	⊖	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Phase separation and mixture level behavior	●	-	⊖	-	⊖	●	●	●	●	-	-		-	-	-	-	-	-	-
Guide tube and lower plenum flashing	●	-	●	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Natural circulation - core and downcomer	●	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Natural circulation - core bypass, hot and cold bundles	⊖	-	⊖	-	-	-	-	-	-	-	-		-	-	-	-	-	-	-
Mixture level in core	●	-	●	-	-	-	●	●	●	-	-	-	-	-	-	-	-	-	-
Mixture level in downcomer	●	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
ECC mixing and condensation	⊖	⊖	⊖	●	-	-	-	-	-	-	-	-	-	-	●	●	●	-	-
Pool formation in upper plenum	⊖	●	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	●	-
Structural heat and heat losses	⊖	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Phase separation in T - junction and effect on break flow	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table 4-6. Cross Reference Matrix for Large Break LOCA in PWRs

Key:

● = simulated
⊖ = partially simulated
- = not simulated or measured

	Blowdown	Refill	Reflood	LOFT	UPTF	CCTF	SCTF	Marviken	Moby Dick	Super MobyDick	THTF Blowdown	FLECHT-SEASET	RBHT Reflood	GOTA Reflood	FLECHT-SS SG Tests	MB-2 SG Tests	MIT Pressurizer Test	Bankoff CCFL Tests
Break Flow	●	●	●	⊖	⊖	-	-	●	●	●	-	-	-	-	-	-	-	-
Phase Separation	⊖	●	●	-	-	●	●	-	-	-	-	●	●	●	-	-	-	-
Mixing and condensation during injection	-	●	●	●	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Core wide void and flow distribution	⊖	●	●	⊖	-	●	●	-	-	-	-	-	-	-	-	-	-	-
ECC bypass and penetration	⊖	●	⊖	⊖	●	-	-	-	-	-	-	-	-	-	-	-	-	-
CCFL (UCSP)	⊖	●	●	⊖	-	⊖	●	-	-	-	-	-	-	-	-	-	-	●
Steam binding	-	⊖	●	⊖	-	⊖	-	-	-	-	-	-	-	-	●	●	-	-
Pool formation in UP	-	●	●	⊖	-	⊖	●	-	-	-	-	-	-	-	-	-	-	-
Core heat transfer incl. DNB, dryout, rewet	●	●	●	●	-	●	●	-	-	-	●	●	●	●	-	-	-	-
Quench front propagation	⊖	⊖	●	●	-	●	●	-	-	-	-	●	●	●	-	-	-	-
Entrainment (Core, UP)	⊖	⊖	●	⊖	-	●	●	-	-	-	-	●	●	●	-	-	-	-
Deentrainment (Core, UP)	⊖	⊖	●	⊖	-	●	●	-	-	-	-	-	-	-	-	-	-	-
1- and 2- phase pump behavior	●	⊖	⊖	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table 4-7. Cross Reference Matrix for Small Break LOCA in PWRs

Key: ● = simulated; ⊖ = partially simulated; - = not simulated or measured	Blowdown	Natural Circulation	Loop Seal Clearance	Core Boiloff	Core Recovery	LOFT	ROSA-IV	BETHSY	Semiscale	SCTF	UPTF	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	MIT Pressurizer Test	Bankoff CCFL Tests
Natural circulation in 1-phase flow, primary side	-	●	-	-	-	⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Natural circulation in 2-phase flow, primary side	-	●	⊖	-	-	⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Reflux condensation mode and CCFL	-	●	⊖	⊖	-	⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Asymetric loop behavior	⊖	●	●	●	●	⊖	⊖	●	⊖	-	-	-	-	-	-	-	-	-	-	-
Leak flow	●	●	●	●	●	●	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Phase separation without mixture level formation	●	●	-	-	-	⊖	-	-	⊖	-	-	-	-	-	-	-	-	-	-	-
Mixture level and entrainment in vert. comp. (SG)	⊖	●	●	-	-	⊖	●	●	●	-	-	-	-	-	-	-	-	-	-	-
Mixture level and entrainment in core	-	-	●	●	●	-	●	●	●	●	-	●	-	●	-	-	-	-	-	-
Stratification in horizontal pipes	-	●	●	●	●	●	●	●	⊖	-	-	-	-	-	-	-	-	-	-	-
ECC mixing and condensation	-	⊖	●	●	●	⊖	●	●	●	-	●	-	-	-	-	●	●	●	-	-
Loop seal clearance	-	-	●	-	-	-	●	●	●	-	-	-	-	-	-	-	-	-	-	-

Table 4-7. Cross Reference Matrix for Small Break LOCA in PWRs

Key:

● = simulated
⊖ = partially simulated
\- = not simulated or measured

	Blowdown	Natural Circulation	Loop Seal Clearance	Core Boiloff	Core Recovery	LOFT	ROSA-IV	BETHSY	Semiscale	SCTF	UPTF	THTF Mix Level and Uncovery	FRIGG	RBHT Mixture Level	RBHT Steam Cooling	UCB Condensation	Dehbi Condensation	U. Wisc. Condensation	MIT Pressurizer Test	Bankoff CCFL Tests
Pool formation in UP/CCFL (UCSP)	-	●	●	-	-	-	●	●	●	●	-	-	-	-	-	-	-	-	-	●
Corewide void and flow distribution	⊖	⊖	●	●	●	⊖	●	●	●	●	-	-	-	-	-	-	-	-	-	-
Heat transfer in covered core	●	●	-	-	-	●	●	●	●	-	-	●	●	●	-	-	-	-	-	-
Heat transfer in partially covered core	-	-	●	●	●	●	●	●	⊖	●	-	●	-	-	●	-	-	-	-	-
Heat transfer in SG primary side	●	●	●	●	●	●	●	●	⊖	-	-	-	-	-	-	-	-	-	-	-
Heat transfer in SG secondary side	●	●	●	●	●	●	●	●	⊖	-	-	-	-	-	-	-	-	-	-	-
Pressurizer thermodynamics	●	⊖	-	-	-	●	●	●	-	-	-	-	-	-	-	-	-	-	●	
Surgeline hydraulics	●	⊖	-	-	-	●	●	●	●	-	-	-	-	-	-	-	-	-	-	-
1- and 2-phase pump behavior	●	⊖	-	-	-	⊖	-	-	●	-	-	-	-	-	-	-	-	-	-	-
Structural heat and heat losses	●	●	⊖	-	-	⊖	⊖	⊖	⊖	-	-	-	-	-	-	-	-	-	-	-
Noncondensible gas effects	-	-	⊖	●	●	-	-	-	-	-	-	-	-	-	-	●	●	●	-	-
Phase separation in T-junction and effect on leakflow	●	●	●	●	●	-	●	-	-	-	-	-	-	-	-	-	-	-	-	-

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5

Summary of Code Results and Performance

The appendices to this report provide a detailed presentation of TRACE calculations and comparisons to experimental data. Each appendix section provides a description of the test facility and TRACE model for the facility along with predictions of the experimental data. This section is intended to provide a summary of TRACE performance for important LOCA processes. Previous sections have discussed LOCA phenomena and the PIRTs that rank those physical processes. Here, assessments from several applicable tests are grouped in order to help identify overall code performance and determine deficiencies.

Approach to Assessment

Independent Assessment of a "Frozen" Code Version

The assessment process used in the validation of TRACE Version 5.0 has been that of Independent Assessment. That is, the assessment has largely been performed by individuals who were not involved in the code development. Model and correlation development and programming of those models were performed by one set of individuals, and assessment by another set. As TRACE was initially being developed, a large number of code versions were tested. Problems observed with those versions were reported to the code development group and corrections were implemented. During the later stages of development, the Independent Assessment process was applied. The code was "frozen" and subjected to calculations against the assessment matrix. If updates were necessary to obtain reasonable agreement with a case or to enable a case to execute without aborting, a new code version was generated, frozen, released to the assessment team and the assessment repeated. During these "freeze" and correct cycles, close coordination is maintained between the code development and code assessment groups. The TRACE Version 5.0 version was "frozen" in December 2006 and all 500+ assessment simulations re-run. No updates were implemented during the assessment reported here. Thus, this report presents the Independent Assessment of the strictly "frozen" TRACE Version 5.0.

Release of future TRACE versions are expected to follow this Independent assessment process, and re-calculation of all cases in the assessment matrix. As new capabilities are added to TRACE, the assessment matrix is expected to increase in size. However, public release of new TRACE versions will follow re-calculation of all cases in the assessment matrix and revision to the supporting documentation.

Code User Effects on Code Results

In the past, the code users often had to make their own selections from among numerous "user options," without proper guidance provided in the code documentation. During TRACE 5.0 assessment emphasis was placed on uniformity and consistency in nodalization and in selection of these user options. Thus, while TRACE has retained a flexible and general input structure, a set of modeling guidelines were applied to the assessment cases in this report in order to minimize the user effect. For example, facilities with regions that represent a core have common nodalizations with the height of axial cells determined by the distance between spacer grids. The same nodalization has used in separate effects tests like FLECHT-SEASET and integral tests such as CCTF and SCTF for validation of reflood heat transfer as well as for separate effects tests such as THTF and ROSA-IV which are important for validation of mixture level swell and other small break processes. User options that select particular model and correlations were also specified so that the assessment group used them consistently. Details on input specifications and common guidelines are described in the TRACE User’s Guide. Users should consult this in setting up plant models or before making modifications to any assessment simulation.

Selection of Key Parameters

Key assessment parameters, also referred to as "figures of merit" are used to define the code's accuracy. The regulatory criteria specified in 10 CFR 50.46 are mainly concerned with integrity of the fuel cladding. Thus, thermal-hydraulic processes that contribute to cladding temperature and its physical condition determines the variables requiring quantification of the code accuracy. In a small-break LOCA with no core heatup, the primary key parameter is the core inventory. Table 5-1 and Table 5-2 summarize the key parameters used in assessing TRACE for different transients for PWRs and BWRs.

Table 5-1. Key Assessment Parameters (Large Break Processes)

Continuous-Valued Key Parameters
Cladding temperature
System pressure
Break flow rate
Break flow (integrated)

Table 5-1. Key Assessment Parameters (Large Break Processes)

Continuous-Valued Key Parameters
ECC injection flow rate
Single-Valued Key Parameters
Blowdown peak cladding temperature
Reflood peak cladding temperature
Times of blowdown and reflood PCT
Time of ECC injection initiation
Time of signal generation for MSIV closure (BWR only)
Time of rewet or quench

Table 5-2. Key Assessment Parameters (Small Break Processes)

Continuous-Valued Key Parameters
Vessel or core collapsed liquid level
Cladding temperature
System pressure
Secondary side pressure (PWRs only)
Break flow rate
Break flow (integrated)
Secondary side valve flow (if any for PWRs only)
ECC injection flow rate
Single-Valued Key Parameters
Blowdown or loop seal clearance period peak cladding temperature
Boiloff / Core recovery period peak cladding temperature
Times of PCT(s)
Minimum core liquid level
Time of accumulator injection (PWRs only)
Time of HPSI and/or LPSI initiation

Table 5-2. Key Assessment Parameters (Small Break Processes)

Time of rewet or quench
Time of loop seal clearance (PWRs only)
Time of reverse primary to secondary heat transfer initiation (PWRs only)
Times of core uncovery and uncovery

Characterization of Assessment Results

Application of TRACE to evaluate plant transients requires a an estimate of the code accuracy, that is, how good are the calculated results and what is the uncertainty in the code predictions. The primary goal of the assessment simulations and comparisons to data are to provide a code accuracy statement. Two approaches are used in this report. The preferred manner is quantitative, in which the accuracy in prediction of a particular parameter is expressed in terms of a bias and an uncertainty. When possible, a bias and uncertainty is determined for the several relevant parameters associated with a process. This helps to identify if there are compensating errors in the models used to predict that process.

The second manner of characterizing code accuracy is qualitative. While this method is subjective, the following Table has been used to characterize those comparisons between predictions and measured results where a quantifiable estimate is not practical.

Table 5-3. Subjective Descriptors for Characterizing Code Accuracy

"Excellent" or "Outstanding" Agreement:	No code or model deficiencies
	1. Major and minor trends predicted correctly.
	2. Calculation usually within measurement uncertainty bounds.
	3. TRACE can be used for similar transients.
"Moderate" or "Reasonable" Agreement	Minor code or model deficiencies
	1. Major trends predicted correctly
	2. Calculation frequently outside measurement uncertainty bounds.
	3. Correct conclusions can be obtained if TRACE is used for similar transients.
"Minimal" or "Poor" Agreement	Significant code or model deficiencies
	1. Some major trends incorrectly predicted.
	2. Some predicted values well outside measurement uncertainty bounds.
	3. Incorrect conclusions may be reached if deficiencies are not considered.

Table 5-3. Subjective Descriptors for Characterizing Code Accuracy

"Insufficient" or "Unacceptable" Agreement	Major code or model deficiencies
	1. Major trends not predicted.
	2. Most predicted values well outside measurement uncertainty bounds.
	3. Conclusions should not be based on code predictions.

Summary of Results for Highly Ranked Processes

In this section results obtained for several processes are summarized.

Critical Break Flow

The critical break flow modeling with TRACE is validated using assessment from separate effects tests Marviken, Moby Dick, and Super Moby Dick, with additional supporting information from the integral effects tests. Results from Marviken are most important, as they provide the most comprehensive set of measurements. In the six tests simulated. TRACE provided reasonable agreement with most measurements. In only one case TRACE was found to provide poor agreement with break flow rate measurements, underpredicting the flow rate in the critical two-phase discharge period. The break flow results have not yet been characterized by a bias and uncertainty.

ECC Bypass

The ability to simulate ECC bypass, an important large break process, was validated by assessment using the UPTF tests. Because this facility is essentially full-scale, subscale tests were not simulated. For UPTF, TRACE was found to provide reasonable to excellent agreement with test results for tests in which the fluids were near saturation. Results were minimal (poor) however if the ECC water was highly subcooled. Key parameters for ECC bypass include the lower plenum liquid penetration rate, which was predicted to within 20% of the test data for most tests. Condensation efficiency was usually predicted to within 25%. The results also tended to be consistent with other findings in the tests. Wall temperature measurements for example indicated that penetration was more likely at the intact loops opposite the broken cold leg. TRACE predictions of ECC bypass were consistent with this.

Blowdown Heat Transfer

TRACE was found to provide reasonable agreement with blowdown heat transfer thermal-hydraulics, based on predictions of steady-state and transient THTF Blowdown Test data. There was moderate agreement with cladding temperatures (provided the DNB location was correct), but there was significant scatter in predicting heat transfer coefficients in this period. For the three steady-state blowdown tests, TRACE overpredicted the heat transfer coefficients by approximately 23% with an uncertainty of 26% at elevations well above the quench point for each test. In the transient blowdown test simulations, TRACE tended to underpredict the mean cladding temperatures at a particular elevation, and underpredicted the peak cladding temperatures by approximately 40 K. The overprediction of the heat transfer coefficients in the steady-state tests is consistent with the underprediction of cladding temperatures in the transient tests. Assessments were performed using both the VESSEL and CHAN Components to model the THTF tests, with similar performance obtained for each Component type.

Reflood Heat Transfer

Reflood period thermal-hydraulics was investigated by several assessment cases. These included the FLECHT-SEASET reflood test, GOTA reflood and radiation tests, and RBHT reflood and steam cooling heat transfer tests. TRACE was found to have reasonable to excellent agreement in comparisons to FLECHT-SEASET forced reflood results. Comparison to FLECHT-SEASET measurements showed moderate agreement with cladding temperatures and quench times. The peak cladding temperatures and quench times at each elevation were generally predicted within 10% of the measured quantity. Based on measurements at five elevations in the bundle, heat transfer coefficients were overpredicted by approximately 12% with an uncertainty of 27%.

GOTA Reflood Test 42 examined reflood in a BWR bundle with both top-down and bottom-up quench. TRACE provided reasonable predictions of the cladding temperatures and quench times of the average rods at a particular bundle elevation. There is a wide spread however in cladding temperatures at an elevation however such that TRACE underpredicted the peak cladding temperatures. Agreement with GOTA Radiation Test 27 results were reasonable to excellent, depending on the number of rod groups used in the model of the bundle. TRACE slightly underpredicted the cladding temperatures when 5 rod groups were used, but agreement improved if the number of groups were increased to at least 10.

Only minimal agreement was obtained with TRACE simulations of RBHT reflood data. In general, TRACE over predicts peak rod temperatures at higher bundle elevations. TRACE has a tendency to under predict quench times for low power and low flow cases and over predict quench times for high power and high flow cases. TRACE was found to have the difficulty with low subcooling cases, with the poorest predictions occurring for those cases involving were seen in low power, low flow and low subcooling. Code performance improved with higher power and flow. These apparent problems in prediction of the RBHT reflood results are under investigation.

TRACE was found to provide reasonable agreement with cladding temperatures in the RBHT steam cooling tests, but minimal (poor) agreement with local values of heat transfer coefficients which is attributed to the lack of spacer grid models in TRACE. For steam cooling, TRACE was found to underpredict heat transfer coefficients by 18% over a broad range of Re.

Mixture Level Swell

Validation of TRACE models for interfacial drag, wall drag, bubble rise, and flow pattern transition are accomplished through assessments against FRIGG, THTF Core Uncovery and Level Swell, RBHT Mixture Level, the GE Level Swell, and Wilson Bubble Rise test cases. In general, agreement between TRACE predictions and measurements is moderate to excellent.

The FRIGG tests provided measurements of axial void fraction profiles at high pressure for a broad range of inlet flow rate and subcooling, typical of a BWR bundle. The performance of TRACE was characterized by comparison of the predicted and measured local void fractions. On average, the predicted and measured local void fractions agreed to within 0.5% with a 2.5% uncertainty for both the VESSEL and CHAN Components. (The CHAN Component is used in BWR analysis.) There is good agreement in prediction of axial void fraction profiles, and good agreement with collapsed liquid levels.

The THTF Code Uncovery and Level Swell Test comparisons were characterized by TRACE predictions of level swell, defined as $(Z_{2\phi}-Z_{CLL})/Z_{CLL}$ and the collapsed liquid level, Z_{CLL}. For the VESSEL Component, TRACE overpredicted the measured level swell by 7.25% with an uncertainty of 27%. The CHAN Component, which uses slightly different models for wall and interfacial drag overpredicted by 20% with a 24% uncertainty. The collapsed liquid levels were underpredicted by 9% and 10% for the VESSEL and CHAN Components respectively.

For the RBHT Level Swell Tests, code performance was characterized by comparisons of local void fraction, bundle exit void fraction, collapsed liquid level and onset of significant void. In these assessments TRACE was found to overpredict the local void fraction by 19.6%, and overpredict the exit void fraction by 12.5%. The collapsed liquid level was underpredicted by 11.5% and the onset of significant void was overpredicted by 32.3%. In general, the comparisons were reasonable to excellent, with most of the deviation driven by cases with high power and high subcooling. The onset of significant void may be large due to uncertainty in the measurements.

The so called Wilson Bubble Rise tests provided data over a wide range of pressure on void fraction for air-water and steam-water two phase mixtures. While a large diameter pipe rather than a rod bundle was used as the test apparatus, the data provides unique information on two-phase flow. TRACE assessment against these tests include data for pressures from 4.24 to 13.9 MPa (815 to 2015 psia). In these assessments, TRACE on-average overpredicted the void fraction by 10.1% with an uncertainty of 3.45%. The code to data comparisons did not indicate any sensitivity to pressure. The comparison is considered reasonable to excellent, as TRACE correctly predicted trends observed in the data and the code to data agreement was well within the experimental measurement uncertainty.

CCFL and Flooding

Validation of TRACE for flooding and countercurrent flow was accomplished using assessments against the Bankoff flooding data and Wallis type correlations for single tube flooding. Agreement with flooding data by TRACE is considered to be reasonable to excellent. For steam-water mixtures TRACE slightly underpredicted liquid flow, but agreement was within the 7% measurement uncertainty. The assessment also showed that TRACE can predict the air/water countercurrent flow data for the 15-hole plate over a significant range of air flows. The predictions are better when saturated steam is used as the gas instead of air.

Condensation

Three sets of separate effects test data were used to validate TRACE for condensation processes. These were the Dehbi-MIT condensation data, the University of Wisconsin data, and the UCB-Kuhn condensation data. The UCB-Kuhn tests provided film condensation heat transfer coefficients for the downward flow of mixtures of steam and noncondensable gases inside a vertical tube. Comparison of TRACE predicted and measured heat transfer coefficient showed that the TRACE predictions were in reasonable agreement with the test data. Agreement was slightly better for laminar films. Nearly all data was within the experimental uncertainty.

The Dehbi condensation tests examined condensation in the presence of a non-condensable gas in relatively stagnant conditions. The tests considered air mass fractions from 0.3 to 0.9, and pressures from 1.5 to 4.5 atmospheres (0.152 to 0.456 MPa). The comparison of TRACE predicted to measured condensation heat transfer coefficients showed excellent agreement. Nearly all points were well within the experimental measurement uncertainty of 15%, and TRACE was found to accurately predicted the trends with air mass fraction and pressure.

The University of Wisconsin condensation tests were simulated with TRACE in order to assess performance of condensation models in TRACE for conditions that may occur on containment walls during a LOCA. Condensation in the presence of non-condensable gas would occur on exterior cold surfaces. TRACE simulations of these tests were used to compare predicted and measured condensation heat transfer coefficients. Comparison of predicted to measured heat transfer coefficients showed excellent agreement for low steam velocities, but poor agreement at high velocities. At high steam velocities (3 m/sec) TRACE was found to significantly underpredict condensation heat transfer.

Steam Generator Hydraulics

Steam binding is an important process that occurs during the reflood period of a PWR LOCA. This phenomenon is characterized by a pressure increase in the upper plenum of the reactor, and is caused by the evaporation of liquid that has been entrained in the reactor vessel and carried over into the steam generator. Four FLECHT-SEASET Steam Generator Separate Effects Tests (FLECHT-SET Phase B) were simulated with TRACE to examine the code' ability to predict steam binding. The purpose of the FLECHT-SET Phase B tests was to measure and characterize the steam generator secondary side to primary side heat release under postulated inlet fluid conditions for a hypothetical PWR LOCA. TRACE predictions of the primary to secondary heat transfer were compared to measurements were reasonable, showing that TRACE, on-average, overpredicted the total heat transfer by 6.4%. TRACE tended to evaporate all of the incoming entrained liquid, while test data showed some portion of the liquid could exit the steam generator tubes.

TRACE was also used to simulate Model Boiler 2 (MB-2) Test 2013, which examined the behavior of the steam generator response to a main steam line break while at 100% power. In this particular transient, test simulated a steam line break with subsequent loss of feedwater flow causing a complete loss of secondary side inventory. Comparison of TRACE predicted and measured results showed that TRACE overpredicted the overall primary to secondary heat transfer.

Summary of Results for Integral Tests

This section summarizes results from assessment against integral effects tests.

PWR Large Break Integral Test Facilities

The large scale facilities used to assess TRACE for large break processes included LOFT, CCTF, and SCTF. The purpose of assessment against these cases is to demonstrate TRACE's capability to simulate system-wide behavior, and physical processes that occur during large break LOCA transients.

LOFT

The LOFT facility is unique in that it is the only integral facility with nuclear rods and can simulate an entire large break transient from blowdown through reflood. LOFT Tests L2-5, L2-6, and LB-1 transient results have been assessed using TRACE and results were compared to measured data. Prediction of thermal-hydraulic parameters in each of these tests were reasonable based on comparisons to the major parameters of interest. TRACE predictions of fuel rod cladding temperatures were not as well predicted however. Insufficient liquid entrainment to the upper regions of the core resulted in overprediction of the rod clad temperature. An overprediction, and high uncertainty in prediction of the LOFT cladding temperatures is expected because of the exterior mounting of the thermocouples in the LOFT core.

CCTF

The Cylindrical Core Test Facility (CCTF) was an integral test facility used to conduct several experiments to examine core thermal hydraulics and ECC behavior during the refill and reflood periods. Data from CCTF provided information on multidimensional core thermal hydrodynamics, the effect of radial power distribution, flow behavior in the upper plenum and hot legs, the behavior of deentrained water in the upper plenum, steam binding, and the behavior of water in the downcomer. Data from several tests conducted at the test facility were used to assess TRACE. The tests simulated included C2-4 (Run 62), C2-5 (Run 63), C2-6 (Run 64), C2-8 (Run 67), C2-1 (Run 55), and C2-12 (Run 71). These tests include variations in total power, radial power profile, and system pressure.

In general, comparisons of TRACE to experimental data were found to be reasonable. TRACE was found capable of predicting the trends due to variations in initial conditions. TRACE also demonstrated the ability to calculate reflood behavior for variations in core power, radial power profile, and system pressures. The rod cladding temperatures in the upper half of the core however, were significantly over-predicted resulting in a significant overprediction of core quench compared to the test data. Typically, cladding temperatures near the top of the bundle can be overpredicted by nearly 300 K and quench can be predicted to occur nearly 300 seconds later than was determined by the data. Prediction of cladding temperatures at the peak power elevation were reasonable. The reasons for the overprediction of the cladding temperatures at the upper elevations in CCTF are considered due to excessive entrainment at the quench front but insufficient interfacial heat transfer that would decrease predicted steam temperatures. At a result, quench front propagation was delayed and the overprediction of steam temperatures undercooled the upper elevations producing excessively high cladding temperatures in the TRACE predictions. The lack of spacer grid models to account for droplet breakup and local convective enhancement also contribute to the deficiencies at upper elevations. TRACE overpredicted the peak cladding temperatures in these tests, and predicts the PCT elevation to be higher in the bundle than in the tests.

SCTF

The Slab Core Test Facility (SCTF) was designed to study two-dimensional heat transfer and hydrodynamics in a reactor core during the reflood phase of a postulated LOCA. A unique feature of the facility was that eight rod bundles were arranged to model a "slice" of a core. TRACE was used to simulate seven of the SCTF tests; Runs 604, 605, 606, 607, 611, 621, and 622. Runs 604, 605, 606, 607 and 611 were gravity reflood tests, while Runs 621 and 622 were forced reflood tests. In the forced reflood tests, the vessel downcomer bottom was blocked off from the lower plenum.

The TRACE simulations of SCTF are in reasonable agreement with the experimental data overall and in quite good agreement with measured cladding temperature at the core midplane. Calculated cladding temperatures at the upper core elevations however, exceeded the measured values by between 50 and 200 K. It is believed that this over prediction of upper elevation temperatures is due to same reasons as in the CCTF simulations. That is, TRACE tends to overpredict entrainment at the quench front and insufficient evaporation of that entrained liquid to provide effective cooling at the top of the core. The lack of a grid spacer models in TRACE may also contribute to the poor prediction of upper elevation cladding temperatures.

The TRACE simulations all show the amount of liquid in the upper plenum is greater than what was measured during the first half of each test, but less than what was measured in the latter half of the tests. TRACE calculated that almost no liquid was carried from the upper plenum into the steam/water separator, in agreement with the test data. Calculated quench times are in good agreement with measured ones for the bottom 1/3 of the core. For higher elevations the calculated quench times fall further and further behind the data as elevation increases. For the top 1/3 of the core calculated quench times lag the data by up to 200 seconds.

Overall Large Break Test Performance

Overall comparison between TRACE predictions and experimental results are considered reasonable. However, there are several concerns and possible code deficiencies observable in the calculations. TRACE was found to significantly overpredict cladding temperatures in the CCTF and gravity reflood driven SCTF simulations. This was attributed to the lack of spacer grid models in the reflood thermal-hydraulics package, poor prediction of entrainment / deentrainment processes in the core, and a sensitivity to condensation in the loops and upper downcomer. In was not clear from the assessments that the liquid that is entrained at the quench front by TRACE results in the correct net carryover to the upper plenum and steam temperatures high in the core. This underpredicts the heat transfer and causes the cladding temperatures to be overpredicted. Condensation in the cold legs and upper downcomer frequently causes large oscillations in the inner vessel collapsed water levels, which in turn causes a misprediction in water reaching the core.

PWR Small Break Integral Test Facilities

Integral small break phenomena was assessed using Semiscale, BETHSY, ROSA-IV, and LOFT small break tests. The purpose of assessment against these cases is to demonstrate TRACE's capability to simulate complex system-wide behavior, event timing, and several physical processes that occur during small break LOCA transients. The integral facilities selected for TRACE assessment cover a broad range of physical scale, as well as a range of transient scenarios. This subsection provides a summary of TRACE performance against these integral tests.

Semiscale

Four Semiscale tests were simulated with TRACE; two natural circulation tests (S-NC-2 and S-NC-3) and two small break LOCA tests (S-NH-1 and S-NH-2). For the natural circulation tests, the objective of these code-to-data comparisons is to evaluate TRACE capabilities for predicting the single and two-phase natural circulation phenomena in an integral facility. In these tests, the primary inventory was systematically drained, and the flow rates measured. The TRACE simulations of Semiscale natural circulation tests S-NC-2 are in good agreement with the experimental data overall. For test S-NC-3 the agreement between TRACE and the data is also good but TRACE slightly underpredicts flowrate at high secondary liquid levels and overpredicts flow rates at very low levels.

Semiscale Tests S-NH-1 and S-NH-2 were Small Break Loss of Coolant Accident (SBLOCA) tests with complete loss of all High Pressure Injection System (HPIS). Test S-NH-1 simulated a 0.5% cold leg pipe break while Test S-NH-2 simulated a 2.1% cold leg pipe break. The two different break sizes resulted in similar core heatup responses during the SBLOCA tests, but with different transient timing of events. The TRACE simulation of Test S-NH-2 showed reasonable agreement between predicted and measure results. The events that occurred during the transient were predicted fairly well, except for the actuation of the steam line automatic depressurization valves (ADVs). The ADVs were opened manually when the peak rod temperature reached 811 K. The calculated rod temperature excursion was delayed and the ADVs were not opened until about 120 seconds after the operator action in the experiment. Although the calculated core dryout appeared reasonable, the calculated rod clad temperature heatup was slower at the upper elevations in the core and TRACE underpredicted the peak clad temperature. Possible reasons include the prediction of excessive liquid entrainment during the boil off and a poor prediction of the critical heat flux.

The TRACE simulation of Semiscale test S-NH-1 was not as good as the simulation of test S-NH-2. The break flow rate was underpredicted resulting in an overprediction of the system pressure. The data showed the broken loop flow rate did not stagnate and liquid was entrained out the break. However, in the simulation the broken loop stagnated. Subsequently, the break flow became mostly steam flow and more liquid remained in the system. With more liquid in the system there was a delay in the core boil off and consequently a delay in the rod heatup. When the rods began to heat-up similar behavior was observed as was observed in the simulation of test SNH-2, i.e. the heatup was slowed and the predicted PCT was lower. Rod temperatures predicted near the top of the core had only a short heatup period. Possible reasons for this include prediction of excessive entrainment and a poor estimate of the critical heat flux.

BETHSY

TRACE was used to simulate experimental data from BETHSY Test 9.1b and Test 6.2TC. BETHSY Test 9.1b (i.e., International Standard Problem No. 27 or ISP-27) simulated a 0.5% (5.08 cm or 2-inch) cold leg break and BETHSY Test 6.2TC simulated a 5.0% (15.24 cm or 6 inch) cold leg break. In both tests, the break was simulated by a side-oriented break nozzle and high pressure injection system (HPIS) was not available. The transients were terminated by the "Ultimate Procedure" in which the steam generator secondaries were dumped to atmosphere when the maximum core heater rod cladding temperature reached 723 K. Depressurization of the steam generator secondary sides allows the primary coolant system to depressurize to the point of accumulator injection followed by low pressure injection system (LPIS) actuation. In the BETHSY simulations, the calculated results generally agree reasonably well with the test data. Although timing of the key events, such as core uncovery, loop seal clearing, etc. in each of the simulations differed when compared to the available test data, the difference is not large and the events were captured reasonably well. Break flow was the most important parameter which resulted in differences between calculations and data. For BETHSY Test 9.1b, differences in predicted and measured behavior result from TRACE predicting two-phase flow earlier at the break location when the measured flow is still liquid. Because the physical layout of the break nozzles for BETHSY Tests 9.1b and 6.2TC are horizontal side connections to the cold leg, the flow in this cold leg could be highly stratified, especially for Test 9.1b. For Test 6.2TC with a larger break size, TRACE overpredicted the integral break flow during an early time period when both the calculation and the test are transition from single-phase to two-phase break flow. The TRACE side offtake model could be the principal source for the deviations in break flow.

The prediction of the heater rod cladding temperatures during core heatup also differed with test data for both test simulations. For Test 9.1b, the differences between predictions and data are related to water inventory and distribution in the pressure vessel. The principal core heatup occurs in Test 9.1b prior to and shortly after initiation of Ultimate Procedure when inventory levels are the lowest. Prior to the Ultimate Procedure, the radial temperature distribution is relatively even. Once the Ultimate Procedure is initiated, the temperature response in the test differs between the inner region and the periphery of the core. This difference was attributed to accumulation of liquid in the upper plenum with draining into the central or inner region of the core while steam flowed upward along the core periphery. While the inner core temperatures quickly turned around following initiation of the Ultimate Procedure, the temperatures continued to rise along the core periphery until accumulator injection was able to reverse and bring uniformity in the core's radial temperature distribution. The TRACE model includes inner and peripheral core regions. Thus, there is the potential to capture the difference in temperature response across the core. However, the TRACE model may not be sufficiently detailed for the events in this particular test.

In Test 6.2TC, core heatup occurs towards the end of the transient when enough inventory has boiled off as a result of the larger break size, the continued isolation of the steam generators because no Ultimate Procedure was initiated, and the unavailability of LPIS. Because TRACE overpredicts the inventory loss out the break and an earlier accumulator injection, the beginning of the final core heatup occurs earlier than in Test 9.1b.

ROSA-IV

Six (6) selected ROSA-IV SBLOCA tests were simulated using the TRACE V5.0 code. In these tests, four different break sizes (0.5%, 2.5%, 5%, and 10%) and three break orientations (top, side, and bottom) were investigated. The performance of TRACE in simulating these tests was characterized by separating the tests into two groups; tests with break sizes larger than 0.5% and the two tests with the 0.5% break.

For the tests with break sizes larger than 0.5%, there was good agreement between the predicted and measured parameters, particularly over the first two phases of the transient, the blowdown and natural circulation or loop-seal clearance phases. As shown in Figure C.5-235, the rate of the blowdown depressurization and the timing and magnitude of the subsequent core-level depression are matched quite well for the 5.% and 10% breaks, with a slightly premature level depression occurring for the smaller 2.5% break. This good agreement was obtained for several processes, such as primary to the secondary heat transfer and break flow and good agreement with cladding temperature.

The primary discrepancies appear just before and shortly after accumulator injection. In tests SB-CL-05, SB-CL-14, and SB-CL-18, there appears to be a delayed affect of the accumulator injection flow, which results in a noticeable drop in core level before the injecting flows begin to aid in core recovery. Then, once the core level begins to rise, it increases too quickly, giving the indication of too much liquid injection. Figure C.5-236 suggests that the accumulator system can be modeled better to bring about better agreement; however, in the SB-CL05 results, the injection flow is significantly under-predicted, yet the subsequently predicted core level surpasses the core-level measurement. Therefore, there appears to be a code problem related to cold leg condensation and fluid mixing. In predictions, almost immediately after the accumulators inject, the temperature in the cold legs plummet to near the injection fluid temperature of 322 K. In the tests however, the temperature drop does not occur as quickly, nor does it reach temperatures as low. This possibly causes the fluid in the downcomer and the core to become more dense, and thus, give indications of higher levels.

The results for the 0.5% break tests were not as good, but they did not indicate any significant code problems. The main problem appears to be in the prediction of the primary-system pressure. Although the agreement between the calculation and the data does not appear unreasonable, the locations and the time over which the differences remained resulted in the predicted core uncovery occurring approximately 350 seconds earlier than expected and inaccurate behavior in the loop seals. It appears that the prolonged unfolding of the transient, as a result of the relatively small break size, magnified any inaccuracies in model geometry, causing inaccurate fluid characterizations at key times during the transient.

Overall, considering the entire range of break sizes, the TRACE code predicted the ROSA-IV SBLOCA transients with *Reasonable Agreement* for all major phenomena.

LOFT

Two Loss-of-Fluid Test (LOFT) small break loss-of-coolant accident (SBLOCA) tests, L3-7 and L3-1, were simulated with TRACE. Tests L3-7 and L3-1 simulated 1-inch (2.54-cm) and 4-inch (10.16-cm) equivalent diameter breaks in the cold leg of a four-loop PWR. The TRACE simulation of LOFT Test L3-7 compared reasonably well with the measured data except for hot and cold leg densities. TRACE overpredicted both primary and secondary side pressures when the SG steam control valve was modeled to behave as described in the experimental data report. However, the measured secondary side pressure, as well as the measured position of the steam valve, suggests that there may have been considerable steam leakage through the valve. Taking into account this leakage, the TRACE calculation was found to be in was in excellent agreement with primary side measured pressures and temperatures.

TRACE did not accurately simulate the drainage of the cold leg when HPIS injection was terminated at 1800 s in Test L3-7. This may be because there is no downcomer to upper head leakage in the TRACE model and there is too much inventory in the vessel in TRACE. A revised TRACE simulation was made with 2% DC/UH leakage flow and a 33% increase in break area. This simulation resulted in hot and cold leg densities that were in much better agreement with the data. However, the agreement of primary system pressure was not as good. The simulation strongly suggests that TRACE, given the actual break geometry, is accurately predicting break flow for the first 400 s of the test but is then underpredicting the break flow throughout the rest of the subcooled blowdown.

The TRACE simulation of LOFT Test L3-1 did not compare well with the test data. The TRACE calculation of primary and secondary pressures were well above the data. The Test L3-1 code-data comparison indicates that an underprediction of critical flow may be responsible for TRACE's overprediction of pressures.

Overall Small Break Test Performance

In general, reasonable agreement was obtained between TRACE and the experimental results for small break integral tests. Most event times were reasonably predicted. Most differences between predicted and measured event times were found to be related to break flow and the effect that mis-predicting the break flow has on primary inventory. Prediction of loop seal clearing however may be problematic and is considered to have minimal agreement. While TRACE generally predicted loop seal clearance at about the correct time, there was not always good agreement in which loop(s) cleared.

BWR Integral Test Facilities

BWR related integral effects behavior was assessed using data from FIST, TLTA, and SSTF. The purpose of assessment against these test is to demonstrate TRACE's capability to simulate system-wide behavior in BWRs, event timing, and several physical processes that occur in BWR transients. This subsection provides a summary of TRACE performance against these integral tests.

FIST

The FIST integral test facility was scaled to a GE BWR/6-218 standard plant. Two tests were simulated with TRACE; Test 6SB2C which was a 0.05 ft^2 recirculation line break without High Pressure Core Spray (HPCS), and Test 6SB1. Test 6SB1 was similar to Test 6SB2C, but included the effect of a stuck open Safety Relief Valve (SRV).

Both FIST SBLOCA tests were simulated using TRACE Code Version 5.0. In both cases, the trends and results agreed reasonably well with the test data. A shift in timing of key events is noted and is attributed to calculated break flow. All key events and sequences of thermal hydraulic phenomena were captured, including the depressurization, vessel flashing, channel uncovery, and rod dryout. The main deviation between the tests and their prediction was associated with the degree of bundle uncovery. In the simulation of Test 6SB1, the bundle was calculated to uncover only partially. This appears to be associated with the amount of liquid entrainment and its deposition during the automatic depressurization system (ADS) blowdown and occurrence of CCFL at the bundle inlet (SEO), the upper tie plate and the bypass top. With the initiation of the ADS blowdown, a large amount of water was calculated to be deposited in the upper plenum, which provided a small but steady supply of water to the bundle until it all drained. This did not occur in tests. The quench front in the TRACE calculation proceeded from the bottom up while in tests both the bottom up and top down quench fronts were observed.

TLTA

The Two Loop Test Apparatus (TLTA) was a single bundle facility scaled to standard BWR/6-218 reactor system, and was used to obtain data for large break loss-of-coolant accident (LBLOCA) phenomena in a boiling water reactor (BWR). TRACE calculations are compared with experimental data from TLTA Test 6425 Run 2 and Test 6424 Run 1. Test 6425 Run 2 was run assuming nominal conditions; average power (5.05 MW), average Emergency Core Cooling (ECC) flow and nominal ECC temperature, and simulates a double-ended rupture of a recirculation line. Test 6424 Run 1 was similar to Test 6425 Run 2, but was conducted with a higher power (6.49 MW). The key events and phenomena of interest in these tests include critical flow, counter current flow limitation (CCFL), and core heatup. In both simulations the code to data results generally agree reasonably well. Major events and thermal hydraulic phenomena were captured. The timings of major events were well predicted in general. However, in both simulations, the system pressure was underpredicted. This is due to an underprediction of the break flow and/or flashing within the system.

SSTF

SSTF was a full-scale mock-up of a 30° sector of a GE BWR/6-218 (624 bundles) design. In Section C.9 of this report, TRACE code calculations are compared with experimental data from SSTF Test EA 3.1 Run 111 and Test EA 3.3-1 Run 119. Test EA3.1 Run 111 was a system transient response test and was conducted to study controlling phenomena related to the BWR/4 ECCS configuration (Ref. 1). It simulated a 1.0-m design basis accident (DBA) break (or 100% recirculation line break). Test EA3.3-1 Run 119 was identical to Test EA3.1/111 except that it simulated a 0.73-m DBA break with some variations in the initial conditions (e.g., mass inventory distribution in the system). The phenomena of interest in these two tests include counter-current flow limiting (CCFL), ECCS injection mixing in the upper plenum and lower plenum, parallel channel phenomena, etc.

The simulation results generally agree reasonably well with the test data. There were however, major deviations in both simulations (e.g., underpredicted pressure, bundle refill, CCFL breakdown timings at the upper core plate) that were attributed to low pressure core spray (LPCS) injection. An improved LPCS model that generates a more realistic LPCS injection liquid distribution in the upper plenum is expected to improve the simulation globally.

Overall BWR Integral Test Performance

The large scale facilities used to assess TRACE for BWR LOCA process were FIST, TLTA, and SSTF. In general agreement between TRACE and measurements is considered reasonable. Problems are noted in SSTF simulations however in which major deviations were observed (e.g., underpredicted pressure, bundle refill, time of CCFL breakdown at the upper core plate) due to inaccuracies and uncertainties in the low pressure core spray distribution prediction. Results in the FIST and TLTA simulations were found to be reasonable considering the impact of mispredicting the break flow.

Known Code Errors

Since the date TRACE V5.0 was "frozen" and released for independent assessment, several code errors have been identified. Errors and problems associated with TRACE are files as "Trouble Reports" that are described and documented in TRACE Developer's Web Site (http://www.nrccodes.com). In many cases these errors have no impact on the results of the assessments reported in this document, or are such that the assessment could be completed without encountering the error. For example, Trouble Reports #335, #336, and #338 involve problems in coupling TRACE and the neutronics code PARCS. In these assessments, PARCS has not been used so these errors or problems have no impact. Others such as Trouble Report #315 (involving input error diagnostics) pertain to code features that are not essential, can be circumvented by making use of other options in the code, or can be resolved by carefully checking the input.

Two errors have been identified however that can affect many of these assessment cases and for which there no work-around options. These errors are discussed here.

Trouble Report #xxx: In three subroutines used in the reflood module the enthalpy was inadvertently calculated as $h = e + P\rho$ instead of $h = e + \frac{P}{\rho}$. The terms are only used in modification of an interfacial heat transfer coefficient and in determination of static quality for use in CHF calculations. Based on a review of the affected coding, the effect on interfacial heat transfer is estimated to be negligible. The CHF calculation however can be significantly impacted at low flow conditions. Assessment cases that do not invoke the reflood module are unaffected.

The code update to correct this error is straight-forward and a test version of TRACE was compiled with the corrections. Several reflood assessment cases were repeated with the new code version. The effect was found to be noticeable, but small; on the order of about 10 K on cladding temperatures. These preliminary results suggest that the effect on the assessment cases in this report is minor, and that correction of this error will not change the conclusions on any particular model or assessment case.

Trouble Report #326: In this error report, large unrealistic pressure gradients were detected for certain geometries. Several test cases to examine the momentum equation showed that for some locations in which fluid velocity changes direction such as at the bottom of the reactor vessel, the pressure drop in TRACE is significantly larger than what one would obtain from a hand calculation. The error appears to cause excessively large pressure drops where axial flow from a downcomer turns into the lower plenum and the turn is accomplished by a single axial level in a VESSEL Component. Several integral test cases have this type of geometric arrangement and are affected by this error. Adding additional axial levels may alleviate this problem, but a code update to resolve the error or provide a sensitivity to its impact is currently unavailable.

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Conclusions

TRACE Version 5.0 has undergone a thorough assessment and validation process in order to determine code accuracy and identify potential deficiencies. The assessment matrix was established by reviewing available PIRTs to large and small break LOCAs, and developing a set of processes that a thermal-hydraulics code such as TRACE must be able to simulate. The assessment matrix that resulted from this consisted of both separate and integral effects tests, and required over 500 simulations. All simulations in this report were performed with a strictly "frozen" version of the code. No updates were applied during the assessment period.

In general, comparisons between code predictions and experimental results were found to be reasonable. That is, major trends were predicted but the code prediction was not always within the measurement bounds. Deficiencies have been identified, and will be the subject of future code corrections.

TRACE can be used reliably for large and small break LOCA in conventional light water reactors based on this report. Users should be aware of the problems observed in the assessments in order to assist in making conclusions based on plant analyses using TRACE. TRACE can also be used for some new and advance plants. The basis for use of TRACE with those plants depends not only on the assessment here, but also on tests specifically for those advanced plants. Simulations with TRACE for advanced plant experimental results are found in other reports.