DEVELOPMENT, VALIDATION AND ASSESSMENT OF THE TRACE THERMAL-HYDRAULICS SYSTEMS CODE

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ABSTRACT

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. Completion of this initial stage of development required an extensive validation and assessment effort, made more complex with the additional requirement to demonstrate the applicability of TRACE to several advanced light water reactors. To accomplish this task a series code assessment exercises were performed generally following the Code, Scaling, Applicability, and Uncertainty (CSAU) approach. This assessment and validation effort is discussed and summarized, as well as the impact the results have had on subsequent code development.

KEYWORDS

Code assessment, TRACE, Thermal-hydraulics

1.0 INTRODUCTION

Efforts to develop the TRAC/RELAP Advanced Computational Engine (TRACE) began in 1997 with the goal of combining the capabilities of TRAC-P, TRAC-B, RELAP5, and RAMONA codes into a single computational platform. The first several years of the project were dedicated into modernization of the coding and selection of various models from the predecessor codes for use in TRACE. Incorporation of features unique to each code, such as the CHAN Component in TRAC-B and side junctions in RELAP5 was undertaken in order to preserve those capabilities. The development effort was accompanied by an extensive verification and validation to assess the accuracy and performance of this essentially new systems code. This V&V effort was, and remains challenging because of the very broad range of intended applications for TRACE.

The TRACE development effort was in addition complicated by the so-called "Nuclear Renaissance" that took place starting in about 2001. Over the decade, there was significant interest by the nuclear industry in gaining Design Certification for new and advanced light water reactors. Many of the novel concepts inherent to the design of these advanced reactors required model and correlation development as well as assessment for phenomena that had lesser importance in conventional LWRs or created conditions for which systems analysis codes had not been validated. Passive safety systems for example, can rely on small gravity driven heads rather than forced flows from pumped injection. Thus, assessment for natural circulation phenomena and low Reynolds number flows took on additional importance.

This paper describes the verification and validation process used in the development of TRACE Version 5.0 and subsequent versions. TRACE Version 5.0 was released in 2007 to NRC staff and several organizations. There have been numerous code "patch" releases since then, as well as four releases of versions that have been subject to the full verification and validation process. The more frequent "patch" releases have limited V&V and are considered developmental. The process described here applies to the full V&V that is performed with these major releases.

1.1 VERIFICATION

The NRC uses a multistep approach to code V&V. Verification is the initial step and ensures software quality. The code developer and programmer are assigned the task of demonstrating that the model(s) or programming changes made to TRACE perform their intended functions. Software quality assurance is detailed in two NUREG reports [1, 2], and adherence to these guidelines is the responsibility of the programmer and the code custodian. This represents the first step in the NRC V&V process. Updates to TRACE not applied to a new code version until they satisfy the guidelines, subjected to the "regression suite" of test cases, and approved by the code custodian. The regression suite, which is the second step in the V&V process, consists of a series of calculations that exercise the code over an extensive range of conditions to test functionality. These regression suite test cases range from simple tests of valve closing logic to cases to ensure that individual models and correlations perform as intended. The regression suite is not static. New regression tests are incorporated into the suite as TRACE features are added or refined, and as code developers examine potential errors. The regression suite has grown to over 2500 cases at this time. This regression suite is automated, so as each update is applied to TRACE problems can be identified before a new executable is released. Table 1 categorizes the regression suite listing the number of cases that address a particular requirement. (The total exceeds 2500 because some tests address the requirements of more than a single category.)

An additional step related to verification is a test of code "robustness." In this step a limited number of assessment cases are automatically re-run and examined for their effect on simulation run-time and agreement with selected parameters (such as peak cladding temperature, steady-state pressure or flow rate). The purpose of the robustness suite is to quickly determine if changes to TRACE have adversely impacted execution time or have unexpected effects on results. If either occurs, the recent code updates are re-examined and revised. Thus, the robustness suite prevents the more resource intensive step of code validation from starting until there is a high degree of confidence in new code updates.

1.2 VALIDATION

While the code verification steps help to ensure that new coding is correctly applied, it does not necessarily assure that the code simulates the correct physics for nuclear plant analysis. Determination of code accuracy and identification of a bias or shortcoming is the objective of validation, as referred to as code assessment. Thus, while verification is a necessary first step it must be accompanied by extensive validation. TRACE validation is categorized into three groups; fundamental problems, separate effects tests, and integral effects tests. Fundamental problems refers to assessment on a very basic level for phenomena such as two-phase pressure drop in vertical and horizontal flows, single-phase frictional pressure drop and single tube counter-current flow. These are closely related to cases in the regression test suite, but usually require a comparison to experimental data. Table 2 lists the fundamental problems used for TRACE assessment.

Central to NRC validation is the Code Scaling, Applicability and Uncertainty (CSAU) methodology [3]. CSAU provides a systematic approach to the application of a systems code to a large scale facility subjected to a complex transient scenario. This is where separate effects tests and integral tests are vital. The methodology is structured and is sufficiently general so that it can be applied to a wide variety of plant designs. The overall framework of CSAU is designed to address three important questions regarding a code to be used for nuclear power plant safety calculations:

1. Has the code the capability to scale up phenomena observed in small-scale test facilities to full-size nuclear power plants (NPPs)?
2. Can the code be applied to safety studies of a particular scenario or a set of scenarios for a given plant design?
3. What is the uncertainty with which the code calculates important parameters, say the peak cladding temperature, in a full scale NPP?

TRACE verification and validation makes particular use of the first two Elements of CSAU; *Requirements and Capabilities, and Assessment and Ranging of Parameters.* Code requirements depend on the scenario, and the models necessary to simulate the scenario. Assessment requires the comparison of code performance against experimental data to determine potential code limitations. Central to these two steps is development of a Phenomena Identification and Ranking Table (PIRT) to identify those physical processes most important to successful simulation of the scenario. For TRACE assessment, three PIRTs were used to identify processes of "generic" interest to large and small break loss-of-coolant accidents (LOCAs). For large break LOCA, the original PIRT by Boyack et al. [4] identified parameters of interest and those processes have found general agreement with PIRTs developed by industry [5, 6]. Table 1 lists the processes of most interest for large break LOCA and the test series used in the assessment of TRACE.

Small break LOCA has been the subject of a fewer number of PIRT efforts, and NRC assessment has made use of two PIRTs developed for PWRs [7, 8]. Of significant interest is that for small break LOCA the dominant processes differ considerably from those in large break LOCA. Different facilities are needed for small break LOCA assessment, and the cases used for TRACE in Table 3 show this.

While not the result of a PIRT process, Reference 9 provides a list of important phenomena for BWR LOCA. There are twenty-four phenomena that occur at some point during a BWR LOCA that must be accounted for in the code assessment. Table 4 lists these phenomena and the assessment tests used for TRACE.

The assessment cases in Tables 2, 3, and 4 represent what is sometimes described as the initial "generic" assessment performed for TRACE. In the heading of each table the number in parenthesis is the number of individual tests simulated. These cases cover a broad range of processes common to most light water reactors at some point in accident scenarios important to reactor licensing. Results of these assessment cases are documented in the TRACE Developmental Assessment Manual [10]. In summary, this generic assessment includes ten different integral test facilities, and total of nearly 300 simulations in the IETs and SETs in the test matrix.

2.0 CODE ASSESSMENT AND MAINTENANCE PROGRAM

Additional assessment is obtained through the Code Assessment and Maintenance Program (CAMP), which provides members with TRACE, RELAP5, and the Purdue Advanced Reactor Core Simulator

(PARCS). PARCS is a multidimensional reactor kinetics code that has been coupled to TRACE and RELAP5. CAMP is a successor to the International Code Assessment and Application Program (ICAP) that was developed by the NRC in 1985 to assess and improve its thermal-hydraulic computer codes. Approximately 30 nations have bilateral cooperative agreements with the United States, providing contributions in the form of model development, code assessment, and application of the codes to nuclear power plants. CAMP members share experience with NRC computer codes to identify errors, perform assessments, and identify areas for additional experiments and model development.

The CAMP program has provided more than 30 NUREG/IAs that contributed to the development, assessment, and application of the NRC T/H analysis codes. Technical areas span the entire range of accident and transient analysis. These studies have significantly contributed to assessment of TRACE, and in identifying features that need to be improved. Separate effects test assessments have included additional examinations of reflood heat transfer [11-14], break flow [15], steam-generator hydraulics [16], loop seal clearance [17], and flooding at the upper core plate [18]. Most of these studies [11-14, 17, 18] provide new assessment not part of the TRACE Developmental Assessment Manual [10].

Integral tests have also been an area of activity in the CAMP program. Assessment has been performed for seven cases [19-25] other than those in the TRACE Assessment Manual, considering a range of break locations including hot leg [24, 25], and upper and lower head [19, 20] small breaks simulated in the ROSA facility. Other integral effects test assessment has been performed using the PKL facility [26-29], IIST [30], and ATLAS [31].

Of particular benefit to TRACE has been assessment that has extended its application to non-domestic reactor systems. Several studies have assessed TRACE for application to the VVER design such as simulation of PACTEL integral tests [32-36], and studies that have used VVER plant data [37, 38]. Additional assessment has also been performed to extend TRACE to CANDU reactor systems through simulation of tests in the RD-14 integral facility [39].

2.1 INTERNATIONAL COOPERATIVE ACTIVITIES

An important component of TRACE assessment activity is derived from international cooperative research. These include tests recently competed in ROSA, and currently being conducted in PKL and ATLAS. The current project in PKL is an OECD-fostered study investigating steam generator hydraulics and transients that may lead to boron precipitation under postulated accident conditions. ATLAS is also an OECD sponsored project that is investigating transients that extend the applicability of systems codes by providing data for scenarios involving station blackout.

3.0 EXTENDED APPLICABILITY OF TRACE

The primary objective of the initial assessment was to validate TRACE for large and small break LOCA in conventional light water reactors. Since then, a greater emphasis was placed on the extension of TRACE to new and advanced reactors to support Design Certification reviews and to assess TRACE for a broader range of accident scenarios.

3.1 NEW AND ADVANCED LWRS

A major area of interest over the previous decade for the NRC has been the development and assessment of TRACE for applicability to new and advanced reactors. For each reactor system undergoing Design Certification, confirmatory calculations of selected accident scenarios are performed by the NRC staff. These confirmatory calculations enable the staff to better evaluate the applicant's submittal, and to assess safety margins in the plant design. While the new designs offer significant improvements to safety, accurate analysis of their performance can be challenging. Several of the new designs place a greater emphasis on passive system systems or components that do not exist in conventional light water reactors, which require additional code assessment and in some cases, model development and new experimental data.

For each new and advanced design, the NRC has used CSAU as a guide. An independent PIRT is first developed, and then the TRACE models and correlations are reviewed for applicability. The PIRT is developed as early as possible in the licensing review, in order to allow as much time as possible in the schedule for model development and independent testing (if necessary). Specific assessment is performed and included in an applicability report, which serves to document the PIRT and evaluation of TRACE models for the particular design. TRACE applicability reports have been completed for ESBWR [40], EPR [41], APWR [42], ABWR [43] and TRACE was subsequently then used for confirmatory analysis. More recently, applicability reports have been completed for small modular reactors [44, 45] with modifications to TRACE for SMR unique features and assessment underway.

Finally, an applicability report was developed for TRACE application to the AP1000 [46]. Since confirmatory analysis used RELAP5 for the AP600 review, it was also used for AP1000. However, since an agency objective is to phase out support for RELAP5, assessment was performed to evaluate TRACE for application to AP1000. Assessment of TRACE for AP1000 made extensive use of confirmatory testing in the APEX facility [47].

3.2 BWR AOO and ATWS

Anticipated operational occurrences (AOO) and anticipated transient without scram (ATWS) represent two scenarios of interest. For AOOs, a PIRT developed by Boyack et al. [48] was used to identify phenomena and determine tests for assessment of TRACE. Applicability of TRACE was demonstrated through simulation of 19 integral and separate effects tests in FIST, FRIGG, BFBT in addition to assessment against the Christianson subcooled boiling experiments with results documented in Reference 49.

ATWS has been addressed by development of a PIRT and assessment of TRACE [50]. Assessment included tests with parallel channel instabilities using FRIGG and plant data from Peach Bottom and Ringhals.

4.0 RECENT TRACE DEVELOPMENT

TRACE verification and validation is a continuing effort. As new problems are addressed additional features must be added to TRACE, and biases and shortcoming found in the assessment must be corrected. Current efforts are directed at several improvements, described briefly next.

4.1 Fuel Rod Models

The fuel rod models in TRACE Version 5.0 are legacy models from TRAC and RELAP5. Recent evaluations identified several shortcomings in one-to-one comparison with FRAPCON, which is considered to be the NRC's state-of-the-art code for fuel performance. To correct for deficiencies, FRAPCON models for gap conductance and fuel thermal conductivity have been implemented into TRACE. Current efforts are expected to modify models in TRACE such that they are nearly identical to those in FRAPCON.

4.2 Automatic Backup Logic

A common problem in systems codes is premature aborts due to numerical difficulties in achieving a solution. The solution is frequently to reduce the time step size to enable the code to run through the "rough spots" which occur when there are sharp transitions presented to various models in the calculation. Traditionally, a code such as TRACE would stop, backup a single time step while cutting it in half and then attempting to proceed. Sometimes this worked, often it did not.

A recent update to TRACE, with alternate backup logic has been applied, and found considerable success. The alternate backup logic places periodic "holding points" in the calculation. If a numerical problem occurs, the code stops and backs up several seconds and then cuts the time step size. In many cases this automated procedure that has been implemented in TRACE allows the TRACE runs to complete significantly faster. For example, the APEX and ROSA models were able to complete roughly an order of magnitude faster because of being able to run with much larger average timestep size. Basically, the analyst's input strategy is now to deploy reasonably large timestep cards into the deck and let TRACE sort out any instabilities. This is a significant improvement over wasting analyst's time by having them manually lower the timestep size when it may not be necessary.

In summary, the alternate backup logic that has been placed in TRACE makes the code more robust, often times faster, and most importantly it saves TRACE analysts' time by automating what used to be an arduous manual process of running and re-running TRACE with smaller time step sizes over sensitive portions of the transient calculation.

4.3 Higher Order Numerics

TRACE now has an option to solve the conservation equations using a second order spatial numerical scheme for the mass and energy advection terms. This option significantly reduces numerical diffusion in regions where there are steep gradients compared to the semi-implicit or SETS first order advection schemes. It gives increased accuracy for problems like BWR density waves and advecting temperature or concentration fronts [51]. Users can also select the second order scheme, the semi-implicit scheme, or SETS on a component by component basis. This allows using the second order scheme where it would be known to be important like in a BWR channel in an ATWS instability calculation and still use SETS in high velocity regions such as steam line and relief valves to avoid very small Courant limited timestep sizes.

4.4 Boron Transport and Precipitation

Boron transport and local concentration is of interest because of its effect on local power due to its impact on kinetics, and because of its potential to precipitate out of solution as its concentration becomes high. TRACE has been modified to account for the effect of boron concentration on coolant density and viscosity. As the coolant becomes highly concentrated with boron, the density and viscosity increase considerably. This can change natural circulation flow in the core during long-term cooling.

4.5 Code Uncertainty

TRACE has been modified so that individual models and correlations can be ranged assuming a distribution and uncertainty selected by the User. Input parameters and boundary condition parameters can also be sampled and ranged. Current efforts are directed toward development of a recommended set of parameters along with distributions based on TRACE assessment results. The objective is to enable a User to examine various code uncertainty methods and the effect of parameter ranging assumptions on results.

An important contributor to code uncertainty is the so-called “User Effect” where equally proficient Users can simulate a transient with options and modeling features allowed in a code obtaining different results. To at least minimize this effect, a set of TRACE User Guidelines are being developed. These Guidelines maintain uniformity between code options and nodalizations used in assessment and plant models.

4.6 Multifield Capability

A long-term objective for TRACE, which is currently in progress, is its transition from a 2-fluid, 2-field formulation to a 2-fluid, 4-field formulation. Rather than liquid and gas fields, the next version of TRACE will separately model a droplet field, a continuous liquid film field, and two gas fields; one for small bubbles and a second for slug bubbles/continuous gas. This will enable TRACE to more effectively simulate the effect of spacer grids and interfacial area transport.

5.0 SUMMARY AND CONCLUSIONS

Development and assessment of TRACE has continued since its initial release in 2007. A significant forcing function for additional assessment has been the need for the NRC to perform confirmatory calculations for new and advanced reactors. For each of these new designs Applicability Reports and assessment specific to the design are produced using the CSAU approach as a guide. Efforts continue in order to extend the range of applicability of TRACE to new accident scenarios such as BWR ATWS with instability and to improve its overall accuracy.

6.0 NOMENCLATURE

ABWR	Advanced BWR	LOCA	Loss of Coolant Accident
ANL	Argonne National Laboratory	LOFT	Loss of Flow Test
AOO	Anticipated Operational Occurrences	MB	Model Boiler
ATWS	Anticipated Transient Without Scram	NPP	Nuclear Power Plant
APEX	Advanced Plant Experiment	NRC	Nuclear Regulatory Commission
APWR	Advanced PWR	OECD	Organization for Economic Cooperation and Development
AP1000	Advanced Passive 1000	PACTEL	Parallel Channel Test Loop
ATLAS	Advanced Thermal-hydraulic Test Loop for Accident Simulation	PARCS	Purdue Advanced Reactor Core Simulator
BETHSY	Boucle d'Etudes Thermalhydrauliques Systèmes	PIRT	Phenomena Identification and Ranking Table
BFBT	BWR Full-Size Fine-Mesh Bundle Tests	PKL	PrimarKreisLauf
BWR	Boiling Water Reactor	RBHT	Rod Bundle Heat Transfer
CAMP	Code Assessment and Maintenance Program	ROSA	Rig of Safety Assessment
CANDU	Canada Deuterium Uranium	RELAP	Reactor Excursion and Leak Analysis Program
CCFL	Countercurrent Flow Limiting	SBLOCA	Small Break LOCA
CHF	Critical Heat Flux	SCTF	Slab Core Test Facility
CL	Cold Leg	SET	Separate Effects Test
CSAU	Code, Scaling, Applicability and	SETS	Stability Enhancing Two-Step

	Uncertainty		
DNB	Departure from Nucleate Boiling	SG	Steam Generator
ECC	Emergency Core Cooling	SMR	Small Modular Reactor
EPR	Evolutionary PWR	SSTF	Steam Sector Test Facility
ESBWR	Economical Simplified Boiling Water Reactor	THTF	Thermal Hydraulic Test Facility
FIST	Full Integration Simulation Test	TLTA	Two Loop Test Apparatus
FLECHT	Full Length Emergency Core Heat Transfer	TPTF	Two Phase Test Facility
FRAPCON	Fuel Rod Analysis Program	TRAC	Transient Reactor Analysis Code
GE	General Electric	TRACE	TRAC/RELAP Computational Engine
HL	Hot Leg	UCB	University of California, Berkeley
ICAP	International Code Assessment and Application Program	UCSP	Upper Core Support Plate
IET	Integral Effects Test	UP	Upper Plenum
IIST	Institute of Nuclear Energy Research Integral System Test	UPTF	Upper Plenum Test Facility
LBLOCA	Large Break LOCA	VVER	Voda-Vodyanoi Energetichesky Reaktor
LP	Lower Plenum		

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Table 1: NRC Regression Suite for TRACE Verification

Regression Test Category	Number of Cases
Numerics and Solution Procedure	1296
Input and Output	301
Control Systems	144
Power and Kinetics	225
Flow Process Models	232
Closure Models	301

Heat Structures	982
Integral Behavior	318
Component Models	1336

Table 2: Fundamental Assessment Cases for TRACE

Fundamental Assessment Case	Cases	Purpose
Radial and Axial Heat Conduction	2	Compare heat conduction calculation to exact solutions.
Drain – Fill	4	Examine ability to track water level across node boundaries.
Oscillating Manometer	1	Compare calculation of two-phase interface to exact solution.
ANL Vertical Two-Phase Flow	71	Prediction of void fraction in vertical two-phase upflow.
TPTF Horizontal Flow Tests	110	Prediction of two-phase flow in a large diameter horizontal pipe.
Single and Two-Phase Wall Friction	27	Prediction of wall friction component of pressure drop.
Single Tube Flooding	3	Examine calculation of flooding and CCFL correlations.
CISE Adiabatic Tube	1	Assess interfacial shear under adiabatic conditions.

Table 3: PWR Large Break LOCA "Generic" Processes and Assessment

Key: ● = simulated ⊖ = partially simulated - = not simulated or measured	LOFT - LBLOCA (3)	UPFT (17)	CCTF (7)	SCTF (7)	Marviken (27)	Moby Dick (10)	Super Moby Dick (8)	THTF Blowdown (4)	FLECHT-SEASET (8)	RBHT Reflood (6)	GOTA Reflood (1)	FLECHT-SET SG Tests (4)	MB-2 SG Tests (1)	MIT Pressurizer Test (2)	UCB Condensation (30)	Dehbi Condensation (7)	U. Wisc. Condensation (6)	Bankoff CCFL (1)
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, cold leg)	●	●	●	⊖	-	-	-	-	-	-	-	-	-	-	⊖	⊖	⊖	-
Hot wall heat transfer (downcomer)	●	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Two-phase performance (pump)	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
ΔP, form losses (pump)	⊖	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Noncondensible gas (CL, accumulator)	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	⊖	⊖	⊖	
Two-phase ΔP (loop)	●		●	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Oscillations (loop)	●		●	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Flow split (loop)	●		●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Table 4: Assessment for Small Break LOCA Processes.

Key:

● = simulated

⊖ = partially simulated

- = not simulated or measured

	LOFT – SBLOCA (2)	ROSA-IV (6)	BETHSY (2)	Semiscale (4)	SCTF (7)	UPTF (17)	THTF Mixture Level and Uncovery (12)	FRIGG (29)	Wilson Bubble Rise (60)	RBHT Mixture Level (10)	RBHT Steam Cooling (7)	UCB Condensation (30)	Dehbi Condensation (7)	U. Wisconsin Condensation (6)	Bankoff CCFL Tests (1)	Single Tube Flooding (2)
Oxidation (fuel rod)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Decay heat (fuel rod)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Local power (fuel rod)	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
3D 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/HL 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)	-	●	●	●	-	⊖	-	-	-	-	-	-	-	-	-	-
Mixture level/Flashing/Void fraction (downcomer/LP)	●	●	●	-	●	●	-	-	-	-	-	-	-	-	-	-
Critical flow in complex geometries (break)	-	●	●	-	-	-	-	-	-	-	-	-	-	-	-	-
Upstream flow regime (break)	●	●	●	●	-	-	-	-	-	-	-	-	-	-	-	-

Table 5: Assessment Matrix for BWR LOCA.

Key:

● = simulated

⊖ = partially simulated

- = not simulated or measured

	FIST (2)	SSTF (2)	TLTA (2)	UPTF (17)	Marviken (27)	GE Vessel Blowdown (2)	THTF Mixture Level and Uncovery (12)	FRIGG (29)	RBHT Mixture Level (10)	GOTA Reflood (1)	GOTA Radiation (1)	FLECHT-SEASET Reflood (8)	RBHT Reflood (6)	RBHT Steam Cooling (7)	Dehbi Condensation (30)	UCB Condensation (7)	U. Wisconsin Condensation (6)	SCTF (7)	CCTF (7)
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 (DNB, dryout, rewet, radiation heat transfer)	●	-	●	-	-	-	-	-	-	●	●	●	●	●	-	-	-	●	●
Quench front propagation (fuel rods and channel walls)	●	-	●	-	-	-	-	-	-	●	-	●	●	●	-	-	-	●	●
Entrainment and de-entrainment in core and UP	⊖	⊖	⊖	-	-	-	-	-	-	●	-	●	●	●	-	-	-	●	●
Separator behavior (flooding, steam penetration and carryover)	●	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Spray cooling	⊖	-	⊖	-	-	-	-	-	-	●	-	-	-	-	-	-	-	-	-
Spray distribution	-	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Steam dryer – hydraulic behavior	⊖	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Single and two-phase pump recirculation behavior including jet pumps	⊖	-	⊖	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Phase separation and mixture level behavior	●	-	⊖	-	⊖	●	●	●	●	-	-	-	-	-	-	-	-	-	-
Guide tube and lower plenum flashing	●	-	●	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
Natural circulation – core & 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	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-