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THE CSNI SEPARATE EFFECTS TEST AND INTEGRAL TEST FACILITY MATRICES FOR VALIDATION OF BEST-ESTIMATE THERMAL-HYDRAULIC COMPUTER CODES

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ABSTRACT

Internationally agreed Separate Effects Test (SET) and Integral Test Facility (ITF) matrices for validation of realistic thermal hydraulic system computer codes were established. ITF development is mainly for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). These matrices were established by sub-groups of the Task Group on Thermal Hydraulic System Behaviour as requested by the OECD/NEA Committee on Safety of Nuclear Installations (CSNI) Principal Working Group 2 on Coolant System Behaviour.

Firstly, the main physical phenomena that occur during considered accidents are identified, test types are specified, and test facilities suitable for reproducing these aspects are selected. Secondly, a list of selected experiments carried out in these facilities has been set down. The criteria to achieve the objectives are outlined. In this paper some specific examples from the SET and ITF matrices will also be provided. In addition, a short summary on the status of validation matrices for Russian Pressurised Water-cooled and Water-moderated Energy Reactor (WWER) is presented.

The matrices will be a guide for code validation, will be a basis for comparisons of code predictions performed with different system codes, and will contribute to the quantification of the uncertainty range of code model predictions. In addition to this objective, the construction of such a matrix is an attempt to record information which has been generated around the world over the last 25 years, so that it is more accessible to present and future workers in that field than would otherwise be the case.

LECTURE OBJECTIVES

Lecture on this subject will provide an idea about how the validation matrices are established and how they are used with a possible extension to natural circulation phenomena and to passive decay heat removal systems.

1. INTRODUCTION

For the analyses of transients and loss-of-coolant accidents (LOCAs) in Light Water Reactors (LWRs) thermal-hydraulic computer codes have been developed over the last thirty years.

Starting with relative simple computer codes in the early 1970's, a continuous development of the codes has been performed with respect to a more realistic description of thermal hydraulic phenomena and a more detailed system representation.

At the beginning of the 1970's, codes for the analysis of large break LOCAs had been requested. The codes were based on the homogeneous equilibrium model, assuming equal velocities and temperatures of vapour and liquid phases. The next effort in code development was directed by the demand for the simulation of transients and small break accidents. The implementation of new models allowed for the separation of vapour and liquid by gravity. The representation of primary and secondary side with control systems and balance of plant models were extended.

In the middle of the 1970's the development of a new generation of thermal-hydraulic codes were initiated to provide analytical tools for a more realistic simulation of LWR behaviour under transient and accident conditions. Thermal and mechanical non-equilibrium phenomena have been taken into account. The effects of non-condensables and boron tracking have been considered. These codes allow the simulation of transients, the entire range of break sizes as well as beyond design basis accidents including accident management procedures with operator interventions.

Parallel to the development of the analytical tools a large variety of experimental programmes have been executed to improve the understanding of thermal-hydraulic phenomena, to study system behaviour, and to provide the required data base for code development and code validation.

A very high number of separate effects tests have been performed for the development and validation of code models. Separate effects tests investigate individual phenomena under clear boundary conditions. While in the 1970's the experiments were conducted mainly on small scale test facilities, in the 1980's more attention has been directed to scaling. For example, in 1986, the first tests at the test facility UPTF, a representation of a four loop 1300 MWe PWR with upper plenum, downcomer and the main coolant pipes in full scale reactor geometry, were performed.

The overall results of the code calculations are validated mainly by data from integral test facilities representing the primary and secondary coolant systems. While in the early 1970's the experiments were focused on large break issues, in the following, up to now, parallel to the advancement in code development, integral tests have been carried out to investigate LWR system behaviour during transients, small breaks, transients under shutdown conditions, and beyond design basis accidents. In addition to the results of integral tests LWR plant data of transients or accidents are being used to validate the predictive capability of the codes.

Construction of validation matrices is an attempt to collect together the best sets of test data for code validation and improvement from the wide range of experiments that have been carried out world-wide in the field of thermal-hydraulics. The first formulation of a validation matrix was proposed by Wolfert and Frisch from GRS [1]. This activity was taken by a CSNI sub-group to establish matrices for PWR and BWR.

In addition, to set-up validation matrices for Russian Pressurized Water-cooled and Water-moderated Energy Reactor (WWER) analyses, an international Working Group was formed on the initiative of the Federal Ministry for Research and Technology (BMFT) of the Federal Republic of Germany. A further evaluation of the WWER matrices was performed by a CSNI Support Group.

Based on these CSNI matrices the lists of phenomena have been reviewed and adopted to the characteristics of WWER-440 and WWER-1000 systems respectively, and the lists of test facilities suitable for code assessment have been completed.

2. DEFINITIONS

Computer codes simulate the system behaviour of nuclear power plant as realistic as possible („best estimate“). These computer codes are used to investigate:

- Incidents and accidents of different scenarios and their consequences,
- the effectiveness of emergency procedures.

The process carried out by comparing code predictions with experimental measurements or measurements in a reactor plant (if available) are called validation. A code or code model is considered validated when sufficient testing has been performed to ensure an acceptable level of predictive accuracy over the range of conditions over which the code may be applied. Accuracy is a measure of the difference between measured and calculated quantities taking into account uncertainties and biases in both. Bias is a measure, usually expressed statistically, of the systematic difference between a true mean value and a predicted or measured mean. Uncertainty is a measure of the scatter in experimental or predicted data [2]. The acceptable level of accuracy is judgmental and will vary depending on the specific problem or question to be addressed by the code. The procedure for specifying, qualitatively or quantitatively, the accuracy of code predictions is also called code assessment.

The international literature often distinguishes between the terms validation and verification. A mathematical model, or the corresponding computer code, is verified when it is shown that the code behaves as intended, i.e., it is a proper mathematical representation of the conceptual model and that the equations are correctly encoded and solved. In this context, the comparison with measured values is not part of the verification process. The term verification, however, is often used synonymously with validation and qualification [2]. Therefore, the term verification has also been used in the code validation work, including comparisons between calculations and measurements.

3. SEPARATE EFFECTS TEST VALIDATION MATRIX

In March 1987, the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) published a document that identified a set of tests which were considered to provide the best basis for the assessment of the performance of thermohydraulic codes, "CSNI Code Validation Matrix of Thermohydraulic Codes for LWR LOCA and Transients", [3], [4], and [5]. The set of tests was chosen to include examples of all phenomena expected to occur in plant transients and LOCA analyses. Tests were selected on the basis of the quality of the data, variety of scaling and geometry, and appropriateness of the range of conditions covered. A decision was made to bias the validation matrix towards integral tests in order that code models were exercised, and interacted, in situations as similar as possible to those of interest in LWR plant. This decision was taken on the assumption that sufficient comparison with separate effects tests data would be performed, and documented, by code development, that only very limited further assessment against separate effects test data would be necessary. This last expectation has proved unrealistic; it is now recognized that continued comparison of calculations with separate effects test data is necessary to underwrite particular applications of codes, especially where a quantitative assessment of prediction accuracy is required, as well as for code model improvement.

It has been decided to develop a distinct Separate Effects Test Matrix rather than extend the original CSNI Code Validation Matrix (CCVM), which consisted almost entirely of integral tests.

Only in some specific cases where integral test facility data were not available, were separate effects tests used in the CCVM. The development of the separate effects test matrix was found to require an extension of the methodology employed for the CCVM both in the scope and definition of the thermal hydraulic phenomena and in the categorization and description of facilities.

There are several reasons for the increased importance now placed on the comparison of codes with separate effects test data. Firstly, it has been recognized that the development of individual code models often requires some iteration, and that a model, however well conceived, may need refinement as the range of applications is widened. To establish a firm need for the modification or further development of a model it is usually necessary to compare predictions with separate effects data rather than rely on inferences from integral test comparisons.

Secondly, there is the question of uncertainties in predictions of plant behaviour. A key issue concerning the application of best estimate codes to LOCA and transient calculations is quantitative code assessment. Quantitative code assessment is intended to allow predictions of nuclear power plant behaviour to be made with a well defined uncertainty. Most schemes for achieving this quantification of uncertainty rely on assigning uncertainties to the modelling by the code of individual phenomena, for instance by the determination of reasonable ranges which key model parameters can cover and still produce results consistent with data. This interest has placed a new emphasis on separate effects tests over and above that originally envisaged for model development.

In the thermohydraulic codes, the physical processes are simulated by mechanistic models and by correlations. The prediction of particular phenomena, such as level swell or counter-current flow limitation, by a code, are usually dominated by one, or perhaps a few, code models. Comparison of code predictions of basic phenomena with events observed in the relatively simple situations contrived in separate effects test facilities, often allows a better assessment of the accuracy of code models than it is possible to make with data from integral tests. This may be, for instance, because steady state rather than transient observations are possible in the separate effects tests; or because in a separate effects test facility dedicated to the study of one particular phenomenon, the measurement instrumentation can be chosen more appropriately, with less need to compromise. The more highly controlled environment of the SET is likely to lead to a more systematic evaluation of the accuracy of a model across a wide range of conditions.

A further incentive to conduct separate effects tests, in addition to those carried out in integral facilities, is the difficulty encountered in scaling predictions of phenomena from integral test facilities (which of necessity are in some sense small scale) to plant applications. Where a phenomenon is known to be highly scale dependent and difficult to model mechanistically, there is a strong case for conducting separate effects tests at full scale. In general, it is desirable to have a considerable overlap of data from different facilities; successfully predicting data from different facilities provides some confirmation that a phenomenon is well understood. The main objective in producing the SET cross reference matrix is to identify the best available sets of data for the assessment, validation and, finally, the improvement of code predictions of the individual physical phenomena. While both integral test data and SET data are appropriate for code validation and assessment, for model development and improvement there should be a strong preference for SET data.

3.1 The Methodology Developed

In the process of establishing the SET validation matrix, a methodology has been developed. This methodology helps to collect and present the data and information collected in a comprehensive and systematic manner. It is a general methodology and therefore, in principal, also applicable to the other type of validation matrices (e.g. on severe accidents). The methodology can be summarized as follows:

1. Identification of phenomena relevant to two-phase flow in relation to LOCAs and thermal-hydraulic transients in light water reactors (LWRs).
2. Characterization of phenomena, in terms of a short description of each phenomenon, its relevance to nuclear reactor safety, information on measurement ability instrumentation and data base. In addition to these points, the present state of knowledge and the predictive capability of the codes is included in the characterization of each phenomenon.
3. Setting up a catalogue of information sheets on the experimental facilities, as a basis for the selection of the facilities and specific tests [6b].
4. Forming a separate effects test facility cross-reference matrix by the classification of the facilities in terms of the phenomena they address.
5. Identification of the relevant experimental parameter ranges in relation to each facility that addresses a phenomenon and selection of relevant facilities related to each phenomenon.
6. Establishing a matrix of experiments (the SET matrix) suitable for the developmental assessment of thermal-hydraulics transient system computer codes, by selecting individual tests from the selected facilities, relevant to each phenomenon.

3.2 Forming a SET Cross-Reference Matrix

The main objective in producing the Separate Effects Test Facility Cross Reference Matrix (SET CRM) is to identify the best available sets of data for the assessment, validation and, finally, the improvement of code predictions of the individual physical phenomena. While both integral test data and SET data are appropriate for code validation and assessment, for model development and improvement there should be a strong preference for SET data.

The thermohydraulic phenomena of interest in LWR LOCA and transients are listed in Table 1. A set of basic two-phase flow and heat transfer processes which are important for the thermohydraulic codes in the form of basic constitutive relations have been added explicitly to the list under the heading "Basic Phenomena". The scope of the SET Facility CRM has been restricted to those phenomena directly affecting the thermohydraulic behaviour in a transient or LOCA.

The resulting list of 67 thermohydraulic phenomena forms one axis of the SET Facility CRM. The second axis of the Matrix consists of the 187 facilities identified as potential sources of separate effects data. The test facilities in 12 OECD member countries are compiled (Table 2) according to the country in which they operate: Canada, Finland, France, Germany, Italy, Japan, Netherlands, Sweden, Switzerland, United Kingdom, USA, Norway. An example for SET facility CRM is shown in Table 3. The SET facility CRM tables for each country can be seen in [6a]. For each test facility the phenomena addressed by the corresponding experimental research programme have been indicated in these Matrix tables, yielding the SET CRM for test facilities and thermohydraulic phenomena.

The correlation between phenomena and SET Facility is assigned to one of three levels:

- suitable for model validation, which means that a facility is designed in such a way as to simulate the phenomenon assumed to occur in a plant and is sufficiently instrumented (x);
- limited suitability for model validation: the same as above with problems due to imperfect scaling, different test fluids (e.g. Freon instead of water) or insufficient instrumentation (o);
- not suitable for model validation: obvious meaning, taking into account the two previous items (-).

This Matrix shows both the number of different phenomena covered by the experimental investigation with one test facility, and the number of different facilities in which an individual phenomenon has been investigated. The test facilities differ from each other in geometrical dimensions, geometrical configuration and operating capabilities or conditions. Therefore, the number of facilities relevant to an individual phenomenon provides some indication of the range of parameters within which a phenomenon has been investigated and experimental data generated. For instance, it is obvious from the SET CRM presented in [6a] that heat transfer phenomena, especially post critical heat flux, departure from nucleate boiling/dryout and quench front propagation/rewet, were investigated in many SET facilities.

For the systematic evaluation of the capabilities of a thermohydraulic code, appropriate experiments have to be identified which provide data over the range of conditions of interest (as far as such data is available), for each phenomenon listed.

3.3 Establishing the Separate Effects Tests (SET) Matrix

For each of the 67 phenomena, a table presents the tests, which have been identified as suitable for code validation with respect to that phenomenon, from the test facilities selected. An example for a phenomena and related tests are given in Table 4. The arguments for the selection of the facilities for a given phenomenon are already identified with the previous step of the methodology.

In order to try to be practical, the number of facilities has been limited to 3 on the average, though in some special cases up to 5 are used. For heat transfer, a larger number was used, because of the large number of parameters affecting heat transfer and its high degree of importance. The total maximum number of tests has been fixed at up to 20 per phenomenon. Here a test is considered to be a set of data points involving one key parameter variation (e.g. a flooding curve at a single pressure and tube geometry). These numbers indicate the large amount of work, which is necessary to assess a code.

It must be emphasized that tests have been chosen on the basis of available information: It is not always possible to determine how satisfactory data is for code validation until it is actually used (completeness of boundary condition information; measurement accuracy, internal consistency etc.) The situation of the various experimental programs and chosen tests varies greatly in this respect.

The tests have been selected in order to cover the experimental data range as defined, knowing that the plant range is not always covered. Particular attention has been given to the geometric scaling problem and small, medium and large scale separate effect facilities have been integrated whenever possible.

As some facilities are useful with respect to several separate effects phenomena, a cross check and a tentative harmonization of the selected tests have been made when possible, in order to try to minimize the number of input data needed for code validation.

In this matrix the selected tests are ordered following one arbitrary chosen main parameter (for example system pressure) with, optionally, additional parameters (for example, representative diameter). This will give the user an indication of the available range of data for code validation, and the possible need for additional tests.

At the bottom of the table the main references, if identified, are given for the chosen tests. The reader is supposed to have enough information in these references to be able to compute the test. Some examples of the SET matrix for selected number of phenomena are given in Table 5. Further tables for each of the 67 phenomena are given in detail in [6a].

Additional information related to the type of tests, or parameter ranges for instance are also provided in the listed references. This matrix has been published as a first attempt. It may be updated by new and additional input from the owners and by remarks from the users. Nevertheless, as it is, this separate effect test matrix covers a large number of phenomena within a large range of selected parameters. If a thermal-hydraulic code is to be used to cover a certain number of phenomena then calculation of the relevant identified tests in the matrix is considered to be a basic step toward the achievement of code qualification.

4. INTEGRAL TEST FACILITY VALIDATION MATRICES

The validation of codes is mainly based on pre-test and post-test calculations of separate effect tests, integral system tests, and transients in commercial plants. An enormous amount of test data, available for code validation, has been accumulated. In the year 1987 the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (NEA) in the Organization for Economic Co-Operation and Development (OECD) issued a report compiled by the Task Group on the Status and Assessment of Codes for Transients and ECC [3, 4, 5]. It contains proposed validation matrices for LOCA and transients, consisting of the dominating phenomena and the available test facilities, and the selected experiments.

Since the issue of the Validation Matrix Report in 1987, new tests have been performed and an update of the validation matrix was published in the year 1996 [7]. In this report a revision and update of the matrices, including experimental facilities and identified experiments was performed. Two new matrices were included, those for „accident management for a non degraded core in PWRs“ and „transients at shutdown in PWRs“. Additional phenomena and test types were identified for these new matrices. A special chapter on counter-part tests, similar tests and International Standard Problem tests was introduced in this revision of the report. Counter-part tests and similar tests in differently scaled facilities are considered highly important for code validation. International Standard Problem experiments are carefully controlled, documented and evaluated. Therefore, these experiments are a good basis for code validation, and they were included in the tables of selected experiments. Additional work was performed to describe the content of the validation matrices, i.e. the test types, the phenomena, and most of the selected tests. A brief description of thermal-hydraulic aspects of severe accidents was included. The thermal-hydraulic codes are being extended to the thermal-hydraulics prevailing under severe accident conditions. They cannot be considered validated at the present time. Experimental data are limited. The important phenomena for severe accident conditions, with particular emphasis on the thermal-hydraulic phenomena were summarized in the report [7].

It is to be noted that the methodology established for the SET validation matrix (as described in section 3, above) has been applied during the establishment of the Integral Test Facility validation matrices.

4.1 Integral Test Cross Reference Matrices

To systematize the selection of tests for code validation, so called „cross reference matrices“ have been established for the first step. Based on these matrices, phenomenologically well founded sets of experiments, for which comparison of measured and calculated parameters form a basis for establishing the accuracy of test calculation results, have been defined in a second step.

In the cross reference matrices the important physical phenomena which are believed to occur during the transient or LOCA, the experimental facilities suitable for reproducing these effects, and the test types of interest are listed. The relationships:

- phenomenon versus test type indicate which phenomena are occurring in which test types,
- test facility versus phenomenon indicate the suitability of the test facilities for code validation of the different phenomena, and
- test type versus test facility indicates which test types are performed in which test facilities.

For PWR facilities six individual matrices were prepared, differentiating between:

- large breaks,
- small and intermediate breaks for PWR with U-tube steam generators,
- small and intermediate breaks for PWR with once-through steam generators (OTSG),
- transients,
- transients at shut-down conditions,
- accident management for a non-degraded core.

The matrix for small and intermediate breaks in PWRs with once-through steam generators have been prepared to address in particular phenomena which are unique to this reactor type.

For BWR facilities two individual matrices have been prepared, differentiating between:

- loss of coolant accidents (LOCA),
- transients.

In Tables 6 to 10 cross reference matrices for PWR facilities with U-tube steam generators are shown. Among the integral system test facilities, the category „PWR“ is included under „test facilities“. The analysis of accidents in actual nuclear power plants is potentially valuable with reference to scaling and simulation problems. Descriptions of phenomena and test types can be found in reference [7].

The relationship phenomenon versus test type is rated at one of three levels:

- occurring: which means that the particular phenomenon is occurring in that kind of test (plus sign in the matrix);
- partially occurring: only some aspects of the phenomenon are occurring (open circle in the matrix);
- not occurring (dash in the matrix).

The relationship test facility versus phenomenon is rated at one of three levels:

- suitable for code assessment: a facility is designed in such a way as to simulate the phenomenon assumed to occur in the plant and it is sufficiently instrumented to reveal the phenomenon (plus sign in the matrix);
- limited suitability: the same as above with problems due to imperfect scaling or insufficient instrumentation (open circle in the matrix);
- not suitable: obvious meaning, taking into account the two previous items (dash in the matrix).

The relationship test type versus facility is rated at one of three levels:

- performed: the test type is useful for code assessment purposes (plus sign in the matrix);
- performed but of limited use: this kind of test has been performed in the facility, but has limited usefulness for code assessment purposes, due to poor scaling or lack of instrumentation (open circle in the matrix);
- not performed (blank).

Based on these cross reference matrices, phenomenologically well founded sets of experiments have been defined in a second step. Criteria for the selection of these tests are listed in the following Section. These selected tests form a basis for establishing the accuracy of test calculation results comparing measured and calculated values. A total number of 177 PWR and BWR-specific integral tests have been selected as potential source for thermal hydraulic code validation.

4.2 Selection of Individual Tests

A number of specific experiments were selected from those facilities, which are included in the cross reference matrices described before. These selected tests versus phenomena establish the individual code validation matrices. During the selection process a number of factors were considered, including:

- Typicality of facility and experiment to expected reactor conditions,
- quality and completeness of experimental data (measurement and documentation),
- relevance to safety issues,
- test selected must clearly exhibit phenomena,
- each phenomenon should be addressed by tests of different scaling (at least one test if possible)
- high priority to International Standard Problems (ISP), counterpart and similar tests (for more explanations see [7]),
- challenge to system codes.

Where counterpart tests or similar tests were identified between two or more facilities, they were included in order to address questions relating to scaling and facility design compromises. For the accident management matrix, priority was given on how realistically the test represented typical accident management procedures [8].

4. CROSS REFERENCE MATRICES FOR WWER ANALYSIS (INTEGRAL AND SEPARATE EFFECTS TESTS)

A multi-national Working Group consisting of experts from Czech Republic, Finland, France, Germany, Hungary, Russia, Slovak Republic, Poland and Ukraine has been formed on the initiative of the Federal Minister for Research and Technology (BMFT) of the Federal Republic of Germany, giving the task to GRS in close co-operation with the Nuclear Protection and Safety Institute (IPSN) of France in May 1993 to elaborate the topic "Verification Matrix for Thermal-hydraulic System Codes Applied for WWER Analysis".

The topic was combined with the objective of a co-operation to formulate an internationally agreed WWER-specific validation matrix as a supplement to the existing CSNI matrix for PWRs with U-tube steam generators.

Based on the CSNI cross reference matrices the lists of phenomena have been reviewed and adopted to the characteristics of WWER-440 and WWER-1000 systems respectively, and the lists of test facilities suitable for code assessment have been completed.

The above tasks have been performed successfully by the Working Group under the leadership of GRS in close co-operation with IPSN during 1993-1995, and the results were published by Liesch and Réocreux [9].

The selection of tests from the large number of experiments proposed has to be continued, in order to get the ones which are the most suitable for code assessment with respect to a given phenomenon or test type. In order to support the selection, detailed explanations of the choices for the selected data have to be given.

As a consequence these activities will continue under the auspices of the OECD/NEA. Therefore, in June 1995 a new Support Group has been installed to continue with the further evaluation of the matrices, concentrating on three tasks:

- description of WWER-specific phenomena and safety relevance,
- optimization of the WWER-specific code validation matrices,
- development of criteria for the data bank storage of experimental data valid for the matrices.

As a result of this work, WWER validation matrix has been completed and published in 2001 [10].

5. CONCLUSIONS

A systematic study has been carried out to select experiments for thermal-hydraulic system code validation. The main experimental facilities for SETs, PWRs, BWRs and WWERs have been identified.

Matrices have been established to identify, firstly, phenomena assumed to occur in LWR plants during accident conditions and secondly, facilities and tests suitable for code validation. The matrices also permit identification of areas where further research may be justified [11], and [12]. While the activities for code validation matrices for SETs, PWRs, BWRs and WWERs are completed and the validation matrices, which are established, are ready for use by the research community [13].

A periodic updating of the matrices will be necessary to include new relevant experimental facilities and tests (e. g. investigating boron dilution or behaviour of advanced reactors) and to include improved understanding of existing data as a result of further validation.

The first volume of the SET matrix report [6a] provides cross references between test facilities and thermal-hydraulic phenomena, and lists tests classified by phenomena. As a preliminary to the classification of facilities and test data, it was necessary to identify a sufficiently complete list of relevant phenomena for LOCA and non-LOCA transient applications of PWRs and BWRs. The majority of these phenomena are also relevant to Advanced Water Cooled Reactors and to WWERs. To this end, 67 phenomena were identified for inclusion in the SET matrix. Phenomena characterization and the selection of facilities and tests for the SET matrix are included in volume I of the report [6a]. In all, about 2094 tests are included in the SET matrix.

To validate a code for a particular LWR plant application, it is recommended that the list of tests in the relevant matrix be viewed as the phenomenologically well founded set of experiments to be used for an adequate validation of a thermal hydraulic computer code. This set of data could serve as a basis for the estimation of code accuracy and quantification of code uncertainty.

The development and application of methods to quantify uncertainties in plant calculations is a major task for the future. This requires a determination of code uncertainties, which is based on a systematic code validation. The validation matrices are a necessary prerequisite to achieve such a systematic validation.

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This paper also summarizes the work performed by A. Annunziato (JRC, Ispra, Italy), J. N. Lillington (AEA Technology, Winfrith, UK), P. Marsili (ENEA, Rome, Italy), C. Renault (CEA, Cadarache, France), A. Sjöberg (Studsvik Eco &Safety AB, Nykoping, Sweden), and H. Glaeser (GRS Garching, Germany). Their contributions to the results presented in this paper are gratefully acknowledged.

NOMENCLATURE

BWR	Biling Water Reactor
CCVM	CSNI Code Validation Matrix
CRM	Cross Reference Matrix
CSNI	Committee on Safety of Nuclear Installations
ECC	Emergency Core Cooling
ISP	International Standard Problem
ITF	Integral Test Facility
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
NEA	Nuclear Energy Agency
OECD	Organization for Economic Co-operation and Development
OTSG	Once-through Steam Generator
PWR	Pressurized Water Reactor
SET	Separate Effects Test
UPTF	Upper Plenum Test Facility
WWER (VVER)	Water-cooled and Water-moderated Energy Reactor

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0	BASIC PHENOMENA	1	Evaporation due to Depressurisation
		2	Evaporation due to Heat Input
		3	Condensation due to Pressurisation
		4	Condensation due to Heat Removal
		5	Interfacial Friction in Vertical Flow
		6	Interfacial Friction in Horizontal Flow
		7	Wall to Fluid Friction
		8	Pressure Drops at Geometric Discontinuities
		9	Pressure Wave Propagation
1	CRITICAL FLOW	1	Breaks
		2	Valves
		3	Pipes
2	PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL	1	Pipes/Plena
		2	Core
		3	Downcomer
3	STRATIFICATION IN HORIZONTAL FLOW	1	Pipes
4	PHASE SEPARATION AT BRANCHES	1	Branches
5	ENTRAINMENT/DEENTRAINMENT	1	Core
		2	Upper Plenum
		3	Downcomer
		4	Steam Generator Tube
		5	Steam Generator Mixing Chamber (PWR)
		6	Hot Leg with ECCI (PWR)
6	LIQUID-VAPOUR MIXING WITH CONDENSATION	1	Core
		2	Downcomer
		3	Upper Plenum
		4	Lower Plenum
		5	Steam Generator Mixing Chamber (PWR)
		6	ECCI in Hot and Cold Leg (PWR)
7	CONDENSATION IN STRATIFIED CONDITIONS	1	Pressuriser (PWR)
		2	Steam Generator Primary Side (PWR)
		3	Steam Generator Secondary Side (PWR)
		4	Horizontal Pipes
8	SPRAY EFFECTS	1	Core (BWR)
		2	Pressuriser (PWR)
		3	Once-Through Steam Generator Secondary Side (PWR)
9	COUNTERCURRENT FLOW / COUNTERCURRENT FLOW LIMITATION	1	Upper Tie Plate
		2	Channel Inlet Orifices (BWR)
		3	Hot and Cold Leg
		4	Steam Generator Tube (PWR)
		5	Downcomer
		6	Surgeline (PWR)
10	GLOBAL MULTIDIMENSIONAL FLUID TEMPERATURE, VOID AND FLOW DISTRIBUTION	1	Upper Plenum
		2	Core
		3	Downcomer
		4	Steam Generator Secondary Side
11	HEAT TRANSFER: NATURAL OR FORCED CONVECTION	1	Core, Steam Generator, Structures
	SUBCOOLED/NUCLEATE BOILING	2	Core, Steam Generator, Structures
	DNB/DRYOUT	3	Core, Steam Generator, Strucutres
	POST CRITICAL HEAT FLUX	4	Core, Steam Generator, Strucutres
	RADIATION	5	Core
	CONDENSATION	6	Steam Generator, Structures
12	QUENCH FRONT PROPAGATION/REWET	1	Fuel Rods
		2	Channel Walls and Water Rods (BWR)
13	LOWER PLENUM FLASHING		
14	GUIDE TUBE FLASHING (BWR)		
15	ONE AND TWO PHASE IMPELLER-PUMP BEHAVIOUR		
16	ONE AND TWO PHASE JET-PUMP BEHAVIOUR (BWR)		
17	SEPARATOR BEHAVIOUR		
18	STEAM DRYER BEHAVIOUR		
19	ACCUMULATOR BEHAVIOUR		
20	LOOP SEAL FILLING AND CLEARANCE (PWR)		
21	ECC BYPASS/DOWNCOMER PENETRATION		
22	PARALLEL CHANNEL INSTABILITIES (BWR)		
23	BORON MIXING AND TRANSPORT		
24	NONCONDENSABLE GAS EFFECT (PWR)		
25	LOWER PLENUM ENTRAINMENT		

Table 1: List of Phenomena

Table 2: List of Facilities

		Info sheet	Selected in the CCVM
1	CANADA		
1.1	Elbow Flooding Rig	a	
1.2	CWIT (CANDU reactors)	a	
1.3	Pumps		
1.4	Header Test Facility (CANDU reactors)	a	
2	FINLAND		
2.1	REWET-I	a	
2.2	REWET-II	a	x
2.3			
2.4	VEERA	a	
2.5			
2.6	IVO-CCFL (air.water)	a	x
2.7	IVO-Thermal Mixing	a	x
2.8	IVO-Loop Seal Facility (Air/Water)	a	x
3	FRANCE		
3.1	MOBY-DICK	a	x
3.2	SUPER MOBY-DICK	a	x
3.3	CANON and SUPER CANON (Horiz)	a	x
3.4	VERTICAL CANON	a	
3.5			
3.6	TAPIOCA (Vertical)	a	x
3.7	Dadine (Vertical Tube, Inside)	a	x
3.8	PERICLES Rectangular	a	x
3.9	PERICLES Cylindrical	a	x
3.10	PATRICIA GV 1	a	X
3.11	PATRICIA gv 2	a	X
3.12	ERSEC Tube (Inside)	a	x
3.13	ERSEC Rod Bundle	a	x
3.14	OMEGA Tube (Inside)	a	x
3.15	OMEGA Rod Bundle	a	x
3.16	ECTHOR Loop Seal (Air/Water)	a	x
3.17	COSI	a	x
3.18	SUPER MOBY-DICK TEE	a	x
3.19	PIERO (Air/Water)	a	x
3.20	EPOPEE		
3.21	EVA	a	x
3.22	SEROPS		
3.23	BETHSY Pressuriser		
3.24	SUPER MOBY-DICK Horizontal	a	x
3.25	REBECA	a	x
3.26	ECOTRA		
4	GERMANY		
4.1	UPTF	a	x
4.2	HDR Vessel	a	x
4.3	BATTELLE PWR RS 16	a	x
4.4	BATTELLE BWR 150396	a	x
4.5	Blowdown Heat Transfer RS 37		
4.6	Het Transfer Refill/Reflod RS 36		
4.7	Steady state DNB Exp. RS 164		
4.8	Trans. Boil. Inst. Tube (Freon) RS 370	a	
4.9	Rewet RS 62/184	a	
4.10	Thermodyn. Nonequilibrium RS 77	a	x
4.11	LOCA Pump Behaviour RS 92	a	
4.12	Thermalhyd. UP-BBR 373		
4.13	Pressuriser- Valve RS 240, 347 636		
4.14	Steam/Water Disch. Flow RS 93, 397	a	
4.15			
4.16	T-Junction Test Facility (KfK)	a	x
5	ITALY		
5.1	Pressuriser (Vapore Plant) ENEA	a	x
5.2	Pressuriser Spray, TURIN	a	x
5.3	Pressuriser Flooding, CISE		
5.4	JETI-4 Fuel Channel SIET	a	x
5.5	Safety VALVE SIET	a	x
5.6	Gen 3x3 (Steam Generator), SIET	a	x
5.7	8x8 Bundle, CISE		
5.8	FREGENE (Steam Generator) ENEA		
5.9	ARAMIS (Separator) ENEA		
5.10	Jet Condensation, TURIN		
5.11	Jet Condensation, ENEA		
5.12	CHF, ENEA		
5.14	FOB Blowdown, ANSALDO		
5.15	GEST-SEP, SIET	a	x
5.16	GET-GEN (20 M W SG), SIET		
5.17	PIPER (Blowdown), PISA	a	x
5.18	JF Blowdown, ENEA		
6	JAPAN		
6.1	TPTF, JAERI	a	x
6.2	Air/Water Horiz. Flow Loop JAERI	a	
6.3	T-Break TF (Air/Water), JAERI	a	
6.4	Air/Water Rod Bundle TF, JAERI		
6.5	SG U-Tube TF, JAERI		
6.6	Single Pin Heat Transf. TF, Jaeri	a	
6.7	SRTF (Reflood), Toshiba	a	
6.8	ESTA (18 Degree Sector), Toshiba		
6.9	ESTA-KP (KWU-PWR), Toshiba		
6.10	RRTF (Refill/Reflood), Toshiba		
6.11	SHTF (Spray Heat Transf.) Toshiba		
6.12	Guide Tube CFL TF, Toshiba		
6.13	Swell Level Tests, Toshiba		
6.14	SCTF, JAERI	a	x
6.15	CCTF, JAERI	a	
6.16	HICOF (Hitachi Core and Fuel Tests		
6.17			
6.18	Hot Leg CCFL Rig, JAERI	a	
7	NETHERLANDS		
7.1	Bcn Boiloff/Reflood Tests (36 rods)	a	
7.2			
7.3	NEPTUNUS	a	x
8	SWEDEN		
8.1	GÖTA BWR ECC Tests	a	x
8.2	MARVIKEN	a	x
8.3	FRIGG/FRÖJA	a	x
8.4	120 bar Loop		
8.5	SIV		
8.6	SEPA		
9	SWITZERLAND		
9.1	NEPTUN-I (Boiloff)	a	x
9.2	NEPTUN-I and II (Reflood)	a	x
9.3	PEANUT (Reflood Inside Tube)	a	
10	UNITED KINGDOM		
10.1	ACHILLES Reflood Loop	a	x
10.2	THETIS Bundle	a	x
10.3	REFLEX Tube Reflood		
10.4	Post Dryout Ins. Tube (HP, Winfrith)	a	x
10.5	TITAN/9 MW Rigs	a	
10.6	High Pressure Rig	a	
10.7	Post Dryout Ins. Tube (LP, Harwell)	a	x
10.8	Air/Water Pipeline Fac. (Large Sc.)		
10.9	Hot Leg (Air/Water, Offt., Large Sc.)	a	
10.10			
10.11	Horiz. CCFL Rig (Air/Water, Small Sc.)	a	
10.12	Air/Water Rigs (Small Scale)		
10.13	LOTUS (Air/Water Ann. Flow in Tube)	a	x
10.14	Single Tube Level Swell (Harwell)	a	x
10.15	Single Tube Reflood (Harwell)	a	
10.16	Crossflow Two-Phase Wind Tunnel	a	
10.17	Loop Seal Air/Water Rig		
10.18	Hot Leg Co and CCF Rig		
10.19	Single tube Reflood (Leatherhead)	a	
10.20	Boiler Dynamics Rig	a	
10.21	Valve Blowdown Test Facility	a	x
10.22	Single Pin Reflood		
10.23	Multipin Cluster Rig		
10.24	Blowdown Rig		
10.25	ECCS Condensation Rig		
10.26	$1/6^{th}$ Sc. Broken Cold Leg Nozzle Rig	a	
10.27	$1/10^{th}$ Scale PWR Refill Strath Clyde		
10.28	R113 Vertical Forced Circul. Loop		
10.29	R113 Horiz. Forced Circul. Loop		
10.30	Vertical Flow Rigs		
10.31	High Press. Steam/Water Forced Circ.		

5.13	CCF, ENEA				10.32	Low Pressure Boiling Fac. (Harwell)	a	

Table 2 (Cont.): List of Facilities

		Info sheet	Selected in the CCVM
11	USA		
11.1	LTSF 1/6 Scale Jet Pump	a	x
11.2	Univ. California SB. LP BWR	a	x
11.3	THEF Post CHF Ins.Tube	a	x
11.4	Battle Columbus Laboratory		
11.5	Wyle Lab. Marshall Steam Station TF		
11.6	Micellaneous Sources		
11.7	Univ. California SB. Vert. Tube		
11.8	Univ. California B. Tube Reflood	a	x
11.9	Univ. California Berkeley		
11.10	Columbia rod Bundle	a	x
11.11	State Univ. New York at Buffalo		
11.12	State Univ. New York at Buffalo		
11.13	1/30, 1/5 + 1/5 VESSEL CREARE	a	x
11.14	1/5 DC + CL CREARE	a	
11.15	CDN DART Bubbly Flow Nozzles	a	
11.16	VERT TUBE PL/DART Annular CCF	a	x
11.17	TUBE + CHANNEL DART Air/Water		
11.18	SNTF DART BWR Spray Nozzle		
11.19	CE + MIT		
11.20	J-Loop Test Fac. Westinghouse		
11.21	HCNTL Univ. of Cincinnati		
11.22	Heat Transf. Loop Baboock and Wilcox		
11.23	FLECHT SEASET Westinghouse	a	x
11.24	Univ. California Los Angeles		
11.25	SCTF Univ. California LA	a	x
11.26	Univ. California Santa Barbara		
11.27	Univ. California Berkeley		
11.28	HST, SSTF, VSF/GE Spray Tests	a	x
11.29	Four Loop Natural Circulation/SRI		
11.30	U-Tube SG Two-Loop Test Fac/SRI	a	
11.31	1/5 EPRI-CREARE Mixing Facility		
11.32	EPRI-SAI Thermal Mixing Test Fac.	a	
11.33	½ Scale Test Facility/CREARE	a	x
11.34	EPRI-Wyle Pipe Rupture Test Fac.		
11.35	TPFL/INEL Tee Critical Flow	a	x
11.36	EPRI-SAI Carryover Large Dim.		
11.37	PHSE/PURDUE ½ Scale Facility		
11.38	Thermal Hydr. Test Fac/ORNL		
11.39	INEL Pump Charcterisation	a	x
11.40	Semiscale/INEL		
11.41	BWR-FLECHT/GE	a	x
11.42	LEHIGH Post CHF Heat Tr. Bundle	a	x
11.43	MIT Pressuriser	a	x
11.44	LS/GE Level Swell in Blowdown	a	
11.45	HOUSTON		
11.46	Cocurrent Hor, Flow/Northwest	a	x
11.47	ANL Power-Void Transf. Funct. BWR	a	x
11.48	Natural Circulation Boiling/ANL	a	
11.49	G2 Loop/Westinghouse		
11.50	Air/Water TF/B. Willamette Pump		
11.51	Univ. California Berkley		
11.52	MB-2 SG Transient/Westinghouse	a	x
11.53	Strat. Condens. Flow/Northwest	a	
11.54	Critical Flow Rig/GE	a	x
11.55	Reflux Rig/Univ. Cal. St. Barbara	a	x
11.56	LTSF Blowdown Quench/INEL	a	x
11.57	LEHIGH Post CHF Vertical Tube	a	x
12	NORWAY		
12.1	Halden Reactor, Reflood Tests	a	x

a : info sheet available in [6, volume 2] - x : selected in the SETs matrix [6, volume 1, chapter 6]

Phenomena | **Separate Effects Test Facilities**

3. France

LEGEND

x suitable for model validation

o limited suitability for model validation

- not suitable for model validation

Phenomena		MOBY-DICK	SUPER MOBY-DICK	CANON and SUPER CANON (Horiz)	VERTICAL CANON		TAPIOCA (Vertical)	DADINE (Vertical Tube, Inside)	PERICLES Rectangular	PERICLES Cylindrical	PATRICIA GV 1	PATRICIA GV 2	ERSEC Tube (Inside)	ERSEC Rod Bundle	OMEGA Tube (Inside)	OMEGA Rod Bundle	ECTHOR Loop Seal (Air/Water)	COSI	SUPER MOBY-DICK TEE	PIERO (Air/Water)	EPOPEE	EVA
	Facility No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
	Info Sheet available	x	x	x	x		x	x	x	x	x	x	x	x	x	x	x	x	x	x		x
0 BASIC PHENOMENA	1 Evaporation due to Depressurisation	x	x	x	x		x	-	-	-	-	-	-	-	o	o	-	-	-	-	-	-
	2 Evaporation due to Heat Input	-	-	-	-		-	o	-	-	-	-	o	o	o	o	-	-	-	-	-	-
	3 Condensation due to Pressurisation	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	4 Condensation due to Heat Removal	-	-	-	-		-	-	-	-	o	-	-	-	-	-	-	o	-	-	-	-
	5 Interfac. Frict. Vertic. Flow	-	-	-	o		o	x	-	-	-	o	-	-	o	o	o	-	-	-	-	-
	6 Interfac. Frict. Horiz. Flow	-	-	o	-		-	-	-	-	-	-	-	-	-	-	o	o	-	-	-	-
	7 Wall to Fluid Friction	x	x	x	-		-	-	-	-	-	-	-	-	-	-	-	-	x	-	-	-
	8 Press. Drops at Geometr. Discontinuities	-	x	-	o		o	-	-	-	-	-	-	-	-	-	-	-	x	-	-	-
	9 Pressure Wave Propagation																					
1 CRITICAL FLOW	1 Breaks	-	x	x	x		x	-	-	-	-	-	-	-	x	x	-	-	x	-	-	-
	2 Valves	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	3 Pipes	x	x	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2 PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL	1 Pipes/Plena	-	-	-	x		x	o	-	-	o	x	-	-	-	-	-	-	-	o	-	-
	2 Core	-	-	-	-		-	-	x	x	-	-	-	-	-	-	-	-	-	-	-	-
	3 Downcomer	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
3 STRATIFICATION IN HORIZ. FLOW	1 Pipes	-	-	-	-		-	-	-	-	-	-	-	-	-	-	o	o	x	-	-	-
4 PHASE SEPARATION AT BRANCHES	1 Branches	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	x	-	-	-
5 ENTRAINMENT/DEENTRAINMENT	1 Core	-	-	-	-		-	o	x	x	-	-	-	x	-	x	-	-	-	-	-	-
	2 Upper Plenum	-	-	-	-		-	-	o	x	-	-	-	-	-	-	-	-	-	-	-	-
	3 Downcomer	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	4 SG-Tube	-	-	-	-		-	-	-	-	x	-	-	-	-	-	-	-	-	-	-	-
	5 SG-Mix. Chamber (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	6 Hot Leg with ECCI (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
6 LIQUID-VAPOUR MIXING WITH CONDENSATION	1 Core	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	2 Downcomer	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	3 Upper Plenum	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	4 Lower Plenum	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	5 SG-Mix. Chamb. (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	6 ECCI in Hot and Cold Leg (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	x	-	-	-	-
7 CONDENSATION IN STRATIFIED CONDITIONS	1 Pressuriser (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	2 SG-Primary Side (PWR)	-	-	-	-		-	-	-	-	x	-	-	-	-	-	-	-	-	-	-	-
	3 SG-Secondary Side (PWR)	-	-	-	-		-	-	-	-	-	x	-	-	-	-	-	-	-	-	-	-
	4 Horizontal Pipes	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	x	-	-	-	-
8 SPRAY EFFECTS	1 Core (BWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	2 Pressuriser (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	3 OTSG Second. Side (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
9 CCF/CCFL	1 Upper Tie Plate	-	-	-	-		-	-	o	o	-	-	-	-	-	-	-	-	-	-	-	-
	2 Channel Inlet Orifices (BWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	3 Hot and Cold Leg	-	-	-	-		-	-	-	-	-	-	-	-	-	-	o	-	-	-	-	-
	4 SG-Tube (PWR)	-	-	-	-		-	-	-	-	x	-	-	-	-	-	-	-	-	-	-	-
	5 Downcomer	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	6 Surgeline (PWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
10 GLOBAL MULTIDIMENSIONAL FLUID TEMPERATURE, VOID AND FLOW DISTRIBUTION	1 Upper Plenum	-	-	-	-		-	-	o	-	-	-	-	-	-	-	-	-	-	-	-	-
	2 Core	-	-	-	-		-	-	x	-	-	-	-	-	-	-	-	-	-	-	-	-
	3 Downcomer	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
	4 SG-Secondary Side	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
11 HEAT TRANSF.: NAT. FORC. CONV.	1 Core, SG, Structures	-	-	-	-		-	x	-	-	-	-	-	-	-	-	-	-	-	-	-	-
SUBC./NUCL. BOIL.	2 Core, SG, Structures	-	-	-	-		-	x	-	-	-	-	-	-	-	-	-	-	-	-	-	-
DNB/DRYOUT	3 Core, SG, Structures	-	-	-	-		-	x	-	-	-	x	x	-	x	x	-	-	-	-	-	-
POST CHF	4 Core, SG, Structures	-	-	-	-		-	x	-	-	-	-	x	-	x	x	-	-	-	-	-	-
RADIATION	5 Core	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
CONDENSATION	6 SG, Structures	-	-	-	-		-	-	-	-	x	-	-	-	-	-	-	-	-	-	-	-
12 QUENCH FRONT PROPAG./REWET	1 Fuel Rods	-	-	-	-		-	-	x	x	-	-	o	x	-	-	-	-	-	-	-	-
	2 Channel Walls and Water Rods (BWR)	-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
13 LOWER PLENUM FLASHING		-	-	-	-		o	-	-	-	-	-	-	-	x	x	-	-	-	-	-	-
14 GUIDE TUBE FLASHING (BWR)		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
15 ONE AND TWO PHASE IMPELLER-PUMP BEHAVIOUR		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	x	x
16 ONE AND TWO PHASE JET-PUMP BEHAVIOUR (BWR)		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
17 SEPARATOR BEHAVIOUR		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
18 STEAM DRYER BEHAVIOUR		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
19 ACCUMULATOR BEHAVIOUR		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
20 LOOP SEAL FILLING AND CLEARANCE (PWR)		-	-	-	-		-	-	-	-	-	-	-	-	-	-	x	-	-	-	-	-
21 ECC BYPASS/DC PENETRATION		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
22 PARALLEL CHANNEL INSTABILITIES (BWR)		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
23 BORON MIXING AND TRANSPORT		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
24 NONCONDENSABLE GAS EFFECT (PWR)		-	-	-	-		-	-	-	-	x	-	-	-	-	-	-	x	-	-	-	-
25 LOWER PLENUM ENTRAINMENT		-	-	-	-		-	-	-	-	-	-	-	-	-	-	-	-	-	x	-	-

Table 3: Separate Effects Test Facility Cross Reference Matrix

FACILITY IDENTIFICATION			KEYWORDS	RELEVANT PARAMETERS RANGES			REASONS FOR SELECTION OR NOTES
No.	Status in the matrix	Name		Pressure (MPa)	Inlet mass flow ($kg/m^2/s$)	Heat flux (W/cm^2)	
3.7	a x	DADINE (VERTICAL TUBE INSIDE)	Vertical tube, Steady-state, Boil-off	0.1-0.6	20-150	1-3	
3.12	a x	ERSEC TUBE (INSIDE)	Tube, reflooding	0.1-0.6	10-120	1-7	1 5 6
3.14	a x	OMEGA TUBE (INSIDE)	Blowdown	16	–	60-125	5 6 7
3.15	a x	OMEGA ROD BUNDLE	Blowdown	13-15	–	44-60	5 6 7
4.5	a x	BLOWDOWN HEAT TRANSFER RS 37	Blowdown Rod bundle	15-1.3	3828-3300	163-74	5 6 7
4.9	a x	REWET (RS 62/184)	Reflooding, tube, single rod	0.1-0.45	2-10 cm/s	2-6	5 6
5.6	a x	GEN 3x3 (STEAM GENERATOR) ENEA	SG Secondary, Steady-state, transient	3.5-8	200-600	–	
5.7	a x	8x8 BUNDLE CISE	BWR-6 Bundle, Steady state	7.1	125-1600	–	6 7
5.12	x	CHF ENEA					
6.1	a x	TPTF JAERI	Core heat transfer, Boil-off, Reflooding, BWR and PWR bundle	0.5-12	20-410	3-25	2 3 5 6
6.16	x	HICOF (HITACHI CORE AND FUEL TESTS)					
8.4	x	120 BAR LOOP					
9.1	a x	NEPTUN-I (BOIL-OFF)	Bundle	0.15	–	25-75 kW	2 3 5 6
10.3	x	REFLEX TUBE REFLOOD					
10.4	a x	POST DRYOUT INST. TUBE (HP, WINFRITH)	Hot patch	0.2-7	50-2000	1-30	2 3 5 6
10.7	a x	POST DRYOUT INST. TUBE (LP, HARWELL)		0.2-0.4	25-200		2 3 5 6
10.20	a x	BOILER DYNAMICS RIG	SG, transient boundary conditions	28	12 kg/s	12 MW	6 7
10.23	x	MULTIPIN CLUSTER RIG					
11.3	a x	THEF POST CHF INS. TUBE	Steady state, quasi-steady state	0.2-7	12-70	0.8-22.5	2 3 4 5 6
11.7	x	UNIV. CALIFORNIA B. TUBE REFLOOD					
11.8	a x	UNIV. CALIFORNIA B. TUBE REFLOOD	Reflooding	0.1-0.3	2.5-18 cm/s		1 5 6

Table 4: Phenomenon No. 11.4 - Heat Transfer: POST-CHF in the Core, in the Steam Generator and at Structures (Part A)

FACILITIES IDENTIFIER		11.56	11.57	12.1		
Main parameters						
P (MPa)	Inlet fluid velocity (m/s)					
6.86	3.7	12				
6.92	0.4	7				
	Mass Flux (kg/m^2s)					
0.378	14.8		100			
0.255	14.9		105			
0.409	20.7		112			
0.396	42.7		124			
0.39	29.5		130			
0.272	42.9		158			
0.302	60		174			
0.395	29.9		191			
	Reflood rate (cm/s)					
				IFA-511-2		
0.2-0.4	9.6			5236		
	5.6			5239		
	7.4			5247		
				IFA-511-3		
	9.6			5258		
	5.6			5261		
	7.4			5265		
	2.1			5266		
		SELECTED TESTS				

References:

11.56 N. Aksan: "Evaluation of Analytical Capability to predict cladding Quench" EGG-LOFT 5555, August 1982.

11.57 D.G. Evans, et al. "Measurement of Axially Varying Nonequilibrium in Post-Critical Heat-Flux Boiling in a Vertical Tube" NUREG/CR-3363, Vols. 1 and 2, June 1983.

12.1 C. Vitanza et al.:"Blowdown/reflood tests with Nuclear Heated Rods (IFA-511.2)" OECD Halden Reactor Project, HPR-248, May 1980.

T. Johnsen, C. Vitanza: "Blowdown/Reflood Tests with Semiscale Heaters (IFA-511.3)" OECD Halden Reactor Project HWR-17, May 1981.

Table 5: Heat Transfer: Post-CHF in the Core, in the Steam Generator and at Structures (7/7)

Matrix I
CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

Phenomena versus test type
+ occurring
o partially occurring
- not occurring

- test facility versus phenomenon
+ suitable for code assessment
o limited suitability
- not suitable

- test type versus test facility
+ performed
o performed but of limited use
- not performed or planned

		Test Type			Test Facility and Volumetric Scaling						
		Blowdown	Refill	Reflood	CCTF 1:25	LOFT 1 : 50	BETHSY 1 : 100	PKL 1 : 145	LOBI 1 : 712	SEMISCALE 1 : 1600	UPTF 1 : 1 (a)
Phenomena	Break flow	+	+	+	o	o	o	o	o	o	o
	Phase separation (condition or transition)	o	+	+	+	+	+	+	+	+	+
	Mixing and condensation during injection	o	+	+	o	o	o	o	o	o	+
	Core wide void + flow distribution	o	+	+	o	o	o	o	o	-	o
	ECC bypass and penetration	o	+	o	+	+	-	o	o	-	+
	CCFL (UCSP)	o	+	+	o	o	o	o	o	-	+
	Steam binding (liquid carry over, ect.)	-	o	+	o	o	-	o	o	o	o
	Pool formation in UP	-	+	+	o	o	o	o	o	o	+
	Core heat transfer incl. DNB, dryout, RNB	+	+	+	o	+	+	+	o	o	-
	Quench front propagation	o	o	+	+	+	+	+	-	+	-
	Entrainment (Core, UP)	o	o	+	o	o	+	o	o	o	+
	Deentrainment (Core, UP)	o	o	+	o	o	o	o	o	o	+
	1 - and 2-phase pump behaviour	+	o	o	-	o	-	o	+	+	-
	Noncondensable gas effects	-	o	o	-	-	o	-	-	-	o
Test Facility	CCTF	-	o	+							
	LOFT	+	+	+							
	BETHSY	-	-	+							
	PKL	o	+	+							
	LOBI	+	+	-							
	SEMISCALE	+	+	+							
	UPTF	o	+	+							

important test parameter
- break location/break size
- pumps off/pumps on
- cold leg injection/combined injection

(a) UPTF integral tests

Table 6: Cross Reference Matrix for Large Breaks in PWRs

	Matrix II **CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE BREAKS**	**Test Type**							**Test Facility and Volumetric Scaling**								
	Phenomenon versus test type + occurring o partially occurring - not occurring **- Test facility versus phenomenon** + suitable for code assessment o limited suitability - not suitable **- Test type versus test facility** + performed o performed but of limited use - not performed or planned	Stationary test addressing energy transport on primary side	Stationary test addressing energy transport on secondary side	Small leak overfeed by HPIS, secondary side necessary	Small leak without HPIS overfeeding, secondary side necessary	Intermediate leak, secondary side not necessary	Pressurizer leak	U-tube rupture	PWR 1 : 1	LOFT 1 : 50	LSTF 1 : 50	BETHSY 1 : 100	PKL-III 1 : 145	SPES 1 : 430	LOBI-II 1 : 712	SEMISCALE 1 : 1600	UPTF, TRAM 1 : 1 (2)
Phenomena (3)	Natural circulation in 1-phase flow, primary side	+	+	+	o	-	+	+	+	+	+	+	+	+	+	+	-
	Natural circulation in 2-phase flow, primary side	+	-	o	+	+	o	-	-	+	+	+	+	+	+	+	o
	Reflux condenser mode and CCFL	+	-	-	+	+	-	-	-	o	+	+	o	o	o	o	+
	Asymmetric loop behaviour	-	-	+	+	-	o	+	-	-	o	+	+	+	o	o	+
	Break flow	-	-	+	+	+	+	+	-	+	+	+	+	+	+	+	o
	Phase separation without mixture level formation	+	-	o	+	+	+	o	-	o	+	+	+	+	+	o	+
	Mixture level and entraiment in SG second side	-	+	+	+	+	+	+	-	-	+	+	+	o	o	-	-
	Mixture level and entraiment in the core	+	-	-	+	+	+	-	-	o	+	+	+	o	o	o	o
	Stratification in horizontal pipes	+	-	-	+	+	-	-	-	+	+	o	o	+	o	o	+
	Phase separation in T-junct. and effect on breakflow	-	-	-	+	+	-	-	-	o	o	o	o	o	o	-	+
	ECC-mixing and condensation	-	-	o	+	+	+	+	-	o	o	o	o	o	o	o	+
	Loop seal clearing	-	-	-	+	+	o	-	-	+	+	+	+	+	+	+	+
	Pool formation in UP/CCFL (UCSP)	+	-	-	o	+	+	-	-	o	o	o	o	o	-	o	+
	Core wide void and flow distribution	+	-	-	o	+	+	-	-	o	o	o	o	-	-	-	o
	Heat transfer in covered core	+	+	+	+	+	+	+	o	+	+	+	+	+	+	+	-
	Heat transfer in partly uncovered core	+	-	-	o	+	-	-	-	+	+	+	+	o	o	o	-
	Heat transfer in SG primary side	+	o	o	+	+	o	o	-	o	+	+	+	+	+	o	-
	Heat transfer in SG secondary side	o	+	+	+	+	+	+	-	o	+	+	+	o	+	o	-
	Pressurizer thermohydraulics	o	-	o	o	+	+	+	o	o	o	o	o	o	o	-	+
	Surgeline hydraulics	o	-	-	o	+	+	o	-	o	o	o	o	o	o	o	+
	1- and 2-phase pump behaviour	-	-	-	o	+	-	-	o	o	o	o	o	o	+	+	-
	Structural heat and heat losses (1)	+	-	o	+	+	o	o	-	o	o	o	o	o	o	o	o
	Noncondensable gas effects	+	-	-	-	-	-	-	-	-	o	o	o	-	-	o	+
	Boron mixing and transport	+	-	+	+	+	+	+	-	-	-	-	-	-	-	-	o
Test Facility	PWR	-	-	o	-	-	+	+									
	LOFT	-	-	+	+	+	+	-									
	LSTF	+	+	+	+	+	+	+									
	BETHSY	+	+	+	+	+	+	+									
	PKL-III	+	+	+	+	+	+	+									
	SPES	+	+	+	+	-	-	-									
	LOBI-II	+	+	+	+	+	+	+									
	SEMISCALE	o	o	+	+	+	+	+									
	UPTF, TRAM	-	-	-	-	+	+	-									

(1) problem for scaled test facilities
(2) UPTF integral tests
(3) for intermediate breaks phenomena included in large break reference matrix may be also important

Table 7: Cross Reference Matrix for Small and Intermediate Breaks in PWRs

Matrix IV
CROSS REFERENCE MATRIX FOR TRANSIENTS IN PWRs

Phenomenon versus test type
- \+ occurring
- o partially occurring

- test facility versus Phenomenon
- \+ suitable for code assessment
- o limited suitability
- \- not suitable

- test type versus test facility
- \+ performed
- o performed but of limited use
- \- not performed or planned

		Test Type								Test Facility and Volumetric Scaling							
		ATWS	Loss of feedwater, non ATWS	Loss of heat sink, non ATWS (c)	Station blackout	Steam line break	Feed line break	Reactivity disturbance	Over-cooling	PWR 1 : 1	LOFT 1 : 50	LSTF 1 : 50	BETHSY 1 : 100	PKL-III 1 : 134	SPES 1 : 430	LOBI-II 1 : 712	SEMISCALE 1 : 1000
Phenomena	Natural circulation in 1-phase flow	+	+	+	+	+	+	o	o	+	o	+	+	+	+	+	+
	Natural circulation in 2-phase flow	+	+	+	+	-	-	o	-	-	o	+	+	+	+	+	+
	Core thermohydraulics	+	+	+	+	o	o	+	o	o	+	+	+	+	+	+	+
	Thermohydraulics on primary side of SG	+	o	o	+	o	o	o	+	o	o	+	+	+	+	+	o
	Thermohydraulics on secondary side of SG	+	+	+	+	+	+	o	+	o	o	+	+	+	o	+	o
	Pressurizer thermohydraulics	+	+	+	+	o	o	o	+	o	o	o	o	o	o	o	o
	Surgeline hydraulics (CCFL, choking)	+	+	+	+	o	o	o	o	o	o	o	o	o	o	o	o
	Valve leak flow (a)	+	+	+	+	+	+	+	+	-	o	o	o	o	o	o	o
	1- and 2-phase pump behaviour	+	+	+	+	o	o	o	+	o	o	+	o	o	o	+	+
	Thermohydraulic-nuclear feedback	+	-	-	-	-	-	+	-	+	+	-	-	-	-	-	-
	Structural heat and heat losses (b)	o	o	o	o	o	o	o	o	-	o	o	o	o	o	o	o
	Boron mixing and transport	-	-	-	-	o	-	-	o	-	-	-	-	-	-	-	-
	Separator behaviour	o	-	-	-	+	-	-	-	-	-	-	-	-	o	o	-
Test Facility	PWR	-	-	-	-	-	-	-	o								
	LOFT	+	+	+	o	-	-	+	+								
	LSTF	-	+	-	+	+	+	-	+								
	BETHSY	-	+	+	-	+	+	-	-								
	PKL-III	-	+	+	+	+	+	-	+								
	SPES	-	+	-	+	-	-	-	-								
	LOBI-II	+	+	+	+	+	+	-	-								
	SEMISCALE	-	+	+	+	+	+	-	+								

(a) valve flow behaviour will be strongly design-dependent, specific experimental data should be used if possible
(b) problem for scaled test facilities
(b) problem for scaled test facilities

Table 8: Cross Reference Matrix for Transients in PWRs

Matrix V **CROSS REFERENCE MATRIX FOR TRANSIENTS AT SHUT-DOWN CONDITIONS IN PWRS**		**Test Type**				**Test Facility and Volumetric Scaling**		
Phenomenon versus test type + occurring o partially occurring - not occurring - Test facility versus phenomenon + suitable for code assessment o limited suitability - not suitable - Test type versus test facility + performed o performed but of limited use - not performed or planned		Loss of RHR with no opening	Loss of RHR with openings	Loss of RHR with dam in HL	Boron dilution at shut-down	LSTF	BETHSY	PKL III
Phenomena	Pressurization due to boiling	+	+	+	-	+	+	+
	Reflux condenser mode and CCFL	+	+	o	-	+	+	o
	Asymmetric loop behaviour	-	o	+	-	+	+	+
	Flow through openings (manways, vents)	-	+	+	-	+	+	-
	Mixture level formation in upper plenum and hot legs	+	+	+	-	+	+	+
	Mixture level and entrainment in the core	+	+	+	-	+	+	+
	SG syphon draining	-	-	+	-	+	-	-
	Asymmetry due to the presence of a dam	-	-	+	-	+	-	-
	Stratification in horizontal pipes	+	+	+	-	+	o	+
	Phase separation in T-junctions and effect on flow	-	+	+	-	o	o	o
	ECC mixing and condensation	+	+	+	-	o	o	o
	Loop seal clearing and filling	+	+	+	-	+	+	-
	Pool formation in UP/CCFL (UCSP)	-	-	-	-	-	-	-
	Core 3D thermalhydraulics	+	+	+	+	o	o	o
	Heat transfer in covered core	+	+	+	-	+	+	+
	Heat transfer in partially uncovered core	+	+	+	-	o	o	-
	Heat transfer in SG primary side	+	+	+	-	+	+	+
	Heat transfer in SG secondary side	+	+	+	-	+	+	+
	Pressurizer thermalhydraulics a)	-	x	x	-	o	o	o
	Surge line thermalhydraulics a)	-	x	x	-	o	o	o
	Structural heat and heat losses	-	-	-	-	-	-	o
	Non-condensible gas effects	+	+	+	-	+	+	+
	Boron mixing and transport	-	-	-	+	-	-	-
	Thermalhydraulics-nuclear feedback	-	-	-	+	-	-	-
Test Facility	LSTF	+	+	+	-			
	BETHSY	-	+	-	-			
	PKL III	+	-	-	-			

a) x is dependent on opening location
+ pressuriser manway open
- pressuriser manway shut

Table 9: Cross Reference Matrix for Transients at shut-down conditions in PWRs

Matrix VI **CROSS REFERENCE MATRIX FOR ACCIDENT MANAGEMENT FOR A NON DEGRADED CORE IN PWRs**		**Test Type**					**Test Facility and Volumetric Scaling**						
- Phenomenon versus test type + occurring o partially occurring - not occurring - Test facility versus phenomenon + suitable for code assessment o limited suitability - not suitable - Test type versus test facility + performed o performd but of limited use - not performed or planned		High pressure primary side feed and bleed	Low pressure, primary side feed and bleed	Secondary side, feed and bleed	RCP-Restart in a highly, voided PCS	Primary to secondary break with multiple failures	LOFT 1 : 50	LSTF 1 : 50	BETHSY 1 : 100	PKL-III 1 : 1 : 145	SPES 1 : 430	LOBI-II 1 : 712	UPTF, TRAM 1 : 1 (2)
Phenomena	Natural circulation in 1-phase flow, primary side	+	-	+	-	+	+	+	+	+	+	+	-
	Natural circulation in 2-phase flow, primary side	+	+	+	-	+	+	+	+	+	+	+	o
	Reflux condenser mode and CCFL	-	-	+	-	+	o	+	o	o	o	o	+
	Asymmetric loop behaviour	+	+	+	+	+	-	o	+	+	+	o	+
	Break flow	+	+	o	+	+	+	+	+	+	o	+	o
	Phase separation without mixture level formation	+	+	+	+	+	o	+	+	+	+	+	+
	Mixture level and entraiment in SG secondary side	-	-	+	-	+	-	+	+	+	o	o	-
	Mixture level and entraiment in the core	+	+	+	o	+	o	+	+	+	o	o	o
	Stratification in horizontal pipes	+	+	+	o	+	+	+	o	o	o	o	+
	Phase separation in T-junct. and effect on breakflow	+	+	o	-	+	o	o	o	o	o	o	+
	ECC-mixing and condensation	+	+	+	-	+	o	o	o	o	o	o	+
	Loop seal clearing (3)	o	o	+	o	+	+	+	o	o	+	+	+
	Pool formation in UP/CCFL (UCSP)	+	+	+	-	+	o	o	o	o	o	-	+
	Core wide void and flow distribution	+	+	+	+	+	o	o	o	o	-	-	o
	Heat transfer in covered core	o	o	+	-	+	+	+	+	+	+	+	-
	Heat transfer in partly uncovered core	+	+	+	+	+	+	+	+	+	o	o	-
	Heat transfer in SG primary side	-	-	+	o	+	o	+	+	+	+	+	-
	Heat transfer in SG secondary side	-	-	+	o	+	o	+	+	+	o	+	-
	Pressurizer thermohydraulics	+	+	o	o	+	o	o	o	o	o	o	+
	Surgeline hydraulics	+	+	o	o	+	o	o	o	o	o	o	+
	1- and 2-phase pump behaviour	o	o	+	+	+	o	o	o	o	o	+	-
	Structural heat and heat losses (1)	+	+	+	+	+	o	o	o	o	o	o	o
	Noncondensable gas effects	o	+	+	+	+	-	o	o	+	-	-	+
	Accumulator behaviour	-	+	+	-	o	o	+	+	+	+	+	+
	Boron mixing and transport	+	+	+	+	+	-	-	-	-	-	-	o
	Thermohydraulic-nuclear feed back	-	-	-	+	-	-	-	-	-	-	-	-
	Separator behaviour	-	-	-	-	-	-	-	-	-	-	-	-
Test Facility	LOFT	-	-	+	-	-	(1) problem for scaled test facilities						
	LSTF	+	+	+	-	o	(2) UPTF integral tests						
	BETHSY	+	+	+	-	+	(3) long term cooling not included						
	PKL-III	o	+	+	+	-							
	SPES	+	+	+	-	+							
	LOBI-II	+	+	+	-	+							
	UPTF, TRAM	o	+	-	-	-							

Table 10: Cross Reference Matrix for Accident Management for non-degraded core in PWRs

Matrix VII **CROSS REFERENCE MATRIX FOR LOCA IN BWRs**		**Test Type**						**Test Facility and Volumetric Scaling**						
- Phenomena versus test type + occurring o partially occurring - not occurring - Test facility versus phenomenon + suitable for code assessment o limited suitability - not suitable - Test type versus test facility + performed o performed but of limited use - not performed or planned		Large Steam Line Break with Fast Depressurization	Large Break Below Water Level with Fast Depress.	Small Break without Depress. before ADS Actuation	Intermediate Break with Slow Depress.	Spray Line Break	Refill - Reflood	BWR 1 : 1 (a)	TBL, 1 : 382, 2 Chan., Full Pow., Full Height	ROSA III, 1 : 424, 4 Channels	TLTA, 1 : 624, 1 Chan., Full Power	FIST, 1 : 624, 1 Chan., Full Pow., Full Height	FIX 2, 1 : 777, 1 Chan., Full Pow., Full Height	PIPER 1, 1 : 2200, 1 Chan., Full Height
Phenomena	Break flow	+	+	+	+	+	o	-	o	o	o	o	o	+
	Channel and Bypass Axial Flow and Void Distribution	+	+	+	+	+	+	o	+	o	+	+	+	+
	Corewide Radial Void Distribution	o	o	+	+	+	+	o	o	+	o	o	o	-
	Parallel Channel Effects-Instabilities	-	-	+	+	+	+	-	o	+	-	-	-	o
	ECC Bypass	-	-	o	o	o	+	-	o	o	o	o	-	+
	CCFL at UCSP and Channel Inlet Orifice	o	+	-	+	+	+	-	o	o	-	o	o	o
	Core Heat Transf. incl. DNB, Dryout, RNB. Surf. to Surf Radiation	+	+	o	+	o	+	-	+	+	+	+	+	+
	Quench Front Propagation for both Fuel Rods and Channel Walls	-	-	-	-	-	+	-	+	+	+	+	-	+
	Entrainment and Deentainment in Core and Upper Plenum	+	+	o	o	o	+	-	-	o	o	o	-	o
	Separator Behavior incl. Flooding, Steam Penetration and Carryover	+	+	o	o	o	-	o	+	o	o	+	o	o
	Spray Cooling	-	-	o	o	o	+	-	o	o	o	o	-	+
	Spray Distribution	-	-	o	o	o	+	-	-	o	-	-	-	-
	Steam Dryer - Hydraulic Behavior	+	-	o	o	-	-	o	o	o	o	o	-	o
	One and Two Phase Pump Recirc. Behavior incl. Jet Pumps	o	o	+	+	+	o	o	o	o	o	o	o	-
	Phase Separation and Mixture Level Behavior	+	+	+	+	+	+	-	o	+	o	+	+	o
	Guide Tube and Lower Plenum Flashing	+	+	-	o	o	-	-	+	+	+	+	+	+
	Natural Circulation- Core and Downcomer	-	-	+	o	o	+	+	+	o	o	+	+	+
	Natural Circulation Core Bypass, Hot and Cold Bundles	-	-	+	o	o	+	-	o	o	o	o	o	o
	Mixture Level in Core	-	-	+	o	o	+	-	+	+	+	+	+	o
	Mixture Level in Downcomer	+	+	+	+	+	+	-	+	o	o	+	+	o
	ECC Mixing and Condensation	-	-	+	o	+	+	-	o	o	o	o	-	o
	Pool Formation in Upper Plenum	o	o	-	o	o	+	-	o	o	o	o	o	o
	Structural Heat and Heat Losses	o	o	o	+	+	+	-	+	o	o	o	o	o
	Phase Separ. in T - Junction and Effect on Break Flow	-	-	+	o	+	-	-	-	-	-	-	-	+
Test Facility	BWR	-	-	-	-	-	-	(a) These are non-LOCA data but may be used for assessment						
	TBL	+	+	+	+	-	+							
	ROSA III	+	+	+	+	-	+							
	TLTA	+	+	-	+	-	+							
	FIST	+	+	+	+	-	+							
	FIX 2	-	+	-	+	-	-							
	PIPER 1	-	+	+	+	-	+							

Table 11: Cross Reference Matrix for LOCA in BWRs

Table 12: Cross Reference Matrix for Transients in BWRs

Matrix VIII
CROSS REFERENCE MATRIX FOR TRANSIENTS IN BWRs

Phenomenon versus test type
- \+ occurring
- o partially occurring
- \- not occurring

Test facility versus phenomenon
- \+ suitable for code assessment
- o limited suitability
- \- not suitable

Test type versus test facility
- \+ performed
- o performed but of limited use
- \- not performed or planned

		Test Type									Test Facility and Volumetric Scaling			
		Stationary Test Measuring Power Flow Map	Recirculation Pump Trip	Core Stability	Loss of Main Heat Sink	Feedwater Flow or Temperature Disturbance e.g. LOFW	Loss of Feedwater (LOFW) up to time of Const. Pressure	Inadvertent Increase in Steam Flow	ATWS	Station Blackout (Loss-of-Offsite Power)	BWR 1 : 1	ROSA III, 1 : 424, 4 Channels	FIST, 1 : 642, 1 Channel, Full Power, Full Height	FIX 2, 1 : 777, 1 Channel, Full Power, Full Height
Phenomena	Natural Circulation in One- and Two-Phase Flow	+	+	+	+	-	-	-	+	+	+	o	+	o
	Collapsed Level Behaviour in Downcomer	-	+	o	+	+	+	+	+	+	+	o	+	+
	Core Thermal Hydraulics	o	+	+	+	o	o	o	+	+	o	+	+	+
	Valve Leak Flow	-	-	-	+	-	-	-	+	+	o	o	o	-
	Single Phase Pump Behaviour (a)	o	+	o	+	o	o	+	+	+	o	o	o	o
	Parallel Channel Effects and Instabilities	-	+	+	o	-	-	-	+	+	o	+	-	-
	Nuclear Thermalhydraulic Feedback Including Spatial Effects	o	o	+	-	o	o	o	+	-	+	-	-	-
	Nuclear Thermalhydraulic Instabilities	-	o	+	-	-	-	o	+	-	+	-	-	-
	Downcomer Mixing	-	-	-	-	+	+	-	+	+	o	o	-	-
	Boron Mixing and Distribution	-	-	-	-	-	-	-	+	-	-	-	-	-
	Steam Line Dynamics	-	-	-	+	-	-	+	+	+	o	-	o	-
	Void Collapse and Temp. Distribution During Pressurization	-	-	-	+	-	-	-	+	+	o	+	+	+
	Critical Power Ratio	-	+	+	+	+	+	+	+	+	o	+	+	+
	Rewet after DNB at High Press. and High Power Incl. High Core Flow	-	+	-	+	-	-	o	+	o	-	o	+	+
	Structural Heat and Heat Losses	-	o	-	o	-	o	o	o	o	-	o	o	o
Test Facility	BWR	+	+	+	+	+	+	+	-	o				
	ROSA III	-	+	+	+	-	+	-	-	+				
	FIST	-	o	-	+	-	+	+	o	+				
	FIX 2	-	+	-	+	-	-	-	-	-				

(a) Two-phase pump behaviour is of interest for certain special ATWS and inadvertent increase of steam flow transients