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CSNI Report 132

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CSNI CODE VALIDATION MATRIX OF THERMO-HYDRAULIC CODES FOR LWR LOCA AND TRANSIENTS

Prepared by a Committee of the
TASK GROUP ON THE STATUS AND ASSESSMENT
OF CODES FOR TRANSIENTS AND ECCS
OF
PRINCIPAL WORKING GROUP No 2 ON
TRANSIENTS AND BREAKS

MARCH 1987

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
OECD NUCLEAR ENERGY AGENCY
38, boulevard Suchet, 75016 Paris, France

CSNI CODE VALIDATION MATRIX FOR THE ASSESSMENT OF THERMAL-HYDRAULIC CODES FOR LWR LOCA AND TRANSIENTS

Prepared by a Committee of the PWG 2

"Task Group on Status and Assessment of Codes for Transients and ECCS"

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Nuclear Energy Agency

Organisation for Economic Co-operation and Development

Pursuant to article 1 of the Convention signed in Paris on 14th December, 1960, and which came into force on 30th September, 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April, 1964), Finland (28th January, 1969), Australia (7th June, 1971) and New Zealand (29th May, 1973).

The Socialist Federal Republic of Yugoslavia takes part in some of the work of the OECD (agreement of 28th October, 1961).

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA) on the adhesion of Japan as a full Member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan, and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

- *encouraging harmonisation of governements' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear fuel cycle and the potential future contribution of nuclear power to overall energy demand;*
- *developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;*
- *setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.*

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

CSNI

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member countries, for example by improving the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is reinforced by the creating of co-operative (international) research projects, such as PISC and LOFT, and by a novel form of collaboration known as the international standard problem exercise, for testing the performance of computer codes, test methods, etc. used in safety assessments. These exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The principal areas covered are operating experience and the human factor, reactor system response during abnormal transients, various aspects of primary circuit integrity, the phenomenology of radioactive releases in reactor accidents, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

The Sub-Committee on Licensing, consisting of the CSNI Delegates who have responsibilities for the licensing of nuclear installations, examines a variety of nuclear regulatory problems and provides a forum for the review of regulatory questions, the aim being to develop consensus positions in specific areas.

FOREWORD

The "Task Group on Status and Assessment of Codes for Transients and ECCS" was given the mandate from the CSNI Principal Working Group No. 2 (PWG-2) to formulate a validation matrix for the assessment of large thermal-hydraulic computer codes. Three committees of the Task Group were set up to achieve this objective, basing their work on discussions held during the Task Group meetings and on a document prepared by K. Wolfert and W. Frisch(10.1). The first committee prepared a code validation matrix for PWRs using U-tube steam generators (UTSG). The resulting matrix was generally approved at the Task Group meeting of May 1984. The second committee was formed to extend the PWR matrix to also be applicable to PWRs with once-through steam generators (OTSG). The PWR matrix for both PWRs with UTSG and OTSG was approved in the Task Group meeting of 16th-18th December 1985, and published as SINDOC(86)12. The third committee prepared a code validation matrix for BWRs. The BWR matrix was approved at the Task Group meeting of May 1986, and published as SINDOC(86)13. As a final step, the PWR and BWR SINDOCs were combined into one LWR code validation matrix for publication as a CSNI report.

This report represents the result of a joint effort of members and participants of the Task Group who formed the various committees. A list of these Task Group and Committee members is given at the end of this report.

Special Note

The tests included in the tables 1 to 10 of the combined matrix have been selected solely according to their technical quality for the purposes of code assessment, and irrespectively of the availability of their data.

Tests whose data at the time of the publication of this report on the combined matrix had not been available have been maintained in the matrix since their release might be expected at a later date. This regards in particular experimental data from the multi-national programmes OECD-LOFT and 2D/3D. Their release has in the meantime, however, been requested by letters to the OECD-LOFT Management Board and to the 2D/3D Steering Committee. US data from commercial plants and industrial laboratories are available on special conditions. Written requests have been made also for the release of still classified data of (1) matrix table 3 for PWRs with once-through steam generators, (2) matrix tables 6 to 10 for BWRs, and (3) the new BETHSY and SPES test facilities.

CONTENTS

Abstract

This report deals with the definition of a set of tests (matrix) for assessment of thermal-hydraulic best estimate codes for LWR LOCA and Transient Analyses. The work has been divided for application to PWR and BWR facilities and then sub-divided into two parts. In the first sub-division, the main physical phenomena that occur during the considered accidents are identified and test facilities suitable for reproducing these aspects are selected. In the second sub-division, a list of selected experiments already carried out or planned in these facilities has been set down. The criteria to achieve the objectives and areas for future development of the work are outlined.

1. INTRODUCTION

Prediction of nuclear reactor thermal-hydraulic behaviour under off-normal conditions can potentially be made in two separate ways:

(a) direct extrapolation of thermal-hydraulic behaviour in smaller scale facilities to real plants;

(b) assessment of large computer codes and application of qualified versions to plant situations.

The group considers that (b) is the only technically satisfactory route.

The objective of this report is to define a minimum set of experiments, for which comparison of the measured and calculated parameters forms a basis for establishing the accuracy of the test predictions. A further step is the estimation of the capability to simulate real plant behaviour.

Two types of matrices have been defined in this study. In the first, which served as a basis for the second, the physical phenomena which are assumed to occur during LOCAs and transients are listed and the experimental facilities suitable for reproducing these aspects are selected. In the second matrix, a set of tests carried out in the facilities which cover the physical phenomena are identified and recommended for code assessment applications.

The following observations should be kept in mind:

- developmental assessment is assumed to have been completed by the code developers, although the Group realises that in reality there is on-going iteration between independent and developmental assessment;

- the proposed assessment process is aimed at Best Estimate (BE) thermal-hydraulic system codes applied to integral tests; however, the comparison of codes against a limited number of separate effects experiments may also be of value;

- while attempts were made to formulate general validation matrices by including the phenomena of interest for most classes of plants, it should be recognised that under certain conditions some special classes of reactors may display phenomena not adequately addressed by the attached matrices. In such cases, additional validation against experiments addressing these phenomena might be needed.

The test matrices have been compiled only on the basis of technical suitability without regard to the availability of data. Some indication is given in the matrices of whether or not the facility and experiment data are currently available for general use. In those cases for which data are not available for general use, special arrangements will be necessary to obtain the data from their sources.

Domestic programmes for PWR configuration have aimed at the basic objective of this work (10.6-10.14). Within the international nuclear community the programme of International Standard Problems (ISPs) has contributed to code assessment(10.3). This contribution is recognised as very valuable, and the Group recommends that the programme should be continued.

Finally, it must be emphasised that this work represents the 'Current State-of-the-Art'; it must be updated after new important findings and particularly after the beginning of the operation of new test facilities.

Tests whose data will not become available in due time will have to be replaced by valid substitutes.

2. GENERAL CRITERIA FOR DEFINING THE MATRICES

At present the required accuracy criteria for best-estimate codes have not been established. In reality, this is a matter of subjective judgement, and in any case the codes are used for a variety of purposes, often requiring different accuracy criteria. Thus, the Task Group has restricted its activities to the selection of the technically best minimum set of experiments to be used in the process normally known as Independent Assessment. The details of how the results of the assessment will be used remain to be worked out.

The first question arising concerns the code area(s) which must be assessed; at least three main aspects can be identified:

- basic models or theories, including balance equations;
- numerical methods, convergence, truncation errors, etc.;
- implementation of the basic models within the computer program (i.e. way of calculating a quantity related to a junction between two adjacent volumes, pre-integration models, etc.).

The Task Group decided that these topics are properly addressed during code development, and therefore they would be excluded from the present study.

The second question concerns the type of test to be considered:

- basic tests including numerical and analytical benchmark exercises, or analyses of a single thermal-hydraulic phenomenon;
- separate effects or component tests;
- tests on integral facilities including transients measured in real plants.

Due to the objective of the work, emphasis has been given to this last topic, referring to the previous two issues only when a basic aspect is considered to be of fundamental importance and if its simulation in an integral facility is judged unsatisfactory.

The last question concerns the analysis and the classification of off-normal events which may happen in a real system. These classifications are slightly different for PWRs and BWRs, although in both cases a simple classification based on rupture area has been preferred.

For PWRs three classes of accidents were selected:

(1) large breaks (rupture area greater than 25 per cent of the maximum pipe area connected with the pressure vessel A_{max});

(2) small and intermediate breaks (rupture area less than or equal to 25 per cent A_{max});

(3) pressurised transients, where upset conditions are created by single or multiple failures of one of more systems in the plant.

Large break LOCAs (LB LOCA, item (1)), are characterised by a strong turbulence inside the primary loop, at least during the depressurisation period, roughly up to the time when the fluid pressure inside the circuit reaches 0.5 MPa.

For Small Break LOCAs (SB LOCA, item (2)), gravity and stratification effects are important phenomena to be simulated. Intermediate Break LOCAs (IB LOCA, item (2)) may contain features related to both LB and SB LOCAs. The actual sub-division between classes (1) and (2) depends in general upon the size of the plant, the position of the break and the status (on, off) of the pumps; the boundary selected (25 per cent A_{max}) must be seen as somewhat arbitrary.

In class (3) the transient is mainly determined by the sequence of operation of valves, pumps, and engineered safety features.

For BWRs, almost all the plants now in operation are equipped with an ADS (Automatic Depressurisation System), and therefore, even a small break LOCA is characterised by a fast depressurisation following the actuation of this device. Thus, as a difference from the PWR case, only two matrices have been considered in the present analysis: a LOCA matrix and a Transient matrix. In particular, with reference to the primary fluid pressure trend during the blowdown period, the curves shown in Fig. 1 have been taken into account for defining the test types in the LOCA matrix. Finally, in the Transient matrix start-up tests, frequent operational transients and ATWS have been included.

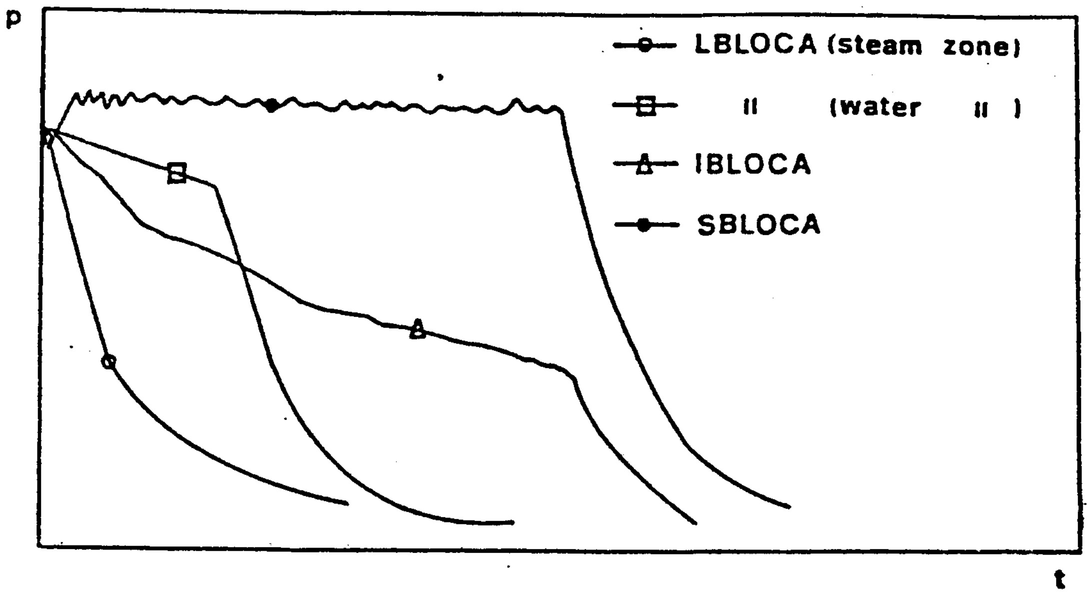

Fig. 1: Reference pressure trends in different types of accidents in BWR plants

3. STRUCTURE AND USE OF THE MATRICES

To systematise the preparation of the validation matrix, matrices related to LOCA and Transients, were set down, with the objective of allowing a systematic selection of tests suitable for code assessment.

Each matrix is composed of three sub-matrices SM1, SM2 and SM3, related to the following items:

(a) phenomena versus test types;
(b) phenomena versus test facilities (both system and separate effects tests);
(c) test facilities (only system tests) versus test types.

In the term 'phenomena' all the important thermal-hydraulic aspects expected to occur during the accident are included; type of test relates to the definition of the experiment; the meaning of 'test facilities' is evident; integral facilities and separate effects facilities are included in this term.

Phenomena, test types and test facilities were selected essentially on the basis of personal experience of the participants of the Committee. As already mentioned, emphasis has been given to large-scale integral systems.

Three observations should be noted:

- developmental assessment is assumed to have been completed by the code developers; although, realistically, there is a continuous iterative process between independent and developmental assessment;

- the validation matrices should provide the basis for the definition of future International Standard Problems;

- only tests already executed or planned through 1985 are considered for filling the matrices.

3.1 Use of the Matrices

The correlation between phenomena and test type (sub-matrices SM1) is given three levels:

- occurring, which means that the particular phenomenon is occurring in that kind of test (closed circle in the matrix);

- partially occurring: only some aspects of the phenomenon are occurring (open circle in the matrices);

- not occurring (dash in the matrices).

The phenomena and test facilities (sub-matrices SM2) is given four levels:

- suitable for code assessment, which means that a facility is designed in such a way to simulate the phenomenon assumed to occur in the plant and it is sufficiently instrumented to reveal it (closed circle in the matrices);

- limited suitability: the same as above with problems due to imperfect scaling or insufficient instrumentation (open circle in the matrices);

- not suitable: obvious meaning, taking into account the two previous items (dash in the matrices);

- expected to be suitable: definition introduced in some cases to emphasise that new facilities still under construction particularly address the simulation of this aspect; clearly a conclusive comment cannot be made at present (x in the matrices).

The correlation between test facilities and type of tests (sub-matrices SM 3) is given three levels:

- already performed or planned within 1985: the test type is useful for code assessment purposes (closed circle in the matrices);

- 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 to lack of instrumentation (open circle in the matrices);

- not performed or planned (blank).

With the exception noted above, all spaces related to facilities in construction have been left blank, awaiting experimental evidence.

3.2 Matrix for Large Breaks in PWRs

The Large Break Matrix is given in Matrix 1. Thirteen phenomena, three test types, seven integral test facilities and six separate effects test facilities are included.

Widely used nomenclature has been adopted in the identification of phenomena and test types. The explanation of the reasons for all the choices would be too long; nevertheless for a better understanding of the matrices, it may be useful to describe one or two lines of each of the three sub-matrices.

Sub-matrix SM1

For PWRs without vent valves, steam binding is assumed to be of high importance for reflood tests, of limited interest in refill tests and of no interest for blowdown tests. For PWRs with vent valves, steam binding is assumed to be of little interest.

Sub-matrix SM2

The steam binding phenomenon is suitable for code assessment when it is detected in PKL facility, of limited interest when it is detected in CCFT, LOFT and SEMISCALE, of no interest in the remaining facilities.

Sub-matrix SM3

Blowdown, refill and reflood types of test have been performed in LOFT and are useful for code assessment purposes; also in PKL all three kinds of tests have been performed, but only the last two are really useful for code assessment purposes, while the blowdown type of test is of limited suitability (due to the relatively low design pressure of PKL, 4.5 MPa).

3.3 Matrix for Small and Intermediate Breaks in PWRs

The Small and Intermediate Breaks Matrices are given in Matrices 2 and 3. Twenty three phenomena, seven test types, eight integral test facilities and seven separate effects test facilities are included for U-tube PWRs and twenty six phenomena, seven test types and five additional integral tests facilities are included for OTSG PWRs. Among the integral test facilities the category PWR has been included. The analysis of accidents in actual nuclear power plants is potentially valuable, especially with reference to scaling and simulation problems.

The same observations as in paragraph 3.2 apply here. Also it should be noted that, among the phenomena, the 'structural heat and heat losses' has been considered in order to emphasise the noticeable distortions introduced by the heat release from structures in the scaled-down facilities with respect to the plant behaviour; this is due to the larger values of structural masses and structure-to-fluid heat exchange areas relative to the volume-scaled values.

A relatively large number of test types (7) have been considered to emphasise that a number of different transients are possible under the general category of Small Break LOCA.

The symbol X just introduced for this matrix (see paragraph 3.1), shows that new facilities, still in construction, are especially suitable for the simulation of some of the identified issues not covered by previous facilities.

3.4 Matrix for Transients in PWRs

The transient Matrix is given in Matrix 4. Eleven phenomena, nine test types and eight integral test facilities are included.

The following observations can be added to those in the preceding two sections:

- almost all the phenomena discussed in the SB and IB matrix are important in the present case and for brevity they are not listed in this table;

- thermal-hydraulic nuclear feedback is the phenomenon which characterises some of the Transients: for this reason, LOFT (which was equipped with a nuclear core) was judged to be more suitable than the other experimental loops in operation;

- spesrate effects test facilities are not considered because relevant experiments are dealt with in the preceding table;

- BETHSY, LSTF and SPES facilities are expected to improve substantially the understanding of Transient behaviour;

- the experience gained from accidents occurring in real plants is potentially of great importance, as is that from the analysis of data from start up, shut down, and other manoeuvres.

3.5 Matrix for LOCAs in BWRs

The LOCA Matrix is given in Matrix 5. Twenty four phenomena, six test types, seven integral test facilities and eight separate effects test facilities are included. Among the integral test facilities, the category BWR has been included. The analysis of off-normal events in actual nuclear plants is potentially valuable, especially with reference to scaling and simulation problems.

It should be noted that, among the phenomena, the 'structural heat and heat losses' have been considered in order to emphasise the noticeable distortions introduced by the heat release from structures in the scaled-down facilities with respect to the plant behaviour; this is due to the larger values of structural masses and structure-to-fluid heat exchange areas relative to the volume-scaled values.

A relatively large number of test types (5) has been considered to emphasise that a number of different transients are possible under the general category of blowdown.

3.6 Matrix for Transients in BWRs

The Transient Matrix is given in Matrix 6. Fifteen phenomena, nine test types, four integral test facilities, and four separate effects test facilities are included.

The following observations can be added to those in the preceding section:

- almost all of the phenomena discussed in the LOCA matrix are also important for transients, but for brevity they are not repeated in this table;

- thermal-hydraulic nuclear feedback is the phenomenon which characterises some of the transients: for this reason, real BWR plants have been judged much more suitable than the experimental loops already in operation.

- the experience gained from off-normal events occurring in real plants is potentially of great importance, as is that from the analysis of data from start up, shut down, and other manoeuvres.

4. BASIS FOR SELECTION OF EXPERIMENTS

During the Validation Matrix Committee meetings, a number of experimental facilities and specific experiments from those facilities were proposed for inclusion in the CSNI Validation Matrix for thermal-hydraulic system computer codes. During the selection process a number of factors were considered, including:

(1) Companion matrices which relate phenomena of interest, test facility and test type, (which became the sub-matrices SM1, SM2 and SM3).

(2) Typicality of facility and experiment to expected reactor conditions.

(3) Quality and completeness of experimental data.

In all cases attempts were made to find plant results or integral experiments to address each of the phenomena of interest. Only in cases where suitable plant results or integral experiments could not be found were separate effects experiments selected. Where counterpart tests or nearly counterpart tests were identified between two or more facilities, they were included in order to address questions relating to scaling and facility design compromises.

5. PWR FACILITIES

5.1 Large Breaks (25 per cent)

(a) PKL-I

PKL-I data were selected as valuable for assessment of codes during the reflood phase. SEMISCALE and LOFT experiments were also considered, but PKL-I was considered to be better instrumented and to have fewer scaling compromises. Also, larger size made PKL preferable to SEMISCALE. The specific experiments selected were judged to cover the range of types of injection of interest (cold leg only, hot leg only, and combined injection). When PKL-II experiments are completed some may be considered as replacements for the PKL-I tests. PKL-II has significant design and instrumentation improvements over PKL-I, and experiments are initiated at a high pressure giving more realistic initial conditions.

(b) LOBI

LOBI experiments cover only the blowdown phase of an accident. It was considered that use of LOBI for the blowdown phase coupled with PKL for the reflood phase would cover the full range of the accident. The experiments selected were judged to cover the various types of ECC injection of interest and to allow a sensitivity study related to required scaling compromises in downcomer design. The 50 mm gap gives good CCFL scaling but 7 times too large a downcomer volume, resulting in unrealistic core flows. The 12 mm gap gives more accurate volume scaling but much too large a hot-wall effect and unrealistic ECC delivery. However, LOBI's larger size and full height components were considered an advantage over SEMISCALE. Two experiments, B-R1M and A1-04R were considered opportunities to investigate counterpart tests. B-R1M is a 25 per cent cold leg break.

(c) CCTF

The Cylindrical Core Test Facility, part of the 2D/3D programme, offered the advantage of a large size, well-instrumented facility which has produced some very good reflood data. Its size allows consideration of scaling effects through comparison with PKL, SEMISCALE and LOFT results.

(d) BETHSY

This new facility currently under construction in France, will take advantage of what has been learned about the various scaling compromises present in older facilities and include the most advanced instrumentation available. The test matrix is not yet established, however, BETHSY has been included in the validation matrix because it may be used to address large break problems during the refill/reflood phase, although the main objectives of the BETHSY programme are in the area of small break LOCA and Special Transients.

(e) LOFT

LOFT's short core and steam generator, excessive core bypass, other scaling compromises, and lack of adequate measurements in certain areas, were considered to be problems. However, it is a large, generally well instrumented facility with a large data-base, and is the only integral research facility with a nuclear core. The four experiments included in the matrix were selected as LOFT's best double-ended cold leg break experiments to cover various core power, ECC injection, and pump operating conditions. In particular, the pumps on/off, L2-5/L2-6 pair offer a unique test of the codes' abilities to predict the effects of pump operation under LOCA conditions.

(f) SEMISCALE

Many more SEMISCALE experiments than those included in the matrix were considered, but generally the larger size and other factors in LOBI, PKL and LOFT made those facilities better choices. SEMISCALE did offer the advantage of running the full blowdown through reflood transient and the two tests selected offered counterpart tests to LOBI and LOFT.

(g) SPES

The comments made on BETHSY also apply to SPES.

(h) SUPER MOBY DICK and MARVIKEN

Tests from these two facilities were selected to supply the most needed data to address break flow modelling. Many other experiments were considered in this area including SUPER MOBY DICK experiments 234R31 and 30B.9X, MARVIKEN experiment 22, and HDR experiments V31.1 and V33. However, it was decided that, since much attention is given to critical flow modelling in the developmental assessment process and each integral facility experiment also offers data related to critical flow, the validation matrix should be limited to two break flow separate effects tests. The two tests selected were judged most appropriate based on size, typicality to reactor conditions and quality of data.

(i) SCTF

The comments on CCTF also apply here. This facility, due to its large size and design features offers a unique opportunity to look at multi-dimensional, two-phase phenomena of interest in the tie-plate, core and upper plenum region. CCFL and liquid distribution in the upper plenum and core as functions of power profiles and ECC injection locations are particularly well addressed in this facility. The experiments selected offer two different core power profiles and two different ECC injection positions. It was also felt that, at a later date, experiments from the companion 2D/3D UPTF program should be considered in conjunction with similar SCTF experiments.

(j) BCL and CREARE

Experiments from these facilities were selected to supply needed downcomer ECC delivery and bypass data. Both facilities are 1/15 scale with the BCL experiment supplying steady state data and CREARE supplying ramped steam, transient data. These two separate effects tests supply detailed measurements to supplement downcomer data obtained in the various integral experiments, addressing in particular hot wall questions.

5.2 Small and Intermediate Breaks (25 per cent) in U-Tube PWRs

In general, comments concerning the strength and limitations of facilities included in the large break section apply here also. In some cases scaling compromises in older facilities, which were originally designed to address large breaks, take on increased significance for small breaks and transients.

(a) PKL

Two steady state tests were chosen covering a range of primary system and steam generator liquid levels with the primary system at 30 bar. Two additional transients, 2 per cent cold leg breaks, were also chosen, one with cold leg injection and one with hot leg injection. In this manner, a large range of small break post-blowdown conditions, including the reflux condenser mode, were covered.

(b) LOBI-MOD 2

Two tests from the A2 matrix were selected to supply information at higher pressures than in the PKL tests. These are a 1 per cent cold leg break with two out of four cold leg HPIS systems operating, and a steady state test with decreasing primary system inventory. When reviewing the LOBI-MOD 2 preliminary B-Matrix, concern was expressed that the tests considered were somewhat benign with no significant core uncovery experiments planned. It is hoped that this will be corrected and perhaps three further tests can be selected for the matrix. The LOBI-MOD 2 system is much improved over the MOD 1 system and it will include a well instrumented full height steam generator which is expected to yield valuable information for code assessment in the steam generator area.

(c) LOFT

While the LOFT facility vertical elevation, high core bypass and poorly instrumented steam generator were considered significant disadvantages for small break testing, its large size, nuclear core and unique data base nevertheless makes it valuable for code assessment. Thus, seven tests have been included in the matrix.

These tests include an intermediate break with deep core uncovery after which the pumps were restarted, and a series of small break tests covering one, two, three and four inch equivalent break sizes, hot leg and cold leg breaks, and primary pumps on and off. It was agreed that, after BETHSY and ROSA IV results became available, they should be reviewed as possible substitutes for some of the LOFT experiments.

(d) SEMISCALE

Two SEMISCALE experiments were considered unique and important for inclusion in the matrix. One, UT-1, was a 10 per cent cold leg break and the other, PL-3E, was a station blackout with feed-and-bleed recovery.

(e) Crystal River and Ginna

It is considered preferable to utilise, when available, some full-sized plant data in the matrix and these two facilities were felt to offer the best opportunity. It was recognised that data will be of limited quality with only a few parameters measured, and that information may not be generally available. However, even limited assessment at full scale is considered essential.

(f) DOEL-2

This Belgian plant experienced a steam generator tube rupture similar to Ginna, but at zero power. This transient was considered as an alternative to Ginna. However, the event has been analysed extensively and a large quantity of information is available.

(g) BETHSY, SPES, ROSA IV

These new facilities have been scaled taking into account the lessons learned on older facilities, in particular regarding scaling compromises necessary for SB-LOCA and Special Transients simulations.

Experiments in these new facilities should be reviewed as they become available. It is expected that some will be suitable for inclusion in the matrix, including possible substitutions for existing tests in other facilities.

(h) THL

The Thermal-Hydraulic Laboratory (THL) at the INEL was planning to perform a series of T-junction breakflow tests in the large two-phase flow loop during the summer of 1984.

These tests will be in LOFT-sized pipes simulating the L3-5 and L3-6 experimental conditions, and will investigate break flow from a Tee exiting at the bottom, side and top of the pipe with various liquid levels in the pipe. Two tests, one with the break at the bottom of the pipe and one at the top, were selected for inclusion in the matrix to address small break critical flow from a branch.

(i) G-2

The three boil-off tests selected from this facility show the effects of both pressure and power. The facility is well instrumented and the data appear consistent. These tests have been used successfully in the assessment of both RELAP-5 and TRAC.

(j) PATRICIA GV-1 and GV-2 and GEN 3 x 3

It was generally agreed that none of the integral facilities have yet supplied the detailed steam generator data needed, primarily due to lack of instrumentation. Steam generator heat transfer is highly important in small breaks and currently it appears the codes need improvement in this area. These three separate effects facilities were selected as most appropriate. PATRICIA GV-1 will supply steady state data in the primary side under various conditions including refluxing. PATRICIA GV-2 will supply steady state data in the secondary side under various conditions. GEN 3 x 3 is a large, 10 metre high, well instrumented, 3 x 3 U-tube steam generator to which primary and secondary fluid can be supplied under various conditions. The test proposed is under transient conditions with varying secondary flow and enthalpy and primary temperature.

(k) Pressuriser Flooding

Another area, in addition to the steam generator, found to be a problem for the codes during small breaks, is the pressuriser. The codes in general have not only not done a good job of predicting pressuriser behaviour, but, recently, consideration is being given to whether nor not the heat transfer mechanisms being used are correct.

For this reason, more detailed pressuriser measurements than are currently available from the various integral facility tests are needed. The PRESSURISER FLOODING tests performed in Italy offer such data under CCFL conditions in both horizontal and vertical portions of the surge line.

5.3 Small and Intermediate Breaks (25 per cent) in OTSG PWRs

The following five additional test facilities were selected to address phenomena which are unique to OTSG PWRs. The primary features of an OTSG plant which produce these unique phenomena include: the 2 x 4 loop configuration, the two OTSGs, the vertical hot legs with candy canes, the horizontal section of pressuriser surge line which enters the vertical hot leg through a trap, and the vent valves.

(a) UMCP 2 x 4 B&W Simulation Loop

This well instrumented and versatile facility at the University of Maryland in College Park Maryland (UMCP) is a 2 x 4 loop simulation of the Babcock & Wilcox TMI unit 2 lowered loop reactor. All active components are simulated with a volume scale of 1 : 500. The electrically heated core has a maximum power of 200 KW (24.7 per cent of full power for a 2800 MW_{th}

reactor) and a maximum pressure of 300 psi (20.7 bar). The loop was designed to simulate natural circulation and SB LOCA behaviour under a wide range of conditions.

Currently six shake-down tests and seven characterisation tests have been performed. The test program consists of five natural circulation parametric studies in five SB LOCA parametric studies and is scheduled to be completed by mid-1986.

A unique feature of this facility is its scaling concept which differs from all other PWR facilities included in this PWR validation matrix; thereby offering the opportunity to investigate the effects of scaling by comparisons with other facilities.

The scaling concepts used are somewhat similar to those proposed by Messrs. Ishii and Kataoka and, while timing of events are designed to be the same as in the full sized plant, elevations of the various components are not preserved. The scaling concept is somewhat similar to, but in some ways different from, that used for SRI-2, discussed in paragraph 5.3(e).

(b) <u>MIST</u>

the Multi-Loop Integral System Test (MIST) Facility was constructed by B&W at Alliance, Ohio. Its test program, consisting of 41 experiments to study SB LOCA, SGTR, Feed and Bleed, pump operation, non-condensables and natural circulation, will be conducted from September 1985 through September 1986.

MIST is a 2 x 4 loop simulation of a lowered loop B&W plant. It has a volume scale of 1 : 817 with critical elevations preserved. The design approach and priorities are similar to the OTIS facility (below) and it uses some of the components, including one of the steam generators, from that facility. All active components are simulated including primary coolant pumps.

The electrically heated core has 45 full length, 10.9 mm diameter heater rods and power equal to 10 per cent scaled power based on a 2584 MW_{th} PWR. Two 19-tube OTSGs will be used with prototypical tubing, tube pitch, and tube support plate characteristics. Hot and cold leg piping are sized to preserve two-phase phenomena and are approximately 64 mm and 51 mm diameter, respectively. The facility has some 687 instruments and operates at full PWR pressures.

(c) <u>OTIS</u>

The Once-Through Integral Systems, OTIS, facility was constructed by B&W in Alliance, Ohio, by modifying the GERDA facility (below) to be more representative of B&W raised-loop plants. Modifications were also made, based on GERDA results, to improve measurements and accident simulation. Modifications included:

(1) installing a reactor head vent;
(2) installaing heaters (active insulation) on the reactor head and pressuriser surge line to limit heat losses to typical plant levels;
(3) installing thermocouples for measuring primary fluid temperatures in selected OTSG tubes;

(4) installing pitot tubes for determining primary flows in selected OTSG tubes;
(5) relocating the cold leg flow meters;
(6) incorporating multiple critical flow orifices at the cold leg break site, and
(7) resetting sub-systems such as HPI capacity, to be consistent with U.S. plant characteristics.

OTIS is a 1 : 1686 volume-scaled single loop (one hot leg and one cold leg) facility with critical elevations preserved. The single OTSG contains 19 tubes prototypical of a B&W plant. The electrically heated core has a power capability of 180 KW which is representative of 10 per cent scaled power based on a 2584 MW_{th} plant. It contains no primary pump but all other active components are simulated and it operates at full PWR system pressure.

Approximately 225 instruments were used for measuring thermal-hydraulic conditions.

The experimental program, which included 13 tests designed to expand the extensive GERDA data base, was conducted between February and May 1984. All tests were SB LOCA with capacity and/or control of one system being varied in each test.

(d) <u>GERDA</u>

The Geradrohr Dampferzeuger Anlage, GERDA, or straight-tube steam generator facility, was constructed and tested at B&W in Alliance, Ohio as a joint effort of Brown Boveri Reactors, BBR and B&W. It simulated a raised loop BBR plant and had 1 : 1686 volume scaled components. The scaling, design philosophy and components were similar, and in most cases identical, to those described for OTIS.

Over 100 separate effects and integral tests were performed in GERDA covering a wide range of SB LOCA and natural circulation conditions. The experimental program was completed in 1983.

(e) <u>SRI-2</u>

SRI-2 is a 2 x 4 loop simulation of a B&W lowered loop plant (TMI Unit 2) built and operated at Stanford Research Institute (SRI). All active components are simulated in 1 : 1296 volume-scaling. The electrically heated core has a maximum power capability of 88 KW (17 per cent full power) with 18 heater rods, and a maximum pressure of approximately 7 bar. The facility is unique in that it is 'Ishii scaled' throughout. All critical elevations are 1/4 full size and cross sectional areas are 1/324 full size. Two sets of hot legs, both two inch (51 mm) ID, with different radii of curvature allow studies of scaling of that parameter. The steam generators have 48 tubes of full size plant diameter and 1/4 height. There are more than 100 instruments.

SRI-2 is now in its shakedown testing phase. Characterisation testing is scheduled to start in September 1985, and the experimental program of some 15-20 tests is to be completed and a final report issued by March 1986. Most experiments will be performed in a steady state mode and will include investigations of SB LOCA, SGTR, feed-and-bleed, natural circulation, boiler condenser, and loop asymetry.

5.4 Plant Transients

It becomes apparent when looking at the Plant Transients matrix, that far fewer experiments are included than in the other two categories. This is for two reasons. First, much of the data obtained from experiments in the small break matrix also apply to plant transients, and second, there is a limited amount of plant transient data available.

(a) LOFT

The LOFT programme, which has included a number of plant transient experiments, is important for investigation of a number of phenomena in the transient area due to its nuclear core. Experiments selected include a turbine trip cooldown, a loss-of-feedwater ATWS, and a loss-of-feedwater, primary feed and bleed.

(b) ANO-2

As was stated earlier, code assessment against full-sized plant data is highly desirable. The ANO-2 turbine trip-cooldown not only offers such an opportunity, but the LOFT L6-7/L9-2 experiment included in the matrix offers a simulation of this incident and supplies more detailed data for understanding the ANO-2 transient. As with other plant data, data from ANO-2 will be of limited quality without as many measurements as desired, and the data may not be generally available. However, ANO-2, coupled with the LOFT simulation, offers a unique opportunity to test the codes against a full-size plant transient.

(c) LOBI-MOD2, BETHSY, SPES, ROSA IV

These new facilities are included in the matrix in the belief that they will include meaningful plant transient experiments in their matrices and, thus, supply additional data against which the codes can be assessed.

6. PWR VALIDATION MATRICES

This section contains the PWR matrices and tables.

The matrices described in Section 3 relating phenomena, test type and transients are given in matrices 1,2,3 and 4.

The lists of selected tests are given in Tables 1 - 4, the structure of which is self-explanatory.

Matrix I

CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
- Test type versus test facility
  - ● already performed or planned until 12/84
  - ○ performed or planned until 12/84, but of limited use

		Test Type			Test Facility System Tests							Test Facility Sep. Effects Tests					
		Blowdown	Reflood	Refill	CCTF 1:25 *	LOFT 1:50	BETHSY 1:100	PKL 1:134	SPES 1:430	LOBI 1:712	SEMISCALE 1:1600	UPTF 1:1	CREARE 1:25	SCTF 1:25	MARVIKEN	BCL	SUPER MOBY DICK
Phenomena	Break flow	●	●	●	-	○		○		○	○		-	-	●	-	●
	Phase separation	○	●	●	●	●		●		●	●		-	-	○	-	-
	Mixing and condensation during injection	○	●	●	○	○		○		○	○		●	●	-	●	-
	Core wide void+flow distribution	○	●	●	○	○		○		-	-		-	●	-	-	-
	ECC bypass and penetration	○	○	●	●	●		○		-	-		●	○	-	●	-
	CCFL (UCSP)	○	●	●	○	○		○		○	-		-	●	-	-	-
	Steam binding (liquid carry over, etc.)	-	●	○	○	○		●		-	○		-	-	-	-	-
	Pool formation in UP	-	●	●	○	○		○		-	○		-	●	-	-	-
	Core heat transfer incl. DNB, dryout, RNB	●	●	●	○	●		○		○	○		-	○	-	-	-
	Quench front propagation	○	●	○	○	●		○		-	○		-	○	-	-	-
	Entrainment (Core, UP)	○	●	○	○	○		○		○	○		-	●	-	-	-
	Deentrainment (Core, UP)	○	●	○	○	○		○		-	○		-	●	-	-	-
	1- and 2-phase pump behaviour	●	○	○	-	○		○		●	●		-	-	-	-	-
Test Facility System Tests	CCTF	-	●	○													
	LOFT	●	●	●													
	BETHSY																
	PKL	○	●	●													
	SPES																
	LOBI	●		●													
	SEMISCALE	●	●	●													

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

* volumetric scaling

Matrix II

CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE LEAKS IN PWRs

- -Phenomenon versus test type
  - ● simulated
  - ○ partially simulated
- -test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - X expected to be suitable
- -test type versus test facility
  - ● already performed or planned until 12/84
  - ○ performed or planned until 12/84, but of limited use

Phenomena****

Phenomena	Test Type: Stationary test adressing energy transp. on prim. side	Stationary test adressing energy transp. on sec. side	Small leak overfeed hy HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak, sec. side not necessary	Pressurizer leak	U-tube rupture	System Tests: PWR 1:1*	LOFT 1:50	LSTF 1:50	BETHSY 1:100	PKL-I 1:134	SPES 1:430	LOBI-II 1:712	SEMISCALE 1:1600	Separate Effects Tests: UPTF 1:1	THL 1:15	GEST GEN 1:50	Patricia GV-1	Patricia GV-2/GEN 3x3	G-2	Pressurizer Test (CISE)
Natural circulation in 1-phase flow, primary side	●	●	●	○	–	●	●	●	●			●		●	●		–			–	–	–
Natural circulation in 2-phase flow, primary side	●	–	○	●	●	○	–	–	●			●		●	●		–			–	–	–
Reflux condenser mode and CCFL	●	–	–	●	●	–	–	–	○			○		○	○		–		●	–	–	–
Asymetric loop behaviour	–	–	●	●	–	○	●	–	–	X	X	–	X	○	○		–			–	–	–
Leak flow	–	–	●	●	●	●	●	–	●			○		●	●		●			–	–	–
Phase separation without mixture level formation	●	–	○	●	●	●	○	–	○			○		●	○		○			–	–	–
Mixture level and entrainment in vertic. comp. SG**	–	●	●	●	●	●	●	–	–			○		○	–		–	X		●	–	–
Mixture level and entrainment in the core	●	–	–	●	●	●	–	–	○			○		○	○		–			–	●	–
Stratification in horizontal pipes	●	–	–	●	●	–	–	–	●	X	X	○		○	○		●			–	–	–
ECC-mixing and condensation	–	–	○	●	●	●	●	–	○			○		○	○		–			–	–	–
Loop seal clearance	–	–	–	●	●	○	–	–	●			○		●	●		–			–	–	–
Pool formation in UP/CCFL(UCSP)	●	–	–	○	●	●	–	–	○	X		○		–	○	X	–			–	–	–
Core wide void and flow distribution	●	–	–	○	●	●	–	–	○	X		–		–	–		–			–	–	–
Heat transfer in covered core	●	●	●	●	●	●	●	○	●			●		●	●		–			–	–	–
Heat transfer in partially uncovered core	●	–	–	○	●	–	–	–	●			○		○	○		–			–	●	–
Heat transfer SG primary side	●	○	○	●	●	○	○	–	○			○		●	○		–	X	●	–	–	–
Heat transfer in SG secondary side	○	●	●	●	●	●	●	–	○			○		●	○		–	X		●	–	–
Pressurizer thermohydraulics	○	–	○	○	●	●	●	○	○			–		○	–		–			–	–	●
Surgeline hydraulics	○	–	–	○	●	●	○	–	○			–		○	○		–			–	–	●
1- and 2- phase pump behaviour	–	–	–	○	●	–	–	○	○			–		●	●		–			–	–	–
Structural heat and heat losses***	●	–	○	●	●	○	○	–	○			○		○	○		–			–	–	–
Noncondensible gas effects	●	–	–	–	–	–	–	–	–			○		●	○		–			–	–	–
Phase separ. in T-junct. and effect on Leakflow	–	–	–	●	●	–	–	–	○			–		○	–		●					

Test Facility – System Tests

Test Facility	Stationary test adressing energy transp. on prim. side	Stationary test adressing energy transp. on sec. side	Small leak overfeed hy HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak, sec. side not necessary	Pressurizer leak	U-tube rupture
PWR	–	–	○	–	–	●	●
LOFT	–	–	●	●	●	●	–
LSTF							
BETHSY							
PKL-I	●	●	–	●	●	●	–
SPES							
LOBI-II	●	●	●	●	●	●	●
SEMISCALE	○	○	●	●	●	●	●

* volumetric scaling
** secondary side
*** problem for scaled test facilities
**** for intermediate leaks phenomena included in large break reference matrix may be also important

Matrix III
CROSS REFERENCE MATRIX FOR SMALL
AND INTERMEDIATE LEAKS IN PWRs WITH OTSG
(IN ADDITION TO THE MATRIX FOR PWRs WITH UTSG)

- Phenomena versus test type
  - ● simulated
  - ○ partially simulated
- test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - X expected to be suitable
- test type versus test facility
  - ● already performed or planned until 12/85
  - ○ performed or planned to be performed by 12/85, but of limited use

		Test Type							Test Facility					
		Stationary test adressing energy transp. on prim. side	Stationary test adressing energy transp. on sec. side	Small leak overfeed by HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak, sec. side not necessary	Pressurizer leak	OTSG-tube rupture	PWR	Univ. Maryland (lowered loop) 1 : 500*	MIST (lowered loop) 1 : 817	SRI (lowered loop) 1 : 1296	OTIS (raised loop) 1 : 1686	GERDA (raised loop) 1 : 1686
Phenomena****	Natural circulation in 1-phase flow, primary side	●	●	●	○	–	●	●	●	●	●	●	●	●
	Natural circulation in 2-phase flow, primary side	●	–	○	●	●	○	○	–	●	●	●	–	–
	Boiler condenser mode	●	–	–	●	●	–	–	–	●	●	●	●	●
	Refill of loops	–	–	●	–	○	●	–	–	●	●	●	○	●
	Asymetric loop behaviour	–	–	●	●	–	○	●	–	●	●	●	–	–
	Leak flow	–	–	●	●	●	●	●	–	●	●	●	●	○
	Phase separation without mixture level formation	–	–	○	●	●	●	○	–	●	●	●	–	–
	Mixture level and entrainment in vertic. comp. SG**	–	●	●	●	●	●	●	–	○	●	○	○	○
	Mixture level and entrainment in the core	–	–	–	●	●	●	–	–	–	–	–	–	–
	Stratification in horizontal pipes	–	–	–	●	●	–	–	–	X	X	–	–	–
	ECC-mixing and condensation	–	–	○	●	●	●	●	–	X	X	–	–	○
	Loop seal clearance	–	–	–	●	●	○	–	–	●	X	–	–	–
	Pool formation in UP/CCFL(UCSP)	–	–	–	○	●	●	–	–	–	–	–	–	–
	Core wide void and flow distribution	–	–	–	○	●	●	–	–	○	○	–	–	–
	Heat transfer in covered core	●	●	●	●	●	●	●	○	X	X	X	●	●
	Heat transfer in partially uncovered core	–	–	–	○	●	–	–	–	–	–	–	–	–
	Heat transfer SG primary side	●	○	○	●	●	○	○	–	●	●	●	○	○
	Heat transfer in SG secondary side	○	●	●	●	●	●	●	–	○	○	○	○	○
	Pressurizer thermohydraulics	–	–	○	○	●	●	●	○	X	X	–	–	–
	Surgeline hydraulics	–	–	–	○	●	●	○	–	–	–	–	–	–
	1- and 2- phase pump behaviour	–	–	–	○	●	–	–	○	○	○	–	–	–
	Structural heat and heat losses***	●	–	○	●	●	●	○	–	○	○	○	○	○
	Noncondensable gas effects	●	–	–	–	–	–	–	–	●	●	○	●	●
	Phase separ. in T-junct. and effect on Leakflow	–	–	–	●	●	–	–	–	–	–	–	–	–
	Intermittent two-phase natural circulation	–	–	–	●	●	○	○	–	●	●	●	●	●
	Natural circulation-core, vent valve, downcomer	●	○	●	●	●	●	●	○	●	●	●	●	●
	Superheating in secondary side	●	●	●	●	●	●	●	●	●	●	●	●	●
Test Facility	PWR	–	–	○	–	–	●	○						
	Univ. Maryland	●	●	●	●	–	●	●						
	MIST	●	●	●	●	–	●	●						
	OTIS	●	●	●	●	–	●	–						
	GERDA	●	●	●	●	–	●	–						
	SRI	●	●	●	●	–	●	●						

* volumetric scaling
** secondary side
**** problem for scaled test facilities

Matrix IV

CROSS REFERENCE MATRIX FOR TRANSIENTS IN PWRs

-Phenomena versus test type

●simulated
○partially simulated

-test facility versus phenomena

●suitable for code assessment
○limited suitability

-test type versus test facility

●already performed or planned until 12/84
○performed or planned until 12/84, but of limited use

		Test Type									Test Facility System Tests							
		ATWS	Loss of feedwater, non ATWS	Loss of heat sink, non ATWS	Station blackout	Steam line break	Feed line break	Cooldown, prim. feed and bleed	Reactivity disturbance	Over-cooling	PWR 1:1*	LOFT 1:50	LSTF 1:50	BETHSY 1:100	PKL-1 1:134	SPES 1:430	LOBI-II 1:712	SEMISCALE 1:1600
Phenomena	Natural circulation in 1-phase flow	●	●	●	●	●	●	●	○	○	●	○			○		●	●
	Natural circulation in 2-phase flow	●	●	●	●	-	-	○	○	-	-	○			○		●	●
	Core thermohydraulics	●	●	●	●	○	○	○	●	○	○	●			-		●	●
	Thermohydraulics on primary side of SG	●	○	○	●	○	○	●	○	●	○	○			-		●	○
	Thermohydraulics on secondary side of SG	●	●	●	●	●	●	●	○	●	○	○			-		●	○
	Pressurizer Thermohydraulics**	●	●	●	●	○	○	○	○	●	○	○			-		○	○
	Surgeline hydraulics (CCFL, chocking)**	●	●	●	●	○	○	○	○	○	○	○			-		○	○
	Valve leak flow***	●	●	●	●	●	●	●	●	●		○			○		○	○
	1- and 2- phase pump behaviour	●	●	●	●	○	○	○	○	○	○	○			-		●	●
	Thermohydraulic-nuclear feedback	●	-	-	-	-	-	-	●	-	●	●			-		-	-
	Structural heat and heat losses ****	○	○	○	○	○	○	○	○	○	-	○			○		○	○
	Boron mixing and transport	-	-	-	-	○	-	-	-	○	-	-			-		-	-
	Separator	○	-	-	-	-	-	-	-	-	-	-			-		-	-
Test Facility System Tests	PWR	-	-	-	-	-	-	-	-	○								
	LOFT	●	●	●	○	-	-	●	●	●								
	LSTF																	
	BETHSY																	
	PKL-1	-	-	-	-	-	-	-	-	-								
	SPES																	
	LOBI-II	○																
	SEMISCALE	-	○	○	○	○	○	○	-	○								

* volumetric scaling

** for phenomena requiring separate effects test, e. g. pressurizer behaviour, refer to small leak cross reference matrix

*** valve flow behaviour will be strongly design-dependent, specific experimental data should be used if possible

**** problem for scaled test facilities

TABLE 1

TESTS PROPOSED FOR VALIDATION MATRIX

LARGE BREAKS (> 25 %)

COUNTRY	FACILITY	INTEGRAL or S.E.	TEST NO.	BRIEF DESCRIPTION	COMMENTS	ACCEPTANCE CONDITIONS	REFERENCE (CHAP 11)	DATA AVAILABLE
Germany	PKL-I	I	K9	DECLG, CL Injection (ISP 10)	Reflood tests only	✓	(a) 7	YES
			K10	DECLG, combined Injection		✓	8	YES
			K11	DECLG, HL Injection		✓		YES
JRC Ispra	LOBI	I	B-R1M	2% CLG, CL Injection	SEMISCALE approx. counterpart S-IB-3	✓	(b) 6	YES
			A1-06	DECLG, combined Injection, 12mm D/C		✓	8	YES
			A1-66	DECLG, CL Injection, 12mm D/C		✓	9	YES
			A1-04R	DECLG, CL Injection, 50mm D/C	LOFT approx. counterpart L2-3	✓	7	YES
Japan	CCTF	I	C(1-11)Run 20		Reflood tests only	✓	(c) 1	NO
			C(1-19)Run 38			✓	2	NO
			C(1-20)Run 39			✓	3	NO
France	BETHSY	I			? possible LB applications			
USA	LOFT	I	L2-3	DECLG, CL Injection, 36 MW	Pumps powered	✓	(e) 2	YES
			L2-5	DECLG, CL Injection, 36 MW (ISP 13)	Pump flywheels disconnected at t_o	✓	3	YES
			L2-6	DECLG, CL Injection, 46 MW	'EM' transient	✓	4	YES
			LB-1	DECLG, CL Injection, 50 MW	Pump flywheels disconnected at t_o	✓	5	NO
	SEMISCALE	I	S-06-3	DECLG, CL Injection	Approx. Counterpart for LOFT L2-3	✓	(f) 2	YES
			S-IB-3	2% CLG, CL Injection	Approx. Counterpart for LOBI B-R1M	✓	3	YES
Italy	SPES	I			? possible LB applications			
France	SUPER MOBY DICK	SE	12 R 305C	120 bar, long nozzle, D= 20mm, SS		✓	(h) 1	YES
Sweden	MARVIKEN	SE	22	L/D= 1.5, $\Delta T_{sub} \sim 50°$. D= 500 mm		✓	(i) 1	YES
Germany	UPTF	SE		UP Water Distribution and D/C penetration	To be included at a later date			NO
Japan	SCTF	SE	S1-01 Run 507	2 power profiles	core void distribution tests	✓	(k) 2	NO
			S1-08 Run 514			✓		NO
USA	BCL	SE	26502-7	SS	Downcomer bypass tests	✓	(l) 2	YES
			29402	Transient with ramped steam flow		✓		YES
	CREARE		H 195	1/15 scale, cold ECCS, hot wall, ramped steam flow		✓	(m) 2	YES

Notes: 1. HL and combined injected tests are not necessary for countries with only CL injection plants.

2. Failure to predict severe hot wall effects (eg in LOBI) will not invalidate code for plant applications.

TABLE 2

SMALL AND INTERMEDIATE BREAKS (≤25 %) FOR U-TUBE PWR's

COUNTRY	FACILITY	INTEGRAL or S.E.	TEST NO.	BRIEF DESCRIPTION	COMMENTS	ACCEPTANCE CONDITIONS	REFERENCE (CHAP 11)	DATA AVAILABLE
Germany	PKL	I	ID1	SS cases with reducing primary inventory, 30 bar	10 cases	✓ ✓	(a) 9	YES
			ID1B	SS cases with reducing secondary inventory, 30 bar	3 cases. Reflux mode in primary		10	YES
			ID5	2% CL, CL injection, 30 bar		✓		YES
			ID6	2% CL, HL injection, 30 bar		✓		YES
JRC Ispra	LOBI-MOD2	I	A2-81	1% CL, CL injection, 2/4 HPIS pumps (ISP18)		pending outcome of tests	(b) 10	YES
			A2-77	SS cases with reducing primary inventory, 90 bar	3 further tests to be specified after first results from MOD2 are available and 'B' matrix is finalised		11	YES
USA	LOFT	I	L8-2	Intermediate CL, delayed ECCS	Deep core uncovery, primary pumps restarted	✓*	(e) 6	YES
			L3-5	4" equiv. CL, pumps off		✓*	7	YES
			L3-6	4" equiv. CL, pumps on	Partial core uncovery	✓*	8	YES
			L3-7	1" equiv. CL	SG 'feed and bleed' test	✓*	9	YES
			SB-1	3" equiv. HL, pumps off	International Consortium tests	✓*	10	NO
			SB-2	3" equiv. HL, pumps on		✓*	11	NO
			SB-3	2" equiv. CL, delayed pump trip, no HPIS		✓*	12	NO
	SEMISCALE	I	UT-1	10% CL,		✓	(f) 4	YES
			PL-3E	Station blackout, 'feed and bleed' recovery		✓	5	YES
	CRYSTAL RIVER	I	-	Stuck open PORV	Data are limited	✓	(n) 1	NO
	GINNA	I	-	SG tube rupture at power	Pressurizer behaviour dominates	✓	(o) 1	NO
Belgium	DOEL-2	I	-	SG tube rupture at zero power			(p) 1	YES
France Italy Japan	BETHSY SPES ROSA-IV	I I I			To be included at a later date			YES

* to be reviewed as results from BETHSY and ROSA-IV become available

TABLE 2 (Continued)

SMALL AND INTERMEDIATE BREAKS FOR U-TUBE PWR's (Continued)

COUNTRY	FACILITY	INTEGRAL or S.E.	TEST NO.	BRIEF DESCRIPTION	COMMENTS	ACCEPTANCE CONDITIONS	REFERENCE (CHAP 11)	DATA AVAILABLE
USA	THL	SE		Break flow at T-junctions, 900 psia simulating LOFT L3-5, L3-6	2 tests, top and bottom of pipe, to be selected by US	✓ ✓	(r) 1	YES
	G-2	SE	716	800 psia, low power boil-off test		✓	(s) 1, 2	YES
			718	800 psia, high power boil-off test		✓		YES
			736	15.3 psia, low power boil-off test		✓		YES
FRANCE	PATRICIA GV-1	SE	109-110	70 bar, 400 $Kg/m^2.sec$, X = 0 - 0.06	SS tests of SG primary, including reflux condensing	✓	(t) 2	YES
			246-247	70 bar, 400 $Kg/m^2.sec$, X = 0.25 - 1.0		✓		YES
	PATRICIA GV-2	SE	6-4.2	56 bar, 502 $Kg/m^2.sec$, Xout = 0.19	SS test of SG secondary	✓	3	YES
ITALY	GEN 3x3	SE	GTPU-08	SG test with variation of secondary side flow and enthalpy, and primary side temperature		✓	(u) 2	YES
	PRESSURIZER FLOODING	SE	106/70	CCFL in Surge Line, vertical pipe		✓	(v) 2	YES
			221/70	CCFL in Surge Line, horizontal pipe		✓		YES

Notes: 1. HL and combined injection tests are not necessary for countries with only CL injection plants.

2. For actual plants data are limited and unreliable, hence conclusions from analyses may be limited.

TABLE 3

SMALL AND INTERMEDIATE LEAKS IN PWR's WITH OTSG

COUNTRY	FACILITY	INTEGRAL OR S.E.	TEST NO.	BRIEF DESCRIPTION	COMMENTS	ACCEPTANCE CONDITIONS	REFERENCE (CHAP 11)	DATA AVAILABLE
Germany	GERDA	I	1605AA	10 cm^2 reactor vessel break	entire transient covered (single phase, natural circulation, intermittent circulation, boiler condenser mode, loop refill)		(y) 1	NO
		S	070100	10 cm^2 CL-suction break	natural circulation in reactor vessel and loop during cooldown			
		S	0900AA	Steady boiler condenser mode	no break, no HPIS-injection			
USA	MIST	I	320201	50 cm^2 cold leg break (SBLOCA)	steam generator heat transfer	✓		YES
			330100	PORV lift, feed and bleed experiment	pressuriser T/H	✓		YES
	UMCP	I	NC-2	Two-phase and one-phase natural circulation				YES
			NC-5	Asymetric flow between loops				YES
	OTIS	I	220100	Two-phase natural circulation and reflux				YES
			220402	Cold leg small break LOCA				YES
	SRI2	I	?	Small break LOCA (to be specified when test matrix formalized)				NO

TABLE 4

PLANT TRANSIENTS

COUNTRY	FACILITY	INTEGRAL or S.E.	TEST NO.	BRIEF DESCRIPTION	COMMENTS	ACCEPTANCE CONDITIONS	REFERENCE (CHAP 11)	DATA AVAILABLE
USA	LOFT	I	L6-7/L9-2	Turbine Trip cooldown	Simulation of ANO-2 transient	✓	(e) 13	YES
			L9-3	Loss-of-Feedwater, ATWS		✓	14	YES
			FW-1	Loss-of-Feedwater, primary 'feed and bleed'	International Consortium test	✓	15	NO
	ANO-1 Unit 2	I	-	Turbine Trip Cooldown		✓	(w) 2	NO
JRC Ispra	LOBI	I			To be included at a later date			
France	BETHSY	I						
Italy	SPES	I						
Japan	ROSA-IV	I						

Notes: 1. For phenomena requiring SE tests, eg pressurizer behaviour, refer to Small Break table.

2. Valve flow behaviour will be strongly design-dependent, and specific experimental data should be used if possible.

7. BWR FACILITIES

7.1 Plant Results

(a) Kernkraftwerk Leibstadt (KKL)

The Nuclear Power Plant, Leibstadt (KKL) is a General Electric direct cycle BWR/6-Mark III type boiling water reactor. It is designed for a net power of 942 MW_e. The reactor core consists of 648 fuel bundles and 84 control rods. Each fuel bundle contains 62 fuel rods and two water rods in an 8 x 8 array.

Two start-up tests, a main steam isolation valve (MSIV) test and a total loss of feedwater with HPCS unavailable test have been included in the Transient matrix.

The purpose of the main steam isolation valve test is to determine the boiling water reactor transient behaviour that results from the simultaneous full closure of all MSIVs. The specific phenomenon which can be observed in this test is the level drop due to void collapse.

The purpose of total loss of feedwater with HPCS unavailable test is to demonstrate the capability of the Reactor Core Isolation Cooling (RCIC) system to maintain reactor water level above level 1 following a total loss of feedwater whereby the HPCS system is not available. The phenomenon which can be observed in this test is the liquid level decrease.

(b) Peach Bottom-2

The Peach Bottom-2 Reactor is a General Electric BWR/4 with a rated power of 3293 MW. The core contains 764 fuel assemblies and 185 control rods. The plant is well instrumented with 43 in-core detector strings, each having local power range monitors at four axial levels. Each of the detector strings can be entered by a travelling in-core probe to record detailed axial power distribution. Pressure measurements are available in the steam dome, core exit plenum and steam lines. Differential pressure measurements are used to measure steam, recirculation and feedwater flows as well as pressure drop through the core and pressure vessel collapsed liquid level. Temperature measurements and signals from the turbine stop valve and scram relay are also available.

Three turbine trip tests were performed in April 1977, at

(1) 47.4 per cent rated power, 98.8 per cent rated flow;
(2) 61.6 per cent rated power, 80.9 per cent rated flow;
(3) 69.1 per cent rated power, 99.4 per cent rated flow.

These tests were all initiated by a manually triggered closure of the turbine stop valves with the reactor protection system delayed to allow a power escalation (due to pressure-void-reactivity feedback) and scram initiation on high average power range monitor (APRM) power level. The APRM scram level had been lowered to a limit that would guarantee a satisfactory minimum critical power ratio during the transient. At the time of the tests, end-of-cycle 2, 576 of the fuel assemblies were of the original 7 x 7 type and 188 were of the reload 8 x 8 type. The third turbine trip test has been included in the Transient matrix.

(c) Philippsburg 1 (KPP 1)

The KPP 1 reactor is a KWU-BL69 type BWR with a rated power of 2575 MW_{th}. Core flow is driven by nine recirculation pumps located inside the pressure vessel. More than 20 commissioning tests were recorded on digital tape -- 67 analog channels including 20 Local Power Monitoring signals.

A "trip of all recirculation pumps without scram" starting from rated core flow and 75 per cent of rated power has been included in the matrix. (Initial run down speed 50 per cent of rated speed/sec.) The purpose of this test is to determine the BWR transient behaviour resulting from a fast core flow reduction. The specific phenomena which can be observed in this test are:

- fast decrease of power from 75 per cent to appr. 25 per cent of rated
- fast increase of water level in the pressure vessel by appr. 50 cm due to the increase of void fraction in the core plenum and separator standpipes.

7.2 System Experiments

(a) TBL

The dimensions of the Two Bundle Loop (TBL) facility (volume scaling ratio = 1/328) and the height scaling ratio (1 : 1) make TBL a very useful apparatus to simulate LOCAs in BWRs.

Four experiments have been selected from the TBL programme to be included in the LOCA matrix: two large break LOCAs (in the steam line and in the recirculation line) and two small break LOCAs (in the steam line and in the recirculation line). In all cases, the intervention of ECCS, and therefore the reflood period, is considered during the test.

(b) ROSA-III

The ROSA-III facility is characterised by a short core (height = 1.88 m) and by the availability of four bundles which allow (at least partially) the simulation of the multi-dimensional spatial effects occurring in the core of a real BWR. The parallel channel oscillation and the natural circulation between hot and cold bundles can be mentioned in this regard.

The recirculation loops, including the jet pumps, are completely external to the main vessel.

Two large break LOCAs, one intermediate break LOCA and two small break LOCAs (run 912, used by CSNI as International Standard Problem 12 and run 984, which is a counterpart test to FIST and TLTA tests) have been selected from the ROSA-III test programme to be included in the LOCA matrix.

Three transient tests are included in the ROSA-III test matrix: runs 919, 923 and 971.

The power decay curve adopted during LOCA tests has been used in the first two runs, while a controlled power decay was calculated for run 971 based on neutronic feedback. Run 971 was initiated by MSIV closure and is considered to be the most interesting for code assessment purposes.

In the Transient Matrix both runs 971 and 919 have been chosen; the latter is characterised, with respect to run 923, by the absence of the vessel Pressure Control System (PCS).

(c) TLTA

The Two-Loop Test Apparatus (TLTA) is a 1 : 624 volume scaled BWR/6 simulator. It was the predecessor of the better scaled FIST facility which used some of its components.

The facility was capable of full BWR system pressure and had a simulated core with a full size 8 x 8, full power single bundle of indirect electrically heated rods. All major BWR systems are simulated including lower plenum, guide tube, core region (bundle and bypass), upper plenum, steam separator, steam dome, annular downcomer, recirculation loops and ECC injection systems. The fundamental scaling consideration was to achieve real-time response. A number of the scaling compromises present in TLTA were corrected in the FIST configuration. These compromises include a number of regional volumes and component elevations. Several TLTA and FIST counterpart tests were performed to assist in understanding the effects of these compromises. The system was significantly modified nine times to accomplish specific test objectives.

The experimental programme was conducted from 1975 to 1980, and included investigations of fuel bundle variation (7 x 7 and 8 x 8), BWR/4 and BWR/6, baseline data with and without ECC, small breaks, large breaks, bundle uncovery and boil-off separate effects experiments.

Three TLTA experiments are included in this validation matrix; two small break LOCAs, one of which is a FIST and ROSA-III counterpart, and one large break LOCA.

(d) FIST

The FIST (Full Integral Simulation Test) facility is a 1 : 624 volume scaled BWR/6 simulator. It is capable of full BWR system pressure and has a simulated core with a full size, full power single bundle of indirect electrically heated rods. Other key features include:

(1) full height test vessel and internals;
(2) correctly scaled fluid volume distribution;
(3) simulation of all ECC systems, safety relief valves and the automatic depressurisation system;
(4) level trip system, and
(5) heated feedwater supply system to allow for steady state operation.

The experimental programme was performed in two phases with phase I being performed in 1983 and phase II in 1984. Phase I involved eight tests including large break LOCAs, natural circulation and power transients. Phase II involved nine tests including BWR/6 LPCI line break, BWR/6 intermediate size recirculation break, BWR/4 LPCI large break, steady state natural circulation tests with feedwater makeup performed at high and low pressure and at high pressure with HPCS makeup, a transient without control rod insertion, a transient with controlled depressurisation, a simulation of not maintaining water level, and a simulation of the Peach Bottom turbine trip test.

Ten FIST experiments were selected for inclusion in the BWR validation matrices: five LOCAs and five transients.

(e) FIX-II

The FIX-II facility is a volume scaled (1 : 777) representation of a Swedish BWR with external pumps. The pressure vessel contains a 36 rod full length bundle and a spray condenser at the top to allow steady state operation. The downcomer, bypass channels and guide tube volumes are represented by external piping. The intact loop represents three of the four external reactor loops. The broken loop is constructed such that both guillotine breaks and split breaks may be simulated. The facility is equipped with ADS-simulation, but no ECCS injection systems are included.

The FIX-II loop is also used to investigate response of pump trips and MSIV closures in internal pump reactors. These experiments were conducted with a blind broken loop and pump coast-down characteristics representing low pump inertia.

The test results show that the cooling modes are dependent on break size. One typical intermediate size break with positive core flow (test 3025) and one typical guillotine break with large negative core flow (test 5052) were selected, in addition to a 100 per cent split break which gives the highest clad temperature (test 3061).

Two transient experiments were selected; one from each test series. The selection (tests 2032 and 6261 from the pump trips and MSIV closures respectively), highlights assessment of a code's capability to predict transient dryout, post dryout clad temperatures and rewet of a high power channel.

(f) PIPER-ONE

PIPER-ONE is a BWR simulator with a volume scaling ratio of 1/2200 and a height scaling ratio of 1/1. The one-dimensionality of volumes, the absence of recirculation loops (in the present version) and the availability of a structures cooling/heating system, make the facility mainly suitable for SBLOCA studies. In particular, the structures cooling system makes it possible to remove at specific times in a transient, fixed amounts of thermal power from the piping walls in order to simulate the energy balance of the fluid which would occur in the prototype plant.

This considered, only small and intermediate break LOCAs can be selected from the PIPER-ONE programme for consideration in the matrix.

7.3 Separate Effects Tests

(a) SSTF

The 30° Sector Steam Test Facility (SSTF) utilises a full scale 30 degree sector of a BWR/6 with 58 complete or partial fuel bundle simulators. The fuel bundle simulators also use individual vapour injectors, and actual reactor hardware is extensively used to assure typicality of the results.

There were two categories of tests carried out with this facility. The first group of tests were separate effect experiments, investigating the refill/reflood phenomena under quasi-steady state conditions. In these tests, component CCFL and plenum mixing characteristics were emphasised. Also bypass mixing and channel wall heat transfer were measured by injecting steam into the bottom of the region from the guide tubes, establishing a two-phase mixture in the bypass, and then injecting sub-cooled water through the LPCI.

The second group of experiments were system response tests, investigating the refill/reflood performance during a blowdown transient from 10 bars. Mainly, the effects of ECC systems, LPCI effectiveness and effects of break size were tested and compared with base case experiments.

Four tests have been selected from SSTF experimental programmes to be included in the validation matrix, emphasising ECC bypass, CCFL at upper tie plates and plenum mixing and spray distribution.

(b) GEST-SEP/GEST-GEN

GEST-SEP and GEST-GEN are two experimental facilities to simulate the separators in a PWR steam generator and the overall steam generator behaviour respectively. The GEST-GEN loop is able to exchange up to 20 MWt between primary and secondary sides.

Both apparatus have been included in this matrix due to their relatively large dimensions, and also to the difficulties in obtaining reliable data on separator and dryer performance from BWR integral facilities.

The specific experiments to be included in the LOCA matrix will be determined after full operation of GEST-GEN begins (scheduled for 1986).

(c) GÖTA

The GÖTA facility was used to investigate modes of ECCS injection. It has two pressure vessels: a test vessel and a pressuriser which also serves as a water reservoir for the injection systems. The test bundle consists of 64 full length, electrically heated rods in a canister. Outside the canister there is a shroud which allows simulation of bypass behaviour. ECCS water may be injected at four locations: downcomer, lower plenum, bundle spray and bypass spray. Tests were conducted by heating the bundle to a specified maximum clad temperature followed by ECCS injection. The initial temperature distribution is useful for validation of radiation calculations. The subsequent ECCS injection highlights quench behaviour.

Two separate effects tests are selected: one with lower plenum injection only (21), and one with essentially top spray (23).

(d) HDR

The Heißdampfreaktor (HDR) is a decommissioned BWR and therefore offered the possibility to run both feedwater line (FWL) -- and steam line blowdown experiments under typical reactor initial thermodynamic conditions and partly using prototypical piping, fluid controlled FWL-Check Valves (FCV) and Main Steam Line -- Isolation Valves (MSIV). Initial thermodynamic conditions, closing/damping characteristics of the FCV and delay time of the MSIV were parametrically varied. Measurements include break flow, two phase jet behaviour and containment loading, transient temperature and pressure wave phenomena including the structural response of the system and the closing function of the valves. The swell level behaviour is indirectly determined. The LOCA experiments were chosen with particular emphasis on break flow and transient loading of the piping. In addition, transient experiments include FCV and MSIV response.

(e) MARVIKEN

The facility consists of a 425m^3 vessel equipped with a discharge pipe at the bottom. Nozzles of various lengths and diameters could be attached to the discharge pipe. Transient conditions were controlled by establishing specified vertical temperature distributions in the vessel water. Most of the tests were conducted from an initial pressure of 5 MPa. Steam line breaks were simulated by introducing a standpipe inside the vessel which connects the steam volume to the discharge pipe.

One of the two tests selected (CFT-22) gives critical flow data in the range from 50° C sub-cooled to saturated. This is the largest span for one single test. The other test is a steam line break (JIT-11) which gives high steam quality critical flow data and a large level swell. Density distributions in the vessel were recorded by local differential pressure measurements.

(f) NEPTUN

In the NEPTUN test facility, bundle boil-off and reflood experiments were performed. The data obtained from NEPTUN I and II experiments are useful in assessing computer codes to predict thermal-hydraulic response during reflooding and also during core uncovery conditions. Emphasis mainly lays on level behaviour and on entrainment phenomena.

The NEPTUN heater rod bundle contains 37 rods in a 15 x 15 PWR configuration. Each heater rod has an axial heated length of 1.68 m and the bundle is contained in a housing. A continuously variable axial profile is used. The instrumentation allows the measurement of cladding (at eight equidistant axial levels), housing, thermal insulation and coolant temperatures, absolute and differential pressures at several axial positions, flow rates, carry-over rates and heating power. System pressure is between 1 and 5 bars.

Three tests have been selected from the NEPTUN test programme, one core uncovery (boil-off) and two reflooding tests to be included in the validation matrix.

(g) Karlstein Recirculation Pump Tests

Original internal recirculation pumps for KWU-PL 69 -- BWRs (max. power 800 kW) and KWU-PL 72 -- BWRs (max. 1300 kW) have been tested under single phase flow conditions, using two different types of impellers (axial and mixed flow) in a comprehensive full-scale pump test programme. The tests covered a wide range of temperatures (50° C - 279° C), pump speeds (600 - 2000 rpm) and suction heads (-0,50 H 1,75, where $H = 2. \cdot g \cdot NPSH/u^2$ and u = velocity at impeller outlet).

These tests give clear information about reduction of pump head and flow under cavitation.

(h) UPTF

The Upper Plenum Test Facility (UPTF) is a full size 3D simulation of the Grafenrheinfeld German PWR with steam and water to be supplied from a fossil fueled conventional power plant. There are 30 experiments included in the test matrix scheduled to be run between 1986 and 1988.

(i) CISE-Safety Valve Experiments

The actuation of valves represents one of the main features of the transients in nuclear power plants. In particular, the flow rate through the valves may substantially affect the evolution of off-normal events. For this reason the assessment of code capabilities in describing valve behaviour is of interest and a specific experiment has been included in the matrix.

The facility consists of a large scale safety valve (scaling ratio = 1/7.4 with respect to a safety valve adopted in a commercial PWR). high pressure piping and instrumentation.

A vessel containing high pressure (up to 12 MPa), high temperature (up to saturation) fluid supplies water to the valve. The mass flow rate of the water is measured.

Two series of steady state tests consisting of 28 experiments (and a similar number of measurement points) have been considered in the matrix. The two series are characterised by the fluid quality at the valve inlet which ranged between 0.25 and 0.75 in the first and is 0 in the second.

8. BWR VALIDATION MATRICES

This section contains the BWR matrices and tables.

The matrices described in Section 3 relating phenomena, test type and transients are given in Matrices 5 and 6. The lists of selected tests are given in Tables 5 to 10, the structure of which is self-explanatory.

Matrix 5

Cross referencu matrix for LOCAs in BWRs

- Phenomena versus test type
  - ● occurring
  - ○ partially occurring
  - – not occurring
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not suitable for code assessment
  - × expected to be suitable
- Test type versus test facility
  - ● already performed or planned until 12/85
  - ○ performed, but of limited use
  - – not performed or planned

		Test Type: Blowdown						Test Facility: System Tests							Test Facility: Separate Effects Tests							
		Large Steam Line Break with Fast Depressurization.	Large Break Below Water Level with Fast Depress.	Small Break without Depress. before ADS Activity.	Intermediate Break with Slow Depress.	Spray Line Break.	Refill - Reflood.	BWR (f) 1 : 1(g).	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 I 1 : 2200 1 Chan., Full Height.	SSTF 30° Sector of PV.	GEST GEN	COTA	HDR	MARVIKEN	NEPTUN 1 Channel, 6x6 Bundle, Half Length (h).	Karlstein Tests, Internal Recirc. Pump, Full Scale.	UPTF (i)
Phenomena	Break Flow.	●	●	●	●	●	○	–	○	○	○	○	○	×	–		–	●	●	–	–	
	Channel and Bypass Axial Flow and Void Distribution.	●	●	●	●	●	●	○	●	○	●	●	●	×	–		–	–	–	–	–	
	Corewide Radial Void Distribution (a).	○	○	●	●	●	●	○	○	●	○	○	○	×	–		–	–	–	–	–	
	Parallel Channel Effects - Instabilities.	–	–	●	●	●	●	–	○	●	–	–	–		–		–	–	–	–	–	
	ECC Bypass.	–	–	○	○	○	●	–	○	○	○	○	–	*	●		–	–	–	–	–	
	CCFL at UCSP and Channel Inlet Orifices.	○	●	–	●	●	●	–	○	○	–	○	○		●		○	–	–	–	–	
	Core Heat Transf. incl. DNB, Dryout, RNB, Surf. to Surf. Rad.	●	●	○	●	○	●	–	●	●	●	●	●	×	–		●	–	–	●	–	
	Quench Front Propagation for both Fuel Rods and Channel Walls.	–	–	–	–	–	●	–	●	●	●	●	–	×	–		●	–	–	●	–	
	Entrainment and Deentrainment in Core and Upper Plenum.	●	●	○	○	○	●	–	–	○	○	○	–		○		–	–	–	●	–	
	Separator Behavior incl. Flooding, Steam Penetration and Carryover.	●	●	○	○	○	–	○	●	○	○	●	○		–	×	–	–	–	–	–	
	Spray Cooling.	–	–	○	○	○	●	–	○	○	○	○	–	×	●		●	–	–	–	–	
	Spray Distribution.	–	–	○	○	○	●	–	–	○	–	–	–		●		–	–	–	–	–	
	Steam Dryer - Hydraulic Behavior.	●	–	○	○	–	–	○	○	○	○	○	–		–	×	–	–	–	–	–	
	One and Two Phase Pump Recirc. Behavior incl. Jet Pumps. (b)	○	○	●	●	●	–	○	○	○	○	○	○		–		–	–	–	–	○	
	Phase Separation and Mixture Level Behavior.	●	●	●	●	●	●	–	○	●	○	●	●	×	–		–	○	●	●	–	
	Guide Tube and Lower Plenum Flashing.	●	●	–	○	○	–	–	●	●	●	●	●	×	–		–	–	–	–	–	
	Natural Circulation - Core and Downcomer.	–	–	●	○	○	●	●	●	○	○	●	●	×	–		–	–	–	–	–	
	Natural Circulation - Core Bypass, Hot and Cold Bundles. (c)	–	–	●	○	○	●	–	○	○	○	○	○		–		–	–	–	–	–	
	Mixture Level in Core.	–	–	●	○	○	●	–	●	●	●	●	●	×	–		–	–	–	●	–	
	Mixture Level in Downcomer.	●	●	●	●	●	●	●	●	○	○	●	●	×	–		–	–	–	–	–	
	ECC Mixing and Condensation. (d)	–	–	●	○	●	●	–	○	○	○	○	–		○		–	–	–	–	–	×
	Pool Formation in Upper Plenum.	○	○	–	○	○	●	–	○	○	○	○	○		●		–	–	–	–	–	
	Structural Heat and Heat Losses.	○	○	○	●	●	●	–	●	○	○	○	○	×	–		–	–	–	–	–	
	Phase Separation in T - Junction and Effect on Break Flow. (e)	–	–	●	○	●	–	–	–	–	–	–	–		–		–	–	–	–	–	
Test Facility System Test	BWR	–	–	–	–	–	–															
	TBL	●	●	●	●	–	●															
	ROSA III	●	●	●	●	–	●															
	TLTA	●	●	–	●	–	●															
	FIST	●	●	●	●	–	●															
	FIX 2	–	●	–	●	–	–															
	PIPER I																					

a) Experiments performed in single bundle facilities at different power levels can supply valuable information for assessment of this phenomenon.

b) No two-phase data for full scale and jet pumps could be identified

c) No completely sufficient data base could be identified

d) Data are provided by SSTF for the upper plenum, and by UPTF for downcomer

e) Can be assessed using THL-data from PWR Validation Matrix

f) These are non-LOCA data but may be used for assessment

g) Volumetric scaling

h) Deentrainment not addressed

i) Structurally different from BWR

Phenomena included in this matrix may be also important for transients.

Matrix 6

Cross reference matrix for transients in BWRs

- Phenomena versus test type
  - ● occurring
  - ○ partially occurring
  - – not occurring
- Test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - – not suitable for code assessment
  - × expected to be suitable
- Test type versus test facility
  - ● already performed or planned until 12/85
  - ○ performed but of limited use
  - – not performed or planned

		Test Type									Test Facility: System Tests				Test Facility: Sep. Eff. Tests			
		Stationary Test Measuring Power Flow Map.	Recirculation Pump Trip.	Core Stability.	Loss of Main Heat Sink.	Feedwater Flow or Temperature Disturbance a. 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(d).	ROSA III 1 : 424 4 Channels.	FIST 1 : 624 1 Chan., Full Pow., Full Height.	FIX 2 1 : 777 1 Chan., Full Pow., Full Height.	HDR.	CISE-Valve Exp.	UPTF (e).	Karlstein Recircul. Pump Test.
Phenomena	Natural Circulation in One- and Two-Phase Flow.	●	●	●	●	–	–	–	●	●	●	○	●	○	–	–		
	Collapsed Level Behavior in Downcomer.	–	●	○	●	●	●	●	●	●	●	○	●	●	–	–		
	Core Thermal Hydraulics.	○	●	●	●	○	○	○	●	●	○	●	●	●	–	–		
	Valve Leak Flow (a)	–	–	–	●	–	–	–	●	●	○	○	○	–	●	●		
	Single Phase Pump Behavior. (b)	○	●	○	●	○	○	●	●	●	○	○	○	○	–	–		●
	Parallel Channel Effects and Instabilities.	–	●	●	○	–	–	–	●	●	○	●	–	–	–	–		
	Nuclear Thermalhydraulic Feedback Including Spatial Effects.	○	○	●	–	○	○	○	●	–	●	–	–	–	–	–		
	Nuclear Thermalhydraulic Instabilities.	–	○	●	–	–	–	○	●	–	●	–	–	–	–	–		
	Downcomer Mixing.	–	–	–	–	●	●	–	●	●	○	○	–	–	–	–	×	
	Boron Mixing and Distribution. (c)	–	–	–	–	–	–	–	●	–	–	–	–	–	–	–		
	Steam Line Dynamics.	–	–	–	●	–	–	●	●	●	○	–	○	–	●	–		
	Void Collapse and Temp. Distribution During Pressurization.	–	–	–	●	–	–	–	●	●	○	●	●	×	–	–		
	Critical Power Ratio.	–	●	●	●	●	●	●	●	●	○	●	●	●	–	–		
	Rewet after DNB at High Press. and High Power Incl. High Core Flow.	–	●	–	●	–	–	○	●	○	–	○	●	●	–	–		
	Structural Heat and Heat Losses.	–	○	–	○	–	○	○	○	○	–	○	○	○	–	–		
Test Facility: System Tests	BWR	●	●	●	●	●	●	●	–	○								
	TBL	–		–														
	ROSA III	–		–														
	FIST	–	○	–	●	–	●	●	○	●								
	FIX 2	–	●	–	×	–	–	–	–	–								

Phenomena included in LOCA reference matrix may be also important.

a) Relevant data were identified for only one type of feedwater check valve, one type of main isolation valve and one type of safety valve

b) Two-phase pump behavior is of interest for certain special ATWS and inadvertent increase of steam flow transients

c) No data base could be identified

d) Volumetric scaling

e) Structurally different from BWR

Table 5: LOCA - BWR Plant Results

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
-	-	-	no data available		-	-

*) related to BWR Plant specific References, see chapter 12 (a)-(c)

Table 6: LOCA - System tests

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
Japan	TBL	108	200 % break of recirculation line	incl. reflood period	(d)2	no
		314	200 % break of steam line	incl. reflood period	(d)3	no
		311	small break in recirculation line	incl. reflood period	(d)4	no
		309	small break in steam line	incl. reflood period	(d)3	no.
Japan	ROSA-III	912	5 % break in recirculation line	incl. reflood period	(e)2	yes
		926	200 % break of recirculation line	no HPCS	(e)3	yes
		952	100 % break in steam line	full ECCS	(e)4	yes
		916	50 % break in recirculation line	no HPCS	(e)5	yes
		984	2.8 % break in recirculation line	no HPCS, counterpart test to FIST test 6SB2C and TLTA test 6432/R1	(e)6	yes
USA	TLTA	6432/R1	64,45 cm² small break	no HPCS, counterpart test to FIST test 6SB2C and ROSA-III test 984	(f)2	yes
		6431/R1	64,45 cm² small break	full ECCS, without ADS activation	(f)2	yes
		6525/R2	200 % break	all systems operational	(f)3	yes
USA	FIST	4DBA1	200 % break of recirculation suction line		(g)3	yes
		GIB1	intermediate break		(g)4	yes
		6SB2C	46,45 cm² small break in recirculation suction line of BWR/6	no HPCS, counterpart test to ROSA-III test 984 and TLTA test 6432/R1	(g)3	yes
		6SB1	46,45 cm² small break in recirculation suction line of BWR/6	no HPCS, SRV stuck open	(g)3	yes
		6MSB1	200 % break of steam line upstream of flow limiter		(g)3	yes
Sweden	FIX-II	3025	31 % break in recirculation line	ISP-15	(h)3,4	yes
		5052	200 % break in recirculation line		(h)5	yes
		3061	100 % break in recirculation line	test with maximum clad temperature	(h)5	yes
Italy	Piper-one	-	tests will be specified after experiments have been performed	facility suitable for small break and intermediate break LOCA	(i)1	-

*) related to System Experiment Specific References, see chapter 12

Table 7: LOCA - Separate Effects Tests

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
USA	SSTF	SE5-2A	ECC bypass		(j)3,6	no
		SE1	CCFL at upper tie plate		(j)3,6	no
		SE3-8A 343	Spray distribution		(j)3,6	no
		SRT-3 Run26	Parallel channel test		(j)4,5,6	no
Italy	GEST-SEP	-	tests will be specified after data become available	facility suitable for assessment of separators and dryers	(k)1	-
Sweden	GÖTA	21	bottom reflood test	tests 21 and 23 are performed at	(l)1,2	yes
		23	top spray test	1 bar, special emphasis on - radiation heat transfer - rod quenching	(l)2	yes
Germany	HDR	V 45	steam line break		(m)3	no
		V 21.2	feedwater line break		(m)3	no
Sweden	MARVIKEN	CFT-22	500 mm diameter break	subcooled critical flow	(n)3	yes
		JIT-11	300 mm diameter steam line break	high steam quality critical flow, level swell	(n)4	yes
Switzerl.	NEPTUN	5007	core uncovery (boil off) test	level behaviour	(o)2	yes
		5050	reflooding test, high flooding rate	special emphasis was given in tests	(o)3	yes
		5052	reflooding test, low flooding rate	5050 and 5052 to entrainment and deentrainment	(o)3	yes
Germany	KARLSTEIN	Series 1	degradation of pump head as a function of NPSH and fluid	axial flow impeller	(p)1,2	no
	Pump Tests	Series 2	temperature in full scale recirculation pumps	mixed flow impeller	(p)1,2	no
Germany	UPTF	-	tests will be specified after experiments have been performed		(q)1	-

*) related to Separate Effects Specific References, see chapter 12

Table 8: Transients - BWR Plant Results

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
Switzerl.	LEIBSTADT	STP-25	main steam isolation valve (MSIV) closure	start up test	(a)1	yes**)
		STP-2001	total loss of feedwater and HPCS unavailable	start up test	(a)2	yes**)
USA	PEACH BOTTOM 2	Test Nr. 3 April 1977	turbine trip test		(b)1	no
Germany	Philippsburg I	Test 24th Aug. 1979	recirculation pump trip test	start up test	(c)1,2	no

*) related to BWR-Plant References, see chapter 12

**) test data are open available, but plant description information is limited

Table 9: Transients - System Tests

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
Japan	ROSA-III	971	MSIV closure, pump coastdown	void reactivity	(e)7	yes
		919	reference transient	usual decay curve	(e)8	yes
USA	FIST	4PTT1	Peach Bottom turbine trip test simulation		(g)4	yes
		T23C	water level test (loss-of-feedwater)		(g)4	yes
		T1QUV	water level test, without ECC or ADS		(g)4	yes
		GPMC3	power transient with controlled depressurization (ATWS)		(g)4	yes
		GPMC1	BWR/6 steam line isolation valve closure without power scram		(g)3	yes
Sweden	FIX-II	2032	Pump trip in internal pump reactor		(h)6	no
		6261	MSIV closure		(h)7	no

*) related to System Experiment Specific References, see chapter 12

Table 10: Transients - Separate Effects Tests

COUNTRY	FACILITY	TEST No.	BRIEF DESCRIPTION	COMMENTS	REF. *)	DATA Avail.
Germany	HDR	PHDR-DIV-V23.3	investigation of steam isolation valve		(m)4	no
		PHDR-SRV 350-V60.6	investigation of feedwater in non-return valve		(m)5	no
Italy	SIET/SAFETY VALVE	1 - 19	steady state tests p_{inlet} = 6., 9., 12. MPa x_{inlet} = 0.25, 0.5, 0.75	two-phase tests	(r)2	no
		20 - 28	steady state tests p_{inlet} = 6., 9., 10. MPa x_{inlet} = 0	saturation water at valve inlet	(r)3	no
Germany	UPTF	-	test will be specified after experiments have been performed		(q)1	-
Germany	KARLSTEIN Pump Tests	Series 1	degradation of pump head as a function of NPSH and fluid temperature in full scale recirculation pumps	axial flow impeller	(p)1,2	no
		Series 2		mixed flow impeller	(p)1,2	no

*) related to Separate Effects Specific References, see chapter 12

9. CONCLUSIONS

A systematic study has been carried out to select experiments for thermal-hydraulic system code assessment.

Matrices have been established to identify, first, phenomena assumed to occur in LWR plants during accident conditions and secondly, facilities suitable for code assessment (Chapter 3). Tables identify the experiments selected for validation of computer codes (Chapters 4 to 8).

To assess a code for a particular LWR plant application, it is recommended that the list of tests in the relevant matrix be viewed as the minimum necessary for Independent Assessment. the Task Group has assumed that the selection of individual models in the code has been adequately justified during the Developmental Assessment phase by the code developers in comparison with appropriate separate effects tests.

A periodic updating of the matrices will be necessary to include findings from new facilities and improved understanding of existing data as a result of further assessment. The matrices also permit identification of areas where further research may be justified.

Criteria have not been addressed by which to judge code performance and, ultimately, the assessment of uncertainties in plant calculations.

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13. LIST OF TASK GROUP AND COMMITTEE MEMBERS INVOLVED IN THE PREPARATION OF THIS REPORT

S.N. Aksan	Swiss Federal Institute for Reactor Research (EIR) CH-5303 Würenlingen, Switzerland
D. Bessette	Nuclear Safety Division OECD Nuclear Energy Agency 38 boulevard Suchet F-75016 Paris, France
I. Brittain	Atomic Energy Establishment Winfrith, UKAEA Dorchester, Dorset DT2 8DH United Kingdom
F. D'Auria	Facolta d'Ingegneria Dipartimento di Costruzioni Meccaniche e Nucleari Via Diotisalvi 2 I-56100 Pisa, Italia
P. Gruber	Kraftwerk Union A.G. Hammerbacher Strasse 12+14 D-8520 Erlangen, Federal Republic of Germany
H.L.O. Holmström	Nuclear Engineering Laboratory Technical Research Centre of Finland P.O. Box 169 SF-00181 Helsinki, Finland
R. Landry	Nuclear Safety Division OECD Nuclear Energy Agency 38 boulevard Suchet F-75016 Paris, France
S. Naff	European Attaché for USNRC Kraftwerk Union A.G., R15 Hammerbacher Strasse 12+14 D-8520 Erlangen, Federal Republic of Germany
R. Pochard	Departement de Sureté Nucléaire Centre d'Etudes Nucléaires B.P. No. 6 F-92260 Fontenay-aux-Roses, France
G. Preusche	Kraftwerk Union A.G. Hammerbacher Strasse 12+14 D-8520 Erlangen, Federal Republic of Germany
M. Reocreux	Département d'Etudes et Recherche en Sécurité SEAREL, CEN/FAR B.P. No. 6 F-92265 Fontenay-aux-Roses, France
O. Sandervag	Studsvik Energiteknik AB S-61182 Nyköping, Sweden

H. Städtke	C.E.C. - Joint Research Centre Ispra Establishment I-21020 Ispra/Va, Italy
K. Wolfert	Gesellschaft für Reaktorischerheit (GRS) m.b.H. Forschungsgelände D-8046 Garching, Federal Republic of Germany
N. Zuber	US Nuclear Regulatory Commission Washington, DC 20555