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Plans for Assessment of Best Estimate LWR Systems Codes

U.S. Nuclear Regulatory Commission

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S. Fabic
P. S. Andersen, Consultant

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S. Fabic
P. S. Andersen, Consultant

Division of Accident Evaluation
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

Systems codes play a very important role in evaluation of safety of nuclear power plants. In contrast to the Evaluation Model codes in which the most pessimistic and conservative combination of events and processes are assumed, the Best Estimate systems codes attempt to describe the physical processes as realistically as possible. They are, therefore, amenable to an indepth assessment through confrontation with experimental evidence gathered, worldwide, in the course of reactor safety research.

That confrontation has many facets and the purpose of this report is to describe the issues, considerations, and a recommended course.

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1. INTRODUCTION

The Best Estimate systems codes are designed to provide realistic, rather than conservative, predictions of LWR plant behavior during a variety of accidents and transients. This report describes the manner in which the predictive capabilities of such computer codes can be assessed.

The final goal the code assessment is to arrive at is a qualified judgment of the accuracy with which the code can predict full-scale LWR plant accidents and transients.

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

a. Establish the accuracy in predicting the best available integral system data both from full-scale LWR plants and from sub-scale test facilities.

b. Determine whether relevant thermal-hydraulic phenomena are modeled well enough to justify application of the code to LWR plant conditions for which full-scale data are unavailable. This task implies application of the code to Separate Effects Tests and Basic Tests.

c. Using the results from tasks 1 and 2 above, estimate the accuracy in predicting full-scale LWR plant accidents and transients.

The plant upset conditions for which the code is intended to provide reliable predictions are of practically unlimited variety. This is particularly true of the most important of the intended applications, namely to small breaks and transients, where multiple failures and operator actions are of the essence. Therefore, one cannot possibly hope to compare code predictions with integral systems data for all intended applications of the code. With very few exceptions, the available full-scale plant data are so incomplete and of so little challenge to the codes as to barely qualify for use in code assessment where code accuracy, rather than just trends, needs to be quantified. On the other hand, the integral systems test facilities, from which challenging data are available, are inflicted with various atypicalities.

It follows that it is not enough to confront the codes with the available integral systems data (assessment task number 1). One needs reasonable assurance that the code is not just a black box tuned to particular data. This assurance can only be obtained by examining, in detail, the performance of the codes when also applied to data from separate effects and basic thermal-hydraulic tests (assessment task number 2).

The final outcome of the code assessment process will be a qualified judgment of the accuracy with which the code can predict full-scale LWR plant accidents and transients (assessment task number 3).

Whether the code accuracy, thus quantified, is "good enough" depends upon the code mission: The same code might be acceptable for some applications while unacceptable for others, so that different acceptance criteria might have to be developed for different missions. However, code assessment, as described in this report, can proceed without reference to any particular set of acceptance criteria since the object is to quantify the code accuracy. Acceptance criteria may then be used to determine whether further code improvements are necessary. A subsidiary goal of code assessment is to help develop a basis for arriving at mission-oriented acceptance criteria.

The issues that need to be addressed during code assessment are described in Section 2. The five sections that follow it contain the road map and methodology for accomplishing the stated goals. The basic approach is described in Section 3. The selection of the Assessment Matrix and the rationale for the selections made are described in Section 4. The Assessment Matrix covers various missions of thermal-hydraulic codes. Codes with only one particular mission (e.g., large-break LOCA) will only be subjected to the relevant parts of the overall Assessment Matrix. Hence, the Assessment Matrix contains tests from different test categories, selected from among the available sources to challenge the code's predictive capabilities and help quantify its performance. The manner through which the cases are sampled from the Assessment Matrix and the overall logistics of code assessment are presented in Section 5. Systems codes produce a very large quantity of results. Selection of the computed results for comparisons with test data and for quantification of code accuracy is described in Section 6. The meaning of code accuracy is dealt with in Section 7. Finally, the topic of code acceptance criteria is touched upon (rather gingerly) in Section 8.

The assessment process described and recommended in this report is aimed at best estimate thermo-hydraulic systems codes (or code versions) that have been completed and released to the public. The work is performed by personnel who were not involved in the code's development. For this reason, the process has been referred to as Independent Assessment. During Independent Assessment the code remains "frozen," i.e., unchanged from its released configuration. In contrast, the Developmental Assessment - not focused on in this report - is performed by code developers during the code development process to help them with the choice of models.

The USNRC Standard Problem exercise was instituted by the regulatory staff about five years ago to study performance of computer codes used by industry in the course of the plant licensing process. RES contractors engaged in the development of the best estimate systems codes also joined this exercise. Soon thereafter, the CSNI group of OECD/NEA sponsored a similar program known as the International Standard Problem Program (ISP). Both of these programs typically produced one or two standard problems per year. It was recognized that, at that pace, the breadth and depth of code assessment needed for quantifying the accuracy of the best estimate codes could not be achieved. While RES contractors continue participation in both the domestic and the international standard problems exercises, the bulk of the code assessment effort will be accomplished through the process described in this report. The Assessment Matrix here described also includes the cases considered in the standard problem exercises, as annotated in the Tables.

2. CODE ASSESSMENT ISSUES

The code performs a numerical integration of a set of conservation equations. These equations contain correlations that model pertinent physical processes.

The issues that need to be addressed during code assessment therefore involve:

a. The Conservation Equations: It needs to be established whether their number and form adequately account for the mass, momentum, and energy of all the fluids of interest.

b. Models and Correlations: The right-hand side of the thermal-hydraulic conservation equations contain all the important terms that model the exchange processes, as functions of the local or global flow and heat transfer regimes; this is illustrated in Figure 1. The most challenging part of code assessment is to determine whether the selected models are complete enough, whether they are sufficiently mechanistic to contain a scale-up potential and whether the empiricism built into the models has a wide enough data base that applicability to LWR conditions can be expected.

c. Numerical Analysis: One needs to evaluate the stability, convergence, truncation errors, and numerical diffusion of the numerical integration scheme.

During the course of code development, and as part of the so-called "developmental assessment" phase, the NRC contractors involved in code development are required to perform studies of the above aspects of the numerical analysis, including comparisons of code results against analytical solutions, wherever possible. The results of that effort are audited during the independent assessment phase that is performed by personnel who were not involved in code development.

Comparisons of code results against analytic solutions are of very limited value primarily because these are possible only for extremely restricted cases which generally do not challenge the treatment of nonlinearities. A recent international attempt to examine the numerical solution techniques through comparison of results from different codes (for a fairly complex benchmark problem void of test data) was not successful because the experts could not agree which calculation result ought to be used as a yardstick.

Clearly, more research needs to be done in this area of qualification of the numerical solution technique. For the time being, we rely on numerous comparisons of code results against test data, for both simple and complex situations. The drawback of this approach is that when it is established that code improvements are needed, the usual tendency is to start modifying the physical models, thereby absolving the numerics from any culpability.

d. User Convenience and Quality of Documentation: Code accuracy is a necessary but not sufficient attribute. The code must be designed in such a way that it could be used by engineers who were not involved in its development; the labor necessary for code input generation must be minimized; multiple input options of the kind that cause two people to get different results for the same case, can be avoided through adequate documentation. These issues need also to be addressed during code assessment. We find it helpful to ask, from time to time, engineers from different institutions to use the code in making comparisons against data for the same test, to see if different results are obtained and why.

The issues mentioned in (d) above will not be further discussed in this report.

3. COMMON DENOMINATOR

The heart of the code assessment process involves comparisons of code results with test data.

Most workers in the reactor safety field are familiar with the overall classification of the experiments that produce the test data base:

a. Tests conducted in LWR plants during their commissioning or start-up procedure, as well as actual transients or accidents. This source of data was neglected in the past, primarily due to preoccupation with the large-break LOCA analyses. The change of emphasis initiated during 1979 gives more attention to those plant transients and accident scenarios which have higher probabilities of occurrence. Assessment of computer codes designed to handle such cases cannot ignore the measurements obtained in LWR plants. While many of these measurements may be too coarse for quantification of code accuracy, they can certainly be utilized to check the validity of the computed trends as the boundary or the initial conditions are parametrically varied.

b. The Integral Systems Tests are designed to reproduce, as closely as possible, the overall reactor coolant system thermo-hydraulic behavior under conditions duplicating various postulated accidents and transients, including the effects of parametric variations. Test facilities that fall in this category exist in the United States and abroad, in different geometric scales.

c. The Separate Effects Tests are designed to produce much more detailed information on the behavior of the individual system components or parts of the overall system, subject to the imposed set of initial and boundary conditions which are parametrically varied to cover the ranges expected during postulated accidents and transients. These types of tests feature the widest range of geometric scales (a number of them in full-scale) and the widest range of parametric variations.

d. The Basic Tests, sometimes referred to as the "model development tests," are used to study thermo-hydraulic interactions on a very idealized and basic level, and to collect empirical information needed to define various constitutive relations.

Further information about the available test data base in the last three categories can be found in References (1) and (2).

4. SELECTION OF TEST DATA (Assessment Matrix)

The selected cases for code assessment will involve all of the major categories: Test data obtained from LWR plants, Integral Systems Tests, Separate Effects Tests, and Basic Tests. Different assessment strategies may emphasize different categories. The choices made for assessment of the advanced, best estimate codes and their rationale are described below.

4.1 LWR Plant Tests

A systematic search of test data obtained during the commissioning and start-up of LWR plants was recently initiated and, therefore, only few selections of such data has been made so far. It is clear that this source of data will be very useful for benchmarking of systems codes in their application to operational transients in specific LWR designs. Such information will also be valuable for checkout of computer input decks for the selected plants and for checkout of their control system models. Emphasis will be given to the representative LWR designs, to challenging test conditions, and to the tests featuring the best and the most extensive instrumentation.

4.2 Integral Systems Tests

The parameters that influenced our selection were as follows:

- Coverage of accident and transient scenarios relevant to various code missions.

- Facility design (PWR or BWR). Facilities featuring a nuclear core are given more weight.

- Facility scale - not only involving the volumetric scale but also the number of active loops, core length, and steam generator height. The larger the test scale the more emphasis in the selection.

- Quality, quantity, and diversity of measurements.

- "Virginity": Tests scheduled on a new facility or new and significantly different tests with an existing facility play a very prominent part in the selected test data base, offering the best opportunity to examine the predictive capability of the code.

The selected tests simulating the large and intermediate size cold leg breaks in PWR geometries are listed in Table I-1. The number of cases selected was influenced by the fact that all, except two, of the specified tests have already been calculated or are being calculated as part of the scheduled analytical support to experimental programs. Systems codes whose mission does not include Large-Break LOCA analysis will, of course, not be assessed against these tests. The columns labelled Type, ECC Injection Location, Initial Conditions, and Comments identify the breadth of coverage. For example, the comment "Simulation of L2-3" made in connection with the Semiscale/MOD-1 test S-06-6 indicates that the latter was chosen to test the effects of facility scale, other features being either the same or similar. The abbreviations are explained behind Table I-2. The crosses in both columns under the Type heading indicate that the test spanned the blowdown, refill, and the reflood stage of LOCA. The list of references appended to the Table indicates the source of detailed information on the related test. The test number designation is shown only for those tests that have already been performed or for the tests for which plans and schedules have been firmed up at the time of writing of this report.

The selected tests featuring Small Breaks and non-LOCA Transients in PWR geometries are listed in Tables I-2 and I-3 respectively.

Finally, Tables II-1 and II-2 show the selected Integral Systems Tests pertinent to BWR geometry. Tests featuring double-ended pump suction breaks are listed in Table II-1 while the Small-Break and Transient tests are listed in Table II-2.

Since many transients challenge the pressure relief valves - which may not be able to close - and some transients may involve ruptured steam generator tubes, the authors believe that the codes designed to address non-LOCA transients must also be able to address loss of coolant through small breaks. For this reason assessment of such codes should include all of the cases listed in Table I-2 and I-3 for PWR application and in Tables II-2 and II-3 for BWR application. Due to the Automatic Depressurization System (ADS) built into BWR plants, the codes designed to address small-break LOCAs (SBLOCA) in BWRs must also be able to handle most of the processes associated with the large-break LOCA (LBLOCA). This calls for consideration of all the cases listed in Tables II-1 and II-2 as part of their assessment.

4.3 Separate Effects Tests

The parameters influencing our selection of tests for code assessment included:

- Coverage of system components, except for those, such as a centrifugal pump, that the code describes purely empirically. Whether the adopted empiricism for such a component is adequate can be determined from the Integral Systems Test data.

- Design: The more faithful the geometric simulation and the greater the capability to reproduce the processes expected in the particular LWR component, the stronger the candidacy.

- Scale: The larger the scale the stronger the candidacy. Moreover, it is also important to have tests in different scales to assess the scale-up capability of the code.

- Quality, quantity, and diversity of measurements.

- Virginity.

- Potential for studying the code capability to serve as a scaling tool. This includes potential to study the validity of physical models in the code.

- Diversity of the initial and boundary conditions.

The core flow tests selected for code assessment are listed in Tables III-1 and III-2. Those featured in Table III-1 are limited to PWR and BWR reflood process. The selection involves electrically heated and nuclear rods, single bundle and multiple bundle tests, and a variety of pressure, flooding and spray rate, heat flux, and critical temperature conditions. In contrast, the tests listed in Table III-2 are not limited to the reflood process. They include blowdown, steady and transient flow, boiloff, and level swell tests. The BWR systems codes designed either for LBLOCA or SBLOCA missions should be assessed against data specified in Table III-1. Since some SBLOCA scenarios in PWRS may lead to core uncovery it is advisable to assess all PWR LOCA codes against data listed in Table III-1. Finally, all systems codes, regardless of their mission, should be assessed against data listed in Table III-2.

The ECC bypass tests with PWR downcomers of different scales are listed in Table III-3 Only the PWR systems codes designed for LBLOCA analysis ought to be assessed against these tests.

Tests featuring ECC flow within the BWR upper plenum and ECC penetration through the tie plate are listed in Table III-4. The cases listed in this table are pertinent to both the LBLOCA and the SBLOCA (BWR) codes.

The BWR jet pump behavior tests are listed in Table III-5. Tests featuring centrifugal pumps (PWR and BWR) were not considered because such pumps are described empirically, through homologous curves, rather than through a set of basic conservation equations. Table III-5 applies to all BWR codes.

Table III-6 lists tests for critical flow through pipe breaks. The large-break LOCA cases are represented by data sources featuring the largest test geometries. Those pertinent to small-break LOCA are also shown.

The selected tests for thermo-hydraulic processes within PWR steam generators are listed in Table III-7. Both U-Tube and Once-Through (B&W) steam generators are represented. The two-phase flow conditions within the primary side were available only in the UTSG tests. Table III-7 applies to all PWR codes, regardless of their mission.

Finally, the PWR pressurizer tests are shown in Table III-8, applicable to all PWR codes.

4.4 Basic Tests (Applicable to all codes, regardless of mission)

A very large data base can be found in the technical literature.

Reduction of the number of candidate cases came from the decision to exclude tests that were utilized for the development of correlations and other empiricism utilized by the code.

Tests utilizing fluids other than air, water, and steam were eliminated because they would require development of specific equations of state and because the empiricism built into the code involving the fluids of interest in reactor safety, may not be valid for other fluids (Freon, etc.).

The emphasis was on tests, in the same facility, where gradual introduction of complexities could help in assessment of "physics" of fluid flow and heat transfer built into the code. Tests featuring simple geometries and two-phase fluid transients are also helpful for assessment of the numerical solution technique.

Tests sponsored by the NRC were given priority because of greater familiarity with the tests.

A summary of the selected Basic Tests is shown in Table IV to illustrate the data sources, fluids employed in the tests, dimensionality, type (steady or transient flow), and the processes that were studied.

More detailed information concerning the particular tests selected from each of the test facilities, the reasons for their selection, description of test parameters, and the desired code output are given in Appendix C, sections (1) through (13).

4.5 Assessment Matrix Flexibility

The current overall code Assessment Matrix involves some 124 Integral Systems Tests, 95 Separate Effects Tests, and 63 Basic Tests, for a total of 282 cases covering all code missions to be considered over the next 4-5 years. New surprises found in the course of research may lead to new tests that need to be performed. Hence, a degree of flexibility is always required in code assessment, including the awareness of the available resources (funds, manpower, computer access), and the agency needs.

The Assessment Matrix presented here is a result of considerable pruning enticed by the fiscal rather than technical considerations. The authors believe that further reductions may prohibit quantification of code accuracy.

5. LOGISTICS OF CODE ASSESSMENT

5.1 Factors that can Influence the Logistics of Code Assessment

5.1.1 Test Schedules

Integral Systems Tests as well as the large-scale Separate Effects Tests in the area of LWR safety require lengthy periods of planning, design, construction, and execution. They require expenditures of considerable resources. Hence, they are embarked upon only if the expected information is important. Data from these tests ought to be utilized in code assessment as long as codes are being used to help determine whether the plants are safe and to establish the margin of safety.

Several of such important tests are scheduled over the next 4 to 5 years. Nevertheless, a carefully planned code assessment process is capable of providing interim information regarding the accuracy with which the code is likely to calculate LWR response. Reliability of this interim information improves as larger fractions of the relevant parts of the overall Assessment Matrix are considered in the intervening years.

Tests scheduled to be performed through 1985 provide the source of the "virgin" cases that many regard as essential in studying predictive capabilities of computer codes.

5.1.2 Sampling Sequence and Rate

Some assessment strategies may choose sequential completion of the selected cases, starting with the Basic Tests category and leaving the Integral Systems Tests for the last task.

This strategy will appear most attractive to those who appreciate a bottom-up approach. It should be recalled, however, that before the code is ready for an independent assessment, the code developers had to resort to "sampling" from all test categories in order to ensure that the code will be able to at least address the relevant LOCAs and transients with a reasonable chance of success. An important disadvantage of this strategy is the fact that the sponsor would have no intermediate information about the code capability to handle its missions, until the whole code assessment process is completed. The regulatory agencies and institutions involved in the design of the Integral Systems Test facilities - who need best estimate systems codes to help them in the design and conduct of tests and interpretation of test results - could hardly afford this assessment strategy.

In the parallel sampling approach, one strategy may require accelerated sampling from the most important test categories (LWR Plant Tests and Integral Systems Tests) to facilitate gathering of information concerning the code capability and accuracy at yearly intervals. Conversely, a strategy could be devised that assigns the largest fraction of sampling from the Integral Systems Tests category to the final year of code assessment. The primary goal of the latter strategy could be accomplished only if test data for the cases to be sampled in the last year are also made inaccessible to code developers. However, locking-up of test data, especially from the important test facilities, appears to be an impractical task. Therefore, a less extreme strategy would be desirable that allows for both the intermediate information on code accuracy and for the opportunity to apply the code to "virgin" tests.

5.1.3 <u>Releases of Improved Code Versions</u>

As codes are being subjected to Independent Assessment, areas of needed improvements are being identified. Such improvements are incorporated into the new code versions. Several such versions are expected prior to completion of the overall Assessment Matrix. It does not appear feasible to repeat, with every new code version, all of the test cases previously considered. One possibility is to select, for repetition, one test case from each integral test facility <u>and</u> those test cases that identified particular weaknesses in the last code version. However, if important coding errors are discovered during assessment of the code, corrections must be made right away and communicated to all code users as well as to the National Energy Software Center. If such corrections are made in the middle of the assessment process, it may be necessary to repeat a few selected cases.

5.1.4 <u>Code User Effects on Code Results</u>

In the past, the code user had to make his own selections from among numerous "user options," without a proper guidance provided in code documentation. There were many instances in which knowledgeable and experienced code users obtained quite different results on the same problem, using the same code. One of the important tasks in the development of the advanced systems codes was the removal of user options or, at the minimum, specification of a careful set of instructions in the users manual, recommending particular options for the expected applications.

During the assessment of TRAC-P1A it became clear that this goal was not yet reached although this code contained only a small fraction of user options available in the previous codes.

Since the important systems codes are not meant to be used by only one person, or even by only one organization, it becomes important to establish and quantify the impact of code users on the computed key results. This can be accomplished by asking several specialists (from different organizations) to predict the same test case using the same code.

5.1.5 Code Tuning

Recalculations of the same test case (where test data are already available), with adjustments in input parameters, user options, nodalization, or even model changes - until the best possible agreement is obtained, is called code "tuning." On the surface, this process may be useful in determining what the code may be capable of calculating, after optimizing all input data. However, it was observed that those same "optimum" parameters are often not utilized in calculating the next experiment, even in the same test facility. A new set of "optimized" input coefficients is then defined, without technical justification. Such a process must be avoided during Independent Assessment of a released code since it will prevent gathering of objective information on code accuracy.

A code that was subjected to such a tuning process in a succession of test cases featuring gradual increases in complexity, may show excellent agreement in many cases. However, that "success" does not guarantee predictive capability, nor does it shed any light on the accuracy with which LWR response can be calculated.

5.2 The Recommended Approach

a. A portion of the overall Assessment Matrix is defined for the assessment process during the fiscal year. It contains the relevant test cases from each of the major test categories depending on the code mission. If a newly released code version is to be assessed, then the selected segment of test cases must also include those already considered cases that the previous code version predicted poorly; and one previously considered integral test case, from each important (integral) test facility. The total number of test cases selected from this segment depends on the magnitude of code assessment funding approved for that fiscal year.

b. The selected segment of the Assessment Matrix is apportioned among all the contractors participating in the assessment process. Cost of computation (dollars per hour of CPU - CDC 7600 equivalent) varies between contractors. During assessment of TRAC-P1A, the long running cases were assigned to contractors where the computation costs were the lowest. With improvements made to the code running time, the computing cost has become overshadowed by manpower expenditures - which are considerable. Cognizance of the contractor's strength in the area of thermo-hydraulics of two-phase flow or his intimate knowledge of particular test facilities plays a role in determining the fraction of the assigned test category. Groups of test cases featuring gradual increase in complexity, unless they fall in the Basic Tests category, are not assigned to one contractor; they are usually split and the individual cases assigned to different contractors.

c. At least one important test case is assigned to more than one contractor to determine the user effect on code results.

d. The contractors are asked to perform a best estimate calculation, involving their best estimate (or guess) as to the geometric nodalization and input options. Results of that calculation must be submitted. Should the results be particularly poor - and if the resources allow - the contractor is asked to perform diagnostic recalculations to ascertain the impact of different input options or nodalizations, or (in the case of the Basic Tests) even of modeling changes. Results of such diagnostic studies are immediately communicated to code developers.

Listing and specification of the desired computed results, especially for all Integral Systems Tests, are communicated to the contractors. The contractors issue detailed reports on the cases considered during the fiscal year.

e. A unified report, integrating the information from all contractors, is prepared at RES headquarters. During that task the key results from all integral systems tests are condensed into single plots - as explained in Section 6 - to examine effects of the geometric scale and to project the code accuracy (probability distribution function) to LWR. A Research Information Letter (RIL) is issued reporting results of independent assessment of each publicly released code version.

6. SELECTION OF COMPUTED RESULTS

6.1 Results for Code Prediction/Test Data Comparisons

The choices of results selected for code assessment could involve:

- Global and local, single-valued results,
- Time histories of results, and
- Statistical measure of fit of time histories.

It is certainly necessary to select those key results that reflect the basic mission of the code and for which information on code accuracy needs to be obtained and compared against code acceptance criteria. In addition, those results must be identified that provide information concerning the code ability to model the relevant physical processes.

a. Single-Valued Global and Local Results:

For code comparisons against test data from the Basic and the Separate Effects Tests, local, single-valued results need to be determined on a case-by-case basis, depending on the individual processes or system components.

The following global and local, single-valued results pertain to comparisons with Integral Systems Test data:

- Results defining the reactor core clad temperature "signatures." These are described in detail in Appendix A, and are applicable to all types of accidents and transients. Incidentally, they are also applicable to Separate Effects Tests featuring single fuel bundles or bundle arrays.

Other results, particularly applicable to small-break LOCAs, may involve:

- The minimum liquid or froth level (whichever measured) reached in the reactor vessel.

- The amount of heat removed by each steam generator during a specified length of time, t*. The latter may be the final core quench time or the time at which some operator action is initiated, etc.

- Amount of coolant mass lost through the break during time, t*.

- Amount of coolant energy released through the break during time, t*.

- Times when coolant pressure in the upper plenum reaches 10 and 5 MPa, respectively. These quantities provide the pressure "signature" for certain types of small-break LOCA. For types featuring very small break sizes, the time and magnitude of the minimum pressure may be more relevant.

Other global results, pertaining to large-break LOCA and to non-LOCA transients may involve:

- times to start and end the discharge of ECC accumulators into the intact loop(s)

- time to start LPIS

- time of the minimum coolant inventory within the lower plenum

- time when the lower plenum liquid inventory first exceeds 90% of maximum, during the refill stage
- time when the minimum liquid inventory is reached on the secondary side of a specified steam generator
- time of the first activation of the steam generator relief valve
- time of the first activation of the pressurizer safety valve
- the minimum coolant pressure reached, etc.

Examples of various plotting formats for the quantity Ø representing any of the above global results are illustrated in Figure 2.

b. Time Histories of Results

Plotted overlays of time histories of the predicted and the measured results provide the most useful information regarding the code capability and, in particular, regarding the consistency of the calculated trends.

In the case of the Basic and the Separate Effects Tests, it is important to plot all results for which measurements are made, as well as other results that shed light on the consistency of trends.

For comparisons against data from Integral Systems Tests, examples include the time histories of clad temperature, the mass of liquid within the lower and the upper plenum, upper plenum pressure, local void fractions within regions of the reactor vessel (where measured), the froth or liquid level positions within the vessel for small-break cases, and results for all the important measurements recorded in the loop spool pieces (local void fraction, coolant temperature, fluid velocity, pressure differentials, metal temperature) and within other system components (steam generator, pressurizer, etc.).

c. Statistical Measure of Fit of Time Histories

The overlays of time histories are not amenable to condensation of results and to application of acceptance criteria. Some researchers have therefore proposed using statistical means of quantifying the discrepancies between the calculated and the measured results. For example, the shaded areas in Figure 3, indicating the amount of discrepancy, could be weighted differently for different time segments of the transient (e.g., blowdown, refill, reflood), for different results (e.g., flows, pressures, temperatures), and even for different regions of the system. The idea is to produce "statistics" of the code accuracy expressible by a figure of merit related to the sum of all the weighted areas of discrepancies, perhaps normalized by the number of terms in the sum. Other "statistical" approaches can be concocted, with an endless variety of weighting factors and figures of merit.

The final aim would be to compare the figure of merit, for each test case, against some acceptable bound and counting the percentage that remained within. The main disadvantage of this approach is that it obscures the information regarding the validity of physical models and the computed trends. In addition, it may be extremely difficult to specify various weighting factors and other assumptions that would be widely acceptable.

6.2 Recommended Approach

The current approach relies heavily on numerous overlays of time histories and subjective judgment of their validity, for all test categories. In addition, it should be attempted to utilize the global and local key results enumerated above for the case of Integral Systems Tests.

7. CHARACTERIZATION OF CODE ACCURACY

The measured magnitude of some physical property, Ø, is reported in terms of its best estimate (or mean, or nominal) value and its uncertainty band, supplemented (in some cases) by the information on the confidence level. Measurement uncertainty is caused by imperfections in the measuring instrument, in signal processing, and in the models through which certain indirect measurements are combined to define the physical property Ø. The narrower the uncertainty band the more accurate is the measurement.

The best estimate code predictions also contain uncertainties. In addition, the nominal or the best estimate value of the code prediction may differ from the nominal or best estimate value of the measurement. As pointed out in Reference 3, the smaller that difference (or the offset) and the narrower the uncertainty "band" of the code prediction, the more accurate is the code.

In what follows, the causes of the prediction uncertainty will be described, together with two methods for its quantification. The preferred method will be indicated.

7.1 Sources of Code Prediction Uncertainty

The uncertainty in the prediction of key single-valued results for LWR can be viewed as being the result of:

a. Uncertainty in the plant condition at the onset of any given accident scenario. The plant condition may include fuel burnup, peaking factors, core power, water levels in ECC accumulators and in the steam generator secondary side, etc. These uncertainties are not considered in the course of code assessment, because they are not due to imperfections in the code.

b. Uncertainties in modeling of the reactor fuel rods' thermal and mechanical properties, such as UO_2 thermal conductivity, gap conductance as affected by the gap size, gap gas composition and pressure, pellet deformation, clad deformation, etc. Information concerning the nuclear fuel rod modeling uncertainty is obtained from a separate assessment program involving fuel behavior codes. That information is pertinent to systems codes since, eventually, the latter will include all models and correlations that were found to be important.

   Even though the majority of the test data used for assessment of systems codes feature electric heaters for simulation of nuclear fuel rods, uncertainties related to their modeling are still present. For example, electric heaters may contain nonuniformities in properties of their materials, nonuniformities in centering of the heater coils or tubes, plus uncertainties in heater coil spacing and installation of clad thermocouples. Effects of these nonuniformities, the shadow and/or fin cooling effects of clad thermocouples are not accounted for in mathematical/physical models of fuel simulators. Their effects should, however, not be ignored in forming conclusions about the code accuracy.

c. Uncertainties in modeling of the reactor primary and secondary coolant systems thermal hydraulics. Their causes are listed below:

   (1) Code input uncertainties related to physical properties or to those coefficients whose specification is left to code user's discretion. Current trend in design of advanced codes is to eliminate, as much as possible, input choices left to user discretion.

   (2) Coefficients embedded in the code that are related to physical models and correlations.

   (3) Degree of system geometry discretization used for numerical solution.

   (4) Upper limits on time steps and on convergence criteria.

   (5) Adequacy of the set of conservation (field) equations solved in the code.

   (6) Adequacy of the thermo-hydraulic models for interphasic and fluid/wall interactions, and for the flow and heat transfer regime recognition criteria.

   (7) Truncation and numerical diffusion errors inherent in the numerical solution strategy.

   (8) Inability to address phenomena of stochastic nature.

   (9) Coding (programming) errors.

7.2 Quantification of Code Uncertainty Through Statistical Code Uncertainty Study

A summary of this approach is given in Appendix B. This method is applicable only to quantification of code results uncertainties caused by the uncertainties in items listed in (a), (b), and (c) parts (1) through (4) above. This is a very important limitation of the method. The second limitation is that the information obtained on code uncertainty is tied to the particular accident scenario used in the study. The third limitation is that the method requires knowledge of the uncertainty range and the probability distribution function for each of the "input" parameters. Only a limited number of such parameters can be considered since very significant expenditures in computing resources are involved. Consequently, prior knowledge of the importance of each parameter is needed to select only those that are judged to significantly affect the uncertainty of the final result. Finally, the sampling strategy may be affected by the choice of the final result for which the code uncertainty is sought.

One of the important key results is the global peak clad temperature, GPCT. Various code uncertainty studies, utilizing the techniques summarized in Appendix B, have shown that, for the case of the so-called Design Basis LOCA, the computed peak clad temperature is normally distributed about its mean or the "best estimate" value.

Information at hand indicates that for the case of the design basis LOCA, prediction uncertainty concerning the logarithm of the core-wide amount of clad oxidation and, by inference, of the local amount of clad oxidation (logarithm), is also normally distributed. Therefore, the use of the standard deviation is applicable for description of code accuracy for, at least, the peak clad temperature and the logarithm of clad oxidation.

7.3 Information on Code Uncertainty From Scatter Plots

In the statistical code uncertainty study, input parameters are varied around their best estimate or nominal values. If, on the other hand, code predictions are made of many test situations, using only the nominal or best estimate input values, plots of the predicted minus the measured (nominal or best estimate) values of key, single-valued results will exhibit scatter, as illustrated in Figure 2. That scatter will not only reflect the uncertainties associated with the nominal (BE) values of the code input and the embedded coefficients but, in fact, it will account for all of the effects listed under (b) and (c) in Section 7.1.

Through proper normalization of the ordinate in the scatter plot the abscissa can account for a large variety of test conditions, in different geometric scales.

Scatter plots are amenable to quantification of the code uncertainty probability distribution function and of the offset, providing sufficient number of entries are present to provide for a statistically meaningful count.

This approach also influences the selection strategy for the number and type of cases to be considered.

7.4 Summary

Any best estimate type analysis is assocated with an "uncertainty band" reflecting the code accuracy. The narrower the band (or scatter) the more accurate is the code. The prediction accuracy must be tested for a variety of key results that characterize the important thermo-hydraulic processes, in different geometric scales and with different boundary conditions. Overlay plots of a variety of calculated time histories (measured and predicted, or just predicted if the measurements are not available) must be obtained to give indications of the computed trends and consistency. Plots of the spatial distribution of the local results (predicted minus measured) are also used to infer whether the physical models are adequate.

8. CODE ACCEPTANCE CRITERIA

Code acceptance criteria are aimed at providing a yardstick for judging whether the code is accurate enough to fulfill its intended mission. If the criteria are met, further efforts in code development would be either terminated or greatly diminished. There is no universal agreement about the need for code acceptance criteria. Some people feel that a subjective judgment, based on knowledge of the code uncertainty and mission, could serve equally well.

It has been suggested that predicted results that lie between the measurement uncertainty bounds are automatically acceptable. However, some measurements are poor enough to provide little challenge even for simple codes. In other instances, the computational mesh is made coarse to such a degree that many measurements are taken within a computational cell. Their combined scatter is then used to define such a wide measurement "uncertainty band" that most codes would pass the accuracy test. A typical example may be a very coarse nodalization of the reactor core and comparison of clad temperature signatures. This approach serves no useful purpose in code assessment, unless one is trying to prove that, due to special conservatisms used in the code, the computed results upper-bound the measured temperatures. On the other hand, some of the measurements (temperatures, pressures, pressure differentials) are so accurate that the code acceptance criteria based on their measurement uncertainty bands are unnecessarily stringent.

Realizing, therefore, that acceptance criteria based on measurement uncertainties are not going to be helpful, let us examine whether acceptance criteria may be connected to some regulatory requirements or may originate from an accuracy goal that is thought to be achievable.

The current regulatory requirements for conservative analyses of the design basis LOCA prohibit the computed peak clad temperature from exceeding

2200°F. The best estimate analyses of the design basis LOCA yield much lower peak clad temperatures. How accurate should be such best estimate analyses? Having concluded in the preceding section, that the best estimate analysis is associated with a probability distribution - which appears to be "normal" in the case of peak clad temperature - it may be possible to define an acceptable standard deviation as function of the regulatory limit. For example, it could be required that the standard deviation, σ_{PCT}, be of such magnitude that the probability of the peak clad temperature exceeding the regulatory limit (2200°F) be equal to or less than, say 5% or less. As illustrated in Figure 4, such a criterion would tolerate fairly large uncertainty (for the global peak clad temperature, GPCT) if the best estimate (or the mean) value were much lower than 2200°F. Conversely, if the best estimate value of GPCT happened to be much closer to the regulatory limit, the required standard deviation could be so small as to be unattainable.

It is very unlikely, however, that best estimate predictions of the key results that directly affect reactor safety, are going to be very much smaller than the regulatory limit, <u>for every conceivable</u> type of the accident or transient. Considerations of multiple failures and operator actions may, in some cases, lead to cases in which the regulatory limit is not only reached but even exceeded. It appears, therefore, that the above described prescription for code acceptance may not be very useful.

This leads us to acceptance criteria that are based on an accuracy goal which is thought to be achievable. The proof of the code accuracy must come from an in-depth assessment of the code involving many comparisons with test data. The available, or the achievable, test data base may, therefore, in itself impose a limitation on the code accuracy requirement that could be substantiated as pertinent to LWRs.

These thoughts lead us to believe that a reasonable accuracy goal, reflecting the current state-of-the-art in code development, could only be posed after a good deal of experience has been gained in the assessment of the current generation of codes. From what is known today and based on the experience gained thus far in the assessment of an advanced systems code (such as TRAC), a reasonable accuracy goal for the peak clad temperature may amount to about 250-320°F for calculation of accidents and transients in LWRs that do not involve any significant core damage. The major contribution to this uncertainty comes from modeling of nuclear fuel.

Twenty percent accuracy on times t_{PCT}, and t_{LQ} may be achievable. No experience exists so far to forecast the achievable accuracy on $I_{\Delta T}$ or on ΔR_{OX} (defined in Appendix A).

One may be tempted to invoke the code sensitivity studies for prioritization of various systems effects, thus of other key results, on the clad temperature signatures; more stringent accuracy would be required for those systems effects that affect more strongly the core thermo-hydraulics. If, however, the same code is to be used to analyze different accidents and transients, it appears that such prioritization efforts could lead to conflicting requirements. For example, good description of the steam generator thermal-hydraulics plays a minor role for large-break LOCAs. Yet, a very good description of steam generator behavior is extremely important for certain small-break LOCAs. Similarly, a very good description of the pressure-time history is not essential for the large-break LOCA, yet it is very important for small-break LOCAs and for certain non-LOCA transients.

The authors believe that towards the end of 1981 enough code assessment experience will have been gained to specify the achievable goals for the key results defined in this report as functions of code mission. Acceptance criteria could then be related to the achievable goals.

It would take much longer to specify acceptance criteria for the derived results based on statistical manipulation of the predicted vs measured time histories. This approach is not being recommended.

In the meantime, emphasis should be placed on the displays (overlays) of the measured and predicted time histories of all results listed in Section 6, to ascertain whether correct trends are predicted and to make subjective judgments about the code adequacy.

9. SUMMARY AND CONCLUSIONS

The conclusion was reached that the most important aspect of code assessment involves numerous comparisons of code prediction results, with measurements obtained in full-scale LWR plants and in various domestic and foreign test facilities.

It was shown that planning of the code assessment process involves making many choices from among alternative approaches. The choices are not all black and white; many are subject to valid criticism and some are controversial.

Five parameters were identified as influencing the code assessment strategy.

The first parameter deals with the type of data selected from among the available test data base. The rationale employed for this selection has been described.

The second parameter has to do with the rate of sampling from among different categories of the selected test data base. Preference was indicated for the parallel rather than sequential sampling and without accelerated sampling from among one particular test data category.

The third parameter involves the choice of the computed results for which the code accuracy is to be determined. It was shown that overlays of time histories of the predicted and the measured results are most informative concerning the code validity in general and validity of the physical model in particular. These overlays form the backbone of the code assessment. Means were described for condensing the information so that focussed, quantitative assessment can be made.

The fourth parameter concerns the adopted method for quantifying the code accuracy. It is shown that, for a variety of reasons, results of code predictions involve an uncertainty band and may be an offset between the measured mean and the predicted nominal (best estimate) value of each key result. Hence, the adopted measure of the code accuracy is expressed by the width of the uncertainty band and by the amount of offset. For reasons discussed in the paper, preference was indicated for finding this information by means of scatter plots rather than through statistical studies of code uncertainty.

The fifth parameter involves the code acceptance criteria. It was shown that an approach in which the acceptance criteria for a few of the key results are tied to the current regulatory limits may not be generally applicable. The other approach discussed, and recommended in this report, identifies the accuracy goal for each key result, consistent with what is judged to be achievable with the current state-of-the-art. It is projected that, by the end of 1981, sufficient experience in code assessment could be gained to allow these goals, hence code acceptance criteria, to be stated

Every currently adopted strategy will find its supporters and opponents. It is anticipated, however, that as experience is gained in this field the advantages and disadvantages of various approaches will become more discernable, hopefully leading towards a strategy acceptable to most of the technical community.

10. REFERENCES

1. Tong, L. S., "USNRC Research Program," Paper No. IAEA-CN-39/99 presented at the International Conference on Current Nuclear Power Safety Issues, October 20-24, 1980, Stockholm, Sweden.

2. Fabic, S., "Data Sources for LOCA Code Verification," J. Nuclear Safety, Vol. 17, No. 6, 1976

3. Murley, T. E., "Verification of Reactor Safety Codes," Trans. ENC '79 Conf. European Nuclear Society, Volume 31 TRANSO 31 1-666 (1979) ISSN:0003-018X, ENS/ANS.

Figure 1

Relationship Between System Region Geometry, Flow Regime, and Physical Models in Field Equations

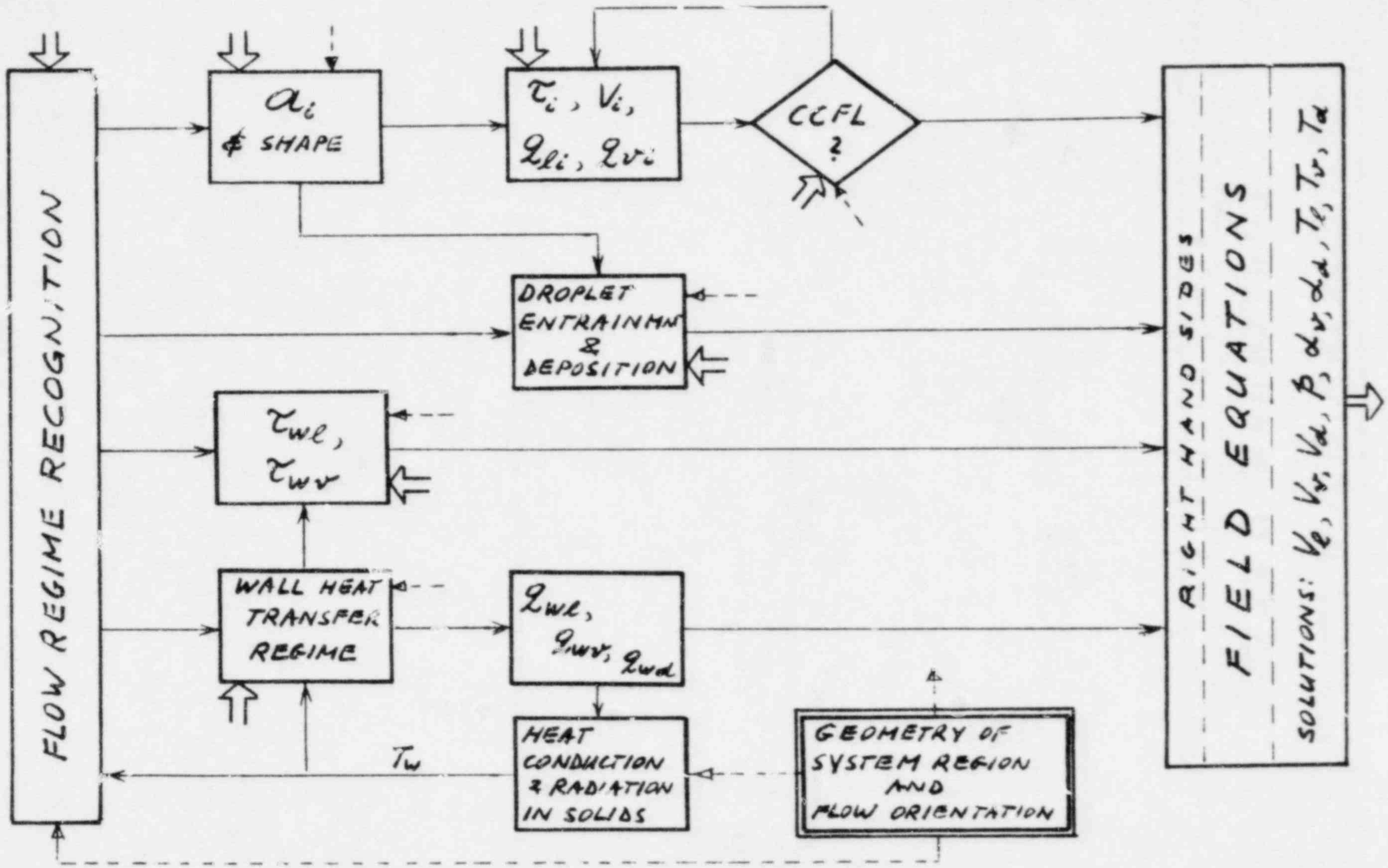


Figure 1 - Continued

Legend:

a_i = interfacial surface area

τ_i, V_i, q_{li}, q_{vi} = interfacial shear, velocity, and heat fluxes, from the liquid and the vapor sides, respectively.

CCFL = empirical correlation for counter-current flow limitation

τ_{wl}, τ_{wv} = shear between wall and liquid or vapor, respectively.

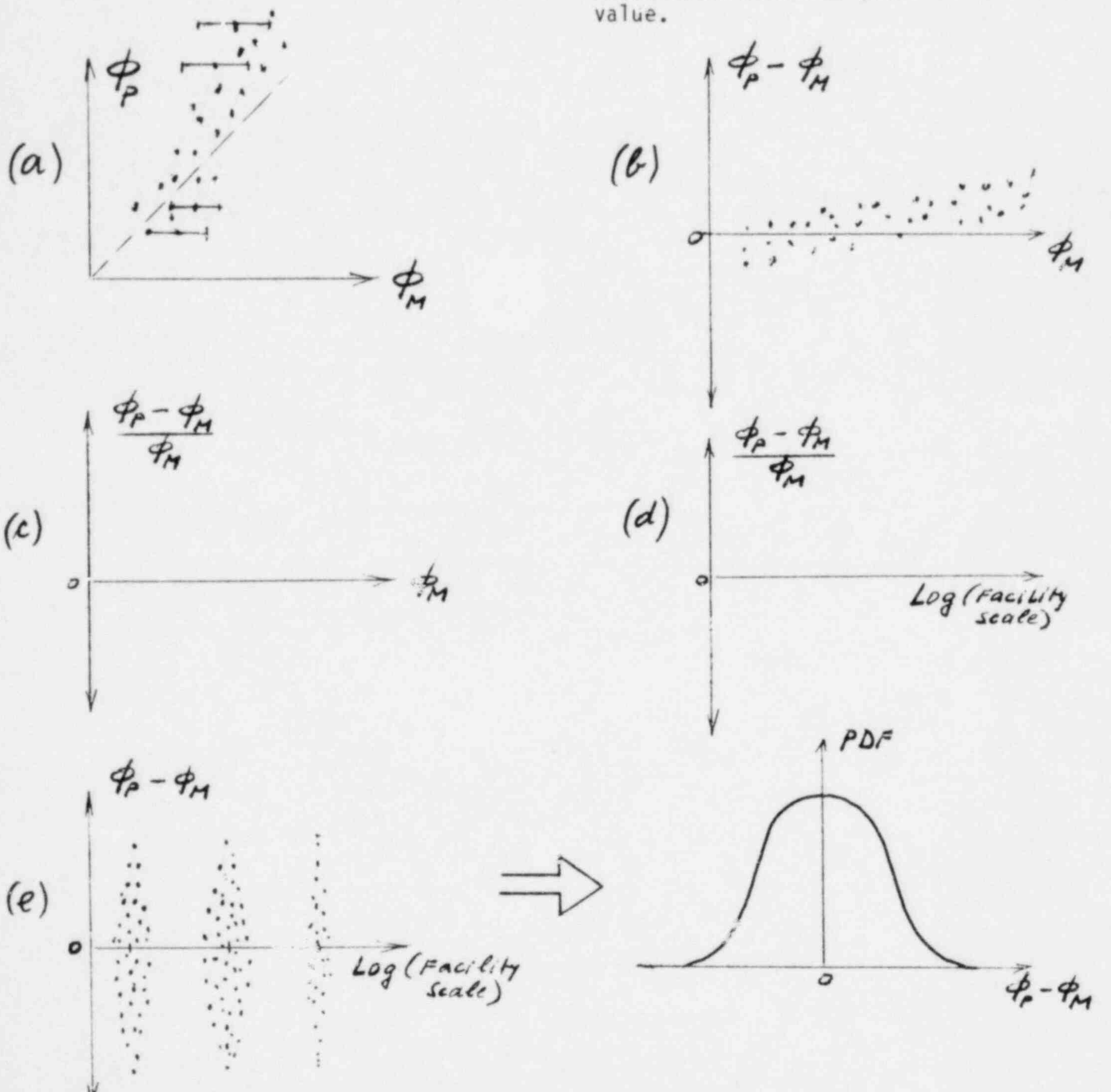
q_{wl}, q_{wv} = heat flux from the wall to liquid and vapor, respectively.

The main point is to illustrate that each of the above models is affected by the flow and heat transfer regimes which, in turn, can be strongly dependent on the geometry of the region of the system being analyzed. The short arrows indicate the dependence on the results of the field equations solutions and on the local geometry. The latter may not be resolvable by the adopted spatial discretization.

Figure 2

Alternative Plots of Global Results

ϕ =single-valued key result
Subscript P indicates predicted value
M indicates measured (best estimate) value.

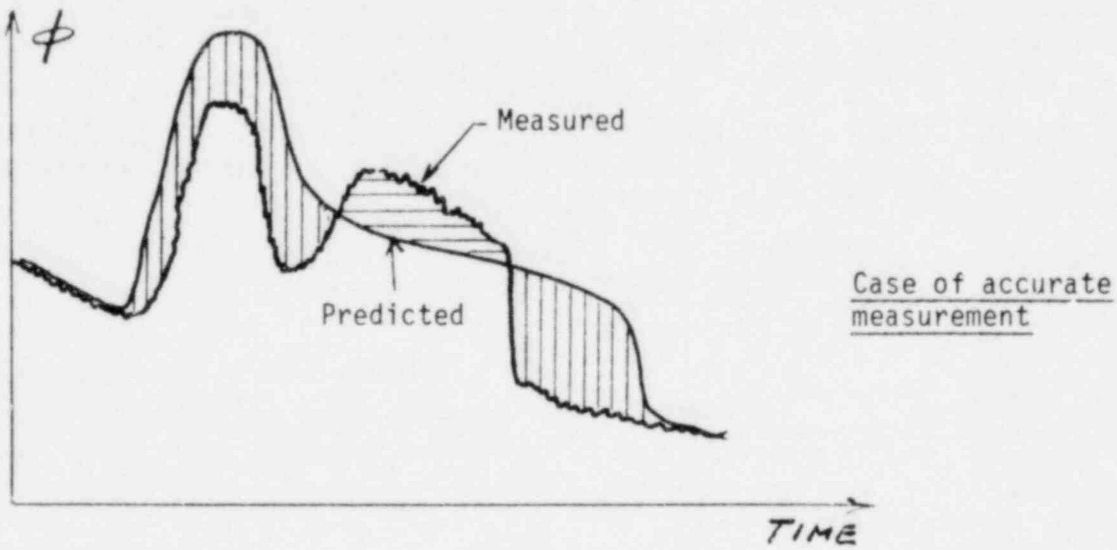


Cases (c) and (d) are suitable for ϕ=time of...
Case (e) illustrates prediction uncertainty which is scale-independent.
If scale effects exhibit clear trends it may be possible to extrapolate to LWR.

Figure 3

In:formation for statistical manipulation of overlay plots of the predicted and measured time histories.

Shaded areas indicate zones of disagreement.



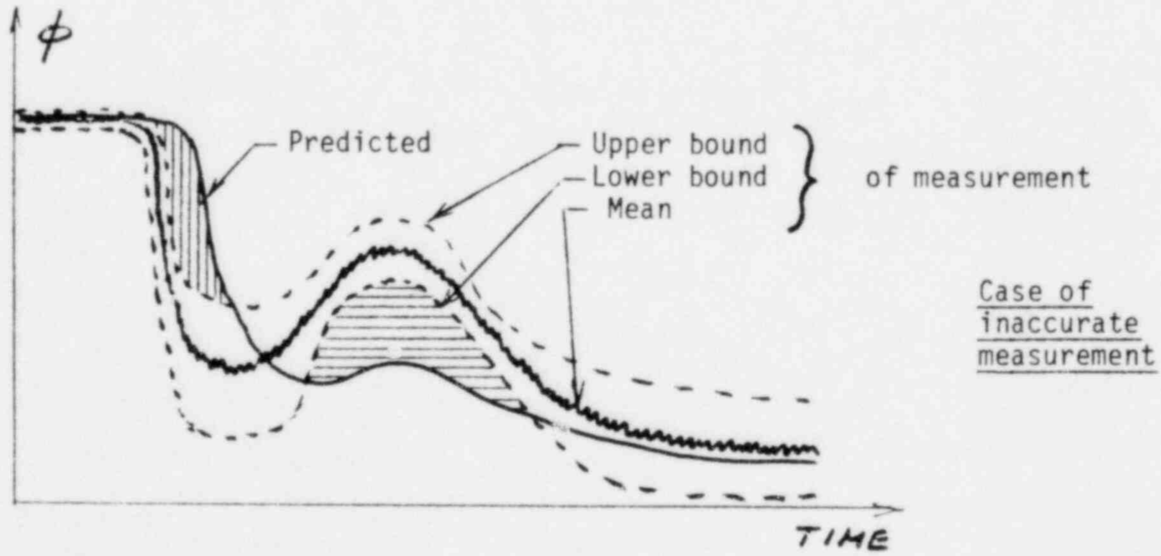


Figure 4

Illustration of two probability distribution functions for peak clad temperature, both obeying the limitation on the probability of PCT exceeding some regulatory limit (EM).

The case featuring the Best Estimate (BE) value of PCT which is closer to the regulatory limit demands a more accurate code, i.e. smaller standard deviation.

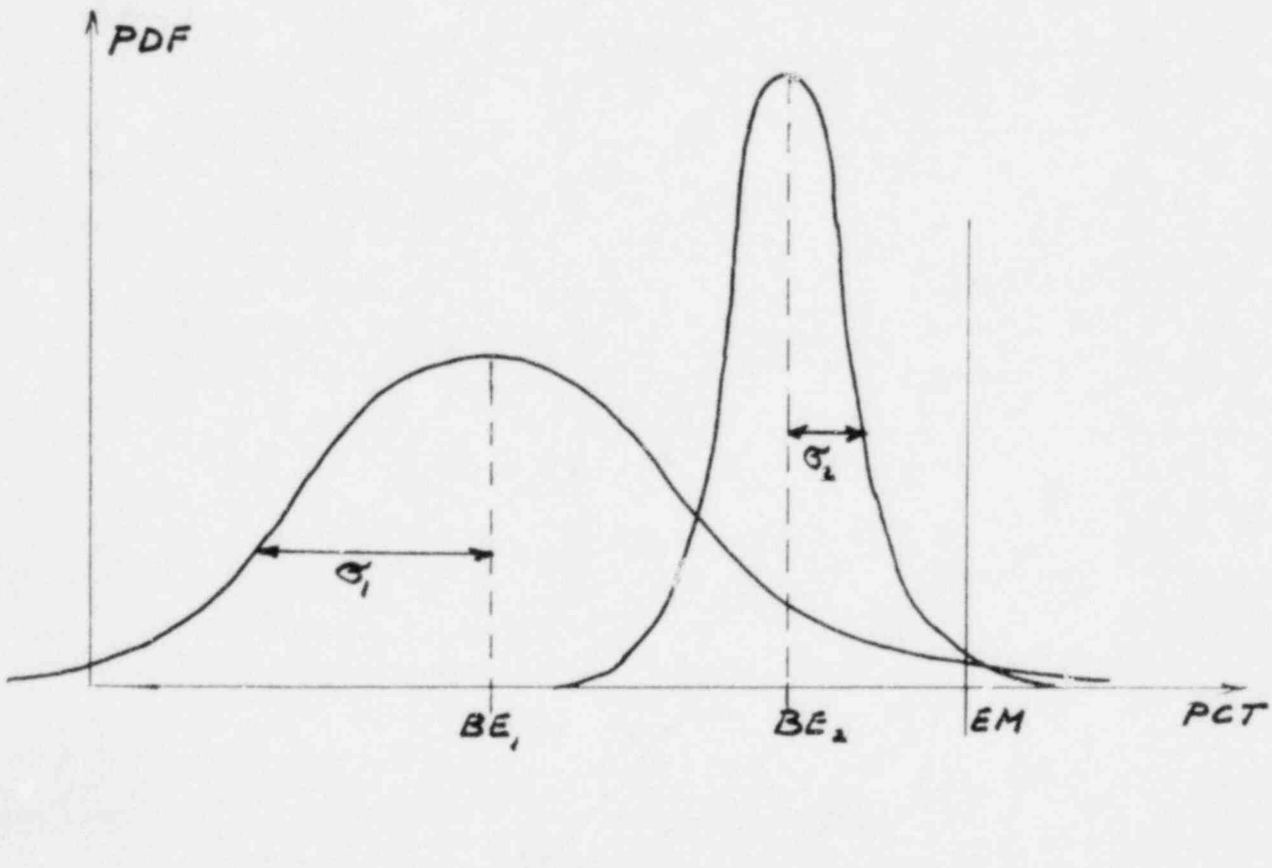


Table I-1 PWR-DESIGN INTEGRAL SYSTEMS-LARGE AND INTERMEDIATE SIZE BREAKS

Facility	Test #	Type: Blowdown	Type: Reflood	ECC Injection Location	Initial Conditions: p (MPa)	Initial Conditions: q'_{rod} (kW/m)	Comments	Reference
Semiscale/ MOD-1	S-04-4	x	x	ILCL, ILHL, LP	15.6	23.1	Reduced volume of lower plenum	2
	S-04-5	x	x	ILCL, ILHL	15.6	23.9	All rods powered. Compares to S-04-6	3
	S-04-6	x	x	ILCL, ILHL	15.6	21.5	Four unpowered rods. See S-04-5	4
	S-28-1	x	x	ILCL	15.6	20.7	Simulates 60 SG tube ruptures	6
	S-28-3	x	x	ILCL	15.6	21.5	Simulates 12 SG tube ruptures	7
Semiscale/ MOD-3	S-07-6	x	x	ILCL	15.6	21.5	Simulates S-04-6. Test influence of new geometry	8
Semiscale/ MOD-2A	1 test						It is likely that at least one test in this category will be run before 1983	
Semiscale/ MOD-5	1 test						At least one test in this category may be run in 1983 or 1984	
LOFT	L1-4	x		ILCL	15.5	0	Tests scale effects by comparing to S-01-4A. USNRC Std. Prob. No. 7	10
	L1-5	x		ILCL	15.5	0	First test with nuclear core installed.	11
	L2-2	x	x	ILCL	15.6	13.3	First test with nuclear heating	12
	L2-3	x	x	ILCL	15.3	16.8	As L2-2 except higher power. USNRC Std. Prob. No. 10	13
	L5-1/ L8-2	x	x	ILCL	15.5	22.9	Intermediate size break line Test Scheduled for Sept. 1981.	14
	L2-5	x	x	ILCL	15.5	16.0	"Beginning of Life" fuel pressure. Test scheduled for January 1982	14
	L2-6	x	x	ILCL	15.5	16.0	"End of Life" fuel pressure. Test scheduled for March 1983	14
LOBI	A1-04	x		No ECC	15.5	21.6	1.8 full power seconds after break. "Virgin" facility. Std. Prob. "PREX"	9
	A1-03	x	x	ILCL, ILHL	15.5	21.6	10.2 full power seconds after break	15
	A1-04R	x	x	ILCL	15.5	21.6	9.0 full power seconds after break	16
	A1-01	x	x	ILCL, ILHL	15.5	21.6	3.9 full power seconds after break	

Table I-1 (Continued)

Facility	Test #	Type: Blowdown	Type: Reflood	ECC Injection Location	Initial Conditions: p (MPa)	Initial Conditions: q'_{rod} (kW/m)	Comments	Reference
CCTF-I	010		x	ILCL,LP	0.20		High pump resistance	
	020		x	ILCL,LP	0.20		Base case	
	025		x	LP	0.41		Simulates FLECHT 3105B	
CCTF-II	5 tests						Testing scheduled to start in October 1981	
PKL-I	K1.3		x	ILCL,ILHL	0.4	1.1	Base case for combined injection	
	K5A		x	ILCL	0.4	1.1	Base case for cold leg injection	
	K7A		x	ILCL	0.4	1.1	As K5A except smaller ECC rate	
	K9		x	ILCL	0.4	1.1	As K5A except hardware changes. OECD Std. Prob. No. 10	
FLECHT-SEASET/ (Reflood)	3 tests		x		0.3	0.7	Final test matrix not defined. Testing scheduled to start in January 1982	

References for TABLE I-1

2. TREE-NUREG-1003

3. TREE-NUREG-1045

4. TREE-NUREG-1122

6. TREE-NUREG-1148

7. TREE-NUREG-1150

8. NUREG/CR-0467

9. EUR 6970 EN, LEC 80-01

10. TREE-NUREG-1086

11. NUREG/CR-0265

12. NUREG/CR-0492

13. NUREG/CR-0792

14. Proposed LOFT Test Program, March 13, 1981

15. LOBI Project, LEC 80-02, December 1980

16. LOBI Project, LEC 80-03, December 1980

Table I-2 PWR-DESIGN INTEGRAL SYSTEMS - SMALL BREAKS

Facility	Test #	Break Size (%)	Break Location	Break Type	Pumps	Core Uncovery	Comments	Reference
Semiscale/ MOD-3	S-07-10B	10	CL	C	On	Yes	Blowdown on SG secondary side. USNRC Std. Prob. No. 11	1
	S-SB-4	2.5	CL	NC	On	No	LOFT configuration. Test of scaling effects	2
	S-SB-4A	2.5	CL	NC	On	Yes	Higher power than S-SB-4	2
	S-SB-2A	2.5	CL	C	On	Yes	Higher power than S-SB-2	3
	S-SB-P1	2.5	CL	C	Off	Yes	Pumps on/off tests	4
	S-SB-P7	2.5	CL	C	On	No		4
	S-SB-P3	2.5	HL	C	Off	No		5
	S-SB-P4	2.5	HL	C	On	No		5
	S-TR-2	NA	SRV	NA	Off	Yes	Boil-off test. No ECC.	6
Semiscale/ MOD-2A	S-UT-1	10	CL	C	Off		Similar to S-07-10D. No UHI	13
	S-UT-2	10	CL	C	Off		As S-UT-1 but with UHI	14
	S-UT-3	2.5	CL	C	Off		No UHI	
	NC-2	No break			Off		Single loop; baseline 1Φ, 2Φ, reflux	
	NC-6	No break			Off		Single loop; reflux with noncondensable	
	NC-7	No break			Off		Two loops; 2Φ; loop imbalance	
Semiscale/ MOD-5	5 tests						Test matrix not defined. Testing may start in 1983	
LOFT	L3-0	2.5	CL	NC	On	No	Zero power	8
	L3-1	2.5	CL	NC	On	No	Break flow greater than HPIS flow	9
	L3-7	0.14	CL	NC	On	No	Break flow equal to HPIS flow	11
	L3-5	2.5	CL	C	Off	No	Pumps on/off tests	12
	L3-6/L8-1	2.5	CL	C	On	Yes		
	L3-3/L9-1	2.5	SRV	NA	On/off	No	Loss of feedwater followed by stuck-open SRV	15
LOBI	SLST-02	1	CL	NC	Off		SG secondary cooled down at 100K/h.	
LOBI/SB	4 tests						The test matrix has not yet been defined. Testing is scheduled to start in 1982	
PKL-I	ID1-13	No break			NA	Yes	Steady state reflux mode	7
	ID1-8	No break			NA	Yes	Steady state two-phase natural circulation	7
ROSA-IV/ LSTF	10 tests						Test matrix not defined yet. Testing scheduled to start in 1984	
FLECHT-SEASET/ (Nat. Circ.)	3 tests	No break			NA	Yes	Test matrix not defined yet. Completion of testing scheduled for January 1982.	

References for Table I-2

1. EGG-SEMI-5201
2. NUREG/CR-1293; EGG-2021
3. NUREG/CR-1459; EGG-2038
4. NUREG/CR-1640; EGG-2053
5. NUREG/CR-1727; EGG-2063
6. EGG-SEMI-5227
7. D. Hein and F. Winkler, WRSR Information Meeting October 27 to 31, 1980
8. NUREG/CR - 0959
9. NUREG/CR - 1145
11. NUREG/CR-1570
12. EGG-LOFT-5242
13. EGG-SEMI-5331
14. EGG-SEMI-5333
15. EGG-LOFT-5430

Abbreviations Used in Tables I-1 and I-2

C Communicative

CL Cold Leg

HL Hot Leg

ILCL Intact Loop Cold Leg

ILHL Intact Loop Hot Leg

LP Lower Plenum

NC Non-Communicative

SRV Safety Relief Valve

Table I-3 PWR-DESIGN INTEGRAL SYSTEMS - TRANSIENTS

Facility	Test #	Comments	Reference
LOFT	L6-5	Loss of feedwater	1
	L6-2	Loss of primary coolant flow	2
	L6-1	Loss of steam load	2
	L6-3	Excessive load increase	2
	L6-7/L9-2	Turbine trip (ANO-2)	3
Semiscale/ MOD-5	5 tests	Test matrix not defined Testing may start in 1982	
Rosa-IV	5 tests	Test matrix not defined Testing scheduled to start in 1984	
Rancho Seco (B&W Plant)	March 18, 1978	Feedwater pump trip from 72% power	
St. Lucie (CE Plant)	February 2, 1977	Test of RCP trip from 40% power	
Prairie Island (W Plant)	October 2, 1979	RCP trip and SG tube rupture	
Sequoyah-1 (W plant)	1	Natural Circulation at 3% power	
	4	Natural Circulation at 1% power with asymmetric SG isolation	
ANO-2 (CE Plant)		Turbine trip followed by steam dump valve failure	

References for Table I-3

1. NUREG/CR-1520

2. NUREG/CR-1797

3. Proposed LOFT Test Program, March 1981

Table II-1 BWR-DESIGN INTEGRAL SYSTEMS - 200% PUMP SUCTION BREAKS

Facility	Test #	ECC		Comments	
TLTA-4	6006/3	No	Peak		1
TLTA5	6406/1	Yes		High clad temperatures. Low ECC flow rate.	
TLTA-5A	6425/2	Yes	Ave.	HPCS, LPCS and LPCI	
	6426/1	No	Ave.		
	6423/3	Yes	Peak	Low ECC flow rate	
	6424/1	Yes	Peak	As 6423/1 except average ECC flow rate	
FIST (upgraded TLTA)	1 test			Test matrix not defined yet. Testing may start in late 1982	
ROSA-III	708	No	Ave.	Reference test	2
	733	Yes	Ave.	Low LPCS and LPCI flow rates	

References for Table II-1

1. GEAP-NUREG-23977

2. JEARI-M8738

TABLE II-2 BWR-DESIGN INTEGRAL SYSTEMS - SMALL BREAKS AND TRANSIENTS

Facility	Test #	Comments	Reference
TLTA-5A	6432/1 6431/1	No HPCS HPCS. "Reversed" natural circulation	
FIST (upgraded TLTA)	6 tests	Test matrix not defined yet. Testing may start in late 1982	
ROSA-III	ISP 12	5% Small Break.	
Peach Bottom-2 (BWR/4 Plant)	TT-1 TT-2 TT-3	Turbine trip from 47, 62 and 69% power	1
Browns Ferry (BWR/4 Plant)		Generator load rejection	2
		Feedwater pump trip	3
		Recirculation pump trip	4
KRB		Pressure setpoint oscillation	5
Dresden-3		Oscillator data, neutron flux to reactivity	5
Oyster Creek		One and five pumps trips from various power levels	6

References for Table II-2

1. EPRI NP-564, June 1978.

2. S. L. Forkner, D. L. Bell, and E. N. Winkler, TVA, July 1978.

3. E. N. Winkler, S. L. Forkner, and D. L. Bell, TVA, July 1978.

4. D. L. Bell, S. L. Forkner, and E. N. Winkler, TVA, July 1978.

5. NEDO - 21506, January 1977

6. NEDO - 10802, February 1973.

Table III-1 SEPARATE EFFECTS - REFLOOD PROCESSES

Facility	Test #	P (MPa)	$T_{clad\ init}$ (K)	q'_{max} (kW/m)	V_{Flood} (mm/s)	Spray	Types of Rods	No. of Rods	Comments	Reference
FLECHT-	31504	0.276	1136	2.3	24.6	No	Electrical	161		1
SEASET	35807	0.276	1136	0.89	10.4	No		161		
(Un-	34209	0.138	1136	2.3	24.6	No		161		
blocked	31701	0.276	1136	2.3	76.0	No		161	USNRC Std.	
bundle)	31805	0.276	1136	2.3	20.3	No		161	Prod. No. 9	
GOTA	22	0.1	877	1.9	12	Yes	Electrical	64	Combined reflood and	2
	21	0.1	972	1.9	12	Yes		64	spray tests for	
	29	0.1	985	1.9	8	Yes		64	BWR geometry	
	42	1.0	870	1.9	12	Yes		64		
NRU	104	0.28	997	1.8	97	No	Nuclear Unpressurized	32		3
	108	0.28	816	1.8	36	No		32		
	109	0.28	899	1.8	33	No		32		
	115	0.28	1181	1.8	241	No		32		
	122	0.28	1080	1.8	1[illegible]	No		32		
	127	0.28	791	1.8	25	No		32		
SCTF	10 tests						Electric	2000	Test matrix not defined as yet. Testing scheduled to start in 1981	

References for Table III-1

1. NRC/EPRI/W Report No. 7; NUREG/CR-1532

2. Studsvik/RL-78/59; NORHAV S-046

3. NUREG/CR-1882

Table III-2 SEPARATE EFFECTS - CORE THERMAL HYDRAULICS (OTHER THAN REFLOOD)

Facility	Test #	Steady State	Stability Limit	Blowdown	Reflood	Boil-off	Level Swell	p_{max} (MPa)	q'_{ave} (kW/m)	Comments	Reference
FRIGG	301047		x					5	40.6	Full length 36 rod bundle. Axial and radial void distribution	
	613130		x					6.9	28.7		
	613132		x					6.9	38.7		
	613123		x					3.0	28.6		
	613124		x					3.0	28.6		
THTF	177			x				15		Full-length	2
	3.02.10F				x	x	x	7	0.95	49 rod bundle	3
FLECHT-SEASET (Un-blocked bundle)	35557					x		0.4	0.84	Full-length 161 rod bundle	4
KWU/SWR (Two-bundles)	B3	x					x	0.5	0.64/0.87	Two parallel 49 rod full-length BWR bundles	5
	B4	x					x	0.5	1.28/1.75		
	B11	x					x	0.1	0.64/0.78		
	B5	x					x	1.0	0.64/0.87		
THETIS	Fig. 5 & 6	x					x	0.2-4	0-0.68	Full-length 61 rod bundle	6
TLTA-5A	6441/7-T5					x		5.5	1.5	Full-length BWR bundle; 64 rods	7
	6441/6-TP1				x	x		2.3	1.5		7
PBF	LOC-11C			x	x			15	67	Nuclear fuel rod	8
	LLR-3			x	x			15	41		9
	LOC-3			x	x			15	56		10

Table III-2 (Continued)

References for Table III-2

1. FRIGG-2, AB Atomenergi, Sweden, 1968
2. NUREG/CR-1476
3. ORNL/BDHT-2370
4. NRC/EPRI/W Report No. 7; NUREG/CR-1532
5. Forderungsvorhaben BMFT RS36-RS36/C
6. G. L. Shires, K. B. Pearson and A. D. Richards, The Thermal Performance of a Partially Filled Fuel Cluster, BNES Journal, October 1980
7. Preliminary Draft Report by D. Seely and R. Muraridharan
8. NUREG/CR-303
9. TFBP-TR-315
10. TFBP-TR-328

Table III-3 SEPARATE EFFECTS - ECC BYPASS IN PWR DOWNCOMER

Facility	Test #	Type of Test			T_{ECC}	T_{Wall}	$J^*_{1,in}$	Ramp Time	Reference
		CCFL	Ramped Steam	CIT	(°F)	(°F)		(s)	
BCL (2/15-scale)	26202-26207	x			77		0.1		
	26306-26310	x			210		0.1		
	26502-26507	x			82		0.06		
	26508		x		80	T sat	0.1	6	1
	2[illegible]11		x		210	T sat	0.1	18.5	
	29402		x		400	T sat	0.1	12.7	
	29302		x		525	T sat	0.1	13.4	
CREARE (1/15-scale)	8.0193			x					2
	H 195		x		78	350	0.116	20	
	H 196		x		80	350	0.116	27	3
	H 197		x		81	350	0.116	33	
	H 198		x		82	350	0.116	64	
UPTF (Full-scale)	10 tests	The test matrix is not defined yet. Testing is scheduled to start in 1985							

References for Table III-3

1. NUREG/CR-1657
2. CREARE TN-271
3. CREARE TN-252; NUREG-0281

Table III-4 SEPARATE EFFECTS - BWR UPPER PLENUM PROCESSES

FACILITY: SSTF

1. Blowdown experiments with upper plenum mixing and upper tieplate CCFL breakdown. Test to start in July 1981.

   Test matrix not available yet

Table III-5 SEPARATE EFFECTS - BWR JET PUMPS

Facility	Comments	Reference
G. E. Full Scale	Subcooled Steady State, 1st Quadrant	NEDO-10329
INEL 1/6 Scale	Subcooled Steady State, Four Quadrants	EGG-LOFT-5063 LTR 20-105

Table III-6 SEPARATE EFFECTS - BREAK FLOW

Facility	Test #	ΔT sub (K)	D (mm)	L/D	Comments	Reference
Marviken-CFR	1	0-15	300	3.0	Flared Nozzle Exit	The Marviken Full Scale Critical Flow Tests. MXC-201 through MXC-226 Sweden 1979
	2	0-30	300	3.0		
	4	0-30	509	3.1		
	7	0-15	300	1.0		
	13	0-30	200	3.0	Straight Nozzle Exit	
	17	0-30	300	3.7		
	19	0-5	300	3.7		
	20	0-5	500	1.5		
	21	0-30	500	1.5		
	22	0-50	500	1.5		
	23	0-5	500	0.3		
	24	0-30	500	0.3		
	25	0-5	300	1.7		
	26	0-30	300	1.7		
Wyle/L3-1 Nozzle	WSB03R	0-25	16.2		Stratified flow upstream of nozzle	G.E. Grueu, Small Break Calibration Data Draft C, 10-29-79

Table III-7 SEPARATE EFFECTS - STEAM GENERATOR PROCESSES

Facility	Test #	Type		Primary Fluid		Pressure		Comments	
		UTSG	OTSG	Subcooled	Two-phase	Reflood	High		Reference
FLECHT-SEASET/SG	22010	x			x	x		Reference run	1
	21806	x			x	x		Low inlet quality	1
	22608	x			x	x		Low secondary level	1
	23402	x			x	x		High primary flow rate	1
B&W	68,69,70		x	x			x	Steam load increase	2
	74,75,76		x	x			x	Feedwater flow increase	2
	28,29		x	x			x	Loss of feed water	3

References for Table III-7

1. NRC/EPRI/W Report No. 4, January 1980

2. G. W. Loudin and W. T. Oberjohn, Transient Performance of a Nuclear Integral Economizer Once - Through Steam Generator (Proprietary)

3. H. R. Carter and D. D. Schleappi, Nuclear Once - Through Steam Generator (OTSG and IEOTSG); Loss of Feedwater Flow (LOFW) Test. (Proprietary)

Abbreviations used in Table III-7

UTSG U-tube Steam Generator
OTSG Once Through Steam Generator

Table III-8 SEPARATE EFFECTS - PRESSURIZER

References:

1. "Dynamic Response of Reactor Plant to Load Swings, Core 1, Seed 2," Duquesne Light Company Shippingport Atomic Power Station Report DLCS-3630101, Test Results T-643738, April 20, 1961.

2. E. E. Bruckner and K. W. Tong, "The Compression of Initially Saturated Vapours," Syracuse Universe Research Institute (Report No. ME 761-790A), June 1961.

Table IV BASIC TWO-PHASE FLOW PROCESSES-SUMMARY

Facility	Fluids		Dimensions		Type								
	A/W	S/W	1-D	2-D	Steady State	Tran-sient	Phase Change	Gravity Dominated	Critical Flow	CCFL/Entrainment	Phase Distribution	Pressure (MPa)	Reference
RPI (1' x 3')	x			x	x			x			x	0.1	1
RPI (3' x 3')	x			x	x			x		x	x	0.1	
Battelle Vessel		x	x			x	x	x	x			7.1	2
GE Small Vessel		x	x			x	x	x	x			6.9	3
GE Large Vessel		x	x			x	x	x	x			6.9	3
Moby Dick (N_2/W)	x		x		x				x		x	1.0	4
Moby Dick (S/W)		x	x		x		x		x		x	0.8	5
KfK-Nozzle (S/W)		x	x		x		x		x			5.0	6
KfK-Nozzle (A/W)	x		x		x				x			0.8	6
BNL-Nozzle		x	x		x		x		x		x	1.0	7
Canon		x	x			x	x		x			3.2	8
Super-Canon		x	x			x	x		x			15.0	9
Univ. of Houston	x		x		x			x		x	x	1.0	10
Dartmouth Tubes	x		x		x			x		x		1.0	11
Dartmouth Tubes	x		x		x			x		x		1.0	12

References for Table IV

1. NUREG/CR-0418
2. OECD Std. Prob. No. 6, Battelle Institute Frankfurt, February 1977
3. G.E. Draft Report, April 22, 1980
4. DTCE/STT/SETRE Note T.T. no. 599, Fevrier 1977
5. M. Reocreux, Thesis, Grenoble 1974
6. KFK 2902, Juli 1980
7. BNL-NUREG-26003
8. C.E.A.-C.E.N.G., Note T.T. No. 547, Avril 1977
9. TT/SETRE/79-2-B/BR6, Fevrier 1979
10. NUREG/CR-0617
11. EPRI NP-1165
12. G. B. Wallis, et al., Int. J. Multiphase Flow, Vol. 7, pp. 1-19, 1981

Abbreviations Used in Table IV

1-D One-Dimensional
2-D Two-Dimensional
A/W Air-Water
S/W Steam-Water
N_2/W Nitrogen-Water

APPENDIX A

Quantification of Clad Temperature Signatures

The measured time behavior signature of fuel clad temperature reflects the local core hydraulics. For example, the time when the clad temperature commences its first excursion in the case of large cold leg break LOCA (signature illustrated in Figure A-1) denotes the condition of an increase in the local void fraction and a decrease in the local mass velocity. The temperature decrease after its first peak is a consequence of an enhanced local cooling caused by a surge of coolant either from above or from below. The last local quench, at time t_{LQ} is caused by the reflood process. A small-break LOCA clad temperature signature is illustrated in Figure A-2 where the onset of temperature excursion indicates the local increase in void fraction caused by the falling liquid or froth level or the depletion of the vessel coolant inventory.

Given the fact that (a) the majority of experiments are conducted employing electric simulators of the nuclear fuel rods, (b) most tests feature fairly extensive measurement locations for the clad temperature, and (c) very few tests provide indication of the local void fraction in the core while measurements of the local fluid velocities in the core interior are extremely rare, the information provided by the clad temperature signatures presents the only feasible way of evaluating the code performance where it matters most.

A computer code calculates such signatures for each computational cell representing a given core region. There may be more than one measured signature within individual computational cells. Some weighted mean -to diminish the influence of thermocouples facing the unpowered rods or the control rods - must be employed in making comparisons with test data. In the comparisons of the computed and measured clad temperature histories, the upper and the lower bounds of the measured histories should also be shown.

The issues at hand are: (1) how to represent a signature, (2) how to quantify the difference between the measured and the predicted signatures, and (3) how to specify and apply acceptance criteria.

Three single-valued parameters are indicated in Figures A-1 and A-2 that collectively aid in identifying the signature: The local peak clad temperature, LPCT, the local time of peak clad temperature, t_{LPCT}, and the local final quench time, t_{LQ}. If these three parameters are insufficient to identify the signature, one may also consider some forms of the weighted integral.

One such integral may be of the form:

$$I_{\Delta T_{sat}} = \int_0^{t_{LQ}} [T_{CL}(t) - T_{sat}(t)]^n dt$$

where n>1 (say, n=2) emphasizes the peaks above T_{sat}. The quantities in the integral and the upper limit of integration would come either from the code (for $I_{\Delta Tsat,calc}$) or from measurements (for $I_{\Delta Tsat,meas}$), utilizing the weighted average mentioned above.

Another type of the signature integral may assume the same form as that used in the computation of the local amount of clad oxide penetration:

$$\Delta R_{OX} = 2A \int_{o}^{t_{LQ}} \exp\,[B/T_{CL}(t)]dt$$

where it is assumed that no oxide existed at t=0. A and B are given constants featured in the Cathcart-Pawel model.

This is an attractive form because it is relatable to the current regulatory limit for the maximum local clad oxidation. In addition, a summation of all ΔR_{ox} (times the cummulative clad surface within a computational cell), over all computational cells in the reactor core, can be related to the global (core-wide) amount of clad oxidation and, therefore, hydrogen generation. Allowable upper bounds for both are currently specified in the Appendix K acceptance criteria.

Admittedly, the oxidation thickness is not directly measured and the peak clad temperatures reached in experiments may not be high enough to give a significant contribution to ΔR_{ox}. Nevertheless, the difference log $(\Delta R_{ox})_{calc}$ - log $(\Delta R_{ox})_{meas}$ is a meaningful representation of the code ability to calculate the clad temperature signature.

The knowledge of how well the local signatures are predicted sheds light on the code's ability to calculate multidimensional behavior where it matters most. Even in one-dimensional calculations it is important to know whether the code calculates the axial distribution of signatures well. If these comparisons are not adequate then it is questionable whether the code has predictive capabilities, even if the core-wide properties - such as the global peak clad temperature (GPCT) and the global t_{GPCT} and $(t_{LQ})_{max}$ are well predicted.

Should all of the above parameters (LPCT, t_{LPCT}, t_{LQ}, $I_{\Delta Tsat}$ or ΔR_{ox}) be used in quantifying the prediction accuracy or only some of them? Some strategies may ignore the local signatures and only quantify the accuracy for the global parameters, such as GPCT (= the largest PCT anywhere), and the summation of $I_{\Delta Tsat}$ (or of ΔR_{ox}) over all cells. One should bear in mind that the differences between the measured and the predicted times, t_{LPCT} or t_{LQ} or t_{GPCT} or $(t_{LQ})_{max}$ may differ greatly for the large and the small-break LOCAs, or for the short vs long duration transients. For such situations it may, therefore, be more convenient to display the differences in predicted and the measured times, divided by the measured time, to fit many comparisons on the same scale. It appears that condensation of results of comparisons with many test cases could only be made for the global parameters. The local parameters can be plotted as illustrated in Figure A-3.

The most informative way of displaying the calculated vs measured signatures is shown in Figure A-4 pertaining to a vertical stack of computational cells at a given azimuthal location illustrated in Figure A-5. However, such a display is useful for a qualitative rather than a quantitative assessment of the code and is not amenable to confrontation with acceptance criteria.

Figure A-1

Sample T_{CL} signature for large break LOCA

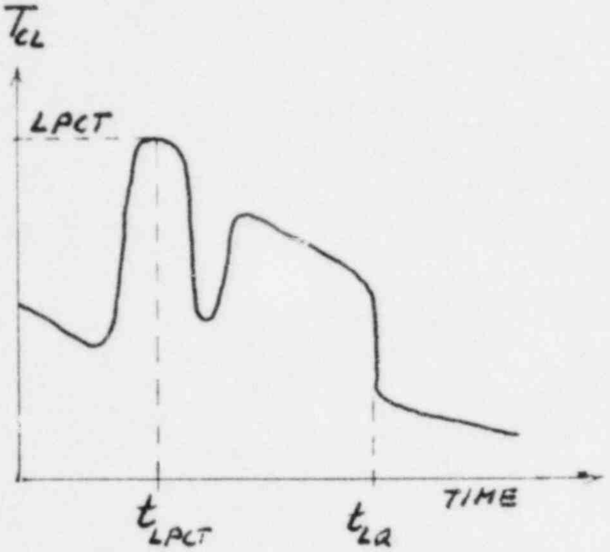


Figure A-2

Sample T_{CL} signature for small break LOCA

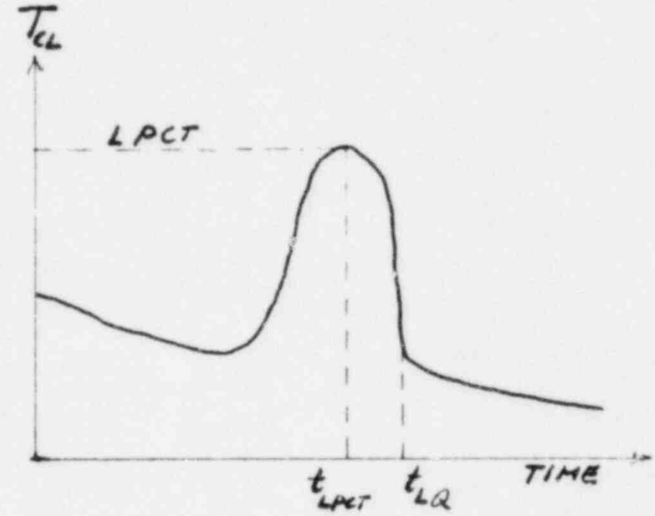
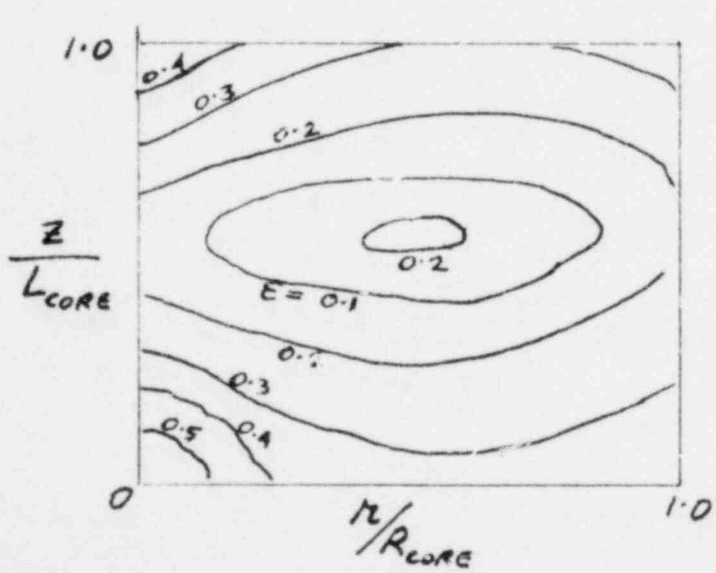


Figure A-3

Contours of equal values of ε for local parameters



$$\varepsilon = \left| \frac{\phi_P - \phi_M}{\phi_M} \right|$$

EXAMPLES FOR ϕ:

LPCT, t_{LPCT}, t_{LQ}, $I_{\Delta T}$, $\log \Delta R_{ox}$ etc.

Figure A-4

Clad Temperature Histories in the Stack of Cells Pertaining to a Specific Circumferential Zone.

(Used for Qualitative Assessment)

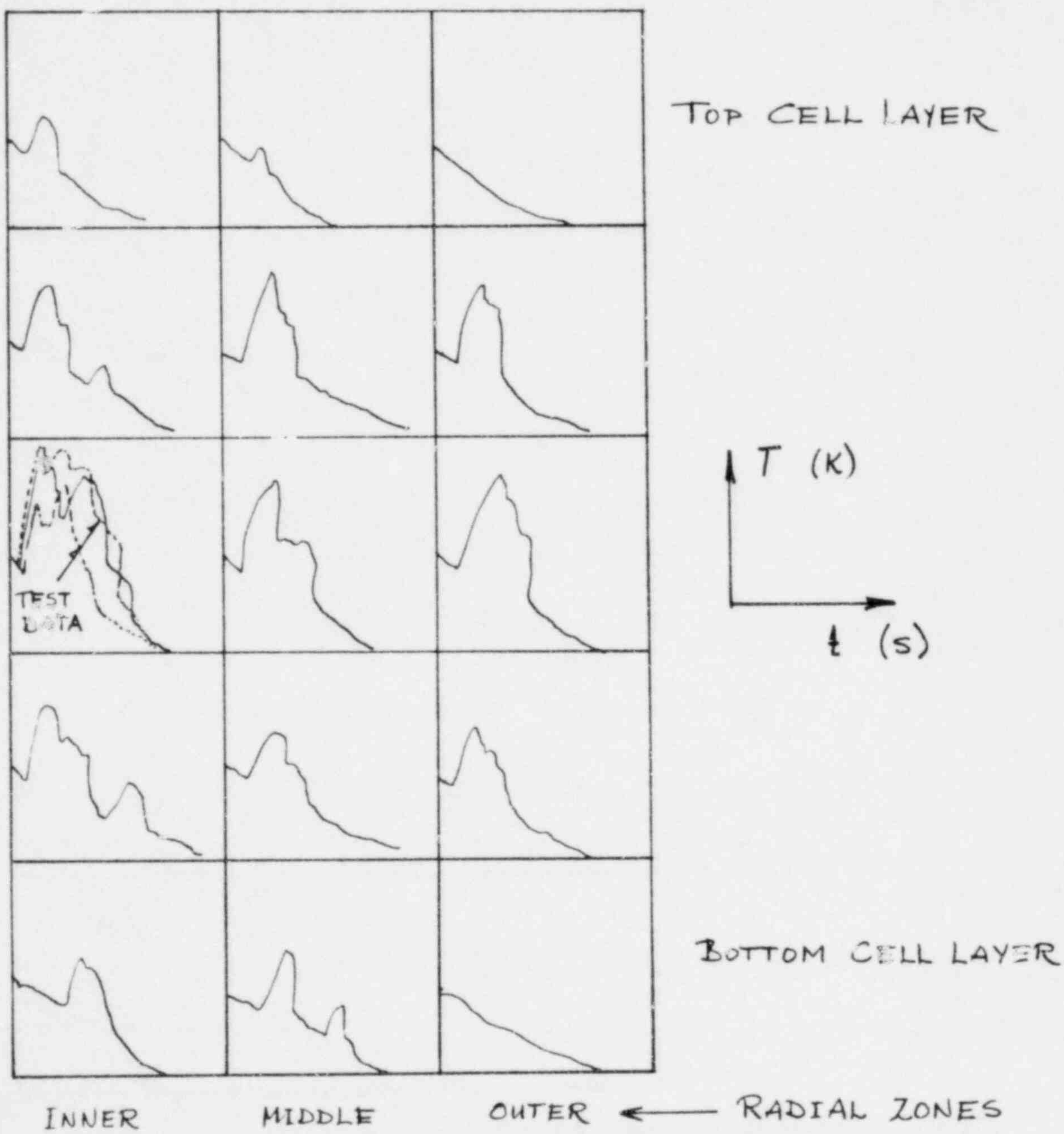
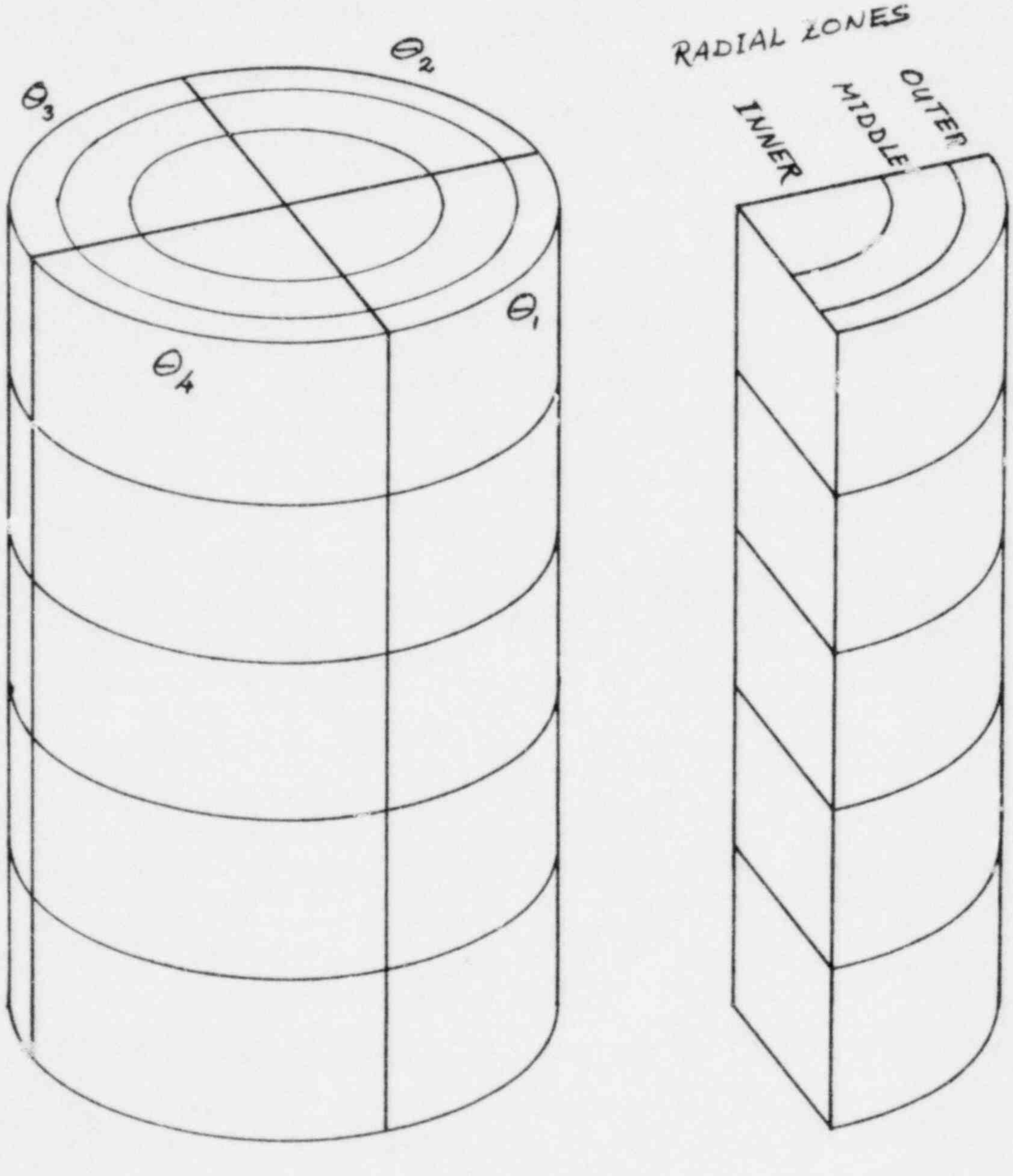


Figure A-5

Example of Three-dimensional Noding of Reactor Core With Illustration of a Stack of Cells for Which the Temperature Histories are Plotted, as in Figure A-4



APPENDIX B

Code Uncertainty Studies

Computer codes contain many empirical correlations. Each such correlation has its own uncertainty "band" or probability distribution. The so-called scatter plots that feature the measured vs the predicted parameter (e.g., heat transfer coefficient) provide information as to the uncertainty range and probability distribution.

Code sensitivity studies performed during code development are designed to indicate which of the many parameters (coefficients) contained in the code are important with respect to the calculated key result. Those that are deemed to be important are then utilized in the statistical analysis of code uncertainty. Suitable sampling procedures such as the "experimental design" and "latin hypercube" are devised for the purpose of developing a Response Surface with a minimum number of computer code runs.

Many runs with a best estimate computer code are performed, featuring a value of each important parameter, X_i, selected within the range of its particular uncertainty. Results of all these runs are fitted into the Response Surface, which is an algebraic expression defining the effect of variation of each parameter, X_i, on the key result (such as peak clad temperature, or the % of clad oxidation in the core) predicted by the given systems code, and for a particular accident/transient scenario.

Other effects that can influence the key results include system discretization (nodalization), time step control, the choice of "user options" in selecting models, and the uncertainties concerning the plant condition at the start of the accident. Advanced LOCA codes are developed in such a manner as to minimize user options. Optimum nodalization schemes for LWR plants should be prescribed during the so-called developmental assessment stage of the code, including the recommendations for the optimum time step control. Nevertheless, these variables may still be present and can influence the total uncertainty in the key results. Other, less tangible phenomena can also affect the code results, as discussed in the text (Section 7.1).

The next step is to perform Monte Carlo calculations with the response surface to obtain the probability distribution for the key result. This step requires the knowledge of the probability distribution function, PDF (X_i), for each of the variable parameters, X_i, defining the Response Surface. Many of these are not sufficiently known; however, effects of different distributions could easily be tested since the Monte Carlo calculations with the Response Surface are fast and economical.

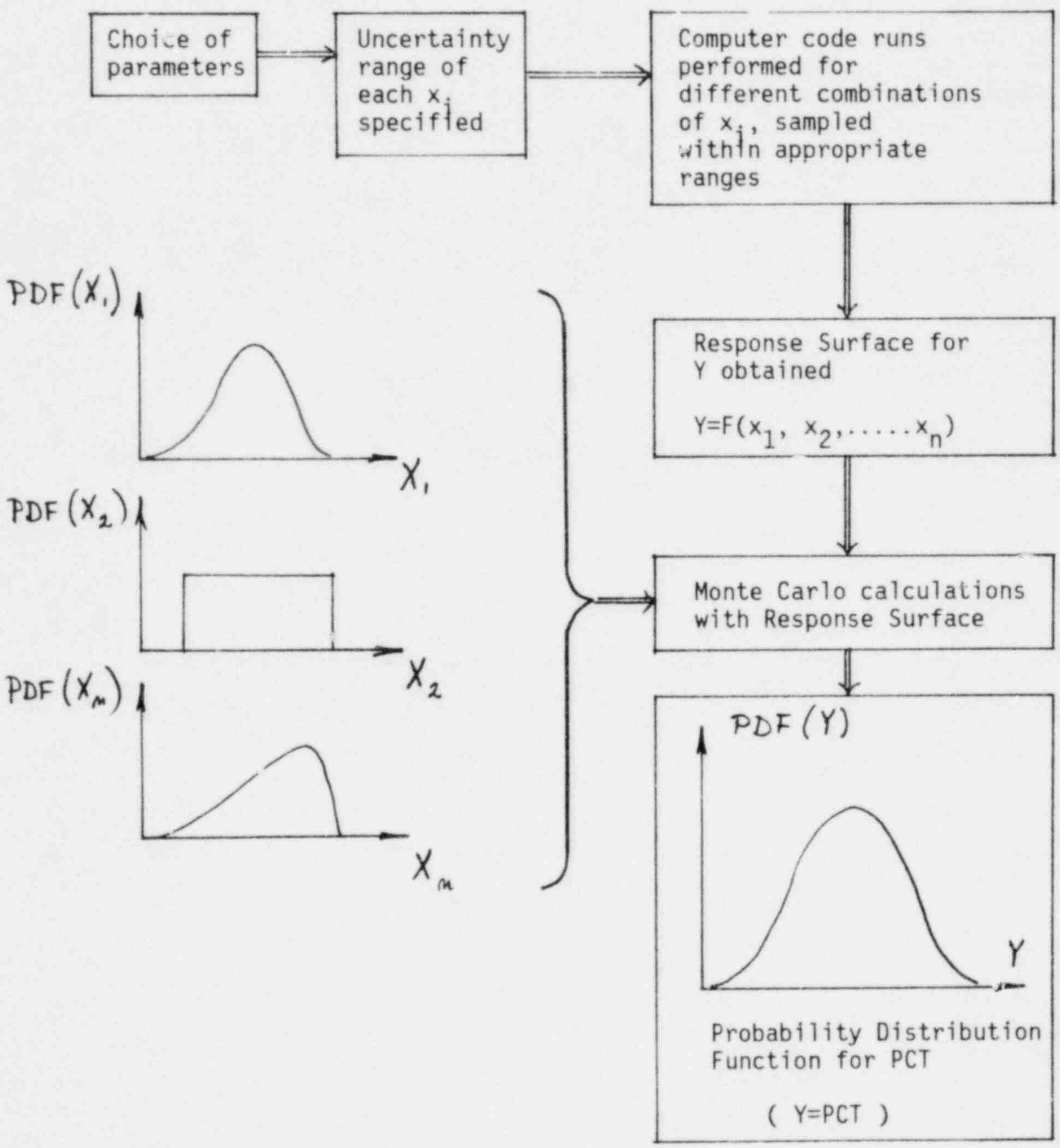
Figure B-1 illustrates the steps in the code uncertainty study.

Code statistical uncertainty studies are very expensive and their results are restricted to the selected accident scenario. In the past, when the main emphasis was placed on the large-break loss-of-coolant accident, it was reasonable to argue for the need for such studies to (a) obtain a feel for the probability distribution of the computed key result (the peak clad temperature, etc.), (b) prioritize the efforts in code development, and (c) prioritize the experimental programs.

The new research direction that does not focus on one particular accident scenario may refrain from extensive use of the statistical uncertainty studies - unless very fast-running and economical tools for analyses become available.

Figure B-1

Illustration of Procedure For Statistical Study of Code Uncertainty



APPENDIX C

Expanded Description of the Basic Tests Selected for Code Assessment

The Basic Two-Phase Flow tests listed in Table IV are described in more detail in the following under the general headings:

- Code Modeling Assessed
- Type of Experiment
- Test Apparatus
- Test Parameters
- Test Data
- Tests Selected for Code Assessment

This expanded description of the Basic Tests is provided because these facilities cannot be described by reference to a reactor component or to the reactor itself.

1. RPI (1' x 3') TWO-DIMENSIONAL PHASE SEPARATION EXPERIMENT

Code Modeling Assessed:

- Void fraction distribution in 2-D upflow
- Flow regimes in 2-D flow
- Interfacial friction

Type of Experiment: Steady state 2-D air-water tests.

Test Apparatus: The test section is a 0.91 m high, 0.30 m wide and 13 mm deep rectangular box. Mixture inlet is at the bottom and outlets are in side walls close to the top. 24, 6 mm diameter rods may be mounted vertically.

Test Parameters:

- Inlet flow rate
- Inlet quality
- Mixture outlet: One or two

Test Data:

- Void distribution

Tests Selected for Code Assessment:

Test #	Rods	# of Outlets	Flowrate	Quality
1	Out	1	low	low
3	Out	2	low	
6	In	1	low	
8	In	2	low	
11	Out	1	low	high
14	Out	2	high	
15	In	1	low	
18	In	2	high	

2. RPI (3' x 3') TWO-DIMENSIONAL PHASE SEPARATION EXPERIMENT

Code Modeling Assessed:

. Void distribution in 2-D countercurrent flow
. Flow regimes in 2-D flow
. Interfacial friction

Type of Experiment: Steady state. 2-D air/water tests.

Test Apparatus: The test section is a 0.91 m high, 0.91 m wide, and 13 mm deep rectangular box. The water inlet is in a side wall at the top while the air inlet is at the bottom.

Test Parameters:

. Water flow rate
. Air flow rate

Test Data:

. Void distribution
. Velocity distribution

Tests Selected for Code Assessment

Test #	Air Flow Rate	Water Flow Rate
1	low	low
2	low	high
3	high	low
4	high	high

3. BATTELLE VESSEL BLOWDOWN EXPERIMENT

Code Modeling Assessed:

. Critical flow through orifice
. Initial flashing delay
. Level swell due to flashing

Type of Experiment: Blowdown of large vessel.

Type Apparatus: 11.2 m tall vessel with ID = 0.77 m. No internals

Test Parameters:

. Nozzle at top or at the bottom
. Nozzle diameter

Test Data:

. Mass flow rate through orifice
. Liquid level in vessel

Test Selected for Code Assessment:

SWR-2R (Top blowdown through 64 mm orifice)

4. G.E. LARGE VESSEL BLOWDOWN EXPERIMENT

Code Modeling Assessed:

. Critical flow through Venturi nozzle
. Initial flashing delay
. Phase separation due to gravity
. Interfacial friction

Type of Experiment: Top or bottom blowdown of vessel initially partially water filled.

Test Apparatus: 4.3 m tall vessel with ID = 1.19 m. Blowdown through a Venturi nozzle from top (3.20 m) or bottom (0.76 m) of vessel.

Test Parameters:

- Nozzle diameter
- Blowdown from top or bottom
- Initial water level

Test Data:

- Flow rate through nozzle
- Δp distribution in vessel

Tests Selected Code Assessment:

Test #	D_{nozzle} (mm)	Type	Water Level (m)
5801-15	63.5	Top	1.68
5803-2	76.2	Bottom	2.90

5. G.E. SMALL VESSEL BLOWDOWN EXPERIMENT

Code Modeling Assessed:

- Critical flow through orifice
- Initial flashing delay
- Phase separation due to gravity
- Interfacial friction

Type of Experiment: Top blowdown of vessel initially partially water filled.

Test Apparatus: 4.3 m tall vessel with ID = 0.30 m. Blowdown through orifice at top of vessel.

Test Parameters:

- Diameter of orifice
- Initial water level

Test Data:

. Flow rate
. Wp distribution in vessel

Test Selected for Code Assessment:

8-21-1 (Diameter of orifice = 9.53 mm; initial water level = 2.71 m)

6. MOBY DICK STEAM-WATER TESTS

Code Modeling Assessed:

. Subcritical and critical flow of steam-water
. Frictional pressure drop in single-phase and two-phase flow

Type of Experiment: Steady state steam-water flow in an approximately 2.7 m long, 20 mm ID vertical pipe followed by a 7° diffusor.

Test Parameters:

. Inlet pressure and temperature
. Back pressure

Test Data:

. Axial pressure distribution
. Axial void distribution
. Radial void distribution upstream of throat
. Identification of critical flow rates

Tests Selected for Code Assessment:

Test #	G crit (kg/m^2S)	Back pressure
400	6526	Low
401	6465	High
406	8718	
455	10176	

7. MOBY DICK NITROGEN WATER TESTS

Code Modeling Assessed:

. Subcritical and critical flow of two component two-phase mixtures
. Frictional pressure drop for single-phase and two-phase flow

Type of Experiment: Steady state nitrogen-water flow in an approximately 2.7 m long, 14 mm ID vertical pipe, followed by a 7° diffusor.

Test Parameters:

- Inlet quality
- Inlet pressure
- Back pressure

Test Data:

- Axial pressure distribution
- Axial void distribution
- Radial void distribution upstream of throat
- Identification of critical flow rates

Tests Selected for Code Assessment:

Test #	Quality 10^4	Back pressure	Comments
3095	0		Subcritical flow
3167	0.75		Choked Flow
3176	0.94	High	Choked Flow
3177	0.93	Low	Choked Flow
3087	5.91	High	Choked Flow
3089	5.90	Middle	Choked Flow
3091	5.95	Low	Choked Flow
3052	8.72		Choked Flow
3141	51.3		Choked Flow

8. KfK NOZZLE EXPERIMENT

Code Modeling Assessed:

- Critical flow of air-water and steam-water
- Frictional pressure drop of two-phase flow

Type of Experiment: Steady state air-water and steam-water flows through a nozzle contracting from 80 to 16 mm ID, followed by a 700 mm long horizontal tube with abrupt exit expansion.

Test Parameters:

- Fluids: Nitrogen-water and steam-water
- Inlet quality (including subcooled liquid)
- Inlet pressure
- Back pressure

Test Data:

. Axial pressure distribution
. Axial liquid temperature distribution
. Void fractions upstream and downstream of contraction

Tests Selected for Code Assessment:

Test #	Test Conditions
V02.08.78/13.59	Cold water (subcritical)
V15.09.78/11.11	Steam-water (choked)
V02.08.78/15.20	Air-water (choked)

9. BNL NOZZLE EXPERIMENT

Code Modeling Assessed:

. Choked flow in convergent-divergent nozzle
. Phase distribution in choked flow
. Flashing and condensation in high speed nozzle flow

Type of Experiment: Steady state steam-water upflow in a linearly convergent-divergent nozzle. The upstream and downstream diameter of the nozzle is 51 mm ID. The throat is 25 mm ID. Total nozzle length is 550 mm.

Test Parameters:

. Inlet pressure and temperature
. Back pressure

Test Data:

. Main flow rate
. Axial pressure distribution
. Axial and radial void distribution

Tests Selected for Code Assessment:

Test #	Inlet Pressure (kPa)
141	239.7
145	306.2
133	350.3
140	465.2

10. CANON EXPERIMENT

Code Modeling Assessed:

- Pipe blowdown processes: flashing, interfacial friction and wall friction
- Critical flow from constant area pipe and through orifice

Type of Experiment: Blowdown of a 4.39 m long, 102.3 mm ID, horizontal pipe. An orifice may be mounted at the outlet.

Test Parameters:

- Initial pressure ($\leq$ 3.2 MPa)
- Initial temperature
- Orifices from 30 mm to full opening

Test Data:

- Transient pressure and temperature distributions
- Transient, area average void fraction at one axial location.

Tests Selected for Code Assessment:

T_{init} (K)	$D_{orifice}$ (mm)
473	50
503	
473	Full opening
503	

11. SUPER-CANON EXPERIMENT

Code Modeling Assessed:

- Pipe blowdown processes: flashing, interfacial friction
- Critical two-phase flow through orifice

Type of Experiment: Blowdown of a 4.389 m long ID, horizontal pipe. An orifice may be mounted at the outlet.

Test Parameters:

- Initial pressure ($\leq$ 15.0 MPa)
- Inital temperature
- Orifices from 30 mm to full opening

Test Data:

- Transient pressure and temperature distribution
- Transient area average void fraction at one axial location
- Reaction force of pipe.

Tests Selected for Code Assessment:

T_{init} (K)	$D_{orifice}$ (mm)
553	50
573	50
553	Full opening
573	Full opening

12. UNIVERSITY OF HOUSTON FLOODING TUBE

Code Modeling Assessed:

- Liquid entrainment in the annular flow regime
- CCFL

Type of Experiment: Cocurrent and countercurrent air-water flow in a vertical 3.96 m long, 50.8 mm ID tube. The air inlet is at the bottom; water is injected through a porous sinter at the tube middle.

Test Parameters:

. Air flow rate
. Water flow rate

Test Data:

. Film flow rates (down and up)
. Film thicknesses
. Flow rate of entrained water
. Axial pressure drops

Tests Selected for Code Assessment:

For water flow rates of 12.6 g/s and 126 g/s the air flow rate was increased stepwise to above the flooding point.

13. DARTMOUTH TUBE COUNTERCURRENT FLOW EXPERIMENTS

Code Modeling Assessed:

. CCFL in annular flow

Type of Experiment: Countercurrent air-water flow in vertical tubes with IDs from 6.4 to 152 mm. Sharp and rounded inlets at the top.

Test Parameters:

. Tube diameter
. Air flow rate

Test Data:

. Liquid flow rate

Data Chosen for Code Assessment: Tube diameters 6.4, 25, and 152 mm. Air flow rate varied to produce the "flooding curve."

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16. ABSTRACT (200 words or less)

Systems codes play a very important role in evaluation of safety of nuclear power plants. In contrast to the Evaluation Model codes in which the most pessimistic and conservative combination of events and processes are assumed, the Best Estimate systems codes attempt to describe the physical processes as realistically as possible. They are, therefore, amenable to an indepth assessment through confrontation with experimental evidence gathered, worldwide, in the course of reactor safety research.

That confrontation has many facets and the purpose of this report is to describe the issues, considerations, and a recommended course.

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