NUREG-1271

Guidelines and Procedures for the International Code Assessment and Applications Program

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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ABSTRACT

This document presents the guidelines and procedures by which the International Code Assessment and Applications Program (ICAP) will be conducted. The document summarizes the management structure of the program and the relationships between and responsibilities of the United States Nuclear Regulatory Commission (USNRC) and the international participants. The procedures for code maintenance and necessary documentation are described. Guidelines for the performance and documentation of code assessment studies are presented. An overview of an effort to quantify code uncertainty, which the ICAP supports, is included.

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PREFACE

An International Code Assessment and Applications Program (ICAP) is being carried out by a number of countries and organizations and coordinated by the U.S. Nuclear Regulatory Commission (USNRC). Its purpose is to support the effort to obtain a considered view of the accuracy and validity of USNRC thermal-hydraulic codes over their range of applicability, while suggesting possible improvements, as needed, to the codes. Quantitative methods will be used by several of the ICAP members to characterize the accuracy of the codes and code assessment activities within the international community will help support these methods. Coordination is proposed because of the recognized need to conserve resources throughout the international community by sharing the effort and benefit among participants. This document describes the procedures by which the coordinated international assessment program will be managed. This document is intended to be maintained in an active and current status.

ACKNOWLEDGEMENTS

Credit is given to the following individuals who have contributed to the completion of the Guidelines and Procedures for the ICAP. At the Nuclear Regulatory Commission those persons are D. Bessette, P. Ting, L. Shotkin, F. Odar and P. Larkins. At the Idaho National Engineering Laboratory those persons are R. Hanson, G. Wilson and G. Case.

1. INTRODUCTION

Increasing numbers of light water reactors (LWRs) are becoming operational throughout the world. It is generally recognized that sound management of this growing resource requires the development of advanced thermal-hydraulic computer codes for use in safety related studies. Such codes enhance the ability to understand and predict the response of LWRs to postulated transients and breaks.

This common interest has led the United States Nuclear Regulatory Commission (USNRC) to organize an international cooperative code assessment and application program that is intended to:

- Support the efforts of the USNRC to determine the ability of the code to appropriately represent important physical phenomena and support the quantitative determination of code accuracy;
- Share user experience on code assessment and to present a well documented assessment data base;
- Share experience on code errors and inadequacies and cooperate in removing the deficiencies to maintain a single, internationally recognized code version; and
- Establish and improve user guidelines for applying the code.

The sharing of effort by the international nuclear community will allow these goals to be met through an optimum use of limited resources. The cooperative effort is planned to continue over a period of approximately five years beginning in 1985.

Towards these ends, the participating organizations will share information relating to analysis methods and code assessment activities. In general, the USNRC will furnish its advanced, best-estimate thermal-hydraulic codes, test data and code assessment results in exchange for code assessment studies and the experimental data necessary for code verification. Other product exchanges will be considered as specifically provided for in the individual agreements between the USNRC and program participants. Organizations with either signed or anticipated agreements and the codes being utilized are listed in Table 1.

The use of advanced thermal-hydraulic codes has been promoted by the increasing trend to perform best-estimate analyses of transients and breaks. Formerly, the tendency was to use prescriptive criteria and to insert conservatisms wherever uncertainty existed or information was judged to be lacking or imprecise. This approach to modeling was intended to ensure a "conservative" prediction of plant behavior. With the increased emphasis on small breaks and transients, however, the capability to accurately predict plant response becomes more important. This is particularly true regarding operator procedures and training.

Use of an advanced, best-estimate code for safety analyses requires that the code accuracy be quantified through assessment over the range of applicability. Also required is a sound physical understanding of the underlying reasons for its accuracy and limitations.

The USNRC has plans to replace its present 10 CFR 50 Appendix K LOCA/ECCS rule with a best-estimate approach and an evaluation of uncertainty. The USNRC plans to employ results from code assessment studies to provide conclusions relative to code accuracy and uncertainty. In addition, the assessment results will be used to help define where future resources should be expended for experimentation and code improvement.

Management of the International Code Assessment and Applications Program (ICAP) will be according to the procedures defined in Section 2. Discussion of code management within this program is provided in Section 3. Reporting and documentation of the program and information flow is presented in Section 4. Section 5 discussed guidelines for performing and documenting code assessment studies. An overview of the plans for quantifying the code accuracy are described in Section 6.

This report revises the guidelines followed by the USNRC for thermal-hydraulic code assessment as described in NUREG-0676 "Plans for Assessment of Best Estimate LWR Systems Codes," S. Fabic, P. Anderson, July 1981.

TABLE 1. PARTICIPATING COUNTRIES AND CODES

Country	Organization	RELAP5	TRAC-PWR	TRAC-BWR	COBRA-TF
Austria (1)	Austrian Research Center Siebersdorf	X			
Belgium	Tractebel	X			
Brazil (1)	The Comissao National De Energia Nuclear of Brazil	X	X		
Finland	Technical Research Center of Finland	X			
France	Commissariat a l'Energie Atomique	X	X		X
F.R. Germany	Federal Ministry for Research and Technology Kraftwerk Union Aktiengesellschaft Gesellschaft fur Reaktorsicherheit	X	X	X	
JRC-Ispra (1)	The Joint Research Center ISPRA Establishment of the European Atomic Energy Community	X			
Italy	ENEA	X	X	X	
Japan (1)	Japan Atomic Energy Research Institute	X	X	X	X
Korea	Korea Advanced Energy Research Institute	X	X		
Mexico (1)					
Netherlands (1)	Netherlands Energy Research Foundation	X			
Spain	Consejo de Seguridad Nuclear	X	X		

TABLE 1. (Continued)

Country	Organization	RELAP5	TRAC-PWR	TRAC-BWR	COBRA-TF
Sweden	Swedish Nuclear Power Inspectorate Studsvik Energiteknik AB	X	X		
Switzerland	Swiss Federal Institute for Reactor Research	X		X	
CCNAA (Taiwan)	Coordinating Council for North American Affairs	X			
United Kingdom	United Kingdom Atomic Energy Authority Central Electricity Generating Board Nuclear Installations Inspectorate National Nuclear Corporation British Nuclear Fuels Ltd.	X	X		
United States	Nuclear Regulatory Commission	X	X	X	X
Yugoslavia (1)	The Josef Stefan Institute of Yugoslavia	X			

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(1) Agreement Under Discussion

2. MANAGEMENT OF THE INTERNATIONAL CODE ASSESSMENT AND APPLICATIONS PROGRAM

This section summarizes guidelines for management of the ICAP program. In general, these guidelines conform to the individual agreements between the USNRC and the other participants, however, the agreements are governing in the event of possible conflicting provisions. The significant relationships and responsibilities for management of the program are summarized in Figure 1.

2.1 Management Structure and Relationships

2.1.1 Program Coordination

Program success and efficiency can best be achieved with the prime management functions focused in a single coordinator. As developer of the code and common signatory to the bilateral agreements, the USNRC has assumed the responsibility for code support and program coordination and integration.

Coordination and integration of the program support effort for the quantification of code accuracy will be carried out by the USNRC and its contractors at U.S. Department of Energy (USDOE) laboratories with Idaho National Engineering Laboratory providing technical integration. The USDOE is responsible for ensuring that contractors perform all USNRC work within the provisions of their operating contract with the USDOE and that the work is delivered within the approved scope, schedule and funding. The USDOE-Idaho Program Manager or his designated representative for the ICAP program will participate in the Program Group.

2.1.2 Program Management Group

The purpose of the Program Management Group is to serve as a review and coordination body for ICAP. The intent is to have a sufficiently small group for effective interchange, typically one or two members for each participating organization. The terms of reference of the Program Management Group include discussion of:

- The plan of work of the respective organizations for the coming year, e.g., assessments to be performed;
- From an overview perspective, the status and performance of the frozen code and work on the next code version;
- The assessment plan and status of the assessment effort;
- Results from Task Group and Specialist Meetings;
- Program documentation, including the code and its documentation, user guidelines, newsletters and assessment reports; and

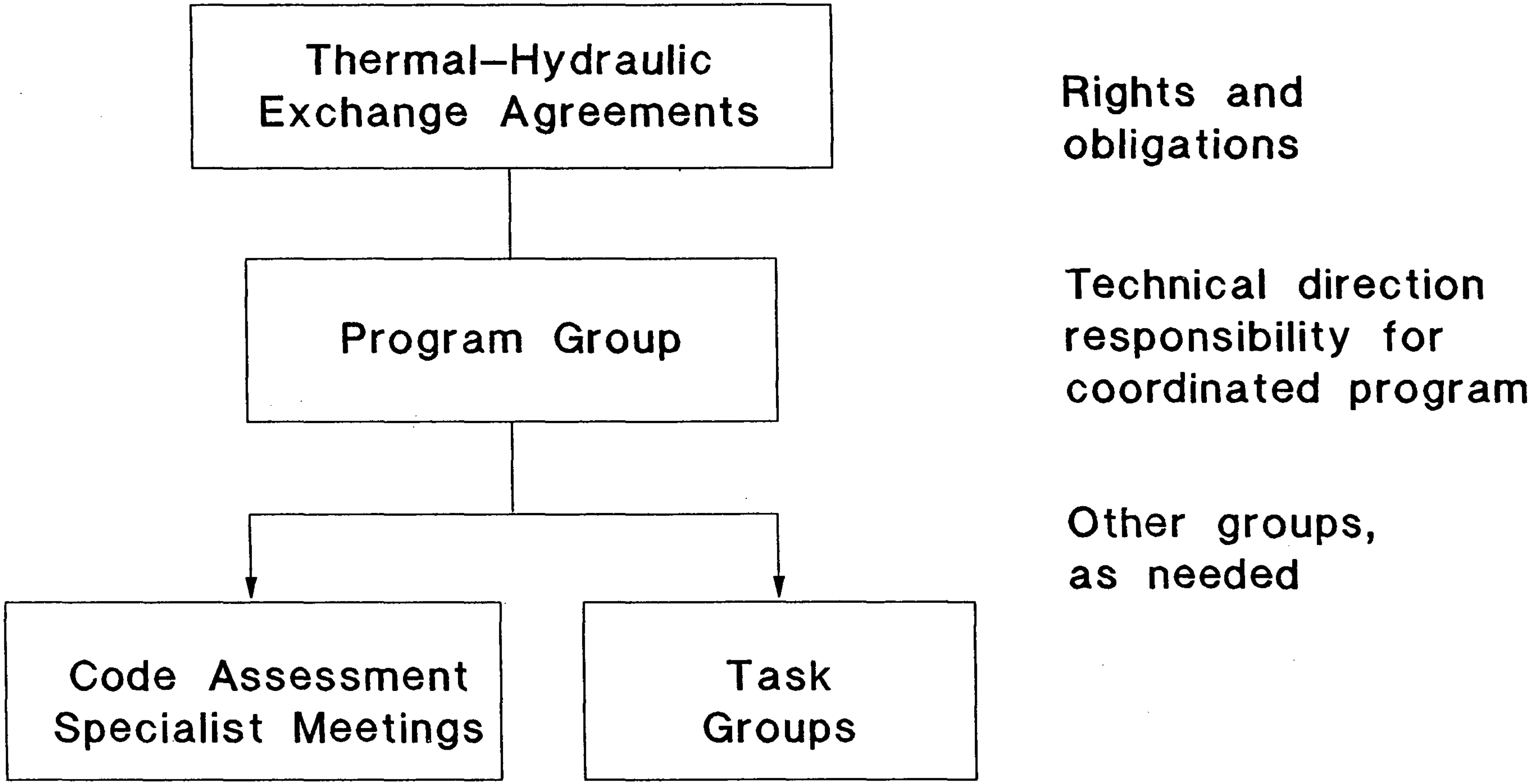

Figure 1. International Code Assessment and Applications Program Management.

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- Procedures for support to the quantification of code uncertainty.

The Program Management Group will meet at least once per year and may meet more often at the discretion of its members. Locations for the meetings may be divided between the United States and countries of other program participants. The Chairman will be provided by the USNRC. It is not the intention of the USNRC to dominate or necessarily lead the discussion of the group, rather, each participant has equal voice. Consensus views on points of discussion will be sought, wherever possible. The USNRC will strongly consider the recommendations of the group in prioritizing its allocation of resources for code maintenance and improvement.

2.1.3 Task Group

A Task Group will meet to resolve specific issues as decided by the Program Management Group and issue written recommendations. The intent is to bring together those specialists actively involved in experimentation and modeling to share information and reach a common understanding relative to specific tasks. Membership of the Task Groups and attendance at meetings may depend on the particular tasks to be addressed. Each participating country represented in the Task Group has equal representation within the group. The Task Group will meet as often as the Program Management Group deems necessary at locations in participating countries. The following are the principal tasks that must be carried out. These tasks may be terminated or new ones created depending on the progress made or problems that may arise:

1. Code Changes. On a regular basis, the Task Group will meet to discuss:

    - Errors discovered, priorities for error corrections, error corrections implemented, input/output and user conveniences for the frozen code version;
    - Model deficiencies and priorities for model improvements for the next code version;
    - Noding studies;
    - Implementation and run time experience;
    - Newsletter; and
    - Code documentation.

2. Assessment Reports. On a regular basis, the Task Group will meet to discuss nonproprietary assessment reports and the review of these reports performed by the USNRC. Discussion will include the qualitative and quantitative assessment carried out by the USNRC, with the objective being to provide a peer review and recommendations.

3. Model Development and Supporting Experimental Work. The Task Group will meet as needed to discuss specific code model deficiencies (e.g., post-CHF heat transfer, two-phase flow), supporting experimental work, and modeling for implementation in a later code version.

4. Plant Applications. The Task Group will meet as needed to discuss plant application studies including noding and sensitivity studies, plant modeling, and results from application of the code to startup tests and plant transients.

2.1.4 Specialist Meetings

Specialist Meetings will be held, as decided upon by the Program Management Group, at locations in participating countries. The purpose will be to bring together code users for presentation of assessment results, both qualitative and quantitative, and exchange information on user experience. There will be at least one such meeting per year.

2.2 Responsibilities of Program Participants

2.2.1 United States Nuclear Regulatory Commission (USNRC)

The USNRC will have the responsibility and provide the resources for:

1. Provision of the code and its documentation;

2. Consultant service and troubleshooting according to agreements;

3. Implementation of code error corrections and enhancements as recommended by participants, but according to the schedule and priorities established by technical and safety needs and available USNRC resources;

4. Development, maintenance and issuance of a quarterly Newsletter for each code;

5. Composition and issuance of the program documentation (see Section 4);

6. Coordination of data supporting the quantification of code accuracy;

7. Providing USNRC and contractor staff to the groups identified in Section 2.1; and

8. Providing such staff for other functions in accordance with the individual agreements between the USNRC and participants.

2.2.2 International Participants

The responsibilities of each participant are defined in its agreement with the USNRC. Generally, the agreements provide for the following responsibilities and resource allocations by each participant:

1. Performing code assessment studies according to the agreement. The guidelines by which the studies will be conducted are described in Section 5;
2. Issuing code assessment reports in English. The content and format of the reports are described in Section 5; and
3. Providing experimental and calculational data used in the code assessment studies necessary to evaluate the studies, to support quantification of code accuracy and to effect code improvements. The necessary forms of information transmittal are covered in Section 5.

In addition to the above responsibilities, participants should send representatives to the Program Management Group, appropriate Task Group and Specialist Meetings identified in Section 2.1.

2.3 Responsibilities of Contact Persons

2.3.1 United States Nuclear Regulatory Commission (USNRC)

The USNRC Program Manager will be responsible for execution of the bilateral thermal-hydraulic exchange agreements. All correspondence concerning the use of the code should be sent to the USNRC Program Manager including errors noted, requests for user assistance and assessment reports. The USNRC Program Manager will work with the responsible USNRC contractor to ensure that appropriate, timely responses are provided to notifications of errors and requests for user assistance. The USNRC Program Manager will have oversight responsibility for the issuance of Assessment Report Reviews, Newsletters, Topical Reports and support to code uncertainty quantification. The Program Manager will also organize and attend Program Group, Task Group and Specialist Meetings of code users. Correspondance to the USNRC Program Manager is to be sent to the following address:

ICAP Program Manager
Reactor and Plant Systems Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Telephone: (301) 427-4920
Telex Number: 908142 NRC BHD WSH

A second copy of technical correspondence should also be sent directly to the USNRC contractor responsible for the code concerned. The USNRC contractors will assist the USNRC in carrying out the program at the direction of the USNRC, and will attend and participate in all appropriate meetings. The responsible individuals are as follows:

TRAC-PWR:
Rick Jenks
Mail Stop K555
Los Alamos National Laboratory
Post Office Box 1663
Los Alamos, New Mexico 87545

Telephone: (505) 667-2021
Telex Number: 9109881773 LOS AL LAB

TRAC-BWR, RELAP5, and COBRA-TF:
Robert Hanson
EG&G Idaho, Incorporated, TSB
Post Office Box 1625
Idaho Falls, Idaho 83415

Telephone: (208) 526-9701
Telex Number: 9105775915 USDOE IDAH

2.3.2 International Participants

Each participant should designate an individual to be responsible for execution of its bilateral thermal-hydraulic exchange agreement within the appropriate organization(s). An alternate may also be designated. All correspondence concerning the use of the code will be sent through the designated contact. This includes errors noted, requests for user assistance, assessment reports, data packages, etc. The USNRC will send project documents including Assessment Reports, Newsletters and Topical Reports to the designated contacts, who will be responsible for further distribution within their organization or country. The designated contacts will also receive notification of all Program Group meetings, relevant Task Group meetings and Specialist Meetings of code users. The designated contacts or their alternates should attend Program Group meetings. Table 2 provides a list of individuals serving as designated contacts.

2.4 Protection of Information

The purpose of this program is to further the knowledge and understanding of the participants in the validation of thermal-hydraulic computer codes. With due protection of information that is proprietary, confidential or privileged, information will flow freely among the participants.

Participants and their contractors are responsible for the protection of proprietary, confidential or privileged information both within and outside the program. It shall be the responsibility of the originating party to clearly identify and mark documents containing proprietary, confidential or privileged information so that such information can be protected from disclosure.

TABLE 2. LIST OF DESIGNATED CONTACT PERSONS FOR PARTICIPATING ORGANIZATIONS

Country	Contact	Organization
Belgium	Dr. E. Stubbe	Tractebel
Finland	Mr. H. Holmstrom	Valtion Teknillinen Tutkimuskeskus
France	Dr. M. Reocreux	Institut De Protection De Surete Nucleaire
F. R. Germany	Prof. E. Hicken	Gesellschaft Fur Reaktorsicherheit
	Mr. F. Winkler	Kraftwerk Union Aktiengesellschaft
Italy	Mr. G. Saponaro	ENEA-DISP
Japan	Dr. K. Sato	Japan Atomic Energy Research Institute
Korea	Dr. Sang Hoon Lee	Korea Advanced Energy Research Institute
Multinational	Dr. H. Stadtke	Commission of the European Communities Joint Research Center ISPRA
Spain	Dr. J. Decarlos	Consejo De Seguridad Nuclear
	Dr. J. Izquierdo	Consejo De Seguridad Nuclear
Sweden	Dr. O. Sandervag	Studsvik Energiteknik AB
Taiwan	Dr. Y. Chen	Institute of Nuclear Energy Research
The Netherlands	Dr. J. Speelman	Energieonderzoek Centrum Nederland
United Kingdom	Mr. J. Fell	United Kingdom Atomic Energy Authority
United States	Dr. N. Lauben	United States Nuclear Regulatory Commission

3. PROCEDURES FOR CODE MANAGEMENT

3.1 Freezing of the Code for Assessment and Application

A principal intention of ICAP is to maintain a standard, internationally recognized code version. To accomplish this goal, changes to the code must be properly managed. Code assessment efforts are hindered when a code is undergoing continuous development. The code must, therefore, be frozen, documented and distributed to code users so that all users have the same code, and assessment results and user experience can be exchanged knowing that results are interrelated and are not specific to a unique code version.

Freezing of the code is defined as the avoidance of code development whereby models and correlations are changed to improve the code predictive capability. Changes to the frozen code are allowed to correct errors, and for improved input/output and user conveniences.

The intent is to freeze each code for a sufficient period of time to be able to establish a considered view of its capability and accuracy over its range of applicability. The current frozen versions are as follows:

TRAC-PF1/MOD1	Version 12.1
RELAP5/MOD2	Cycle 36
TRAC-BF1	Version G1J1

3.2 Procedure for Error Corrections

The procedure for correcting errors is shown in Figure 2 and described in this section. If a participant discovers an error, the information should be sent to the USNRC Program Manager with a copy to the responsible USNRC contractor (see Section 2.3.1). The problem should be described and suggestions for possible fixes may be included. The problem will be reviewed by the USNRC and its contractor to ensure that it is indeed an error and that the problem is relevant to conditions that might conceivably be expected to occur in a power plant. The USNRC and its contractor will correct the problem, within the limits of its available resources and priorities. Error corrections will result in code updates to the frozen version which will typically be released quarterly. News of the error and associated resolution will be included in the next issue of the code newsletter. This includes errors discovered through use of the code within the United States.

3.3 New Code Versions

The use of the code for independent assessment and applications can be expected to reveal deficiencies that need to be addressed through code development. The quantitative and qualitative assessment of the code should be particularly valuable in defining code adequacies and inadequacies. The information generated during the period when the code is kept frozen will be considered continuously and collectively in preparing new versions of the code.

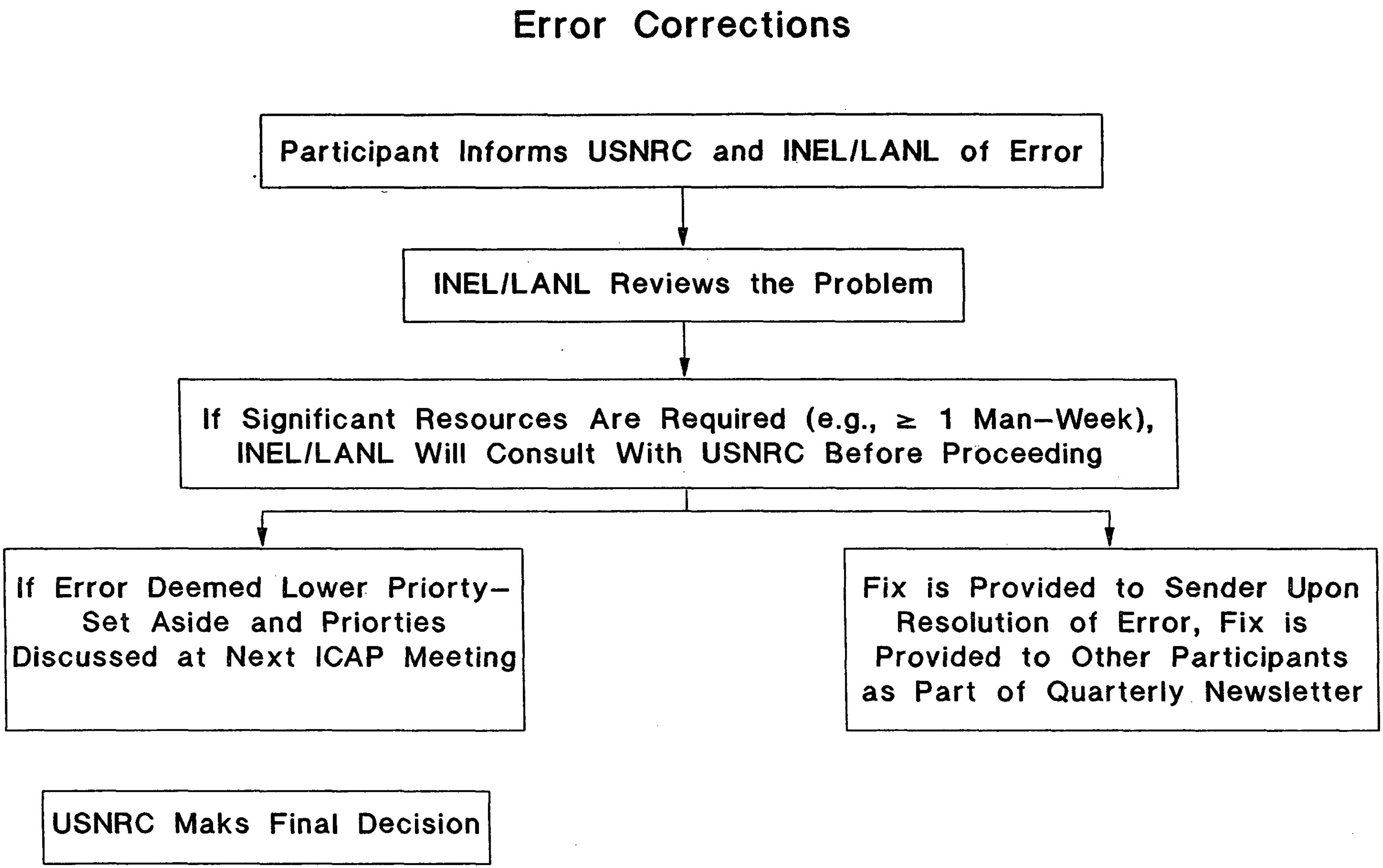

Figure 2. Procedure for error corrections.

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The USNRC will maintain a list of all code revisions suggested in the assessment process and provide priorities for their implementation. The list and associated documentation will form a basis for review by the Program Group, which may provide recommendations for reprioritization. Model improvements will be made in parallel to the assessment of the frozen code, according to priorities and available USNRC resources. User requirements to resolve safety and regulatory needs is the foremost determinant of priorities. The new version of the code will be issued to all participants using that code.

International participants may produce suggested code enhancements resulting from their code assessment studies and experimental and modeling work. Participants may provide new or improved models or correlations. The USNRC will make use of enhancements and data supplied by participants in preparing new code versions.

Keeping in mind the intention that there shall be an officially released, documented version of the code such that all users are working with the same code, the participant supplying enhancements must indicate whether the enhancements should not be incorporated into a new code version because the information is considered to be proprietary, confidential or privileged. In this event, such enhancements cannot be incorporated in an official version of the code.

4. REPORTING AND DOCUMENTS

To meet the goals of ICAP, a well designed and conducted documentation effort is necessary. This section defines the required documentation of program information. The major program documents are summarized in Figure 3. When activities are specified to be carried out by the USNRC, the USNRC itself or USNRC contractors including the Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL) and Sandia National Laboratory, will perform the activities.

4.1 Distribution of Information

All program documents will be available to the USNRC. The USNRC may publish Assessment Reports prepared by participants that do not contain proprietary, confidential or privileged information as NUREG/IA reports, with proper credit to the originator. Documents containing proprietary, confidential or privileged information shall not be published or distributed by the USNRC. Data Packages, whether proprietery or nonproprietary, shall not be published or distributed by the USNRC.

The following documents shall have unrestricted distribution to the program participants:

- Guidelines and Procedures (including assessment plans and associated matrices);
- Executive Summaries of Assessment Reports;
- Nonproprietary Assessment Reports;
- USNRC reviews of Assessment Reports;
- Newsletters; and
- Topical Reports.

If an assessment report will contain proprietary, confidential or privileged information, the participant may prepare the assessment report, if possible, with a view to having the main body of the report as unrestricted and having sensitive information in an appendix. In this case, the main body of the report may be published by the USNRC with proper credit to the originator and would be given unrestricted distribution within the program while the appendix would not be distributed. Proprietary assessment reports will not be published or distributed by the USNRC.

4.2 Code and Related Documentation

The USNRC will provide to each participant entitled to receive the information the code and accompanying documentation, including the code structure, documentation and discussion of models, correlations and numerics, developmental assessment and user guidelines. Included are code updates and new code versions that may result during the course of the ICAP

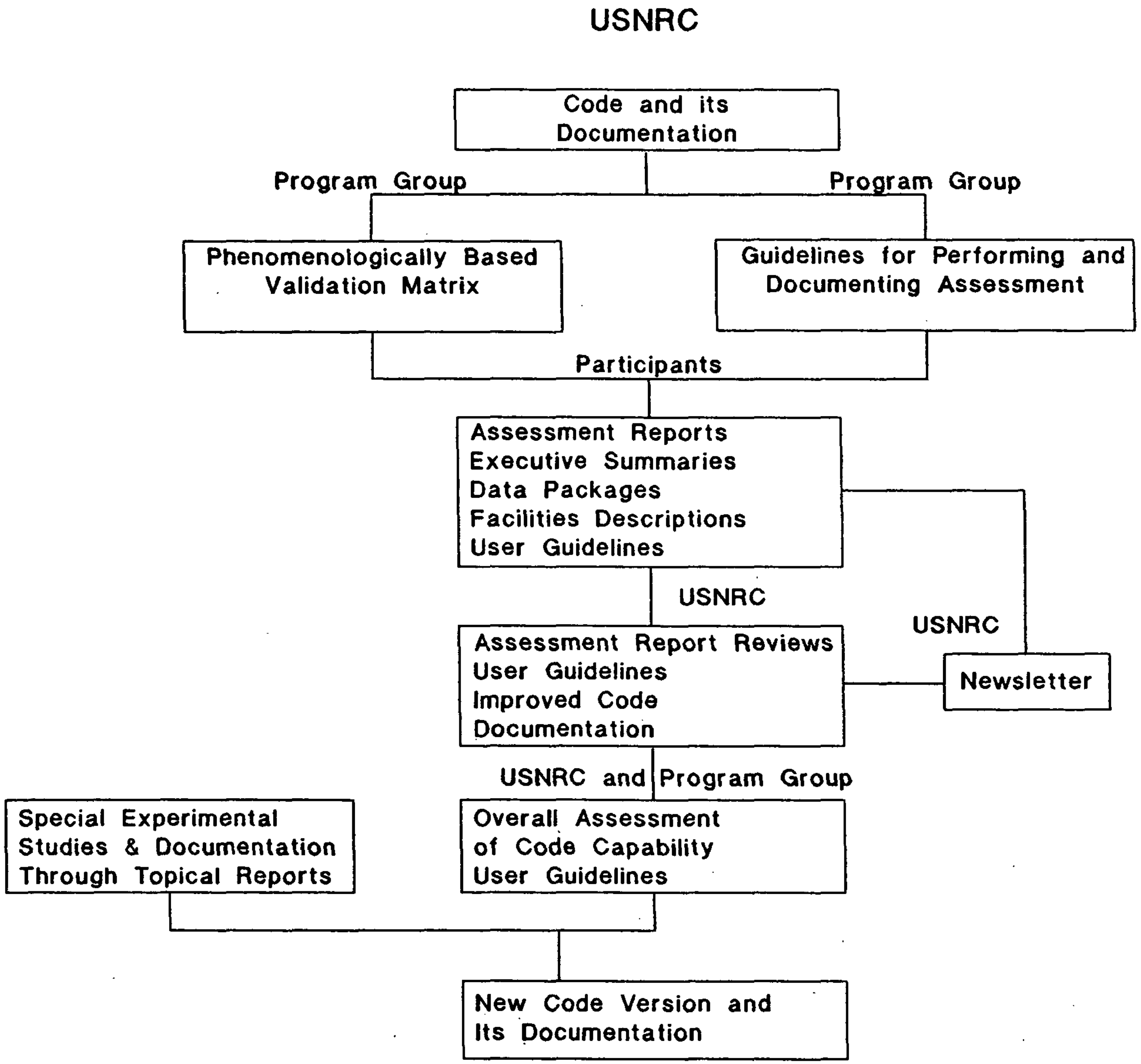

Figure 3. International Code Assessment and Applications Program documents.

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program. Normally, one copy of the code will be provided to each participant, who is then responsible for copying and providing the information to other users within that participant's organization or country that are entitled to receive the code. As part of the assessment effort, participants should indicate improvements needed to the code documentation.

4.3 Code Assessment and Validation Matrices

This is an active plan summarizing the current direction to be followed in the conduct of the program to achieve its goals. First, a matrix (shown in the Appendix) will be maintained which will present the current activities of the international participants. Secondly, a set of matrices will be maintained which focus on the broad needs for code assessment and validation for the thermal-hydraulic codes. These needs will reflect the current status of the codes through regular update to the matrices given in the Appendix (the validation are a refinement of the CSNI matrices for the range of transients in PWR and BWR facilities). The validation matrix will be updated annually.

4.4 Assessment Reports, Executive Summaries and Data Packages

Participants conducting assessment studies will document those studies in assessment reports and associated data packages. Participants should prepare an Executive Summary for each assessment report, which will be openly distributed by the USNRC to all participants. Executive Summaries should not, therefore, contain proprietary, confidential or privileged information.

Facility and plant descriptions sufficient to construct input decks may be provided as part of the data package to assist the USNRC review and evaluation of code assessments and to effect code error corrections and enhancements. The USNRC may request such information on a case-by-case basis where needed to evaluate the code assessments. Provision of the facility and plant description documents is the responsibility of the participant providing the associated assessment report. A single transmittal and/or minor updates may be sufficient for a series of assessment reports based upon the same experimental facility or plant.

The procedure for sending assessment reports, executive summaries and data packages is as follows:

- Assessment reports and executive summaries for RELAP5 and TRAC-BWR: One copy each to the USNRC program manager and the INEL program manager.
- Assessment reports and executive summaries for TRAC-PWR: One copy each to the USNRC program manager, the INEL program manager and the LANL program manager.
- Data packages for RELAP5 and TRAC-BWR: One copy to the INEL program manager.
- Data packages for TRAC-PWR: One copy to the LANL program manager.

Further information on guidelines for performing and documenting code assessments is given in Section 5.

4.5 USNRC Review of Assessment Reports

The USNRC will undertake to review each assessment report and its executive summary. The reviews will be documented in concise evaluation summaries. Prior to publication of an evaluation summary, the participant providing the assessment report will have the opportunity to review and comment on the draft evaluation summary. The final evaluation summary will be distributed to all participants, therefore, they will not contain proprietary, confidential or privileged information.

The assessment reports will be reviewed by the responsible USNRC contractor, INEL in the case of RELAP5 and TRAC-BWR, and LANL in the case of TRAC-PWR. The primary objectives of the review of assessment reports is to ensure that the results of code assessments are fully understood and assimilated by the code developers and that any questions regarding the data used or how the code was applied are resolved. Additionally, the review will serve to integrate the general body of knowledge developed from all assessment results.

4.6 Newsletter Service

For timely information flow to all program participants, a newsletter service will be provided by the USNRC. The newsletters will be issued quarterly and will contain information of general interest to the program. A separate newsletter will be issued for each code to users of that code. Participants are welcome to provide input for incorporation in the newsletter. Contents of the newsletters will include:

- Recently identified code errors and corrections (updates);
- Schedule for error corrections to be made;
- Executive summaries and USNRC evaluation summaries of assessment reports;
- Planned model improvements and schedule for the next code version;
- Recommended modeling techniques and user guidelines;
- Improved input/output and user conveniences;
- Upcoming meeting notices;
- Bibliography of code usage, etc.

It is expected that from time to time participants may encounter errors or problems in running the code. In such cases, the participants should notify the USNRC Program Manager, who will ensure that a satisfactory resolution is provided within the limitations of priorities and available resources. News of such difficulties and their resolution will be provided to all participants through the newsletter service. This includes any code updates that may result.

4.7 Topical Reports

Certain agreements may contain provisions for specific research. In addition, certain program activities may be of sufficient depth and extent to warrant special consideration. Documentation of such efforts, particularly those conducted by Task Groups, will be provided through special topical reports. The development, composition, publication and issuance of topical reports will be considered on a case-by-case basis.

At times appropriate to the accumulating assessment data base (approximately annually), the USNRC together with the Program Group will evaluate the significant findings of the assessments and support to quantification of the code accuracy. Results and conclusions will be documented and published in topical reports. These documents will be available to all participants, therefore, they will not contain proprietary, confidential or privileged information.

4.8 Information Storage and Retrieval

INEL will operate an information storage and retrieval system for RELAP5 and TRAC-BWR. The system will contain: newsletters, executive summaries of assessment reports and USNRC evaluation summaries of assessment reports. It may also contain information on code errors, corrections and code updates. The INEL system will utilize an IBM PC accessible by phone. Details of the system and access may be obtained by contacting the INEL ICAP program manager. Similarly, LANL will operate an information storage and retrieval system for TRAC-PWR utilizing a VAX computer accessible by phone. Details of the system and access may be obtained through the LANL TRAC user contact.

5. GUIDELINES FOR PERFORMANCE AND DOCUMENTATION OF CODE ASSESSMENT STUDIES

Objectives and guidelines for performing code assessments are presented in Sections 5.1 and 5.2. Key assessment parameters are defined and the information required to quantify the code uncertainty is specified in Section 5.3. Documentation requirements are provided in Sections 5.4 and 5.5.

5.1 Code Assessment Objectives

The primary objective of code assessment is to formulate well-supported judgements relative to code performance. This includes: the ability of the code to physically represent important phenomena; the quantification of code accuracy; recommendations for user guidelines, particularly nodalization; providing feedback to the code developer for suggested code improvements; and providing feedback concerning the code documentation. Code assessments should report conclusions, recommendations (when applicable) and information for evaluation of the code accuracy. In addition to clearly supported conclusions derived from a code assessment, the reporting of perceptions of the assessor is encouraged, however, perceptions should be clearly indicated as such.

A code assessment, of necessity, consists of a code calculation for which qualified experimental data exist. Comparison of the calculated and measured response provides insights into the capabilities of the code. A code assessment consists of a base case calculation, and may also include sensitivity studies to further evaluate code performance. All sensitivity studies should be reported to ensure maximum benefit of the assessment results.

5.2 Code Assessment Guidelines

5.2.1 Evaluation of Experimental Data Uncertainty

The assessor must evaluate and report the uncertainty of experimental data used for code assessment. This uncertainty has the following sources:

- The distribution of individual measurements about the mean for a given node of the calculation input model (e.g., thermocouple measurements in the core, where the differences may result from three-dimensional flow effects, spacer grid effects, etc.);
- The effect of measurement location (e.g., measurements of clad temperature may be influenced by a thermocouple fin effect; measurement of hot leg fluid temperature may be influenced by a hot wall or cold wall effect; measurement of fluid density may be influenced by the location of the densitometer; etc.);
- Limitations in measurement capability (e.g., dead bands in turbine flow meters where the instrument cannot measure below a certain flow rate);

- Experimental boundary conditions (e.g. experimental facility heat losses and distributions, heat input from heat tracing, valve leakage, etc);

- Instrument uncertainty and signal processing, as determined by instrument calibration.

Depending on the type of measurement, one or more effects may dominate the determination of experimental data uncertainty. For temperature, the effects of measurement location and the distribution of individual measurements resulting from three-dimensional effects are important. The assessor must provide a mean value for the node being compared with the code calculation, and a 1-sigma value of the standard deviation. Data should not be used if they are suspect or if their uncertainty is indeterminate. Obviously, this includes such examples as quantities beyond the instrument range, broken or uncalibrated instruments, and signals known to be in error.

5.2.2 Base-Case Calculation

In performing the base case calculation, the user must use best-estimate techniques and recommended code options to provide an objective survey of the code. A best estimate should be made with respect to nodalization. Recommended options from the code users manual should be adopted and identified. The base case calculation must be performed with the formally released code version. In all cases the facility model and code version must be fully documented.

5.2.3 Sensitivity Studies

In addition to the base case calculation, sensitivity studies are needed to develop user guidelines. Sensitivity studies may explore the effect of input modeling, boundary conditions or code options (e.g., loss coefficients). Sensitivity studies may also be performed to study the effect of code model modifications. These are intended to explore the reasons the code results were obtained and to suggest possible enhancements for incorporation in a future version of the code. Whereas sensitivity studies are valuable in elucidating the code performance, mere tuning of the code is to be avoided. The results of the sensitivity studies must be reported as additional analyses to further explore code performance and should include purpose, modifications made, and results.

5.2.4 Nodalization Studies

Nodalization studies should be performed. Proper nodalization can be crucial to obtaining good results. Often, a facility model is adopted and an input deck is established and remains fixed for a large number of calculations that may include a range of conditions. Sufficient attention must be given to studying the sensitivity of the calculation to nodalization. When nodalizing experimental facilities (especially basic and separate effects) the noding detail should be similar in general to typical power plant analyses. That is, highly detailed noding is generally irrelevant if comparable noding is not likely to be used for plant analyses. Studies using detailed noding are encouraged, however, to determine the effect of noding sensitivity. The nodalization studies will permit improved user guidelines and will help to define the user uncertainty.

5.2.5 Evaluation of the Code

In evaluating the code, first the assessor should study the experimental data and understand the phenomena that occurred. Second, the code calculation should be studied to verify whether the code predicts the same phenomena.

Discrepancies between the experimental data and code calculations should be explained, areas of agreement noted and conclusions drawn on code capabilities and limitations. Table 3 presents a list of important phenomena for LWR transients and LOCAs. The code should be evaluated with respect to relevant phenomena from the table for each assessment case performed.

The assessor must keep in mind the importance of the variability of experimental data. For example, the average of several thermocouple readings in a volume is applied when comparing the code-calculated temperature for the volume. Only when the uncertainty and variability of the experimental data is considered can the code be properly evaluated and conclusions drawn. Particular attention should be given to the effect of structural heat and heat losses, as well as boundary conditions. Atypicalities of the experimental facility must be remembered. For example, electrical heater rods are generally a poor simulation of nuclear rods with respect to stored energy and thermal conductivity.

The computational efficiency of the code should be evaluated and reported, including run statistics identified in Table 4. Also, identify whether the calculation time step was determined by the code due to Courant or other time step limits or whether the calculation proceeded at a user specified maximum time step and thereby could have proceeded at a faster rate. The time step evaluation may be presented by providing a comparison of the actual time step and the user specified maximum time step, both as a function of transient time.

The user should evaluate the code documentation, including the user manual, user guidelines, documentation of code models, correlations and numerics and the developmental assessment. Deficiencies in the documentation should be noted along with recommendations for improvements.

5.3 Information Required for Quantification of Code Uncertainty

Code assessment activities contribute to an improved code, improved code documentation, user guidelines and an improved user interface. However, in addition, code assessment provides a base of data that can be used to support the quantification of code uncertainty.

The ICAP has the charter to support the various uses of code assessment data as itemized above. A list of key parameters, shown in Table 5, has been defined for use in quantification of code accuracy. The table includes

TABLE 3. CAPABILITY OF THE CODE TO PREDICT PHENOMENA

PWR Phenomena	Large Break	Small Break	Transient
Break Flow and Valve Leak Flow (subcooled and saturated)	X	X	X
Phase Separation in T-Junction and Effect on Break Flow		X	
Liquid Inventory Distribution		X	
Phase Separation	X	X	
Mixing and Condensation During ECC Injection, Including Thermal Stratification	X	X	
ECC Bypass and Penetration	X		
Steam Binding (liquid carry-over, etc.)	X	X	
Core-Wide Void and Flow Distribution	X	X	
Entrainment and Deentrainment in Core	X	X	
Entrainment and Deentrainment in Upper Plenum	X	X	
CCFL at Upper Tie Plate and Pool Formation in Upper Plenum	X	X	
Mixture Level in Core		X	
Mixture Level in Downcomer	X	X	
Core Heat Transfer, Including Partially Covered Core (DNB, dryout, rewet)	X	X	
Quench Front Propagation	X	X	
Single Phase Natural Circulation	X	X	
Two Phase Natural Circulation	X	X	
Natural Circulation Through Vent Valves (B&W)	X	X	
Stratification in Horizontal Pipes		X	
Reflux Condenser Mode and CCFL		X	
Boiler Condenser Mode (B&W)		X	
Noncondensible Gas Effects		X	
Asymmetric Loop Behavior		X	
Loop Seal Clearance		X	
Primary Side Steam Generator Heat Transfer		X	X
Secondary Side Steam Generator Heat Transfer		X	X
Mixture Level and Entrainment in Steam Generator		X	
One and Two Phase Pump Behavior	X	X	X
Pressurizer Thermal-Hydraulics		X	X
Surgeline Hydraulics		X	X
Refill of Loops (B&W)		X	
Thermal-Hydraulic Nuclear Feedback			X
Boron Mixing and Transport			X
Separator Hydraulics		X	X

TABLE 3. (Continued)

BWR Phenomena	LOCA	Transient
Break Flow and Valve Leak Flow (subcooled and saturated)	X	X
Core-Wide Void and Flow Distribution, Including Channel and Bypass Axial Flow and Void Distribution	X	X
Parallel Channel Instabilities	X	X
Phase Separation	X	
Mixing and Condensation During ECC Injection	X	
ECC Bypass and Penetration	X	
CCFL at Upper Tie Plate and Pool Formation in Upper Plenum	X	X
CCFL at Channel Inlet Orifices	X	
Lower Plenum Flashing	X	
Steam Binding (liquid carry-over, etc.)	X	
Core Heat Transfer, Including Partially Covered Core (DNB, dryout, rewet)	X	
Quench Front Propagation on Fuel Rods and Channels	X	
Entrainment and Deentrainment in Core	X	
Deentrainment at Upper Tie Plate and Upper Plenum	X	
One and Two Phase Pump Behavior, Including Jet Pumps	X	X
Single Phase Natural Circulation		X
Two Phase Natural Circulation	X	X
Mixture Level in Core	X	X
Downcomer Mixing and Mixture Level in Downcomer	X	
ECC Mixing and Condensation		X
Spray Distribution and Spray Cooling	X	
Phase Separation in T-Junction and Effect on Break Flow	X	X
Boron Mixing and Transport		X
Steam Line Dynamics		X
Separator Hydraulics	X	X
Steam Dryer Hydraulics	X	
Nuclear Thermal-Hydraulic Feedback and Instabilities		X
Void Collapse During Pressurization		X

TABLE 4. RUN STATISTICS

- Required plots are: CPU vs. RT
  and DT vs. RT

- Required number is: $(\mathrm{CPU} \times 10^3) \times (\mathrm{C} \times \mathrm{DT})$

  Where
  RT = Transient Time
  C = Total number of volumes in the model
  DT = Total number of time steps
  CPU = Total execution time

- Specify type of machine used to perform the calculation

TABLE 5. KEY PARAMETERS FOR QUANTIFYING THE ACCURACY OF THE CODE

Key Parameter	Applicable to	
	PWRs	BWRs
Primary system pressure	X	X
Primary system break and/or valve flow	X	X
ECC injection flow rate (each operative system)	X	X
Vessel or core collapsed liquid level	X	X
Core cladding temperature at 1/4, 1/2, 3/4 and peak power region	X	X
Core inlet mass flow rate	X	X
Nuclear core power (reactivity transients only)	X	X
Pressurizer liquid level	X	
Secondary side pressure	X	
Secondary side break and/or valve flow	X	
Steam generator primary and secondary liquid levels	X	
Hot leg mass flow rate	X	
Cold leg mass flow rate	X	
Steam line and recirculation loop flow rates		X
Single Values		
Blowdown peak cladding temperature	X	X
Reflood peak cladding temperature	X	X
Minimum core liquid level	X	X
Maximum or minimum primary system pressure	X	X
Peak core power (nuclear simulation only)	X	X
Maximum or minimum secondary side pressure	X	
Times of core uncovery	X	X
Times of CHF at the cladding hot spot	X	X
Times of peak cladding temperature (blowdown and reflood, or other)	X	X
Time to final core quench	X	X
Time to rewet and/or quench at the hot spot	X	X
Times of ECC initiation (each operative system)	X	X
Time of MSIV signal generation	X	X
Time of loop seal clearing (i.e. time of first significant vapor flow through seal)	X	X
Time when primary pressure is less than steam generator pressure	X	
Time to pressurizer empty or full	X	
Times of pressure maxima and/or minima (primary and secondary)	X	X

both single and continuous valued parameters, with the single-valued parameters generally representing event timings or maximum or minimum values of the continuous valued parameters. These parameters will characterize the ability of the code to calculate the response observed in an experiment. The following general notes apply to the table.

- Not all parameters will be applicable to each transient.
- Liquid levels are derived parameters generally based on differential pressure measurements. It is preferable that differential pressures be reported rather than liquid levels. If liquid levels are reported, the exact method by which they are derived must be specified.
- Flow rates (break, valve, core, etc.) should be reported in terms of mass. The method by which each parameter was derived should be clearly stated. The flow area used with each parameter must be given.
- All key parameters included in a data transmittal must be clearly related to their geometric location in the code model and the experimental facility.

For the purpose of support to the quantification to code uncertainty, participants should supply suitable data for the applicable key parameters shown in Table 5 for base case and sensitivity studies. Determination of the quantitative differences between the code and measured results requires information transmittal in an appropriate form. Continuous valued plots normally provided in assessment reports are unsatisfactory for the detail necessary to perform uncertainty analyses. Instead, the qualified data from the experiment should be provided for the applicable key parameters either on magnetic tape, floppy disk or other suitable electronic means. The single-valued parameters should be presented in the form of a table.

There are special considerations when redundant and/or multichannel measurements exist for a key parameter in an experiment. An example is the case of several thermocouples that are combined to provide a volume average cladding temperature corresponding to a particular node in the code calculation. In such cases, all experimental measurements should be included in the data transmittals. Provision of this information will support the determination of the experimental data variability contribution to the overall uncertainty. Additionally, the measurement uncertainty should be provided for each key parameter, including derived parameters.

5.4 Code Assessment Documentation

The usefulness of code assessment is strongly affected by the degree to which the documentation of the study is complete and concise. The following provides guidelines for the documentation of code assessments and describes sections that should be included and their contents.

5.4.1 Executive Summary

An executive summary should be supplied which discusses the objectives, scope, methodology used, conclusions and recommendations. The discussion should include: the ability of the code to represent important physical phenomena, the code accuracy; recommended code improvements, proposed enhancements to the code; and recommendations concerning user guidelines and nodalization. The summary should also include run statistics information identified in Table 4. Proprietary, confidential or privileged information should not appear in the summary since it will be distributed to all program participants.

5.4.2 Abstract

The abstract should contain a brief description of the contents of the document and the sponsoring and performing agencies. The code, experimental facility and tests should be identified. Results, conclusions and recommendations should not be included in the abstract.

5.4.3 Table of Contents

The table of contents should identify sections of the report and include lists of tables and figures.

5.4.4 Introduction

The introduction should include a detailed discussion of the assessment study background, scope and objectives, and should present the assessment methodology used for the study. A brief description of sections to follow in the document may be helpful to the reader.

5.4.5 Facility and Test Description

A brief discussion should be provided of the experimental facility or plant including its geometric layout, instrumentation, operational procedures and other information, as required for the reader's understanding. Reference may be made to detailed facility or plant and test description documents to be part of a data package that may accompany the assessment documentation (see Section 5.5). A single facility and test description document may cover a series of experiments. Transmittal of such documents need only occur once when the first assessment study in the series is transmitted; however facility/test features unique to each assessment study should be described in the assessment document. The information necessary to recreate the input deck may be included as part of a data package to accompany the assessment document (see Section 5.5).

The experiment to be calculated should be discussed including important thermal-hydraulic information, initial and boundary conditions, and operational information pertinent to the calculation. Measurement uncertainty must also be discussed.

5.4.6 Code Input Model Description

The code input model should be discussed in detail including nodalization, assumptions, boundary and initial conditions and operational conditions for the calculation. The nodalization diagram should be presented along with a discussion of the nodalization rationale, including assumptions and modeling compromises. Modifications to the input model or code for sensitivity studies should be discussed.

5.4.7 Results

Results of the base case calculation that lead to major conclusions should be clearly presented and discussed. Applicable assessment parameters from Table 3 should be discussed. The rationale for performing any sensitivity studies should be discussed along with the methodology used to perform them. Modifications to base case conditions and the resulting effect should be fully described and qualified.

The discussion of base case and sensitivity results should include:

- A comparison between the code prediction and the experiment with regard to the important physical phenomena that occurred during the experiment;
- The accuracy of the code with an explanation of the reasons for differences between the data and calculation. The accuracy discussion should include key parameters from Table 5;
- Conclusions from sensitivity studies;
- Suggestions for improved models and correlations or for additional models of features that should be added to the code;
- User guidelines for performing similar analyses; and
- Suggestions for improving input/output features and users conveniences.

Assessment parameters should be overlayed with appropriate experimental results. A discussion should be included if the code fails to provide user flexibility to model certain phenomena or simply fails to predict phenomena. The applicability of the results to full scale power plants should be discussed. Note whether the observed phenomena are likely to occur in a full scale plant.

5.4.8 Run Statistics

A discussion of run statistics should include those listed in Table 4 and a discussion of the time step behavior as described in Section 5.2.5.

5.4.9 Conclusions

A summary of major conclusions and recommendations derived from the code assessment study should be presented. Supportive discussions in the body of the document may be referred to. Conclusions and recommendations not supported by discussions in the body of the document are inappropriate.

5.4.10 References

All drawings, test conditions, and other information required to construct the input deck and perform the calculation and analysis should be referenced.

5.4.11 Appendices

Appendices should be included to supply support information necessary to ensure document completeness, including proprietary, confidential or privileged information, which will be protected from disclosure by the USNRC and its contractors.

5.5 Data Package

Participants should provide the USNRC with a data package to accompany the code assessment document if such information is normally prepared and available. The USNRC may specifically request data packages on a case-by-case basis if additional information is required to evaluate the code assessment work. Data packages should include: the code input listing; all calculated data from the study; all experimental data for the experiment; and a detailed description of the facility in which the experiment was performed. The data package will be used only by the USNRC and its contractors to support the evaluation of submitted assessment studies and shall not be distributed by the USNRC or its contractors to third parties. Proprietary, confidential or privileged information included in the package should be clearly identified as such.

5.5.1 Code Input Model

An input model listing should be provided in both hardcopy and on data tape. Reference should be made to facility information required to perform the calculations. Section 5.5.4 presents guidelines for the facility description document.

5.5.2 Calculated Data

A digital magnetic tape of all calculated data should be included. The tape must be in either EBCDIC or ASCII codes in order to allow the tape to be processed. Information adequate for naming the data channels and their units of calibration (pressure, temperature etc.) must be either on the data tape or must be provided with the tape. Table 6 lists minimum requirements for tape description. It should be noted that formatting routines to process data tapes are expensive to develop. Thus, each participant should standardize their tape format to be consistent within each tape (i.e. data vs. code) and between subsequent tapes.

TABLE 6. MINIMUM REQUIREMENTS FOR TAPE DESCRIPTION

I. Machine

   1. Name or type of computer that generated the tape;
   2. Word size in bits;

II. Tape Format

   1. Number of tracks;
   2. Packing density;
   3. Record size;
   4. Blocking factor;
   5. Description of tape code used (EBCDIC or ASCII). Binary coded tapes are not acceptable;
   6. Specify any record or block control words which are not part of the data. A word-by-word description of each record on the tape is very helpful.

III. Data Format

   1. Description of logical or physical data arrangement on the tape, e.g., multiplexed, demultiplexed, etc.;
   2. Definition and location of timing information;
   3. Outline any unique or special features.

IV. Other

Scan sheets with measurement identification and engineering units.

5.5.3 Experimental Data

The experimental data package should contain all relevant experimental data and their associated uncertainties available for the specified test. The data should be accompanied by a facility description diagram that clearly shows the location of each measurement device. Each type of measurement should be discussed relative to the measurement process and data uncertainty. When data are derived from measured data (i.e. void fraction, mass flow, etc.) the equations used should be given along with the composite uncertainty of the derived parameter. Time-history data should be supplied on digital magnetic tape (see Table 6) as well as hardcopy.

A table of initial and boundary conditions should be supplied for the experiment along with a sequence of events which specifies event timings during the transients.

5.5.4 Facility or Plant Description Document

The facility or plant description should include information relative to sponsoring agencies and type of facility or plant. A general description of the components that make up the facility or plant should be provided. Associated control systems should be briefly described. Scaling information should be provided for experimental facilities.

All sketches, drawings, operational procedures, experiment conditions, setpoints, material specifications, geometric information and other information pertinent to the facility or plant description should be either included such that the input model could be reproduced or should be referenced. If assessment studies are to be performed for a series of experiments in a facility or plant, a detailed description must be supplied only with the first study documentation. Thereafter, only modifications pertinent to subsequent experiments need be documented.

A system schematic drawing should be provided to clearly show how the various components form the overall system. The facility or plant should be described component-by-component, providing all necessary information to convey the component's function and operation as well as its geometry (areas, volumes, etc.). Piping that connects components must be addressed to the same extent as the components themselves. For complex components, a tabulation of volume as a function of elevation should be included if available.

Component operational data should include delay times, rates of change (valve movement), performance curves (pumps) and all other control and performance information necessary to perform a simulation of the experiment.

Control systems associated with a component or group of components should be described to the level of detail necessary to convey their function and operation. Sufficient control system data should be included to allow duplication of the modeled control systems. Trip points and setpoints should be clearly tabulated for control systems functions (ECCS actuation, scram, etc.).

Hydraulic and geometric information necessary to determine loss coefficients and heat transfer coefficients should be included in the data package and referenced. Insulation of components and piping should be clearly identified and, where heaters are used to insulate a component (guard heaters), their control procedure for the experiment should be provided. If available, regionally quantified heat loss information should be provided. Insulation material properties and dimensions must be specified. Heat loss due to instrument cooling or uninsulated regions should be identified and quantified if possible. System coolant leakage estimates should be evaluated and included in the facility description package.

6. QUANTIFICATION OF CODE ACCURACY

6.1 Overview of Code Accuracy Quantification

This section is intended to provide a general overview of quantitative code assessment and its utilization, with details of the methodology to be presented in other documentation. The ICAP has the charter to provide support to the effort to quantify code uncertainty. The primary support effort entails the compilation of code assessment data suitable for code uncertainty quantification.

The code must be sufficiently accurate for its intended use for LWR safety related studies. This requires that the code be able to predict relevant phenomena occurring for postulated transients and breaks. The dominant phenomena will vary depending on the event, however, the code should posses the flexibility to be used with confidence over its intended range of application.

Code assessment is performed to obtain a considered view of the accuracy and validity of the code over its range of applicability. A sound understanding is required of the code's ability to physically represent the phenomena which may be expected to occur during transients and breaks. To achieve this, the code must be exercised against a matrix of experiments. The following considerations must be applied in formulating such a matrix.

- The code should be exercised in its ability to model all relevant phenomena; this leads to a phenomenologically based matrix;
- The code should be exercised for different plant configurations of concern including, for example, cold leg injection, combined injection, once-through steam generators, etc;
- Experiments selected for assessment should cover a range of scales as a test of the ability of the code to correctly model full scale; and
- Experiments other than those used in the code development process should generally be chosen for subsequent assessment studies.

The product of application of the code to experiments is a body of information containing comparisons between code predictions and test data. Thus far, conclusions on code accuracy based on assessment studies have been qualitative (e.g., the code predicted break flow reasonably well) and have generally been ambiguous and difficult to utilize by those engaged in plant safety related studies.

A goal of the present assessment effort is for the USNRC, together with the Program Group, to support formulation of more coherent, quantitative conclusions on the capability of the code by evaluating the accuracy of the code with respect to phenomena. Quantitative conclusions can be constructed only upon the basis of a thorough, comprehensive physical understanding of the code's ability to represent relevant phenomena. Table 5 presents a list of key parameters. The safety barrier concept for fission product release was considered in deriving this list, with the first barrier being the fuel cladding and the second barrier the reactor coolant system pressure boundary. The final barrier is the containment. The ICAP program is focused of the integrity of fuel elements and the reactor coolant system pressure boundary, hence, their condition influenced the choice of key parameters.

As an example, during a LBLOCA, the reactor coolant system pressure boundary does not perform its intended function. In events involving core uncovery, the principal concern is clad temperature. If the core remains covered as may occur in a small break LOCA, however, the clad temperature will follow the saturation temperature. In this case, such concerns as coolant inventory and distribution and steam generator heat transfer influence the choice of key parameters. Similarly, key parameters can be identified for transients.

6.2 Application of Quantitative Assessment Results

From an applications standpoint, best-estimate thermal-hydraulic codes are used for safety related studies involving the response of power plants to transients and breaks. The codes may also be used when planning experiments in scaled facilities. Code assessment is performed to: (a) provide quantitative information on the code accuracy for applications to power plant studies; and (b) provide quantitative information on the code accuracy in order to know what are the code development needs. In each case, it is necessary to quantify the code accuracy.

Along with determining the code accuracy, acceptance criteria must be defined to resolve code improvement priorities. If the code meets the acceptance criteria, then code development can be considered as being completed and further experimental programs are unnecessary. If the code does not meet the acceptance criteria, then one or more models in the code must be improved, within the constraints of available resources. This also implies that new experimental data may be needed.

It is generally believed that significant progress has been made through past code development and the code has attained a measure of reliability. It is probable, however, that the assessment and applications process will reveal areas where the code does not meet the acceptance criteria. This being so, it will be necessary to prioritize possible areas for improvement within the constraints of available resources. This should be approached by addressing the following two questions for each deficiency identified:

1. How important is the deficiency with respect to the understanding of plant behavior?, and

2. How many resources are estimated to be required to reduce the code uncertainty to meet the acceptance criteria?

6.3 Acceptance Criteria for the Code

Simple acceptance criteria, based on a few key parameters for each class of plant and transient, should be chosen to avoid the exceedingly complex situation of establishing a unique criterion for each of many possible parameters. The proposed code acceptance criteria should include:

- Correlations and models used in the code should not be in conflict with any statements or conclusions contained in documentation used to support the model acceptability. If any conflicts exist, they must be reviewed and deemed acceptable.

- Correlations and models used in the code should not conflict with other applicable data, correlations or models. If any conflicts exist, they must be reviewed and deemed acceptable.

The USNRC intends to use the results of the ICAP program to support the effort to quantify uncertainty, and support further definition of the acceptance criteria.

In determining whether the code meets the acceptance criteria for the selected key parameters, care must be taken to understand the phenomena studied in the experiment and the code calculation. For example, the code may incorrectly predict the minimum core inventory during a small break LOCA. This may be the result of incorrect break flow, incorrect ECC flow, incorrect steam generator heat transfer or incorrect mass distribution in the reactor coolant system. Thus the evaluation of the code should indicate which of the models in the code were responsible for the behavior observed.

APPENDIX

APPENDIX

TEST MATRICES FOR THE ICAP

This appendix contains tables that summarize two matrices that are integral to the ICAP. Table A-1 presents a phenomenologically based matrix for pressurized water reactor and boiling water reactor plant configurations and for large break loss-of-coolant accidents, small break loss-of-coolant accidents, and for operational transients. This table presents key phenomena important to the calculation of system response during the respective scenerios.

Table A-2 presents the assessment plans that each of the ICAP participants have identified for fulfillment of their respective bilateral agreements. The table contains the assessment plans relative to the RELAP5, TRAC-BWR and TRAC-PWR codes.

Table A-1.1. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

- Phenomena versus test type
  - ● Simulated
  - o Partially simulated
- Test facility versus phenomena
  - ● Suitable for code assessment
  - o Limited suitability
- Test type versus test facility
  - ● Already performed or planned until 12/84
  - o Performed or planned until 12/84, but of limited use

		Test Type			Test Facility																				
					System Tests							Sep. Effects Tests													
		Blowdown	Reflood	Refill	CCTF 1:25*	LOFT 1:50	BETHSY 1:100	PKL 1:134	SPES 1:430	LOBI 1:712	SEMISCALE 1:1600	UPTF 1:1	CREARE 1:25	SCTF 1:25	MARVIKEN	BCL	SUPER MOBY DICK	DADINE	EPIS	ERSEC	OMEGA	ACHILLES	REFLEX	THETIS	Harwell Post-CHF H.T.
Phenomena	Break flow	●	●	●	-	o		o		o	o		-	-	●	-	●								
	Phase separation	o	●	●	●	●		●		●	●		-	-	o	-	-								
	Mixing and condensation during injection	o	●	●	o	o		o		o	o		●	●	-	●	-		o						
	Core wide void&flow distribution	o	●	●	o	o		o		-	-		-	●	-	-	-								
	ECC bypass and penetration	o	o	●	●	●		o		-	-		●	o	-	●	-								
	Upper plenum hydraulic (CCFL, pool information)	o	●	●	o	o		o		o	-		-	●	-	-	-								
	Steam binding (liquid carryover, etc.)	-	●	o	o	o		●		-	o		-	-	-	-	-								
	Core heat transfer incl. DNB, dryout, RNB	●	●	●	o	●		o		o	o		-	o	-	-	-	o		o	o	o	o	o	o
	Quench front propagation	o	●	o	o	●		o		-	o		-	o	-	-	-	o		o		o	o	o	
	Entrainment (Core, UP)	o	●	o	o	o		o		o	o		-	●	-	-	-								
	Deentrainment (Core, UP)	o	●	o	o	o		o		-	o		-	●	-	-	-								
	1- and 2-phase pump behavior	●	o	o	-	o		o		●	●		-	-	-	-	-								
Test Facility System Tests	CCTF	-	●	o																					
	LOFT	●	●	●																					
	BETHSY																								
	PKL	o	●	●																					
	SPES																								
	LOBI	●		●																					
	SEMISCALE	●	●	●																					

- Leak location/leak size
- Pumps off/pumps on
- Cold leg injection/combined injection

(APPLIES TO ALL TABLES)

* volumetric scaling

Table A-1.2. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE LEAKS IN PWRs ****

- Phenomenon versus test type
  - ● Simulated
  - o Partially simulated
- Test facility versus phenomena
  - ● Suitable for code assessment
  - o Limited suitability
  - x Expected to be suitable
- Test type versus test facility
  - ● already performed or planned unitl 12/84
  - o Performed or planned until 12/84, but of limited use

		Test Type							Test Facility																	
									System Tests								Separate effects Test									
		Stationary test addressing energy transp. on prim side	Stationary test addressing energy transp. on sec. side	Small leak overfeed by HPIS secondary side necessary	Small leak w/o HPIS over-feeding secondary side nec.	Intermediate leak, sec. side not necessary	Pressurizer leak	U-tube rupture	PWR 1:1 *	LOFT 1:50	LSTF 1:50	BETHSY 1:100	PKL-I 1:134	SPES 1:430	LOBI-II 1:712	SEMISCALE 1:1600	UPTF 1:1	THL 1:15	GEST GEN 1:50	Patricia GV-1	Patricia GV-2/GEN 3x3	G-2	Pressurizer Test (CISE)	NEPTUNUS 1:40	CANON	Harwell Post-CHF heat transfer
Phenomena	Natural circulation in 1-phase flow, primary side	●	●	●	o	-	●	●	●	●			●		●	●		-			-	-	-			
	Natural circulation in 2-phase flow, primary side	●	-	o	●	●	o	-	-	●			●		●	●		-			-	-	-			
	Reflux condenser mode and CCFL	●	-	-	●	●	-	-	-	o			o		o	o		-		●	-	-	-			
	Asymmetric loop behavior	-	-	●	●	-	o	●	-	-	x	x	-	x	o	o		-			-	-	-			
	Leak flow	-	-	●	●	●	●	●	-	●			o		●	●		●			-	-	-		o	
	Phase separation without mixture level formation	●	-	o	●	●	●	o	-	o			o		●	o		o			-	-	-			
	Mixture level and entrainment in vertic. comp. SG **	-	●	●	●	●	●	●	-	-			o		o	-		-	x		●	●	-			
	Mixture level and entrainment in the core	●	-	-	●	●	●	-	-	o			o		o	o		-			-	●	-			
	Stratification in horizontal pipes	●	-	-	●	●	-	-	-	●	x	x	o		o	o		●			-	-	-			
	ECC-mixing and condensation	-	-	o	●	●	●	●	-	o			o		o	o		-			-	-	-			
	Loop seal clearance	-	-	-	●	●	o	-	-	●			o		●	●		-			-	-	-			
	Pool information in UP/CCFL (UCSP)	●	-	-	o	●	●	-	-	o	x		o		-	o	x	-			-	-	-			
	Core wide void and flow distribution	●	-	-	o	●	●	-	-	o	x		-		-	-		-			-	-	-			
	Heat transfer in covered core	●	●	●	●	●	●	●	o	●			●		●	●		-			-	-	-			
	Heat transfer in partially uncovered core	●	-	-	o	●	-	-	-	●			o		o	o		-			-	●	-			o
	Heat transfer SG primary side	●	o	o	●	●	o	o	-	o			o		●	o		-	x	●	-	●	-			
	Heat transfer in SG secondary side	o	●	●	●	●	●	●	-	o			o		●	o		-	x		●	-	-			
	Pressurizer thermohydraulics	o	-	o	o	●	●	●	o	o			-		o	-		-			-	-	●	o		
	Surgeline hydraulics	o	-	-	o	●	●	o	-	o			-		o	o		-			-	-	●			
	1- and 2- phase pump behavior	-	-	-	o	●	-	-	o	o			-		●	●		-			-	-	-			
	Structural heat and heat losses ***	●	-	o	●	●	o	o	-	o			o		o	o		-			-	-	-			
	Noncondensible gas effects	●	-	-	-	-	-	-	-	-			o		●	o		-			-	-	-			
	Phase separ. in T-junct. and effect on leakflow	-	-	-	●	●	-	-	-	o			-		o	-		●			-	-	-			
	Upper head mixing			o	●		●	●	-	o			o		o	o					-	-	-			
Test Facility System Tests	PWR	-	-	o	-	-	●	●																		
	LOFT	-	-	●	●	●	●	-																		
	LSTF																									
	BETHSY																									
	PKL-I	●	●	-	●	●	●	-																		
	SPES																									
	LOBI-II	●	●	●	●	●	●	●																		
	SEMISCALE	o	o	●	●	●	●	●																		

* volumetric scaling

** secondary side

*** problem for scaled test facilities

**** for intermediate leaks phenomena included in large break reference matrix may be also important

Table A-1.3. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR TRANSIENTS IN PWRs

- *Phenomena versus test type*
  - ● Simulated
  - o Partially simulated
- *Test facility versus phenomena*
  - ● *Suitable for code assessment*
  - o *Limited suitability*
- *Test type versus test facility*
  - ● *Already performed or planned until 12/84*
  - o *Performed or planned until 12/84, but limited use*

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

* Volumetric scaling

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

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

**** Problem for scaled test facilities

Table A-1.4. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR LARGE BREAKS IN BWRs
- Test facility versus phenomena
  - ● Suitable for code assessment
  - o Limited suitability
- Test type versus phenomena
  - ● Phenomena occurs
  - o N/A
- Test type versus facility
  - ● Facility applies
  - o Facility does not apply

		Test Type				Test Facility													
						System Tests								Separate Effects Tests					
		Small break	Blowdown	Refill	Reflood	TLTA 1:624	Power plants	FIST 1:624	TBL 2:764	ROSA III 1:424	FIX II 1:777	SSTF 1:1	Marviken 1:1	PIPER-1	REWET II	NEPTUN	FRIGG	GOETA 1:676	Harwell Pos CHF
Phenomena	Neutronics		●	o	o		●												
	Metal/water reaction																		
	Interfacial shear	●	●	●	●			●	●	●		●					●	●	
	Interfacial heat transfer	●	o	●	●			●	●	●						●	o	o	●
	Boron mixing and transport																		
	BOP and control systems		o	o	o		●												
	Post CHF heat transfer		●	●	●			●	o	●				●	●	●			●
	Fast numerics	●	●	●	●			●		●	●		●					●	
	Level tracking	●	●	●	●		o	●			●			o		●			
	Jet pump ECC injection	●	o	●	●			●		●	●								
	Containment	●	●	●	●		o						●						
	Core spray	●	o	●	●					●		●		o				●	
	Break flow	●	●	●	●					●	●		●	o					
	Separator	o	o	o	●			o			o								
	Reflood heat transfer	o	o	o	●			●		●				o	o		●	●	
	Recirculation pump behavior	●	●	o	o		o			o	o								
System Facilities	TLTA	●	●	●	●														
	Power plants																		
	FIST	●	●	●	●														
	TBL	●	●	●	●														
	ROSA III	●	●	●	●														
	FIX II	●	●	●	●														
	SSTF	o	o	●	●														

Table A-1.5. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR SMALL BREAKS IN BWRs
- Test type versus phenomena
  - ● Phenomena applies
  - o N/A
- Test type versus Facilities
  - ● Facility applies
  - o Facility does not apply
- Test facility versus phenomena
  - ● Suitable for code assessment
  - o Limited suitability

		Test Facilities															
		Test Type		**System Tests**							**Separate Effects Tests**						
		Recirculation loop break	Steam break	TBL	TLTA	Power plants	FIST	ROSA III	FIX II	SSTF	MARVIKEN	PIPER-1	REWET II	NEPTUN	FRIGG	GORTA	Harwell post CHF
Phenomena	Neutronics	●	●			o											
	Interfacial shear	●	●	●	o		●	●	o	●					o	●	
	Fast numerics	●	●	●			●	●									
	Level tracking	●	●				●	●	●		●						
	Jet pump ECC injection	●	●		o		●	●	●								
	Containment	●	●				●	●	●								
	Core spray	●	●			o					o						
	Break flow	●	●				o	●	●							●	
	Separator	o	●	o			●	●	●		●						
	Recirculation pump behavior	●	o														
System Facilities	TLTA	●	●														
	Power plants	o	o														
	FIST	●	●														
	TBL	●															
	ROSA III	●															
	FIX II	●															
	SSTF	o	o														

Table A-1.6. Phenomenologically based code assessment matrices.

CROSS REFERENCE MATRIX FOR NON-BREAK TRANSIENTS IN BWRs

- Test type versus phenomena
  - ● Phenomena applies
  - o N/A
- Test type versus facilities
  - ● Facility applies
  - o Facility does not apply
- Test facility versus phenomena
  - ● Suitable for code assessment
  - o Limited suitability

		System Facilities												
		System Tests							Separate Effects Tests					
		Loss of feedwater	ATWS	High Press. boil/off	Low press. boil/off	Reactivity transients	Pump trips	Turbine trips	TLTA	Power Plants	FIST	FIX-II	ROSA-III	TBL
Phenomena	Neutronics		●	●	●	●	●	●		o				
	Metal/water reaction		●	●	●									
	Interfacial shear		●	●	●	●			●		●	●	o	
	Interfacial heat transfer	●		●	●				●		●	o	o	o
	Boron mixing and transport		●											
	BOP and control systems		●	●	●	●		o		o				
	Fast numerics										●	o	●	o
	Level tracking	●	●	●	●	●		●		o	●	o	●	o
	Containment		●	●	●			●		o				
	Recirculation pump behavior		●			●	●			o		o		
System Facilities	TLTA			●	●									
	Power plants	●	o			●	●	●						
	FIST		●	●	●									
	FIX II					o								
	ROSA III			●	●									
	TBL			o	o									

Table A-2. ICAP code assessment matrices.

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***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
AUSTRIA		LOFT	INTEGRAL	RELAP5	L9-3	LOFW	LOSS-OF-FEEDWATER WITH PRIMARY AND SECONDARY FEED/BLEED.
AUSTRIA		LOFT	INTEGRAL	RELAP5	SB-3	SBLOCA	SMALL BREAK LOCA WITH NO HPI, DELAYED FEEDWATER
AUSTRIA		FIX-II	INTEGRAL	RELAP5			
AUSTRIA		FIST	INTEGRAL	RELAP5			
AUSTRIA				RELAP5			TO BE DETERMINED
AUSTRIA				RELAP5			TO BE DETERMINED
AUSTRIA				RELAP5			TO BE DETERMINED
BELGIUM		DOEL	POWER PLANT	RELAP5			POWER OPERATED RELIEF VALVE STARTUP TEST
BELGIUM		DOEL	POWER PLANT	RELAP5			PRESSURIZER HEATER STARTUP TEST
BELGIUM		DOEL4	POWER PLANT	RELAP5	SU-PR-101		PRESSURIZER SPRAY STARTUP TEST
BELGIUM		DOEL	POWER PLANT	RELAP5			NATURAL CIRCULATION
BELGIUM		DOEL	POWER PLANT	RELAP5			PUMP COASTDOWN
BELGIUM		LOBI	INTEGRAL	RELAP5	BT-01	SBLOCA	10% STEAMLINE BREAK
BELGIUM				RELAP5			TO BE DETERMINED
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4TTS		TURBINE TRIP AND SCRAM BY HIGH SG LEVEL
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4LOR		LOAD REJECTION TEST AT 100% POWER
BELGIUM		DOEL1	POWER PLANT	RELAP5	D1LOR		LOAD REJECTION TEST AT 75% POWER
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4LRS		LOAD REJECTION TEST AND SCRAM
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4BL30		BLACKOUT AT 30% POWER
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4TPA		TURBINE RUNBACK

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
BELGIUM		DOEL4	POWER PLANT	RELAP5	D4POP2		POWER RAMP TEST
BELGUIN		DOEL2	POWER PLANT	RELAP5	SGTR		STEAM GENERATOR TUBE RUPTURE
F.R. GERMANY	PROF. E. F. HICKEN	FKL	INTEGRAL	RELAP5	ID6	SBLOCA	SMALL COLD LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	RELAP5	IIB4.1	LBLOCA	200% HOT LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	RELAP5	IIB2	LBLOCA	200% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	RELAP5			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	RELAP5			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	RELAP5			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	RELAP5	PWR 1	SCRAM	SCRAM
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	RELAP5	PWR 2		TURBINE TRIP, STEAM GENERATOR WITH ECONOMIZER
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	RELAP5	PWR 3		TURBINE TRIP
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
FINLAND	MR. H. HOLMSTROM	REWET-II	SEPARATE EFFECTS	RELAP5			REFLOOD
FINLAND	MR. H. HOLMSTROM	REWET-II	SEPARATE EFFECTS	RELAP5			NATURAL CIRCULATION
FINLAND	MR. H. HOLMSTROM	LOVISSA	POWER PLANT	RELAP5			STUCK OPEN TURBINE BYPASS VALVE
FINLAND	MR. H. HOLMSTROM	LOVISSA	POWER PLANT	RELAP5			STARTUP TESTS
FINLAND	MR. H. HOLMSTROM	LOBI	INTEGRAL	RELAP5	A2-B1	SBLOCA	SMALL BREAK LOCA
FINLAND	MR. H. HOLMSTROM	LOFT	INTEGRAL	RELAP5	SB-1	SBLOCA	SMALL HOT LEG BREAK LOCA, PUMPS OFF
FINLAND	MR. H. HOLMSTROM	LOFT	INTEGRAL	RELAP5	SB-2	SBLOCA	SMALL HOT LEG BREAK LOCA, PUMPS ON

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
FINLAND	MR. H. HOLMSTROM	LOFT	INTEGRAL	RELAP5	SB-3	SBLOCA	SMALL COLD LEG BREAK LOCA WITH NO HPI, DELAYED FEEDWATER
FINLAND	MR. H. HOLMSTROM	LOFT	INTEGRAL	RELAP5	LB-1	LBLOCA	200% COLD LEG BREAK, FLYWHEEL DISCONNECT, REDUCED ACC. FLOW
FINLAND	MR. H. HOLMSTROM	LOFT	INTEGRAL	RELAP5	L2-6	LBLOCA	200% COLD LEG BREAK, U.S. LICENSING BOUNDARY CONDITIONS
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	SPES	INTEGRAL	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	GEST-GEN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	GEST-GEN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	GEST-GEN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	GEST-GEN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	GEST-GEN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	FW-1		LOSS OF FEEDWATER
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	SB-1		SMALL HOT LEG BREAK, PUMPS OFF
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	SB-2		SMALL HOT LEG BREAK, PUMPS ON
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	SB-3		SMALL BREAK WITH NO HPI, DELAYED FEEDWATER

***** T E S T   M A T R I X *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	LB-1		200% COLD LEG BREAK, FLYWHEEL DISCONNECT, REDUCED ACC. FLOW
ITALY	MR. G. SAPONARO	LOFT	INTEGRAL	RELAP5	L2-6	SBLOCA	200% COLD LEG BREAK, U.S. LICENSING BOUNDARY CONDITIONS
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
ITALY	MR. G. SAPONARO			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			PRESSURIZER PRESSURE CONTROL
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			PRESSURIZER LEVEL CONTROL
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			STEAM DUMP
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			LOSS OF FEEDWATER AT FULL POWER
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			LOSS OF FEEDWATER AT LOW POWER
KOREA	DR. SANG HOON LEE	PWR	POWER PLANT	RELAP5			STATION BLACKOUT
KOREA	DR. SANG HOON LEE	NORTHWESTERN U.	SEPARATE EFFECTS	RELAP5			CONDENSATION HEAT TRANSFER
KOREA	DR. SANG HOON LEE	BCL	SEPARATE EFFECTS	RELAP5			CONDENSATION HEAT TRANSFER
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
KOREA	DR. SANG HOON LEE			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN	BORSELE	POWER PLANT	RELAP5			UNDERCOOLING TRANSIENT
NETHERLANDS	DR. J. SPEELMAN	BORSELE	POWER PLANT	RELAP5			NATURAL CIRCULATION
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
NETHERLANDS	DR. J. SPEELMAN			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS	LOFT	INTEGRAL	RELAP5	SB-3		SMALL BREAK WITH NO HPI, DELAYED FEEDWATER
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5	LP-FP-2		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5	LP-SB-1		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5	LP-SB-2		TO BE DETERMINED

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
SPAIN	DR. JOSE DECARLOS			RELAP5	DOEL(SGTR)		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			RELAP5			TO BE DETERMINED
SWEDEN	DR. ODDBJORN SANDERVAG	MARVIKEN	SEPARATE EFFECTS	RELAP5	22		SUBCOOLED CRITICAL FLOW
SWEDEN	DR. ODDBJORN SANDERVAG	MARVIKEN	SEPARATE EFFECTS	RELAP5	11		HIGH QUALITY CRITICAL FLOW
SWEDEN	DR. ODDBORN SANDERVAG	FIX-II	INTEGRAL	RELAP5			10% RECIRCULATION LINE BREAK
SWEDEN	DR. ODDBJORN SANDERVAG	FIX-II	INTEGRAL	RELAP5			31% RECIRCULATION LINE BREAK
SWEDEN	DR. ODDBJORN SANDERVAG	FIX-II	INTEGRAL	RELAP5			200% RECIRCULATION LINE BREAK
SWEDEN	DR. ODDBJORN SANDERVAG	LOFT	INTEGRAL	RELAP5	L3-5		4" COLD LEG BREAK, PUMPS OFF
SWEDEN	DR. ODDBJORN SANDERVAG	LOFT	INTEGRAL	RELAP5	L3-6		4" COLD LEG BREAK, PUMPS ON
SWEDEN	DR. ODDBJORN SANDERVAG	FRIGG	SEPARATE EFFECTS	RELAP5			SUBCOOLED VOID DISTRIBUTION
SWEDEN	DR. ODDBJORN SANDERVAG	FRIGG	SEPARATE EFFECTS	RELAP5			CRITICAL HEAT FLUX
SWEDEN	DR. ODDBJORN SANDERVAG	INST. OF TECH.	SEPARATE EFFECTS	RELAP5			POST DRYOUT HEAT TRANSFER
SWITZERLAND	DR. O. MERCIER	NEPTUN	SEPARATE EFFECTS	RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			RELAP5			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER	LOFT	INTEGRAL	RELAP5	SB-3		COLD LEG BREAK SBLOCA, NO HPI, DELAYED FEEDWATER
SWITZERLAND	DR. O. MERCIER	LOFT	INTEGRAL	RELAP5	LB-1		200% LOCA, FLYWHEEL DISCONNECT REDUCED ACC.
SWITZERLAND	DR. O. MERCIER	LOFT	INTEGRAL	RELAP5	L2-6		200% LOCA, LICENSING BOUNDARY CONDITIONS
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	LOFT	POWER PLANT	RELAP5	L3-6		SMALL BREAK LOCA
TAIWAN	DR. YI-GIN CHEN	LOFT	POWER PLANT	RELAP5	L9-3		LOSS-OF-FEEDWATER ATWS
TAIWAN	DR. YI-GIN CHEN	LOFT	POWER PLANT	RELAP5	L2-2		LARGE BREAK LOCA
TAIWAN	DR. YI-GIN CHEN	MAANSHAN	POWER PLANT	RELAP5			FULL LOAD REJECTION
TAIWAN	DR. YI-GIN CHEN	MAANSHAN	POWER PLANT	RELAP5			LOSS-OF-OFFSITE POWER
TAIWAN	DR. YI-GIN CHEN	MAANSHAN	POWER PLANT	RELAP5			TURBINE TRIP AT FULL POWER
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
TAIWAN	DR. YI-GIN CHEN	PWR	POWER PLANT	RELAP5			TO BE DETERMINED
USA	MR. PHILLIP TING	GERDA	SEPARATE EFFECTS	RELAP5	1605AA		
USA	MR. PHILLIP TING	NEPTUNUS	SEPARATE EFFECTS	RELAP5			PRESSURIZER

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	MIT	SEPARATE EFFECTS	RELAP5			PRESSURIZER
USA	MR. PHILLIP TING	MIT	SEPARATE EFFECTS	RELAP5			PRESSURIZER
USA	MR. PHILLIP TING	GE	SEPARATE EFFECTS	RELAP5	1004-3		BLOWDOWN
USA	MR. PHILLIP TING	EDWARDS PIPE	SEPARATE EFFECTS	RELAP5			BLOWDOWN
USA	MR. PHILLIP TING	CSE	SEPARATE EFFECTS	RELAP5	B-9		
USA	MR. PHILLIP TING	FLECHT-SEASET	SEPARATE EFFECTS	RELAP5	31504		
USA	MR. PHILLIP TING	CHRISTIANSEN	SEPARATE EFFECTS	RELAP5			HEAT TRANSFER
USA	MR. PHILLIP TING	B&W (SINGLE TUBE)	SEPARATE EFFECTS	RELAP5			HEAT TRANSFER

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 1		LOAD REJECTION OR TURBINE TRIP
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 2		RECIRCULATION PUMP TRIP
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 3		LOSS OF ON-SITE POWER
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 4		FEEDWATER HEADER TRIP
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 5		CLOSURE OF MAIN STEAM ISOLATION VALVES
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 6		POWER RAMPS WITH RECIRCULATION CONTROL
F.R. GERMANY	PROF. E. F. HICKEN	BWR	POWER PLANT	TRAC-BD1	BWR 7		LEVEL CONTROL
MEXICO				TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
SWITZERLAND	DR. O. MERCIER			TRAC-BD1			TO BE DETERMINED
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	4PM01		MSIV CLOSURE ATWS, NO ADS
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6DBA1B	LOCA	LARGE BREAK LOCA
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6MSB1		MAIN STEAM LINE BREAK
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6SB1		STUCK OPEN SRV

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6SB2C		SMALL BREAK LOCA WITH NO HPCS
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6PMC2		TO BE DETERMINED
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	6PMD3B		NATURAL CIRCULATION, LEVEL TRACKING
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	T2302/T1QUV		BOILING HEAT TRANSFER, LEVEL TRACKING
USA	MR. PHILLIP TING	FIST	INTEGRAL	TRAC-BD1	4PTT1		OVERPRESSURIZATION
USA	MR. PHILLIP TING	SSTF	INTEGRAL	TRAC-BD1			UPPER PLENUM MIXING, CCFL
USA	MR. PHILLIP TING	GE	INTEGRAL	TRAC-BD1			BORON MIXING, TRANSPORT AND STRATIFICATION
USA	MR. PHILLIP TING	FRIGG	SEPARATE EFFECTS	TRAC-BD1	613130		CORE VOID DISTRIBUTION
USA	MR. PHILLIP TING	FIX-II	INTEGRAL	TRAC-BD1			INTERMEDIATE SPLIT BREAK
USA	MR. PHILLIP TING	ROSA-III	INTEGRAL	TRAC-BD1			SMALL BREAK LOCA
USA	MR. PHILLIP TING	MARVIKEN	SEPARATE EFFECTS	TRAC-BD1		LOCA	LOCA
USA	MR. PHILLIP TING	GRAND GULF	POWER PLANT	TRAC-BD1			STARTUP TESTS
USA	MR. PHILLIP TING	BROWNS FERRY	POWER PLANT	TRAC-BD1			STARTUP TESTS

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	4		200% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	6		200% COLD LEG BREAK WITH BBR ECCS
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	8		200% COLD LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	10		200% COLD LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	11		200% HOT LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	12		200% COLD LEG BREAK WITH BBR ECCS
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	2		200% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	3		200% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	5		200% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	6		200% HOT LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	7		50% COLD LEG BREAK WITH COMBINED INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	UPTF	SEPARATE EFFECTS	TRAC-PF1	8		200% COLD LEG BREAK WITH BBR ECCS
F.R. GERMANY	PROF. E. F. HICKEN	KARLSTEIN	SEPARATE EFFECTS	TRAC-PF1			CO AND COUNTERCURRENT FLOW AT UPPER TIE PLATE
F.R. GERMANY	PROF. E. F. HICKEN	KARLSTEIN	SEPARATE EFFECTS	TRAC-PF1			CO AND COUNTERCURRENT FLOW AT UPPER TIE PLATE
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	TRAC-PF1	IIB4.1		200% HOT LEG BREAK WITH HOT LEG INJECTION
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	TRAC-PF1	IIB3		200% HOT LEG BREAK WITH HOT LEG INJECTION

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	TRAC-PF1			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PKL	INTEGRAL	TRAC-PF1			TO BE DETERMINED
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR1		SCRAM
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR4		TRIP OF ONE COOLANT PUMP
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR5		TRIP OF ONE COOLANT PUMP, DIFFERENT STEAM GENERATOR
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR6		STARTUP OF ONE COOLANT PUMP
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR7		TRIP OF ALL COOLANT PUMPS
F.R. GERMANY	PROF. E. F. HICKEN	PWR	POWER PLANT	TRAC-PF1	PWR8		STATION BLACKOUT
FRANCE	MR. M. REOCREUX	MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	403		STEADY-STATE CRITICAL FLOW; LOW PRESSURE, LOW FLOW RATE
FRANCE	MR. M. REOCREUX	MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	408		STEADY-STATE CRITICAL FLOW; LOW PRESSURE, HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	455		STEADY-STATE CRITICAL FLOW; INTERMEDIATE PRESSURE, HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	79		STEADY-STATE CRITICAL FLOW; INTERMEDIATE PRESSURE, HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	172		STEADY-STATE CRITICAL FLOW; HIGH PRESSURE, HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	1		STEADY-STATE CRITICAL FLOW; 12 MPA, 305.9C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	2		STEADY-STATE CRITICAL FLOW; 12 MPA, 319.7C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	3		STEADY-STATE CRITICAL FLOW, 12 MPA, 324.6C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	4		STEADY-STATE CRITICAL FLOW; 12 MPA, 0.05% X, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	5		STEADY-STATE CRITICAL FLOW; 12 MPA, 2.4% X, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	6		STEADY-STATE CRITICAL FLOW, 4.82 MPA, 234C, LONG NOZZLE, DIVERGENT

TEST MATRIX

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	7		STEADY-STATE CRITICAL FLOW; 3.33 MPA, 234C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	8		STEADY-STATE CRITICAL FLOW; 3.11 MPA, 234 C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	9		STEADY-STATE CRITICAL FLOW; 3.07 MPA, 234 C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	10		STEADY-STATE CRITICAL FLOW; 3.02 MPA, 234C, LONG NOZZLE, DIVERGENT
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	11		STEADY-STATE CRITICAL FLOW; 12 MPA, 305.9C, LONG NOZZLE, SUDDEN EXP
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	12		STEADY-STATE CRITICAL FLOW; 12 MPA, 319.7C, LONG NOZZLE, SUDDEN EXP
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	13		STEADY-STATE CRITICAL FLOW; 12 MPA, 324.6C, LONG NOZZLE, SUDDEN EXP
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	14		STEADY-STATE CRITICAL FLOW; 3.33 MPA, 234C, SHORT NOZZLE
FRANCE	MR. M. REOCREUX	SUPER MOBY-DICK	SEPARATE EFFECTS	TRAC-PF1	15		STEADY-STATE CRITICAL FLOW; 3.0 MPA, 0.9% X, SHORT NOZZLE
FRANCE	MR. M. REOCREUX	MARVIKEN	SEPARATE EFFECTS	TRAC-PF1	6		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	MARVIKEN	SEPARATE EFFECTS	TRAC-PF1	17		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	MARVIKEN	SEPARATE EFFECTS	TRAC-PF1	24		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	CANON	SEPARATE EFFECTS	TRAC-PF1	D		BLOWDOWN, CRITICAL FLOW; 3.2 MPA, 200C, 100MM DIA.
FRANCE	MR. M. REOCREUX	CANON	SEPARATE EFFECTS	TRAC-PF1	I		BLOWDOWN, CRITICAL FLOW; 3.2 MPA, 230C, 30MM DIA.
FRANCE	MR. M. REOCREUX	CANON	SEPARATE EFFECTS	TRAC-PF1	L		BLOWDOWN, CRITICAL FLOW; 3.2 MPA, 230C, 100MM DIA.
FRANCE	MR. M. REOCREUX	SUPER-CANNON	SEPARATE EFFECTS	TRAC-PF1	P		BLOWDOWN, CRITICAL FLOW; 15.0 MPA, 280C, 100MM DIA.
FRANCE	MR. M. REOCREUX	SUPER-CANON	SEPARATE EFFECTS	TRAC-PF1	Q		BLOWDOWN, CRITICAL FLOW; 15.0 MPA, 300C, 30MM DIA.

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
FRANCE	MR. M. REOCREUX	SUPER-CANON	SEPARATE EFFECTS	TRAC-PF1	X		BLOWDOWN, CRITICAL FLOW; 15.0 MPA, 320C, 100MM DIA.
FRANCE	MR. M. REOCREUX	VERTICAL-CANNON	SEPARATE EFFECTS	TRAC-PF1	9		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	VERTICAL-CANON	SEPARATE EFFECTS	TRAC-PF1	22		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	VERTICAL-CANON	SEPARATE EFFECTS	TRAC-PF1	24		BLOWDOWN, CRITICAL FLOW
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	D		SINGLE TUBE REFLOOD HEAT TRANSFER-STEADY STATE
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	H		SINGLE TUBE REFLOOD HEAT TRANSFER-STEADY STATE
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	Q		SINGLE TUBE REFLOOD HEAT TRANSFER-STEADY STATE
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	312 BIS		SINGLE TUBE REFLOOD HEAT TRANSFER-TRANSIENT
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	310		SINGLE TUBE REFLOOD HEAT TRANSFER-TRANSIENT
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	614		SINGLE TUBE REFLOOD HEAT TRANSFER-TRANSIENT
FRANCE	MR. M. REOCREUX	DADINE	SEPARATE EFFECTS	TRAC-PF1	313		SINGLE TUBE REFLOOD HEAT TRANSFER-TRANSIENT
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	3		SINGLE TUBE BLOWDOWN HEAT TRANSFER-QUALIFICATION RUN
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	9		SINGLE TUBE BLOWDOWN HEAT TRANSFER-QUALIFICATION RUN
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	6		SINGLE TUBE BLOWDOWN HEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	8		SINGLE TUBE BLOWDOWN HEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	29		SINGLE TUBE BLOWDOWN HEAT TRANSFER-SMALL BREAK
FRANCE	MR. M. REOCREUX	OMEGA-TUBE	SEPARATE EFFECTS	TRAC-PF1	30		SINGLE TUBE BLOWDOWN HEAT TRANSFER-SMALL BREAK
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2		36 ROD BLOWDOWN HEAT TRANSFER-QUALIFICATION RUN

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	18		36 ROD BLOWDOWN HEAT TRANSFER-QUALIFICATION RUN
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	3		36 ROD BLOWDOWN HEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	9		36 ROD BLOWDOWN HEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	11		36 ROD BLOWDOWNHEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	19		36 ROD BLOWDOWN HEAT TRANSFER-LARGE BREAK
FRANCE	MR. M. REOCREUX	OMEGA-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	13		36 ROD BLOWDOWN HEAT TRANSFER-SMALL BREAK
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	2080		SINGLE TUBE REFLOOD HEAT TRANSFER-LOW FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	1944		SINGLE TUBE REFLOOD HEAT TRANSFER-LOW FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	1955		SINGLE TUBE REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	1957		SINGLE TUBE REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	2070		SINGLE TUBE REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	2086		SINGLE TUBE REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-TUBE	SEPARATE EFFECTS	TRAC-PF1	2034		SINGLE TUBE REFLOOD HEAT TRANSFER-HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2513		36 ROD REFLOOD HEAT TRANSFER-LOW FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2517		36 ROD REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2523		36 ROD REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
FRANCE	MR. M. REOCREUX	ERSEC-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2625		36 ROD REFLOOD HEAT TRANSFER-INTERMEDIATE FLOW RATE
FRANCE	MR. M. REOCREUX	ERSEC-ROD BUNDLE	SEPARATE EFFECTS	TRAC-PF1	2704		36 ROD REFLOOD HEAT TRANSFER-HIGH FLOW RATE
FRANCE	MR. M. REOCREUX	EPIS-II	SEPARATE EFFECTS	TRAC-PF1	81.23		INJECTION/CONDENSATION
FRANCE	MR. M. REOCREUX	EPIS-II	SEPARATE EFFECTS	TRAC-PF1	80.19		INJECTION/CONDENSATION
FRANCE	MR. M. REOCREUX	EPIS-II	SEPARATE EFFECTS	TRAC-PF1	85.14		INJECTION/CONDENSATION
FRANCE	MR. M. REOCREUX	PATRICIA-GV1	SEPARATE EFFECTS	TRAC-PF1	1		STEAM GENERATOR PRIMARY SIDE HEAT TRANSFER
FRANCE	MR. M. REOCREUX	PATRICIA-GV1	SEPARATE EFFECTS	TRAC-PF1	2		STEAM GENERATOR PRIMARY SIDE HEAT TRANSFER
FRANCE	MR. M. REOCREUX	PATRICIA-GV1	SEPARATE EFFECTS	TRAC-PF1	3		STEAM GENERATOR PRIMARY SIDE HEAT TRANSFER
FRANCE	MR. M. REOCREUX	PATRICIA-GV1	SEPARATE EFFECTS	TRAC-PF1	4		STEAM GENERATOR PRIMARY SIDE HEAT TRANSFER
FRANCE	MR. M. REOCREUX	BETHSY	INTEGRAL	TRAC-PF1	1		FOUR CASES TO BE DETERMINED
FRANCE	MR. M. REOCREUX	BETHSY	INTEGRAL	TRAC-PF1			TO BE DETERMINED
FRANCE	MR. M. REOCREUX	BETHSY	INTEGRAL	TRAC-PF1			TO BE DETERMINED
FRANCE	MR. M. REOCREUX	BETHSY	INTEGRAL	TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1	LP-FP-1		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1	LP-SB-1		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1	LP-SB-2		TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SPAIN	DR. JOSE DECARLOS			TRAC-PF1			TO BE DETERMINED
SWEDEN	DR. ODDBJORN SANDERVAG	PWR	POWER PLANT	TRAC-PF1			GRID DISCONNECT AND TURBINE RUNBACK TO HOUSE LOADS
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	SB-1		SMALL BREAK LOCA (3") WITH RCPS
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	SB-2		SMALL BREAK LOCA (3") WITH RCPS
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	SB-3		SMALL BREAK LOCA WITHOUT HPI; RECOVERY USING SG FEED AND BLEED
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	L2-3		TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	L2-5		TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	LB-1		LARGE BREAK LOCA AT 50MW WITH RCPS DISCONNECTED FROM FLYWHEELS
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	L2-6		LARGE BREAK LOCA AT 46MW; EM TRANSIENT
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	FP-1		TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL	LOFT	INTEGRAL	TRAC-PF1	FP-2		TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL	LOBI-MOD2	INTEGRAL	TRAC-PF1	A2-81		SMALL BREAK LOCA
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TEN CASES FROM TRAC-PF2 ASSESSMENT TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TO BE DETERMINED
UNITED KINGDOM	DR. J. FELL			TRAC-PF1			TEN CASES IN OPTION YEARS 4 AND 5
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-UT-8		SBLOCA
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-SF-1		
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-FS-3		STEAMLINE BREAK
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-SF-4		
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-SF-5		FEEDLINE BREAK
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-IB-3		INTERMEDIATE BREAK
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-S6		STEAM GENERATOR TUBE RUPTURE
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-S8		STEAM GENERATOR TUBE RUPTURE
USA	MR. PHILLIP TING	SEMISCALE	INTEGRAL	TRAC-PF1	S-PL-3		LOSS OF POWER
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	300000		BASE CASE NATURAL CIRCULATION
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	310000		REDUCED FLOW TO THE STEAM GENERATOR
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	310503		REDUCED FLOW TO THE STEAM GENERATOR
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	330301		FEED AND BLEED
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	330302		FEED AND BLEED
USA	MR. PHILLIP TING	MIST	INTEGRAL	TRAC-PF1	360101		SMALL BREAK LOCA
USA	MR. PHILLIP TING	OTIS	INTEGRAL	TRAC-PF1	220100		BASE CASE NATURAL CIRCULATION

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	OTIS	INTEGRAL	TRAC-PF1	220402		STEAM GENERATOR CHARACTERISTIC
USA	MR. PHILLIP TING	GERDA		TRAC-PF1	1605AA		BOILER CONDENSER MODE OF NATURAL CIRCULATION
USA	MR. PHILLIP TING	LOBI	INTEGRAL	TRAC-PF1	A1-04R	LOCA	LARGE BREAK LOCA
USA	MR. PHILLIP TING	LOBI	INTEGRAL	TRAC-PF1	BRIN	LOCA	25% BREAK LOCA
USA	MR. PHILLIP TING	LOBI	INTEGRAL	TRAC-PF1	A2-B1		
USA	MR. PHILLIP TING	LOFT	INTEGRAL	TRAC-PF1	FW-1		
USA	MR. PHILLIP TING	LOFT	INTEGRAL	TRAC-PF1	L2-5		LARGE BREAK LOCA
USA	MR. PHILLIP TING	LOFT	INTEGRAL	TRAC-PF1	L2-6		
USA	MR. PHILLIP TING	LOFT	INTEGRAL	TRAC-PF1	L9-4		
USA	MR. PHILLIP TING	PKL	INTEGRAL	TRAC-PF1	1D1		NATURAL CIRCULATION
USA	MR. PHILLIP TING	PKL	INTEGRAL	TRAC-PF1	1D1-4		
USA	MR. PHILLIP TING	PKL	INTEGRAL	TRAC-PF1	1D1-B-13		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	1		DOWNCOMER AND COLD LEG FLUID MIXING
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	2		COLD AND HOT LEG BREAK (200% AND 50%) WITH COLD LEG INJECTION
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	3		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	4		LOWER PLENUM AND DOWNCOMER FLASHING/CD AND COUNTERCURRENT FLOW
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	5		LOWER PLENUM AND DOWNCOMER FLASHING/CD AND COUNTERCURRENT FLOW
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	7		LOWER PLENUM AND DOWNCOMER FLASHING/CD AND COUNTERCURRENT FLOW
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	9		HOT AND COLD LEG FLOW PATTERN
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	10		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	11		

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	12		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	13		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	14		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	15		
USA	MR. PHILLIP TING	UPTF	SEPARATE EFFECTS	TRAC-PF1	16		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	54		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	57		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	58		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	59		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	70		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	71		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	72		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	75		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	76		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	77		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	78		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	79		
USA	MR. PHILLIP TING	CCTF	SEPARATE EFFECTS	TRAC-PF1	80		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	529		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	604		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	605		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	608		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	610		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	611		

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	613		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	614		
USA	MR. PHILLIP TING	SCTF	SEPARATE EFFECTS	TRAC-PF1	617		
USA	MR. PHILLIP TING	FLECHT-SEASET	SEPARATE EFFECTS	TRAC-PF1	31504		REFLOOD HEAT TRANSFER
USA	MR. PHILLIP TING	FLECHT-SEASET	SEPARATE EFFECTS	TRAC-PF1	31701		REFLOOD HEAT TRANSFER
USA	MR. PHILLIP TING	FLECHT-SEASET	SEPARATE EFFECTS	TRAC-PF1	8		TWO-PHASE NATURAL CIRCULATION
USA	MR. PHILLIP TING	B&W OTSG	SEPARATE EFFECTS	TRAC-PF1	28		LOSS OF FEEDWATER
USA	MR. PHILLIP TING	B&W OTSG	SEPARATE EFFECTS	TRAC-PF1	29		LOSS OF FEEDWATER
USA	MR. PHILLIP TING	NEPTUNUS	SEPARATE EFFECTS	TRAC-PF1	Y05		PRESSURIZER
USA	MR. PHILLIP TING	DARTMOUTH	SEPARATE EFFECTS	TRAC-PF1			COUNTERCURRENT FLOW LIMITATION 3 TUBE
USA	MR. PHILLIP TING	NORTHWESTERN	SEPARATE EFFECTS	TRAC-PF1	253		CONDENSATION
USA	MR. PHILLIP TING	NORTHWESTERN	SEPARATE EFFECTS	TRAC-PF1	259		CONDENSATION
USA	MR. PHILLIP TING	NORTHWESTERN	SEPARATE EFFECTS	TRAC-PF1	293		CONDENSATION
USA	MR. PHILLIP TING	NORTHWESTERN	SEPARETE EFFECTS	TRAC-PF1	479		CONDENSATION
USA	MR. PHILLIP TING	NORTHWESTERN	SEPARATE EFFECTS	TRAC-PF1			COUNTERCURRENT FLOW LIMITATION AT UPPER TIE PLATE
USA	MR. PHILLIP TING	MB-2	SEPARATE EFFECTS	TRAC-PF1			STEAM GENERATOR
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	SB-CL-01	LOCA	SMALL BREAK LOCA
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	SB-CL-02	LOCA	SMALL BREAK LOCA
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	SB-CL-03	LOCA	SMALL BREAK LOCA
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	SB-CL-05	LOCA	SMALL BREAK LOCA, STEAM GENERATOR LIQUID HOLDUP
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	SB-CL-06	LOCA	SMALL BREAK LOCA, STEAM GENERATOR LIQUID HOLDUP
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	ST-NC-01		NATURAL CIRCULATION

***** TEST MATRIX *****

COUNTRY	CONTACT	FACILITY	TRANSIENT TYPE	CODE	TEST NUMBER	EXPERIMENT TYPE	EXPERIMENT DESCRIPTION
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	ST-NC-02		NATURAL CIRCULATION
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	ST-SG-01		STEAM GENERATOR RESPONSE
USA	MR. PHILLIP TING	ROSA-IV	INTEGRAL	TRAC-PF1	ST-SG-02		STEAM GENERATOR RESPONSE

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SEE INSTRUCTIONS ON THE REVERSE.

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13. ABSTRACT *(200 words or less)*

This document presents the guidelines and procedures by which the International Code Assessment and Applications Program (ICAP) will be conducted. The document summarizes the management structure of the program and the relationships between and responsibilities of the United States Nuclear Regulatory Commission (USNRC) and the international participants. The procedures for code maintenance and necessary documentation are described. Guidelines for the performance and documentation of code assessment studies are presented. An overview of an effort to quantify code uncertainty, which the ICAP supports, is included.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
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GUIDELINES AND PROCEDURES FOR THE INTERNATIONAL CODE ASSESSMENT
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