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CERTA-TN

Consolidation
of Integral System Test
Experimental Databases for
Reactor Thermal-Hydraulic Safety Analysis

European Thematic Network

Euratom Framework Programme 5 (1988-2002)
Key Action: Nuclear Fission

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Summary Report
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Foreword

This report has been produced to summarise the results of the CERTA Thematic Network established under the provisions of European Commission Euratom Framework Programme 5 (FP5) which is mainly dedicated at supporting research activities for the exploitation of the full potential of nuclear energy through the enhancement of related safety technologies

The specific objective of CERTA aimed at the consolidation of reactor safety experimental databases acquired in European integral system test facilities in order to ensure their preservation and an user-friendly data access and retrieve capabilities using modern web-based information technologies.

It is maintained that availability of reactor safety integral system test experimental databases will represent a continuing requirement in order to support the refinement of models and numerical schemes adopted in the current version of reactor safety system codes as well as to support development and assessment of advanced analytical methodologies.

The CERTA partners would like to acknowledge the institutional framework and financial support provided by the European Commission and the essential contribution from the experimental and analytical teams from the reactor safety research organisations whic have provided all relevant information for the execution of this project.

Overview

The programmatic objectives of the CERTA Thematic Network were primarily intended at establishing a consolidated framework for the preservation of reactor safety thermal-hydraulic databases acquired in European integral system test facilities and at providing data access/retrieval capabilities using modern web-based information technologies.

CERTA assembled 10 major institutional and industrial reactor safety research organisations from EU member countries as well as from EU Enlargement and from the European Free Trade Area (EFTA) countries; the Organisation for Economic Cooperation and Development – Nuclear Safety Agency (OECD-NEA) has participated in the network as interested observer in view of the NEA long standing initiative aimed at promoting preservation of reference data selected for the Computer Code Validation Matrix (CCVM). As configured, CERTA includes experimental programs and databases relevant to reactors in operation within the western European Countries as well as in the Central and East European Countries and in the New Independent States (i.e.; PWRs, BWRs and VVERs).

The overall CERTA workprogramme comprised three major work-packages which were planned for completion over a 24 months period starting from the official commencement date of October 1, 2000; i.e.:

WP1 - Assessment of current practices adopted within the CERTA participating organisations in the preservation of reactor safety thermal-hydraulic databases and in the maintenance of the related documentation.

WP2 - Compilation of data storage and access/retrieve requirements for the development and verification and/or validation of computer codes used in water reactor safety analysis.

WP3 - Establishment of a user-friendly distributed informatic platform using modern web-based information and communication technologies for data storage and with user-friendly access/retrieval provisions.

Recently, the Senior Group of Experts on Safety Research (SESAR) assembled by the OECD-NEA Committee on the Safety of Nuclear Installations (CSNI), reviewed the research being carried out in the field of reactor safety identifying future requirements and priorities. With respect to water cooled reactor integral system thermal-hydraulic research, SESAR recommended co-operative research programmes in some of the test facilities still in operation and the preservation of the acquired experimental databases focusing, as appropriate, on an international project in this field [1].

Within this overall framework, WP1 and WP2 were intended at providing reference information to complement the underlining rationales of WP3 which aimed at establishing a demonstration platform of web-based distributed databases. CERTA has been based on the STRESA [2] software developed and in use at the European

Commission - Joint Research Centre (EC-JRC) for storage and retrieval of the experimental data acquired in the LOBI and FARO/KROTOS experimental programs conducted in the frame of the JRC Euratom Research Program (http//asa2.stresa.jrc.it/).

It is retained that CERTA has fulfilled its intended scope and has represented an effective approach in networking established centres of excellence as foreseen in the frame of the provisions of the European Research Area (ERA) initiative. It has also been instrumental in the establishment of similar networks in other reactor safety research domains such as reactor safety severe accident research and has the potential for extension also in the non-nuclear field.

The approach used in CERTA has also resulted in spin-off technological implementation initiatives in other reactor safety research organizations; notably, the Electrogorsk Research and Engineering Centre (EREC) of the Russian Federation and OECD-NEA. EREC has adopted the underlining CERTA software for storage and access/retrieval of experimental data acquired in the PSB-VVER large scale thermal-hydraulic test facility. A feasibility study leading to an operational prototype has also been conducted in the frame of a collaboration agreement between the JRC and the OECD-NEA to exploit the CERTA approach for storage and retrieval of the experimental data currently selected for inclusion in the CCVM database.

1. INTRODUCTION

Following the construction of the first European nuclear power reactor in 1956 (i.e., the 50 MWe Magnox type GCR Calder Hall 1 at Seascale in Cumbria, UK), the nuclear power industry has developed across several European countries. The greatest expansion of the commercial European power industry occurred in the period 1970-1980 in response to international crises which encouraged many countries to diversify energy production lessening reliance on unreliable supplies of conventional fuels such as oil.

The Three Mile Island accident in 1979 and the Chernobyl disaster in 1986, however, as well as safety problems related to the treatment and disposal of radioactive waste, have had adverse consequences on further deployment of additional nuclear capacity in many countries. Nevertheless, nuclear power continues to represent an important contribution to the overall European and international energy economy.

It is maintained that continued operation of nuclear installations has to evolve within an adequate safety context to prevent potential adverse consequences to the general population and to the environment. Accident prevention is a widely accepted prerequisite for ensuring the safe operation of nuclear installations. The fundamental principles of accident prevention are the quantification of the safety margins in both operational and anticipated accident or transient conditions and the adherence to the prevailing regulatory framework and best practices guidelines.

Within this context, a general international consensus of opinions emerged in the early 70s on the need to provide reliable methodologies for a realistic estimate of the performance of the engineered safety systems. Notably, the effectiveness of the emergency-core-cooling system (ECCS) during the course of design basis loss-of-coolant accidents (LOCAs) or of any other anticipated abnormal sequence were questioned as the reactor power size was being increased. Emphasis was placed mainly on deterministic experimental and analytical methodologies supported, as appropriate, by probabilistic risk assessment studies with the aim to:

- acquire representative experimental databases in scaled integral system effect and/or separate effect test facilities in order to provide reference information for the fundamental understanding of major physical phenomena governing plant operation and the evolution of prospected accidents and transients,

- provide safety analysis capabilities through the development of system codes in order to realistically predict system and/or component behaviour in operational or accident conditions and quantify the related safety margins as well as the effectiveness of emergency operating procedures and eventual accident management strategies.

Since the experimental data acquired in scaled test facilities cannot be directly extrapolated to the full size plant due to scale-dependent distortions and simulation constraints, it is maintained that reactor safety analysis has thus to rely mainly on computational evidence provided by validated safety analysis codes. However, in order to verify that computer codes can accurately model the specified safety cases, a significant code validation effort against representative experimental data is required. Also, the code predictive capabilities have to be proven to be scale independent so that the full-size plant behavior can be predicted with an acceptable confidence.

The extent to which the existing reactor safety experimental databases are preserved and can be eventually accessed and/or recovered is an issue often debated in the nuclear community. In addition to the loss of related skilled resources, a compounding problem is given by the rapid advancement of computer hardware and software technologies which are making data storage methods rapidly obsolete and in some cases access and retrieval of the data practically impaired.

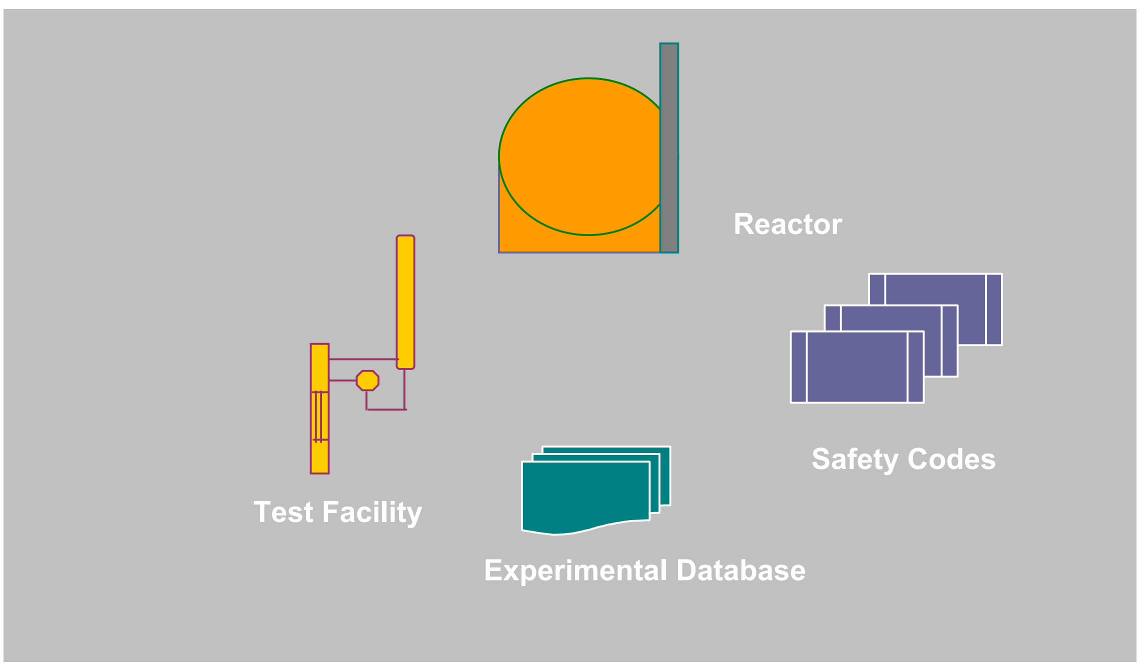
Access to integral system experimental databases will represent a continuing requirement in order to sustain reactor safety analysis activities as well as to support the refinement of models and numerical schemes of current as well as advanced reactor safety analysis codes. An additional element to consider is the capital investment required for the establishment and conduction of such large scale experimental programs which in the presence of current economic constraints facing the nuclear research community will be certainly difficult to revisit.

The CERTA Thematic Network thus attempts at establishing a collaborative framework among European research organisations for the consolidation of the reactor safety experimental databases acquired in integral system thermal-hydraulic test facilities to ensure their preservation and access/retrieve capabilities on the basis of modern information and communication web-based computer technologies.

It is retained that in addition to the inherent technical/scientific merits, CERTA has also provided a forum for the exchange of information on reactor safety research at the pan-European level; as such it has represented also a precursory approach in networking centres of excellence in the context of the European Research Area (ERA) initiative.

2. WORK PROGRAMME

A considerable amount of resources has been devoted at the European level during the past three-decades in order to plan and conduct experimental programs for the generation of reference thermal-hydraulic databases to support code development and assessment. Experimental programmes devoted to reactor thermal-hydraulic safety research have been traditionally based on 1) separate-effect experiments to investigate specific phenomena and develop/validate related analytical models and 2) integral-system-effect experiments to simulate reactor system response and develop/validate and assess reactor safety analysis codes.



Research into reactor safety integral-system-effect thermal-hydraulics originated in the early ‘70s in the US with the definition and execution of experimental programs in the Loss-of-Fluid-Test (LOFT) facility and in the Semiscale test facility. Similar research activities were also initiated in Japan with the establishment of the ROSA and later the Large Scale Test Facility (LSTF) experimental programs.

Referring to the European context, integral system effect thermal-hydraulic investigations initiated in Germany with the construction and operation of the Primärkreisläufe (PKL-I) test facility in the late '70s; other major European experimental programs which followed thereafter and are partners in the CERTA project are listed below. Experimental programs have also been promoted by research organisations in the Russian Federation to address VVER reactor types thermal-hydraulic safety issues such (e.g., the Integral System Test Facility (ITF) and the Large Scale Test Facility PSB experimental programmes).

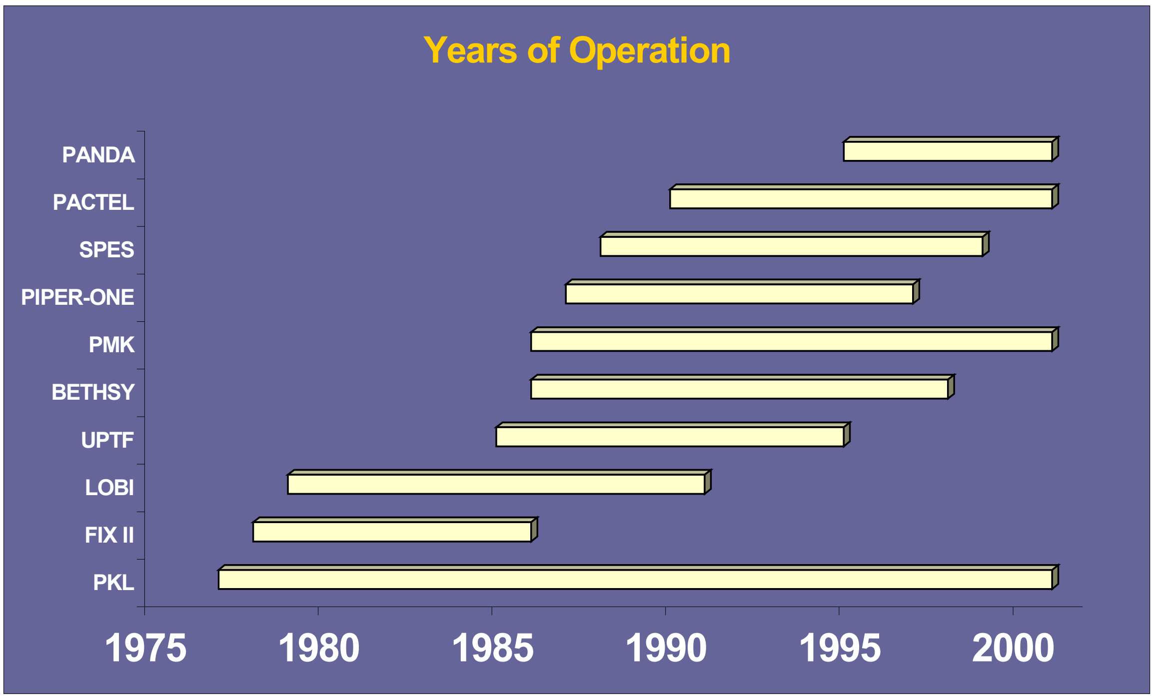
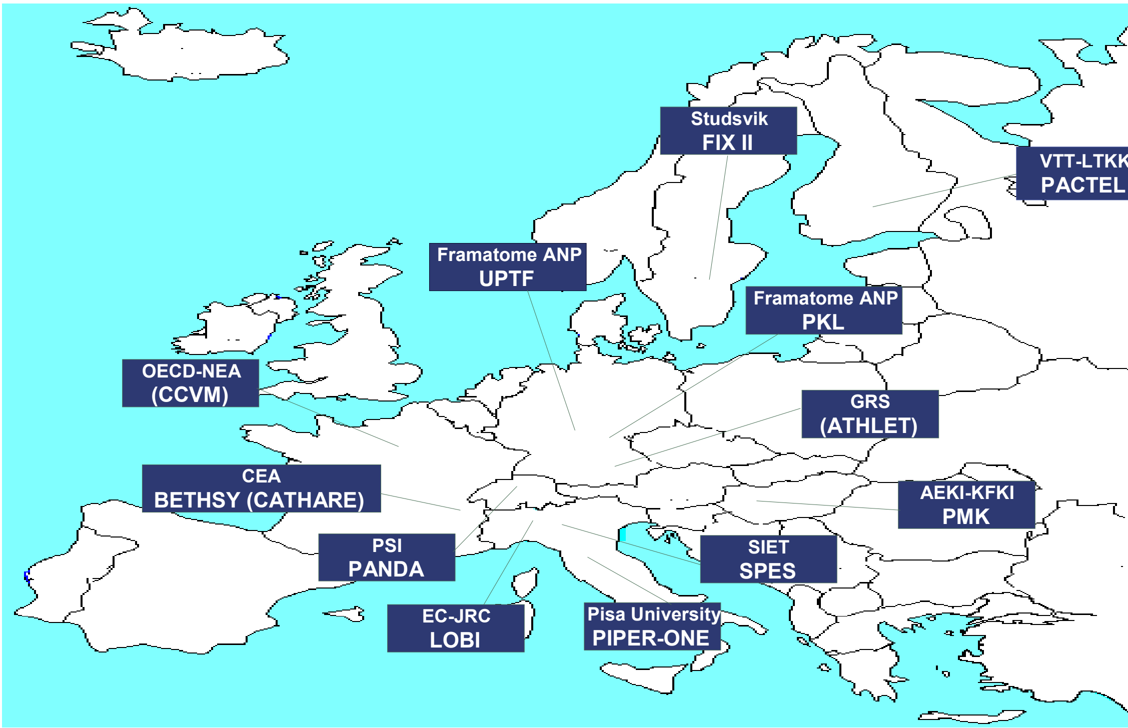


Table 1. The CERTA Consortium

Program	Denomination	Organisation	Country
PKL	PrimaerKreisLäufe	Framatome ANP - Erlangen	Germany
BETHSY	Boucle d'Etudes Thermohydraulique Systeme	CEA - Grenoble	France
SPES	Simulatore PWR per Esperienze di Sicurezza	SIET	Italy
LOBI	Loop Off-normal Behavior Investigations	JRC - Ispra	EC-JRC
UPTF	Upper Plenum Test Facility	Framatome ANP - Mannheim	Germany
PIPER 1	GE-BWR Test Facility	Pisa University	Italy
PACTEL	VVER 440 Test Facility	VTT Energy	Finland
PMK	VVER 440 Test Facility	KFKI	Hungary
FIX-II	ASEA-ATOM BWR Test Facility	Studsvik	Sweden
PANDA	Passive Decay Heat Removal and Depressurization Test	PSI	Switzerland

Networking European reactor safety research organisations.



In view of the transnational aspects related to nuclear reactor safety and taking into account contingent safety issues concerning the EU enlargement agenda, the network has been designed to include experimental programs and data bases relevant to reactors in operation within the EU member countries as well as to reactors in operation within the Central and Eastern-European Countries and the New Independent States.

The acquired experimental databases have been instrumental in the development/assessment of reactor safety thermal-hydraulic system codes for western European type pressurized and boiling water reactors (PWRs and BWRs) such as CATHARE, ATHLET, RELAP and TRAC as well as for safety codes dedicated to the analysis of accident and transients in eastern European type water cooled pressurized reactors (VVER) such as DYNAMICA, TECH, etc.

The design of integral system test facilities generally represents a scaled down mock-up of the full size reactor system. All test facilities suffer from simulation and construction

constraints with respect to the nuclear heat source which, for practical safety and economic reasons, cannot be simulated in a laboratory scale installation (LOFT being an exception) and by economic considerations which pose a limitation to the size of the installation as well as to system maximum operating pressure and temperature.

Tale 2: Test Facilities - Major Design and Scaling Parameters

Test Facility	Scaling Factor		Core Power (MWth)	Pressure (MPa)	Loops (#)	Heater Rods¨ (#)
	Volume	Elevation				
PKL III	135	1	1.5	4.0	1+1+1+1	340
LOBI	700	1	5.4	16.0	1+ 1(3)	64
BETHSY	100	1	3.0	15.5	1+1+1	428
UPTF	1	1	Steam Injection	2.0	4	193 dummy
SPES	420	1	30	15.5	1+1+1	97
PWR*	1	1	3800	15.0-16.0	1+1+1 +1	c. 51000
PIPER 1	2200	1	0.28	7.4	-	16
FIX-II	770	1	3.3	9.0	1 +(3)	36
PANDA	40	1	1.5	1	-	690
BWR*	1	1	3800	7.8-9.0	24	c. 52000
PMK	2070	1	2.0	1.6	6	18
PACTEL	305	1	1	8.0	3	144
VVER-440*	1	1	1375	12.5	6	c. 44000

* typical (#) lumped loops ¨ electrically heated

To meet the programmatic objectives, CERTA has been structured into three main work packages:

Table 3: CERTA – Work Packages Description

WP	Description	Deliverables	Leader
1	Assessment of current practices adopted within the CERTA participating organisations in the preservation of reactor safety thermal-hydraulic databases and in the maintenance of the related documentation.	Status Report EUR 19937 EN	Prof. F. D'Auria Pisa University
2	Compilation of data storage and access/retrieve requirements for the development and verification and/or validation of computer codes used in water reactor safety analysis.	Status Report EUR 20413 EN	F. Steinhoff GRS Garching
3	Establishment of a user-friendly distributed informatic platform using modern web-based information and communication technologies for data storage and user-friendly access/retrieval provisions.	DEMO Package http:/asa2.jrc.it./certa EUR 20421 EN	A. Annunziato JRC-IPSC

3. ACHIEVEMENTS

As previously mentioned, European institutional and industrial organisations initiated systematic research into reactor safety thermal-hydraulics in the late '70s with the PKL, LOBI and FIX test facilities. The major research effort was acquired in the '80s with the construction and commissioning of a number of integral system test facilities such as

BETHSY, SPES, UPTF, PIPERONE, and PMK. It continued with a limited effort in the '90s with the commissioning of the PACTEL and PANDA test facilities.

To date, some of the reported experimental programmes have been terminated and several test facilities dismantled; e.g., LOBI, BETHSY, UPTF and FIX-II. Some of the test facilities are still in operation such as PKL, PACTEL, PMK and PANDA, SPES and PIPERONE are in stand-by conditions. The Table below provides a synopsis on the status of the experimental programs providing also an indication of the related capital investment.

Table 4: European LWR Integral System Test Facilities (status:10.2002)

No.	Program	Reference Reactors System	Operation	Kv	Status	Partial Costs Hard/Software (M€*)	Overall Cost (M€*)
1	PKL I PKL II PKL III A PKL III B PKL III C PKL III D PKL III E	KWU-4loop PWR	1977-1981 1981-1986 1986-1989 1989-1992 1992-1995 1995-1999 since 2000	145	in operation	-	50
2	BETHSY	Framatome PWR		100		15/38	53
3	SPES 1 SPES 2 SPES 99	W-312 PWR W-AP600 2 loop PWR	1988-1991 1991-1994 1994-1999	427 395	in stand by	8/5 2/3 0.2/0.2	18.4
4	LOBI MOD1 LOBI MOD2	KWU-4loop PWR	1979-1982 1982-1991	712	dismantled	50/90	140
5	UPTF 2D/3D UPTF TRAM	KWU-4loop PWR	1985-1990 1991-1995		dismantled	-	215
6	PIPER-ONE	GE-BWR SBWR	1987-1990 1990-1997	2200	in stand by	1.8/2.2	4
7	PACTEL	VVER-440	Since 1990	305	in operation	2.1/2	4.1
8	PMK	VVER-440		2070		1/0.8	1.8
9	FIX-II	ASEA-ATOM BWR	1978-1986	777	dismantled	2.3/3.5	6
10	PANDA	GE-SBWR ESBWR SWR1000	Since 1995	40	In stand by(#)	5/5	10

(*) M€ related to year 2000 (#) Facility being modified for FP5 funded projects TEMPEST and NACUSP

3.1 Current Practices in Databases Maintenance (WP1)

Work Package 1 [3] was aimed at collecting reference information on current practices in the maintenance of experimental database acquired in the participating European reactor safety integral system test facilities. The ensuing report contains information on the status of the experimental programmes, a short description of the test facilities with performed test matrices, an overview of test data format with supporting documentation concerning test facility and test results analysis.

Emphasis has been given on collating information on current test data preservation practices and data format including support medium. Since test data analysis reports is a prerequisite in order to ensure a proper and efficient use of the test data type availability and eventual reference of existing reports is also provided. A short overview of major achievements of each research programme is also given.

The acquired overall experimental database consists of c. 1000 experimental tests covering a wide range of reactor accident conditions and reactor reference system configurations. It includes large and small break loss-of-coolant accidents (LOCAs), anticipated transients with and without scram (ATWS), emergency operating procedures (EOP) and accident management (AM) strategies.

Dedicated test runs have been performed in several of the test facilities to investigate specific thermal-hydraulic phenomena such as natural circulation heat transport mechanisms, steam generator heat transfer modes and heat transport degradation. In addition, test facility characterisation experiments have been conducted in some cases to provide information on non-prototypical heath transfer phenomena such system heath losses and pressure drops which are important to be accounted for in code simulation.

Currently, experimental databases are maintained in variety of forms and format (e.g., paper support, tapes, diskettes, CDs). In addition to data storage and accessibility concerns, other issues must also be considered in order to ensure widespread or selected access and use of the data. Sufficient information on facility design and instrumentation as well as on operational characteristics are required to construct the code input model for system code pre- and post-test prediction calculations.

To date several experimental programmes have been terminated and several test facilities such as LOBI, BETHSY, UPTF and FIX-II have been dismantled. A number of test facilities are either still in operation such as PKL, PACTEL, PMK and PANDA or kept in stand-by conditions such as SPES and PIPER-ONE. The overall financial investment accrued in the establishment and execution of the experimental programmes included in CERTA has been estimated in the order 450 M€. The capital investment is thus rather substantial and needs to be considered in the context of current and prospected financial constraints that can make future large scale research programmes in this field highly unlikely.

Table 5: Status of Experimental Database Maintenance

No.	Program	Number of Tests	Archiving Medium			Other
			Modern e-Support	Old e-Support	Paper Support	
1	PKL I - PKL II	62			62	
	PKL III A PKL III B PKL III C PKL III D PKL III E	74	30	44		
2	BETHSY	82	82			
3	SPES 1	10	-	5		
	SPES 2	15	-	15		
	SPES 99	1	1	-		
4	LOBI MOD1 LOBI MOD2	70	70			
5	UPTF 2D/3D UPTF TRAM	237	237			
6	PIPER-ONE	34	5	25		4
7	PACTEL	184	22	162		
8	PMK	48	27	21		
9	FIX-II	93	5	-	88	
10	PANDA	58	58			

3.2 Databases Access and Retrieve Requirements (WP2)

Experimental databases acquired in scaled experimental installations constitute the primary reference information for benchmarking the predictive capabilities of large system codes used in the safety evaluation of full size reactor systems. The CERTA WP2 [4] aimed at collating code users requirements with specific emphasis on information needs for code input deck built-up and access/retrieve priorities to carry out comparison analysis between predicted and measured parameters.

There is a general consensus of opinions on persisting needs to access integral system experimental databases in order to support the application of the current generation of safety codes as well as to sustain the refinement of models and numerical schemes in advanced or even new code versions. In addition to dedicated reactor safety analysis applications , it is also retained that experimental databases constitute a significant reference test bench to support educational and training activities.

Referring to experimental databases access/retrieve requirements for code verification/validation and assessment, emphasis need to be placed on the following main aspects:

- Preservation of the data using modern information technologies in order to facilitate access/retrieve capabilities as well as upgradability to comply with the introduction/evolution of new hardware/software technologies.

- Maintenance of the reference documentation for the interpretation of the experimental results that should include among others, Test Specifications, Experimental Data Report and Test Results Analysis Report.

- Availability of reference information concerning test facility design and operational characteristics with inclusion of system and components construction drawings, instrumentation longitudinal and axial location to optimize code input deck build-up and nodalization scheme.

Additional specific information requirements for code users include the availability of test facility characterization test data as well as provision of operational data concerning major components especially when they may be affected by non-prototypical behavior:

- Lay-out and geometrical configuration
- Overall and relative elevations and pipework alignment
- Material properties
- System and components pressure losses
- Amount and distribution of heath losses and heat gains
- By-pass flow paths and pipework or components dead-ends
- Location and quantification of eventual system leaks
- Main coolant pumps 1 and 2 phases hydraulic behaviour
- Characteristics of safety and relief valves
- High pressure and low pressure safety injection systems characteristics
- Normal and auxiliary feed-water systems and SG recirculation characteristics
- Volume control system
- Accumulator injection system behaviour
- Control and data acquisition system characteristics
- Instrumentation location, characteristics, eventual uncertainties and error bands

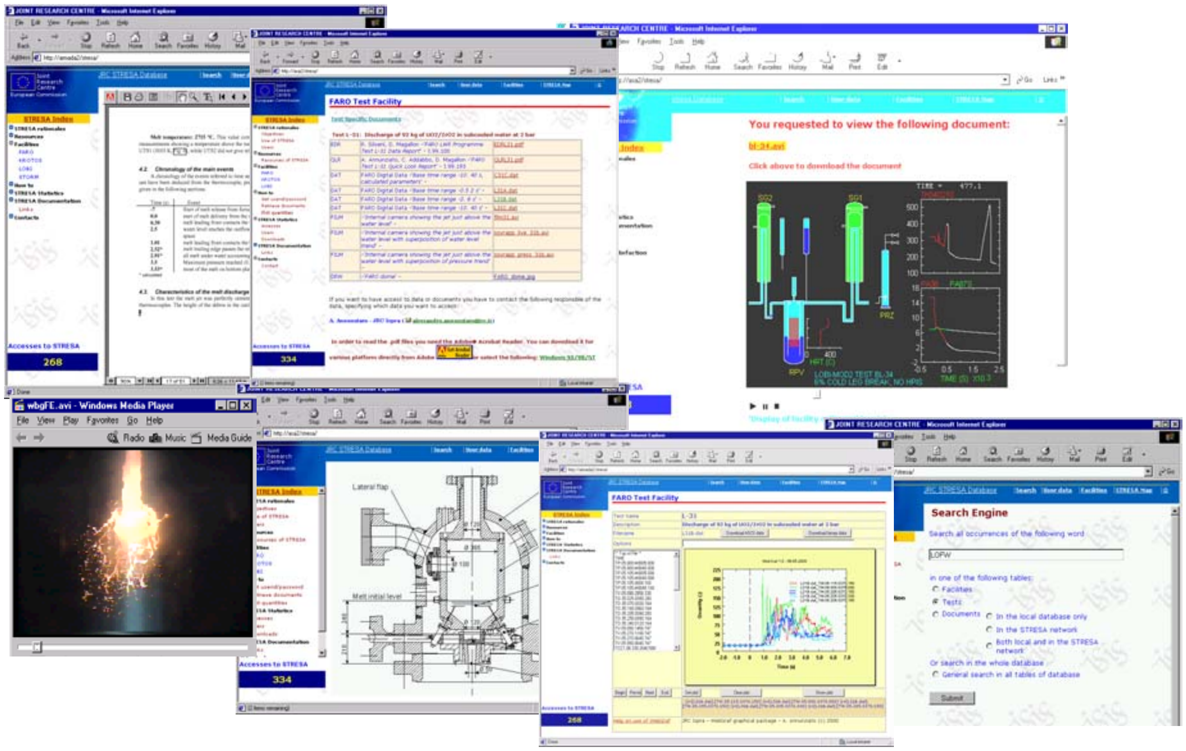
Taking into account the above that is clearly not to be considered exhaustive, special emphasis should be placed on the identification and description of scaling principles and simulation constraints in order to place in the right perspective the scale-up of the code predictive capabilities.

3.3 Establishment of the CERTA Platform (WP3)

The main added value of the CERTA initiative is the exploitation of advanced hardware and software computer technologies to ensure a distributed repository of reactor safety experimental databases with access/retrieval control and authorization from each database originating organisation.

As such, the CERTA WP3 [5] envisaged the provision of a demonstration package consisting of a web-based informatic platform networking the different databases. The CERTA prototype has been established using as supporting software the STRESA [2] approach which has been developed and is being successfully used at EC Joint Research

Centre (JRC) for storage and dissemination of the experimental data acquired in the LOBI and FARO/KROTOS experimental programs which are accessible at http://asa2.jrc.it/stresa.

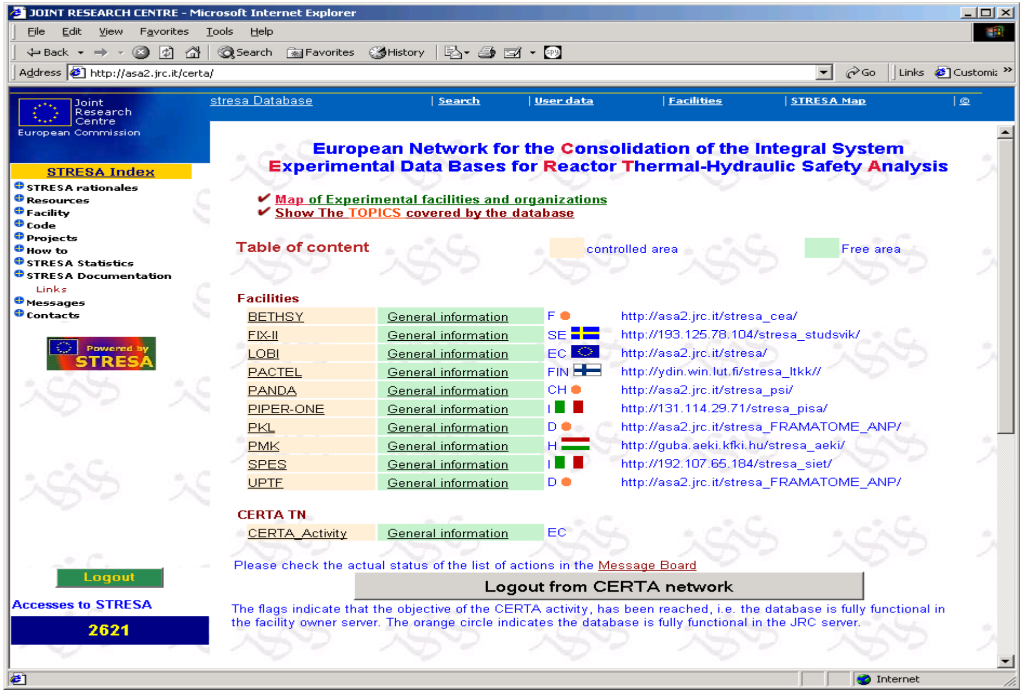


STRESA Views

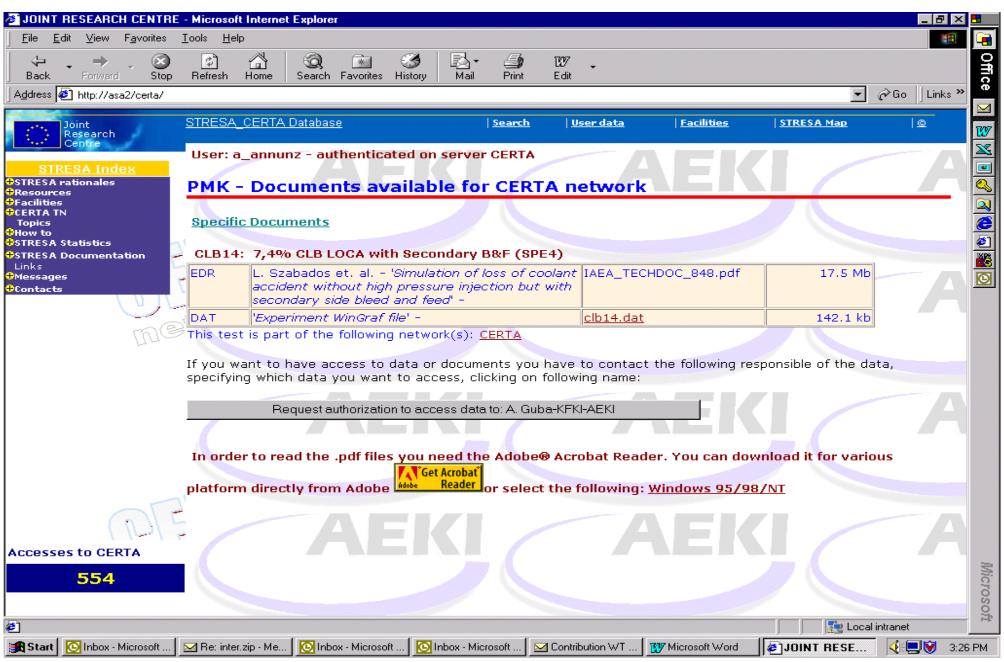
STRESA is a general purpose database designed to archive data from plants or experimental set-ups as well as from code calculations. Access to the data is through the so called three tier arrangement whereby the user connects to a server that will then provide access to a database.

STRESA can be configured as a network of database interconnecting nodes of local databases. Within this arrangement authorization to access and download data rests with the local database manager ensuring thus data property rights. This feature could overcome the reluctance of some organizations to release their data to a central repository and as such it has received attention from the OECD/NEA as a possible alternative to store the data selected for the Computer Code Validation Matrices.

The CERTA platform has been successfully established and is operational; each database is accessible connecting at either the CERTA main portal or at the individual local links.



CERATA Main Portal



CERTA AEKI Link

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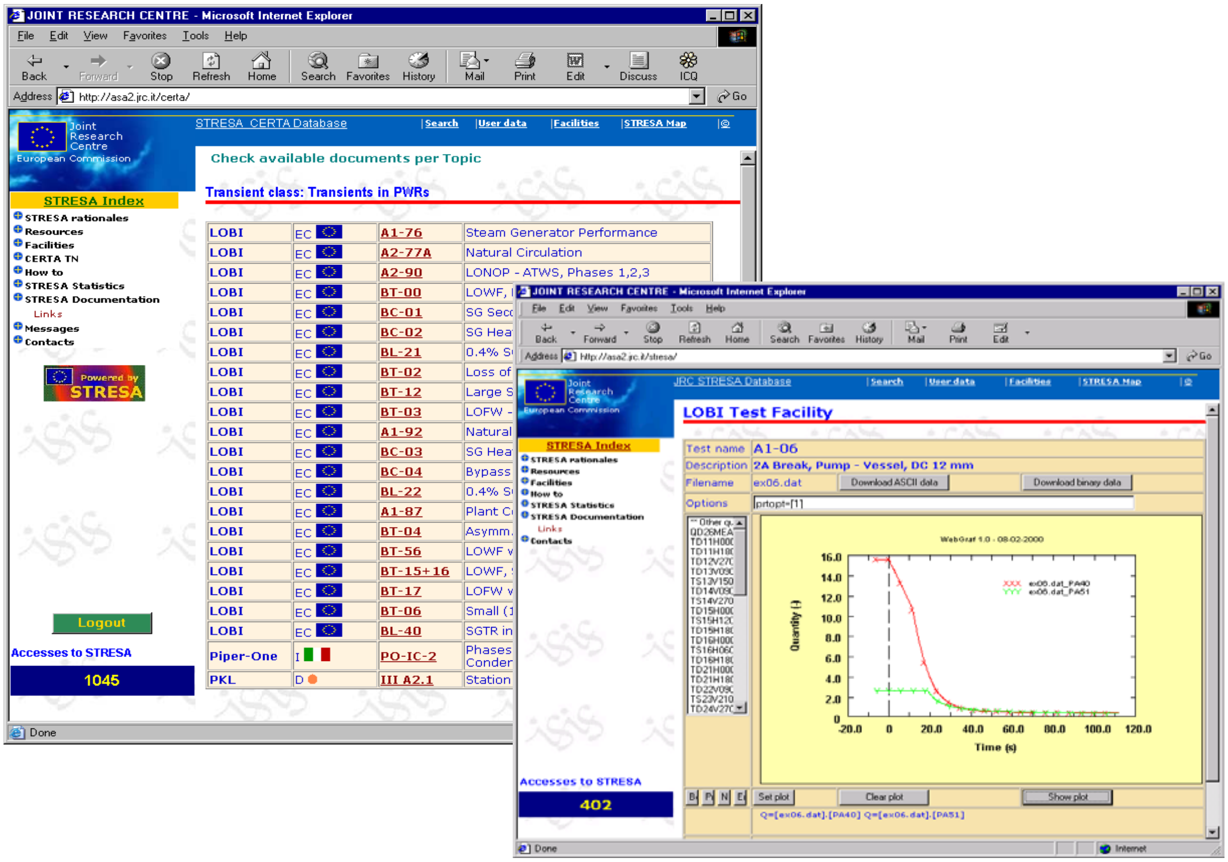
As planned, CERTA was mainly envisaged as a precursory exercise to demonstrate the feasibility of networking distributed databases with controlled access/release provided by the data owner. For each experimental programme two tests have been selected for inclusion in the database; in some cases more tests and in the case of LOBI all 70 performed tests have been included and are accessible via web.

Table 6: CERTA Network Databases

Programme	Tests	URL
BETHSY	6.9c: Loss of RHR at midloop operation, pressuriser and SG outlet man-ways open 9.1b: LOCA with break in cold leg without HPIS and delayed application of an ultimate procedure	http://asa2.jrc.it/stresa_cea
FIX II	2032: RC pump coast down simulation for internal RCP BWR with low pump inertia 3061: 100% break in circulation line 5052: 200% break in circulation line 6261: MSIV closure	http//: 193.125.78.104/stresa_studsvik
LOBI	All 70 Tests : LOCA and Special Transients	http//:asa2.jrc.it/stresa
PACTEL	ISP 33: Stepwise inventory reduction LOF-10: Loss of Feedwater	http://ydin.win.lut.fi/stresa_ltkk
PANDA	ISP 42: standard problem Phases 1 to 5	http://asa2.jrc.it/stresa_psi
PIPERONE	PO-IC-2: A/II and B Isolation Condenser phenomena PO-SB-7: ISP 21 Small Break 2.6% LOCA in recirculation line and core uncovery	http//:131.114.29.71/stresa_pisa
PKL	PKL-III A2.1 Cool-down procedure with 4SGs under loss of off-site power conditions PKL III B3.3 Reflux condenser CCFL Test UPTF	http://sa2.jrc.it/stresa_FRAMATOME_ANP
UPTF	10B-run081 Steam/water flow phenomena at the upper tie-plate 5A-run063: Steam/water flow phenomena in intact cold leg	http://sa2.jrc.it/stresa_FRAMATOME_ANP
PMK	CLB14 7.4% LOCA with secondary B%F (SPE4) PRISE2 SG Collector cover lift-up (SPE3)	http://guba.aeki.kfki.hu/stresa_aeki
SPES	SPW02: Loss of feed-water with bleed and feed SPSB03: SBLOCA 6" with decay power, CPT	http://arancia.arcoveggio.enea.it/stresa_siet

The CERTA overall informatic structure is supported by search and filtering possibilities as well as by on-line data plotting and analysis routines provided by the embedded WinGraf data processing and visualisation package.

The main components of the CERTA informatic structure are a) the files on disk, b) the Access database and c) the html-asp pages. The data sets from the different experimental programmes are converted into a same format, the WinGraf format developed at the JRC that provides an harmonised treatment of the data and, eventually, unified conversion mechanisms in view of the evolution or introduction of new hardware and software computer technologies.

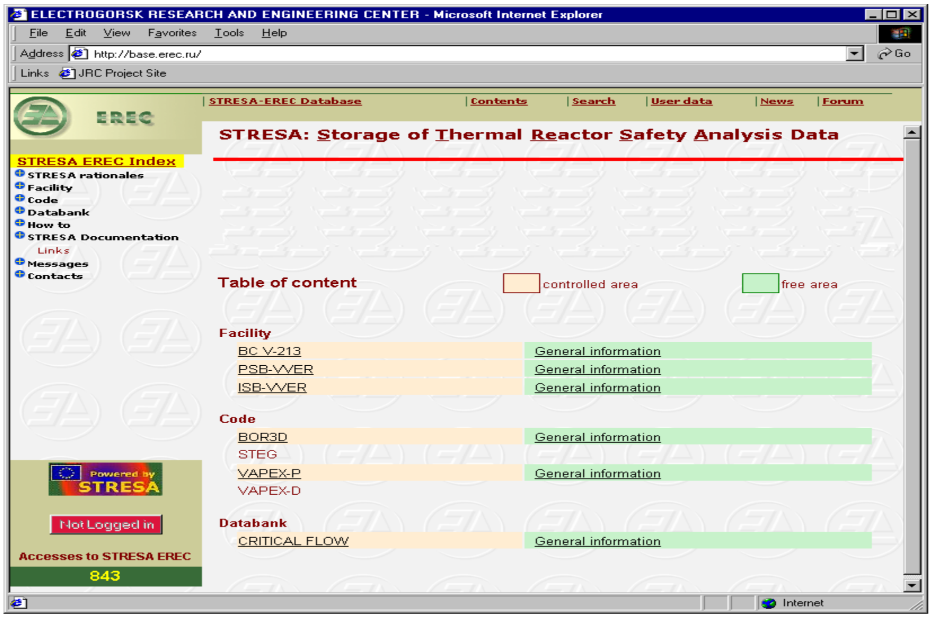


Searching/Plotting/Processing Data

4. DISSEMINATION AND EXPLOITATION O RESULTS

A major risk facing the preservation and dissemination of any type of massive scientific or analytical databases is certainly the obsolescence of storing media that in time are not served by adequate reading and conversion tools. This is in turn compounded by progressive loss of know-how and know-where that often resides in the mind of experts that in time is being dispersed or even lost. The CERTA is an attempt to address this issue at the pan-European level and represent an viable approach to cope with some eventual contingencies in this field.

CERTA has also resulted in spin-off technological implementation initiatives in participating organizations such as the LTKK/FIN and in other reactor safety research organizations outside the CERTA context such a the Electrogorsk Research and Engineering Centre (EREC) of the Russian Federation where the underlining rationales are currently being used for storage and access/retrieval of experimental data acquired in large scale thermal-hydraulic test facilities. A feasibility study has also been conducted in the frame of a collaboration agreement between the JRC and the OECD-NEA to exploit the CERTA approach for storage and retrieval of the experimental data currently available in the CCVM database.



EREC Database Home Page

Prototype of OECD/NEA CCVM Database Home Page

5. CONCLUDING REMARKS

A comprehensive experimental database has been acquired in European integral system test facilities during the last three-decades to support thermal-hydraulic safety analysis of water cooled reactors (i.e., PWR, BWR and VVER) and development/assessment of related analytical methodologies. The acquisition of these data has requested a considerable financial commitment from both institutional and industrial reactor safety research organisations which has been estimated in the order of 450 Million EUROs.

To date, some of the experimental programmes have been terminated and several test facilities dismantled; e.g., LOBI, BETHSY, UPTF and FIX-II. Some of the test facilities are still in operation such as PKL, PACTEL, PMK and PANDA; SPES and PIPERONE are in stand-by conditions.

The integral system test databases that are subject matter of CERTA, are currently maintained in a variety of support media and format. Some support media are not any more serviced by adequate hardware/software and some data are available only on paper format. The extent to which supporting documentation (e.g., test data analysis reports, test facility design drawing, test facility description and information on instrumentation system, etc.) is preserved is in some cases questionable.

It is generally recognised that the CERTA experimental databases represent a unique set of reference information relevant to accident and transient analysis of water cooled reactors. Due to current and even prospected reactor safety research financial constraints that can prohibit execution of such large scale experimental programmes in the near future, it is felt that it is an obligation of the nuclear community to ensure preservation of currently available databases and to provide related user-friendly access/retrieval capabilities based on modern information technologies [6].

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