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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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CSNI INTERNATIONAL STANDARD PROBLEMS (ISP)

Brief descriptions (1975-1999)

88420

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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

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

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

PREFACE

The attached report has been prepared by Prof. Dr.-Ing. Helmut Karwat, acting as a consultant to NEA. The document, which has been reviewed by members of CSNI's Principal Working on Coolant System Behaviour (PWG2) and Principal Working on Confinement of Accidental Radioactive Releases (PWG4), summarises systematically the main features of all International Standard Problem exercises organised so far, from ISP-1 to ISP-43, namely:

- Objectives of the ISP
- Brief Description of the Facility
- Scaling Information
- Parameters Offered for Comparison
- Dominating Experimental Uncertainties
- Findings
- Recommendations
- Total Duration of the ISP exercise
- Participation
- CSNI Report Reference

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FOREWORD

Over the last twenty-five years the NEA Committee on the Safety of Nuclear Installations (CSNI*)* has sponsored a considerable number of international activities to promote the exchange of experience between its Member countries in the use of nuclear safety codes and testing materials. A primary goal of these activities is to increase confidence in the validity and accuracy of analytical tools or testing procedures which are needed in warranting the safety of nuclear installations, and to demonstrate the competence of involved institutions. Three main areas of CSNI-sponsored international comparative studies may be distinguished.

International Standard Problems (ISPs) exercises are comparative exercises in which predictions or recalculations of a given physical problem with different best-estimate computer code are compared with each other and above all with the results of a carefully specified experimental study. ISP exercises are performed as "open" or "blind" problems. In an open Standard Problem exercise the results of the experiment are available to the participants before performing the calculations, while in a blind Standard Problem exercise the experimental results are locked until the calculation results are made available for comparison.

In contrast, **Computational Benchmark Studies (Benchmarks)** are specified to compare numerical solution methods for a given set of mathematical equations describing physical processes (e.g., in the field of neutronics or thermal-hydraulics) with each other and - if available - with the analytical solution of the given equations. More recently, also the numerical procedures for the exchange of data between numerically coupled sets of equations (e.g., between neutron kinetics and thermal-hydraulic equations) are the subject of benchmark exercises. (It is to be noted that, outside the framework of CSNI activities, code comparison exercises similar to International Standard Problem exercises are sometimes also called Benchmarks.)

Both for International Standard Problem exercises and for Benchmark studies, it is desirable to specify a yardstick allowing to judge which of the submitted calculations is closest to "reality" or the "true solution". For an International Standard Problem the measurements of a number of relevant parameters of the experiment serve as "yardstick" to which the submitted calculations must be compared. In a Benchmark exercise the submitted calculations are to be compared to the mathematical solution of the given set of equations, if such a solution exists. However, in most cases a mathematical solution is not available leaving the participants in a Benchmark exercise with the often insurmountable difficulty of deciding which of the submitted calculations is closest to reality.

A third category of comparative studies is the so-called **Round Robin Tests (RRT)** serving to compare non-destructive testing or inspection techniques on materials or material probes to identify specific properties or the quality of such probes. Well known is the Programme for the Inspection of Steel Components (PISC I to III) performed under the auspices of OECD/NEA and the European Commission in which test assembles were studied which contained deliberately introduced flaws and failures. In these cases the final destructive inspection of the probes served as an objective "yardstick" to group or classify the submitted results.

The present document is restricted to International Standard Problems. The CSNI-promoted International Standard Problem (ISP) activity started in the early 70s and is still underway. Parallel to other national and international programs the CSNI has sponsored over more than 25 years forty-seven International Standard Problem exercises. This program has been focused mainly on the applicability of large thermal-hydraulic code systems simulating the behaviour of nuclear coolant and containment systems, fuel behaviour under accident conditions, hydrogen distribution, core-concrete interactions and fission product release and transport. One ISP exercise was organised in connection with a seismic ultimate dynamic response test.

ISP exercises have proven to be very valuable to participating countries. They have been fruitful to identify code application problems and to amplify the contacts between the experimental and analytical working communities on an international level. Moreover, they enable code users to improve their ability, gain experience and demonstrate their competence.

Technical proposals originating from laboratories in 13 countries were submitted and adopted serving as experimental background for the international activity.

An activity complementary to the ISP programme but not directly related to it is the establishment of comprehensive sets of experimental data for use in validating proper code application by experts other than code developers. Known as a Code Validation Matrix, the results of such an extensive list of experiments are collected and stored in the NEA Data Bank for the purpose that they be made available to Member countries wishing to validate relevant codes. ISP tests are important elements of validation matrices for thermal-hydraulic codes. In the field of containment behaviour it has been recommended to preferably base a general code validation matrix on containment-related ISPs as well.

The way of organising ISP exercises has been described in CSNI Report No. 17, CSNI Standard Problem Procedures, revised for the third time in November 1989.

The following report in some detail summarises the main objectives, the underlying experimental background and the main findings and conclusions of the ISPs to provide interested groups an overview on the accumulated experience in this matter.

Supplementing the study of the relevant ISP Reports the interested reader is also advised to consult the minutes of the workshops held in conjunction with the particular ISP exercises which provide some additional information on encountered problems.

Prof. Dr.-Ing. H. Karwat

Countries Contributing to the Experimental Background of ISPs

Country	**Proposals**
Australia	1
Belgium	1
Canada	2
Finland	1
France	4
Germany	12
Italy	2
Japan	4
Sweden	2
Switzerland	1
United Kingdom	3
USA	10
EC/JRC Ispra	3
Total	**46**

General Overview on ISP Topics *

ISP 01 SET Straight Pipe Depressurization Experiment (Edwards' Pipe)

ISP 02 INT Standard Problem 2 (Semiscale Test 1011)

ISP 03 SET Comparison of LOCA Analysis Codes

ISP 04 INT UNITED STATES STANDARD PROBLEM 4 / INTERNATIONAL STANDARD PROBLEM 8
(Simulation of Semiscale MOD 1 Test S-02-6)

ISP 05 INT UNITED STATES STANDARD PROBLEM 7 / INTERNATIONAL STANDARD PROBLEM 5
(Nonnuclear Isothermal LOFT Blowdown Test L1-4)

ISP 06 SET Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase

ISP 07 SET Analysis of a Reflooding Experiment

ISP 08 INT Semiscale MOD 1; Test S-06-03 (LOFT Counterpart Test)

CASP1 CON Steamline Rupture within a Chain of Compartments (Battelle Test D15)

ISP 09 INT LOFT Nuclear Experiment L3-1

ISP 10 INT Refill and Reflood Experiment in a Simulated PWR Primary System (PKL)

ISP 11 INT LOFT Nuclear Experiment L3-6/L8-1

CASP2 CON Water Line Rupture into a Branched Compartment Chain (Battelle Test D16)

CASP3 CON Small-Scale Two-Compartments Basic Containment Experiment

ISP 12 INT ROSA-III 5% Small Break Test, Run 912

ISP 13 INT LOFT Nuclear Experiment L2-5

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* *Type of Specified Experiment:*

INT *Integral Coolant Systems Experiment*

CON *Integral Containment Experiment*

SET *Separate Effects Test*

SEISM *Seismic Test*

ISP 14 SET Behaviour of a Fuel Bundle Simulator during a Specified Heatup and Flooding Period (REBEKA-Experiment) - Results of Post-Test Analyses

ISP 15 INT LOCA Experiment in the Swedish FIX-II Facility Related to BWRs

ISP 16 CON Rupture of a Steam Line within in the HDR Containment leading to an Early Two-Phase Flow

ISP 17 CON Marviken: Pressure Suppression Containment - Blowdown Experiment No.18

ISP 18 INT LOBI-MOD 2 Small Break LOCA Experiment A2-81

ISP 19 SET Behaviour of a Fuel Rod Bundle during a Large Break LOCA Transient with a two Peaks Temperature History (PHEBUS Experiment)

ISP 20 INT Steam Generator Tube Rupture in the Nuclear Power Plant DOEL-2, Belgium

ISP 21 INT PIPER-ONE Test PO-SB-7 on Small Break LOCA in a BWR Recirculation line

ISP 22 INT Loss of Feedwater Transient in Italian PWR (SPES Test SP-FW-02)

ISP 23 CON Rupture of a Large-Diameter Pipe in the HDR Containment

ISP 24 SET SURC-4 Experiment on Core Concrete Interactions

ISP 25 SET ACHILLES Best-Estimate Natural Reflood Experiment with Nitrogen Injection from Accumulators

ISP 26 INT ROSA-IV LSTF 5% Cold Leg Small-Break LOCA Experiment

ISP 27 INT BETHSY Experiment 9.1B; 2" Cold Leg Break without HPSI and with delayed ultimate procedure

ISP 28 SET PHEBUS-SFD B9+ Experiment on the Degradation of a PWR Type Core

ISP 29 CON Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions

ISP 30 SET BETA V5.1 Experiment on Melt-Concrete Interaction

ISP 31 SET CORA-13 Experiment on Severe Fuel Damage

ISP 32 cancelled

ISP 33 INT PACTEL Natural Circulation Stepwise Coolant Inventory Reduction Experiment

ISP 34 SET FALCON Fission Product Experiments FAL-ISP-1 and FAL-ISP-2

ISP 35 CON NUPEC Hydrogen Mixing and Distribution Test (Test M-7-1)

ISP 36 SET CORA-W2 Experiment on Severe Fuel Damage for a VVER-type PWR

ISP 37	CON	VANAM M3 -A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material
SSWISP		SEISM Seismic Shear Wall ISP NUPEC's Seismic Ultimate Dynamic Response Test
ISP 38	INT	BETHSY Experiment 6.9c: Loss of Residual Heat Removal System during Mid-Loop Operation
ISP 39	SET	FARO Test L-14 on Fuel Coolant Interaction and Quenching
ISP 40	SET	Aerosol Deposition and Resuspension in STORM Test SR 11
ISP 41	CON	RTF Experiment on Iodine Behaviour in Containment Under Severe Accident Conditions
ISP 42	INT/ CON	PANDA Test "TEPPS"
ISP 43	SET	Rapid Boron Dilution Test

ISP DESCRIPTIONS

ISP	Date	Host Country	Host Organisation
1	1973	USA	Aerojet Nuclear Company (NRTS); Idaho Falls, USA

Title:

Straight Pipe Depressurization Experiment (Edwards'Pipe)

Objectives:

- check the capabilities of various thermal equilibrium blowdown codes to predict rapid blowdown processes
- comparison of code capabilities

Facility:

- straight pipe, electrically heated, filled with pressurized water; length 4,07 m ; diameter: 72,8 mm
- initial conditions: 1000 psig (7 MPa); 467 F (240 C)
- glass rupture disc; opening time estimated 1ms; effective flow area reduced by approx.13%
- 7 fast-response pressure gauges, located along the pipe
- 7 temperature transducers, located along the pipe; response time 15 ms
- 2 x-ray density sensors

Scaling Information:

basic separate effects test

Parameters offered for Comparison:

- 7 absolute pressures;
- 7 fluid temperatures;

2 local void fraction

ISP	Date	Host Country	Host Organisation
1	1973	USA	Aerojet Nuclear Company (NRTS); Idaho Falls, USA

Dominating Experimental Uncertainties:

- measurements of initial temperature distribution inside pipe unreliable

Findings:

- significance of thermal non-equilibrium conditions for fast depressurization events

Recommendations:

none

Total Duration of Exercise:

January 1973 -October 1973

Participation:

6 institutions from 1 country (USA);
6 lumped parameter codes

CSNI Report:

R.W.Garner;
Comparative Analyses of Standard Problems-STANDARD PROBLEM ISP 1;
(Straight Pipe Depressurization Experiment);
Aerojet Nuclear Company; Interim Report I-212-74-5.1; October 1973

ISP	Date	Host Country	Host Organisation
2	1974/5	USA	Aerojet Nuclear Company, Idaho Falls, USA

Title:

Comparative Analysis of Standard Problems- Standard Problem 2
(Semiscale Test 1011)

Objectives:

Simulation of a Loss of Coolant Accident (LOCA) by the Semiscale Test Facility starting from isothermal conditions of the primary coolant loop

Facility:

- 1 1/2 Loop Semiscale System consisted of a pressure vessel with internals, including 9 electrically heated fuel rod simulators;
- 1 intact loop with steam generator, simulated pump and bypass line
- 1 broken loop simulator
- emergency core cooling injection system

Scaling Information:

- volume to break area equal to volume to break area of a large PWR
- other parameters not scaled, facility should only yield data for comparison to analyses

Parameters offered for Comparison:

- 2 pressures (upper plenum and upstream break nozzle) ; containment pressure
- 4 differential pressures (hot leg-cold leg, across pump, pressure vessel containment)
- 6 local fluid densities
- 7 flow rates and including break flow
- liquid level
- time to start high pressure injection system (HPIS) and low pressure injection system (LPIS)

ISP	Date	Host Country	Host Organisation
2	1974/5	USA	Aerojet Nuclear Company, Idaho Falls, USA

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

mostly related to homogeneity of fluid conditions

Recommendations:

nothing documented

Total Duration of Exercise:

September 1973- January 1975

Participation:

5 institutions from USA only

CSNI Report:

D.J. Barnum;
Comparative Analysis of Standard Problems- Standard Problem 2
(Semiscale Test 1011)
Interim Report ;
Aerojet Nuclear Report I-296-75-1; January 1975, Rev.June 1975

ISP	**Date**	**Host Country**	**Host Organisation**
3	1976	Canada	Atomic Energy of Canada Limited;Pinawa; Canada

Title:

Comparison of LOCA Analysis Codes

Objectives:

Basic experiment of subcooled blowdown with simultaneous heat addition

Facility:

- CISE blowdown test rig
- uniformly heated vertical test section (4000 mm long, 21 mm diameter) with unheated riser section (9995 mm long) and unheated lower feeder section (9845 mm long)
- heated section provided with upstream and downstream valves
- constant heater power 109 KW
- initial pressure 9.9 MPa

Scaling Information:

basic separate effects test

Parameters offered for Comparison:

- pressures of test section at closed and open end
- heater surface temperature
- corresponding fluid temperature
- fluid mass remaining in system

ISP	Date	Host Country	Host Organisation
3	1976	Canada	Atomic Energy of Canada; Limited;Pinawa; Canada

Dominating Experimental Uncertainties:

- instrumentation of test section not sufficient for anticipated purpose of the ISP
- uncertain initial conditions

Findings:

Wide divergence of calculated results attributed to a variety of problems, e.g.
- discharge models
- nodalization of test section
- confusion about correct initial experimental conditions
- participants unable to arrive at common conclusions

Recommendations:

- attempt to find convergent solution by rerunning the problem
- determine sensitivity to empirical content of code calculations
- establish a so-called "benchmark solution of the task"
- repeat activity with agreed specified empirical content to determine impact of chosen numerical solution procedures by comparison to "benchmark solution"

Total Duration of Exercise:

1975/76

Participation:

14 institutions from 14 countries
codes involved: mostly RELAP-4, DANAIDES, RAMA, NAIAD

CSNI Report:

W.T. Hancox, B.H. McDonald;
CSNI STANDARD PROBLEM 3;
Comparison of LOCA Analysis Codes;
Submission to CSNI Ad Hoc Group on Emergency Core Cooling, December 1976;
CSNI Report No.15

ISP	Date	Host Country	Host Organisation
4	1976	USA	EG&G, Idaho Falls, USA

Title:

UNITED STATES STANDARD PROBLEM 4 AND INTERNATIONAL STANDARD PROBLEM 8 **(Simulation of Semiscale MOD 1 Test S-02-6)**

Objectives:

- Simulation of a small-break Loss of Coolant Accident (LOCA) by the modified Semiscale Test Facility including blowdown and refill involving the high and low pressure injection systems

- "blind" pre-test predictions (served also as United States Standard Problem 6)

Facility:

- Semiscale Mod 1 System consisted of a pressure vessel with internals, including a 40 rod core with indirect electrically heated fuel rod simulators ;
- 36 rods at average power, 4 rods at higher power level
- total power 1.56 MW, initial pressure 15.77 MPa, average coolant temperature 590 K
- 1 intact loop with steam generator, simulated pump and bypass line
- 1 broken loop simulator
- high and low pressure coolant injection including accumulator

Scaling Information:

- Volume and power scaling 1/2000 related to commercial PWRs
- Length of fuel rod simulators equivalent to approximately half of commercial PWR fuel rods
- Elevation of steam generators relative to the elevation of the pressure vessel with core simulator preserved

Parameters offered for Comparison:

- 2 pressures (upper plenum and upstream break nozzle)
- 5 differential pressures (cold-hot leg, across pump and pump suction, steam generator, lower-upper plenum)
- 9 local fluid densities and 11 local fluid temperatures
- 9 flow rates and including break flow
- core liquid level
- 4 fuel temperatures of hot rod
- time to start high pressure injection system (HPIS) and low pressure injection system (LPIS)

ISP	Date	Host Country	Host Organisation
4	1976	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

- break flow and accumulator flow in general overpredicted
- significant differences between predictions and measured data for hot rod temperatures
- underprediction of upper plenum pressure
- prediction of accumulator injection rates and subsequent impact on thermal-hydraulic behaviour of primary system flow not satisfactory

Recommendations:

nothing documented

Total Duration of Exercise:

November 1975 - March1978

Participation: (ISP only)

9 institutions from 9 countries;
different versions of thermal equilibrium code RELAP 4 , ALARM-P1 (Japan) and NAIAD (Australia)

CSNI Report:

H.M. Delaney;
UNITED STATES STANDARD PROBLEM 4 AND INTERNATIONAL STANDARD PROBLEM 8;
Final Report ;
CSNI Report No. 50; March 1978

ISP	Date	Host Country	Host Organisation
5	1977	USA	EG&G, Idaho Falls, USA

Title:

UNITED STATES STANDARD PROBLEM 7 AND INTERNATIONAL STANDARD PROBLEM 5 (**Nonnuclear Isothermal LOFT Blowdown Test L1-4)**

Objectives:

- Simulation of a 200% double-ended break Loss of Coolant Accident (LOCA) in the LOFT test facility including blowdown and refill processes with delayed high pressure injection system (HPIS)
- "blind" pre-test predictions (served also as United States Standard Problem 7)

Facility:

- LOFT loss of coolant test facility consisted of a pressure vessel with a core simulator to simulate the pressure drop of a core;
- a pressure suppression containment system
- 1 intact loop representing three intact loops of a four loop PWR, 2 parallel pumps and a pressurizer line
- 1 broken loop simulator with simulated steam generator, pump and blowdown valves
- high and low pressure coolant injection including accumulator
- primary coolant volume 7.71 m^3, initial pressure 15.73 MPa, initial temperature 552.2 K

Scaling Information:

- Volume and flow area scaling 1/50 related to commercial PWRs
- core simulator with flow restricting orifices representative for nuclear core
- heat-up of the pre-pressurized system by energy addition from running recirculation pumps

Parameters offered for Comparison:

- 6 pressures (hot and cold leg of the intact loop, upstream break nozzles of the broken loop)
- 9 differential pressures (cold-hot leg, across pump and pump suction, steam generator, lower-upper plenum)
- 5 local fluid densities and 14 local fluid temperatures
- 3 volumetric flow rates including break flow
- 4 liquid levels (reactor vessel, pressurizer, containment suppression tank)
- 2 pump speeds

ISP	Date	Host Country	Host Organisation
5	1977	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

- integrated measured break flow approx. 1,39 times initial water mass inventory leading to correction factor 0,72 of all flow rate measurements

Findings:

- accumulator modelling proven as inadequate to correctly describe accumulator performance
- codes accounting for nonequilibrium eliminated saturation line crossing (oscillations) and ECC mixing problems encountered using homogeneous equilibrium concepts (observation derived from numerical post-test analyses results)
- sub-partition of circumferential downcomer improving ECC penetration in downcomer region
- critical flow models in equilibrium codes require calibration for accurate break flow prediction

Recommendations:

- error analyses considered necessary to quantify integral effects of model and system uncertainties inherent to the experiment and analytical predictions
- post-test analyses recommended

Total Duration of Exercise:

May 1977 -April 1978

Participation: (ISP only)

8 institutions from 8 countries;
different versions of RELAP 4, BRUCH -D-06 and DRUFAN (Germany), ALARM-P1 (Japan)

CSNI Report:

B.L. Hansen;
UNITED STATES STANDARD PROBLEM 7 AND INTERNATIONAL STANDARD PROBLEM 5;
Final Report ;
CSNI Report No. 29; January 1979

ISP	**Date**	**Host Country**	**Host Organisation**
6	1978	Germany	Gesellschaft für Anlagen-und Reaktorsicherheit (GRS), Germany

Title:

Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase

Objectives:

Comparison of calculations with measurements of processes important for anticipated steam-line ruptures of Boiling Water Reactors (BWR)

- mass discharge rates
- mixture level transient
- pressure and temperature behaviour of the water and vapour phase
- "open" post-test analyses

Facility:

Blowdown test facility at the Battelle Institute - Frankfurt, Germany

- Cylindrical pressure vessel; 11.9 m height, 5.2 m^3 volume, electrical heater elements, no other steam dome internals
- initial water level approximately 2/3 of total elevation
- initial pressure 7.11 MPa, initial temperatures in the water phase between 285.5 ^oC and 289.5 ^oC

Scaling Information:

no specific considerations given to typical separate effects test

Parameters offered for Comparison:

- absolute pressures at 6 distinct locations
- temperatures at 6 corresponding locations
- mixture level
- mass discharge flow rate

ISP	Date	Host Country	Host Organisation
6	1978	Germany	Gesellschaft für Anlagen-und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

Uncertainties with respect to the spatial distribution of initial temperatures in the water phase burden the interpretation of the results with thermal non-equilibrium codes

Findings:

- Calculations performed with thermal equilibrium codes indicate their limits
- different results obtained with same code indicate the importance of code user effects
- proper selection of "discharge factors" important for agreement between calculations and measurements, in particular for two-phase flow release

Recommendations:

- development of non-equilibrium models to simulate fast depressurization and wave-propagation transients
- more post-test analyses to investigate into effects of bubble size parameters, slip ratios, initial liquid subcooling
- interrelations between time step size and nodalization should be studied

Total Duration of Exercise:

December 1976 to August 1978

Participation:

10 countries; 12 institutions applying
various versions of thermal equilibrium codes RELAP-4, BRUCH-S, and ALARM-B
thermal nonequilibrium codes NORA, DRUFAN, FROTH2

CSNI Report:

W. Winkler;
COMPARISON REPORT ON OECD-CSNI STANDARD PROBLEM No. 6;
Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase;
CSNI Report No. 30, August 1978

ISP	Date	Host Country	Host Organisation
7	1979	France	Commissariat à l’ Energie Atomique; IPSN; Département de Sûreté Nucléaire; Grenoble, France

Title:

Analysis of a Reflooding Experiment

Objectives:

- Separate Effects Test investigating into the reflooding phase of a Loss of Coolant Accident
- "blind" pre-test predictions

Facility:

- uniformly heated and internally cooled single tube installed in the ERSEC loop in Grenoble, France
- tube placed in a vacuum chamber with reflecting shield to reduce external heat losses
- constant operating conditions (flooding rate, subcooling of water and power generation)
- intial conditions: Pressure 0.3 MPa; Flooding rate 52 kg/m^2 s; inlet water subcooling 23 K; Power 6.2 KW

Scaling Information:

- analytical experiment, two-phase thermal-hydraulic oriented exercise
- 0.3 MPa total pressure, assumed as constant

Parameters offered for Comparison:

- wall temperatures of tube at 4 distinct elevations versus time
- quench time at different elevations
- water entrainment at tube exit versus time
- steam temperature at tube exit versus time

ISP	Date	Host Country	Host Organisation
7	1979	France	Commissariat à l' Energie Atomique; IPSN; Département de Sûreté Nucléaire; Grenoble, France

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

- experiment characterized by strong non-equilibrium effects(e.g. subcooled boiling and void formation)
- thermal equilibrium codes could not predict the measurements in all respects
- RELAP4-MOD 6 empiricism derived from bundle experiments not applicable for this internally cooled monotube experiment
- ISP 7 results not representative for capabilities to predict nuclear plant system behaviour

Recommendations:

nothing derived from this exercise

Total Duration of Exercise:

1978- June 1979

Participation:

9 institutions from 8 countries (pre-test predictions)
4 institutions from 3 countries (post-test analyses)

CSNI Report:

R. Deruaz, N. Tellier;
Comparison Report on OECD Standard Problem No. 7
Analyses of a Reflooding Experiment ;
CSNI Report No. 55; June 1979

ISP	Date	Host Country	Host Organisation
8	1979	USA	EG&G, Idaho Falls, USA

Title:

Semiscale MOD 1; Test S-06-03 (LOFT Counterpart Test)

Objectives:

- Counterpart test to LOFT nuclear test L2-3, covering the blowdown, refill and reflood phase of a simulated Loss of Coolant Accident
- "blind" pre-test predictions

Facility:

- Semiscale Mod 1 System consisted of a pressure vessel with internals, including a 40 rod core with 36 electrically heated fuel rod simulators and 4 unpowered rods
  - total power 1.004 MW, initial pressure 15.77 MPa, average coolant temperature 598 K
  - 1 intact loop with steam generator, simulated pump and bypass line
  - 1 broken loop simulator
  - high and low pressure coolant injection including accumulator

Scaling Information:

- Volume and power scaling 1/2000 related to commercial PWRs
- Full elevation length of fuel rod simulators
- Elevation of steam generators relative to the elevation of the pressure vessel with core simulator preserved
- increased steam generator resistance to compare with LOFT reactor situation

Parameters offered for Comparison:

- 7 pressures (upper plenum and accumulator, pressurizer, steam generator etc.)
- upper plenum and accumulator core inlet flow
- accumulator flow
- 2 break flows and break flow densities
- intact loop flow and density
- 9 fuel rod temperatures

ISP	Date	Host Country	Host Organisation
8	1979	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

- two-phase flow orifice of break nozzle not calibrated
- considerable error bands of intact loop flow data

Findings:

- pressure predictions in general compare well with measured data
- break flow and accumulator flow in general overpredicted
- temperature predictions of heater rods burdened by unprecise prediction of onset of critical heat flux (CHF), but most calculations overpredicted the temperature evolution in the upper portion of the core simulator
- unrealistic early rewetting by most predictions due to positive core flow

Recommendations:

nothing documented

Total Duration of Exercise:

February 1978 - June 1979

Participation:

9 institutions from 8 countries;
different versions of thermal equilibrium code RELAP 4, non-equlibrium code DRUFAN 01

CSNI Report:

J.L. LaChance;
INTERNATIONAL STANDARD PROBLEM 8;
Semiscale MOD 1; Test S-06-03 (LOFT Counterpart Test)
Preliminary Report ;
CSNI Report No. 38

ISP	Date	Host Country	Host Organisation
CASP 1	1979	Germany	Gesellschaft für Anlagen-und Reaktorsicherheit (GRS), Germany

Title:

Steamline Rupture within a Chain of Compartments (Battelle Test D15)

Objectives:

- Best-estimate post-test calculations of thermal-hydraulic containment parameters resulting from an integral steamline rupture test performed in a simplified compartment arrangement within the BATTELLE Frankfurt containment test facility
- Subdivision of analyses for three typical event phases (short term, medium range, long term)
- "Open" post test analyses

Facility:

- BATTELLE model containment consists of a concrete building with a free volume of 626 m^3, subdivided into several (up to 9) empty sub-compartments
- interconnections by vent openings of variable cross sections
- total vertical extension of internal free volume approximately10 m
- inner and outer walls reinforced concrete, limited leak tightness under elevated pressures
- maximum design pressure ca. 0.5 MPa
- BATTELLE high pressure test rig (see also ISP 06) served as energy reservoir allowing 1/64 volume-scaled mass- and energy source rate into the model containment

Steam-line blowdown test D15 was performed in a chain-type arrangement of six containment compartments

Scaling Information:

- elevation scaling approximately 1/5 related to commercial PWR containments
- volume reduction related to commercial PWR containments approximately 1/100
- internal surface/volume ratio approx. 2.4 (as compared to approximately 0.9 surface/volume ratio of commercial PWR containments)

Parameters offered for Comparison:

- 13 pressure differences between different compartments (up to 2,5 s)
- 12 abs.pressures in compartments involved (up to 50 s)
- 10 fluid temperatures within compartments (up to 50 s)
- 1 global pressure (up to 1500 s)
- 1 average fluid temperature (up to 1500 s)
- 1 separated water mass (up to 1500 s)

ISP	**Date**	**Host Country**	**Host Organisation**
CASP 1	1979	Germany	Gesellschaft für Anlagen-und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

relatively large error bands attributed to input functions provided as mass and energy release rates

Findings:

- results of "open" calculations show margins comparable to margins obtained in the frame of earier "blind" national standard problem (e.g. short term pressure differences and peak pressures)
- pressure loss coefficients and water carryover between adjacent compartments identified as sensitive parameters for short term pressurisation
- sensitivity to nodalization experienced
- medium and long term transient pressure calculations influenced by modelling energy exchange (heat transfer) between structures and containment fluid

Recommendations:

further containment standard problems with improved instrumentation recommended to avoid premature conclusions about code capabilities

Total Duration of Exercise:

May 1978-September 1979

Participation:

12 institutions from 11 countries;
codes involved: ZOCO-V, RELAP 4-MOD4/5, COFLOW, CONDRU, GRUYER, PACO, BEACON-MOD3

CSNI Report:

W. Winkler;
Draft Comparison Report on OECD-CSNI Containment Standard Problem No.1;
Steamline Rupture within a Chain of Compartments (Battelle Test D15)
CSNI Report No. 41; September 1979

ISP	Date	Host Country	Host Organisation
9	1980	USA	EG&G, Idaho Falls, USA

Title:

INTERNATIONAL STANDARD PROBLEM 9

LOFT Nuclear Experiment L3-1

Objectives:

- Simulation of a small break Loss of Coolant Accident (LOCA) in the nuclear LOFT test facility including blowdown and refill processes with high pressure injection system (HPIS) in operation

- "blind" pre-test predictions requested

Facility:

- LOFT loss of coolant experiment L3-1 was carried out using a pressure vessel with a nuclear core, consisting of 1300 unpressurized fuel rods 1.67 m long ;
- 1 intact loop representing three intact loops of a four loop PWR, 2 parallel pumps and a pressurizer
- 1 broken loop simulator with simulated steam generator, pump and blowdown valves
- high and low pressure coolant injection including accumulator
- primary coolant volume 7.71 m^3, initial pressure 14.87 MPa, initial cold leg temperature 554 K and hot leg temperature 574 K; initial power level 48.9 MW
- a pressure suppression containment system

Scaling Information:

- Volume and flow area scaling 1/50 related to commercial PWRs
- length of nuclear core equivalent to 1/2 of commercial PWR fuel
- reactor scrammed 2 s before initiation of blowdown process to protect nuclear fuel elements

Parameters offered for Comparison:

- 5 pressures (steam generator, hot and cold leg broken loop, upper plenum, pressurizer, accumulator)
- 3 local fluid densities and 3 local fluid temperatures
- break flow rate
- 4 rod cladding temperatures at different elevations

ISP	Date	Host Country	Host Organisation
9	1980	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

Two leakage paths identified after completion of test L3-1; impact on test results considered to be small based on supporting analyses

Findings:

- primary system depressurisation during initial 300 s predicted reasonably well
- pressure on secondary side of steam generator generally over-predicted during first 700 s
- time to start onset of voiding over-estimated by most analyses
- significant differences in predicted critical mass flow at break observed, although pressure transient was in reasonable agreement
- significant scatter of secondary side pressure predictions
- significant scatter of local density predictions

Recommendations:

no particular recommendations reported

Total Duration of Exercise:

1980-1981

Participation:

11 institutions from 9 countries;
different versions of RELAP 4, RELAP 5, THYDE-PD (Japan), TRAC -PD2

CSNI Report:

A.C. Peterson, C. Polk, J.A. Sellars;
INTERNATIONAL STANDARD PROBLEM 9(LOFT Test L3-1);
Preliminary Comparison Report ;
CSNI Report No. 66; April 1981

ISP	**Date**	**Host Country**	**Host Organisation**
10	1980	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Refill and Reflood Experiment in a Simulated PWR Primary System (PKL)

Objectives:

PKL Test K9 served to investigate into gravity fed refill- and reflood -process within a pressure vessel connected to primary system loops simulating a PWR system

Post-test analyses

Facility:

- PKL facility consisted of a pressure vessel with external downcomer simulator
- 3 primary system loop simulators: 1 broken loop (simple capacity), 1 intact loop (double capacity), 1 intact loop (simple capacity)
- loops equipped with U-tube steam generators
- pumps simulated by appropriate resistances

containment simulator to maintain correct post-blowdown operating pressure

Scaling Information:

- Volume and power scaling 1/134 related to commercial PWRs
- Full elevation length of fuel rod simulators
- Elevation of Steam generators relative to the pressure vessel with core preserved
- cross sections and lengths of loops designed to preserve nominal pressure losses
- break location representative for cold leg break

Parameters offered for Comparison:

- cladding tube temperatures and heat transfer coefficients for average rod at 7 elevations
- 35 cladding tube temperatures and heat transfer coefficients for 5 specified rods in the bundle
- quench front propagation in pressure vessel
- quench front propagation at 5 specified fuel rods
- collapsed liquid levels in pressure vessel and downcomer
- 2 differential pressures(downcomer - break location, upper plenum - break location)
- mass flow rates and void fractions inside loops and downcomer tube

ISP	Date	Host Country	Host Organisation
10	1980	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

no particular observations reported

Findings:

- refill/reflood process characterized by large scatter of experimental data from rod to rod
- overall fluid dynamics of refill/reflood process predicted satisfactorily with trends to oscillations similar to the experimental observations
- maximal turnaround temperatures slightly underpredicted
- shortcomings in the analytical models observed for quench front behaviour, droplet formation and local reflood heat transfer processes

Recommendations:

Additional separate effects tests with improved instrumentation considered useful to investigate the above observed analytical shortcomings

Total Duration of Exercise:

December 1979- May 1981

Participation:

- 5 institutions from 5 countries;
- different versions of thermal equilibrium code RELAP 4/MOD 6 and 7, REFLOS

CSNI Report:

D.L. Nguyen, W. Winkler;
COMPARISON REPORT ON INTERNATIONAL STANDARD PROBLEM 10;
Refill and Reflood Experiment in a Simulated PWR Primary System (PKL)
Final Report ;
CSNI Report No. 64, September 1981

ISP	Date	Host Country	Host Organisation
11	1981	USA	EG&G, Idaho Falls, USA

Title:

INTERNATIONAL STANDARD PROBLEM 11

LOFT Nuclear Experiment L3-6/L8-1

Objectives:

- Simulation of a small break Loss of Coolant Accident (LOCA) in the nuclear LOFT test facility including blowdown and refill processes with high pressure injection system (HPIS) with ECCS flow directly into the downcomer region
- "Open" post-test analyses requested

Facility:

- LOFT loss of coolant experiment L3-6 was carried out using a pressure vessel with a nuclear core, consisting of 1300 unpressurized fuel rods 1.67 m long ;
  - 1 intact loop representing three intact loops of a four loop PWR, 2 parallel pumps and a pressurizer
  - 1 broken loop simulator with simulated steam generator, pump and blowdown valves
  - high and low pressure coolant injection including accumulator
  - primary coolant volume 7.71 m^3, initial pressure 14.87 MPa, initial cold leg temperature 552.2 K and hot leg temperature 577.1 K; initial power level 50MW
  - a pressure suppression containment system

Scaling Information:

- Volume and flow area scaling 1/50 related to commercial PWRs
- length of nuclear core equivalent to 1/2 of commercial PWR fuel
- reactor scrammed 5.8 s before initiation of blowdown process to protect nuclear fuel elements

Parameters offered for Comparison:

- 5 pressures (hot leg of the intact loop, hot and cold leg broken loop, upper plenum, pressurizer)
- steam generator pressure
- 2 differential pressures (across intact loop steam generator, intact loop pumps)
- 6 local fluid densities and 10 local fluid temperatures
- 2 mass flow rates (HPIS and break flow)
- 4 rod cladding temperatures at different elevations
- integral energy discharge to containment suppression tank

ISP	Date	Host Country	Host Organisation
11	1981	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

nothing reported

Findings:

- primary system depressurisation calculated accurately by most submissions
- pressure on secondary side of steam generator not properly calculated
- duration of subcooled blowdown overestimated by most analyses
- a variety of multipliers associated to critical flow models applied
- time to quench rods in good agreement

Recommendations:

no particular recommendations reported

Total Duration of Exercise:

1981-1982

Participation:

14 institutions from 9 countries;
different versions of RELAP 4 and RELAP 5, DRUFAN (Germany) and THYDE-P (Japan), SATAN-VI, RETRAN, TRAC-PD2

CSNI Report:

A.C. Peterson, C. Cook;
INTERNATIONAL STANDARD PROBLEM 11(LOFT Experiment L3-6/L8-1);
Preliminary Comparison Report ;
CSNI Report No. 73; April 1982

ISP	Date	Host Country	Host Organisation
CASP 2	1981	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Water Line Rupture into a Branched Compartment Chain (Battelle Test D16)

Objectives:

- Best-estimate post-test calculations of thermal-hydraulic containment parameters resulting from a test simulating the blowdown of a pressurized water reservoir into a simplified compartment configuration performed within the BATTELLE Frankfurt containment test facility
- Subdivision of analyses for three typical event phases (short term, medium range, long term)
- "Open" post test analyses

Facility:

- BATTELLE model containment consists of a concrete building with a free volume of 626 m^3, subdivided into several (up to 9) empty sub-compartments
- interconnections by vent openings of variable cross sections
- total vertical extension of internal free volume approximately 10 m
- inner and outer walls reinforced concrete, limited leak tightness under elevated pressures
- maximum design pressure ca. 0.5 MPa
- BATTELLE high pressure test rig (see also ISP 06) served as energy reservoir allowing 1/64 volume-scaled mass and energy source rate into the model containment

Test D16 with discharge of pressurized water was performed in a branched arrangement of six typical containment compartments

Scaling Information:

- elevation scaling approximately 1/5 related to commercial PWR containments
- volume reduction related to commercial PWR containments approximately 1/100
- internal surface/volume ratio approximately 2.4 (as compared to surface/volume ratio approximately 0.9 of commercial PWR containments)

Parameters offered for Comparison:

- 7 pressure differences between different compartments (up to 2.5 s)
- 14 local fluid temperatures (up to 2.5 s)
- 4 absolute pressures in compartments involved (up to 50 s)
- 7 fluid temperatures within compartments (up to 50 s)
- 1 global pressure (up to 1500 s)
- 2 averaged fluid temperatures (up to 1500 s)
- accumulated water mass (up to 1500 s)

ISP	Date	Host Country	Host Organisation
CASP 2	1981	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

inhomogeous initial temperature distribution along the blowdown discharge line reducing enthalpy of initially released pressurized water

Findings:

- results of "open" calculations submitted for CASP 2 show margins comparable to margins obtained in the frame of an earlier "blind" national standard problem based on the same experiment (e.g. for the simulation of short term pressure differences and peak pressures)
- pressure loss coefficients and water carryover between adjacent compartments identified as sensitive parameters for short term pressurization effects
- medium and long term transient pressure calculations influenced by modelling energy exchange (heat transfer) between structures and containment fluid
- sensitivity to nodalization of structures experienced, in particular for the simulation of local heat transfer to structures

Recommendations:

further containment standard problem based on a water blowdown test in the large-scale HDR-facility with improved instrumentation recommended to avoid premature conclusions about code capabilities

Total Duration of Exercise:

April 1980-May 1982

Participation:

12 institutions from 11 countries;
codes involved: ZOCO-V, RELAP 4-MOD5, COFLOW, CONDRU, GRUYER, ARIANNA-0, PRESCON, COPTA-6, PACO, TRAP-SCO, TRAP-CON, CLAPTRAP

CSNI Report:

D.L. Nguyen, W. Winkler;
Comparison Report on OECD-CSNI Containment Standard Problem No. 2;
Water Line Rupture into a Branched Compartment Chain (Battelle Test D16)
CSNI Report No. 65; May 1982

ISP	Date	Host Country	Host Organisation
CASP 3	1982	Australia	Australian Atomic Energy Commission, Lucas Heights Research Laboratories

Title:

Small-Scale Two-Compartments Basic Containment Experiment

Objectives:

- Best-estimate post-test calculations of pressure, temperature and heat transfer coefficients resulting from a two-phase blowdown test performed in a two-compartments arrangement of the Australian Lucas Heights blowdown/containment test rig
- "Open" post test analyses have been requested

Facility:

- Lucas Heights blowdown/containment test rig consists of a subdivided steel vessel with a free volume of approximately 1.8 m^3
- total elevation of internal free volume approximately 2.4 m
- all thin steel walls without insulation

Scaling Information:

- internal surface/volume ratio approximately 6.1 (as compared to commercial PWR containments approximately 0.9)

Parameters offered for Comparison:

- 2 absolute pressures
- 1 pressure difference
- 2 fluid temperatures within compartments
- 1 wall temperature
- 2 heat transfer coefficients

ISP	Date	Host Country	Host Organisation
CASP 3	1982	Australia	Australian Atomic Energy Commission, Lucas Heights Research Laboratories

Dominating Experimental Uncertainties:

nothing reported

Findings:

- split opinions about the merits of the exercise
- more detailed heat transfer measurements would be necessary to generate "mean values" of heat transfer coefficients applicable to present thermal-hydraulic modelling capabilities
- considerable heat transfer through thin-walled separation plate between both parts of the containment vessel affecting pressure transients

Recommendations:

- more sophisticated separate effects test to study local heat transfer conditions

Total Duration of Exercise:

January 1982-October 1982

Participation:

8 institutions from 8 countries;
codes involved: ZOCO, RELAP 4, COBRA, GRUYER, ARIANNA, CLAPTRAP, COMPARE

CSNI Report:

J. Marshall, W. Woodman;
Comparison Report on Containment Analysis Standard Problem No. 3;
CSNI Report No. 77; April 1983

ISP	Date	Host Country	Host Organisation
12	1982	Japan	Japan Atomic Energy Research Institute (JAERI); Tokai-mura; Japan

Title:

ROSA-III 5% Small Break Test, Run 912

Objectives:

- Simulation of thermal-hydraulic conditions representative for a 5% split break in the pump suction recirculation line of a Boiling Water Reactor (BWR)
- Failure of the high pressure core spray system assumed, steam line and feedwater line uncoupled from the primary coolant system
- Post-test analyses requested (open problem)

Facility:

- Pressure vessel with steam dome, upper plenum and steam separators, feed water and steam line, external recirculation system and emergency core cooling system (ECCS), automatic depressurization system (ADS)
- electrically heated test section, 62 rods
- pressure vessel internal downcomer, 2 jet pump recirculation loops
- Initial power level of test section 4.4 MW
- initial steam dome pressure 7.23 MPa

Scaling Information:

- power scaling 1/864 corresponding to BWR-6 type Boiling Water Reactors
- volume scaling 1/437, all nominal flow rates scaled 1/424
- rod bundles half length of commercial BWR-6 fuel

Parameters offered for Comparison:

- 3 absolute pressures (steam dome, lower plenum, upstream break)
- 11 specified differential pressures
- 15 mass flow rate measurements (core inlet, high and average power channel inlet, both recirculation loops, steam relief valve)
- 8 volumetric flow rates
- 2 momentum flux measurements
- 7 mixture levels
- 4 cladding temperatures of heater rods at specified elevations
- 7 fluid temperatures

ISP	Date	Host Country	Host Organisation
12	1982	Japan	Japan Atomic Energy Research Institute (JAERI), Tokai-mura; Japan

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

- pressure transient calculated with reasonable accuracy until actuation of automatic depressurisation system (ADS)
- large uncertainties in break flow simulations
- strong correlation between heater rod surface temperatures and mixture level calculations in the core region, mostly associated to the correct simulation of recirculation pump efficiency
- some problems in analyzing clad temperatures, dryout and quench front propagation

Recommendations:

- improvements of break flow simulation and other two-phase flow models in codes
- improvement of simulation of the heat transfer mode after dryout

Total Duration of Exercise:

July 1981-April 1982

Participation:

8 institutions from 5 countries submitted calculations
Codes involved: RELAP4 MOD 6, RELAP5 MOD1, THYDE-B1, TRAC-BD1

CSNI Report:

K. Tasaka et al;
Comparison Report for CSNI International Standard Problem 12
ROSA-III 5% Small Break Test, Run 912
JAERI-M82-120; CSNI Report No.100; September 1982

ISP	Date	Host Country	Host Organisation
13	1983	USA	EG&G, Idaho Falls, USA

Title:

INTERNATIONAL STANDARD PROBLEM 13

LOFT Nuclear Experiment L2-5

Objectives:

- Simulation of a Loss of Coolant Accident (LOCA) caused by a double-ended, off-shear guillotine cold leg rupture coupled with a loss of off-site power in the nuclear LOFT test facility. Delayed initiation of the high pressure injection system (HPIS) and the low pressure (LPIS) emergency core cooling system.

- Test preceded by 40 hours effective full power to build up a fission product decay heat inventory.

- "Blind" pre-test analyses requested, but late submissions as "open" post -test analyses processed also

Facility:

- LOFT loss of coolant experiment L2-5 was carried out using a pressure vessel with a nuclear core, consisting of 1300 unpressurized fuel rods 1.67 m long ;
- 1 intact loop representing three intact loops of a four loop PWR, 2 parallel pumps and a pressurizer
- 1 broken loop simulator with simulated steam generator, pump and blowdown valves
- high and low pressure coolant injection including accumulator
- primary coolant volume 7.71 m^3, initial pressure 14.94 MPa, initial intact loop hot leg temperature 589 K; initial power level 36 MWth
- a pressure suppression containment system

Scaling Information:

- Volume and flow area scaling 1/50 related to commercial PWRs
- length of nuclear core equivalent to 1/2 of commercial PWR fuel
- reactor scrammed automatically upon initiation of blowdown process

Parameters offered for Comparison:

- 5 pressures (hot leg of the intact loop, hot and cold leg broken loop, upper plenum, pressurizer)
- steam generator pressure
- 4 local fluid densities and 6 local fluid temperatures
- 5 mass flow rates (LPIS, HPIS, core flow and break flow)
- pump speed
- 2 rod cladding temperatures at 0.76 m and 0.99 m elevation
- integrated breakflow
- reactor vessel inventory

ISP	**Date**	**Host Country**	**Host Organisation**
13	1983	USA	EG&G, Idaho Falls, USA

Dominating Experimental Uncertainties:

nothing reported

Findings:

- primary system hydraulic behaviour calculated satisfactorily by most submissions, but fluid temperatures and pressure generally lower than data
- subcooling and superheat histories of primary system not well predicted
- hot leg densities adequately predicted, but large scatter for cold leg densities
- break flow overpredicted by most participants with particular problems experienced with broken loop hot-leg
- model dependent fuel rod temperature profiles with heat up rates well but rod quench not predicted
- similar findings for processed "open" post test contributions

Recommendations:

review of preliminary comparisons and commenting of the particular calculations recommended

Total Duration of Exercise:

1982-1983

Participation:

- 11 institutions from 9 countries;
- different versions of RELAP 4 and RELAP 5, DRUFAN (Germany) and THYDE-P1 and PD2 (Japan), TRAC-PD2

CSNI Report:

J.D. Burtt, S.A. Crowton;
INTERNATIONAL STANDARD PROBLEM 13 (LOFT Experiment L2-5);
Preliminary Comparison Report ;
EGG-NTAP-6276; April 1983 --- (CSNI Report No. 101)

ISP	Date	Host Country	Host Organisation
14	1984	Germany	Inst. for Reactor Dynamics and Reactor Safety, Techn. Univ. Munich, Germany

Title:

Behaviour of a Fuel Bundle Simulator during a Specified Heatup and Flooding Period (REBEKA Experiment)- Results of Post-Test Analyses

Objectives:

Investigation into the non-steady material behaviour of a bundle of electrically heated fuel rod simulators with respect to

- local fuel temperatures,
- cladding strain,
- time to burst and
- local strain at location of burst

Inlet coolant flow conditions and local heat transfer coefficients along the simulator bundle have been provided to participants in "open" post-test exercise to decouple analytical tasks for simulating cooling conditions and local fuel rod material behaviour

Facility:

REBEKA test facility operated at Kernforschungszentrum Karlsruhe Research Center

- 48 rod fuel simulator bundle with cosine shaped power distribution
- internal INCONEL heater rods
- standard Zircaloy-cladding
- pressurized and unpressurized fuel rod simulators
- controlled cooling conditions at the bundle inlet (blowdown, refill and reflood period)

Scaling Information:

- cooling conditions scaled proportional to number of fuel rod simulators
- dimensions of rods and cladding material identical to PWR

Parameters offered for Comparison:

- 4 rods out of 48 rods to be analyzed
- 43 cladding temperatures concentrated at expected location of cladding failures at 4 rods
- axial distribution of fluid temperatures
- internal rod pressures
- location of failure with radial deformation profile

ISP	Date	Host Country	Host Organisation
14	1984	Germany	Inst. for Reactor Dynamics and Reactor Safety, Techn. Univ. Munich; Germany

Dominating Experimental Uncertainties:

none reported

Findings:

- decoupling of thermal-hydraulic cooling predictions from fuel material behaviour predictions not fully successful
- simulation of material behaviour of fuel rod cladding requires high accuracy in local temperature predictions at anticipated location of deformations
- applied fluid dynamic codes do not satisfy above mentioned accuracy requirement
- material behaviour in bundle experiments not directly comparable to predictions or experiments performed with single rods under well defined thermal-hydraulic conditions
- local influence of thermocouple mounting on resulting local deformation important

Recommendations:

- Improvement of common understanding of important interacting processes for transient fuel material predictions under LOCA conditions highly desirable
- at least one more ISP in this field recommended
- further attention to proper use of codes and/or code options addressed
- improvement of generally applicable cladding rupture criteria

Total Duration of Exercise:

September 1983 to September 1984 ("open" exercise)

Participation:

5 institutions from 5 different countries

CSNI Report:

H. Karwat;
INTERNATIONAL STANDARD PROBLEM ISP 14;
Behaviour of a Fuel Bundle Simulator during a Specified Heatup and Flooding Period (REBEKA Experiment)-Results of Post-Test Analyses;
Final Comparison Report;
CSNI Report No. 98; February 1985

ISP	Date	Host Country	Host Organisation
15	1984	Sweden	Studsvik Energiteknik AB; Studsvik; Sweden

Title:

LOCA Experiment in the Swedish FIX-II Facility Related to BWRs

Objectives:

- Simulation of conditions representative for an intermediate size split break in a recirculation line of a Boiling Water Reactor (BWR) with external recirculation lines.
- Specific emphasis was put on heat transfer between rod cladding and coolant
- "Blind" pre-test calculations requested (double-blind)

Facility:

- Pressure vessel with steam dome and mock-up for steam separation including lower plenum section
- 36 rod test section
- external downcomer and bypass tubes with recirculation pump
- initial power level of test section 3.38 MW
- initial steam dome pressure 7MPa

Scaling Information:

- power and volume scaling 1/777 corresponding to Swedish BWR Oskarshamn-2
- full length rod bundles

Parameters offered for Comparison:

- 3 absolute pressures (steam condenser, lower plenum, upstream break)
- 8 specified differential pressures
- 10 fluid temperatures at various locations
- 5 flow rate measurements (core inlet, bypass inlet, both recirculation pumps, steam relief valve)
- 12 cladding temperatures of heater rods at specified elevations

ISP	Date	Host Country	Host Organisation
15	1984	Sweden	Studsvik Energiteknik AB; Studsvik; Sweden

Dominating Experimental Uncertainties:

some errors in data on initial conditions have been under discussion, but remain unresolved

Findings:

- time-wise sequence of events well predicted
- uncertainties in predicted rewet and dryout phenomena experienced
- differences between predictions and measured data attributed to incorrectly used initial conditions, to shortcomings of critical flow modeling and in relating local heat transfer conditions to core mass distribution
- duration of DNB heat transfer regime in general overpredicted
- several post-test calculations performed and submitted to improve comparisons between calculations and measured data by correcting input errors, changing break flow multiplier, modification of nodalizations, etc.

Recommendations:

investigation into the atypicalities of data originating from scaled facilities

Total Duration of Exercise:

March 1983-June 1984

Participation:

7 institutions from 7 countries (blind pre-test predictions)
codes involved: RELAP4 and RELAP5 versions, THYDE-P1, GOBLIN, TRAC-BD1

CSNI Report:

O. Sandervag, B. Kjellen;
ISP 15, Final Comparison Report;
Report Studsvik/NR-84/430; 25.6.1984
CSNI Report No. 102

ISP	Date	Host Country	Host Organisation
16	1983-1985	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Rupture of a Steam Line within in the HDR Containment Leading to an Early Two-Phase Flow

Objectives:

Experiment V44 selected for ISP16 served to better understand the containment pressurisation events expected to happen in typical LOCA events of real nuclear power plants. Specific attention was devoted to

- dynamic and quasi-static pressure loading conditions
- containment internal temperature distributions
- heat transfer to structures during the short and long-term ranges
- transfer of water droplets between adjacent compartments
- local steam /air composition
- generation of natural convection patterns inside containment after termination of blowdown process

Facility:

HDR-containment test facility

- Containment of cylindrical shape, lower section highly substructured into 70 subcompartments; large-volume upper dome section.
- Total free volume: 11,000 m^3; height: 50 m
- original primary system pressure vessel serving as representative energy reservoir for blowdown process

Scaling Information:

- large scale test facility, cylindrical shaped steel shell containment, typicality of compartment arrangements; geometrical similarity not preserved;
- volume scaling compared to full-size PWR approximately 1/6
- energy release rates scaled to preserve power/volume ratio expected for full size PWR;
- time preserving of mass and energy release rates resulting in preservation of typical pressurisation transients
- "as measured" mass and energy-release rates cross-checked by supplementary blowdown calculations

Parameters offered for Comparison:

short-term range: 5 local compartment absolute pressures; 7 pressure differences between compartments; 7 local compartment temperatures:
medium-term range: 4 absolute pressures; 5 local temperatures; 1 local heat transfer coefficient, 1 local heat flux measurement;
long-term range: 1 absolute pressure; 2 local temperatures.

ISP	Date	Host Country	Host Organisation
16	1983-1985	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

- uncertainties in the communicated information about the "as measured" mass and energy discharge rates, due to obvious failure of some sensors in the blowdown line

Findings:

- ISP16, as based on experiment HDR-CONT-V44 considered as an important step towards code application to full-size plants
- results of the "open" ISP16 exercise considered as "satisfactory"
- limits of application of lumped parameter codes to simulate local effects realized

Recommendations:

nothing mentioned explicitly

Total Duration of Exercise:

November 1983-June 1985

Participation:

10 institutions from 7 countries;
10 different codes involved: CONTEMPT-4 (3*), GRUYER, ZOCO-V, COFLOW, CONDRU, COBRA-NC, TRAP, CLAPTRAP II, ARIANNA-2;

CSNI Report:

M. Firnhaber;
INTERNATIONAL STANDARD PROBLEM ISP 16;
Rupture of a Steam Line within in the HDR Containment Leading to an Early Two-Phase Flow, Results of Post-Test Analyses;
Final Comparison Report ;
CSNI Report No. 112; June 1985

ISP	Date	Host Country	Host Organisation
17	1983	Sweden	Studsvik Energiteknik AB; Studsvik Sweden

Title:

Marviken: Pressure Suppression Containment-Blowdown Experiment No.18

Objectives:

obtain insight into modeling capabilities of containment computer codes with respect to

- vent clearing transient
- steam/air transport through vent system
- medium-time pressure suppression characteristics
- periods of oscillatory condensation and/or chugging not considered

Facility:

Containment with pressure suppression system of the decommissioned MARVIKEN nuclear power plant

- containment involves multi-compartment drywell part (1,970 m^3) connected to the
- wet well (2,149 m^3, 556 m^3 water pool) by channel-type vent system and header
- 58 vent pipes, 5.5 m long, 0.3 m diameter; 30 vent pipes blocked during the experiment
- total vent flow area activated during the experiment 1.95 m^2, initial submergence of vent pipes 2.8 m

Scaling Information:

commercial containment, however no similarities to BWR typical pressure suppression system containments

Parameters offered for Comparison:

5 absolute pressures
5 local pressure differences (e.g., drywell-wetwell)
5 gas phase temperatures
1 water pool temperature (averaged)

ISP	Date	Host Country	Host Organisation
17	1983	Sweden	Studsvik Energiteknik AB; Studsvik Sweden

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

- relative to the accuracy of the measurements, the calculated pressure and pressure differences between drywell and wetwell agreed well with experimental data
- calculated time to clear vents slightly too short
- heat transfer between fluid and structures in the drywell underestimated due to low heat transfer coefficients
- small number of nodes sufficient to meet experimental drywell results
- important issues not addressed by this ISP (e.g. periods of oscillatory condensation and/or chugging)

Recommendations:

nothing mentioned in the report

Total Duration of Exercise:

June 1983-August 1984

Participation:

4 institutions from 4 countries
codes involved: CONTEMPT-LT, ARIANNA, ZOCO-V, COPTA

CSNI Report:

J-E Marklund;
SUMMARY REPORT FOR ISP 17;
International Standard Problem for Containment Codes on Blowdown No. 18 in the Marviken Full Scale Experiments;
STUDSVIK/NR-84/464; 19 September 1984;
CSNI Report No. 103

ISP	Date	Host Country	Host Organisation
18	1984	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Title:

LOBI-MOD 2 Small Break LOCA Experiment A2-81

Objectives:

- Simulation of a 1% small break Loss of Coolant Accident (LOCA) in the nonnuclear LOBI test facility at Ispra with high pressure injection system (HPIS) partially in operation

- "Blind" pre-test predictions requested

Facility:

- LOBI test facility using a pressure vessel with an electrically heated core simulator, consisting of 64 rods;
- 1 intact loop with steam generator (triple capacity) representing three intact loops of a four loop PWR,
- 1 broken loop simulator with steam generator, pump and blowdown valves
- pressurizer either connected to broken or intact loop
- high and low pressure coolant injection including accumulator
- primary coolant volume 0.6 m^3, initial pressure 15.8 MPa, initial power level 5.3 MW
- a pressure suppression containment system to simulate back-pressure effects

Scaling Information:

- Volume and flow area scaling 1/712 related to commercial PWRs
- length of fuel rod simulators equivalent to commercial PWR fuel
- relative elevations of components preserved to properly study natural convection processes

Parameters offered for Comparison:

- primary and secondary system pressures
- fluid densities at various locations
- temperature difference between primary and secondary side of SSG
- liquid levels at various locations
- heater rod temperatures and local fluid temperatures (subcooling) at core simulator inlet
- primary system mass inventory

ISP	Date	Host Country	Host Organisation
18	1984	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Dominating Experimental Uncertainties:

Findings:

- Global system behaviour well predicted in general, with a few exceptions
- some local phenomena not well predicted (e.g. mass and temperature distribution in the primary system, mixing of ECC water with saturated steam and resulting condensation processes)
- common modeling problems identified such as loop seal clearing, stratification in vertical and horizontal components and thermal non-equilibrium (mixing, condensation)
- excessive running computer times
- large user influence obvious from submissions based on the same code version
- need for detailed user guidelines evident

Recommendations:

no particular recommendations reported

Total Duration of Exercise:

November 1983-December 1985

Participation:

25 institutions from 12 countries;
different versions of RELAP 4/MOD 6, RELAP 5/MOD 1 and 2, CATHARE, TRAC -PF1, DRUFAN/MOD2

CSNI Report:

H Städtke;
INTERNATIONAL STANDARD PROBLEM -18;
LOBI-MOD 2 Small Break LOCA Experiment A2-81
Final Comparison Report ;
CSNI Report No. 133; April 1987

ISP	Date	Host Country	Host Organisation
19	1987	France	CEA/CEN-Cadarache, IPSN/DERS/SEAREL, France

Title:

Behaviour of a Fuel Rod Bundle during a Large Break LOCA Transient with a two Peaks Temperature History (PHEBUS Experiment)

Objectives:

Post-test investigation into the response of a nuclear fuel bundle to a large break loss of coolant accident with respect to
- local fuel temperatures,
- cladding strain at the time of burst,
- time to burst and

under given thermal-hydraulic boundary conditions of PHEBUS-test 218

Facility:

PHEBUS test facility operated at CEA Research Center CADARACHE consists of a pressurized circuit involving pumps, heat exhangers and a blowdown tank
- 25 nuclear fuel rod bundle, coupled to a separate driver core
- active length 0.8 m, cosine axial power profile
- pressurized and unpressurized fuel rods
- controlled cooling conditions at the bundle inlet (blowdown, refill and reflood period)
- depressurized test rig volume 0.22 m^3

The following "as measured" boundary conditions (B.C.) were offered to participants as options with decreasing challenge to their analytical approach:
Boundary conditions B.C.0:
- full thermal-hydraulic analysis of PHEBUS test rig (was not recommended!)

Boundary conditions B.C.1:
- thermal power level of fuel bundle
- fluid inlet conditions to bundle section

Boundary conditions B.C.2:
- local cladding temperatures of rods
- heat transfer coefficients

Boundary conditions B.C.3:
- cladding temperatures of rods
- internal pressure of rods

Scaling Information:

nuclear test, but relevant scaling ratios unclear

Parameters offered for Comparison:

- fuel cladding temperatures of four rods
- fuel centerline temperatures of four rods
- hoop strain of two rods
- internal gas pressure of two rods
- rupture time and rupture strain for two rods

ISP	Date	Host Country	Host Organisation
19	1987	France	CEA/CEN-Cadarache, IPSN/DERS/SEAREL, France

Dominating Experimental Uncertainties:

uncertainties of testing in a nuclear environment were reflected by the specified options for analytical boundary conditions (B.C)

Findings:

Exercise showed the complexity of the prediction of fuel rod behaviour at high temperatures after a loss of coolant accident. Items for concluding discussions were:

- the effects of specifying the proper task and boundary conditions
- the code user effects
- accuracy of cladding temperature measurements
- the effects of the actual fuel rod geometry (e.g. gap width, gap conductance, internal fuel relocation)
- the correct specification of local heat transfer coefficients
- the limited achievable accuracy of local heat exchange between coolant and fuel (e.g. in connection with B.C.1)
- the effects of azimuthal temperature distribution on rupture strain
- selection of appropriate material properties (e.g. creep law, rupture criterion)

Recommendations:

- to perform another exercise on fuel behaviour with improved specification
- avoid specifying boundary conditions of the B.C.2 type
- development of rupture criteria suited for one- and/or two-dimensional cladding temperature computation

Total Duration of Exercise:

May 1985 - May 1986

Participation:

10 institutions from 8 different countries

CSNI Report:

E. Scott de Martinville, M. Pignard;
INTERNATIONAL STANDARD PROBLEM ISP 19;
Behaviour of a Fuel Rod Bundle during a Large Break LOCA Transient with a two Peaks Temperature History (PHEBUS Experiment);
Final Comparison Report;
CSNI Report No. 131; 1987

ISP	Date	Host Country	Host Organisation
20	1987	Belgium	TRACTEBEL, Belgium with support from Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Steam Generator Tube Rupture in the Nuclear Power Plant DOEL-2; Belgium

Objectives:

- reanalysis of real plant transient in DOEL-2 on June, 25th, 1979
- "open" post-incident analyses

Facility:

- DOEL-2 Nuclear Power Plant, 2 loop Pressurized Water Reactor (PWR) with 1,187MWth (392 MWe), commissioned in 1975
- 2 steam generators (SSG) with recirculation loops,
- SSG consists of 3,260 U-tubes with 4,130 m^2 heat transfer area
- plant in heat-up condition at 15.5 MPa with coolant temperature at 528 K
- initial flow rate of impaired U-tube approximately 15Kg/s
- data characterizing the plant and the incident wer made available to participants in coded form, based on an available RELAP 5-MOD 2 deck for the DOEL-2 plant

Scaling Information:

Full size 2 loop PWR designed by Westinghouse (commercial Nuclear Power Plant)

Parameters available for Comparison:

- fluid temperatures in hot and cold legs of recirculation loops
- reactor coolant system pressure
- pressure and water level in pressurizer
- pressure and water level in steam generators

ISP	Date	Host Country	Host Organisation
20	1987	Belgium	TRACTEBEL, Belgium with support from Gesellschaft für Reaktor-und Anlagen-sicherheit (GRS); Cologne, Germany

Dominating "Experimental" Uncertainties:

- recorded times with 2 minutes uncertainty band, additional time shift of 20 minutes between some data records
- limited number of data with uncertain time syncronization

Findings:

- acceptable simulation of primary system parameters, large scatter for calculated steam generator parameters by user-induced modifications of RELAP-5 input deck
- limited quality and consistency of plant data give special problems in analyzing the event
- critical safety parameters of a steam generator tube rupture necessary for proper recalculation of the event have been assessed
- in view of limited precision of plant data the ISP exercise gave a qualitative indication for scalabilty of modeling concepts for full size plants

Recommendations:

- for an ISP based on full scale plants coded input data an acceptable alternative to proprietary data, however data should be reliable and validated, e.g. on basis of commissioning tests
- additional data, e.g. on steam relief valve actions, steam dump system and feedwater level control system conditions indispensable for plant reanalysis
- future ISP on real plant transients recommended if equipped with high quality recording systems
- more plant data should be included in ccode validation matrix to address scalability of code models supplementing data from test facilities

Total Duration of Exercise:

June1986-December1987

Participation:
8 institutions from 5 countries
codes involved :RELAP5/MOD2, SMABRE, CATHARE

CSNI Report:

E. Stubbe, et.al;
INTERNATIONAL STANDARD PROBLEM No.20;
Steam Generator Tube Rupture in the Nuclear Power Plant DOEL-2;Belgium
Final Report;
CSNI-Report No.154; December 1988

ISP	Date	Host Country	Host Organisation
21	1989	Italy	Dipartimento di Costruzioni Meccaniche e Nucleari, Università di PISA, Italy

Title:

PIPER-ONE Test PO-SB-7 on Small Break LOCA in BWR- Recirculation line

Objectives:

- Simulation of the PIPER-ONE facility under the conditions of a small break loss of coolant accident in the downcomer region
- confirmation of code capabilities describing a test in a geometrically simplified apparatus
- test PO-SB-7 served as counterpart test to BWR-related tests in other facilities (FIX II)
- "blind" pre-test predictions requested

Facility:

- PIPER-ONE facility consists of a one-dimensional pressure vessel simulator including regions for upper plenum, downcomer separators and steam dome
- 16 heater rod core simulation, electrically heated
- Primary system pressure 7.4 MPa, total volume 0.19 m^3
- Maximum core power 0.28 MW, allows for 20 % of scaled nominal power

Scaling Information:

- power scaling 1/13,500 related to BWR-6 plant
- volume scaling 1/2,200
- full length core simulator
- geodetical elevations of major components preserved
- flow cross sections corresponding to volume scaling (one-dimensionality ot test rig!)

Parameters offered for Comparison:

- 2 absolute pressures (steam dome, lower plenum)
- 2 specified differential pressures
- 18 fluid temperatures at various locations
- 8 mass flow rate measurements
- 9 fuel rod surface temperatures at specified elevations
- 1 fluid density (hot leg)
- several event times

ISP	Date	Host Country	Host Organisation
21	1989	Italy	Dipartimento di Costruzioni Meccaniche e Nucleari Università di PISA, Italy

Dominating Experimental Uncertainties:

certain atypical phenomena mentioned, caused by heat losses and small size of the test rig

Findings:

- considerable scatter of data between pre-test predictions and measurements observed
- qualitative judgement on analytical results given in tables, distinguishing phenomenological findings for three transient periods of the experiment
- certain phenomena of the experiment reasonably well predicted by submitted calculations (e.g.mixture level rise, late re-pressurization)
- some phenomena partially or not well predicted (e.g., thermal stratification, depressurization rate, occurrence of dryout, flashing of core liquid)

Recommendations:

no specific recommendations mentioned in the report

Total Duration of Exercise:
July 1988 - November 1989

Participation:

5 institutions from 4 countries
5 different codes involved: RELAP4 and 5 versions, SMABRE, THYDE-B1

CSNI Report:

F. D'Auria, M. Mazzini, F. Oriolo, S. Paci;
Comparison Report of OECD-CSNI International Standard Problem No. 21;
PIPER-ONE Test PO-SB-7 on Small Break LOCA in BWR Recirculation Line;
CSNI Report No. 162; November 1989

ISP	Date	Host Country	Host Organisation
22	1991	Italy	Dipartimento di Costruzioni Meccaniche e Nucleari Università di PISA, Italy

Title:

Loss of Feedwater Transient in Italian PWR (SPES Test SP-FW-02)

Objectives:

- simulation of the integral behaviour of the SPES facility under the conditions of complete loss of feedwater to Secondary Steam Generators
- core dryout due to lack of heat sinks and opening of pressurizer relief valve
- primary side refilling by actuation of emergency feedwater and pressurizer drainage to the primary coolant system
- "blind" pre-test predictions and extensive "open" post test analyses processed

Facility:

- SPES facility consists of 3 active loops simulating a three loop 2,775 MWth PWR
- 97 heater rod core simulation, electrically heated
- 3 secondary steam generators equipped with 13 U-tubes of original dimensions
- Primary system pressure up to 20 MPa, 640 K to allow to study power excursions
- Maximum channel power level of test section allows140 % of scaled nominal power
- HPIS and LPIS available

Scaling Information:

- power and volume scaling 1/427 corresponding to PWR-PUN, Westinghouse 312 type
- full length core simulator, power level and nominal flow rates scaled 1/427
- geodetical elevations of all components preserved 1/1 to simulate gravitational head

Parameters offered for Comparison:

- 4 absolute pressures (pressurizer, steam dome pressures)
- 3 specified differential pressures (core, SG secondary side)
- 1 fluid temperature at vessel upper plenum
- 2 mass flow rate measurements (core inlet, loop seal)
- 4 fuel rod surface temperatures at specified elevations
- 1 fluid density (hot leg)
- primary system heat loss
- several event times

ISP	Date	Host Country	Host Organisation
22	1991	Italy	Dipartimento di Costruzioni Meccaniche e Nucleari Università di PISA, Italy

Dominating Experimental Uncertainties:

overproportionally large thermal capacities of some compact SPES structures

Findings:

- considerable differences between pre-test predictions and measurements observed
- general failure to predict primary recirculation flow after EFW actuation (reflux condenser mode)
- problems with analysis of heat transfer in steam generators
- modelling of heat losses and thermal capacities of some structures of the SPES components problematic
- post-test calculations in general give insight in reasons for unsatisfactory results of several simulations
- RELAP 4 thermal equilibrium concept not suited for simulation of cold water injection (EFW)
- secondary side recirculation ratio of steam generators not met by codes except by tuning parameters
- heat losses of SPES not well specified before starting the task
- code user problems identified (e.g. nodalization problems)

Recommendations:

no specific recommendations mentioned in the report

Total Duration of Exercise:

November 1987 - March 1990 ("blind" pre-test predictions)
March 1990- July 1991 (post-test analyses)

Participation:

17 institutions from 14 countries (blind pre-test predictions)
4 submittals from 4 countries (post-test analyses)
8 different codes involved: RELAP4 and 5 versions, TRAC-PF1, SMABRE, CATHARE 2, NOTRUMP MOOT

CSNI Report:

W. Ambrosini, M.P. Breghi, F. D'Auria, G.M. Galassi;
Evaluation of Post-Test Analyses of OECD-CSNI International Standard Problem No. 22;
Loss of Feedwater Transient in Italian PWR (SPES Test SP-FW-02)
NEA/CSNI/R(92)7; July 1992

ISP	Date	Host Country	Host Organisation
23	1988	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Rupture of a Large-Diameter Pipe in the HDR Containment

Objectives:

Experiment T31.5 selected for ISP23 served to better understand the pressurisation events expected to happen in typical LOCA events of real nuclear power plants. Specific attention was devoted to

- dynamic and quasi-static pressure loading conditions
- temperature distributions
- heat transfer to structures during the short- and long-term ranges
- transfer of water droplets between adjacent compartments
- local steam /air -composition
- generation of natural convection patterns after termination of blowdown process

Facility:

HDR-containment test facility

- Containment of cylindrical shape, lower section highly substructured into 70 subcompartments; large-volume upper dome section.
- Total free volume: 11,000 m3; height: 50 m
- original primary system pressure vessel serving as representative energy reservoir for blowdown process

Scaling Information:

- large scale test facility, cylindrical shaped steel shell containment, typicality of compartment arrangements; geometrical similarity not preserved;
- energy release rates scaled to power/volume ratio;
- time preserving of mass and energy release rates results in typical pressurisation transients
- "as measured" mass and energy release rates cross-checked by supplementary blowdown calculation

Parameters offered for Comparison:

short-term range: 5 local compartment pressures; 8 pressure differences; 5 local temperatures: 3 local mixture velocities; 3 local mass flow rates
medium-term range: 3 absolute pressures; 10 local temperatures; 3 local heat transfer coefficients
long-term range: 2 absolute pressures; 14 local temperatures; 3 local heat transfer coefficients; 6 internal temperatures of concrete structures; various local steel shell temperatures and corresponding surface conditions

ISP	Date	Host Country	Host Organisation
23	1988	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

- uncertainty of measured discharge rates; largely resolved by extensive analysis of blowdown process under thermal nonequilibrium conditions
- local containment flow velocity measurements burdened by large uncertainties

Findings:

- considerable scatter between predictions and measured evidence observed for the short term range of comparisons, in particular the compartment pressurisation and the resulting local pressure differences
- phase separation and water transport from rupture compartment of great influence
- medium range prediction of maximum global pressure related to the quality of treatment of heat absorption process (global heat balance)
- comparison of long-term range predictions difficult to assess on basis of temperature measurements only; pressure predictions in good agreement; relevance of temperature predictions for heat transfer sensors questionable;
- code user effects clearly visible; quality of all submitted predictions related to earlier experience with similar tasks

Recommendations:

- no further LOCA-oriented ISP necessary
- post-test analyses of ISP 23
- synopsis of earlier ISPs addressing LOCA phenomena (CASP1, CASP3 and ISP16) recommended to assess common code application experience for LOCA situations (code validation matrix)
- future ISP should address phenomena relevant for Severe Accident scenarios, involving the use of traceable gases for better assessment of gas distribution processes

Total Duration of Exercise:

May 1987 to June 1989

Participation:

"blind" pre-test predictions submitted by 15 international institutions; 3 post-test submissions additionally presented;
15 different lumped parameter codes and 1 field code involved in ISP23

CSNI Report:

H. Karwat;
INTERNATIONAL STANDARD PROBLEM ISP 23;
Rupture of a large diameter pipe within the HDR Containment;
Final Comparison Report ;
CSNI Report No. 160 Vol.1.and Vol.2

ISP	Date	Host Country	Host Organisation
24	1988	USA	Department of Energy; Brookhaven National Laboratory; USA

Title:

SURC-4 Experiment on Core Concrete Interactions

Objectives:

Understanding the phenomena associated to the molten core-concrete interaction process and improving the predictability by computer codes with specific emphasis on

- long-term interactions of corium with concrete and water
- oxidation process of metallic core-melt components
- formation of carbon monoxide and hydrogen
- gas release rates and gas chemistry
- "blind" post test analyses

Facility:

- Integral experimental facility SURC (Sustained Urania -Concrete) generating data base for core-concrete interaction processes (e.g. gas release chemistry; heat transfer mechanisms, aerosol formation)
- SURC- tests involve MgO crucible, basaltic concrete slab, mixtures of metallic components (stainless steel, zirconium); heating by induction coils (up to 280 kW)
- SURC-4 Test: 200 kg stainless steel, 20 kg Zr metal, 6 kg fission product simulants and basaltic concrete
- duration of SURC-4 test: 162 minutes with specified periods of heating between 98 kW and 245 kW

Scaling Information:

SURC test series of basic nature serving to generate basic data; typicality of involved material components aimed for

Parameters offered for Comparison:

- temperatures of molten pool and released gas as function of time;
- concrete erosion and dehydration based on thermocouple measurements;
- gas generation rates as function of time corresponding to calculated and measured temperatures
- aerosol release rates as function of time
- energy balance of melt of test rig

ISP	Date	Host Country	Host Organisation
24	1988	USA	Department of Energy; Brookhaven National Laboratory; USA

Dominating Experimental Uncertainties:

nothing mentioned in the final comparison report
reliability and accuracy of experimental data confirmed during post-test analyses workshop

Findings:

large discrepancies observed between "blind" post-test predictions and experimental evidence
involved codes failed to predict trend of melt temperature measurements and effects of addition of zirconium to the melt
some codes predicted concrete erosion correctly
gas release predictions partially correct
results from different participants with the same code differ considerably, indicating a strong user effect in choice of options

Workshop on post-test analyses discussed and confirmed principal shortcomings of applied simulation concepts to analyse core-concrete interaction processes

Recommendations:

several areas for code improvements and amendments (e.g. involved chemical processes, heat exchange between anticipated phases, aerosol formation and release etc.)

more integral tests to investigate into phenomena not well understood hitherto (e.g. crust stability and fragmentation, pool heat and mass transfer),

more separate effects tests to investigate into phenomena not well understood hitherto (lateral and vertical heat, melt chemistry, aerosol release and subsequent mechanical aerosol behaviour)

Total Duration of Exercise:

October 1987-September 1988 (blind post-test predictions)/December 1989 (open post-test analyses)

Participation:

8 institutions from 7 countries;
Involved thermal-hydraulic codes: CORCON/2.02 and2.04 , WECHSL, DECOMP-DOE;
involved aerosol codes :VANESA, METOXA-DOE

CSNI Report:

M. Lee, R.A. Bari;
INTERNATIONAL STANDARD PROBLEM ISP 24;
SURC-4 Experiment on Core-Concrete Interactions;
Final Comparison Report; CSNI Report No. 155, Vol. 1, September 1988;
Final Workshop Summary Report; CSNI Report No. 155, Vol. 2, December 1989

ISP	Date	Host Country	Host Organisation
25	1991	UK	AEA-Thermal Reactor Services, Winfrith Technology Centre; Dorset; UK

Title:

ACHILLES Best-Estimate Natural Reflood Experiment with N_2-Injection from Accumulators

Objectives:

- Simulation of the end of accumulator injection in a postulated large break loss of coolant accident
- thermal behaviour in a core section caused by enhanced reflood rates upon nitrogen injection into the cold leg after termination of nitrogen-driven water delivery from accumulators
- separate effects test type ISP
- "blind" pre-test predictions requested, later "open" post-test analyses also processed

Facility:

ACHILLES test facility consisting of a shroud vessel containing a bundle of 69 electrically heated fuel rod simulators connected to a nitrogen vessel representing an accumulator emptied of water downcomer and upper plenum section allowing the injected nitrogen to escape via a simulated breach location

- fuel rod bundle power 2.8 kW, cosine-shaped power profile
- allowable pressure of test section 0.27 MPa

Scaling Information:

- full length of fuel rod simulator corresponding to commercial PWR fuel rods
- geodetical elevations of rod bundle, downcomer and upper plenum preserved

Parameters offered for Comparison:

- 3 absolute pressures (nitrogen vessel, downcomer and upper plenum)
- 3 overall pressure differences in the simulated loop region
- 6 local pressure differences in core region
- 2 collapsed water levels
- 5 local flow rates
- quench front propagation
- 10 fuel surface temperatures at specified elevations
- 5 subchannel fluid temperatures at specified elevations

ISP	Date	Host Country	Host Organisation
25	1991	UK	AEA-Thermal Reactor Services, Winfrith Technology Centre; Dorset; UK

Dominating Experimental Uncertainties:

nothing specifically reported

Findings:

- "blind" pre-test predictions with a wide scatter of results, even when similar versions of the same code were used (code user effects!)
- modelling of non-condensables presented a problem, partially overcome by applying various user options
- downcomer collapsed water level difficult to predict in pre-test calculations
- heat transfer during initial surge of water generally well predicted, at later stages of the experiment generally over-predicted, possibly due to anomalous flow oscillations at core inlet
- "open" post-test analyses showed improvements, mostly attributed to improvement of modelling the non-condensables
- assumptions concerning flow loss factors a decisive parameter

Recommendations:

nothing explicitly mentioned in the report

Total Duration of Exercise:

June 1988 - November 1989 ("blind" and "open" predictions)

Participation:

7 institutions from 5 countries;
codes involved: RELAP5 /Mod2, TRAC-PF1
"open" exercise:
3 institutions from 3 countries

CSNI Report:

B.J. Holmes;
ISP 25 Comparison Report;
AEA-Technology report AEA-TRS-1043 ;
NEA/CSNI/R(91)11; February 1991

ISP	Date	Host Country	Host Organisation
26	1990	JAPAN	Japan Atomic Energy Research Institute (JAERI); Tokai-mura, Japan

Title:

ROSA-IV LSTF 5% Cold Leg Small-Break LOCA Experiment

Objectives:

- assess code capabilities for analytical simulation of the thermal-hydraulic conditions of an integral loop system during a 5% cold leg small break loss of coolant accident experiment with loss of offsite power and deactivated high pressure safety injection
- improve common understanding of PWR thermal-hydraulic response during such a transient
- identify problem areas which require further code/model improvements based on comparisons between experimental data and calculated data
- "open" post-test calculations requested

Facility:

- integral experiment with electrically heated core simulator, pressure vessel and pressurizer
- 1,064 fuel rods with maximum output of 10 MW (corresponding to 14 % of scaled nominal power)
- 2 equal double capacity loops with secondary steam generators (U-tubes with original length)
- total coolant volume 7.23 m^3
- design pressure 16 MPa

Scaling Information:

- 1/48 volume scaled main components and loop system
- number of fuel rod simulators and decay power scaled 1/48 (however, nominal full power not achievable)
- preservation of flow regime transition anticipated by hot leg diameter specification

Parameters offered for Comparison:

- 5 absolute pressures
- 18 differential pressures
- 9 collapsed liquid levels
- 18 fluid temperatures
- 27 core heater rod surface temperatures
- 24 local mass flow rates
- 7 fluid densities
- 21 additional variables such as void fraction, core power, mass inventory etc.

ISP	Date	Host Country	Host Organisation
26	1990	JAPAN	Japan Atomic Energy Research Institute (JAERI); Tokai-mura, Japan

Dominating Experimental Uncertainties:

nothing mentioned in the report

Findings:

- qualitative simulation of important phenomena (e.g. primary system coolant depletion, loop seal clearing core level drop) successful
- quantitative accuracy not satisfactory (e.g., core liquid level depression, core heater rod temperatures) due to inaccurate simulation of the coolant inventory distribution in the primary system
- major difficulty related to an accurate simulation of the break flow rate which was governed by strong thermal and mechanical non-equilibrium
- code user decisions made during preparation of an input deck significantly affect calculated results derived from calculations made with the same code version by different participants in the ISP
- supplementary JAERI sensitivity calculations to identify sources of scatter among submitted RELAP5/MOD2 calculations indicate interfacial drag model at junctions and break location as prinicipal waekness in conjunction with the adopted nodalization concepts

Recommendations:

- future ISPs on similar objectives should include measured break flow rates as given boundary condition (input data)
- ISP on separate effects tests should deserve more emphasis to improve knowledge about code model applicabilty

Total Duration of Exercise:

March 1989 -mid 1991

Participation:

18 institutions from 15 countries;
codes involved: mostly RELAP 5 MOD2; CATHARE, ATHLET, TRAC-PF1, SMABRE, NOTRUMP,

CSNI Report:

Y. Kukita et. al;
OECD/NEA/CSNI INTERNATIONAL STANDARD PROBLEM ISP 26;
ROSA-IV LSTF 5% Cold Leg Small-Break LOCA Experiment
Comparison Report ;
NEA/CSNI/R(91)13; February 1992

ISP	Date	Host Country	Host Organisation
27	1992	France	CEA-Departement de Thermo-hydraulique et de Physique; CEN-Grenoble; France

Title:

BETHSY Experiment 9.1B; 2" Cold Leg Break without HPSI and with Delayed Ultimate Procedure

Objectives:

- Simulation of integral plant behaviour under Small Break Loss of Coolant Accident (SBLOCA) conditions
- loop seal clearing and heat transfer during boil-off period,
- behaviour of an uncovered core under reflux-condenser mode of steam generators,
- primary side refilling by Low Pressure Injection System (LPIS)
- "blind" predictions requested, "open" post test analyses also processed

Facility:

- BETHSY facility is a 3-loop replica of a 900 MWe FRAMATOME PWR
- 428 heater rod core simulation, electrically heated
- 3 secondary steam generators equipped with 34 U-tubes of original dimensions
- Primary system pressure up to 17.2 MPa, secondary side pressure up to 8 MPa
- Initial power level of test section 10 % of scaled nominal power
- heat losses controlled by external heater system
- accumulator, HPIS and LPIS available

Scaling Information:

- power and volume scaling 1/100 corresponding to FRAMATOME BWR of the 900 MWe class
- full length core simulator, power level and nominal flow rates scaled 1/100
- geodetical elevations of all components fully preserved to simulate gravitational head
- loop piping diameter of hot legs dimensioned to preserve FROUDE number criterion of full size plant

Parameters offered for Comparison:

- 6 absolute pressures (steam generator, condenser, pressurizer, accumulators)
- 30 specified differential pressures
- 12 fluid temperatures at various locations
- 10 flow rate measurements (downcomer, steam lines, LPIS, break flow)
- 7 fuel rod temperatures at specified elevations
- 6 primary side void fractions
- 6 mass inventory informations at SGs and accumulators
- 3 pump speeds

ISP	Date	Host Country	Host Organisation
27	1992	France	CEA-Departement de Thermo-hydraulique et de Physique; CEN-Grenoble; France

Dominating Experimental Uncertainties:

nothing reported

Findings:

- all important trends observed during the test qualitatively predicted, overall behaviour of the BETHSY facility well predicted by some calculations
- break flow predictions problematic although characteristics of break nozzle given
- heat transfer in the core during boil-off generally underestimated
- primary coolant depressurisation predictions dependent on quality of secondary side depressurisation treatment
- transition to residual heat removal system operation (RHRS) burdened by inadequate simulation of stratification conditions
- large scatter of results obtained by various participants with the same code indicate the strong user effect (e.g. for RELAP5 MOD2)
- post-test calculations in general resulted in improved simulations
- large number of participants (28 "blind", 20 "open" contributions) in ISP27 influential on generalized assessment of overall results of this activity

Recommendations:

no specific recommendations mentioned in the report

Total Duration of Exercise:

June 1990 - July 1992

Participation:

25 institutions from 19 countries ("blind" pre-test predictions)
9 different codes involved: RELAP5 versions, TRAC-PF1, DYNAMICA, TECH-M4, ATHLET Mod1

CSNI Report:

P. Clement,T. Chataing, R. Deruaz;
ISP 27-International Standard Problem No. 27;
BETHSY Experiment 9.1B;
2" Cold Leg Break without HPSI and with Delayed Ultimate Procedure
Comparison Report
NEA/CSNI/R(92)20; November 1992

ISP	Date	Host Country	Host Organisation
28	1992	France	CEA/CEN-Cadarache, IPSN/DERS/SEAREL; France

Title:

PHEBUS-SFD B9+ Experiment on the Degradation of a PWR Type Core

Objectives:

PHEBUS Test B9+ was devoted to predict phenomena occurring during a severe fuel damage (SFD) accident in a PWR with specific emphasis ("blind" part) on

- cladding oxidation,
- dissolution of UO_2 and ZrO_2 by molten zircaloy,
- relocation of the melt
- mechanical behaviour of the external zirconia layer of the cladding containing molten zircaloy

under given ("open" part) thermal boundary conditions of the shroud surrounding the fuel test bundle

Facility:

PHEBUS SFD test facility operated at the CEA Research Center CADARACHE consists of a test stringer with a pressurized gas supply, a condenser and a storage tank

- 21 nuclear fuel rod bundle, coupled to a separate nuclear driver core
- active length 0.8 m, flat cosine axial power profile
- pre-pressurized fresh fuel rods
- controlled fluid-dynamic conditions at the bundle inlet (high temperature mixture of steam/hydrogen or helium)
- multi-layer insulating shroud consisting of an inner liner, a porous ZrO_2 layer (24,5 mm) surrounded by a dense ZrO_2 layer (1mm) and a stainless steel tube (8mm) to limit heat losses to the surrounding structures

Provided thermal boundary conditions ("open" part):

- axial profile of the inner liner temperature of the shroud
- simulating the heat losses through the different layers of the shroud surrounding the rod bundle, based on a specified function for the thermal conductivity of the porous ZrO2

Scaling Information:

nuclear separate effects test with some atypical confining structures and conditions

Parameters offered for Comparison:

- fuel temperatures of several fuel rods at various elevations
- fuel cladding temperatures of several fuel rods at various elevations
- local fuel temperatures
- liner and shroud temperatures
- oxidation, UO2 and ZrO2 dissolution profiles
- hydrogen production rates
- axial distribution of relocated core materials

ISP	Date	Host Country	Host Organisation
28	1992	France	CEA/CEN-Cadarache, IPSN/DERS/SEAREL; France

Dominating Experimental Uncertainties:

- behaviour of the shroud as an important thermal radial boundary condition for the test B9+
- dependence of the thermal conductivity of the porous ZrO2 material on operating pressure of the test stringer underestimated

Findings:

Exercise showed the existing capabilities of advanced codes for the prediction of the main degradation phenomena, although test-specific but unavoidable uncertainties existed

- predictions using specified conductivity function overestimated the rod temperatures by underestimating the heat losses through the shroud
- generally, results point on the needs for more verification and validation for severe fuel damage computer codes
- most important limitation concerned simultaneous UO_2-ZrO_2 dissolution by molten Zr and the relocation of molten materials
- specific features of testing degraded core conditions would require more code versatility to allow a correct analytical simulation of the experimental atypicalities (e.g., the shroud in case of the PHEBUS SFD test B9+)
- exercise illustrated the need for improved code user guidelines to make code utilization more effective

Recommendations:

- modelling improvements and corresponding validation work must be achieved, e.g., focussing on hydrogen release, unresolved dissolution problems, channel blockage etc.
- complete validation of degradation models for reactor plant calculations should take into account more comparisons between calculations and experimental evidence from a larger set of conditions, e.g. irradiated and cracked fuel, steam-starved conditions, different heating rates etc. before recommendations for plant application should be given

Total Duration of Exercise:

April 1990 - September 1992

Participation:

15 institutions from 12 different countries;
codes involved: SCDAP, SCDAP/RELAP5, MELCOR, ATHLET-SFD, ICARE, FRAS-SFD, KESS-III, MARCH 3

CSNI Report:
B. Adroguer, A. Commande, C. Rongier, M. Mulet;
OECD/NEA/CSNI INTERNATIONAL STANDARD PROBLEM ISP 28;
PHEBUS-SFD B9+ Experiment on the Degradation of a PWR Type Core;
Comparison Report; Vols. I and II
NEA/CSNI/R(92)17; December 1992

ISP	Date	Host Country	Host Organisation
29	1989-1992	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions

Objectives:

study of the long-lasting phenomena characteristic for severe accident scenarios with hydrogen release with specific emphasis on

- the temperature distribution inside the containment during the entire transient
- the distribution of energy inside the containment between atmosphere and structures during and after the phase simulating the long-lasting small break LOCA
- the steam/air/hydrogen distribution within the containment atmosphere under severe accident conditions initiated by a SBLOCA

Facility:

HDR-containment test facility

- Containment of cylindrical shape, lower section highly substructured into 70 subcompartments; large-volume upper dome section.
- Total free volume: 11,000 m^3; height: 50 m
- original primary system pressure vessel serving as representative energy reservoir for blowdown process
- additional steam supply to simulate long-lasting decay heat release
- cooldown period, enhanced by temporary water spray on external steel shell

Scaling Information:

- large scale test facility, cylindrical shaped steel shell containment, typicality of compartment arrangements; however: geometrical similarity not preserved;
- energy release rates scaled to preserved power/volume ratio;
- time preserving of mass and energy release rates results in typical energy discharge transient
- as measured mass and energy release rates cross-checked by blowdown calculation
- hydrogen release into the containment simulated by release of a helium/hydrogen mixture, rates scaled proportional to the reduced containment volume
- hydrogen release initiated 740 min after begin of SBLOCA transient

Parameters requested for Comparison:

- 1 absolute pressure;
- 10 local compartment temperatures;
- 8 local steam concentrations;
- 11 local hydrogen concentrations;
- 2 vertical hydrogen concentration profiles at 10 preselected points in time;
- 2 containment sump temperatures;

3 local steam condensation rates;

ISP	Date	Host Country	Host Organisation
29	1989-1992	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

- measured steam release rates simulating long-lasting decay heat transport burdened by initial calibration error; initial calibration error detected and resolved by careful recalibration of orifice
- local impact of instrument cooling system on local convection pattern uncertain, although global extraction of energy well established; instrument cooling system causing over-proportional heat loss during the experimental period

Findings:

- considerable scatter between predictions and measured evidence observed for the local hydrogen and steam concentrations and concentration profiles
- applied analytical simulation models in general tend to underestimate thermal stratification of the containment atmosphere
- some predictions of the global pressurisation process indicate shortcomings in the overall energy balance of the containment atmosphere, likely to be caused by inadequate exchange of energy with containment structures
- code user effects clearly visible; quality of all submitted predictions related to earlier experience with similar tasks

Recommendations:

- additional separate effects tests proposed to investigate specific phenomena of integral containment behavior, e.g.:
- flow loss coefficients for large area flow junctions typical for containment systems
- subdivision of large free volumes into subcompartments for lumped parameter code applications
- variation of thermal properties of commercial concrete compositions, e.g. by installed rebars and by concrete humidity
- post-test analyses of other hydrogen distribution experiments to fully merit findings of results of ISP 29
- code application guidelines should be improved, incorporating earlier application experience to reduce unavoidable code user effects

Total Duration of Exercise:

April 1990-June 1992

Participation:

10 institutions from 8 countries;
5 different codes involved : CONTAIN(5*), MELCOR, RALOC, GOTHIC, FUMO

CSNI Report:

H. Karwat;
INTERNATIONAL STANDARD PROBLEM ISP 29;
Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions;
CSNI Report;
NEA/CSNI/R(93)4; February 1993

ISP	Date	Host Country	Host Organisation
30	1992	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

BETA V5.1 Experiment on Melt-Concrete Interaction

Objectives:

- Investigation into phenomena expected to occur during an ex-vessel phase of a core melt accident;
- specific emphasis on concrete erosion under high temperatures and effects caused by a large amount of zirconium in the core-melt
- release of aerosols; formation and release rates for hydrogen and carbon monoxide

Facility:

- BETA test facility developed at Kernforschungszentrum Karlsruhe to investigate molten core concrete interaction (MCCI) experiments
- cylindrical siliceous concrete crucible with electrical induction coil heating
- part of the melt prepared from thermite reaction in a separate tank
- envisaged addition of fission product simulants failed

Scaling Information:

nothing mentioned

Parameters offered for Comparison:

- melt front propagation derived from large number of thermocouple measurements positioned in the concrete crucible
- melt temperatures (however: several sensors failed)
- off-gas temperature
- gas release rate and composition

ISP	Date	Host Country	Host Organisation
30	1992	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

- unexpected melt process caused splash of predominantly metallic melt against the upper crucible cover and into the off-gas system
- more than 50% reduction of metallic content of the melt
- erosion of crucible, "as measured" gas release rates and composition impeded
- aerosol rates not representative due to failure to add fission product simulants

Findings:

- ISP 30 was an "open" exercise, all essential data including malfunction of simulant addition were known
- calculations met general tendencies of experimental results, but also show several inconsistencies
- calculated melt temperatures deviate significantly from measured data and amongst each other
- calculated erosion profiles differ from experimental evidence (underestimation of downward erosion vs.overestimation of radial erosion)
- calculations overestimate generation of burnable gases although early splash of metallic material was known to analysts

Recommendations:

- improvements of current MCCI codes generally recommended
- preparation of a code validation matrix to foster further code assessment
- some participants recommend specific model improvements simulating important processes of melt chemistry

Total Duration of Exercise:

September 1990- April 1992

Participation:

7 institutions from 5 countries;
different versions of 2 codes (CORCON, WECHSL)

CSNI Report:

M. Firnhaber, H. Alsmeyer;
ISP 30; INTERNATIONAL STANDARD PROBLEM No. 30;
BETA V5.1 Experiment on Melt-Concrete Interaction;
Comparison Report ;
NEA/CSNI/R(92)9, April 1992

ISP	Date	Host Country	Host Organisation
31	1993	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

CORA-13 Experiment on Severe Fuel Damage

Objectives:

- Analysis of the heat-up and meltdown phases of a PWR type fuel element in the CORA test facility
- specific emphasis on reliability and accuracy of severe accident computer codes
- investigation into the thermal and mechanical behaviour of a fuel bundle at high temperatures (e.g., formation of blockages, fragmentation of rods)
- study of physico-chemical processes during core degradation (e.g., oxidation of cladding and other metallic components, hydrogen formation)
- "blind" post test analyses

Facility:

- CORA- test facility operated at Kernforschungszentrum Karlsruhe serving to study the behaviour of PWR fuel elements under severe accident conditions
- fuel rod bundle with heated and unheated rods under controlled thermal-hydraulic boundary conditions, high temperature radiation shield surrounding the bundle
- heated fuel rods consist of 6 mm diameter tungsten rod surrounded by UO_2 annular pellets
- two absorber rods added to the bundle to simulate interaction of fuel rods with absorber rod materials
- steam supply to provide superheated steam
- refill or quench phase added

Scaling Information:

- heated length of rods 1,000 mm, rod dimensions and pitch corresponding to original PWR fuel

- typical separate effects tests

Parameters offered for Comparison:

- fluid and fuel temperatures
- cladding and absorber temperatures and temperatures of structures surrounding test bundle
- hydrogen generation rates
- various parameters obtained from post-test examination (e.g., molten zircaloy oxidized part, dissolved UO_2 and absorber material, relocation of molten material and core blockage)

ISP	Date	Host Country	Host Organisation
31	1993	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

nothing reported

Findings:

- ISP 31 was a "blind" exercise, only thermal initial and boundary conditions were given
- heat-up phase predicted quite well, however larger deviations observed for onset of oxidation induced temperature escalation
- modeling of interaction of control rod or spacer material with fuel cladding not possible
- hydrogen production overpredicted during heat-up phase, during late refill and quench phase observed intensive hydrogen production generally not amenable to prediction
- final core blockage fairly well predicted by calculations
- code user influence important

Recommendations:

- necessary development of materials interaction
- code validation and verification necessary for severe fuel damage codes
- improved code user guidelines with more detailed information and recommendations on specific items on degradation phenomena and cladding failure needed

Total Duration of Exercise:

June 1991- June 1993

Participation:

9 institutions from 9 countries;
codes involved SCADAP/RELAP5/MOD3 and 4 other codes (ICARE, KESS, FRAS-SFD, MELCOR)

CSNI Report:

M. Firnhaber, et al.;
ISP 31; OECD/NEA-CSNI INTERNATIONAL STANDARD PROBLEM No. 31;
CORA-13 Experiment on Severe Fuel Damage
Comparison Report;
NEA/CSNI/R(93)17, July 1993

ISP	Date	Host Country	Host Organisation
32			

CANCELLED

ISP	Date	Host Country	Host Organisation
32			

CANCELLED

ISP	Date	Host Country	Host Organisation
33	1994	Finland	VTT Energy, Technical Research Center of Finland, Helsinki

Title:

PACTEL Natural Circulation Stepwise Coolant Inventory Reduction Experiment

Objectives:

- Study natural convection circulation in a VVER plant
- series of quasi steady natural circulation periods
- single phase natural convection and two-phase natural convection with continuous liquid flow
- convection flow under reflux -boiler mode conditions
- "double-blind" pre-test analyses; "open" post-test analyses also processed

Facility:

PACTEL , out of pile model of VVER-440 reactor

- 144 fuel rod simulators with electrically heated and chopped cosine axial power distribution (maximum power 1 MW)
- three double-capacity coolant loops to simulate VVER plant with six loops
- horizontally oriented secondary steam generators
- pressurizer, high and low pressure core cooling systems

Scaling Information:

- 1/ 305 volume and power scaled model of Russian-designed PWR of the VVER-440 type
- maximum power 1 MW equivalent to 22 % of scaled full power
- full length fuel rod simulators
- elevation of main components including loop seals preserved
- diameter (overall height) of steam generators reduced

Parameters offered for Comparison:

- fluid and fuel temperatures
- differential pressure measurements
- primary and secondary side pressures
- water levels at various locations
- downcomer and loop mass flow rates
- various event times (e.g. time to reach and duration of two-phase natural convection)

ISP	Date	Host Country	Host Organisation
33	1994	Finland	VTT Energy, Technical Research Center of Finland, Helsinki

Dominating Experimental Uncertainties:

- core power measurement not accurate at low power levels
- safety valve on top of upper plenum leaking during pressure peaks

Findings:

- overall transients reasonably well predicted
- main discrepancies noted concerned the predictions of flow stagnations and time of core heat-up
- 2-phase natural circulation flow and refilling rates in general overpredicted
- post-test analyses in general yielded improved results due to inclusion of some experimental problems not known in advance for "double-blind" predictions (e.g., leak of safety valve)
- loop seal behaviour a problem also for "open" post-test analyses

Recommendations:

nothing reported

Total Duration of Exercise:

February 1992 -September 1993

Participation:

12 countries; 15 institutions using a variety of codes, including some of Russian origin

CSNI Report:

H. Purhonen, J. Kouhia, H. Holmström;
ISP 33; OECD/NEA-CSNI INTERNATIONAL STANDARD PROBLEM No. 33;
PACTEL Natural Circulation Stepwise Coolant Inventory Reduction Experiment
Comparison Report ; Vols 1 and 2
NEA/CSNI/R(94)24, Parts 1 and 2 ; December 1994

ISP	Date	Host Country	Host Organisation
34	1994	UK	AEA Technology, Winfrith Technology Centre, United Kingdom

Title:

FALCON Fission Product Experiments FAL-ISP-1 and FAL-ISP-2

Objectives:

Phenomena of fission product and aerosol transport in primary coolant system and in the containment
- physical and chemical behaviour of fission products under simulated severe accident conditions
- multi-component aerosol effects and vapour-aerosol interactions
- exercise carried out in two steps:
- "open" part : low relative humidity in the containment, high particle concentration emphasising study of multicomponent aerosols
- "blind" part : high relative humidity in the containment, low particle concentration emphasising study of hygroscopicity in the absence of agglomeration

Facility:

Test apparatus description not very clear in Comparison Report

- Thermal gradient arrangement (TG): serves study of aerosol deposition inside a tube with controlled temperature gradient supposed to simulate a primary system condition

- Containment Configuration (CT): serves to provide data related to a containment under severe accident conditions

Scaling Information:

nothing reported, basic experiments

Parameters offered for Comparison:

Thermal gradient arrangement:
- deposition profiles for specified species of aerosols

Containment configuration:
- time functions of specified containment structure temperatures, gas temperature, relative humidity and mole fraction of helium
- mass deposition of specified aerosols at end of test

ISP	Date	Host Country	Host Organisation
34	1994	UK	AEA Technology, Winfrith Technology Centre, United Kingdom

Dominating Experimental Uncertainties:

- role of revaporisation not clear from experiments
- laminar flow conditions within FALCON pipework not representative for all severe accident sequences
- high surface/volume ratio in FALCON results in some distortions
- uncertainties in deposition data possibly caused by inhomogeneous conditions in the CT
- uncertainties in the mass release rates provided to participants

Findings:

- results in general burdened by a number of uncertainties associated to the experimental conditions and provided input data
- strong interrelations between revaporisation and underprediction of final deposition
- several chemistry issues of aerosol and fission product behaviour require further improvements
- code results may be strongly dependent on code user

Recommendations:

- processes having the major effects on fission product transport behaviour should be identified in order to decide the accuracy to which studies of different phenomena were required
- further relatively simple experiments required to validate specific models of aerosol transport codes

Total Duration of Exercise:

March 1992- June 1994

Participation:

50 submissions from 12 countries;
Codes involved: MELCOR, CONTAIN, VICTORIA, ITHACA, NAUA versions, TRAP-MELT, RAFT, ARES, ATHLET-TRAP

CSNI Report:

D A. Williams;
OECD INTERNATIONAL STANDARD PROBLEM Number 34;
FALCON Fission Product Experiments FAL-ISP-1 and FAL-ISP-2
Comparison Report;
NEA/CSNI/R(94)27; December 1994

ISP	Date	Host Country	Host Organisation
35	1994	Japan	Nuclear Power Engineering Corporation (NUPEC);Tokyo, Japan

Title:

NUPEC Hydrogen Mixing and Distribution Test (Test M-7-1)

Objectives:

- study helium (in lieu of hydrogen) distribution phenomena in a model containment
- particular objectives associated to natural convection effects, effects of spray water addition in inner containment regions
- validation of containment codes
- "blind" post-test predictions requested; supplementary "open" post test calculations were possible

Facility:

NUPEC Hydrogen Mixing and Distribution test facility consists of a

- 1/4 linearly scaled steel containment vessel with 1,300 m^3 free volume, height 17.4 m, inner diameter 10.8 m
- design pressure 0.15 MPa
- test vessel subdivided into 25 inner compartments
- internal and external walls fabricated from carbon steel, external wall covered with thermal insulation
- test vessel equipped with spray system

Scaling Information:

- test vessel linearly scaled 1/4
- typicality of compartment arrangement preserved
- spray rates based on area scaling principles
- thermal properties of containment internal structures not representative for massive concrete structures of prototype facilities

Parameters offered for Comparison:

- 1 global containment pressure
- 28 compartment temperatures
- 29 helium gas concentrations
- 24 locally specified wall temperatures
- 18 junction related mass flow rates

ISP	Date	Host Country	Host Organisation
35	1994	Japan	Nuclear Power Engineering Corporation, (NUPEC); Tokyo, Japan

Dominating Experimental Uncertainties:

- conditioning of the test vessel (pre-heating phase of the test) burdened with some uncertainties, partially resolved by some post-test analyses

Findings:

- qualitative agreement on modeling the main parameters of hydrogen mixing concluded
- some discrepancies between calculated results and experimental observations noted
- local flow measurements impeded by wet conditions (due to spray effects), hence comparisons to predictions impossible
- spray efficiently enhanced mixing process
- use of steel-designed internal compartment walls neglects heat capacities of actually existing large concrete masses inside the containment
- calculation models for heat transfer, flow loss coefficients and stratification need improvements to better understand mixing phenomena

Recommendations:

- future experiments on hydrogen mixing should more carefully specify input and operating conditions to improve the global energy balance
- better local mass flow measurements should be available in a future experiment
- mixing experiments with spray addition should be designed on basis of volume-scaling instead of area scaling to reduce distortions of compartment temperature distribution to more accurately reflect commercial power plant conditions
- future hydrogen mixing and distribution experiments should be more scenario oriented and consider the effects of heat capacities of inner containment structures to better reproduce transient thermal-hydraulic phenomena

Total Duration of Exercise:

January 1993 - March 1994

Participation:

12 institutions from 10 countries ("blind" exercise);
codes involved: CONTAIN, RALOC, MAAP , WGOTHIC, WAVCO, FUMO, COMPACT, MELCOR;
3 institutions from 3 countries ("open" post test analyses)
codes involved: JERICO, CONTAIN, MELCOR, MAPHY-BURN

CSNI Report:

NUPEC;
INTERNATIONAL STANDARD PROBLEM ISP 35;
NUPEC Hydrogen Mixing and Distribution Test (Test M-7-1)
Final Comparison Report;
NEA/CSNI/R(94)29; December 1994; also referenced as OCDE/GD(95)29

ISP	Date	Host Country	Host Organisation
36	1995	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

CORA-W2 Experiment on Severe Fuel Damage for a VVER-type PWR

Objectives:

Analysis of the heat-up and meltdown phases of a VVER-type fuel element in the CORA-test facility

- specific emphasis on reliability and accuracy of severe accident computer codes
- investigation into the thermal and mechanical behaviour of a fuel bundle at high temperatures (e.g. formation of blockages, fragmentation of rods)
- study of physico-chemical processes during core degradation (e.g., oxidation of cladding and other metallic components, hydrogen formation)
- "blind" post-test analyses

Facility:

CORA test facility operated at Kernforschungszentrum Karlsruhe serving to study the behaviour of PWR fuel elements under severe accident conditions

- Fuel rod bundle with heated and unheated rods under controlled thermal-hydraulic boundary conditions, high temperature radiation shield surrounding the bundle
- heated fuel rods consist of 6 mm diameter tungsten rod surrounded by UO_2 annular pellets and Zr-Nb cladding material, arranged in hexagonal array to represent VVER type fuel elements
- two absorber rods added to the bundle to simulate interaction of fuel rods with absorber rod materials
- steam supply to provide superheated steam
- slow cooldown phase terminated the experiment

Scaling Information:

- heated length of rods 1,000 mm, rod dimensions and hexagonal arrangement corresponding to original VVER fuel

- typical separate effects tests

Parameters offered for Comparison:

- fluid and fuel temperatures
- cladding and absorber temperatures and temperatures of structures surrounding test bundle and surrounding shroud
- hydrogen generation
- various parameters obtained from post-test examination (e.g., oxidized part of molten zircaloy, dissolved UO_2 and absorber material, relocation of molten material and core blockage)

ISP	Date	Host Country	Host Organisation
36	1995	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Dominating Experimental Uncertainties:

nothing reported

Findings:

- ISP 36 was organized as a "blind" exercise, only thermal initial and boundary conditions were given
- heat-up phase predicted quite well, however larger deviations observed for onset of oxidation induced temperature escalation
- modeling of the interaction of control rod or spacer material with fuel cladding not possible
- hydrogen production and release rates not described correctly, several codes did not properly treat oxidation of steel components and boron carbide oxidation
- final core blockage predicted by only some calculations
- code user influence important

Recommendations:

- necessary development of materials interaction subroutines
- code validation and verification generally necessary for severe fuel damage codes
- improved code user guidelines with more detailed information and recommendations on specific items (e.g. degradation phenomena and cladding failure) needed

Total Duration of Exercise:

February 1994 - February 1996

Participation:

17 institutions from 9 countries
Codes involved: SCADAP/RELAP5/MOD3 and 4 other codes (ICARE, KESS, FRAS-SFD, MELCOR, ATHLET-CD, RAPTA-SFD)

CSNI Report:

M. Firnhaber, et al.;
OECD/NEA-CSNI INTERNATIONAL STANDARD PROBLEM No. 36;
CORA-W2 Experiment on Severe Fuel Damage for a Russian-type PWR
Comparison Report;
NEA/CSNI/R(95)20, February 1996; also referenced as OCDE/GD(96)19

ISP	Date	Host Country	Host Organisation
37	1996	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany

Title:

VANAM M3-A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material

Objectives:

- containment thermal-hydraulic analyses, distribution and depletion of aerosols in case of severe accidents
- specific emphasis on atmosphere mixing by natural convection loops in a multi compartment geometry
- investigation into the thermal energy distribution inside the containment including structural heat transfer, wall condensation effects
- study of hygroscopic aerosol distribution and depletion including steam condensation on aerosols and volume condensation effects
- assessment of limits of current computer codes
- "open" exercise under given experimental boundary conditions

Facility:

- VANAM tests have been performed in the BATTELLE model containment
- BATTELLE Model Containment is a concrete building with a free volume of 626 m^3, subdivided into several (up to 9) empty sub-compartments
- inner and outer walls are built of reinforced concrete; limited leak tightness under elevated pressures
- interconnections by vent openings of variable cross sections
- total vertical extension of internal free volume approximately 10 m
- inner maximum design pressure ca. 0.5 MPa

Scaling Information:

- volume scaling related to commercial PWR containments approximately 1/100
- elevation reduction approximately 1/5 related to commercial PWR containments
- internal surface/volume ratio approximately 2.4 (as compared to approximately 0.9 ratio of commercial PWR containments)
- boundary conditions of VANAM test not typically expected for severe accident situations in real plants

Parameters offered for Comparison:

- total containment pressure
- containment temperatures and temperatures of structures at several specified locations
- vent flow velocities
- relative humidities
- aerosol concentrations and particle size distribution at seven locations
- airborne water droplet concentrations
- local condensation rates at 3 distinct condensate collectors of 1 m^2 surface area

ISP	Date	Host Country	Host Organisation
37	1996	Germany	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS); Germany

Dominating Experimental Uncertainties:

- largely unknown local and global leakage behaviour of the model containment, distorting in particular long-lasting experiments
- humidity measurements had large uncertainties

Findings:

- due to the complexity of the experiment and the facility ISP 37 was performed as "open" exercise
- comparisons of calculated pressures impeded by uncertainties about facility leakage effects
- deviations in calculated temperature distributions have been attributed to applied unfavourable modeling concepts (e.g. nodalization)
- atmospheric stratification simulated only by part of submitted calculations
- wide scatter of calculated aerosol concentrations (up to two magnitudes), mostly overestimation of actual concentrations
- scatter of aerosol concentrations mostly attributed to limits in capabilities of codes to predict the relevant thermal parameters like relative humidity, volume condensation and atmospheric flow rates
- due to applied test conditions and other guiding uncertainties limitations of ISP 37 results to application in real plants to be observed
- code user effects stressed

Recommendations:

- code models and experimental techniques concerning thermal-hydraulic variables like humidity, wall condensation and heat transfer to walls should be improved

Total Duration of Exercise:

May 1995- December 1996

Participation:

22 institutions from 12 countries;
codes used mostly: CONTAIN, MELCOR, FIPLOC, GOTHIC, ECART, MACRES etc.

CSNI Report:

M. Firnhaber, T.K. Kanzleiter, S. Schwarz, G. Weber ;
OECD/NEA-CSNI INTERNATIONAL STANDARD PROBLEM No. 37;
VANAM M3-A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material;
Comparison Report;
NEA/CSNI/R(96)26, December 1996; also referenced as OCDE/GD(97)16 and GRS-137

ISP	Date	Host Country	Host Organization
SSWISP	1996	Japan	Nuclear Power Engineering Corporation (NUPEC); Tokyo, Japan

Title:

SEISMIC SHEAR WALL ISP NUPEC'S SEISMIC ULTIMATE DYNAMIC RESPONSE TEST

Objectives:

- To clarify behavior of seismic shear walls up to the ultimate state.
- To improve reliability of techniques for non-linear seismic response analysis and ultimate strength evaluation.
- To provide available information for use in quantifying uncertainties in modeling and in improving the computer codes of the participants in the future.
- To suggest necessary experiments to reduce technical ambiguities in the ultimate behavior of seismic shear walls.

Facility:

The world's largest high-performance shaking table at Tadotsu Engineering Laboratory of NUPEC.
- Two reinforced concrete (RC) shear wall specimens with the same specification.

Scaling Information:

- Web wall: thickness: 75 mm, length: 3,000 mm, height: 2,020 mm,
  shear span ratio: 0.8,
  steel percentage in the concrete: 1.2%.
- Flange wall: thickness: 100 mm, length: 2,980 mm,
- Added mass: 92.9 tonf (to ensure coincidence between the center of gravity and the geometrical center of the top slab of the specimen),
- Total specimen weight: 122.0 tonf.
- Vertical compressive stress in the walls: 0.15 kgf/mm^2.

Parameters offered for Comparison:

- Small-amplitude level in elastic region. (RUN-1)
- Strain level at shear crack initiation. (RUN-2, shear deformation angle of 0.24/1000 rad.)
- Strain level at 3 times step (2). (RUN-3, shear deformation angle of 0.72/1000 rad.)
- Shear deformation angle of about 2/1000 rad. (RUN-4)
- Vicinity of ultimate strength. (RUN-5, shear deformation angle of about 4/1000 rad.)

ISP	Date	Host Country	Host Organization
SSWISP	1996	Japan	Nuclear Power Engineering Corporation (NUPEC); Tokyo, Japan

Dominating Experimental Uncertainties:

Measured displacement had saturated after the occurrence of failure in the specimen at web wall in the test RUN-5.

Findings:

Issues highlighted in the discussions:
- Modeling of damping, particularly damping in plastic region (viscous damping and hysteresis damping).
- Modeling of hysteresis rule of RC-members.
- Handling of non-linear stress-strain relationship of concrete which is prone to vary, largely due to progressive cracking of the concrete.
- Methods of load application in analysis (static, cyclic and dynamic loadings).
- Degree of details of modeling.

Recommendations:

- Simplified and Lumped Mass Methods are highly cost effective and simulation results produced are by no methods inferior to those from FEM. However, it should be noted that when these methods are applied to non-linear analysis of RC structures up to their ultimate condition, rational and physical modeling of the hysteresis rule of the RC members has to be introduced. The lumped mass method will be used continuously for the time being for NPP seismic design.
- FEM is a very effective and useful tool, including application to static analyses. Although FEM involves too much analytical cost to be affordable in practical seismic design analyses, it could soon be a popular tool among engineers in this field, taking into account future progress in performance of computer systems and application experience.
- By that time, FEM researchers have to improve reliability of dynamic FEM analysis and prepare some application guidelines to help engineers determine "basic physical parameters" for general FEM modeling.
- Benchmark analyses based on test data like this SSWISP, simulating non-linear dynamic response of specimens up to their failure, are very useful in learning the and issues for the application of FEM and sophisticated analytical methods.

Total Duration of Exercise:

December, 1993 – February 1997

Participation:

1st workshop: 53 part. from 10 countries; 2nd workshop: 90 part.from 12 countries (38 organis.)
Analyses submitted: FEM Static: 19, FEM Dynamic: 12, Simplified model: 10, Lumped mass model; 6

CSNI Report:

SEISMIC SHEAR WALL ISP NUPEC'S SEISMIC ULTIMATE DYNAMIC RESPONSE TEST
Comparison Report
NEA/CSNI/R(96)10, February 1997; also referenced as OCDE/GD(96)188

ISP	**Date**	**Host Country**	**Host Organisation**
38	1997	UK	AEA Technology / TRACTEBEL / CEA

Title:

BETHSY Experiment 6,9c: Loss of Residual Heat Removal System during Mid-Loop Operation

Objectives:

Simulation of integral plant behaviour under atmospheric conditions (open system) with anticipated loss of Residual Heat Removal System during mid-loop operation

- study physical phenomena under very low system pressure, in particular the behaviour of pressurizer and surge line
- core uncovery and reflooding
- effects of loss of primary coolant and refilling by RHRS
- "open" calculations requested to allow assessment of code applicability for conditions not anticipated during development of codes (extreme low pressure)

Facility:

BETHSY facility is a 3-loop replica of a 900 MWe FRAMATOME PWR

- 428 heater rod core simulation, electrically heated
- 3 secondary steam generators equipped with 34 U-tubes of original dimensions
- primary system pressure up to 17.2 MPa, secondary side pressure up to 8 MPa
- initial power level of test section allows for 10 % of scaled nominal full power
- heat losses controlled by external heater system
- HPIS and LPIS available

Scaling Information:

- power and volume scaling 1/100 corresponding to FRAMATOME BWR of the 900 MWe class
- full length core simulator, decay power level and nominal flow rates scaled 1/100
- geodetical elevations of all components preserved 1/1 to simulate gravitational head
- loop piping diameter of hot legs dimensioned to preserve FROUDE number criterion of full size plant

Parameters offered for Comparison:

- 2 absolute pressures (cold leg, upper plenum)
- 30 locally specified differential pressures
- 7 fluid temperatures at various locations
- 7 flow rate measurements (downcomer, safety injection, pressurizer)
- 4 fuel rod temperatures at specified elevations
- 6 local void fractions
- 6 mass inventory informations supplemented by 9 integrated mass flow rates
- 1 core mixture level

ISP	Date	Host Country	Host Organisation
38	1997	UK	AEA Technology / TRACTEBEL / CEA

Dominating Experimental Uncertainties:

- extensive consistency checks performed for important experimental parameters did not indicate any important experimental problem

Findings:

- although involved codes originally were not developed for low pressure conditions, calculations follow the general transients
- observed deviations partially attributed to the use of correlations derived for high pressure conditions and hitherto applied in numerical schemes (e.g.nodalization concepts) proven under such conditions
- main phenomena of interest for this type of experiment (e.g. mass discharged through open manways, core uncovery and heat-up) qualitatively met by calculations
- on a quantitative basis magnitude of calculated phenomena differs considerably from experimental values and among codes, among different versions of the same code and/or among different submissions employing the same code version
- code user effects visible in particular from submissions employing the same code version for an identical task
- multidimensional character of two-phase flow phenomena under very low system pressure is also considered causal for quantitative differences
- caveat expressed with respect to representativeness of scale BETHSY results for direct application to a Nuclear Power Plant and in view of applicability to other low pressure transients

Recommendations:

- further work for code application to low pressure conditions is required
- separate effects tests recommended to extend range of applicability of models
- RELAP 5 developers encouraged to resolve numerical mass diffusion problem
- attention should be paid to further scaling considerations for low pressure conditions
- future ISP with limited objectives on basis of separate effects test recommended

Total Duration of Exercise:

September 1995-May 1997

Participation:

24 submissions from 17 countries;
5 different code groups involved: RELAP5/MOD3 versions, TRAC-PF1, ATHLET Mod1, CATHARE, MELCOR

CSNI Report:

G. Kimber, C. Leveque, G. Lavialle;
International Standard Problem No. 38;
BETHSY Test 6,9c: Loss of Residual Heat Removal System during Mid-Loop Operation
Final Comparison Report;
NEA/CSNI/R(97)38; Vols. I and II; June 1998

ISP	Date	Host Country	Host Organisation
39	1997	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Title:

FARO-Test L-14 on Fuel Coolant Interaction and Quenching

Objectives:

- Benchmarking the predictive capabilities of computer codes used in the evaluation of fuel coolant interaction (FCI) and quenching phenomena on basis of the FARO test L-14

- Particular emphasis on vessel pressurization, premixing aspects, debris formation and cooling, quenching and steam production rates, quantification of hydrogen formation rate

- "Open" exercise, experimental results provided

Facility:

- FARO facility consists of a furnace capable to handle up to 200 kg oxide fuel simulant at temperatures up to 3,270 K; metallic components can be processed as well
- orifice/valve system provided to control release mode of the melt (gravity or forced) into test vessel
- release vessel designed for a pressure of 10 MPa at 570 K combined with a debris catcher on the lower part of the vessel
- release vessel connected to a steam/water separator and a venting system with a set of 4 pressure relief valves
- test vessel trace-heated to control heat losses and initial test conditions
- FARO test L-14 run with 125 kg of a dioxide mixture (80% UO_2 and 20% ZrO_2) released by gravity in 2 m deep pool of saturated water at 5.1 MPa

Scaling Information:

no information provided in the report

Parameters offered for Comparison:

- 1 vessel pressure
- 2 temperatures in steam dome and water phase
- 2 mixture levels
- energy release information
- quenching rate
- final fragmented mass and debris diameter distribution
- melt-water contact time (event time)
- melt-bottom contact time (event time)

ISP	Date	Host Country	Host Organisation
39	1997	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Dominating Experimental Uncertainties:

some uncertainties in melt release rate, overcome with later reference assumptions,
sensitivity analyses demonstrated small influence of this parameter on results of analyses

Findings:

- in view of "open" nature of exercise considerable scatter of calculated results concluded as not yet acceptable
- released energy and resulting vessel pressure in general too low, result attributed to energy partitioning between steam/water phases in non-equilibrium
- origin and impact of hydrogen production on quenching phase unclear
- great differences in calculated vapour void fraction and melt particle distribution at given point in time (0.9 s)
- code user effects on calculated results evident from multiple use of same code and from supplementary sensitivity analyses on effects of "tunable" parameters

Recommendations:

- future verification of code capabilities on the test L-14 after some time considered useful to evaluate improvements
- addressed areas for further code development (e.g. fragmentation modeling, subcooled film boiling and radiation heat transfer, steam/water energy partitioning, interrelation between hydrogen production and melt quenching)
- better descriptions of rationale behind offered code options (e.g."tunable" parameters)

Total Duration of Exercise:

April 1996-September 1997

Participation:

13 institutions from 10 countries;
8 different codes or code versions participating: COMETA, IFCI, IVA, JASMINE, MC3d, TEXAS, THIRMAL, VAPEX

CSNI Report:

C. Addabbo, etal..;
OECD/CSNI INTERNATIONAL STANDARD PROBLEM 39;
FARO-Test L-14 on Fuel Coolant Interaction and Quenching;
Comparison Report;
NEA/CSNI/R(97)31; Volumes I and II ;1998

ISP	Date	Host Country	Host Organisation
40	1998	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Title:

Aerosol Deposition and Resuspension in STORM Test SR 11

Objectives:

- Demonstration of the capabilities of the computer codes used in safety calculations of nuclear reactors and identification of weak points in the modelling of deposition and resuspension of aerosols
- Particular emphasis on deposition processes (first phase of exercise) and resuspension processes (second phase of exercise)
- "Blind" exercise, boundary conditions provided; followed by "open" recalculations

Facility:

- STORM test section consists of a 5.0 m long straight pipe with 63 mm internal diameter, coupled to a mixing vessel (10m3 volume) and a quench vessel
- Plasma torches for aerosol preparation; steam and nitrogen supply systems providing carrier gas for aerosol transport into test section and gas flow during the resuspension period
- tin oxide (SnO2) used as aerosol simulant
- deposition phase preceded by pre-heating process to stabilize thermal-hydraulic conditions of the test section

Scaling Information:

separate effects test

Parameters offered for Comparison:

- spatial distribution of deposited mass per unit area (kg/m2)
- temporal evolution of total deposited mass in the test section
- spatial distribution of deposition mechanisms
- during second test-phase:
- resuspension rates
- mean particle size at the end of test section
- mass remaining in test pipe

ISP	Date	Host Country	Host Organisation
40	1998	CEC-Ispra	CEC-Ispra, Joint Research Centre Commission of the European Communities

Dominating Experimental Uncertainties:

Error in the steam flow rate during the deposition phase communicated for the "blind" exercise, but detected before starting the second phase of the exercise

Findings:

- aerosol retention strongly depends on correct characterisation of thermal-hydraulic conditions
- submitted analyses indicate tendencies to overpredict aerosol deposition in test pipe
- particle size tracking codes assessed in a preliminary phase of development ; improvements would require input of parameters which are generally not known
- aerosol resuspension models available for ISP 40 assessed as inadequate
- potential of resuspension models dependent on correct characterisation of deposit (state of deposit, not only deposited mass)
- procedures of code qualification and user qualification questioned

Recommendations:

Separate effects tests needed to relate the characteristics of deposits to their chemical composition and the deposition mechanisms

Total Duration of Exercise:

March 1997-July 1998

Participation:

15 institutions from 10 countries and organisations
11 different codes participating: MELCOR, ATHLET-CD, VICTORIA, RAFT, ECART, ART, SOPHEROS, AEROSOLS-B2, DeNIRO, ART, MARIE

CSNI Report:

J. Areia Capitao, A. de los Reyes, G. de Santi;
INTERNATIONAL STANDARD PROBLEM No. 40;
Aerosol Deposition and Resuspension;
Final Comparison Report;
NEA/CSNI/R(99)4, February 1999

ISP	Date	Host Country	Host Organisation
41	1998	Canada	Atomic Energy of Canada Ltd.(AECL) and CANDU Owners' Group (COG)

Title:

RTF Experiment on Iodine Behaviour in Containment Under Severe Accident Conditions

Objectives:

- Evaluation of various iodine behaviour codes in terms of predicting the iodine volatility from the liquid to the gas phase as function of the pH of the aqueous phase under radiation conditions
- basic experiment characterized by its simplicity for ease of modelling and good quality of data to compare the basic components of involved codes
- suppress formation of organic iodides by avoiding any organic material
- two sets of calculations requested: first calculation with input variables chosen by the participant; second calculation based on a standard set of rate constants agreed upon by all participants

Facility:

- Radioiodine Test Facility (RTF) consisted of a cylindrical stainless steel vessel provided with a 60 Co radiation source and electrical heaters around the vessel to control the temperatures
- vessel connected to an aqueous recirculation loop and to two sampling loops for the gas and aqueous phases
- liquid phase volume 0.025 m3 , gas phase volume 0.220 m3
- initial pH-value approximately 10 with stepwise variation during the experiment
- initiation of test by injection of 131 I-labelled CsI into the aqueous phase resulting in a starting concentration of approximately 10 -8 mol / m3 CsI

Scaling Information:

- fundamental experiment not directly scaled to assumed severe accident conditions of NPPs

Parameters offered for Comparison:

- total concentration of iodine in the aqueous phase as function of time
- total concentration of iodine in the gas phase as function of time
- speciation of the iodine in the aqueous phase as function of time
- speciation of the iodine in the gas phase as function of time
- H2O2 and H2 concentrations as function of time resulting from radiation
- distribution of iodine at the end of the test between the gas- and water-phases and on exposed surfaces

ISP	Date	Host Country	Host Organisation
41	1998	Canada	Atomic Energy of Canada Ltd.(AECL) and CANDU Owners' Group (COG)

Dominating Experimental Uncertainties:

- nothing reported

Findings:

Analysis of calculations indicated the key phenomena influencing iodine behaviour, ranked in order of most important to least important were
1. Radiolysis (pool chemistry)
2. Adsorption/desorption behaviour (rate constants)
3. mass transfer

3 out of 9 for the first set of calculations and 4 out of 9 for the second set of calculations (repeated calculations with a standard set of rate constants for adsorption and mass transfer) met the two criteria established for comparison i.e. gas phase iodine concentration reproducible within an order of magnitude, aqueous phase iodine concentration reproducible within 20% as function of time.

In conclusion, based on the established criteria for comparison, the ISP exercise demonstrated that all the codes were capable of giving reasonable reproduction of experimental results under the given restricted test conditions. Proper choice of code parameters for key phenomena is essential for overall performance of codes if used as predictive or interpretive tools

Recommendations:

Two sets of future exercises were proposed to examine other aspects of code applicability:

Parametric calculations covering the effects of
- pH,
- temperatures of the test,
- dose rates
- initial iodine concentrations
- simultaneous presence of additional materials (e.g., leading to organic iodine formation)

"Blind" post test calculations of similar future experiments

Total Duration of Exercise:

1997 -1999

Participation:

8 countries (9 institutions) using codes like LIRIC, MELCOR-I, IMPAIR and IODE

CSNI Report:
International Standard Problem (ISP) No. 41:
Containment Iodine Computer Code Exercise Based on a Radioiodine Test Facility (RTF) Experiment Comparison Report;
NEA/CSNI/R(2000)6

ISP	Date	Host Country	Host Organisation
42	1999	Switzerland	Paul Scherrer Institut(PSI), Villigen, Switzerland

Title:

ISP-42- PANDA Test "TEPPS"

Objectives:

- transient and quasi steady-state operation of a passive containment cooling system (PCCS)
- primary coolant system and containment system behaviour coupled
- primary coolant system under low pressure natural convection conditions
- steam condensation in the presence of non-condensables
- mixing and/or stratification of light or heavy gases in the containment
- mixing and/or stratification in large water pools

Facility:

- multi-purpose facility
- 4 pressure vessels simulating different parts of a containment (e.g., drywell and wetwells, total volume ca. 424 m3)
- reactor pressure vessel (RPV) simulator (22.8 m3)
- connecting pipework
- 3 passive cooler elements
- maximum available thermal power 1.2 MW

Scaling Information:

integral systems experiment, but without scale-relations to existing nuclear power plants
basic testing of new containment features

Parameters offered for Comparison:

- 3 absolute pressures (RPV, drywell, wetwell)
- 9 air partial pressures
- 9 mass inventories
- 10 mass flow rates
- 5 phase detector signals inside vent lines
- 2 valve positions (open/close)
- 45 temperatures of gas and gas/steam phase at various locations
- 4 pool surface temperatures
- 36 water phase temperatures at various locations
- 43 wall and wall surface temperatures at various locations

ISP	Date	Host Country	Host Organisation
42	1999	Switzerland	Paul Scherrer Institut(PSI), Villigen, Switzerland

Dominating Experimental Uncertainties:

ISP 42 still in progress

Findings:

ISP 42 still in progress

Recommendations:

ISP 42 still in progress

Total Duration of Exercise:

ISP 42 still in progress

Participation:

ISP 42 still in progress

CSNI Report:

ISP Specification reports:

D. Lübbesmeyer et al.
ISP 42 - Description of the PANDA Facility
PSI Report TM-42-98-41--ALPHA-836-1;
29 January 1999

C. Aubert, J. Dreier;
ISP-42 PANDA Test Phases;
PSI Report TM-42-98-40--ALPHA-835-1;
28 January 1999

ISP	Date	Host Country	Host Organisation
43	1999	USA	University of Maryland (UM), College Park, Maryland, USA

Title:

Rapid Boron Dilution Test

Objectives:

- examine the applicability of CFD codes to integral test facilities
- verify specific features of CFD codes that are required in the simulation of reactor transients
- collate the vast experience gained in rapid boron dilution experimentation and simulation by organisations in various countries
- boron dilution process simulated by specified local injection of "cold" water supported by qualitative and quantitative flow visualization tests
- "blind" exercise, boundary conditions provided; followed by "open" recalculations

Facility:

- 2x4 Thermal-hydraulic Loop Facility (UM 2x4 Loop)
- reactor vessel with core barrel and downcomer, 2 hot legs and once through steam generators, 4 cold legs and one pressurizer
- 4 reactor coolant pumps
- high pressure safety injection (HPSI)
- design pressure 2.1 MPa

Scaling Information:

- volume scaling 1/500, reduced component elevation levels
- hydraulic diameter ratios of downcomer 1/5
- cold leg internal diameter ratio 1/9
- downcomer nominal velocity ratio 1/18
- scaling of separate effects test based on Reynolds, Strouhal, and Schmidt number considerations

Parameters offered for Comparison:

- differential pressure transducers
- local flow meter measurements
- thermocouple temperature measurements

ISP	Date	Host Country	Host Organisation
43	1999	USA	University of Maryland (UM), College Park, Maryland,USA

Dominating Experimental Uncertainties:

ISP 43 still in progress

Findings:

ISP 43 still in progress

Recommendations:

ISP 43 still in progress

Total Duration of Exercise:

ISP 43 still in progress

Participation:

ISP 43 still in progress

CSNI Report:
