Joint ICTP-IAEA Essential Knowledge Workshop on Deterministic Safety Assessment and Engineering Aspects Important to Safety

Verification and validation of the computer codes

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Content of the lecture

- Definition of V&V
- V&V of the computer code
- Experimental programs
- OECD CCVM projects
  - o Separate effect tests
  - o Integral effect tests
- IAEA validation matrix for competency and skill development
- Qualification of the code input model

Definition

- Verification: Comparison of the source coding with its description in the documentation (“doing thing right ”)
- Validation: Code assessment against relevant experimental data to demonstrate the applicability/ accuracy of the code to predict phenomena expected to occur (“doing right thing”)

Code verification

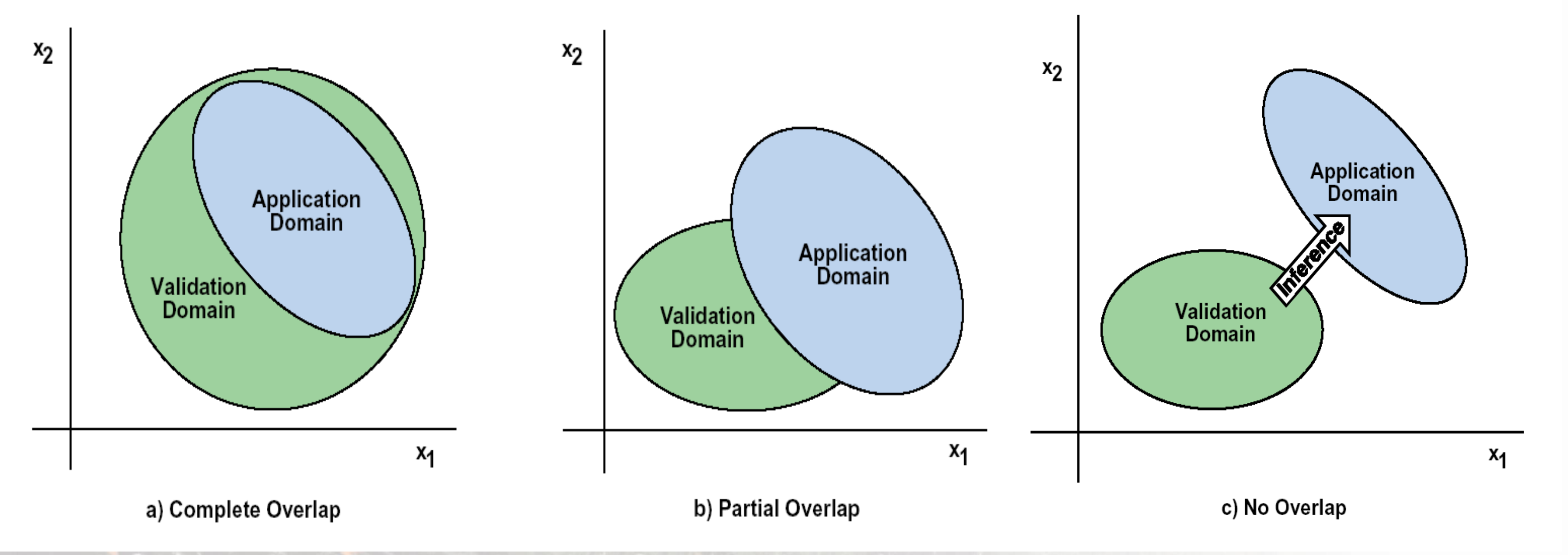
- Almost entirely code developer responsibility
- Verification practice
  - o Formal, major life-cycle reviews and audits
  - o Formal peer reviews
  - o Informal tests such as unit and integration testing
  - o QA (software)

Code validation

- Code validation
  - o Demostration of the code capability to predict facility response to PIE
- Principal way of code validation through comparison to (scaled-down) experimental data

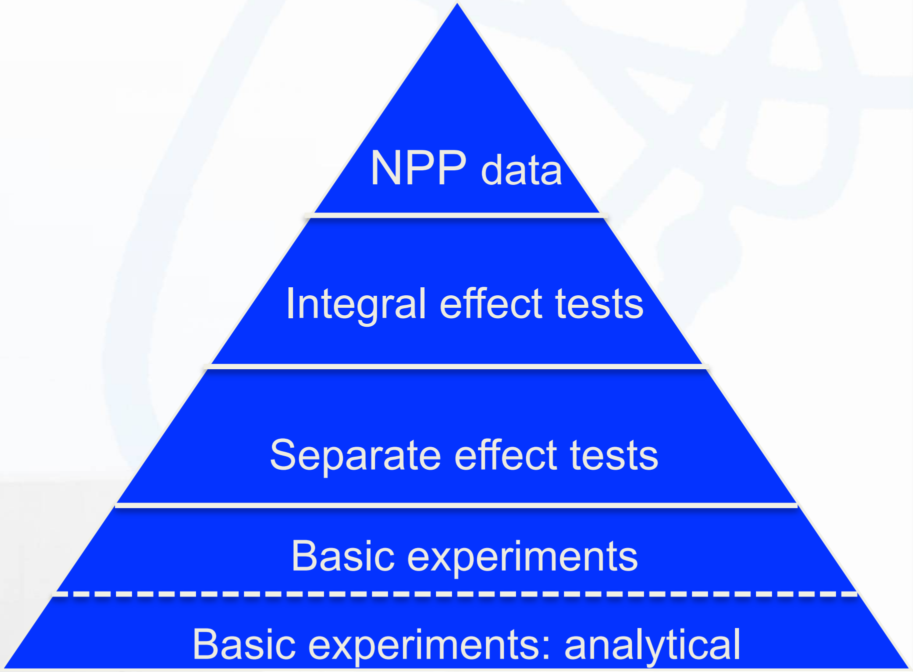
Computer code validity

- Able to simulate the analyzed facility and PIE
- Appropriate for the selected methodology
- Verified and validated

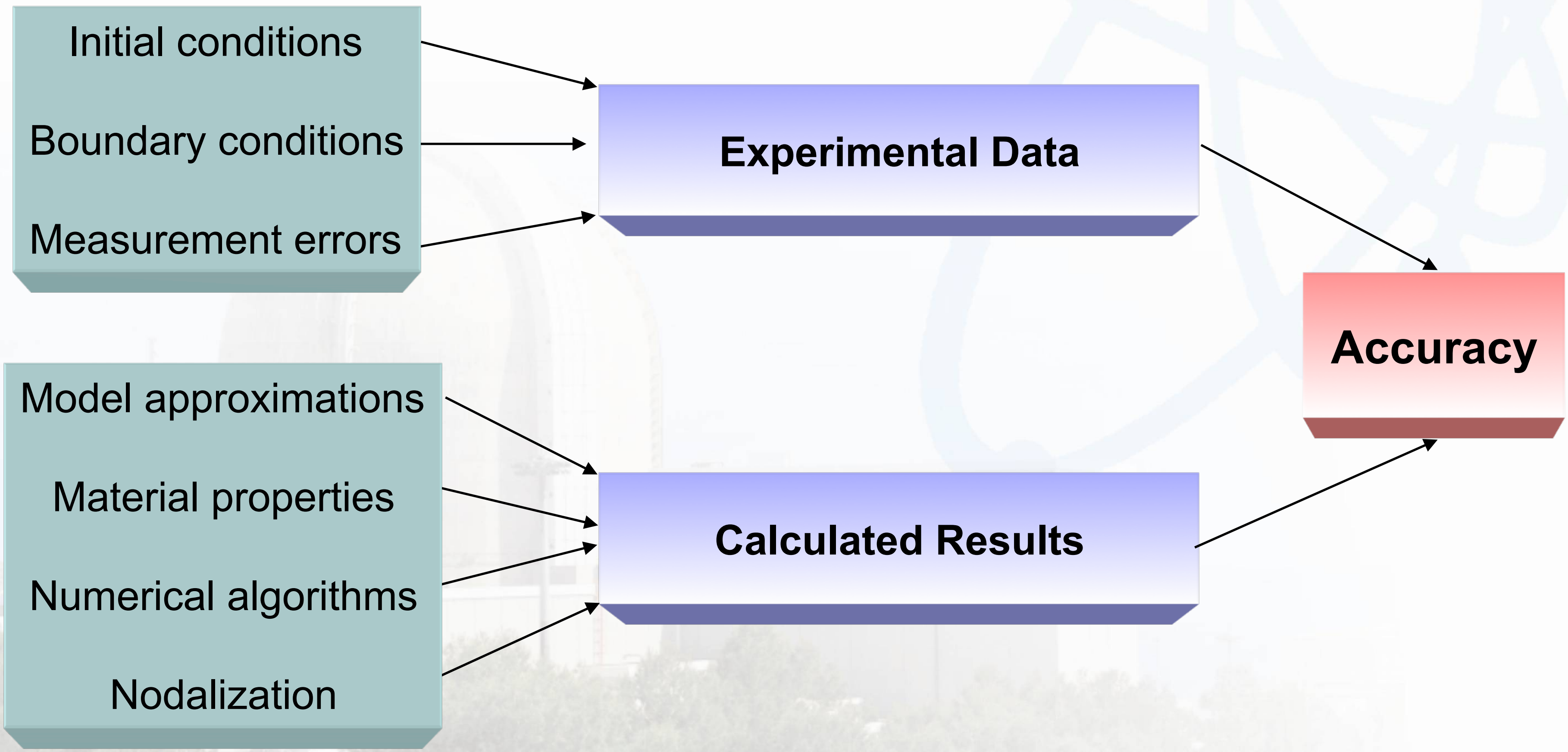


Computer code validation

- Validation practice
  - Basic tests
    - Simple test cases that may not be directly related to an NPP. The tests may have analytical solutions or correlations or data derived from experiments
  - Separate effects tests
    - These address specific phenomena that may occur in an NPP
  - Integral tests
    - These are tests carried out in scaled down test facilities simulating NPPs where the overall behaviour of a plant can be simulated during accident conditions.
  - NPP level tests and operational transients
    - Data from operating plants – planned tests or transients – provide an important means for qualifying the plant model



Computer code validation



Background of CCVM

- Systematic collection of the best sets of openly available test data for code validation, assessment and improvement, including quantitative assessment of uncertainties in the modeling of individual phenomena by the codes

Reports

- OECD/NEA/CSNI: Validation Matrix of Thermal-Hydraulic Codes for LWR LOCA and Transients. CSNI/R132, Paris: NEA, 1987
- OECD/NEA/CSNI: Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients. NEA/CSNI/R(96)17, Paris: NEA, 1996 – update of previous report

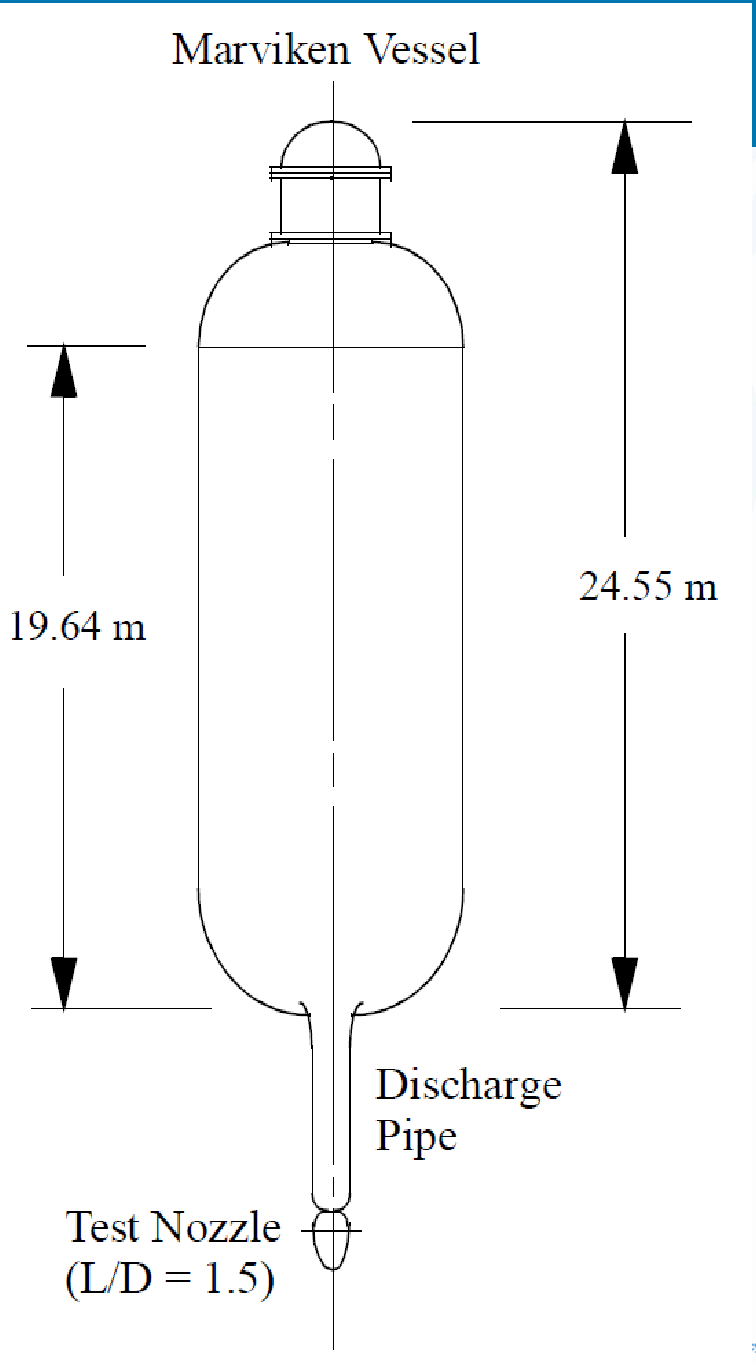
  http://www.oecd-nea.org/nsd/docs/1996/csni-r1996-17.pdf
- OECD/NEA/CSNI: Separate Effects Test Validation Matrix for Thermal-Hydraulic Code Validation. NEA/CSNI/R(93)14, Paris: NEA, 1993

  http://www.oecd-nea.org/nsd/docs/1993/csni-r1993-14.pdf
- OECD/NEA/CSNI: Validation Matrix for the Assessment of Thermal-Hydraulic Codes for VVER LOCA and Transients. NEA/CSNI/R(2001)4, Paris: NEA, 2001

  http://www.oecd-nea.org/nsd/docs/2001/csni-r2001-4.pdf

Separate effect tests

- Behavior of a single component or isolated part of the system or single TH phenomenon
- (Relatively) easy to build and operate
- Full size (no scaling)
- Clear boundary conditions
- Measurement instrumentation can be chosen to study one particular phenomenon
- Reduced possibility of compensating modelling errors during validation of computer codes
- Systematic evaluation of accuracy of code models across a wide range of conditions up to full reactor plant scale



OECD/CSNI SET CCVM

- Report in two volumes
- Volume I
  - o Phenomena characterization and selection of facilities and tests
  - o Matrices
    - – Phenomena vs. SET
    - – Phenomenon vs. facility identification (at least 3 facilities), relevant parameters ranges -> basic info for code validation
- Volume II
  - o Facility and experiment characteristics

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OECD/CSNI SET CCVM

- 67 thermal-hydraulic phenomena
- 185 test facilities
- Information sheets for 113 test facilities available
- Identification of phenomena relevant to two-phase flow in relation to LOCAs and thermal-hydraulic transients in light water reactors (LWRs)
- Characterisation of phenomena, in terms of
  - a short description of each phenomenon,
  - its relevance to nuclear reactor safety,
  - information on measurement ability, instrumentation and data base
  - present state of knowledge and predictive capability of the codes
- Selection of relevant tests
- A total of 1094 tests are included in the SET matrix

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SET – phenomena vs. SET facilities

Phenomena | **Separate Effects Test Facilities**

LEGEND

x suitable for model validation

o limited suitability for model validation

· not suitable for model validation

Phenomena		1. Canada: Elbow Flooding Rig	CWIT (CANDU reactors)	Pumps	Header Test Facility (CANDU reactors)	2. Finland: REWET-I	REWET-II		VEERA		IVO-CCFL (Air/Water)	IVO-Thermal Mixing	IVO-Loop Seal Facility (Air/Water)
	Facility No.	1	2	3	4	1	2	3	4	5	6	7	8
	Info Sheet available	a	a		a	a	a		a		a	a	a
0 BASIC PHENOMENA	1 Evaporation due to Depressurisation	·	·	·	·	·	·		·		·	·	·
	2 Evaporation due to Heat Input	·	·	·	·	·	·		·		·	·	·
	3 Condensation due to Pressurisation	·	·	·	·	·	·		·		·	·	·
	4 Condensation due to Heat Removal	·	·	·	·	·	·		·		·	·	·
	5 Interfac. Frict. Vertic. Flow	·	·	·	·	·	·		·		o	·	o
	6 Interfac. Frict. Horiz. Flow	·	·	·	·	·	·		·		·	·	o
	7 Wall to Fluid Friction	·	·	·	·	·	·		·		·	·	·
	8 Press. Drops at Geometr. Discontinuities	·	·	·	·	·	·		·		·	·	o
	9 Pressure Wave Propagation												
1 CRITICAL FLOW	1 Breaks	·	·	·	·	·	·		·		·	·	·
	2 Valves	·	·	·	·	·	·		·		·	·	·
	3 Pipes	·	·	·	·	·	·		·		·	·	·
2 PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL	1 Pipes/Plena	·	·	·	·	·	·		·		o	·	·
	2 Core	·	·	·	·	·	·		·		·	·	·
	3 Downcomer	·	·	·	·	·	·		·		·	·	·
3 STRATIFICATION IN HORIZ. FLOW	1 Pipes	·	·	·	x	·	·		·		·	·	o
4 PHASE SEPARATION AT BRANCHES	1 Branches	·	·	·	x	·	·		·		·	·	·
5 ENTRAINMENT/DEENTRAINMENT	1 Core	·	·	·	·	·	x		·		·	·	·
	2 Upper Plenum	·	·	·	·	·	·		·		·	·	·
	3 Downcomer	·	·	·	·	·	·		·		·	·	·

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SET – facilities characteristics and parameter ranges

TABLE 5.1.1 : PHENOMENON No. 1.1 - CRITICAL FLOW IN BREAKS

FACILITY IDENTIFICATION			KEYWORDS	RELEVANT PARAMETERS RANGES					REASONS FOR SELECTION OR NOTES
No.	Status in the matrix	Name		Nozzle diam. (m)	Length (m)	Pressure (MPa)	Max Flux (Kg/m²s)	Temperature (K)	
3.2	x a	SUPER MOBY DICK	Steady st. critical flow	0.020	0.100	2-8	8430-54100	465-567	2 3 4 6 7
3.3	x a	CANON + SUPER CANON	Adiabatic blowdown	0.03-0.1 diaphragm	-	3.2-15.0	-	470-590	2 3 6
3.4	x a	VERTICAL CANON	Blowdown	0.003-0.015	0.346-0.395	13.0	-	500-590	
3.6	x a	TAPIOCA/CEA FRANCE	Adiabatic blowdown	0.010, 0.020	-	15.0	-	550	2 3 4 6
3.14	x a	OMEGA-TUBE/CEA FRANCE	Blowdown	0.006-0.010	-	16.0	-	560-610	2 3 4 6 7
3.15	x a	OMEGA-ROD BUNDLE CEA/FRANCE	Blowdown 6 x 6 array	0.007-0.023	0.198	13.0, 15.0	-	558	
3.18	x a	SUPER MOBY-DICK TEE/CEA FRANCE	Critical flow phase sepa	0.020	0.082	2.0	318-17500	Saturated	2 8
3.25	x a	REBECA/CEA FRANCE	High qual. Non condensables	0.030	0.100	5.0	-	Saturated	2 3 6 8
4.2	x a	HDR VESSEL	Blowdown in pipes	0.150-0.450	-	4.0-11.0	-	Up to saturated	
4.3	x a	BATTELLE RS16B	Blowdown	0.05-0143	0.350, 0.550	13.7-14.7	-	535-570	
4.4	x a	BATTELLE BWR 150396	Steam line FW line breaks	0.033-0.076	0.500-0.800	5.4-8.8	-	530-575	
4.14	x a	STEAM/WATER DISCH.	Steady state	0.040	0.150	0.1-0.9	-	Up to sat.	
4.16	x a	T-JUNCTION Test Facility (KfK)	Phase redistribution	0.0042-0.05	0.760-3.115	1.5-10.0	-	Saturated	
8.2	x a	MARVIKEN TEST FACILITY	Blowdown	0.200-0.509	0.100	0.5	-	Up to sat.	2 3 4 6

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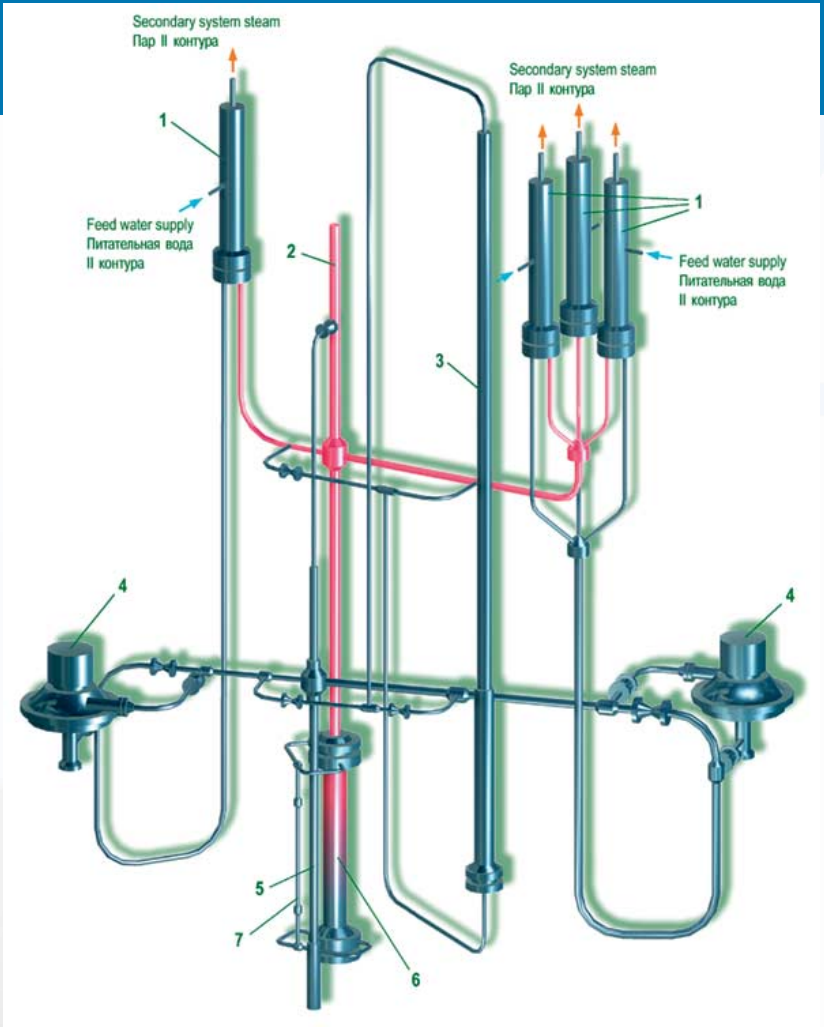
Integral test facilities

- Understanding of physical phenomena on integral level
  - Simulation of the overall facility response
- Validation of code ability to predict:
  - The coupling of complex phenomena
  - The extrapolation from one scale to another
    - – Counter-part tests, similar tests
  - Testing of actions for procedures



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ITFs – U-tube PWRs

- LSTF (Large Scale Test Facility)
  - Operated by JAERI, Japan
  - 4-loop Westinghouse PWR, volume scaling 1:48, height 1:1
  - LOCAs, operational transients, transients at shutdwosn, accident management

Large-scale Test Facility (LSTF): The world's biggest facility to simulate a PWR accident. More than 180 experiments have been conducted using the :LSTF and they confirm that Japanese nuclear power plants have adequate safety margins.

OECD/CSNI ITF CCVM

- Content
  - General considerations
  - Experimental facilities
  - Validation matrices
  - Counterpart tests, similar tests and ISPs
  - TH aspects of SA
  - Appendices
    - Description of test types
    - Characterization of phenomena
    - Information on selected tests
    - Severe accident phenomena

OECD/CSNI ITF CCVM

- PWR
  - Large breaks
  - Small and intermediate breaks, UTSG
  - Small and intermediate breaks, OTSG
  - Transients
  - Transients at shutdown conditions
  - Accident management for a non-degraded core
- BWR
  - Loss of coolant accidents
  - Transient

Matrix II
CROSS REFERENCE MATRIX FOR SMALL AND INTERMEDIATE BREAKS

- **Phenomenon versus test type**
  - + occurring
  - o partially occurring
  - \- not occurring
- **Test facility versus phenomenon**
  - + suitable for code assessment
  - o limited suitability
  - \- not suitable
- **Test type versus test facility**
  - + performed
  - o performed but of limited use
  - \- not performed or planned

		Test Type							Test Facility and Volumetric Scaling								
		Stationary test addressing energy transport on primary side	Stationary test addressing energy transport on secondary side	Small leak overfeed by HPIS, secondary side necessary	Small leak without HPIS overfeeding, secondary side necessary	Intermediate leak, secondary side not necessary	Pressurizer leak	U-tube rupture	PWR 1 : 1	LOFT 1 : 50	LSTF 1 : 48	BETHSY 1 : 100	PKL-III 1 : 134	SPES 1 : 430	LOBI-II 1 : 712	SEMISCALE 1 : 1600	UPTF, TRAM 1 : 1 (b)
Phenomena (c)	Natural circulation in 1-phase flow , primary side	+	+	+	o	-	+	+	+	+	+	+	+	+	+	+	-
	Natural circulation in 2-phase flow , primary side	+	-	o	+	+	o	-	-	+	+	+	+	+	+	+	o
	Reflux condenser mode and CCFL	+	-	-	+	+	-	-	-	o	+	+	o	o	o	o	+
	Asymmetric loop behaviour	-	-	+	+	-	o	+	-	-	o	+	+	+	o	o	+
	Break flow	-	-	+	+	+	+	+	-	+	+	+	+	+	+	+	o
	Phase separation w ithout mixture level formatior	+	-	o	+	+	+	o	-	o	+	+	+	+	+	o	+
	Mixture level and entraiment in SG second side	-	+	+	+	+	+	+	-	-	+	+	+	o	o	-	-
	Mixture level and entraiment in the core	+	-	-	+	+	+	-	-	o	+	+	+	o	o	o	o
	Stratification in horizontal pipes	+	-	-	+	+	-	-	-	+	+	+	+	+	o	o	+
	Phase separation in T-junct. and effect on break	-	-	-	+	+	-	-	-	o	o	o	o	o	o	-	+
	ECC-mixing and condensation	-	-	o	+	+	+	+	-	o	o	o	o	o	o	o	+
	Loop seal clearing	-	-	-	+	+	o	-	-	+	+	+	+	+	+	+	+
	Pool formation in UP/CCFL (UCSP)	+	-	-	o	+	+	-	-	o	o	o	o	o	-	o	+
	Core w ide void and flow distribution	+	-	-	o	+	+	-	-	o	o	o	o	-	-	-	o
	Heat transfer in covered core	+	+	+	+	+	+	+	o	+	+	+	+	+	+	+	-
	Heat transfer in partly uncovered core	+	-	-	o	+	-	-	-	+	+	+	+	o	o	o	-
	Heat transfer in SG primary side	+	o	o	+	+	o	o	-	o	+	+	+	+	+	o	-
	Heat transfer in SG secondary side	o	+	+	+	+	+	+	-	o	+	+	+	o	+	o	-
	Pressurizer thermohydraulics	o	-	o	o	+	+	+	o	o	o	o	o	o	o	-	+
	Surgeline hydraulics	o	-	-	o	+	+	o	-	o	o	o	o	o	o	o	+
	1- and 2-phase pump behaviour	-	-	-	o	+	-	-	o	o	o	o	o	o	+	+	-
	Structural heat and heat losses (a)	+	-	o	+	+	o	o	-	o	o	o	o	o	o	o	o
	Noncondensable gas effects	+	-	-	-	-	-	-	-	-	+	+	+	-	-	+	o
	Boron mixing and transport	+	-	+	+	+	+	+	-	-	-	-	-	-	-	-	o
Test Facility	PWR	-	-	o	-	-	+	+									
	LOFT	-	-	+	+	+	+	-									
	LSTF	+	+	+	+	+	+	+									
	BETHSY	+	+	+	+	+	+	+									
	PKL-III	+	+	+	+	+	+	+									
	SPES	+	+	+	+	-	-	-									
	LOBI-II	+	+	+	+	+	+	+									
	SEMISCALE	o	o	+	+	+	+	+									
	UPTF, TRAM	-	-	-	-	+	+	-									

(a) problem for scaled test facilities

(b) UPTF integral tests

(c) for intermediate breaks phenomena included ir large break reference matrix may be also importar

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Basis of selection of experiments

- Each phenomenon should be addressed in test facilities of different scale
- All test types should be included
- Typicality of facility and experiments to expected reactor conditions
- Quality and completeness of experimental data (measurements and documentation)
- Relevance to safety issues
- Test selected must clearly exhibit phenomena
- Each phenomenon should be addressed by tests of different scaling (at least one test if possible)
- High priority to ISPs, counterpart and similar tests
- Challenge to system codes

Scaling issues

- Experimental data are applicable to the prototype system if the test facilities and BIC of the experiments are scaled properly
  - Scaling distortions or biases are unvoidable and their impact must be evaluated – understanding of essential physics (correct scaling of the significant areas of concern is much more important than the total number of potential concerns)
  - Dimensional analysis and similitude
  - Scaling uncertainties are reduced by employing as large a scale as practical and/or by performing counterpart tests

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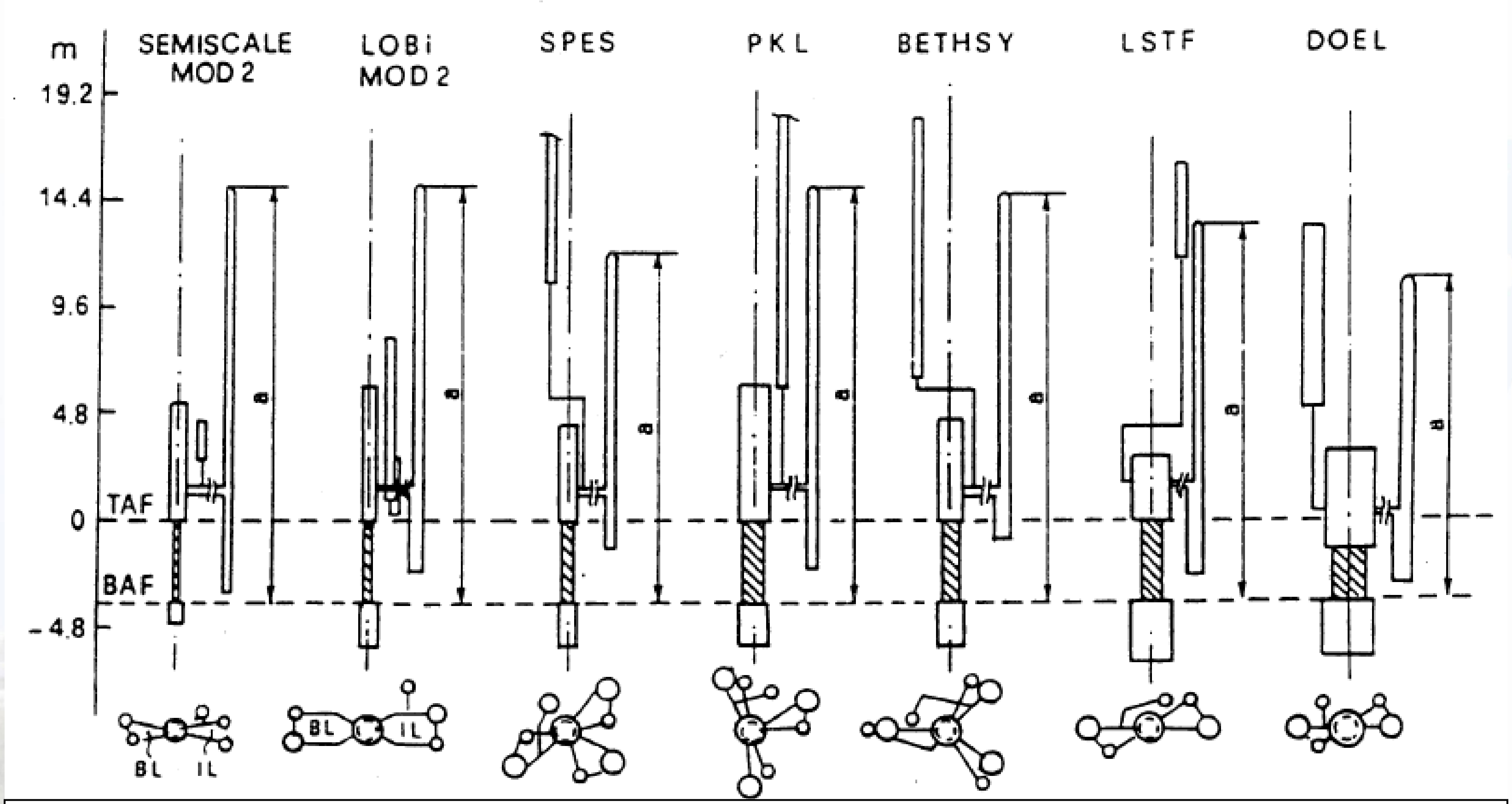
Counterpart tests and similar tests

- Validation performed over wide range of test data from test facilities of different scaling ratios and / or design concepts
- Counterpart tests:
  - Tests specified as counterpart tests with scaled initial and boundary conditions
  - No significant influences of facility distortions, e.g.
    - Geometric dimensions or arrangements of components
    - Downcomer configuration (annulus, pipe(s))
    - Heat losses
    - Heater rods (nuclear vs. electric heating, gaps or not)
    - Bypass flows
- Similar tests:
  - Do not meet conditions for counterpart tests but maintain similar initial and boundary conditions and can reveal important scaling influences

Counterpart tests and similar tests



Semiscale 1:1705 LOBI 1:700 SPES 1:427 PKL 1:145 BETHSY 1:100 LSTF 1:48 Doel 1:1

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Counterpart tests - PWR

	Semi Scale	LOBI	SPES	PKL III	BETHSY	LOFT	LSTF
Reference Reactor Power (MW)	W 3400	KWU 3800	W 2775	KWU 3800	Framatome 2700	W 3250	W 3420
Primary System Volume (m^3)	0.195	0.600	0.630	2.4	2.88	7.22	8.3
Reference Scaled Nominal Power (MW)	2.	5.3	6.5	28.36	27.	50	77.5
Reactor Power/ Ref. Scaled Nom. Power	1700	712	427	134	100	65	48
Large Break LOCA	**S-06-3**	**A1-04**				**L2-3**	
25% Break LOCA	**S-IB-3**	**B-R1M**					
CL Small Break LOCA 5 %	**S-LH-1**						**SB-CL-06** **SB-CL-10**
CL Small Break LOCA 6 % at Low Power		**BL-34**	**SP-SB-03**		**6.2 TC**		**SB-CL-21**
CL Small Break LOCA 6% at Full Power		**BL-44**	**SP-SB-04**				
Cooldown under Natural Circulation		**A1-87**		**A2.1**			
Natural Circulation at 40 bar		**A1-92**		**AC-1**			
Two-phase Nat. Circulation					**4.1aTC**		**ST-NC-06/07**

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Similar tests - PWR

	LOBI	SPES	PKL III	BETHSY	LOFT	LSTF
Reference Reactor Power (MW)	KWU 3800	W 2775	KWU 3800	Framatome 2700	W 3250	W 3420
Primary System Volume (m^3)	0.600	0.630	2.4	2.88	7.22	8.3
Reference Scaled Nominal Power (MW)	5.3	6.5	28.36	27.	50	77.5
Reactor Power/ Ref. Scaled Nom. Power	712	427	134	100	65	48
Large Break LOCA MCPs on/off	**A1-72 S1-66**				**L2-3 L2-5**	
LOFW with Secondary Side F&B	**BT-17**	**SP-FW-02**	**B.1.2**			**TR-LF-04**
LOFW with Primary Side F&B	**BT-02**			**5.2 C**		**TR-LF-07**
Natural Circulation			**B.3.2.B**			**ST-NC-08**

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Counterpart tests - BWR

	ROSA_III	**FISTI**	**TLTA**	**Piper-One**
Reference Reactor Power (MW)	BWR-6 3150	BWR-6 3150	BWR-6 3150	BWR-6 3150
Primary System Volume (m^3)	1.418	0.712		0.199
Reference Scaled Nominal Power (MW)	10.1	5.1		1.42
Reactor Power/ Ref. Scaled Nom. Power	310	618	624	2210
3 % break in recirulation line	**Test 984**	**6SB2C**	**6432/R**	**PO-SB-7**

ISPs

#	Date	Title
1	1975	Edwards' Pipe Blowdown test
2	1975	Semiscale Blowdown Test 11
3	1977	Comparison of LOCA Analysis Codes, CISE, Blowdown
4	1978	Semiscale MODI Test S-02-6
5	1979	LOFT Test L1-4 (isothermal non-nuclear blowdown)
6	1978	Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase
7	1979	Analysis of a Reflooding Experiment, ERSEC
8	1979	Semiscale MODI Test S-06-03 (LOFT Counterpart Test)
9	1981	LOFT Test L3-1
10	1981	PKL-1-K9 Test (Refill and Reflood)
11	1984	LOFT L3-5 and L3-6 Tests
12	1982	ROSA-III 5 % Small Break Test, Run 912
13	1983	LOFT Experiment L2-5 (Large Break LOCA)
14	1985	Behaviour of a Fuel Bundle Simulator during a Specified Heatup and Flooding Period (REBEKA Experiment)
15	1983	FIX-II Experiment 3025 (31 % LOCA)
16	1985	Rupture of a Steam Line within the HDR Containment Leading to an Early Two-Phase Flow
17	1984	Marviken BWR Standard Problem
18	1987	LOBI-MOD2 Small Break LOCA Experiment A2-81
19	1987	Behaviour of a fuel rod Bundle during a large break LOCA transient with a two-peaks temperature history (PHEBUS Experiment)
20	1988	DOEL 2 Steam Generator Tube Rupture Event
21	1989	Piper-One, Test PO-SB-07
22	1990	SPES – Loss of Feedwater Transient, Test SP-FW-02
23	1989	Rupture of a large diameter pipe in the HDR containment
24	1989	SURC-4 - Core-Concrete Interaction Test

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ISPs

#	Date	Title
25	1991	ACHILLES - N2 injection from accumulators and faster (best estimate) reflood rates
26	1992	ROSA-IV LSTF Cold-Leg Small-Break LOCA Experiment, SB-CL-18
27	1992	BETHSY – Small Break LOCA with Loss of HP Injection, 9.1b
28	1992	PHEBUS SFD B9+ - Experiment on the Degradation of a PWR Type Core
29	1993	HDR Experiment E11.2 - Hydrogen distribution inside the HDR containment under severe accident conditions
30	1992	BETA II Core-Concrete Interaction Experiment (Test V5.1)
31	1993	CORA-13 Experiment on severe Fuel Damage
32	-	FLHT-6 Experiment, cancelled
33	1992	PACTEL – WWER-440 Natural Circulation Test Behavior ITE-06
34	1994	Falcon Experiments FAL-ISP-1 and FAL-ISP-2, Fission product transport
35	1994	NUPEC Hydrogen Mixing and Distribution Test M-7-1
36	1996	CORA-VVER Severe Fuel Damage Experiment (Test W2)
37	1996	VANAM M3-A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material
38	1997	Loss of the Residual Heat Removal System during mid-loop operation (BETHSY)
39	1997	Fuel Coolant Interaction and Quenching (FARO)
40	1999	STORM Test SR11 - Aerosol Deposition and Resuspension in the Primary Circuit
41	1999	RTF Experiment on Iodine Behaviour in Containment Under Severe Accident Conditions
42	2003	PANDA tests (six different phases) related to passive safety systems for Advanced Light Water Reactors
43	2001	UMCP Boron dilution test
44	2002	KAEVER aerosol depletion tests with three differently soluble materials and uniform thermal-hydraulic conditions with slight volume condensation
45	2003	QUENCH-06, Fuel rod bundle behaviour up to and during reflood/quench (severe core damage)
46	2004	PHEBUS in reactor experiment (FP-1) on the degradation, fission product release, circuit and containment behaviour following overheating of an irradiated fuel rod bundle
47	2005	Based on experiments performed in the TOSQAN, MISTRA and ThAI facilities for containment thermalhydraulics
48	2005	Containment capacity (Integrity and Ageing of Components and Structures). 1:4 scale model of a pre-stressed concrete containment vessel (PCCV) of a nuclear power plant (SANDIA II mock-up)

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Appendices

- Appendix A – Description of test types
  - Description of test types for PWRs
    - Classification of scenarios listed in the cross reference matrix
  - Description of test types for BWRs
    - Classification of scenarios listed in the cross reference matrix
- Appendix B – description of phenomena
  - Phenomena listed in the cross reference matrix are described in Appendix B.
  - These descriptions are intended to provide a common basis for understanding and interpretation
  - Relevance to nuclear reactor safety
- Appendix C – information on selected tests
  - Test Conditions
  - Major Phenomena
  - References

CCVM for WWERs

- Supplement to the existing OECD Integral (IT) and Separate Effects Test (SET) Validation Matrices
- Consideration of specific features of WWER-440 and WWER-1000 systems and their behavior in normal and abnormal situations
- Enlargement of experimental data base for code assessment, not taken into account in the previous OECD reports

CCVM for WWERs

- WWER Matrices reports contains
  - Large break LOCAs
  - Small and intermediate break LOCAs
  - Transients
- Phenomena identified relevant for WWER primary and secondary systems during LOCAs and transients
- Phenomena of WWER compared with Western PWR and similarities clarified
- Phenomena described in detail as basis of common evaluation and assessment by experimental data
- Facilities and experiments (ITs and SETs) identified that supplement the CSNI Validation matrices and are suitable for WWER specific code assessment

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NEA Databank

- Up to now 71 experiments stored
- Some more expected
- CERTA-TN (European Thematic Network for the Consolidation of the Integral System Effect Experimental Data Bases for Reactor Thermal Hydraulic Safety Analysis) for data collection and data bank (2000-2002)

http://www.oecd-nea.org/databank/

http://www.oecd-nea.org/dbprog/ccvm/

http://www.oecd-nea.org/dbprog/ccvm/indexset.html

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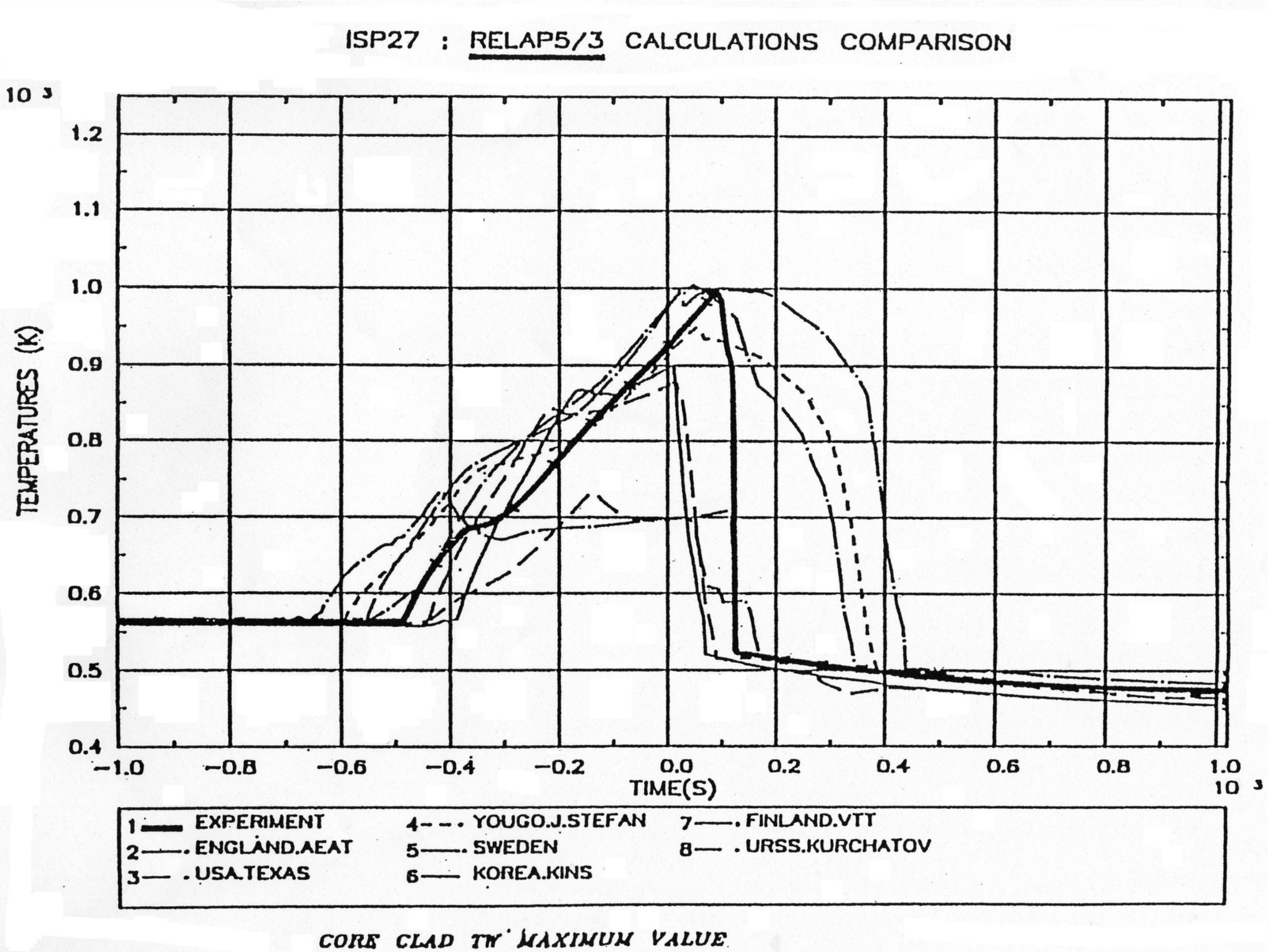
Application of CCVM for user training

- User effect
  - To preserve the code wide applicability and flexibility (different reactor types, different transients) user must make many choices in setting up an input deck and in running a calculation
    - System noding and flow paths
    - Heat structure distribution
    - Material properties
    - Additional options
    - Time step controls
    - ...
- Systematic user training and mentoring

ISP27 Blind test calculation



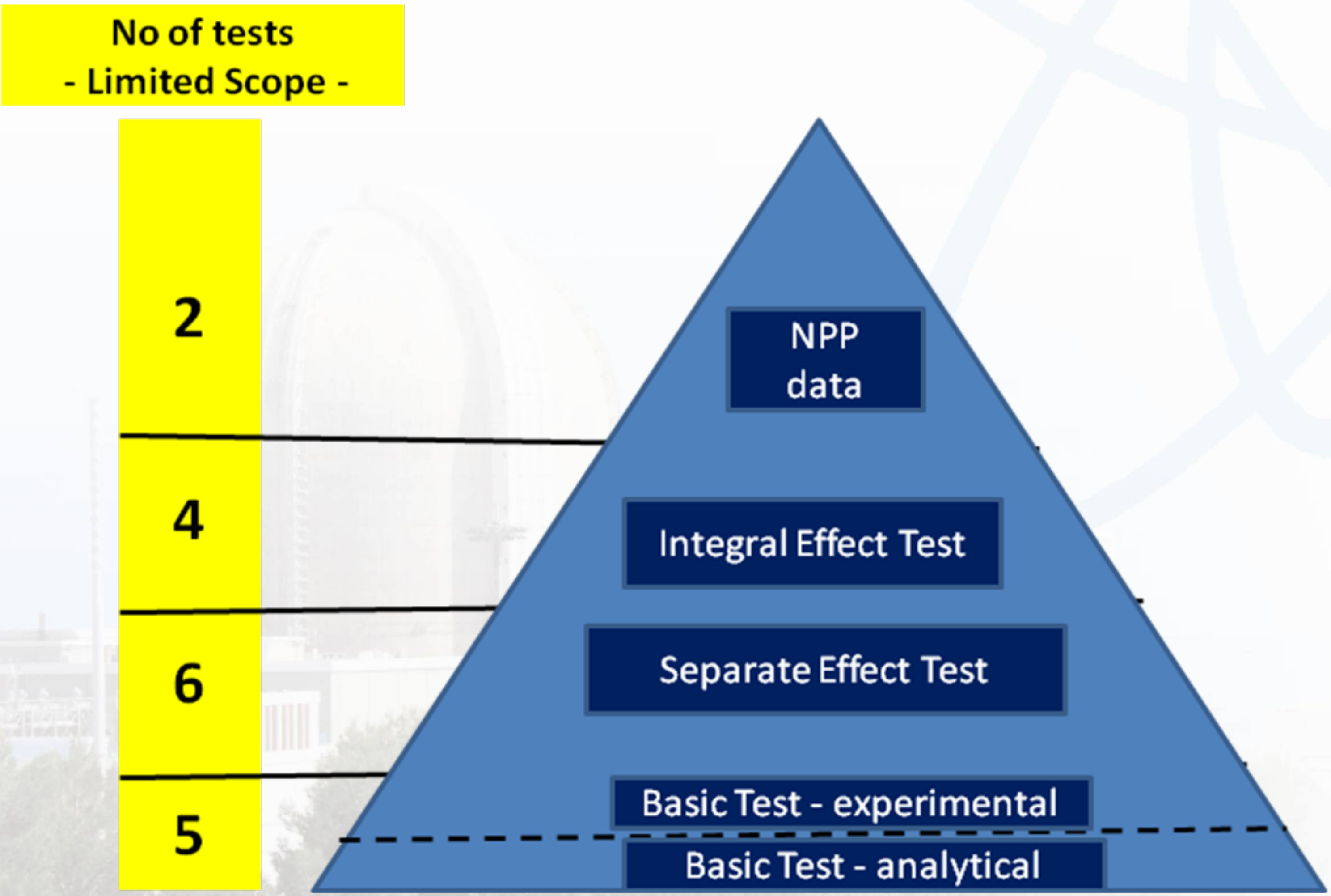
Exercise matrix

- IAEA activity to create thermal-hydraulic test matrix suitable for training and transfer of expertise in the sector of water-cooled nuclear reactors
- Proposed matrix consists of twelve tests that can be analyzed by a single trainee or by a (small) group of trainees
- Scope restricted to PWRs with U-tubes SGs

Exercise matrix

- The modular structure for the overall matrix allows application for training of personnel at different competency levels as well as for interlinking the training to other disciplines like neutron physics and nuclear fuel or to expanding the training scope to other reactor types such as BWR, VVER, CANDU and including reactors with passive systems

Exercise pyramid



List of exercises - draft

Test designation	Description
Basic analytical and experimental tests	
Analytical	
Pressurizer	Condensation phenomena due to pressurization and flashing due to depressurization effect
Experiments	
NCSU1966	Pressure drop in two phase flow
Bennett heated tube	Dry-out, CHF
Separate Effects Tests	
Set 1	
Super Moby-Dick steady state	Two phase flow phenomena in steady state
Edwards pipe	Blowdown, flashing, voiding
Set 2	
ORNL-THTF	CHF, DNBR
Set 3	
Creare	CCFL
Takeuchi	CCFL
ORNL	CCFL
UPTF	CCFL
Set 4	
Neptun	Reflood
Pericles	Reflood
FLECHT-SEASET	Reflood at low and high reflood rates, boil off
Integral Effect Tests	
LOFT L2-5	LB LOCA
LOBI SBLOCA 2”	SB LOCA
LOBI BL-21	SGTR
LOFT L9-3	ATWS
Nuclear Power Plant	
Zion LBLOCA	LB LOCA
Mihama	SGTR

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Safety analysis process

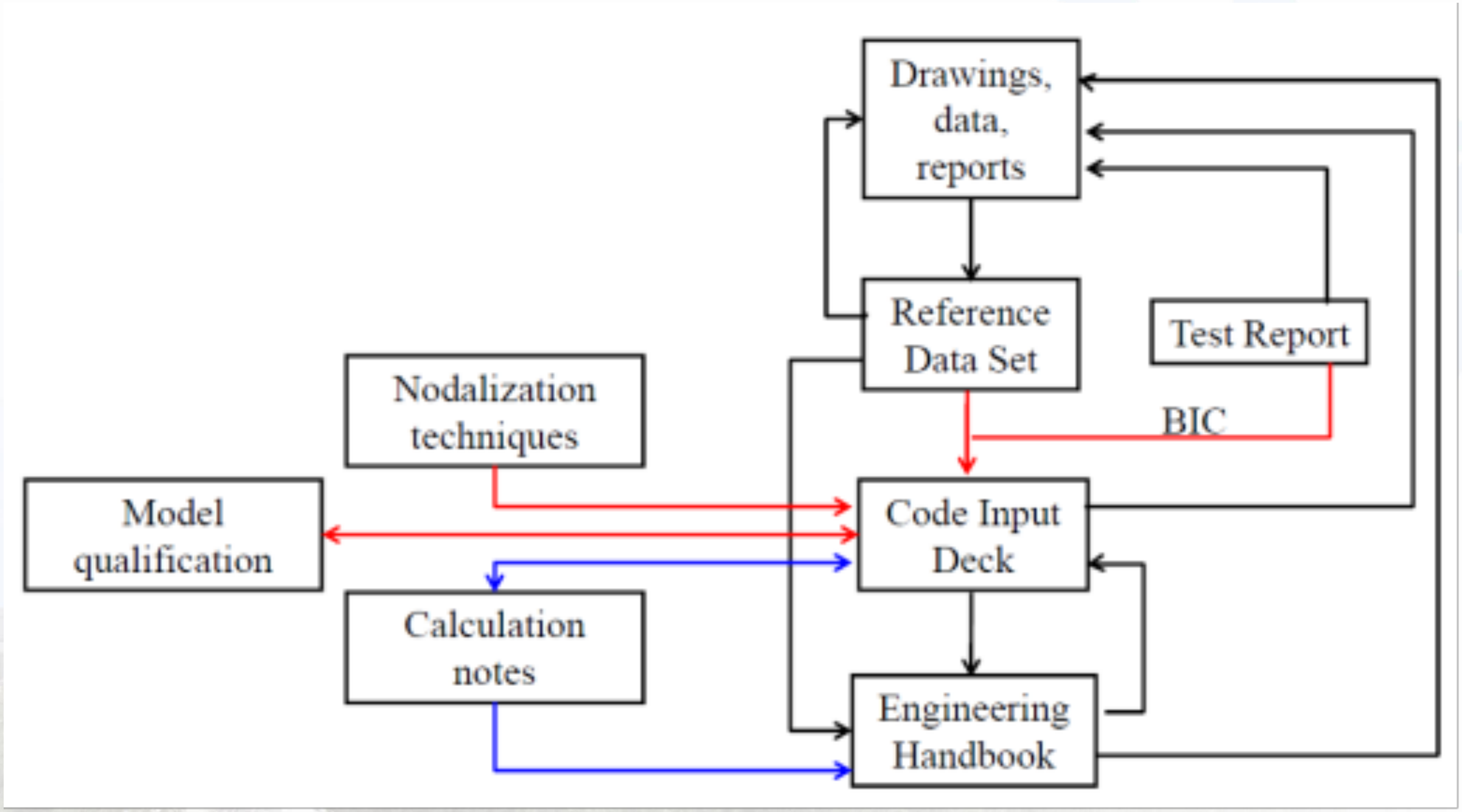
- Objective of the analysis
  - o What facility and which of its systems and components are included?
  - o What transients and accidents will be simulated?
  - o What phenomena are expected to occur?
- Selection of the appropriate computer code
  - o RELAP5 -> best estimate analysis of small and medium break LOCAs and plant transients in light water reactors

Safety analysis process

- Collection of facility data
- Database (code independent)
- Development of the input data deck including calculation notebook
- Verification and validation (qualification) of the input model
- Engineering handbook (code specific)

- Preparation of the scenario
- Execution of the calculation
- Checking of the results
- Presentation of the results

Model development



Input model preparation

Pressurizer

The water volume of the pressurizer is connected via a T element
parallel lines to the hot leg of the main circulation loop No. 1. The steam v
connected via a pressurizer spray line to the cold leg of the same l
connections are on the non-isolable part of the main circulation loop.

Pressurizer pressure vessel and internals:

- Internal diameter 2.382 m [D39
- Internal heights:
  - — Inner height of cylindrical part 8.79 m [D39]
  - — Elliptical head at bottom 2×0.701 m [
  - — Total inner height of pressurizer 10.192 m [D3
- Wall thickness (including inner lining 9 mm thick):
  - — Cylindrical part above pressurizer heaters 0.154 m [D39
  - — Cylindrical part on the level of pressurizer heaters 0.199 m [D39
  - — Elliptical head and bottom 0.169 m [D39
- Total internal volume 44.0 m^3 [R28
- Thickness of the insulation 0.23 m [D43]
- Height of the spray above the pressurizer bottom 9.392 m [D40
- Elevations of heaters above pressurizer bottom (on four levels separated from each other by 0.31 m) 1.111–2.041 m
- Basic material of the pressurizer vessel Steel 22K [R
- Mass of the pressurizer vessel (without coolant) · 127 300 kg [I
- Total power of the pressurizer heaters 1620 kW [R4

Volume Name	Lower plenum boxes
Component Type	BRANCH

Geometrical data	Calculation V
L	V17
0,377	0,269

The input of volume is omitted (0.0), so it is automatically calculated by the coc
the product of flow area A by length L must be equal to the volume. Hydraulic d
according to

$$\Phi_{hyd} = 4\frac{A}{p}$$

Wall roughness is assumed equal to 6.0e-5 m.

Note	L	Lower plenum box length

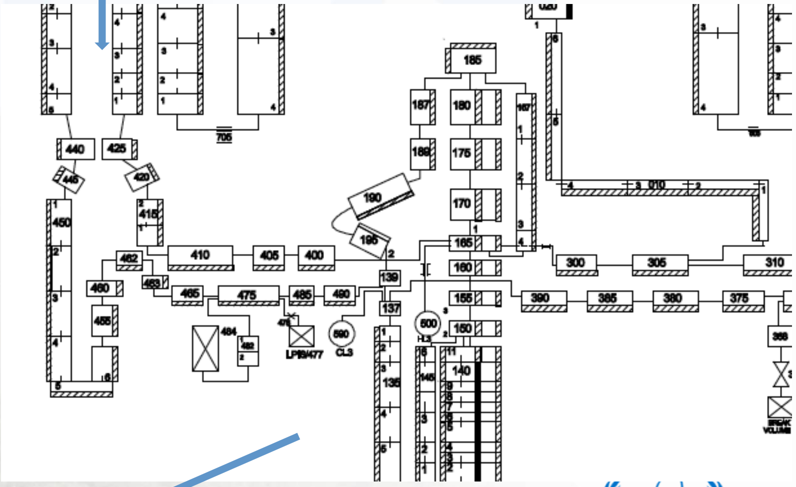
Volume Name	Lower plenum boxes
Component Type	BRANCH

Volumes

	Flow Area	Length	Volume	Inclination	Elevation
1	0,714	0,377	0,269	90,0	0,377

Junctions

	Type	From Vol.	To Vol.	Flow Area	Kf
1	Normal	244-1	17-0	0,0	0,00
2	Normal	17-1	530-0	0,0	0,00
3	Normal	17-1	251-0	0,048659	0,00



bethsy.i

```
0401101 040000000 030010000 0. 1.e-6  1.e-6  00001000
*
* prez level control j
0450000 prz.lec   tmdpjun
0450101 050000000 020010000 0.
*
* prez lvl control vol
0500000 prz.cvvo    tmdpvol
0500101 0. 10. 10. 0. 90. 10. 5.0e-5 0. 0000000
*
* steady state pressure control
0600000 pre.sts  tmdpvol
0600101 0.0121 2. 0. 0. 0.   0. 5.0e-5 0. 0000000
*
* tmdp conn valve to prez
0650000 pr.tmv  valve
0650101 060000000 040010000 0.01   1.e-6 1.e-6 0001100 1. 1.
```

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Input model qualification