VTT

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APROS VALIDATION

SELECTED VALIDATION CASES RELATED TO NUCLEAR SAFETY ANALYSES
and
TRAINING SIMULATORS

Used during code development

1. THERMAL HYDRAULIC SEPARATE EFFECT TESTS **1.1 Homogeneous model**	MAIN PARAMETERS AND PHENOMENA
Hungarian PMK test facility (PWR): small break LOCA	Critical flow, primary/secondary pressure relationship during SBLOCA, two-phase natural circulation, core overheating
Edward's pipe: blowdown of horizontal pipe	Critical flow, fast depressurization during rapid blowdown with delayed flashing
Top blowdown test facility (Battelle Frankfurt), OECD ISP-6, steam blowdown	Depressurization of a vessel, phase separation below swell level, critical flow from single phase steam to two phase mixture
Marviken critical flow test MXC-17	Critical flow, pressure distribution in a large diameter blowdown pipe
FRIGG-loop, two-phase heat transfer	Two phase heat transfer during flow boiling
Christensen experiment, subcooled boiling	Boiling heat transfer with enthalpy non-equilibrium
1.2 Two-phase model (five equation model)	
Marviken critical flow test MXC-17	Critical flow, pressure distribution in a large diameter blowdown pipe
IVO large scale loop seal experiment	Loop seal effect in a full scale experimental facility, stratification in a horizontal pipe (air/water experiment)
FRIGG-loop, two-phase heat transfer	Two phase heat transfer during flow boiling
Christensen experiment, subcooled boiling	Boiling heat transfer with enthalpy non-equilibrium
1.3 Two-phase model (six equation model)	
Edward's pipe: blowdown of horizontal pipe	Critical flow, fast depressurization during rapid blowdown with delayed flashing
Top blowdown test facility (Battelle Frankfurt), OECD ISP-6, steam blowdown	Depressurization of a vessel, phase separation below swell level, critical flow from single phase steam to two phase mixture
IVO large scale loop seal experiment	Loop seal effect in a full scale experimental facility, stratification in a horizontal pipe (air/water experiment)
Becker's dryout test	Flow boiling, high quality dryout and enthalpy non-equilibrium in post dryout heat transfer regime

Ersec reflood test: OECD ISP-7	Reflooding, heat transfer in quenching, quenching front propagation, effect of axial heat conduction
Marviken critical flow test MXC-17	Critical flow, pressure distribution in a large diameter blowdown pipe
FRIGG-loop, two-phase heat transfer	Two phase heat transfer during flow boiling, Interfacial friction, Acceleration, hydrostatic and friction pressure drop distributions, void fraction distribution
Christensen experiment, subcooled boiling	Boiling heat transfer with enthalpy nonequilibrium
UPTF-loop seal experiment (integral experiments it1b and it2a)	Loop seal effect in a full scale reactor geometry, stratification in a horizontal pipe (four loop configuration, 1.5 MPa)
UPTF-loop seal experiment (separate effects tests 01a, 02a, 03a, 04b, 05a, 07a, 08b, 09d, 10e, 11d, 11e)	Loop seal effect in a full scale loop seal, stratification in a horizontal pipe (single loop configuration, 0.3 and 1.5 MPa)
IVO CCFL Experiment	Counter-current flow limitation, interface friction, perforated plate, fuel bundle
REWET-II, reflood test SGI/6, 19 rods	Reflooding, heat transfer in quenching, quench-front propagation, effect of axial heat conduction
LOTUS annular flow experiment	Pressure loss in vertical annular air/water flow
NOKO emergency condenser experiments	Condensation in horizontal tubes
PANDA isolation condenser experiments	Steam and steam/air mixture condensation in vertical tubes
PANTHERS full scale condencer experiment	Steam and steam/air mixture condensation in vertical tubes and heat transfer in a large pool (ICONE-8)
PACTEL pressurizer experiments ATWS 10-13, ATWS 20-21	Compression and expansion of steam, wall condensation and effect of spray.
MIT pressurizer	Compression and expansion of steam, wall condensation.
NEPTUNUS pressurizer experiments	Compression and expansion of steam, wall condensation and effect of spray.
LOVIISA turbine trip, only pressurizer modelled.	Compression and expansion of steam, wall condensation and effect of spray.
UPTF CCFL tests for downcomer and core upper tie plat	Limitation of counter current flow in large scale reactor pressure vessel downcomer and in core upper tie plate.

2. INTEGRAL TESTS **2.1 Homogeneous, 5- and 6-equation thermal hydraulic models, 1-dimensional reactor model and the Loviisa plant model** **Loviisa nuclear power plant (compared to plant data):**	
Steady state (@ 1375, 1500, 1530 MWth)	Capability of reproducing normal operating conditions at different power levels
Natural circulation on various power levels (0.5,1.0,...4.0%)	Coolant flow and temperature distribution during natural circulation
Stepwise load change test (set point of turbine power is changed)	Response of the reactor power caused by coolant temperature induced reactivity change
Reactor trip	Coolant temperatures (hot, cold leg), primary pressure and pressurizer level response, effect of the upper head liquid temperature, secondary pressure and steam generator level in rapidly changing conditions
Turbine trip	Plant response to closing of turbine valve and opening of the turbine bypass line
Trip of one feed water pump	Start-up procedure of the auxiliary feed water pump, plant controller behaviour, if pump does not start
Trip of two primary coolant pumps	Automatic reactor power control with slow shutdown, loop temperatures, setup of reverse loop flow conditions in two loops
Feedwater line break	Partial loss of feedwater, dynamic response of plant safety systems, dynamic behaviour of the feedwater line system during blowdown
Several plant regulation tests	Plant control and protection system behaviour
Blackout test from primary circuit point of view	Transition to natural circulation
High pressure preheater system	Design of a high pressure preheater system, design and preconditioning of control system, dynamic tests
Primary circuit overcooling transient	Turbine bypass valve capacity for reactor system cooldown after reactor trip, primary circuit repressurization due to HPIS startup, pressurizer spray characteristics, pressurizer water level response to cooling, system pressure response to pressurizer level increase
Loviisa nuclear power plant, automation system (compared to Loviisa training simulator data):	
Movement of control rod group	Reactor power and power distribution response to control rod movement, primary circuit response to reactor power change

Small break LOCA	Single and two phase natural circulation characteristics, use of secondary side pressure control to accident management, gradual depletion of the secondary side water inventory, transition to loss-of-feed-water ATWS, primary system dynamics response to heat transfer decrease, energy removal through the break
Loss of feedwater transient (ATWS)	High pressure behaviour of the primary circuit after depletion of the secondary water mass and trip of reactor coolant pumps, two phase natural circulation and reflux cooling of the primary system
Turbine trip with steam dump to condenser	Plant response to turbine bypass opening, plant controller behaviour
Control rod withdrawal (ATWS)	Primary and secondary circuit response to overpower, gradual transition to a loss of feed water incident due to insufficient feed water injection rate, plant controller behaviour
Opening of one pressurizer safety valve	Primary and secondary circuit response to slow pressure decrease
2.2 Homogeneous and 5-equation thermal hydraulic models, TVO nuclear power plant (compared to accident analysis code results (GOBLIN, BISON)	
Steam line break	Steam flow out through the end of steam line, boiling water reactor vessel and steam line dynamics, pressure and water level behaviour in transients, response of reactor power to core inlet flow changes, feed-water and turbine flow controller dynamics
Simultaneous closure of all steam line isolation valves	Pressure and reactor power peaking due to the abrupt steam flow change
Loss of feed water	Plant system response to the delayed reactor scram
2.3 Homogeneous and 5-equation thermal hydraulic models, automation system, Forsmark 3 nuclear power plant model (compared to operational instructions, plant data and plant simulator data)	
Steady state (@ 65, 100, 109 % of nominal power)	Plant data from F3
Plant startup (cold shutdown > 109 %)	Operating procedures
Plant shutdown (109 % > cold shutdown)	Operating procedures
2.4 Homogeneous and 5-equation thermal hydraulic models, automation system, KOLA nuclear power plant	

Erroneous opening of pressurizer safety valve (KOLA 3)	Primary and secondary circuit response to slow pressure decrease
Trip of two main primary coolant pumps (KOLA 4)	Automatic reactor power control with slow shutdown, loop temperatures, setup of reverse loop flow conditions in two loops
2.5 Homogeneous, 5- and 6-equation thermal hydraulic models, LOFT test facility model	
Small break LOCA L3-6 (2.5%) with pumps running	Slow primary loop depressurisation, pump behaviour in two phase flow conditions, two phase heat transfer in steam generators
Small break LOCA LP-SB-03, cold leg break (5-6 kg/s)	Critical flow, two phase heat transfer in steam generator, pump behaviour in two phase flow, core uncovery and reflooding under high pressure, feed and bleed cooling
Medium size LOCA L5-1, cold leg break (110 kg/s)	Critical flow, coolant redistribution in primary circuit, core uncovery and reflooding
Large break LOCA L2-5, 200% double ended break in cold leg	Critical flow, coolant redistribution in primary circuit, core uncovery and reflooding under low pressure, accumulator and LPIS cooling
2.6 5- and 6-equation thermal hydraulic model, PACTEL test facility	
Natural circulation experiment (ISP-33)	Natural circulation as a function of primary coolant mass inventory in a horizontal steam generator, including single phase and two phase natural circulation and reflux boiling (SG scaled according to tube lengths)
SBLOCA experiment, hot leg loop seal behavior (SBL-22)	As above, SG scaled according to SG height, single loop, pressurizer isolated, opening of the loop seal with a pressure peak
SBLOCA experiment, hot leg loop seal behavior (SBL-30)	As above, three loops, pressurizer isolated, asymmetric behavior of the loops, opening of the loops seal(s) with a pressure peak
SBLOCA experiment, validation of the accumulator models (SBL-31)	As above, three loops, pressurizer included, ECC water from two accumulators
SBLOCA experiment, validation of the accumulator and HPI models (SBL-33)	As above, ECC from accumulators and HPIS, boron dilution during boiler-condenser natural circulation
Single phase flow instabilities (CMP-04)	Oscillations of single phase natural circulation during compensated leakage situation (single loop experiment, undamping oscillations)
Single phase flow instabilities (CMP-08)	As above, three loop experiment, damping oscillations
Single phase flow instabilities (CMP-09)	As above, three loop experiment, undamping oscillations

Horizontal steam generator behavior (SG-2, SG-3, SG-4)	One loop operation, heat transfer in primary and secondary side with different secondary side water levels, mixture level behavior in the secondary side pool
Loss of feedwater, horizontal steam generator behaviour (LOF-10)	One loop operation, heat transfer in primary and secondary sides with different secondary water levels, water circulation in primary side tubes (low level)
Gravity Driven Core Cooling test (GDE-11)	Passive safety injection system behaviour, CMT with two pressure balancing lines, rapid condensation in CMT, thermal stratification in CMT
Gravity Driven Core Cooling test (GDE-24)	Passive safety injection system behaviour, CMT with pressure balancing line to cold leg only, condensation and thermal stratification in CMT
Gravity Driven Core Cooling test (GDE-34)	As GDE-24 , CMT initially full of hot water, start up of natural circulation through passive safety injection system with small driving force
Gravity Driven Core Cooling test (GDE-41)	As GDE-24 , 1.5 % break in cold leg
Gravity Driven Core Cooling test (GDE-43)	As GDE-24 , 0.1 % break in cold leg
ATWS-10	Compressibility of steam in the top of pressuriser, pressuriser heating
Dissolved gas behaviour in Pactel facility during LOCA	Comparison of tank and node based models for dissolved gas release during LOCA
2.7 5- and 6-equation thermal hydraulic model, BETHSY test facility	
Loss of Residual Heat removal during mid-loop operation (BETHSY test 6-9C)	• Low pressure range in primary loop • Phase separation by large velocity differences (0-20 m/s) in vertical, inclined and horizontal pipes • Stratification in horizontal hot leg and cold leg pipes • Counter current flow in vertical steam generator • Phase separation in core in low pressures • Liquid holdup in pressurizer during counter-current flow
2 inch cold leg break in a PWR with vertical steam generators and with high pressure safety injection assumed to be unavailable. Low pressure safety injection actuated assisted by the secondary pressure reduction (BETHSY test 9.1b)	• Small break LOCA chain in a Western commercial PWR • Critical break flow in the orifice with subcooled liquid, saturated liquid, two-phase mixture and single phase steam upstream the break • Core uncover • Core quenching by level swell after pressure drop • Final core quenching by low pressure injection • Vertical steam generator (process component) in accident and transient conditions

	• Stratified flow in hot and cold leg • Clearance of hot leg and cold leg loop seals • Counter-current flow in steam generator tubing
2.8 6-equation thermal hydraulic model, PSB-VVER test facility	
OECD PSB-VVER analytical exercise	Primary to secondary leakage (PRISE), SG collector cover lift up
2.9 6-equation thermal hydraulic model, PKL III test facility	
E2.2	SBLOCA, Boron dilution
E3.1	Mid loop operation, Boron dilution
2.10 6-equation thermal hydraulic model, ROSA test facility	
OECD/ROSA research program test 6.1	SBLOCA in pressure vessel upper head
OECD/ROSA research program test 6.2	SBLOCA in pressure vessel bottom

3. CONTAINMENT **3.1 Containment integral tests**	
Marviken full scale suppression pool containment experiment BD 18	LBLOCA in BWR suppression pool containment. Comparison to Contempt results
Victoria 13	General behaviour of containment and ice condenser. Normal operation of IC doors. Both IC block are full of ice.
Victoria 29	General behaviour of containment and ice condenser. Behaviour of formation of the natural circulation loop inside the containment. IC doors are forced open. Other IC block is empty of ice.
Victoria 42	General behaviour of containment and ice condenser. Start of the natural circulation loop inside the containment. IC doors are forced open. Asymmetric ice loading i.e. other IC block is full of ice and other empty of ice.
Victoria 50 (multiple node IC)	General behaviour of containment and ice condenser. Helium distribution. Timing of start of the natural circulation loop inside the containment. Both IC blocks are loaded with half of the normal amount of ice.
Victoria 50 (single node IC)	General behaviour of containment and ice condensers. Helium distribution. Timing of start of the natural circulation loop inside the containment. Both IC blocks are loaded with half of the normal amount of ice.
ISP-42 at Panda facility, Phase F	General behaviour of BWR suppression pool containment during the steam and helium injection.
3.2 Containment separate effect tests	
Battelle tests	Thermal hydraulics and the aerosol behaviour in the ice condenser.
POOLEX STB-20 and STB-21	Water pool (suppression pool) heating and stratification during the steam injection.
Recombiner tests	Function of passive autocatalytic recombiner model.
ISP-35 at NUPEC facility	Effect of containment internal spray system on pressure, gas temperatures, and hydrogen (helium) behaviour.
ISP-47 at MISTRA facility	Steam condensation on structures during the injection of pure steam, and a mixture of steam and helium. Pressure, gas temperatures, helium behaviour.
Spray tests at Mistra facility	Effect of inner spray on vessel pressure and temperature in conditions where initial temperature was stratified.
PACOS x1.2 internal spray test in Battelle model containment (test by GRS)	Effect of containment internal spray system on pressure and gas temperature behaviour.
3.3 Containment benchmarks	

Ice condenser doors	Behaviour of ice condenser doors. Comparison to COCOSYS results.
Blowdown	Containment thermodynamics during the water and steam blowdown. Effect of the mist droplets. Comparison to SUPLES results.
Bubble condenser	Suppression of pressure in the bubble condenser containment by steam condensation in water pools, and by passive internal spray from the pools. Comparison to CONTAIN and MELCOR results.
LLOCA and MSLB in Loviisa ice condenser containment	Behaviour of Loviisa containment during LLOCA and MSLB sequences. Comparison to COCOSYS results.
MSLB in Olkiluoto NPP	Behaviour of Olkiluoto 1&2 primary coolant and containment systems during MSLB sequences. Comparison of containment behaviour to COPTA results. Comparison of behaviour of primary coolant system to COBLIN results

4. NUCLEAR REACTOR **4.1 Comparison with Loviisa Plant Core Measurement Data (3-D Model)**	
Assemblywise power at BOC and EOC	Bundle inlet and outlet temperatures
4.2 OECD/NEA 3-D LWR Core Transient Benchmarks (3-D Model)	No measurement data, code results were required for the following items
PWR core	• Control assembly ejections at full power • Control assembly ejections at hot zero power • Uncontrolled withdrawal of control rods at zero power
PWR in steady state	• Critical boron concentration • Radial power distributions at various axial levels • Maximum power peaking factor • Position of maximum power peaking factor • Axial power distribution
PWR in transient	• Core power versus time • Core averaged fuel temperature versus time • Maximum fuel temperature versus time • Coolant outlet temperature versus time • Radial distribution of power at time of power maximum (at various axial levels) • Radial distribution of power at final time 5 s (at various axial levels)
BWR core	• Cold water injection • Fast and slow core pressurisation
BWR in steady state	• k-eff • Radial power distribution at middle core • Coolant outlet density distribution • Maximum power peaking factor • Position of maximum power peaking factor • Axial power distribution

BWR in transient	• Core power versus time • Core averaged fuel temperature versus time • Maximum fuel temperature versus time • Coolant outlet temperature versus time • Radial distribution of power at time of power maximum (at middle core) • Radial distribution of power at final time 20 s (at middle core) • Coolant outlet density distribution at time of power maximum • Coolant outlet density distribution at final time 20 s
4.3 First AER 3-D Hexagonal Kinetic Benchmark (3-D Model)	No measurement data, code results were required for the following items
VVER-type Core control rod ejection	• Core power versus time • Core power distribution at various axial levels (at steady state and at five time points during the transient)
4.4 Second AER 3-D Hexagonal Kinetic Benchmark (3-D Model)	No measurement data, code results were required for the following items
VVER-type Core control rod ejection with adiabatic feedback	• k-eff of initial state • Core power versus time • Total time integrated power of reactor • Maximum fuel temperature • Reactivity • Core power distributions at various axial levels (at steady state and at five time points during the transient) • Fast flux distributions at various axial levels (at steady state and at four time points during the transient) • Thermal flux distributions at various axial levels (at steady state and at four time points during the transient)
4.5 Third AER 3-D Hexagonal Kinetic Benchmark (3-D Model)	No measurement data, code results were required for the following items
VVER-type Core control rod ejection, whole core dynamics and thermal hydraulics	• Spatial power distributions at 5 specified times • Axial distributions of average coolant density • Time functions: (14 items, including power, temperatuers, mass flows, enthalpies) • Hot channel axial distributions • Hot channel time functions (11 items, including DNB, maximum temperatures, mass flux, void fraction, steam quality, oxide layer thickness)
4.6 Fifth AER 3-D Hexagonal Kinetic Benchmark (3-D Model)	No measurement data, code results were required for the following items

VVER-type Core
Steam line break transient
	• Results of tuning:
	  ○ Subcriticality of the initial state
	  ○ Time of reaching recriticality
	  ○ Time of maximum fission power + value
	• Spatial nuclear power distributions (3 specified situations)
	• Core power versus time
	• Time funtions of global plant parameters: 14 items (pressures, temperaturs, flow rates etc.)
	• Time functions for broken loop: 12 items (pressures, levels, powers, mass flows)
	• Time functions for intact loops: 12 items (pressures, levels, powers, mass flows)

4.7 OECD PWR MSLB Benchmark	Reference solution with TRAC-PF1/NEM code results required for following items
TMI-1 NPP Main steam line break transient Three stages: a) Point kinetics and circuit model b) 3-D core with boundary conditions c) 3-D core and circuit model APROS-model: Point kinetics /3-D core model + 6-equation circuit model.	**Steady state** • k-eff • Radial power distribution • Axial power distribution • Scram and stuck rod worth • Primary system pressure, temperatures and flows **Transient** • Sequence of events • Reactor power • Steamline pressure (broken/intact loop) • RCS pressures • Hot and cold leg temperatures • Break flow rates • Steam generator mass • Reactivity edits • Snapshots at 3 specified situations

Validation cases used regularly at each APROS version change

Separate effects tests

EDWARDS PIPE

BATTELLE TOP BLOWDOWN EXPERIMENT

BECKER'S EXPERIMENTS

ERSEC REFLOODING TEST

MARVIKEN CRITICAL FLOW TESTS

LIQUID BLOWDOWN TO CONTAINMENT

BLOWDOWN EXPERIMENT MX-II AT MARVIKEN CONTAINMENT FACILITY

SPRAY EXPERIMENT (ISP-35) AT NUPEC CONTAINMENT FACILITY

STEAM CONDENSATION EXPERIMENT (ISP-47) AT MISTRA CONTAINMENT FACILITY

WATER LEVEL RISE TRANSIENT IN PACTEL PRESSURIZER

Integral tests with power plant models

LOVIISA NUCLEAR POWER PLANT (2 versions)
- Stopping of main recirculation pump
- Reactor scram
- Small break LOCA
- Large break LOCA
- Stopping of three main recirculation pumps with modified end: 3D-neutronics test

VVER-440 NUCLEAR POWER PLANT
- Stopping of main recirculation pump
- Reactor scram

OLKILUOTO 1 NUCLEAR POWER PLANT
- Steam line break
- Reactor scram

FORSMARK 3 NUCLEAR POWER PLANT (HAMBO)
- Electric load rejection
- Turbine trip, turbine bypass fails
- Loss of condenser vacuum
- Trip of all recirculation pumps
- Feed water line break, ATWS
- Steam line break

LOVIISA ICE CONDENSER CONTAINMENT
- Results of Large Break LOCA

CCGT POWER PLANT MODEL WITH DISTRICT HEATING CIRCUIT
- Power set point change