VTT Nuclear Energy

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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 — MAIN PARAMETERS AND PHENOMENA

1.1 Homogeneous model

THERMAL HYDRAULIC SEPARATE EFFECT TESTS	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 (integral tests)

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 behavior (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 uncovery
  - 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)

VVER-type Core

- Control rod ejection, whole core dynamics and thermal hydraulics

No measurement data, code results were required for the following items

- 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)

VVER-type Core plus Circuit Model

- Steam line break transient

No measurement data, code results were required for the following items

- 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

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.

Reference solution with TRAC-PF1/NEM code results required for following items:

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

Power plant models

LOVIISA NUCLEAR POWER PLANT

- Stopping of main recirculation pump
- Reactor scram
- Small break LOCA

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