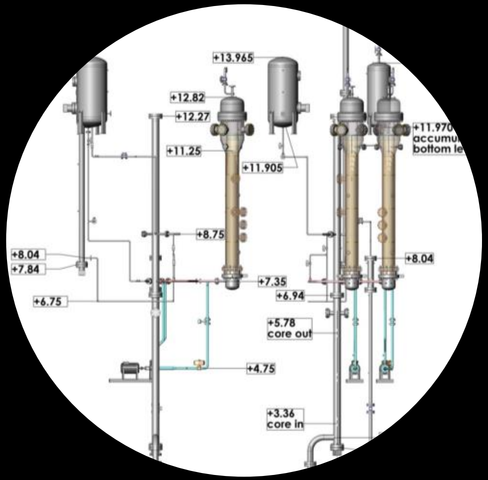
JOINT INTERIM SEMINAR OF SAFIR2022 AND KYT2022

PATE PROJECT



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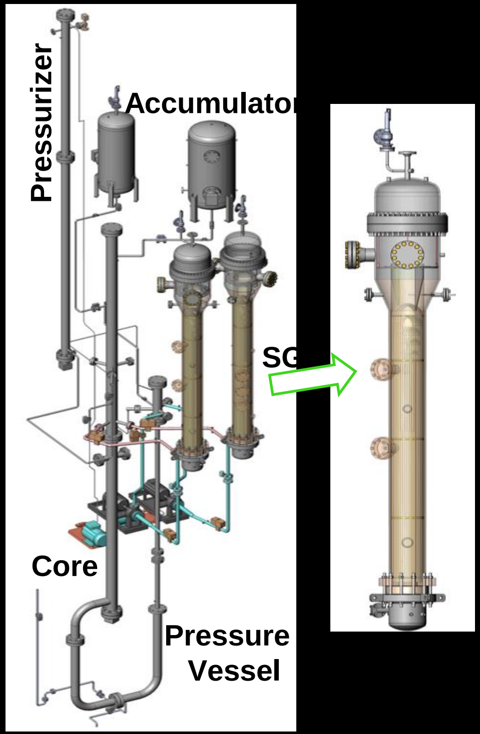
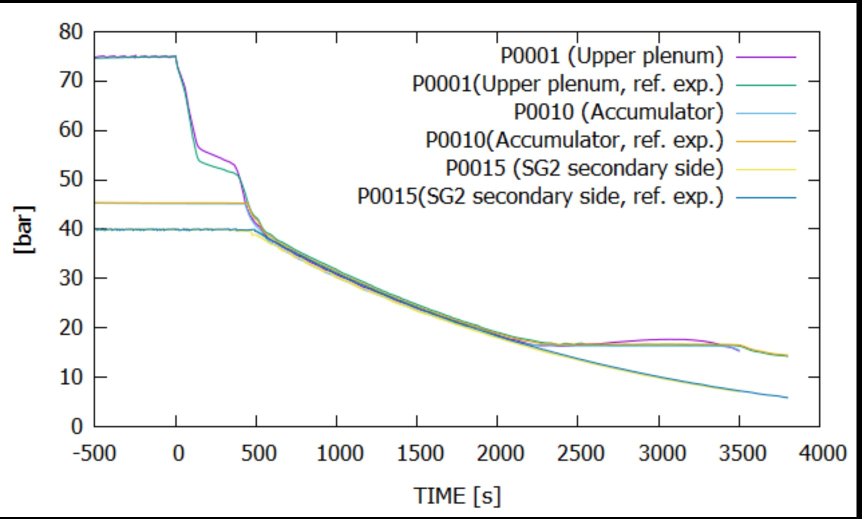
PWR PACTEL FACILITY

- Is designed for safety studies related to thermal hydraulics of the PWRs with EPR type vertical steam generators
- Consists of a reactor pressure vessel model, two loops with vertical steam generators, a pressurizer, and emergency core cooling systems including nitrogen-driven accumulators

CHARACTERISTIC	PWR PACTEL
Reference power plant	PWR / EPR
Volumetric scale: pressure vessel, steam generators, pressurizer	1:405, 1:400, 1:562
Height scale: pressure vessel, steam generators, pressurizer	1:1, 1:4, 1:1.6
Maximum primary / secondary side pressure	8.0 MPa / 5.0 MPa
Maximum primary / secondary side temperature	300 °C / 260 °C
Maximum core power	1 MW
Maximum accumulator pressure	5.5 MPa
Number of fuel rod simulators	144
Maximum rod cladding temperature	750 °C
Number of primary loops	2
Number of U-tubes in steam generator	51
Number of instrumented U-tubes in steam generator 1 / 2	8 / 14 (51)
Average steam generator U-tube length	6.5 m
Steam generator U-tube diameter / wall thickness	19.05 mm / 1.24 mm
Main material of components	Stainless steel
Insulation material	Mineral wool



NITROGEN EXPERIMENTS



- Earlier PWR PACTEL nitrogen experiments with a hot leg break showed that the accumulator nitrogen could stop the primary side depressurization and cause a core heat up at reactor pressure above or very close to a typical low-pressure safety injection shut-off head
- New tests to map the full range of the pressures at which the decoupling of the primary and secondary side pressures takes place, as a function of
  - Break size
  - Number of the accumulators injecting (nitrogen mass)
  - Number of the steam generators participating in the secondary side depressurization (volume available for nitrogen)
- Pre-test simulations with APROS were done to plan the experiments
- Effect of the number of steam generators on the system was studied in 2020 with one loop
  - Transient with either one or two loops and steam generators had a similar effect on the overall appearance of the decoupling of the pressures, the pressure value level when the decoupling appeared, and the length of that period
- In the OECD/NEA PKL Phase 4 project three PWR PACTEL nitrogen experiments were performed in 2018 and 2019 to complement the PKL experiment(s) on IB/SB-LOCA
- LUT also participates in the OECD/NEA ETHARINUS project

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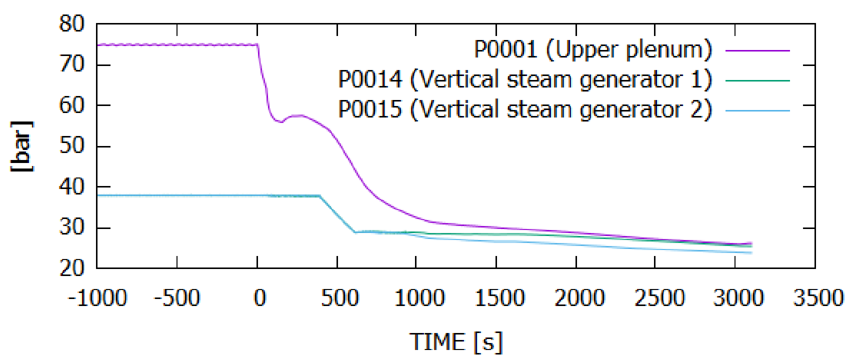
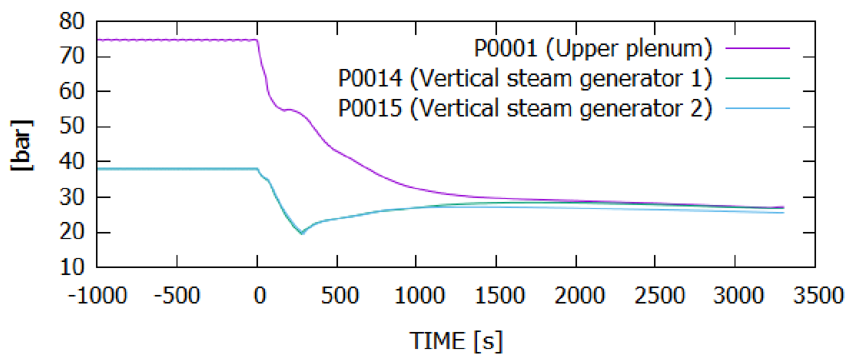
VALVE OPENING EXPERIMENTS

- Inadvertent opening of the pressurizer safety valve with a simultaneous full opening of the valves in the main steam relief trains influences the departure from nucleate boiling ratio in the beginning of a loss-of-coolant accident
  - Goal of the experiments was to clarify the effect of the fast secondary side cooldown
  - Pre-test simulations with APROS were done to get initial insight on the conditions during the experiment procedures
- Two experiments were performed
  - In the first experiment opening of one pressurizer safety release valve was assumed
  - Second experiment was similar but the main steam relief valves were assumed to open at the same time as one pressurizer safety release valve
  - General system behavior was similar in both experiments
  - Core temperatures never increased above the initial values and at the end, they stabilized and decreased slowly





SIMULATIONS

APROS

- Was used in the post-test simulation of one PWR PACTEL nitrogen experiment in the OECD/NEA PKL Phase 4 project
- Overall trends and events during the transient were predicted comparable
- Accumulator injection period was slightly prolonged in the simulation
  - Efficiency of the liquid heat transfer needed modification

TRACE

- Model of the PKL test facility was tested by calculating a IBLOCA test of the PKL facility
- Initial conditions and the general progression of the transient were mostly well predicted
- Leaked water mass was overestimated due the simple nodalization
  - LPSI started much earlier in the calculation since the primary side pressure dropped to the lower level
- Core heat-up was not predicted
- Adjustments of the k-factors at the break
  - Improved the prediction of the leaked mass, behavior of the primary side pressure, the accumulator injection flow rate, and the core cladding temperature behavior during the accumulator water injection

EXPLOITATION OF THE RESULTS AND THE SAFETY SIGNIFICANCE

- Project
  - Ensure the operation of safety related systems or the efficiency of the procedures in accident and transient situations of nuclear power plants
  - Enhances the Finnish nuclear safety assessment capability for solving future safety issues as they appear
  - Maintains and extends the research expertise needed for the experimental work and produces data for the validation of computational tools
  - Educates experts on EPR specific issues
  - Has significant international connection through the OECD/NEA PKL Phase 4 and PKL ETHARINUS projects training new researchers and familiarizing them with international networking, as well as developing their skills in communicating their research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