CHALMERS
UNIVERSITY OF TECHNOLOGY

The Role of Swedish Universities in Supporting SSM Activities in the Field of Deterministic Safety Analysis

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Strål säkerhets myndigheten
Swedish Radiation Safety Authority

Contents

OVERVIEW OF NUCLEAR POWER IN SWEDEN

Development of nuclear industry

1947

- Establishment of the atomic energy research organization, AB Atomenergi.

1954

- R1, experimental research reactor was commissioned at Royal Institute of Technology, Stockholm.
- Heavy water reactor which could have been used to produce weapons grade plutonium.

1960

- Two test reactors, R2 and R2-0, were commissioned at Nyköping.
- Heavy water reactors with a capability for co-generation.

1964

- The first commercial nuclear power plant, Ågesta a.k.a R3, was in operation in Stockholm.
- Heavy water reactor for co-generation: 65 MWt, 10 MWe.
- Another reactor was built, Marviken a.k.a R4, for dual purpose: electricity production (130 MWe) and plutonium production.
- Marviken was abandoned in 1970.

1966
- Oskarshamn 1 was ordered from ASEA and started up in 1972.
- Western-designed boiling water reactor.

Late 60's
- Increased interests in light water reactors.

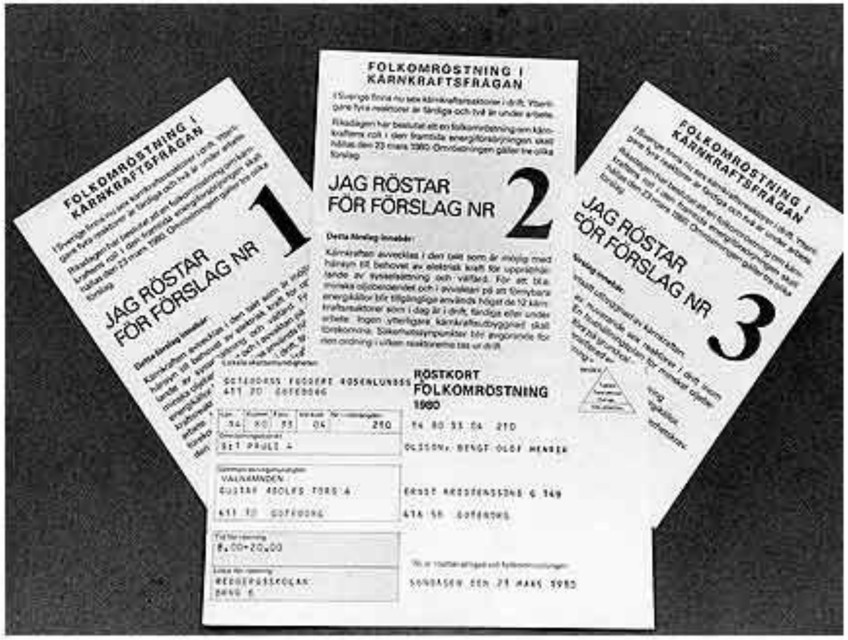
1968
- Ringhals 1 (750 MWe BWR) and Ringhals 2 (800 MWe PWR) were ordered.

1969
- Oskarshamn 2 and Barsebäck 1 were ordered.

70's – 80's
- 6 reactors were in commercial service in the 1970s.
- 6 other reactors were operated in the 1980s.

Public referendum about the future of nuclear power March 23, 1980

- Parliament decided to embargo further expansion of nuclear power and aim for closing the 12 plants by 2010 *if new energy sources were available **realistically*** to replace them.
- Decommissioning of Barsebäck-1 (November 1999) and Barsebäck-2 (June 2005)



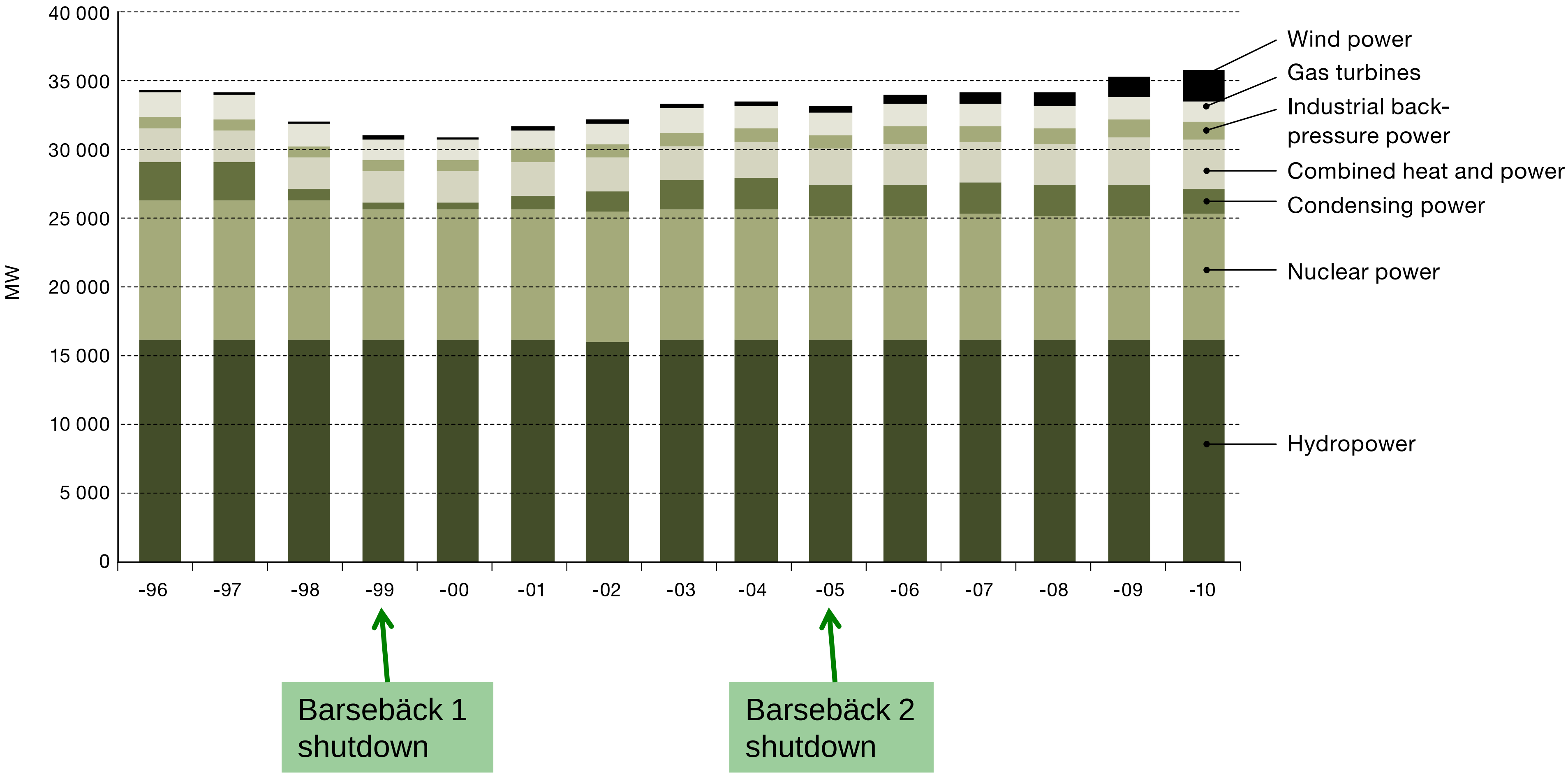
February 2009

- The Swedish coalition government announced an agreement to abolish the act of banning construction of new nuclear reactors.

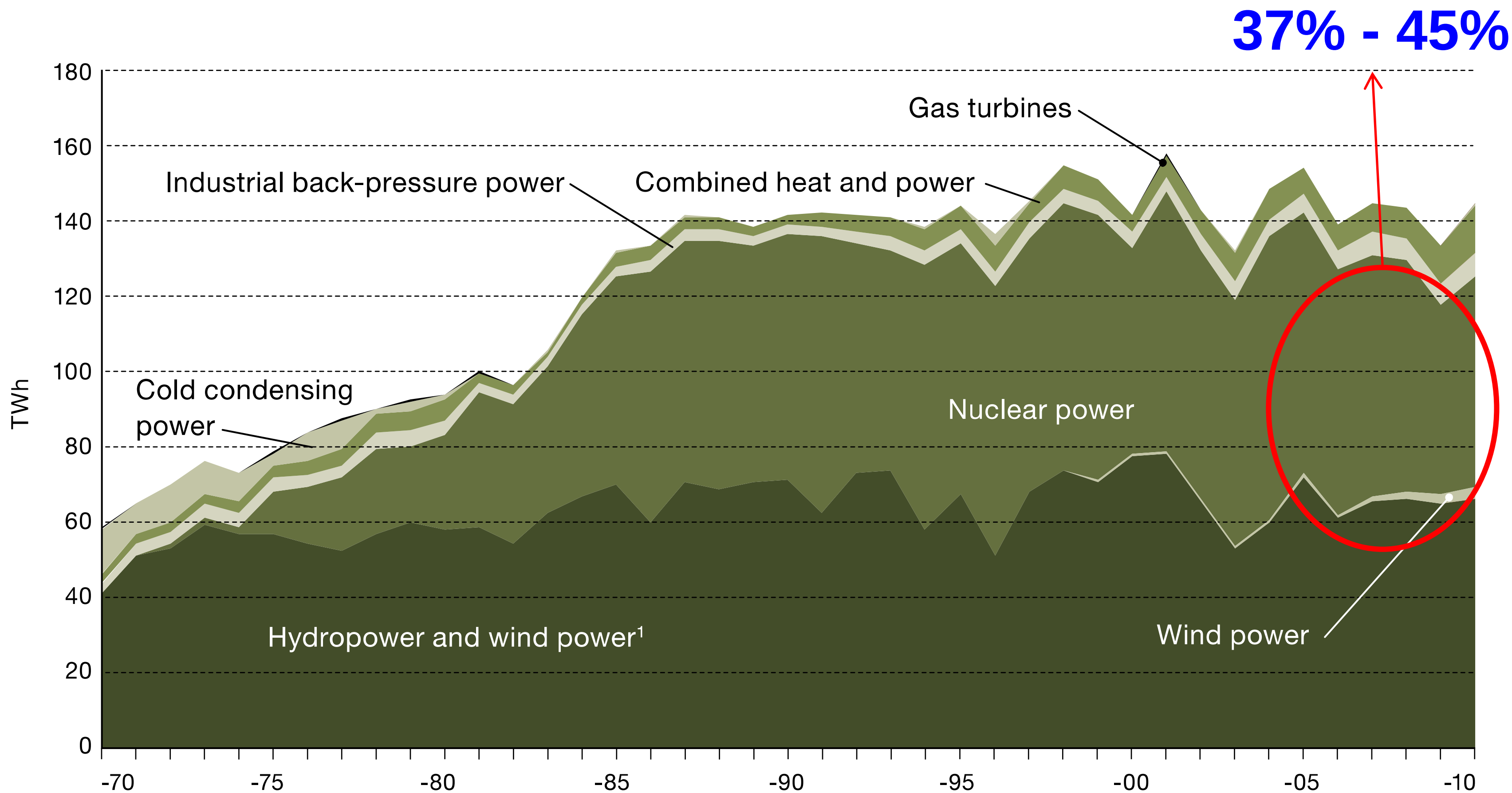
June 2010

- The parliament approved the decision allowing the replacement of the existing reactors with new nuclear reactors, starting effectively from 1 January 2011.

Installed electricity production capacity in Sweden, 1996-2010, in MW



Electricity production in Sweden, 1970-2010, in TWh



Status on Power Uprates

Plant	Type	Initial Power MWt	Uprated Power MWt	Power Uprate %	Year of Uprating	Comment
Forsmark-1	BWR	2711	2928	8%	1986	
			3253	20%	2011	Plant mod
Forsmark-2	BWR	2711	2928	8%	1986	
			3253	20%	2009	Plant mod
Forsmark-3	BWR	3020	3300	9%	1989	
			3775	25%	201x	Delayed
Oskarshamn-1	BWR	1375	1375	0%	1982	
Oskarshamn-2	BWR	1700	1800	6%	1982	
			2300	35%	2011	Delayed
Oskarshamn-3	BWR	3020	3300	9%	1989	
			3900	29%	2009	Test operation
Ringhals-1	BWR	2270	2500	10%	1989	
			2540	12%	2006	
Ringhals-2	PWR	2440	2660	9%	1989	
Ringhals-3	PWR	2783	3000	8%	2006	
			3144	13%	2009	
Ringhals -4	PWR	2783	3300	19%	2011	Plant mod

TSO-DSA ACTIVITIES

Reactor safety research

Support regulation

Contribute to national competence in the area of nuclear safety.

Regulatory challenges

Improved understanding of occurrences in power plants

Power uprates and core optimizations

Review of safety analysis reports, SSM shall ensure that the requirements for safe operation are fulfilled

Fear

National competence will degrade if not directed efforts, particularly towards universities, are made

TSO - DSA

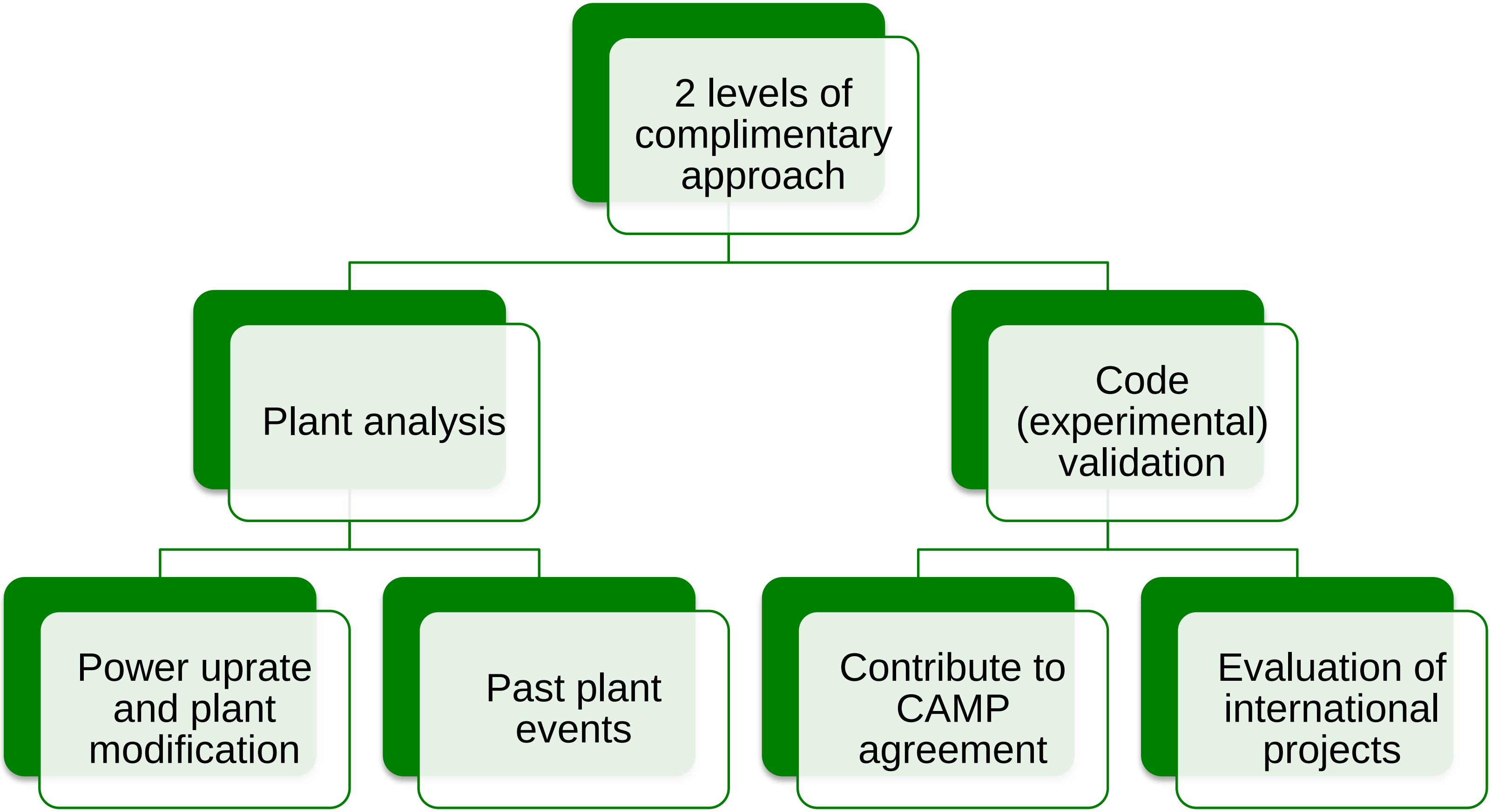
to promote **national competence** within the deterministic safety analysis,

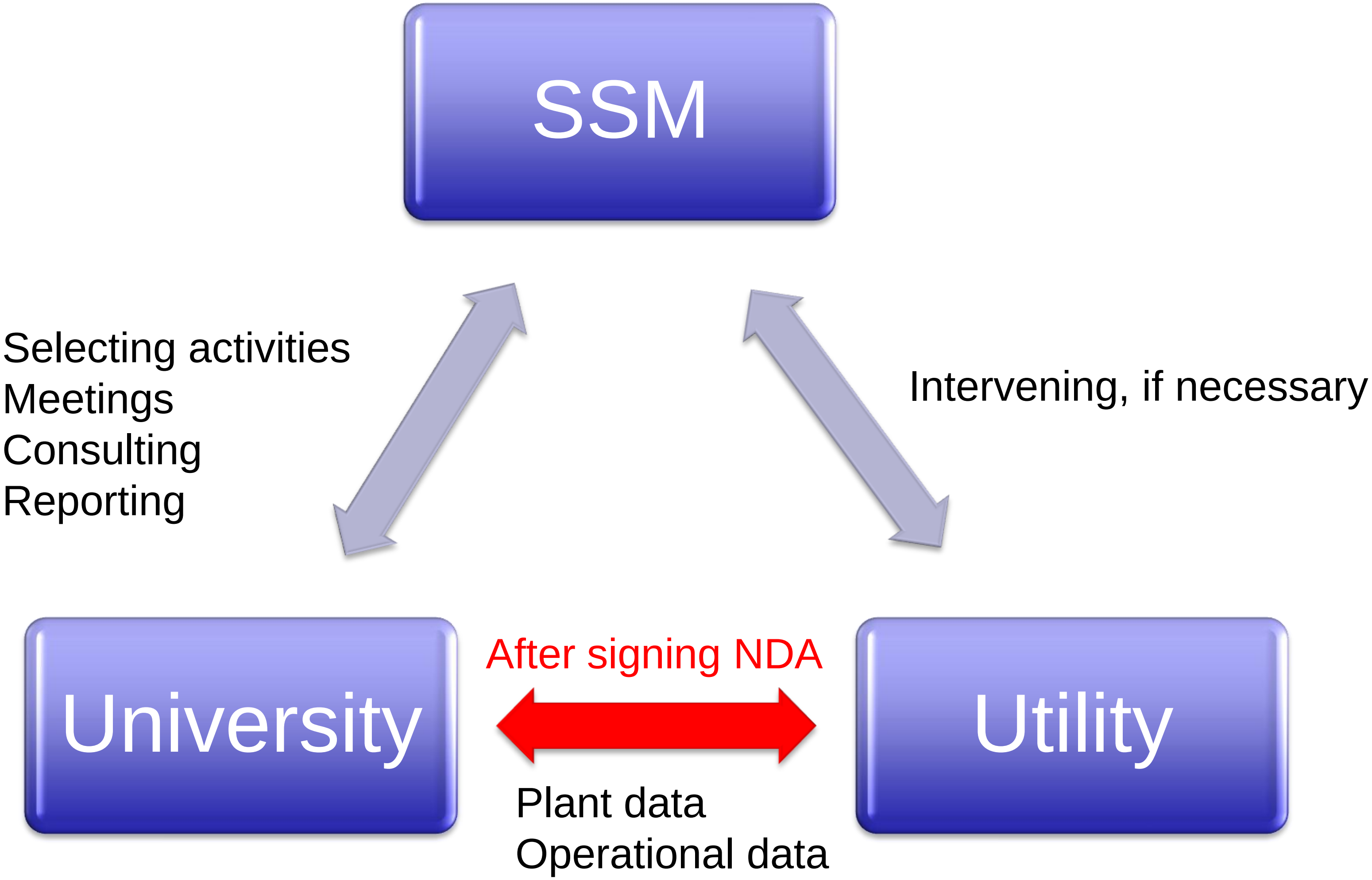
to **establish groups** who can support SSM to perform review and inquiries,

to participate in relevant **international projects** and working groups.

Chalmers University of Technology

Royal Institute of Technology (KTH)


2 levels of complimentary approach
Plant analysis
Code (experimental) validation
Power uprate and plant modification
Past plant events
Contribute to CAMP agreement
Evaluation of international projects


SSM
Selecting activities
Meetings
Consulting
Reporting
Intervening, if necessary
University
After signing NDA
Plant data
Operational data
Utility

Plant analysis

- Chalmers :
  - Ringhals-3 and Ringhals-4 (Westinghouse 3-loop PWR)
  - Forsmark-1 and Forsmark-2 (BWR)
- KTH :
  - Oskarshamn-2 (BWR with external pumps)
  - Oskarshamn-3 and Forsmark-3 (BWR with internal pumps of ASEA design)
- Perform independent analyses of some limiting transients and accident sequences as basis for safety judgements.
- Methodology to perform independent analyses **with coupled neutron kinetics and thermalhydraulics** in three dimensions, using PARCS/RELAP5 and PARCS/TRACE.
- Highlight the effects of power increases through calculations of transients that have occurred at different power levels and core loadings.
- Through best estimate analyses and uncertainties improve SSM knowledge base for judgement of risks associated with various types of transients.
- Through sensitivity analyses determine safety importance of different phenomena.

Examples of past plant events

- Forsmark-1
  - 2006 loss of external power and loss of power supply from 2 of 4 diesel generators
- Oskarshamn-3
  - 2002 turbine trip event
  - 2009 startup tests at uprated power level (stability, CR test, load rejection)
- Oskarshamn-2
  - 1999 FW transient (instability)
  - 2008 FW transient, trip on low water level
- Forsmark-3
  - 1994 unauthorized CR insertion
- Ringhals-3
  - 2005 load rejection transient
  - 2005 LONF transient

Parameters influenced by the power uprate

- increase of the **power density** (possibly leading to a reduction of the **DNB margins)**;
- faster **transients** and **incidents**;
- larger **decay heat** after reactor shutdown;
- larger energy/mass release in case of **pipe break** on the steam lines or on the primary loops;
- reduction of the **shutdown margins**;
- increase of the **steam flow** from the steam generators (leading to larger pressure drops, which can increase the load on some systems and components);
- increase of the load on some **electrical systems** and **components**;
- increase of the **irradiation** levels from the core (possibly leading to more irradiation-induced defects in materials);
- increase of the inventory of radioactive elements of the **nuclear fuel waste**;
- modification of the temperature in some **cooling circuits** (possibly leading to a new distribution of the loads and of the sensitivity to corrosion)

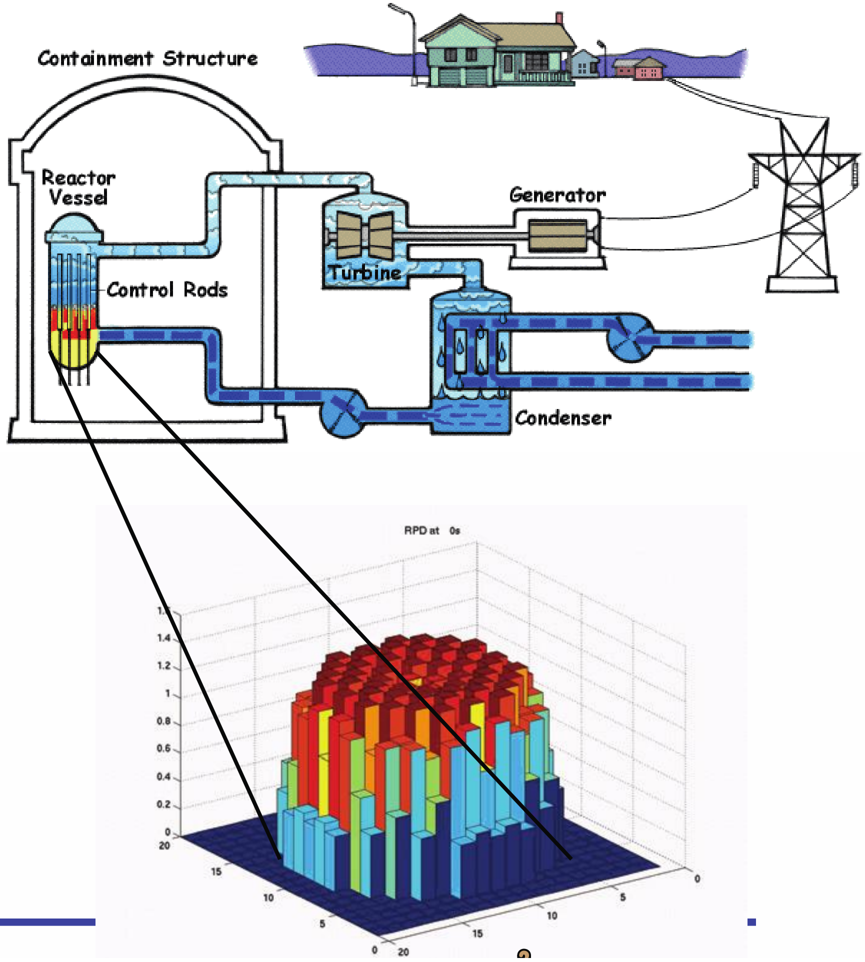
Coupled NK/TH Concept

Primary System Thermal-Hydraulics

- governing equations – 1D two-phase fluid flow
- purpose – simulate response of entire plant
- simulation tools – RELAP5, TRACE, CATHARE, ATHLET

Reactor Core Neutron Kinetics

- governing equations – 3D neutron kinetics
- purpose – simulate detail power distribution inside reactor core
- simulation tools – PARCS, CRONOS, SIMULATE, QUABOX



Coupled NK/TH Theory

Common solution methods

- **Fluid Flow**: 6 PDEs, nearly-hyperbolic

$$\frac{\partial}{\partial t} f + A_f(f)\frac{\partial}{\partial x} f = S_f(f, T, \mathrm{f})$$

Upwind FD
1st order time, 1st order space

- **Heat Conduction**: 1 PDE, parabolic

$$\frac{\partial}{\partial t} T + A_T(T)\frac{\partial^2}{\partial x^2} T = S_T(f, T, \mathrm{f})$$

Center FD
1st order time, 2nd order space

- **Neutron Diffusion**: 2 (typically) PDEs, parabolic

$$\frac{\partial}{\partial t} \mathrm{f} + A_{\mathrm{f}}(f, T)\frac{\partial^2}{\partial x^2} \mathrm{f} = S_{\mathrm{f}}(f, T, \mathrm{f}, c)$$

Nodal
2nd order time, Nth order space

- **Precursor Concentration**: 6 (typically) ODEs

$$\frac{\partial}{\partial t} c = S_c(\mathrm{f}, c)$$

Crank-Nicolson
2nd order time

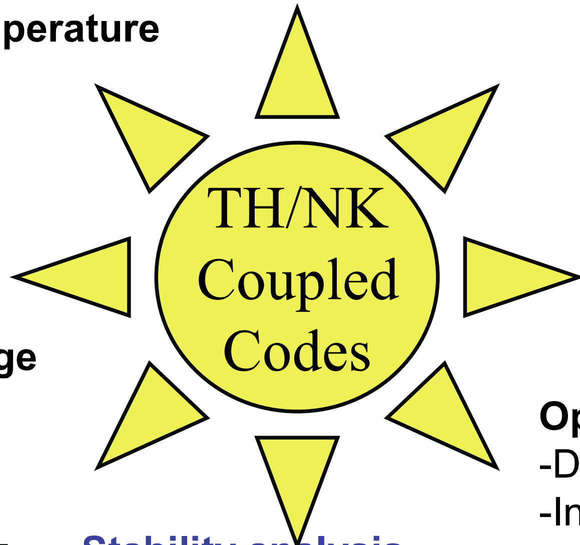
Non-linear, strongly coupled through coefficient matrix **A** and source **S**

Transients for which NK/TH coupled codes could be important

Ability to evaluate local effects

Asymmetric inlet temperature
-Steam line break
-Startup of cold loop
-FW transient
-Pump trip

Asymmetric flow conditions
-Boron dilution events
-Solute tracking



Accurate feedback effects
-Transients without scram
-Turbine trip
-HFP → HZP → cold shutdown

Localized spatial change
-Rod ejection accident
-Rod drop
-Channel blocking

Operational support
-Determining setpoints
-Instrumentation response

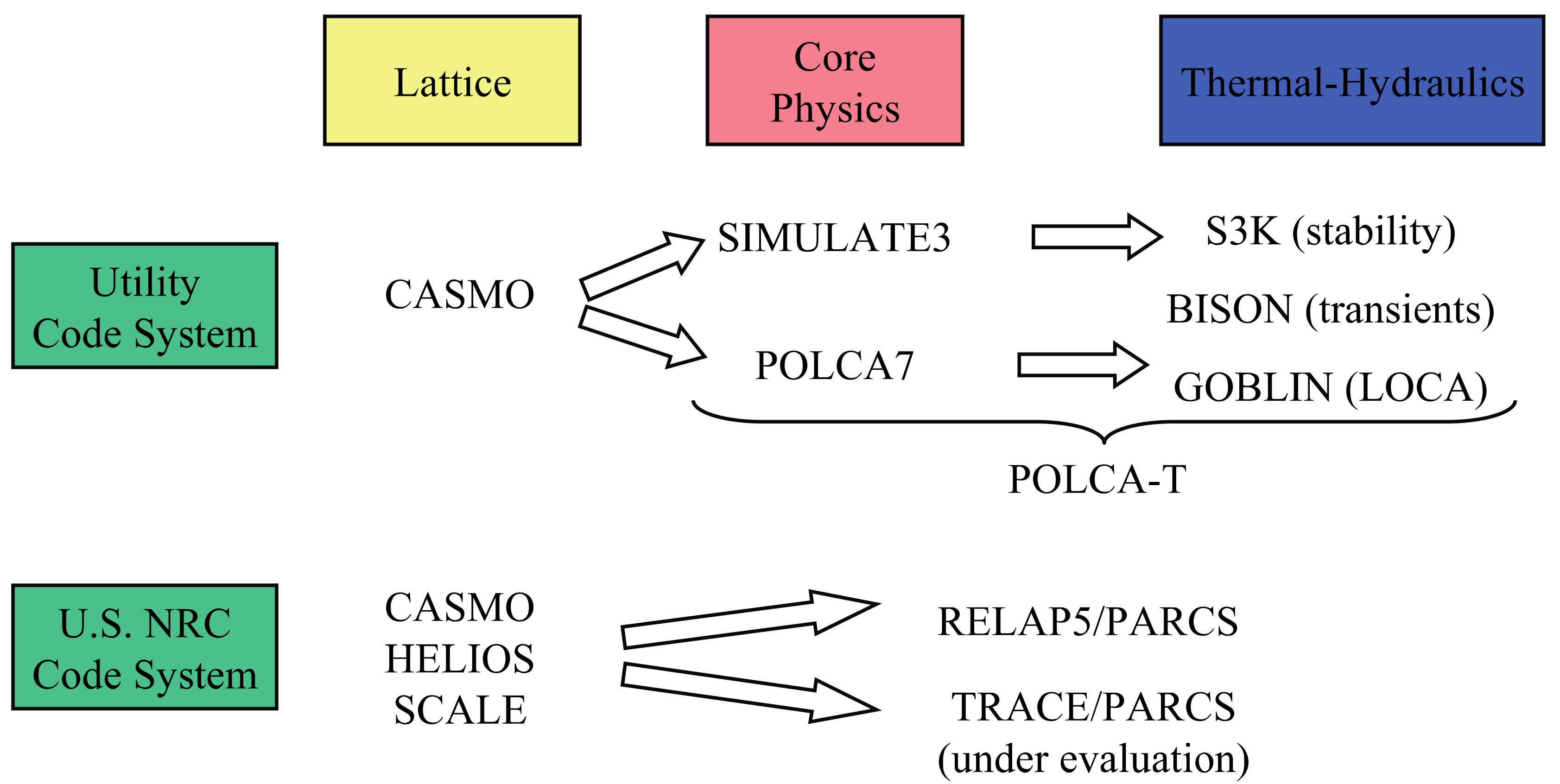
Stability analysis
-Global (in-phase)
-Regional (out-of-phase)
-Local

Asymmetric reactivity change

Calculation accuracy

Consequences of the transients better understood
Potential for relief in restrictive operating limits and increase in safety margins

State-of-the-Art Capability



Code Validation

- assessment against Marviken critical flow tests (NUREG/IA-401)
- Assessment against Marviken level swell observed in jet impingement tests.
- validation of PARCS against TIP (Traversing In-core Probe) measurements in Forsmark-1 and Ringhals-3 (NUREG/IA-414)
- validation against FIX-II experiments (scaled-down facility of Oskarshamn-2): MB-LOCA, LB-LOCA, Guillotine break
- validation against ROSA-II experiment (scaled-down facility of PWR)
- validation against spray cooling and CCFL experiment in GÖTA
- validation against void profile measurement in BFBT
- involvement in the DNB benchmark of PSBT
- PARCS/TRACE code assessment against ISP-50 ATLAS test,
- validation of RELAP5 against SB-LOCA benchmark exercise (SBL-50) in the PWR PACTEL test facility.

MODEL DEVELOPMENT OF RINGHALS-3

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The Ringhals-3 Unit

Type	PWR
Design	Westinghouse 3-loop
Fuel	157 17x17 assemblies
Years of service	31
Generation	146 TWh
Thermal Power / Gross Electrical Output	2778 MWt / 920 MWe
After Extended Power Uprate	3160 MWt / 1128 MWe

Stages of the power uprate process of Ringhals-3

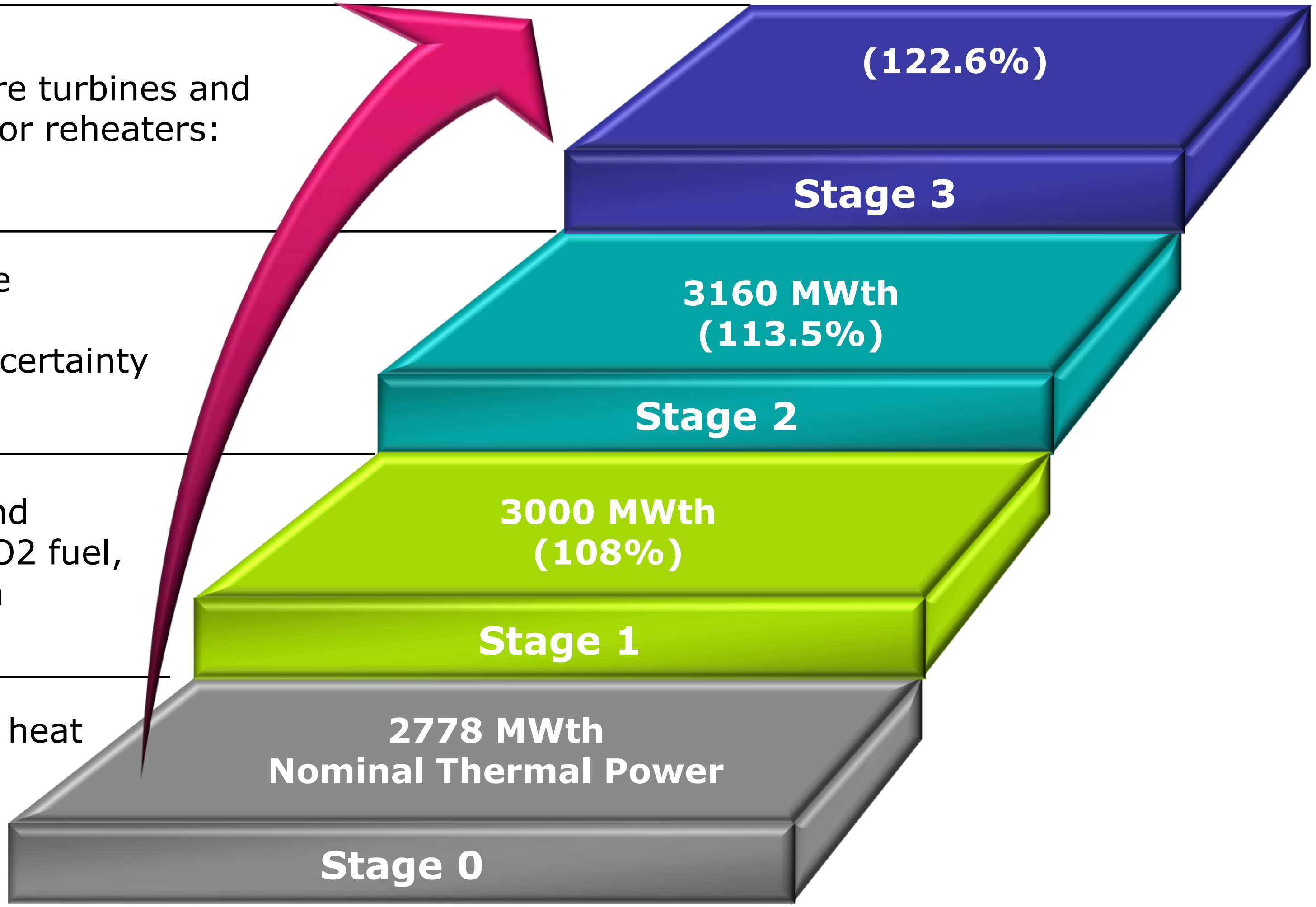
Final goal: +22.6% gain in the gross electric output

New high pressure turbines and moisture separator reheaters: +44 MWe

New low pressure turbines, lower measurement uncertainty (2009)

Higher burnup and higher density UO2 fuel, core optimization (2006)

New SGs: +60% heat exchanger area already built in (1995)



Objectives set by SSM to Chalmers University

1

to perform **independent analyses** of some limiting transients and accident sequences as basis for safety judgments

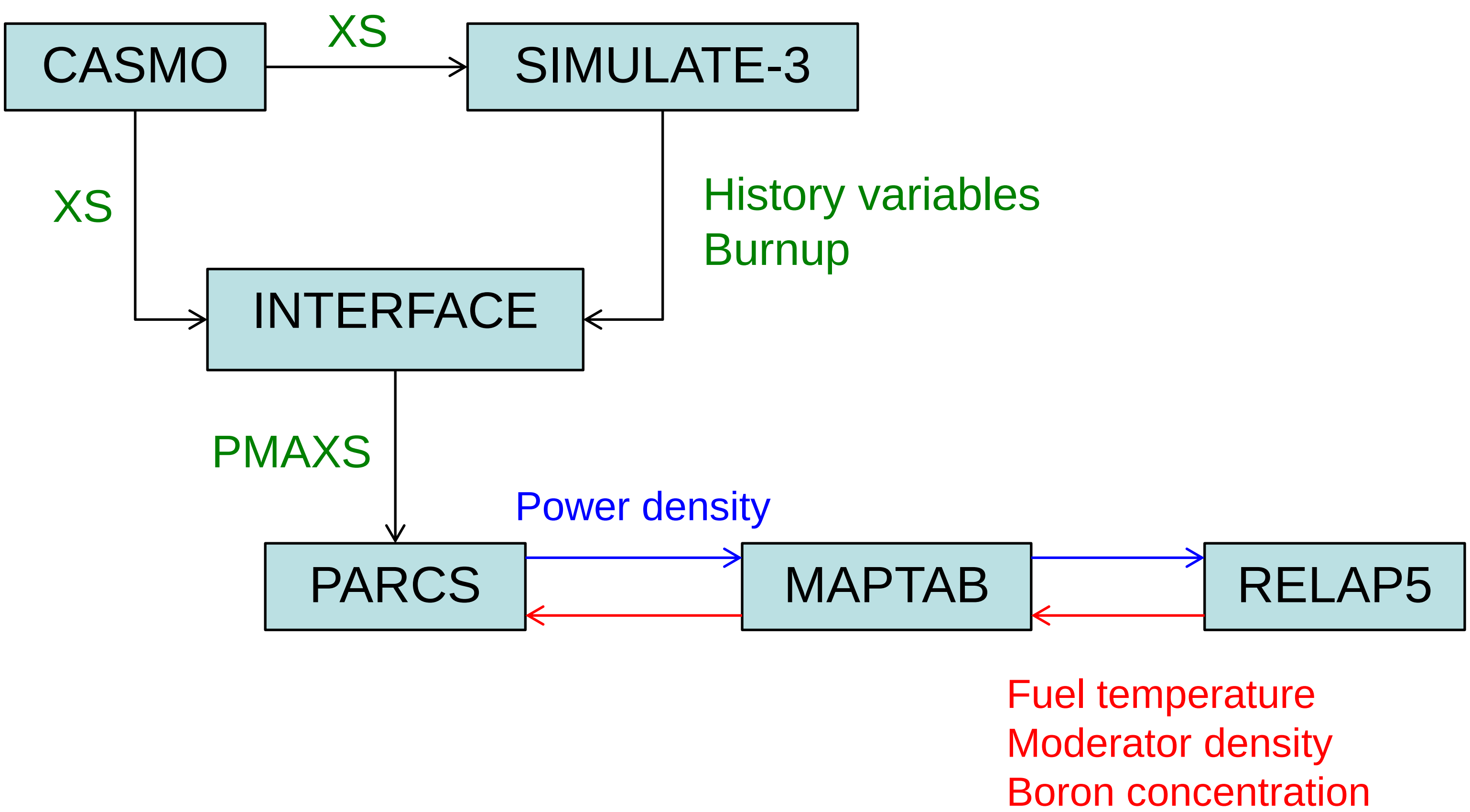
2

to highlight the **effects of power increases** through calculations of transients that **have occurred** at different power levels and core loadings

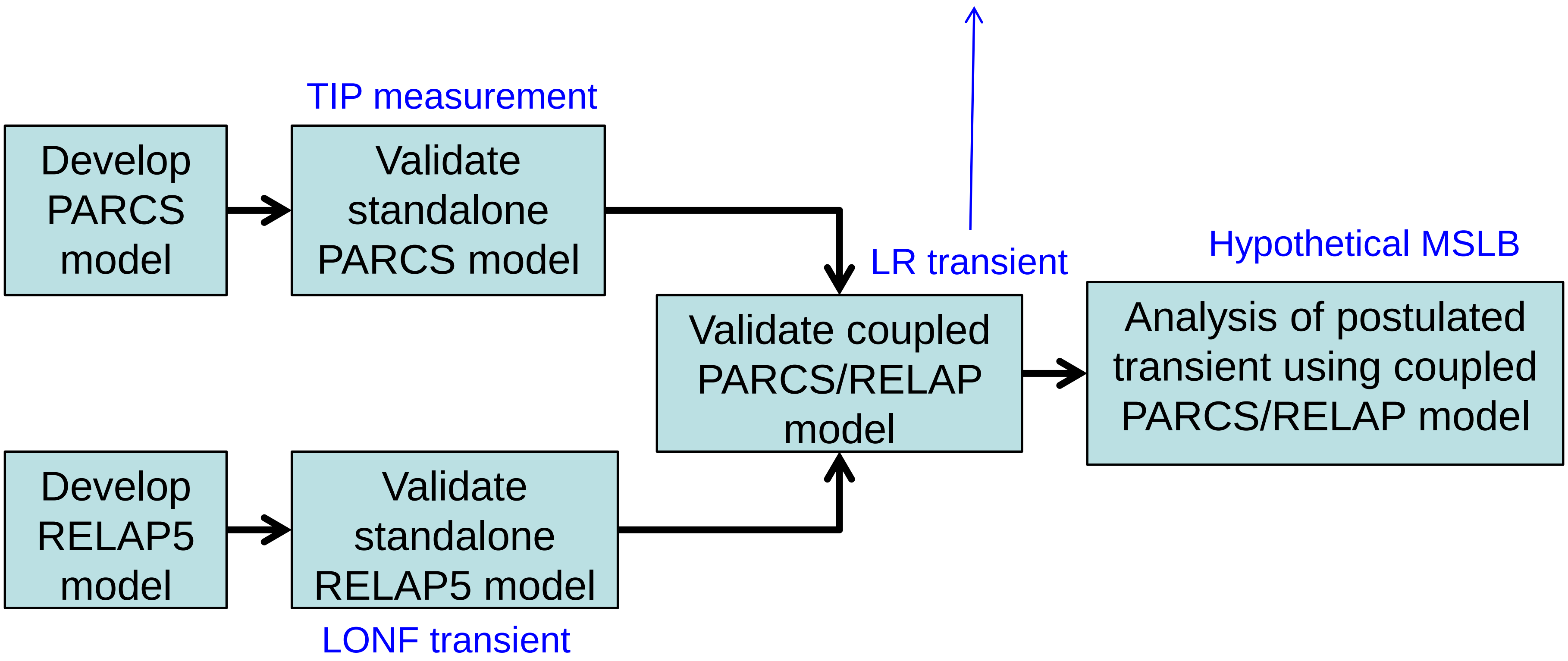
3

to improve **SSM's knowledge base** for **judgment of risks** associated with various types of transients through best estimate analyses and uncertainties

Use of the US NRC PARCS/RELAP5 coupled codes as analytical tools


CASMO
XS
SIMULATE-3
XS
History variables
Burnup
INTERFACE
PMAXS
Power density
PARCS
MAPTAB
RELAP5
Fuel temperature
Moderator density
Boron concentration

1. 2005 LR test (before uprate)
2. 2009 LR transient (after stage 1 uprate)
3. 2010 LR test (after stage 2 uprate)



SIMULATE-3 modelling of Ringhals-3 history

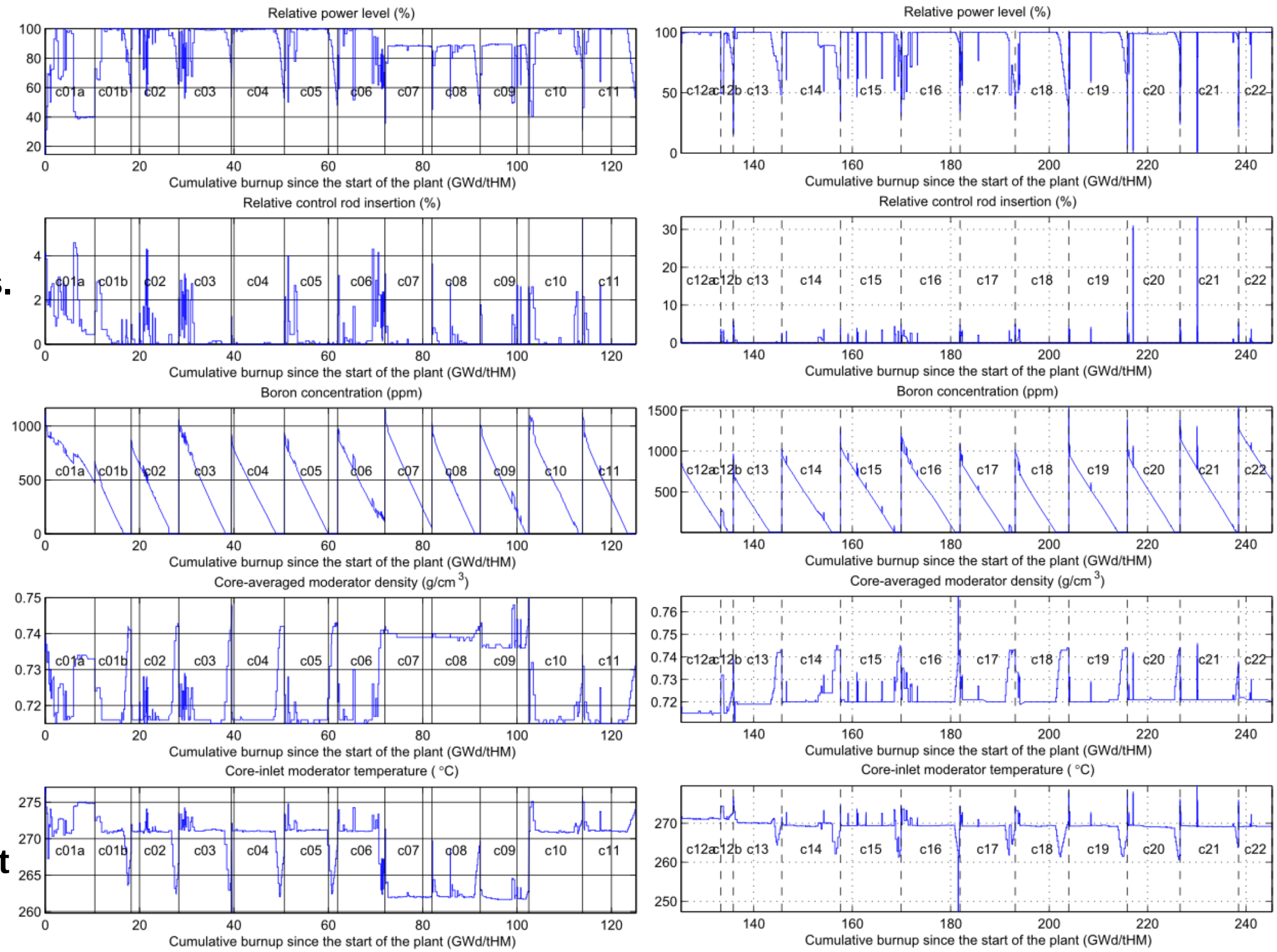
Relative power

Relative control rod ins.

Boron concentration

Moderator density

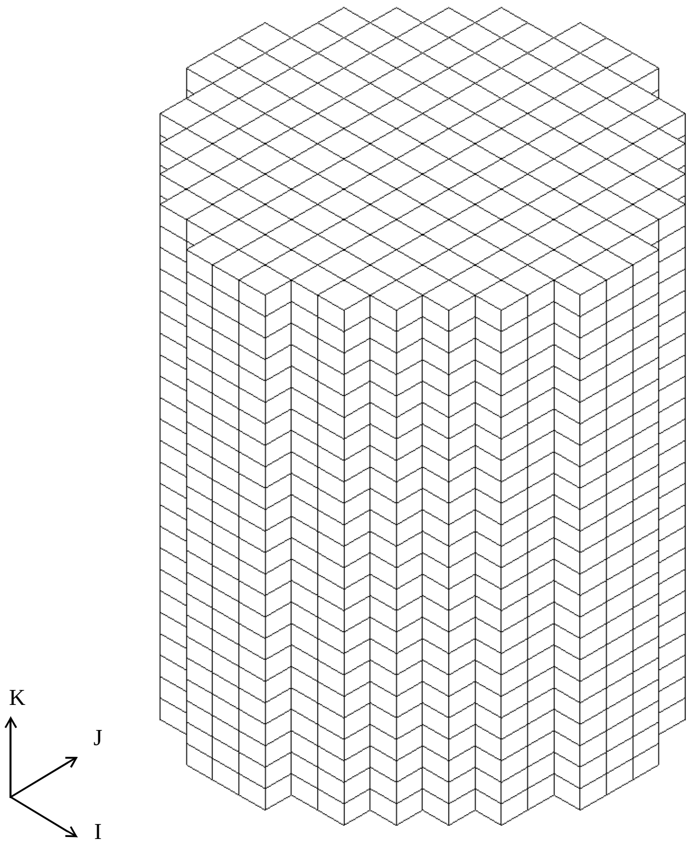
Moderator temp. at inlet



Development of a 2-group fully heterogeneous PARCS model

- Altogether 157 fuel assemblies modelled individually
- Full heterogeneity of the core taken into account (radial and axial zoning of the fuel types), 4 reflector types
- Dependence of the material data on exposure, history variables and instantaneous variables (special interface developed for that purpose)
- Cross-sections based on CASMO-4 calculations (PMAXS files)
- Spatial distribution of the history variables and the burnup retrieved from SIMULATE-3 code.
- One set of kinetic data for the whole core (inverse velocities, fraction of delayed neutrons, decay constants of the precursors).
- Spatial distribution of the instantaneous parameters (e.g. fuel temperature, moderator density and boron concentration) are retrieved from RELAP5.

Nodalization of the active core



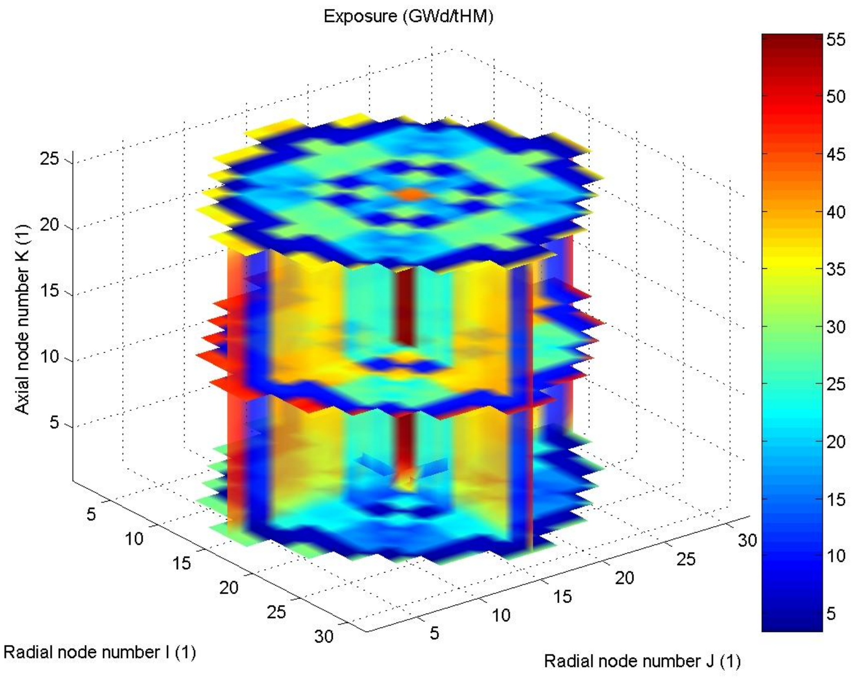
Core loading

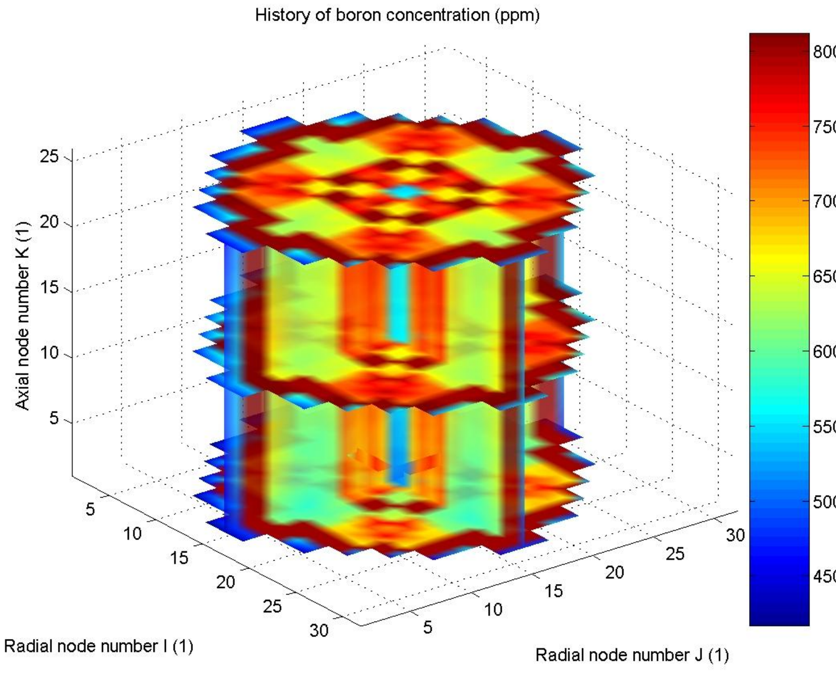
I \ J		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
							11	11	11	11	11						
1					11	12	12	2	2	2	12	12	11				
2				12	12	2	1	6	6	6	1	1	12	12			
3			12	12	1	8	7	7	2	7	7	6	1	12	12		
4		11	12	1	5	5	5	2	1	2	5	5	5	1	12	11	
5		12	1	6	5	1	4	1	2	1	4	1	5	8	2	12	
6	11	12	1	7	5	4	1	7	2	7	1	4	5	7	1	12	11
7	11	2	6	7	2	1	7	1	4	1	7	1	2	7	6	2	11
8	11	2	6	2	1	2	2	4	3	4	2	2	1	2	6	2	11
9	11	2	6	7	2	1	7	1	4	1	7	1	2	7	6	2	11
10	11	12	1	7	5	4	1	7	2	7	1	4	5	7	1	12	11
11		12	2	8	5	1	4	1	2	1	4	1	5	6	1	12	
12		11	12	1	5	5	5	2	1	2	5	5	5	1	12	11	
13			12	12	1	6	7	7	2	7	7	8	1	12	12		
14				12	12	1	1	6	6	6	1	2	12	12			
15					11	12	12	2	2	2	12	12	11				
							11	11	11	11	11						

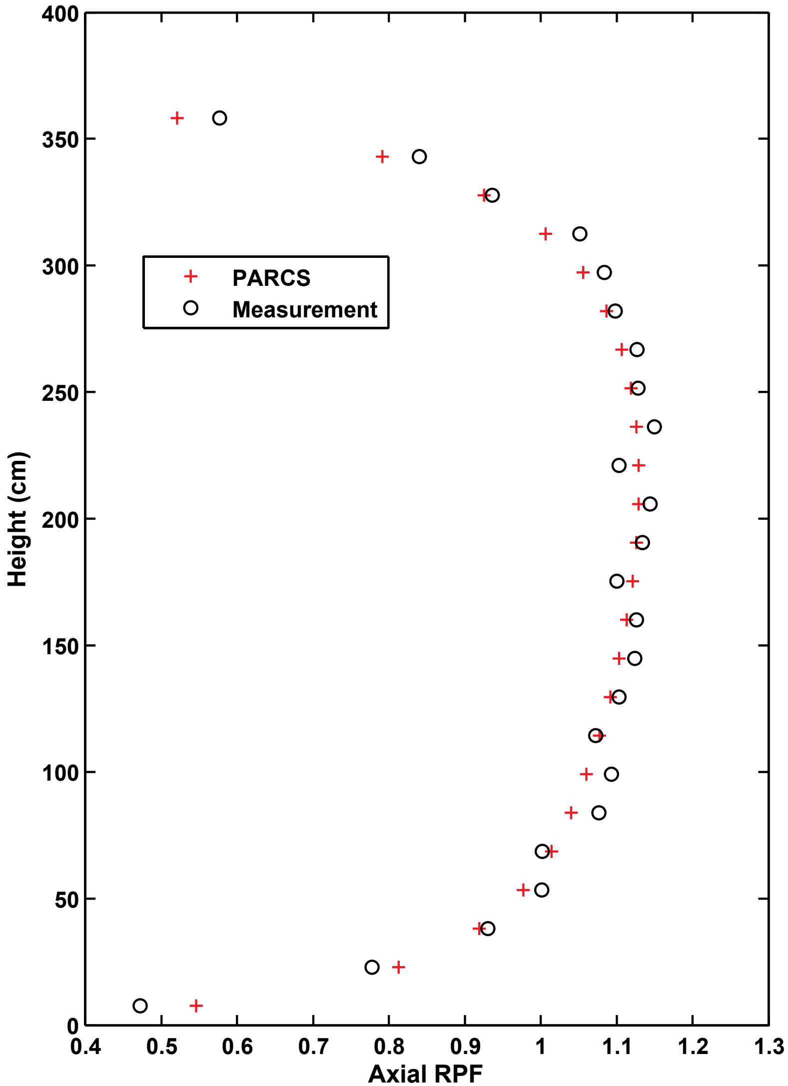
CR banks

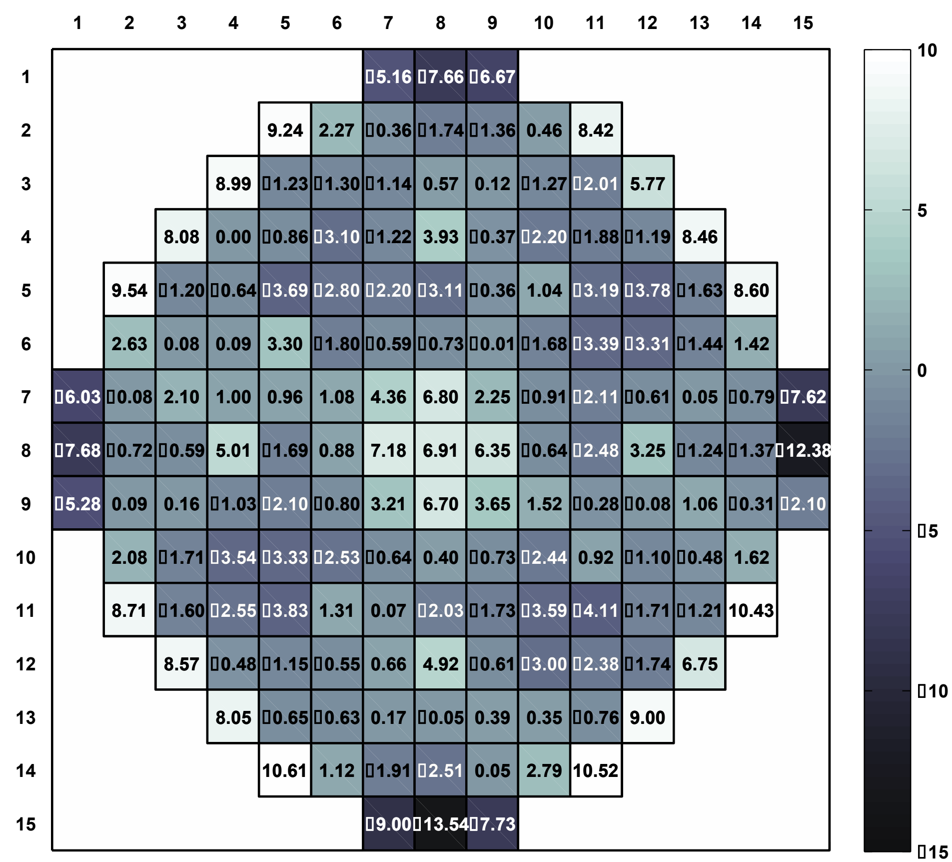
I \ J		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
1																	
2							4		1		4						
3								6		6							
4					2		3				3		2				
5						5						5					
6			4		3		1		2		1		3		4		
7				6				5		5				6			
8			1				2				2				1		
9				6				5		5				6			
10			4		3		1		2		1		3		4		
11						5						5					
12					2		3				3		2				
13								6		6							
14							4		1		4						
15																	

Spatial distribution of the exposure and the history of the boron concentration for the fuel cycle 22






PARCS
Measurement
Height (cm)
Axial RPF
0
50
100
150
200
250
300
350
400
0.4
0.5
0.6
0.7
0.8
0.9
1
1.1
1.2
1.3


1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
1 -5.16 -7.66 -6.67
2 9.24 2.27 -0.36 -1.74 -1.36 0.46 8.42
3 8.99 -1.23 -1.30 -1.14 0.57 0.12 -1.27 -2.01 5.77
4 8.08 0.00 -0.86 -3.10 -1.22 3.93 -0.37 -2.20 -1.88 -1.19 8.46
5 9.54 -1.20 -0.64 -3.69 -2.80 -2.20 -3.11 -0.36 1.04 -3.19 -3.78 -1.63 8.60
6 2.63 0.08 0.09 3.30 -1.80 -0.59 -0.73 -0.01 -1.68 -3.39 -3.31 -1.44 1.42
7 -6.03 -0.08 2.10 1.00 0.96 1.08 4.36 6.80 2.25 -0.91 -2.11 -0.61 0.05 -0.79 -7.62
8 -7.68 -0.72 -0.59 5.01 -1.69 0.88 7.18 6.91 6.35 -0.64 -2.48 3.25 -1.24 -1.37 -12.38
9 -5.28 0.09 0.16 -1.03 -2.10 -0.80 3.21 6.70 3.65 1.52 -0.28 -0.08 1.06 -0.31 -2.10
10 2.08 -1.71 -3.54 -3.33 -2.53 -0.64 0.40 -0.73 -2.44 0.92 -1.10 -0.48 1.62
11 8.71 -1.60 -2.55 -3.83 1.31 0.07 -2.03 -1.73 -3.59 -4.11 -1.71 -1.21 10.43
12 8.57 -0.48 -1.15 -0.55 0.66 4.92 -0.61 -3.00 -2.38 -1.74 6.75
13 8.05 -0.65 -0.63 0.17 -0.05 0.39 0.35 -0.76 9.00
14 10.61 1.12 -1.91 -2.51 0.05 2.79 10.52
15 -9.00 -13.54 -7.73
10
5
0
-5
-10
-15

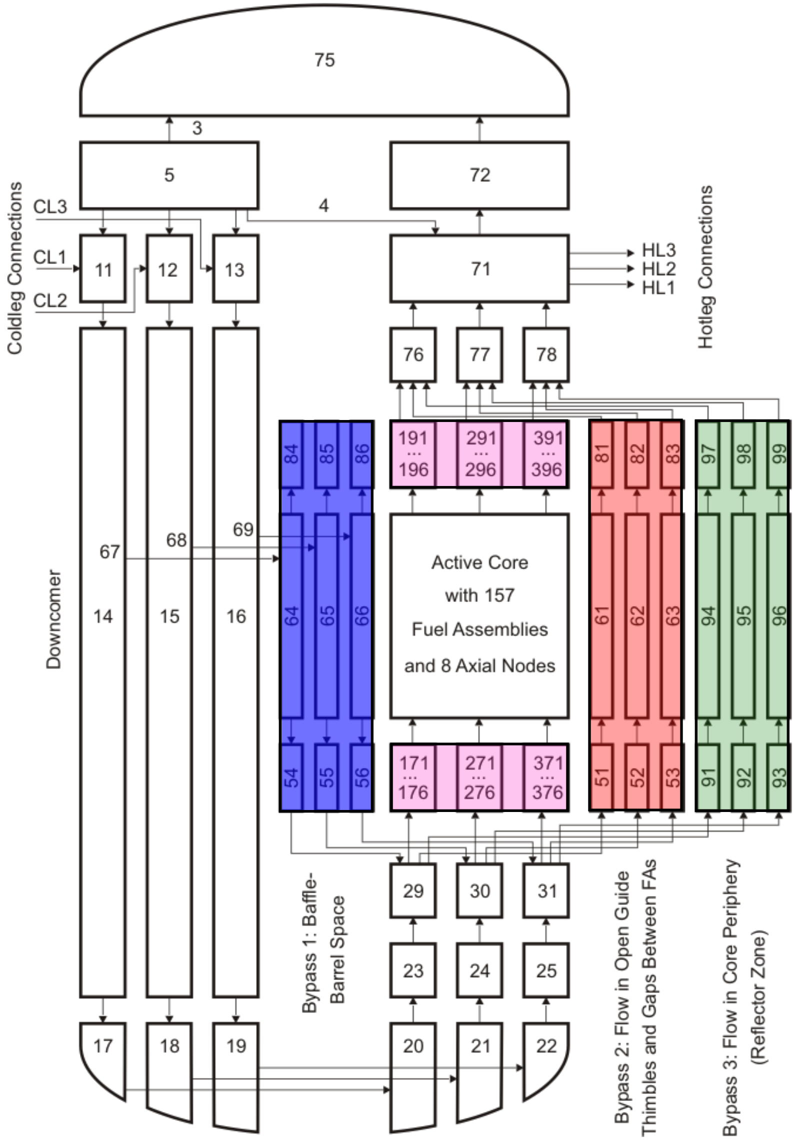
Tools for the neutronic/thermal-hydraulic mapping

- Two mappings are necessary:
  - **neutronic ↔ hydrodynamic cells** (which fuel temperature, moderator temperature/density from the thermal-hydraulic code needs to affected to a specific neutronic node);
  - **neutronic ↔ heat structures** (which fission power needs to be affected to a specific thermal-hydraulic “node”).
- Development of mapping tools was needed. The automatic mapping capabilities of PARCS were not sufficient for the sophisticated Ringhals-3 model.

The current stage of the RELAP5 development

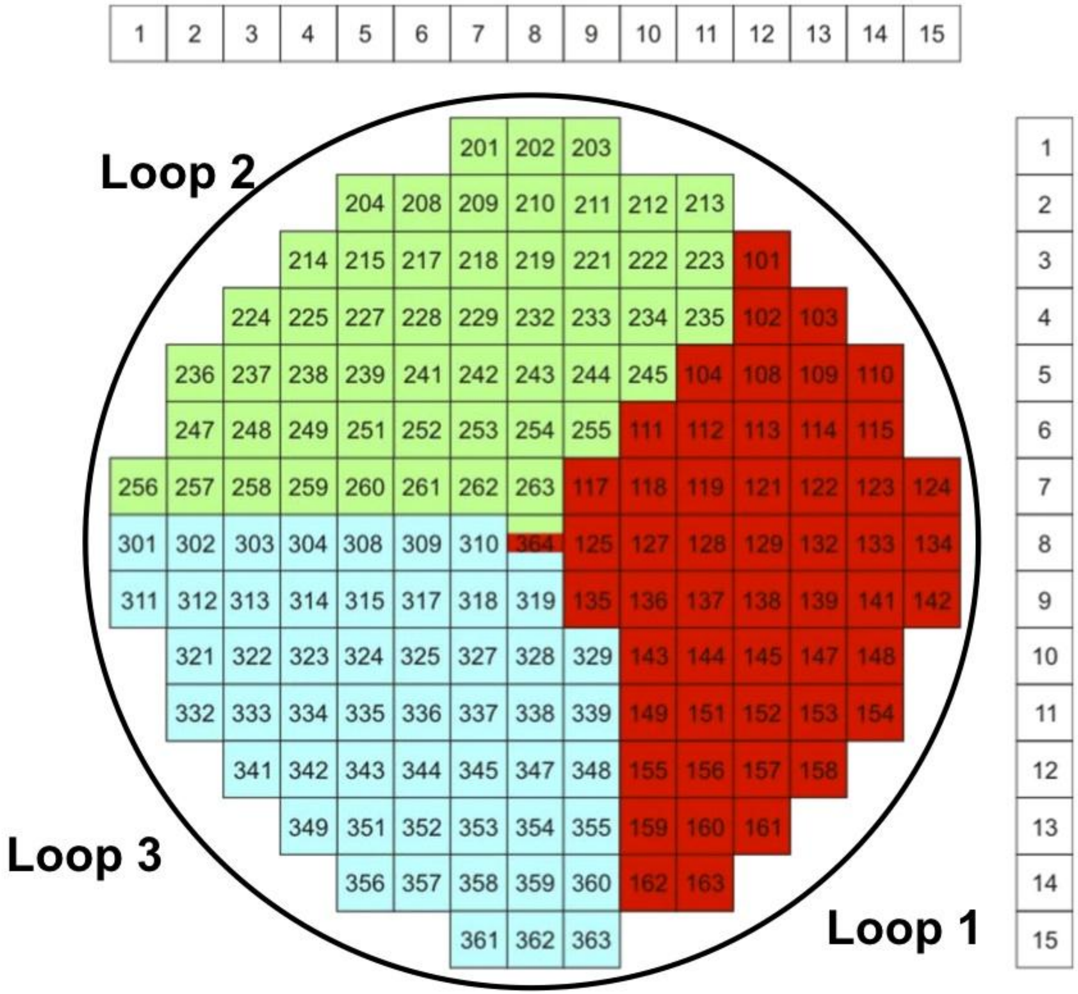
- RPV internals: completely restructured.
  - 3-loops configuration is retained within the RPV. This allows studying of asymmetric behaviour of the system in cases, such as MSLB or feed-water disturbances. New bypass channels are added to represent, for instance, the reflector.
- Now each of the 157 fuel assembly is modeled individually! This arrangement allows 1:1 radial coupling of the heat structures to the neutron kinetics calculation
- Steam-line: in order to capture the pressure pulse propagation, the number of sub-volumes is increased by 10x to achieve better resolution.
- Cleanup and updating of the control system and trip logic
- The full plant model consists of
  - 463 hydrodynamic components
  - 630 control components
  - 202 heat structures

Nodalization of the RPV

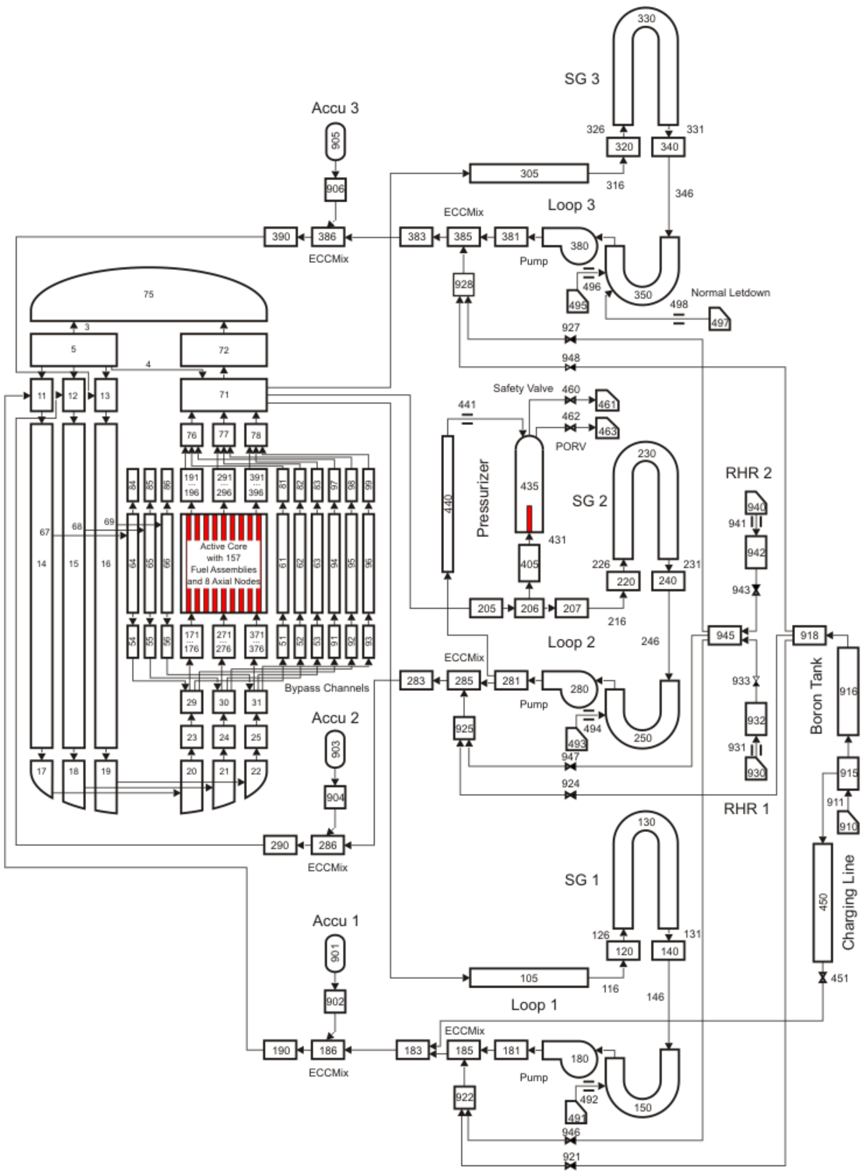


- The 3-loop structure is retained inside the RPV;
- Reflector zones are modeled by separate volumes;
- Heated and unheated bypass channels added;
- 18 branch components had to be added to the core inlet and outlet. (Reason: limit in RELAP5, max 9 connection to 1 branch).
- Accurate bypass flowrates achieved (plant data: 5% of the total primary flowrate)

Numbering scheme of the fuel assemblies

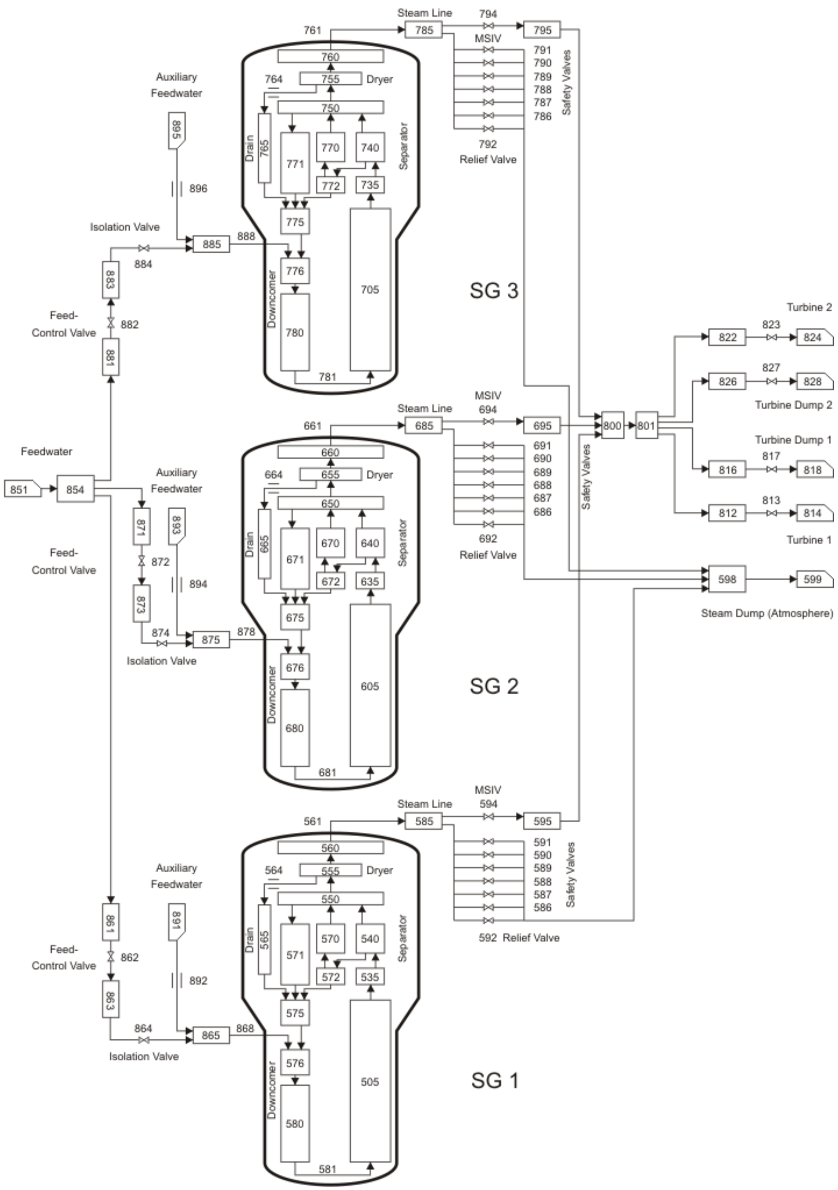


Nodalization of the primary side



- Can also be run in stand-alone mode of RELAP5
- Can be imported into the SNAP model editor
- Passed both steady-state and transient validation tests
- Coupled NK-TH calculations performed successfully

Nodalization of the secondary side



- A steam dump system is added with control components
- Auxiliary feedwater is modeled
- FW control system updated
- Proper recirculation ratio is achieved inside the SGs (~1:3.6)
- The feedwater temperature is given as a boundary condition (can be a function of time)
- The turbine system is modeled in a simplified way

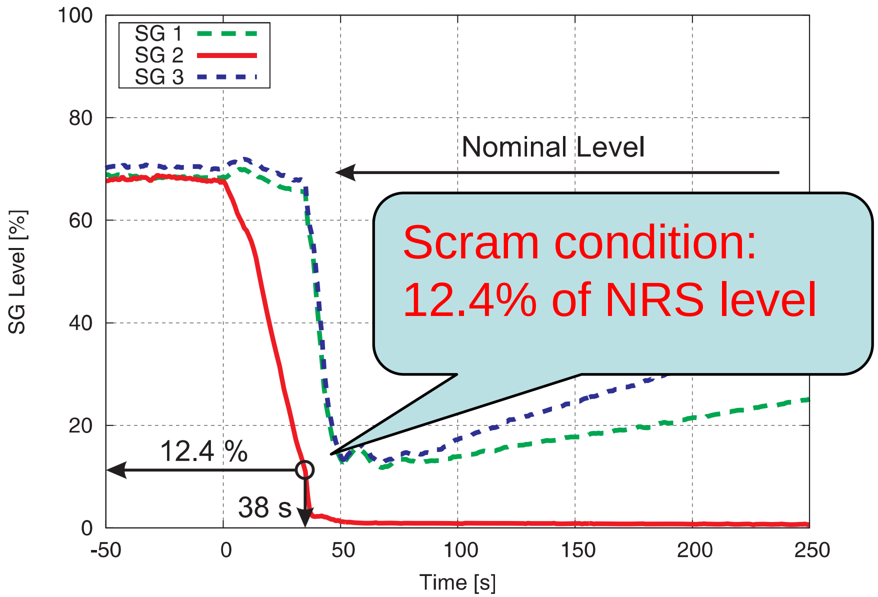
VALIDATION EXAMPLES

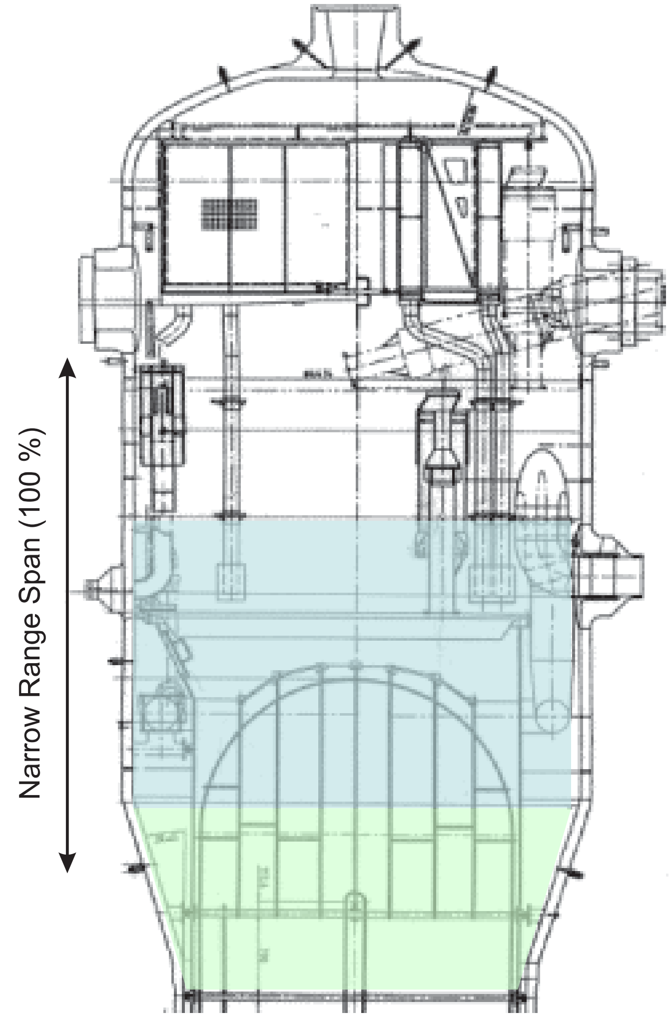
Loss of Normal Feedwater Transient

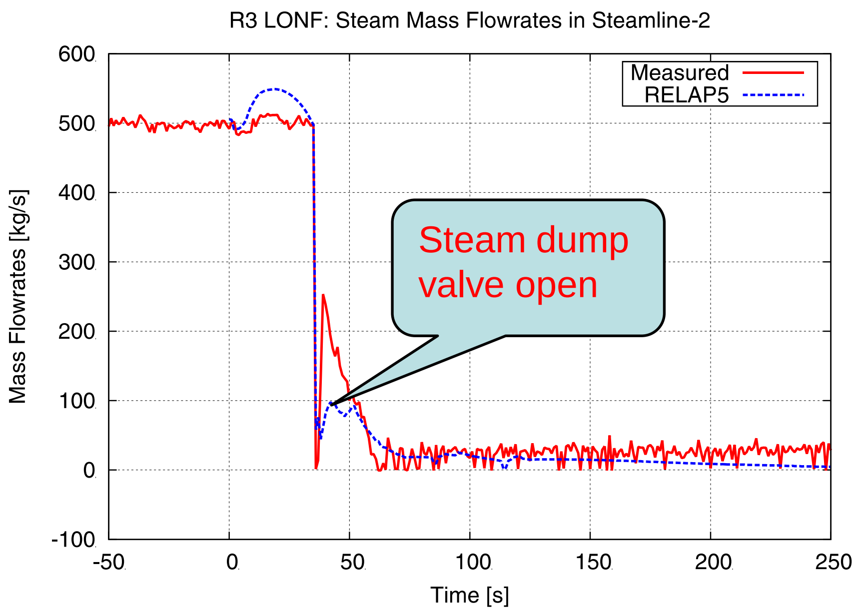
Loss of Normal Feedwater

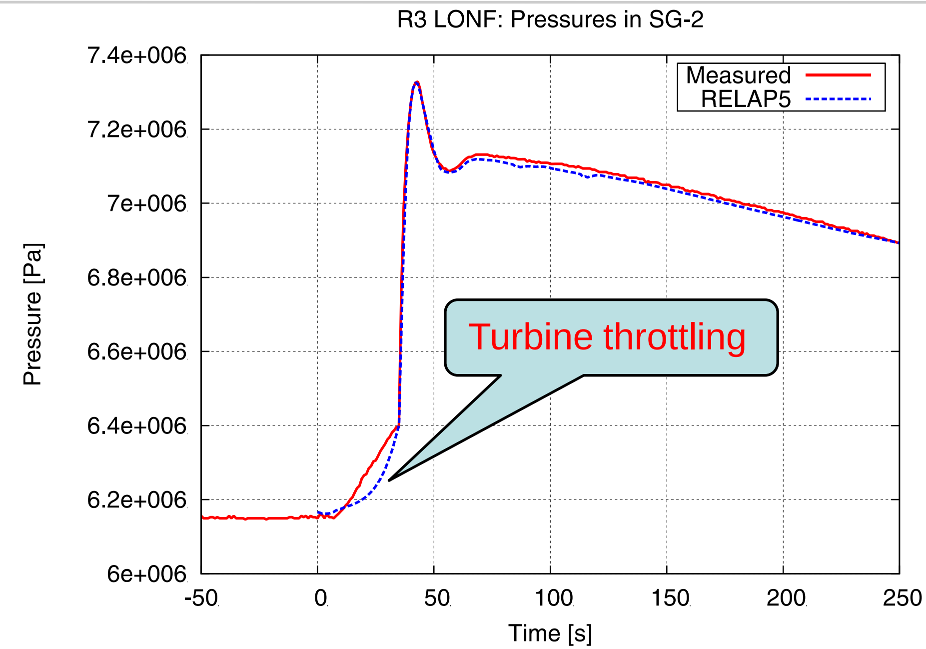
- **Purpose:**
  - **the validation exercise is suitable for evaluation of the model, focusing on the secondary side dynamics**
- **Transient occurred on 2005-08-16**
- **Reason: malfunction of SG-2 FW control valve**
- **SG-2 level drops below the top of the U-tubes**
- **Reactor is fully scrammed**
- **Auxiliary FW is sufficient for preventing of heat transfer from further degradation**

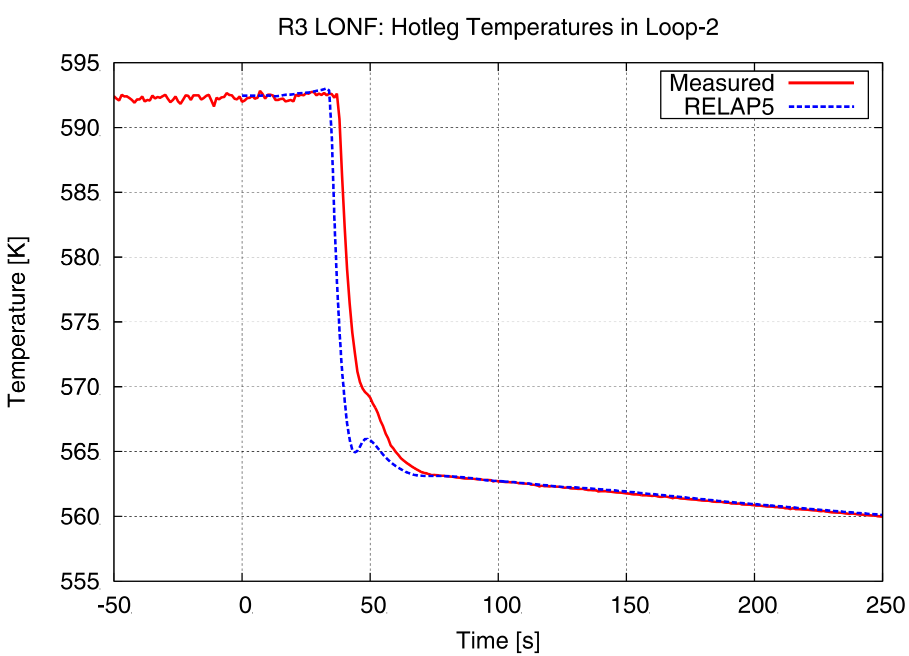
Narrow range SG levels during the LONF

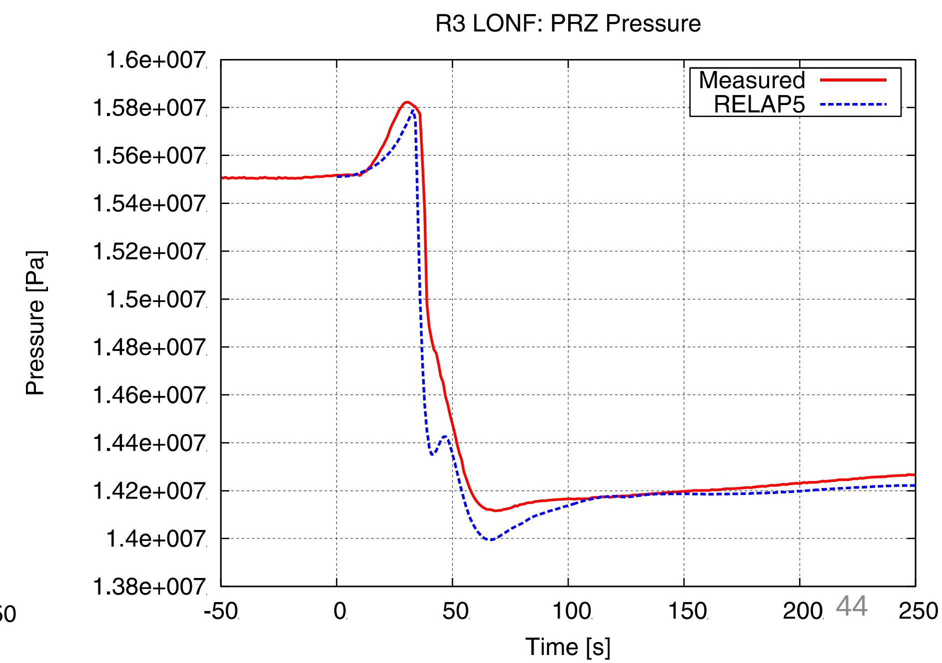




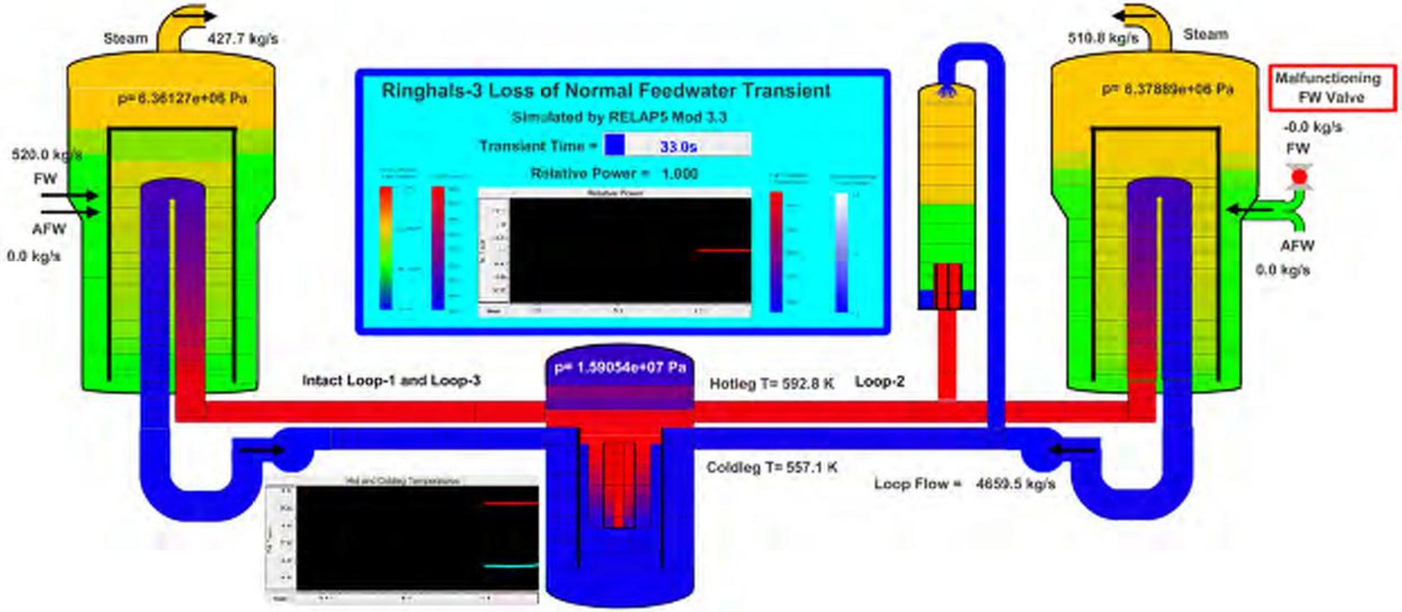

R3 LONF: Steam Mass Flowrates in Steamline-2
Mass Flowrates [kg/s]
Time [s]
Measured
RELAP5
Steam dump valve open
600
500
400
300
200
100
0
-100
-50
0
50
100
150
200
250


R3 LONF: Pressures in SG-2
Pressure [Pa]
Time [s]
Measured
RELAP5
Turbine throttling
7.4e+006
7.2e+006
7e+006
6.8e+006
6.6e+006
6.4e+006
6.2e+006
6e+006
-50
0
50
100
150
200
250

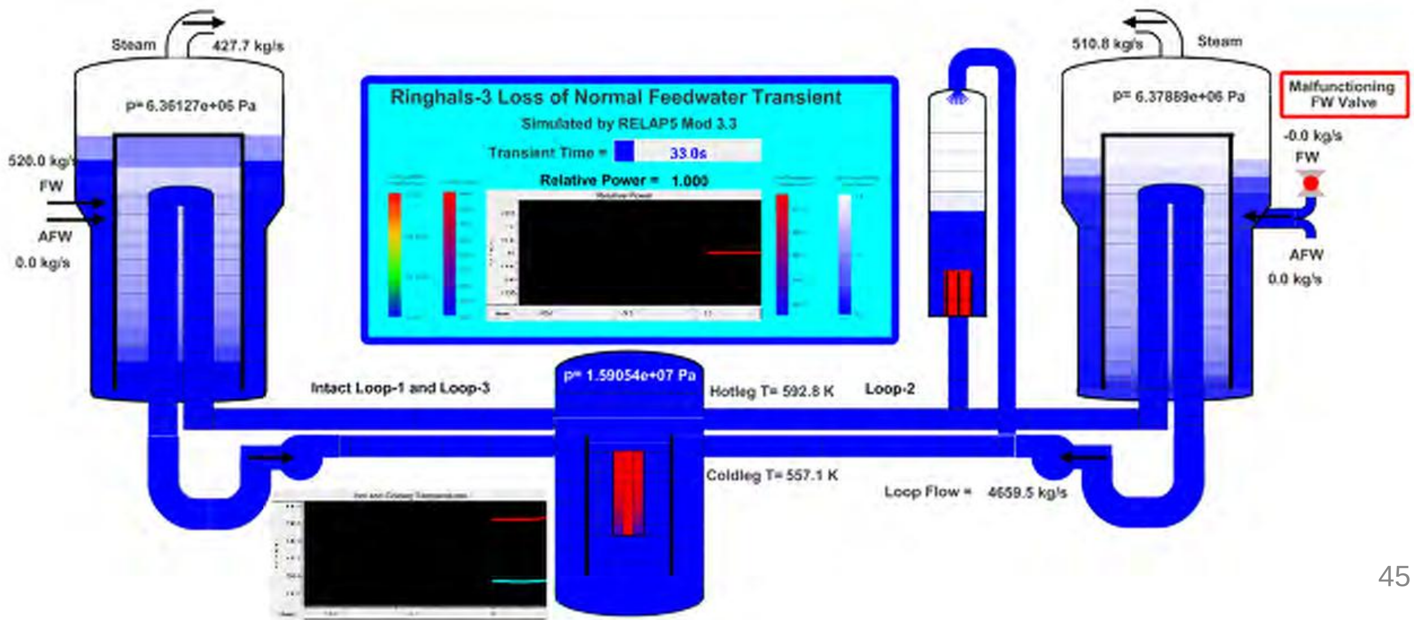

R3 LONF: Hotleg Temperatures in Loop-2
Temperature [K]
Time [s]
Measured
RELAP5
595
590
585
580
575
570
565
560
555
-50
0
50
100
150
200
250


R3 LONF: PRZ Pressure
Pressure [Pa]
Time [s]
Measured
RELAP5
1.6e+007
1.58e+007
1.56e+007
1.54e+007
1.52e+007
1.5e+007
1.48e+007
1.46e+007
1.44e+007
1.42e+007
1.4e+007
1.38e+007
-50
0
50
100
150
200
250

Temperature and fluid condition



Void fraction distribution

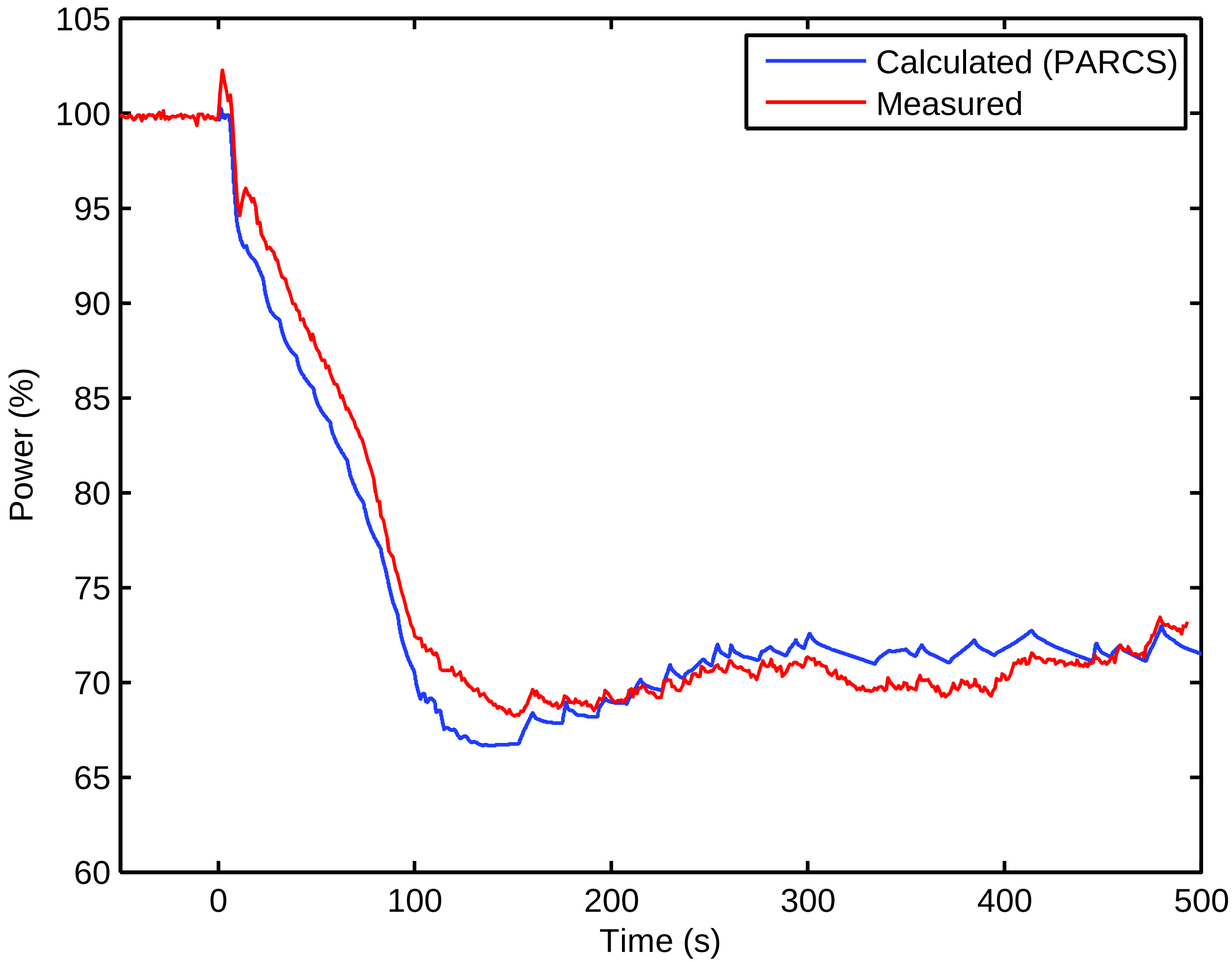


Analysis of Load Rejection Transient at Ringhals-3

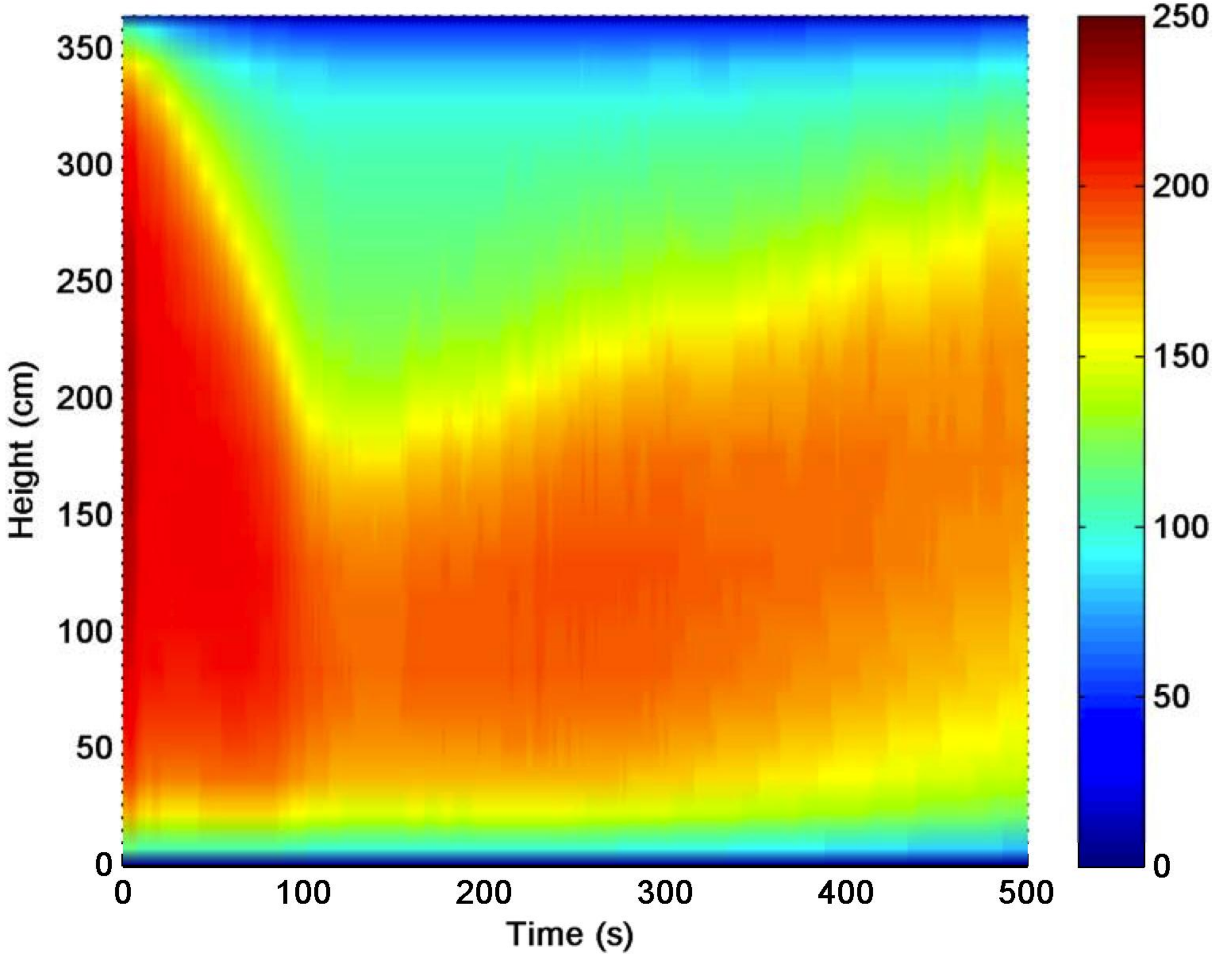
- A coupled RELAP5/PARCS model of R3 NPP was developed and validated against a load rejection transient.
  - Challenging for coupled calculations due to strong feedback effects.
- The transient occurred on November 28, 2010
  - A scheduled fully-instrumented test
  - Purpose: to verify the reactor transient behavior and to observe the capability of the control system of preventing generator trip after the power up-rating to 3144 MWt and after the modernization of the turbine control and protection system.

- Modifications to the existing RELAP5 model have been performed, especially to the control logic of some components (PRZ, SG, bypass dump valves, FW valves) and inclusion of boron injection.
- New cross-section and history data sets for Cycle 28, incorporating the new model of the shielding assemblies.
- Movement of the control rods during the transient is used as boundary condition in PARCS.
- The model produces relatively good agreement with measured data.

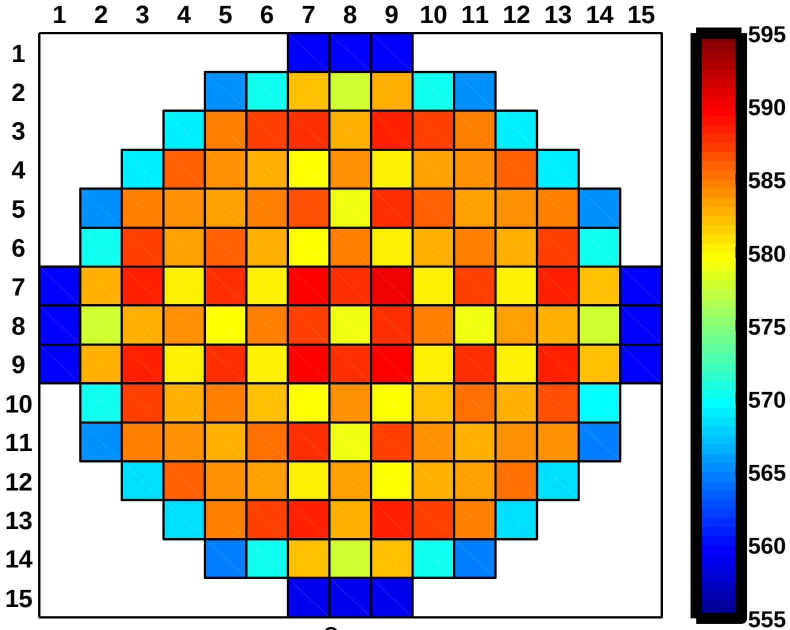
Nuclear power

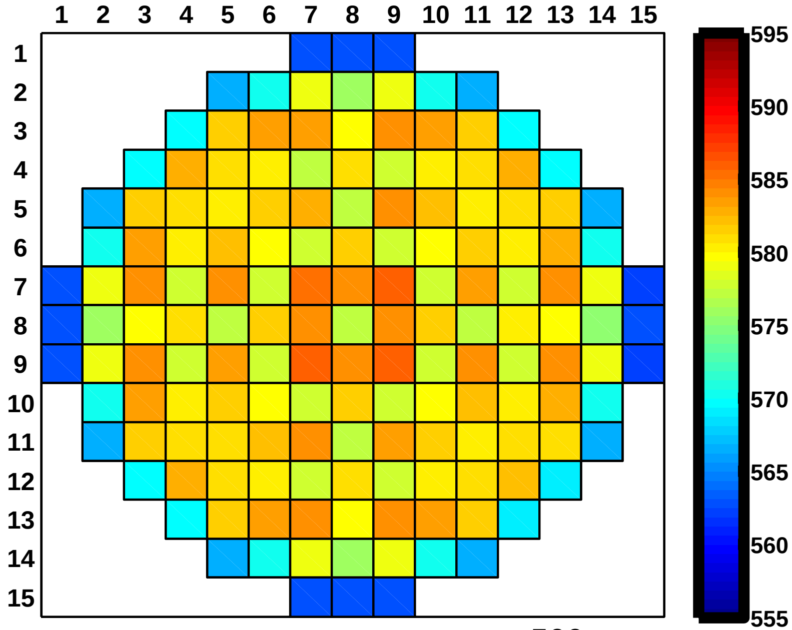


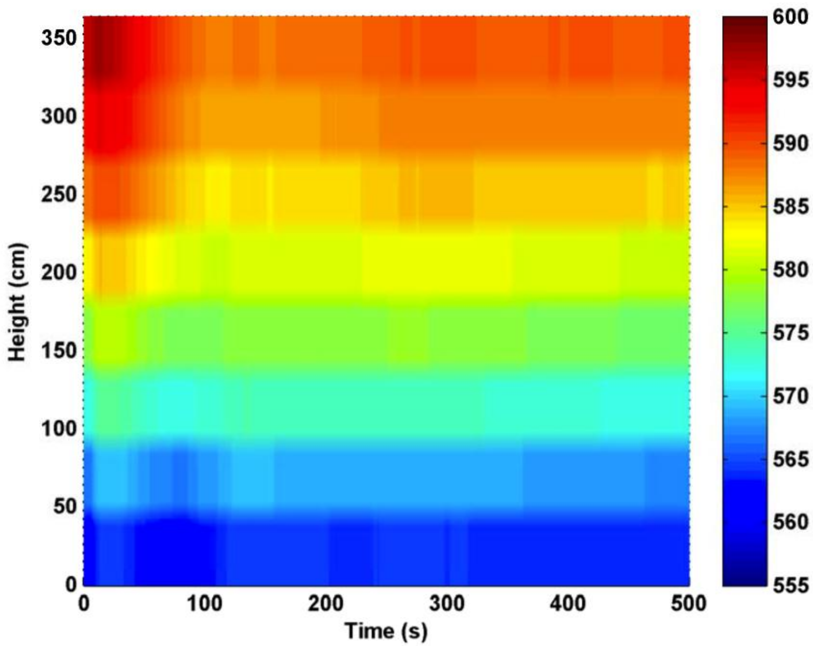
Axial power profile



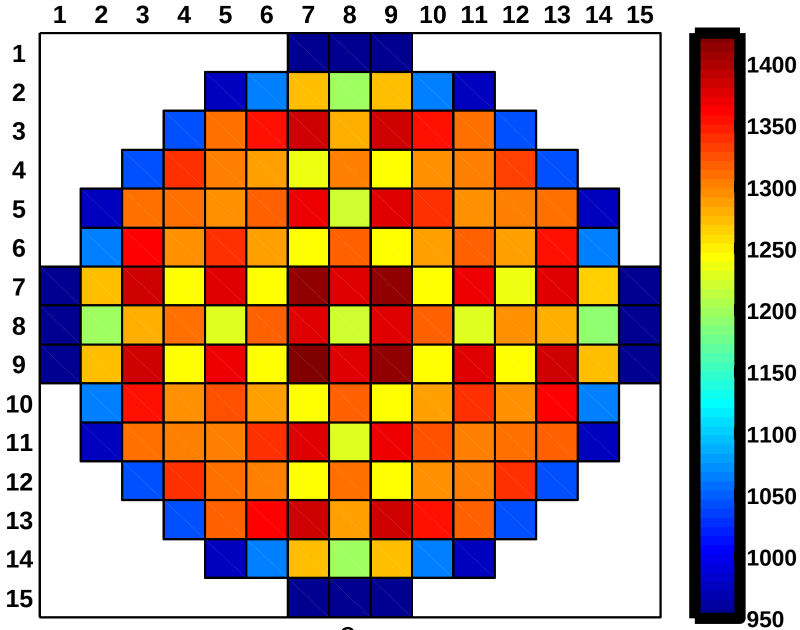
Moderator temperature



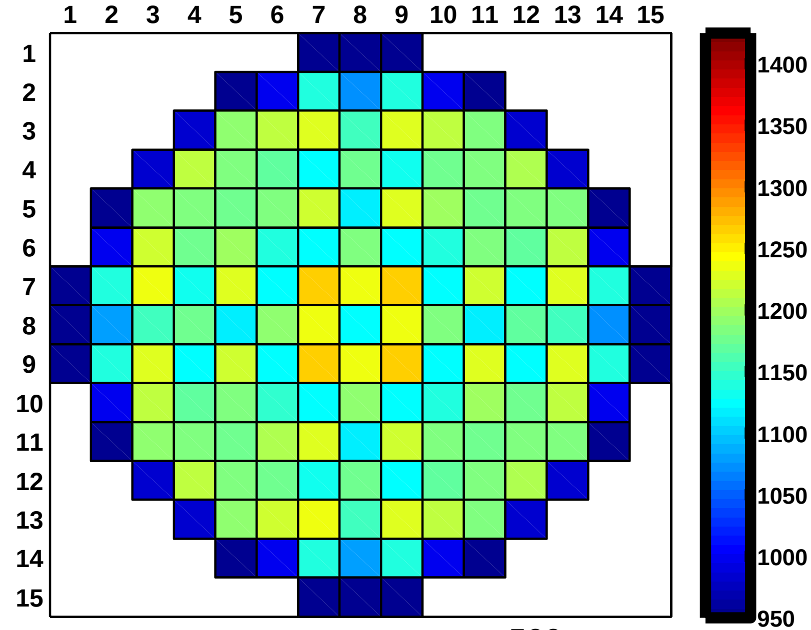




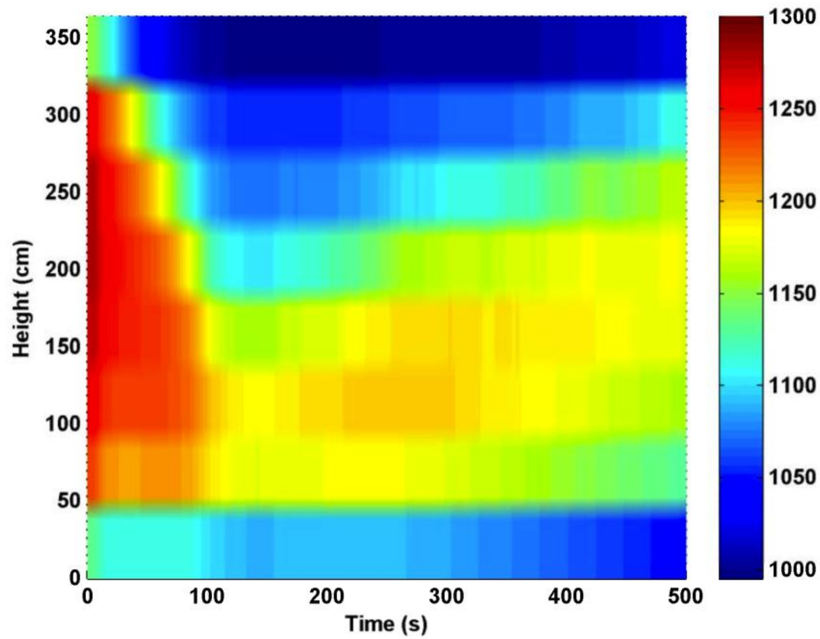
Fuel temperature

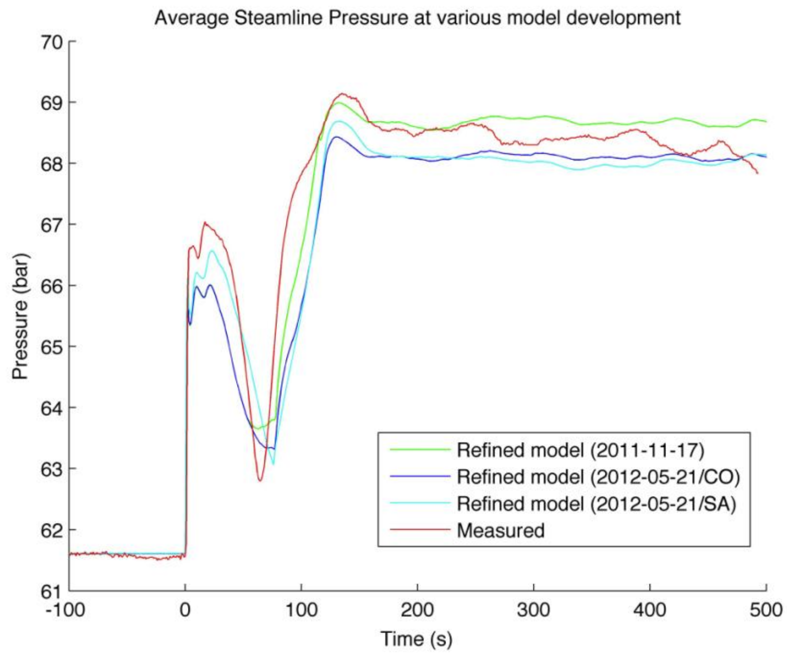


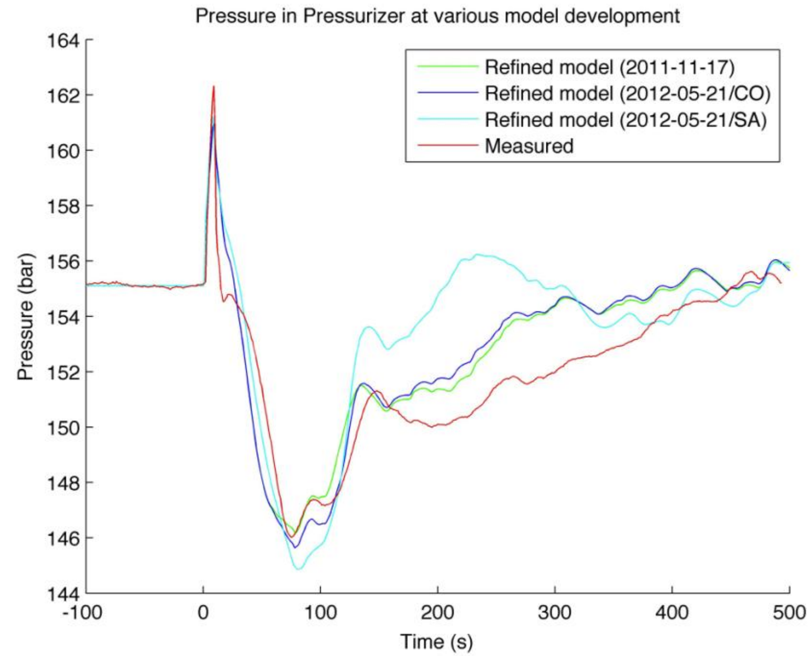
0 s

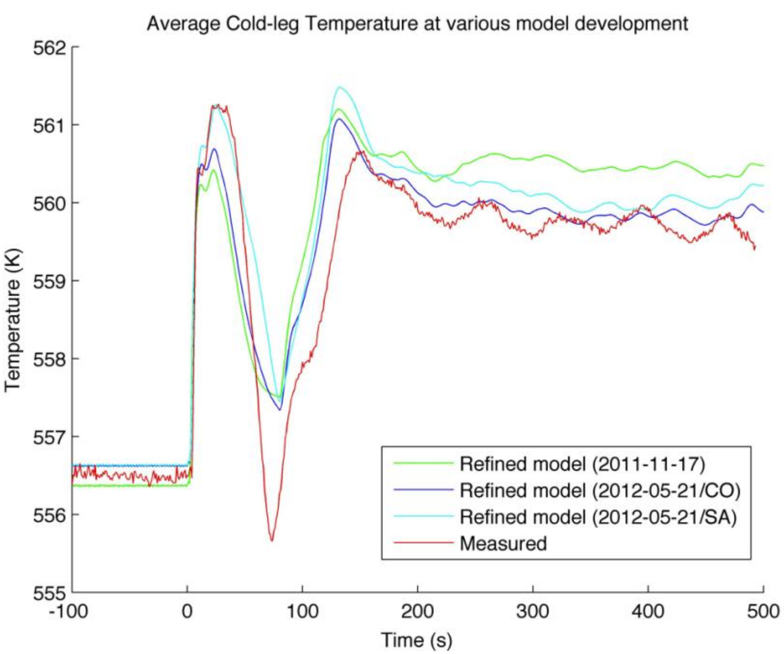


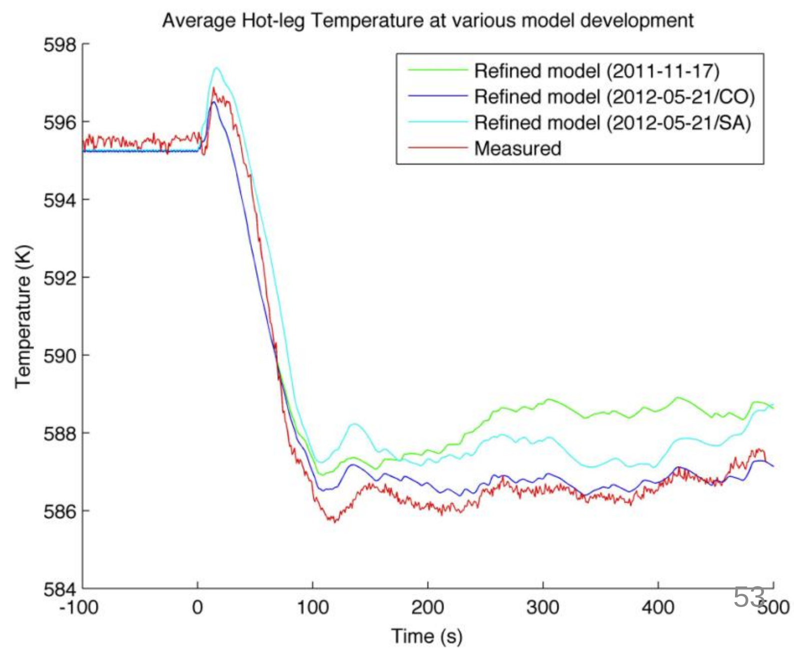
500 s




Average Steamline Pressure at various model development
Pressure (bar)
Time (s)
Refined model (2011-11-17)
Refined model (2012-05-21/CO)
Refined model (2012-05-21/SA)
Measured


Pressure in Pressurizer at various model development
Pressure (bar)
Time (s)
Refined model (2011-11-17)
Refined model (2012-05-21/CO)
Refined model (2012-05-21/SA)
Measured


Average Cold-leg Temperature at various model development
Temperature (K)
Time (s)
Refined model (2011-11-17)
Refined model (2012-05-21/CO)
Refined model (2012-05-21/SA)
Measured

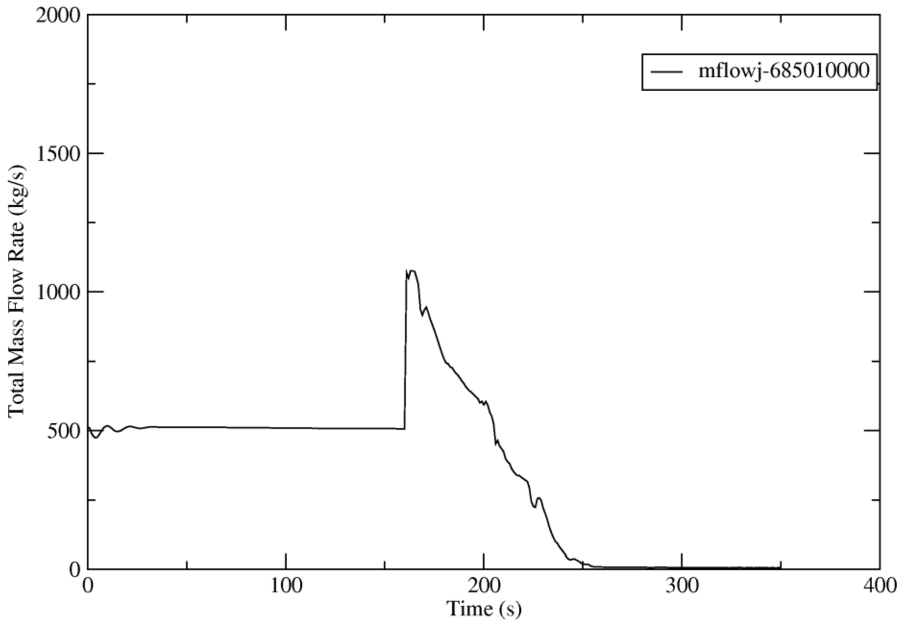

Average Hot-leg Temperature at various model development
Temperature (K)
Time (s)
Refined model (2011-11-17)
Refined model (2012-05-21/CO)
Refined model (2012-05-21/SA)
Measured

Hypothetical Main Steam Line Break

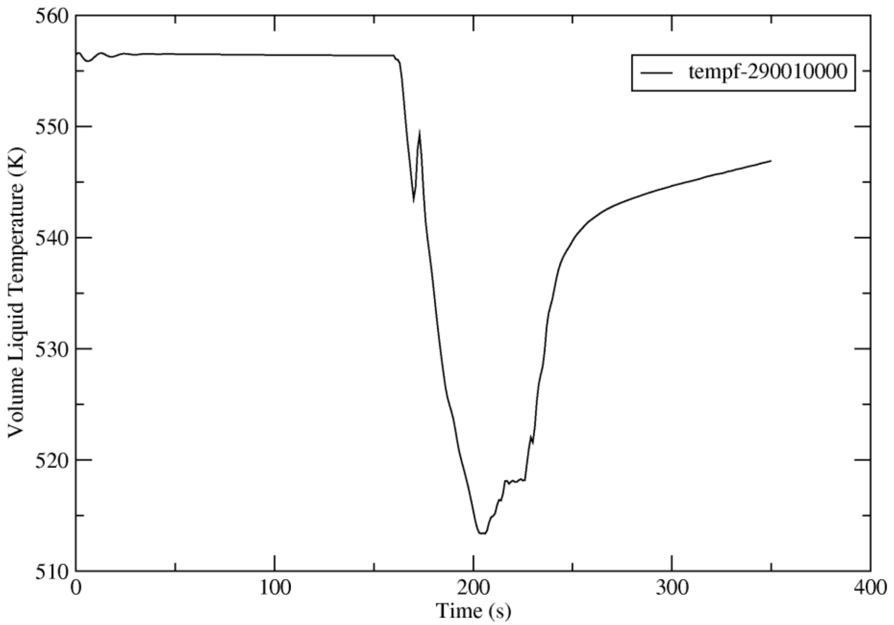
Modelling of the MSLB case

- The break is located in the middle of steamline in the 2nd loop
- Modelled with a trip valve connected to a TDV with atmospheric conditions
- Cycle exposure of 12.9742 GWd/tHM (EOC): most negative Moderator Temp. Coeff. (MTC)
- Hot Full Power (HFP) conditions

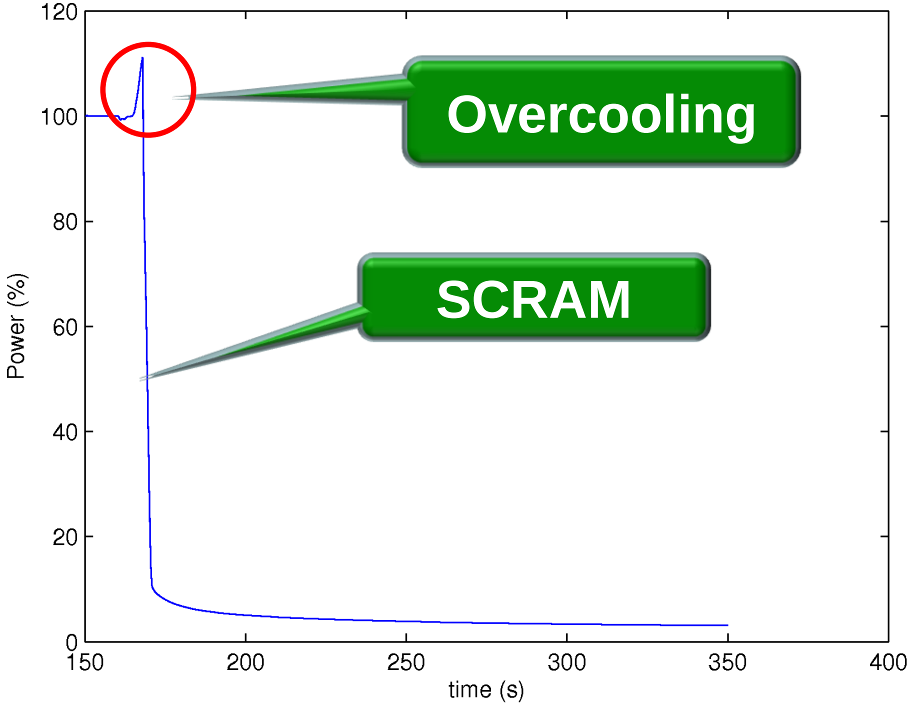
Steam flowrate in SL-2



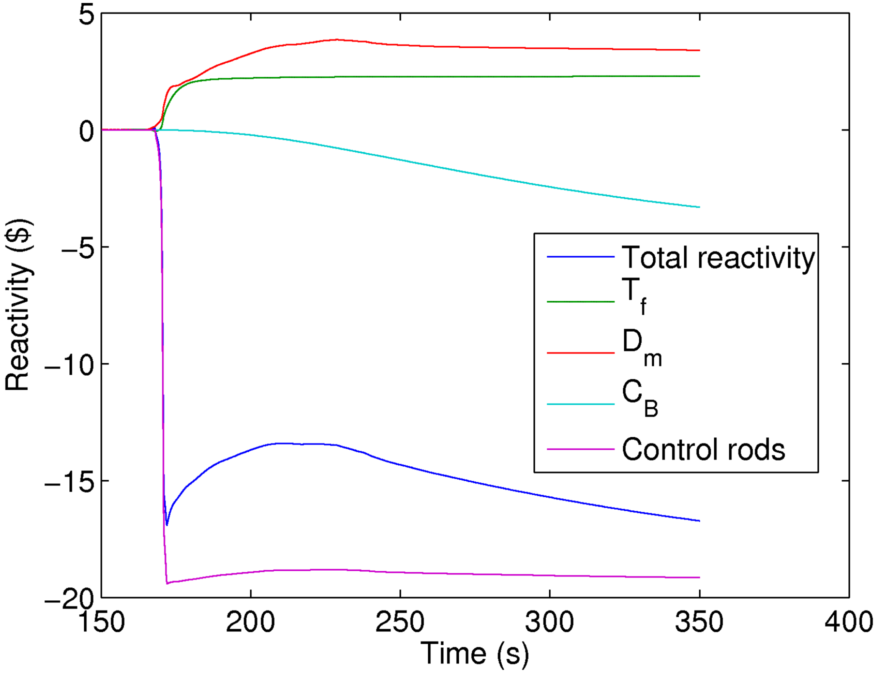
Cold-leg temperature in Loop 2



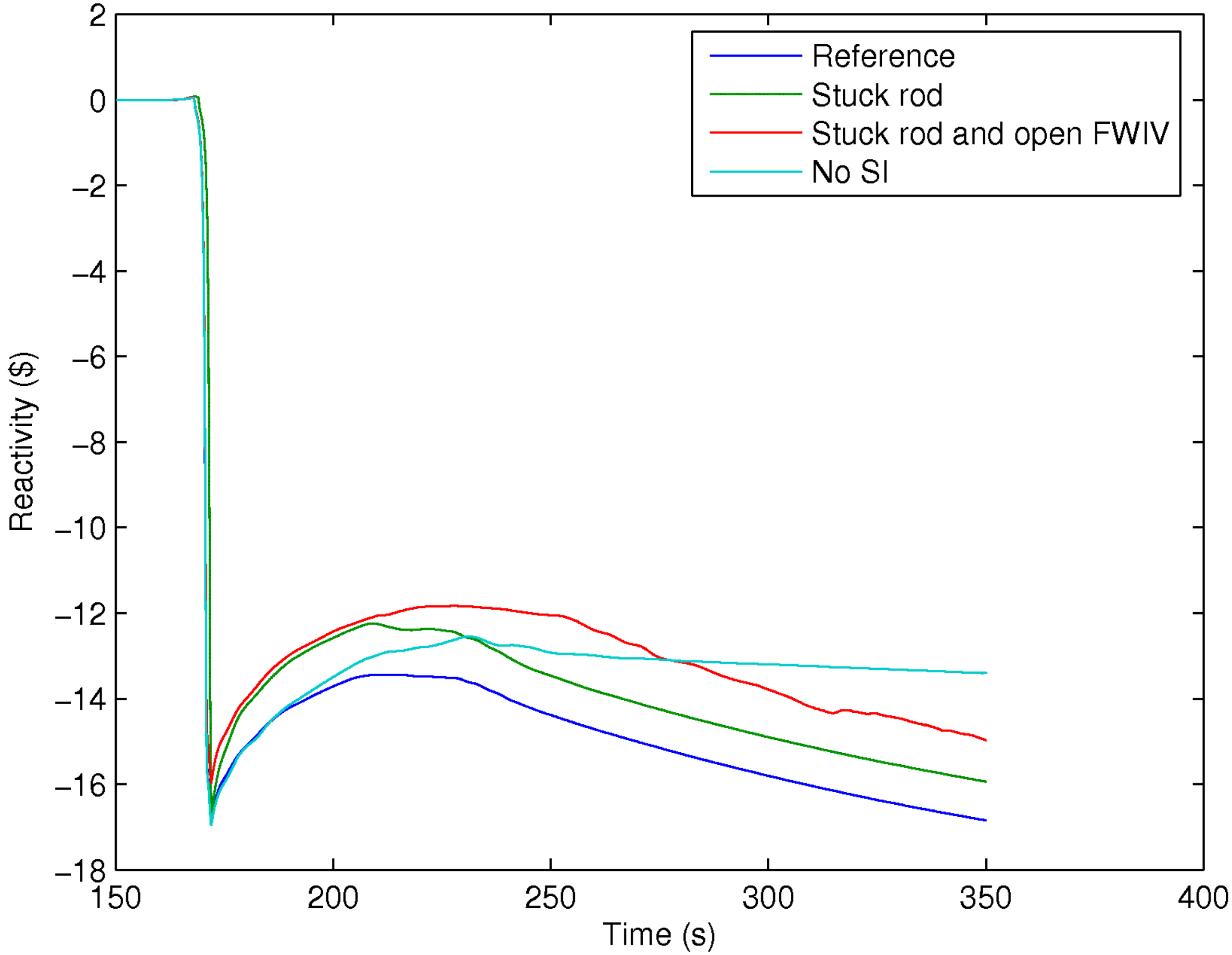
Calculated power



Reactivity components



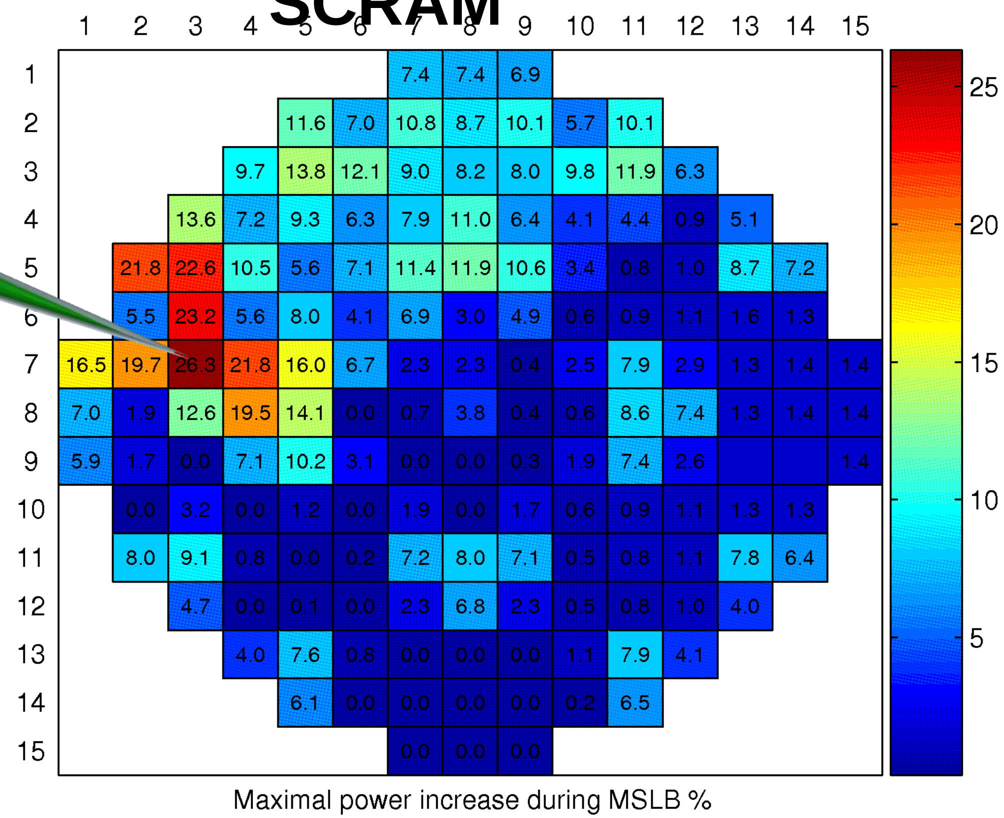
Reactivities in various scenarios



A stuck control rod case

Assembly-wise power increase before SCRAM

Location of the stuck rod



Ringhals 3, c22

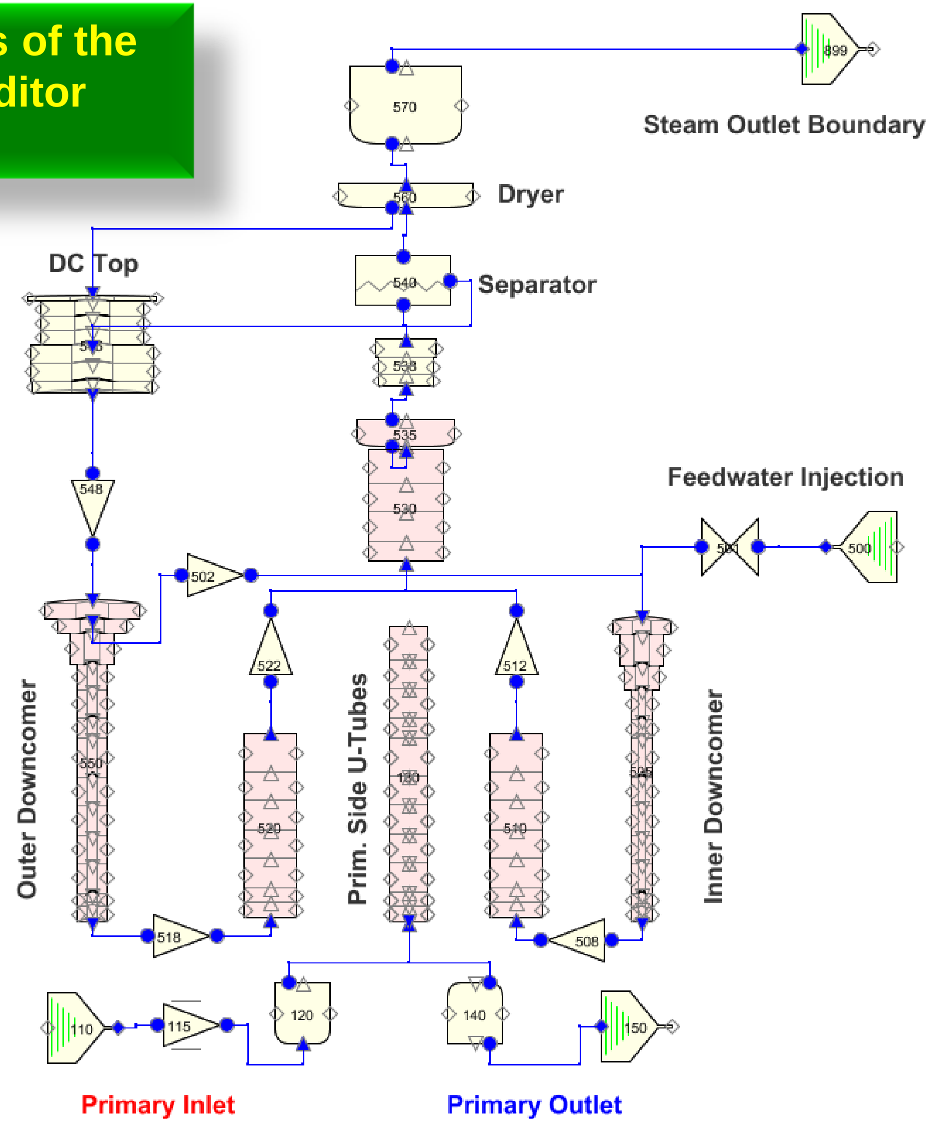
Development of New Component Models for Ringhals-4

Ringhals 4 related projects

- **FREJ**:
  - Replacement of the 3 steam generators and the pressurizer with new AREVA design components
- **TURBO**:
  - Refurbishment of high pressure turbines, replacement of four main steam reheaters, installation of two new low pressure pre-heaters, refurbishment of main feed water pumps etc.
- **NICE**:
  - Modernization of turbine protection and control systems
- **QUATTRO**:
  - The project manages the hot testing period that is undertaken after the conclusion of the projects above. During the test operation the thermal power remains 2783 MW (100 %), but the ultimate goal is to uprate it to 3300 MW (118.6 %).

Development of the new RELAP5 SG Model

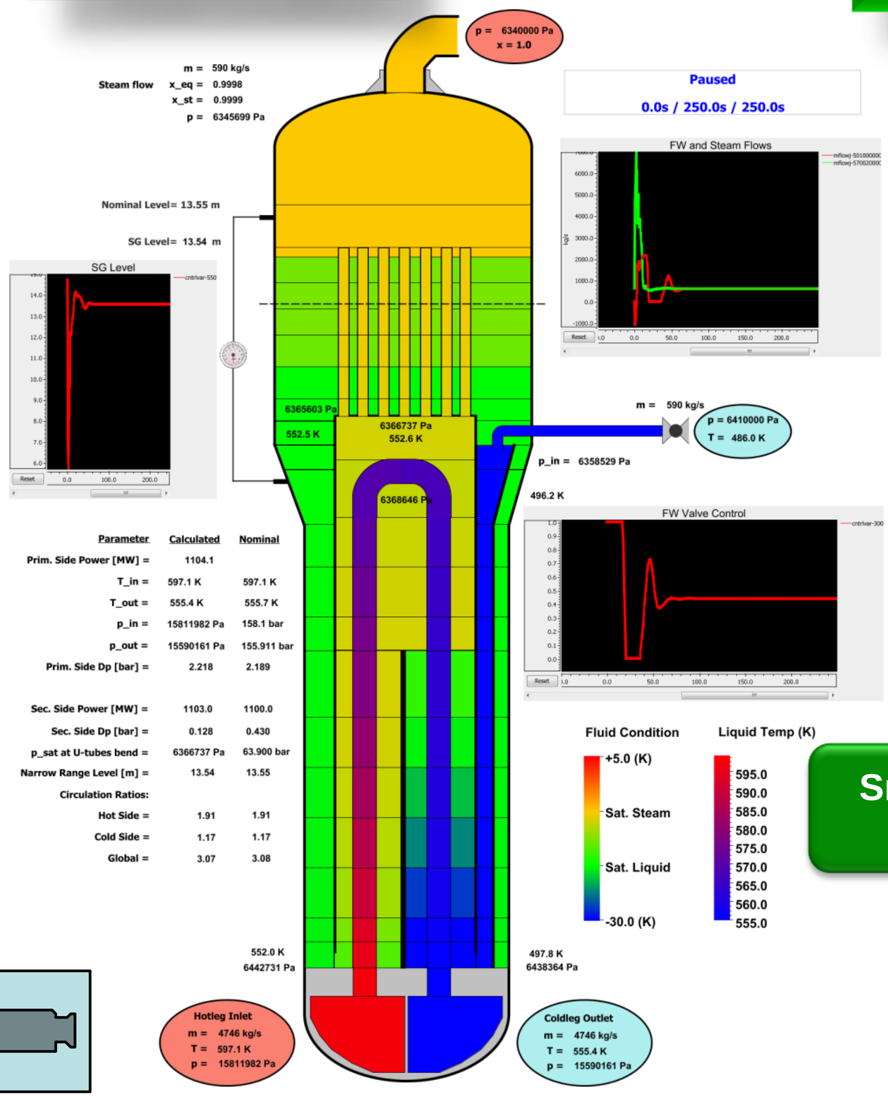
Hydrodynamic components of the SG in the SNAP Model Editor

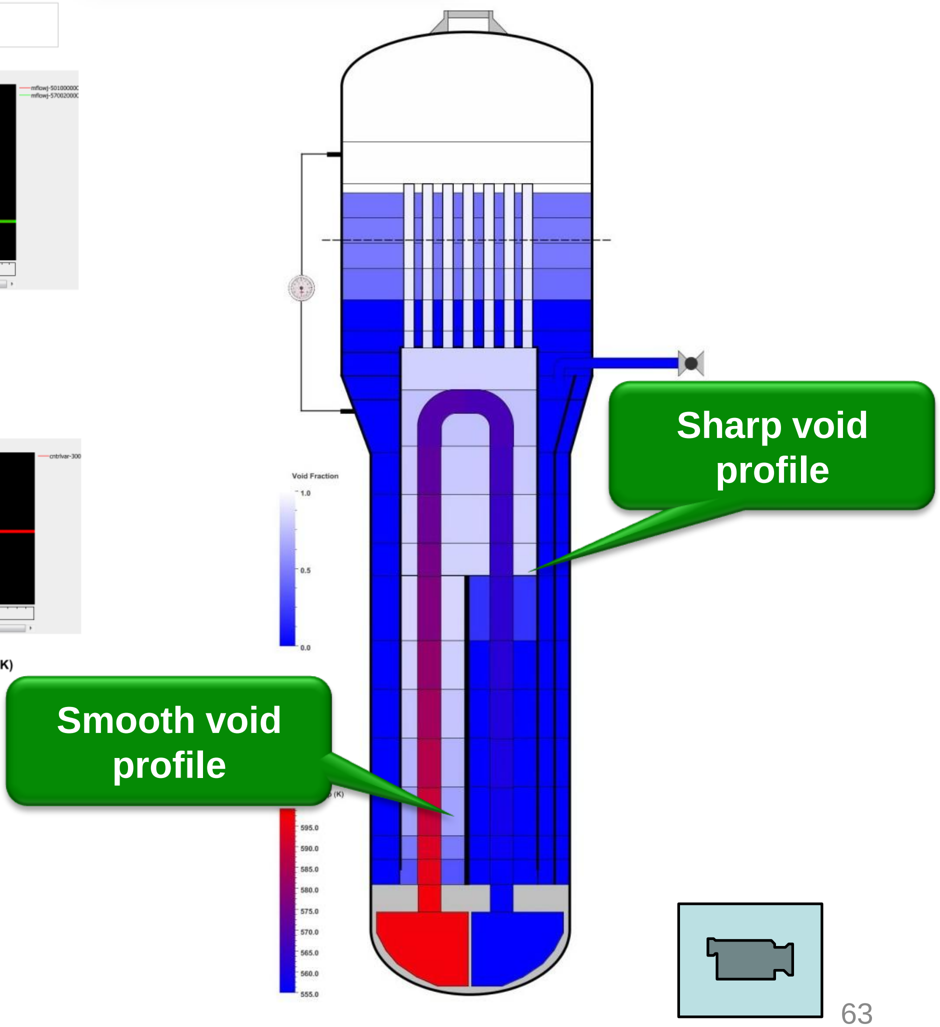


Visualization: proportionally scaled down components

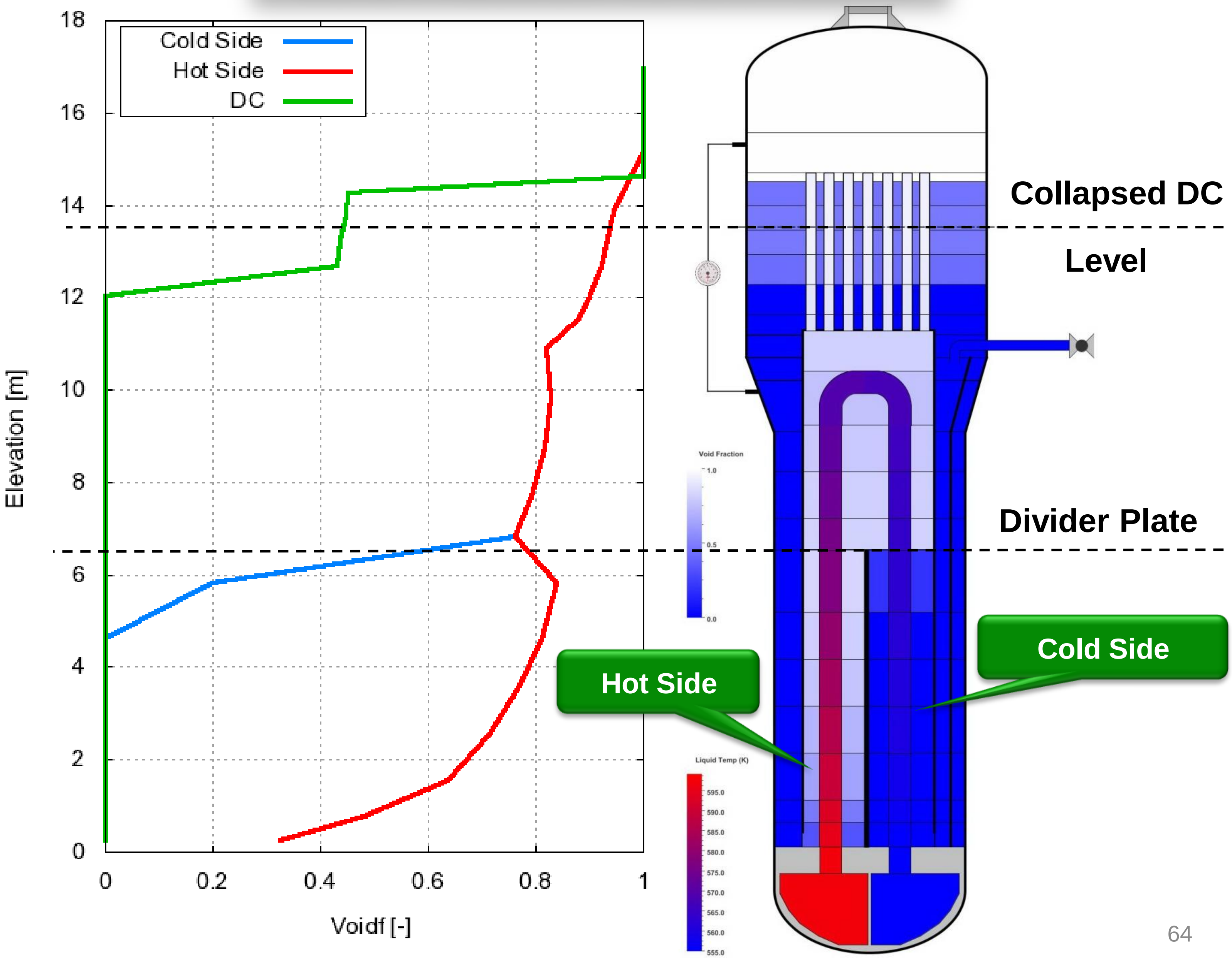
Fluid Condition

Void Fraction





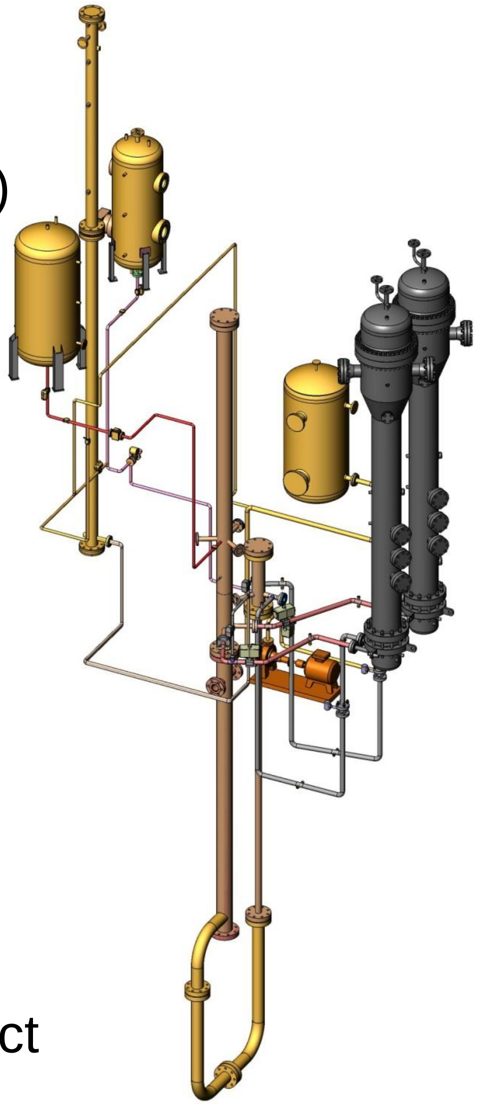
Void Fraction vs Elevation



Participation in the PACTEL SBL-50 Benchmark

The PACTEL SBL-50 Benchmark

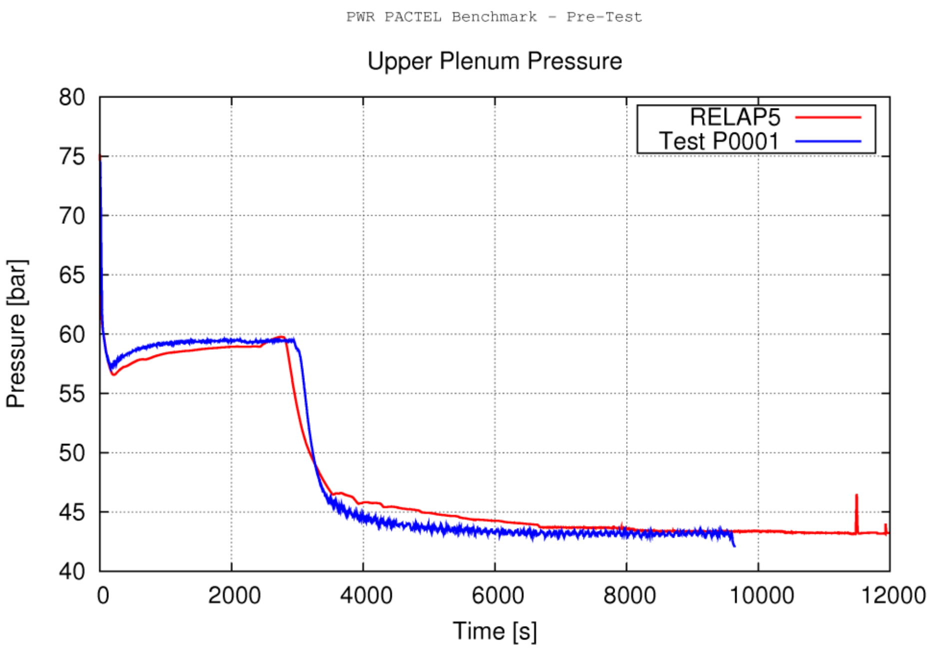
- The Benchmark Test
  - Performed in the PWR PACTEL Facility (Finland)
  - 1 mm SB-LOCA in Cold-Leg of Loop 2
  - Nearly continuous inventory reduction until core heat-up, under natural circulation conditions
  - Structure of SG is similar to EPR SG
  - Sec. side parameters are kept constant

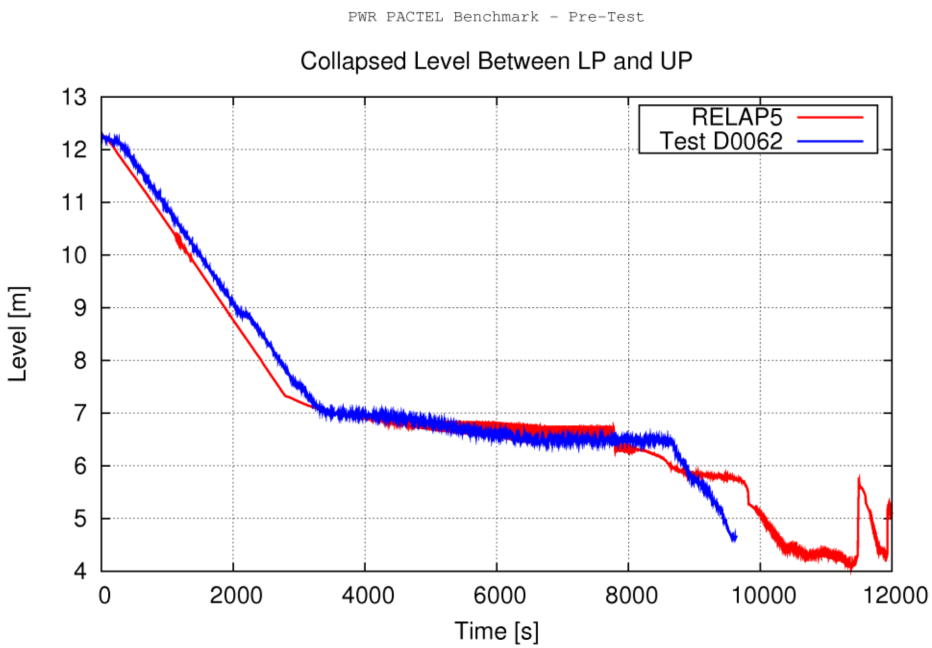


- Participation of Chalmers
  - RELAP5/Mod 3.3 used
  - Pre-test phase (blind calculation)
  - Post-test phase (open calculations)
  - Further development within a M.Sc. Thesis Project

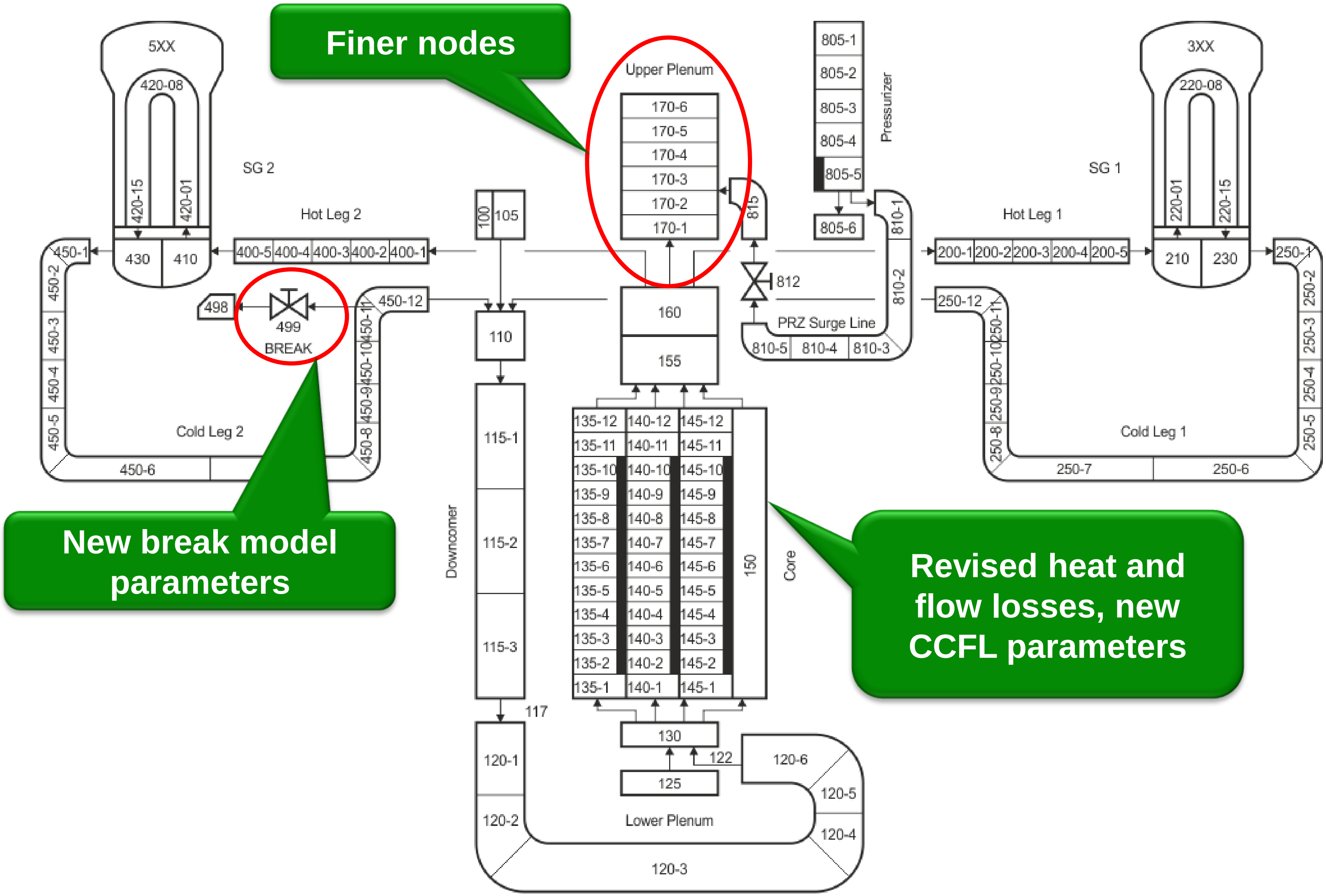
Pre-test results

Good general agreement already in the blind calculations

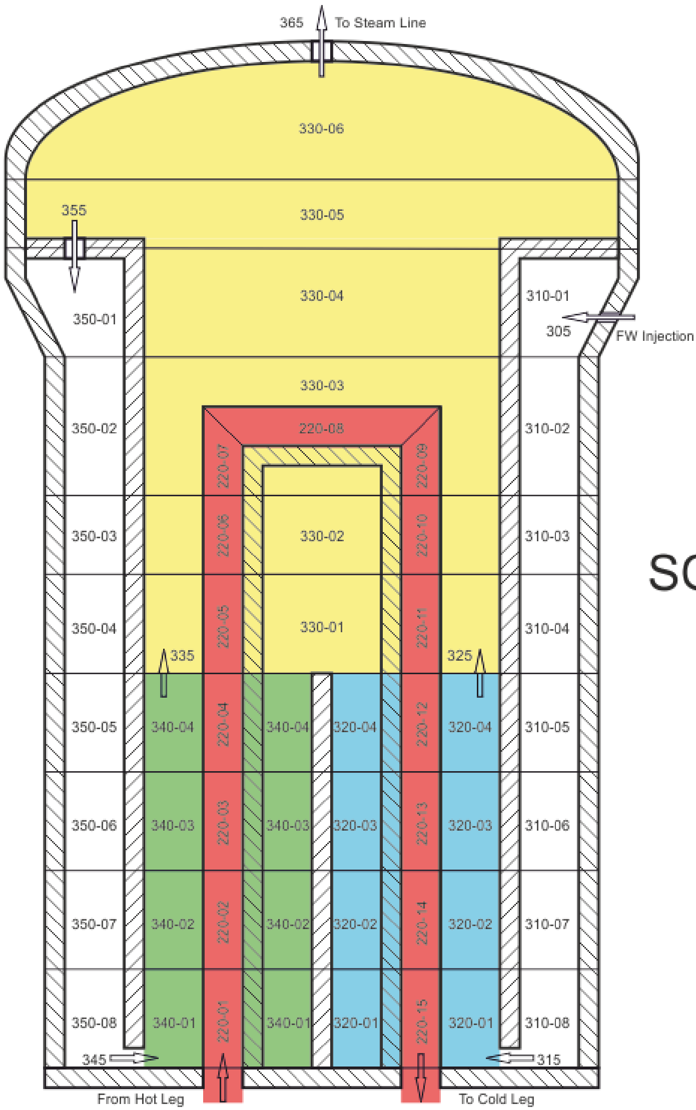




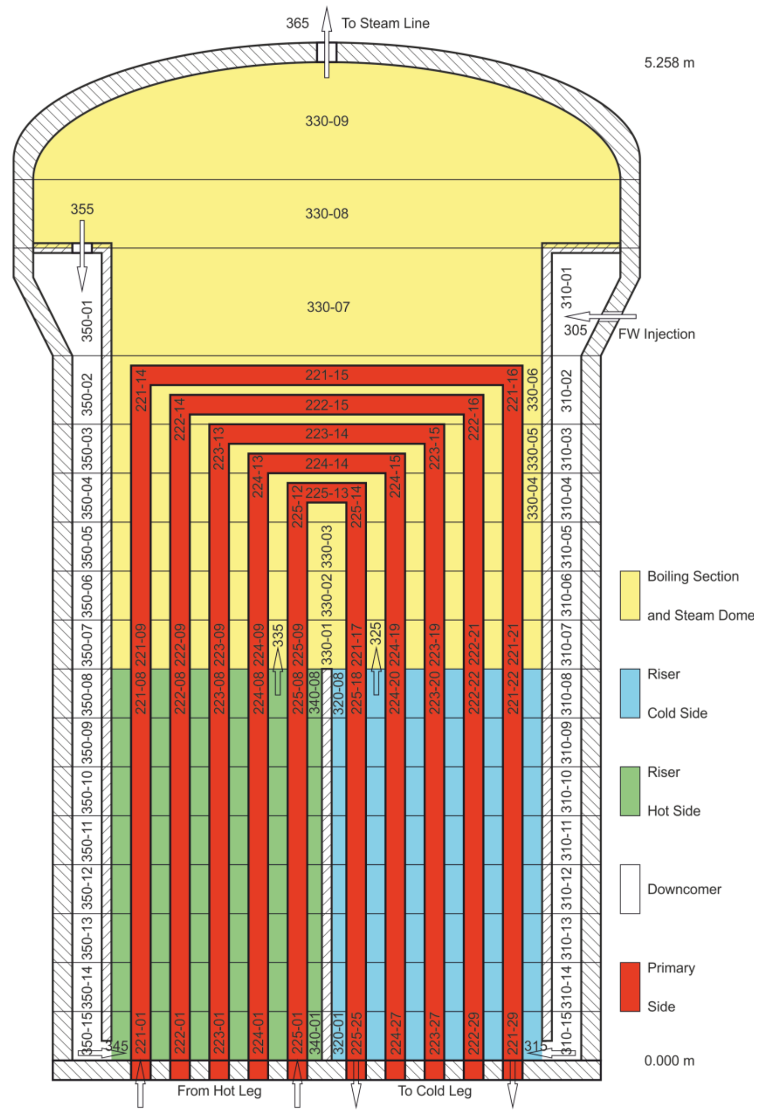
Changes between pre and post-tests



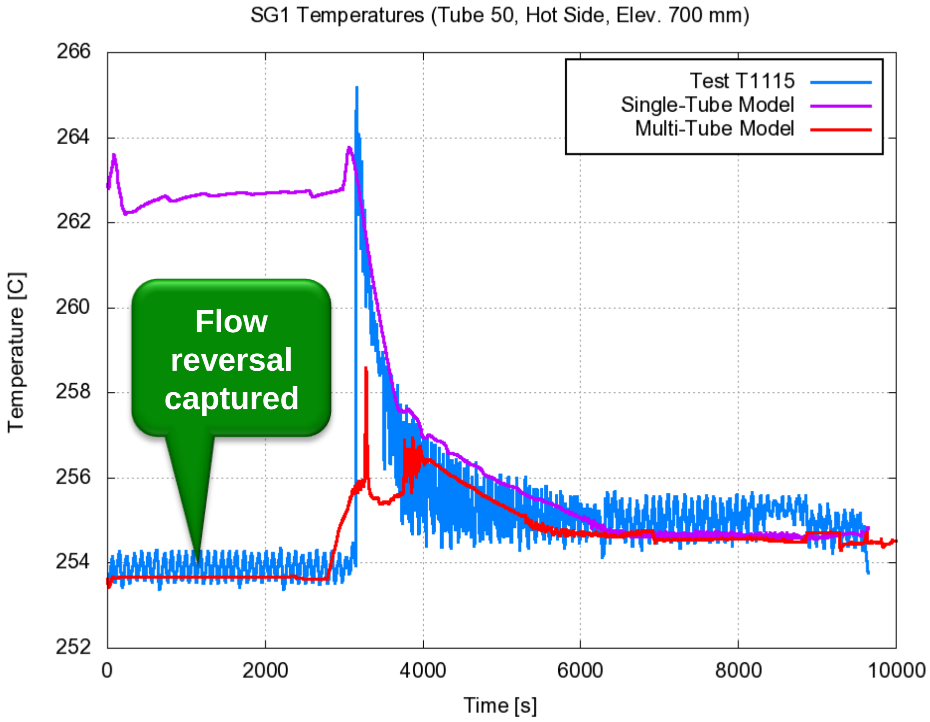
The single and multi-tube SG models

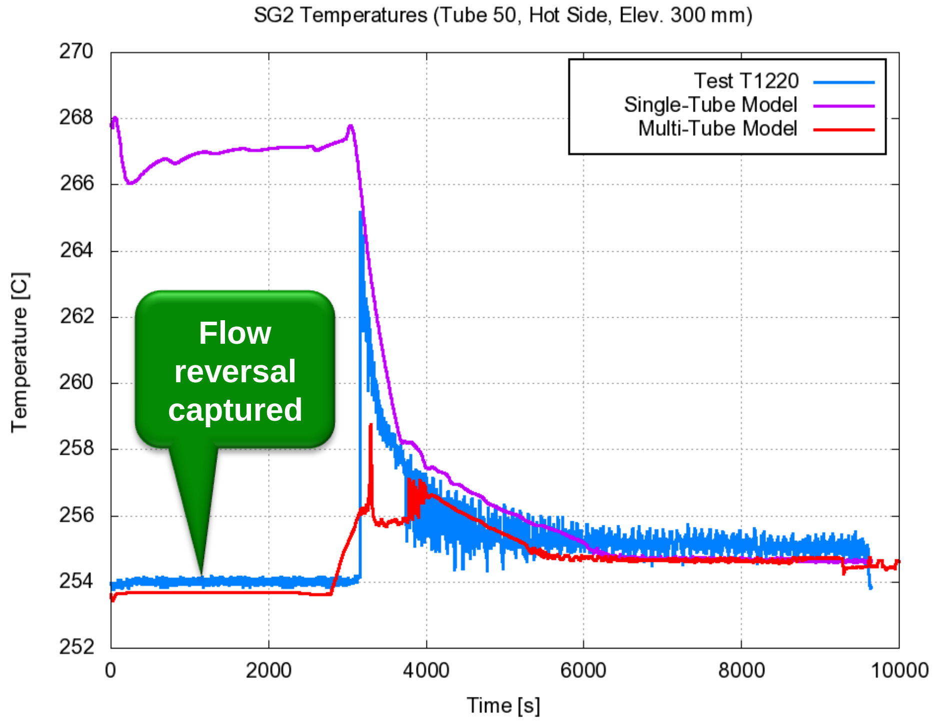


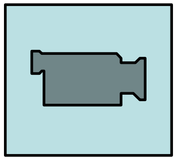
SG 1



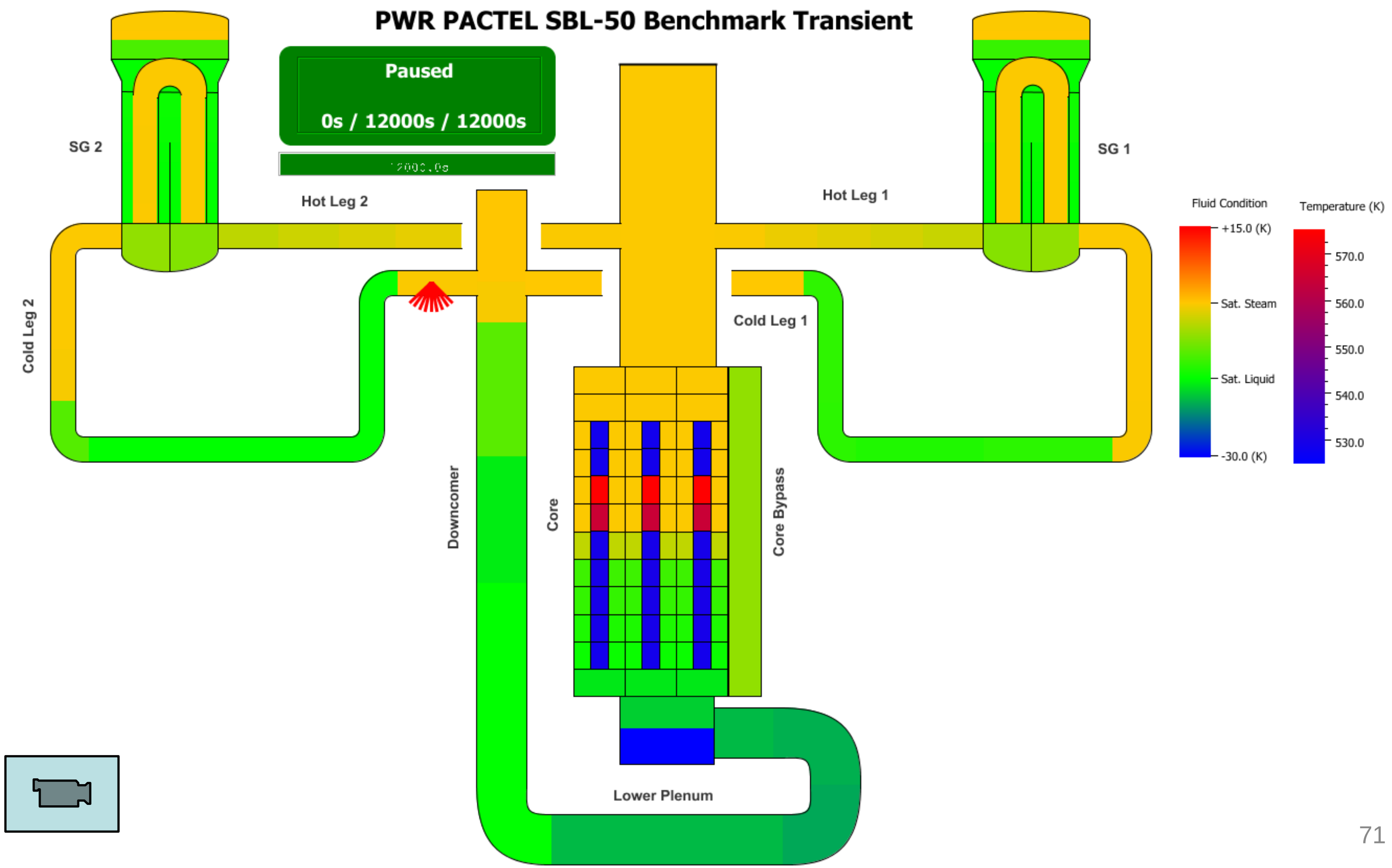
Improvement of temperatures in SG longest tubes



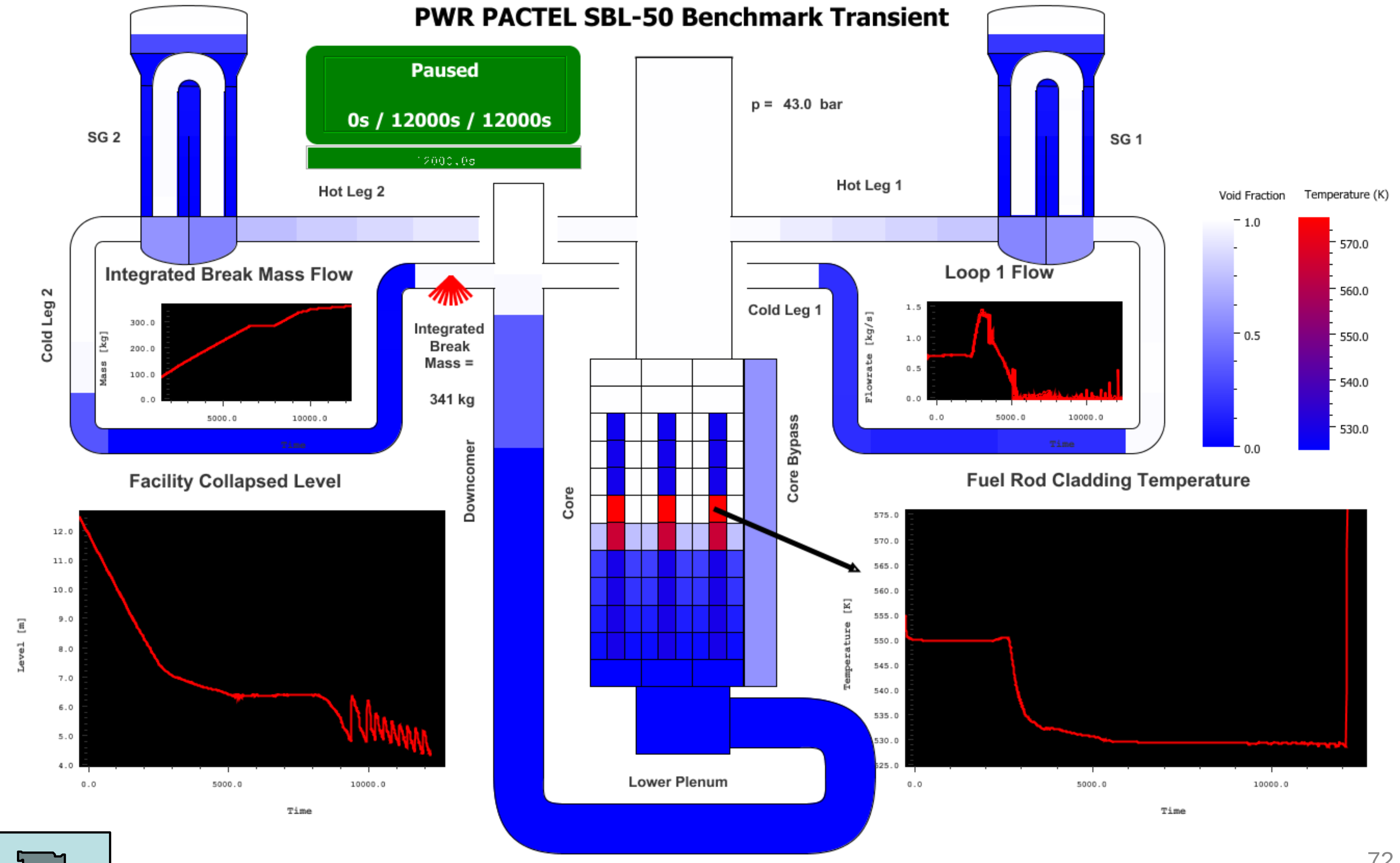




Visualization: fluid conditions by SNAP



Visualization: void distribution by SNAP



Closure

- Activities related to the TSO-DSA function have been described.
- These activities produce good and satisfactory results
  - Organization is functioning well in supporting SSM's tasks.
  - Good example on how the safety authority co-operates with the universities.
- Coupled NK/TH codes give satisfactory results and good agreements with the measured data to a large extent.
  - Coupled simulations combined with three-dimensional discretization might reveal some phenomena that are difficult to capture with stand-alone code.

THANK YOU FOR YOUR ATTENTION

Strål
säkerhets
myndigheten
Swedish Radiation Safety Authority