GERMAN (GRS) APPROCH TO ACCIDENT ANALYSIS PARRT II

VALIDATION OF COMPUTER CODES

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- **Validation of System Codes for SAR in Germany**
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  - **ATHLET**
- **Example of LBLOCA Analysis for KONVOI Plants**
- **Current Approach and Trends Towards Best-Estimate Safety Analyssis (examples of BE calculations)**

1. Genaral Considerations

Typical Code Categories for Deterministic Safety Analysis

- Simulation of Phenomena
  - reactor physics
  - fuel behaviour
  - thermal hydraulics
  - structure dynamics
  - radiological dose calculations

- Depth of Simulation
  - detailed mechanistic models
  - simplified modelling

- Scope of Simulation
  - core (square, hexagonal,....)
  - reactor coolant system
  - dry containment
  - pressure suppression systems

Name	Application	Remarks
S-RELAP5	T/H behaviour of BWR, PWR, WWER	
COCO	PWR-T/H Containment (single compartment)	coupled with S-RELAP5
WAVCO	T/H Containment (multi-compartment)	coupled with S-RELAP5
BETHY	LOCA fuel rod behaviour	
CARO	Fuel rod condition prior to occurrences, accidents	
CARLO	LOCA fuel rod behaviour, core damage	CARO+BETHY+S-RELAP5
GSUAM	Statistics for LOCA behaviour	
SAV	PWR-3D neutronics, stationary	
PANBOX	PWR-3D neutronics, transient	coupled with S-RELAP5
COBRA	PWR-3D T/H in reactor core	coupled with PANBOX
RAMONA	BWR-3D neutronics + T/H in reactor core	coupled with S-RELAP5
FRANCESCA	BWR multi-channel T/H in reactor core	
COSBWR	BWR neutronics +T/H in reactor core	coupled with S-RELAP5
HUXY	BWR LOCA fuel assembly behaviour	
HECHAN	BWR PDO fuel assembly behaviour	
HEXMED	WWER-3D neutronics, stationary	
HEXTIME	WWER-3D, neutronics, transient	coupled with S-RELAP5
KWU-MIX	PWR/WWER Mixing in RPV	PTS scenarios
KWU-DEO	PWR/WWER Mixing in RPV	Boron dilution scenarios
PHOENICS	CFD	
FLUTAN	CFD	
ASTEC	CFD	
CFX	CFD	
STAR-CD	CFD	

AVAILABLE CODES FOR PERFORMING AA IN GERMANY

2. Validation of System Codes for SAR in Germany

The Terms Verification and Validation

- **Verification**:
  The code behaves as intended (proper mathematical representation of the conceptual model, equations are correctly encoded and solved)
  Verification may include:
  Demonstration of convergence of calculated results while reducing time steps and size of nodes, comparison with exact mathematical solutions, benchmark comparison with other codes, check of plausibility

- **Validation:**
  Comparison of calculated results with measured values

"Verification" often used synonymously with validation and qualification

VALIDATION OF S-RELAP (basic information)

- Initially developed by Siemens Power Corporation (SPC) in Richland, USA for performing realistic analysis of LBLOCA for PWRs
- Based on RELAP5/MOD2 and RELAP5/MOD3 further developed from SPC and KWU
- The basic models, the code structure and the numeric solution procedures are identical with those of RELAP5/MOD2
- The code structure for S-RELAP5 was modified to be essentially the same as that for RELAP5/MOD3, with the same code portability features
- The coding for reactor kinetics, control system and trip systems are the same like in RELAP5/MOD3

- As an example, the verification matrix for RELAP5/MOD3, partly repetitions of calculations already performed before with RELAP5/MOD2, contains phenomenological problems like for instance gravitation head effect of falling liquid into a steam atmosphere, non-condensable state oscillatory flow (U-tube oscillations), and several workshop problems simulating a hypothetical two-loop PWR system

- US NRC has approved ANF-RELAP - an SPC-modified version of RELAP5/MOD2, Version 36.02 - for SBLOCA, steam line break, and non-LOCA transient licensing analyses

- The improvements and modifications included are those required to provide congruency with the unmodified literature correlations and those required to obtain adequate simulation of key LBLOCA experiments

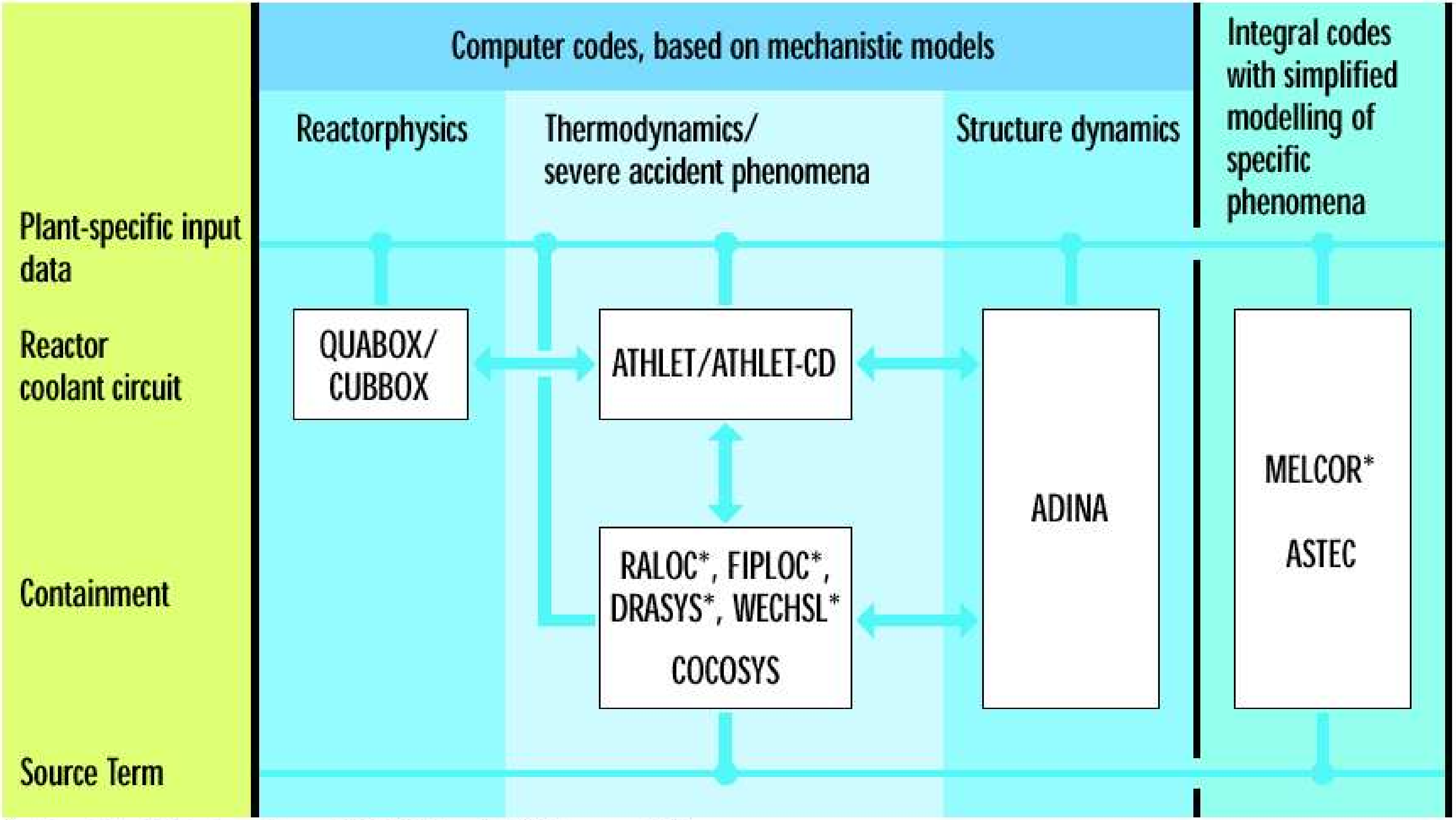
- The developmental verification problems performed for RELAP5/MOD2 and MOD3 can also be considered as applicable for S-RELAP5

List of the facilities and tests previously analysed by SPC and reviewed by the US NRC

- University of Hannover CCFL Test Facility - 8x8 bundle, steam-water tests with various tie plates,
- INEL Semiscale Facility - S-UT-8 small break test,
- INEL Semiscale Facility - S-NC-2 natural circulation test,
- INEL LOFT Nuclear Facility - L3-1 small break test,
- INEL LOFT Nuclear Facility - L6-1 loss of load test,
- INEL LOFT Nuclear Facility - L6-2 loss of flow test,
- INEL LOFT Nuclear Facility - L6-3 excessive steam load test,
- INEL LOFT Nuclear Facility - L6-5 loss of feedwater test, and
- ORNL Thermal Hydraulic Test Facility (THTF) - 3.09.10J,.10M, and 10DD level swell tests.

- The Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology to a LBLOCA (US NRC, 1989) is included as part of the assessment methodology. It provides a logical process for assessing and quantifying the uncertainties of a computer code with respect to blowdown, refill, and reflood periods of a LBLOCA
- According to CSAU, code assessment may be performed at two levels: the developmental assessment and the code assessment. Both levels were used to assess S-RELAP5.
- S-RELAP5 was exercised over a range of integral and separate effects tests in order to demonstrate that the code could predict the dominant phenomena associated with a PWR LBLOCA , for example - phenomena which are characteristically for PWR with cold leg injection.

- Specific phenomena of combined hot leg and cold leg injection mode were assessed:
  - LOFT L2-5/-6
  - CCTF Run 54
  - FLECHT-SESEAT 31504/33056
  - UPTF Test 6/7, 11
  - ORNL THTF (3.09.10)
  - CE/EPRI Pump Test
  - GE Level Swell 1004-3
  - Marviken Test 22/24
  - Bennett Tests 3358 & 5379



*only until validated versions of COCOSYS and ASTEC are available

System of Selected GRS-Codes for AA of LWR

SYSTEM CODE ATHLET

ATHLET simulates flow and heat transfer processes in the reactor coolant systems of PWR, BWR, WWER, RBMK. The scope of application comprises the whole range of incidents from normal operation, anticipated operational occurrences and transients up to accidents with local damages to fuel elements in the reactor core. BDBA in the preventive area also belong to the field of application.

ATHLET has a modular code structure segmented in accordance with the most important processes to be simulated:

- **Module TFD: thermal fluid dynamics**
- **Module HECU: heat conductivity/heat transfer**
- **Module NEUKIN: neutron kinetics (Point Kinetics, 1D and 3D Kinetics)**
- **Module GCSM: instrumentation and control**

VALIDATION OF ATHLET

- The validation of single physical models and the entire computer code ATHLET is performed systematically by pre- and post-calculations of experiments on reactor safety as well as by confirmatory calculations of transients in reactor plants.

- On the basis of the international OECD/NEA validations matrix, an overall validation matrix was derived for ATHLET extended by the experiments which are specific for reactors of German design.

- The ATHLET overall validation matrix contains:
  - 93 single-effects experiments (PWR, BWR)
  - 101 integral experiments (PWR, BWR)
  - 24 integral experiments for VVER-reactors

		Pressurised water reactors (experiments for WWER not included)					Boiling water reactors	
Facility or plant	Scale	Large breaks	Small, inter-mediate breaks	Transients	Shut-down transients	AM	LOCA's	Transients
UPTF/ TRAM	1:1	6	2			3		
CCTF	1:25	4						
LOFT	1:50	2	4	1				
LSTF	1:50		2					
BETHSY	1:100		7		3	5		
PKL	1:145	2	9	6	1	6		
ROSA-III	1:424						5	1
FIST	1:642						2	1
LOBI	1:712	2	7	4		2		
GERDA	1:1686		1					
Konvoi	NPP			3				
Brokdorf	NPP			4				
Gundremmingen	NPP							3
Krümmel	NPP							3
		16	**32**	**18**	**4**	**16**	**7**	**8**

ATHLET Validation Matrix, Survey on Integral Experiments for PWR and BWR

		WWER reactors				
Facility or plant	Scale	Large Breaks	Small breaks, intermediate breaks	Transients	Shutdown transients	AM
PACTEL	1:305		9	2		2
PMK-2	1:2070		7		2	
ISB-WWER	1:3000		2			
			18	**2**	**2**	**2**

ATHLET Validation Matrix, Survey on Integral Experiments for WWER

Phenomena 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						
	Blowdown	Refill	Reflood	CCTF 1:25	LOFT 1 : 50	BETHSY 1 : 100	PKL 1 : 145	LOBI 1 : 712	SEMISCALE 1: 1600	UI : 1
Break flow	+	+	+	o	o	o	o	o	o	o
Phase separation (condition or transition)	o	+	+	+	+	+	+	+	+	+
Mixing and condensation during injection	o	+	+	o	o	o	o	o	o	+
Core wide void + flow distribution	o	+	+	o	o	o	o	o	-	o
ECC bypass and penetration	o	+	o	o	+	-	o	o	-	+
CCFL (UCSP)	o	+	+	o	o	o	o	o	-	+
Steam binding (liquid carry over, etc.)	-	o	+	o	o	-	o	o	o	o
Pool formation in UP	-	+	+	o	o	o	o	o	o	+
Core heat transfer incl. DNB, dry-out, RNB	+	+	+	+	+	+	+	o	o	-
Quench front propagation	o	o	+	+	+	+	+	-	+	-
Entrainment (Core, UP)	o	o	+	o	o	o	o	o	o	+
De-entrainment (Core, UP)	o	o	+	o	o	o	o	o	o	+
1 - and 2-phase pump behaviour	+	o	o	-	o	-	o	+	+	-
Non-condensable gas effects	-	o	o	-	+	+	-	-	-	+

Cross Reference Matrix PWR LBLOCA with Selected Test for ATHLET Validation

Test Facility	CCTF	-	o	+
	LOFT	+	+	+
	BETHSY	-	-	+
	PKL	o	+	+
	LOBI	+	+	-
	SEMISCALE	+	+	+
	UPTF	o	+	+

Important test parameter
- break location / break size
- pumps off / pumps on
- cold leg injection / combined injection

(a) UPTF integral tests

Facility	Run	Description
UPTF	2	2A-CLB,CI,LR
	27B	2A-CLB,CI,BER
	18	2A-CLB,CHCI,LR
	28	2A-CLB,CHCI,BER
	19	0.5A-CLB,CHCI,LR
	24	2A-CLB,DCI,VVT
CCTF	C2-19/79	2A-CLB,CHCI,LR
	C2-20/80	2A-CLB,CHCI,BER
	C2-04/62	2A-CLB,CI,BCT
	C2-12/71	2A-CLB,CI,BER

Facility	Run	Description
LOFT	L2-5	2A-CLB,TBO
LOBI	A1-06	2A-CLB,CHCI
	A1-66	2A-CLB,CI
PKL II	B2	2A-CLB,CHCI
	B3	2A-HLB,HI
	B5	2A-CLB,CI
BETHSY	6.7b	2A-CLB,CI,NR

BCT	Base Case Test
BER	Best Estimate Rates
CHCI	Combined Injection
CI	Cold Leg Injection
CLB	Cold Leg Break
DCI	Downcomer and CI
HI	Hot Leg Injection
HLB	Hot Leg Break
LR	Licensing Rates
NR	Nitrogen Release
TBO	Total Black-out
VVT	Vent Valve Tests

Continuation of Cross Reference Matrix

The ATHLET validation matrix covers a broad spectrum of incident scenarios and concerns numerous single-effects experiments and integral experiments on:

- large, medium and small breaks in PWR with U-tube SGs, PWR with once-through SGs, VVER-PWR with horizontal SGs,
- transients in PWR with U-tube SGs, PWR with straight-tube SGs, in VVER-PWR with horizontal SGs,
- LOCAs and transients in BWR,
- Accident Management Measures.

In this respect, the **international standard problems (ISP)** represent particularly important contributions. The ISP related to LOCAs mostly concerns the simulation of event sequences in reactor plants with cold-leg emergency core cooling injection.

The **transferability** of findings from a scaled down test facility to a reactor plant is of special importance for the validation. It is essential for models validated on the basis of such test results that they are checked selectively according to data from a test facility corresponding with the geometry of a reactor. The experiments of the **UPTF- and UPTF/TRAM-programme** offer outstanding possibilities for it. Due to this special status, many of the UPTF experiments and nearly all experiments of the UPTF/TRAM series A and B have been **incorporated in the ATHLET validation matrix.**

Further validation matrices:

- **Six integral experiments (covers all essential phenomena)**

CCTF C2-20	200% break in cold leg, combined ECC injection, PWR
LSTF-SB-CL-18	5% break in cold leg, cold leg ECC injection, PWR
LOBI A2-90	Loss of offsite power, ATWS, PWR
GERDA 160702	20cm² pump seal break, once - through steam generator, PWR
FIST GMSBI	steam line break, BWR
ROSA III-916	50% break in recirculation line, BWR

Up to now, post-calculations have been performed for **60 integral** and **75 single-effects** experiments within the scope of the validation.

There are **several institutions**, independent of GRS, involved in the validation process.

3. Example of LBLOCA Analysis for KONVOI Plants

- Performed in the mid eighties
- Analyses are compiled in the ECC handbook (part of SAR)
- SAR is prepared by the vendor for the utility (applicant) and was presented to the Minister in charge of nuclear safety in that federal state on the territory of which the NPP was foreseen to be build
- Analyses at that time were performed with CEM-codes (**conservative evaluation models**) by the vendor

Accident Phase	Vendor	Assessor	
Blowdown Phase	LECK 4 /Mod 2	DRUFAN	TRAC-PF 1
Refill and Reflood Phase	HYDRANS	FLUT	TRAC-PF 1
Hot Rod Analysis and Core Damage	BETHY-AZ	TESPA	TESPA

Use of Codes in KONVOI Plant Licensing

Case	Vendor	Assessor
DEGB in Cold Leg	X	X
DEGB in Crossover Leg	X	
DEGB in Hot Leg	X	
1 A in Cold Leg	X	
0.5 A in Cold Leg	X	
0.25 A in Cold Leg	X	

LBLOCA Cases analysed in KONVOI Plant Licensing

- Conservative initial and boundary conditions were used as prescribed in a general way in the safety criteria and more precisely in the RSK guidelines .The RSK guidelines had been established in order to ease the process of assessment within the RSK. However, the guidelines allowed a certain degree of **flexibility to provide sufficient freedom to take into account the continuous development of safety technology**. If the applicant was not willing or not able to fulfill particular requirements, he had to **demonstrate** that alternative measures and solutions assure safety in at least an equivalent way. If reasons of conservatism of model assumptions were obsolete, the model could be replaced by a more realistic assumption, provided the applicant could prove it by appropriate experimental verification.

- The analysis of DEGB was requested in order to demonstrate the EC efficiency and the strength of the containment. Concerning load assumptions for the calculation of reaction and jet forces on pipes and components, and consequences on reactor pressure vessel internals, however, **a break cross section of 0.1 A had to be postulated** (because of the application of the break preclusion concept). **All analyses of break sizes larger than 0.1 A were performed without considering reactor scram**. It was demonstrated for break sizes larger than 0.1 A that reactor scram is not needed to shut down the fission power. Also for the long term reactivity balance the effect of the scram system was not considered.

Conservative model assumptions for the LBLOCA analyses

- used for critical flow rate, burnout time, film boiling heat transfer, heat transfer during refill and reflood, residual water content, flow rate reduction for hot rod temperature calculation, and others.

- **Typical conservative initial conditions** were the increased core power and the increased primary coolant temperature, as well as the reduced pressure in the primary system. Such data were assumed at the boundary of bands of the reactor limitation system, considering also the uncertainty of measurements.

- **Essential conservative boundary conditions**
  - the loss of normal onsite and offsite power (LONOP) simultaneously with the fast break initiation (breaking time of 15 ms),
  - reduced containment back pressure,
  - reactor core power distribution,
  - decay heat power,
  - failure of redundancies of the ECC system due to single failure and maintenance. These failures were applied on those components most favourable for core cooling regardless whether they are active or passive.

- No credit for core cooling was taken from emergency core coolant injected in the vicinity of the break.

The technical limit (acceptance criteria) values for the LBLOCA analyses were:

- Calculated peak clad temperature below 1200°C
- Calculated local cladding oxidation depth below 17 % of the cladding thickness
- Zirconium-water reaction less than 1 % of the total cladding material
- No changes of core geometry which would prevent sufficient core cooling
- Fission product releases due to cladding tube defects below specific values: 10 % of noble gases, 3 % of halogens, 2 % of volatile solids (Cs, Te), and 0.1 % of other solids. It was to be assumed that 10 % of all cladding will fail unless a core damage analysis would result in a lower value.

A core damage analysis was requested by the licensing authorities in order to demonstrate a core damage of less than 10 %.

In addition, requirements on long-term cooling have to be fulfilled with regard to maintaining subcriticality and to avoiding long-term steam release to the containment.

The assessors confirmed the fulfillment of the acceptance criteria including the 10 % core damage.

4. Current Approach and Trends Towards Best-Estimate Safety Analyssis

Best estimate computer codes are used to calculate postulated loss of coolant and transient accidents in a **realistic and not in a conservative way**. There is an increasing interest in computational reactor safety analysis to **replace the conservative evaluation model calculations by best estimate calculations supplemented by a quantitative uncertainty analysis.**

The thermal-hydraulic system codes presently available in Germany for licensing and related purposes are more realistic than at the time of licensing the KONVOI plants.

Prerequisites of BE Analysis

- The codes and models reflect the best available knowledge (exceptions have to be treated still conservatively)
- The code user have high qualification
- The uncertainties are evaluated
- Standards and Guidelines allowed the flexibility to follow the advances in safety technology and transfer reliable R&D results into code models and assumptions

  Deterministic AA was performed under conservative and Best-estimate conditions in all licensing processes

  The RSK guidelines reflected the priority of the LBLOCA as the limiting DBA and did not specifically consider all important phenomena which determine the accident sequence of SBLOCA

Status of Present Analysis Tools

- Available TH codes are able to calculate more realistically than at the time of the last licensing of new NNPs.

- Existing models were **improved**, new models were developed, e.g.
  • six-equation two-fluid model • direct condensation models • models for non-condensable gases • boron tracking models • extended two-phase flow models for special components, e.g. T-junctions, complex balance-of-plant models.

These models reflect the best available knowledge.

- **Coupling of TH codes with 3D reactor physics** codes and containment codes

- Improved robustness of codes, extented accuracy checks

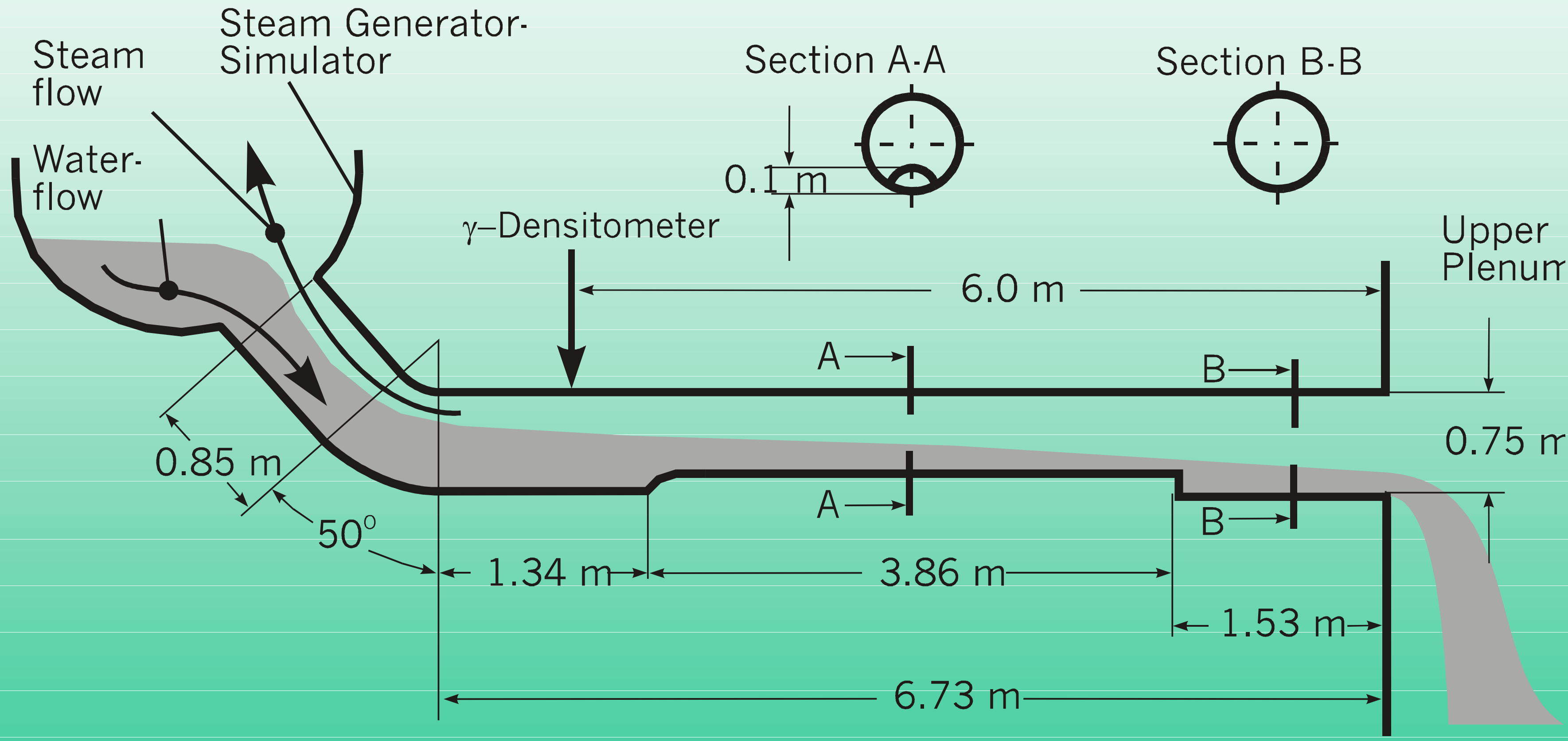
- Some very complex phenomena can still not be described in a complete realistic manner → conservative assumptions

AN EXAMPLE FOR THE DEVELOPMENT TOWARDS BEST ESTIMATE APPROACH

- The analytical description of the drift velocity is not directly related to one of the assumptions recommended in the RSK guidelines but it concerns the general development of the thermal-hydraulic system codes used for the SAR. Until 1984, a conventional drift-flux model was used in the codes for all essential flow regimes in **vertical geometry**, **supplemented by correlations for the counter-current flow limitation**. **For horizontal flow channels, only simplified void-correlations were available**. Validation was performed by means of scaled experiments, e.g. from the test facilities LOBI and LOFT. In order to meet these experiments, **appropriate model approaches with regard to the counter-current flow limitation were made** for the reactor calculations for the KONVOI plants during the licensing process. To compensate the scaling uncertainty a penalizing charge was made, e.g. with the objective to hinder the water downflow from the steam generators to the upper plenum by the steam flow from the upper plenum.

- In 1984 experimental results from the full scale UPTF test series had been obtained. New correlations related to the geometry for the horizontal tube which also includes the inclined tube section at the steam generator inlet were developed and validated against a variety of experiments including the UPTF test 11.

Counter Current Flow in Horizontal Flow Channels



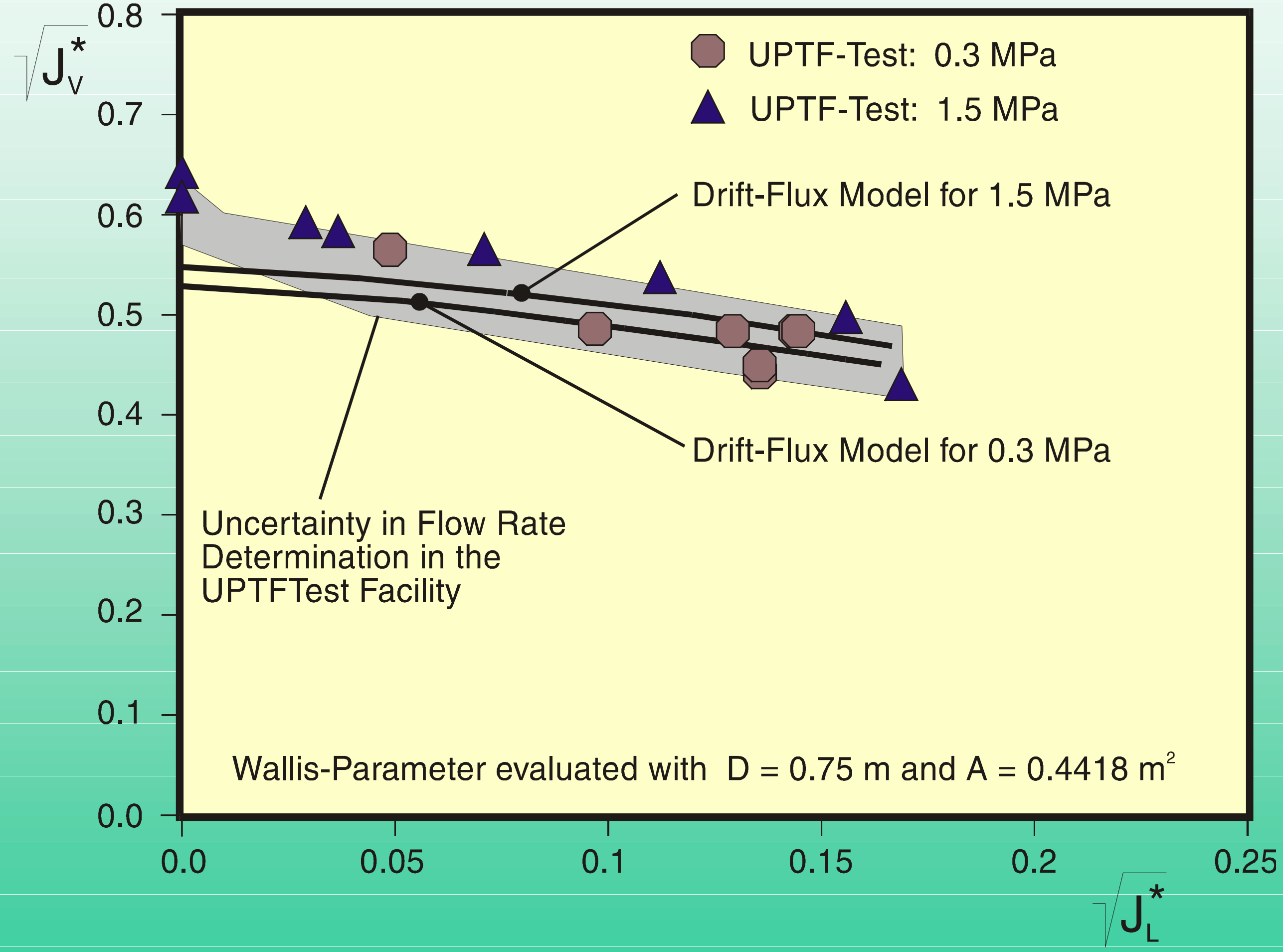
Schematic Representation of the Hot Leg of UPTF

- Saturated water was supplied in the experiment from the steam-generator side and, simultaneously, saturated steam from the vessel side, i. e. from the upper plenum. The steam mass flow rate was increased gradually until occurrence of the counter-current flow limitation. The **specific volumetric flow rates** in terms of Wallis parameters for steam and water for the counter-current flow limitation condition are measured and after that compared with ATHLET calculations.

- The good correspondence for the horizontal tube with original diameter demonstrates that the velocities of **the two phases including the counter-current flow limitation are determined realistically by the correlation**. With the new correlation, further experiments were analyzed successfully, e. g. the depletion of the steam generator U-pipes during a small-leak experiment in the LOBI test facility. After completion of the model validation calculations, the new correlation was incorporated in the ATHLET code.

- Based on the same findings the vendor made the similar improvements in his version of RELAP5

Counter Current Flow in Horizontal Flow channels

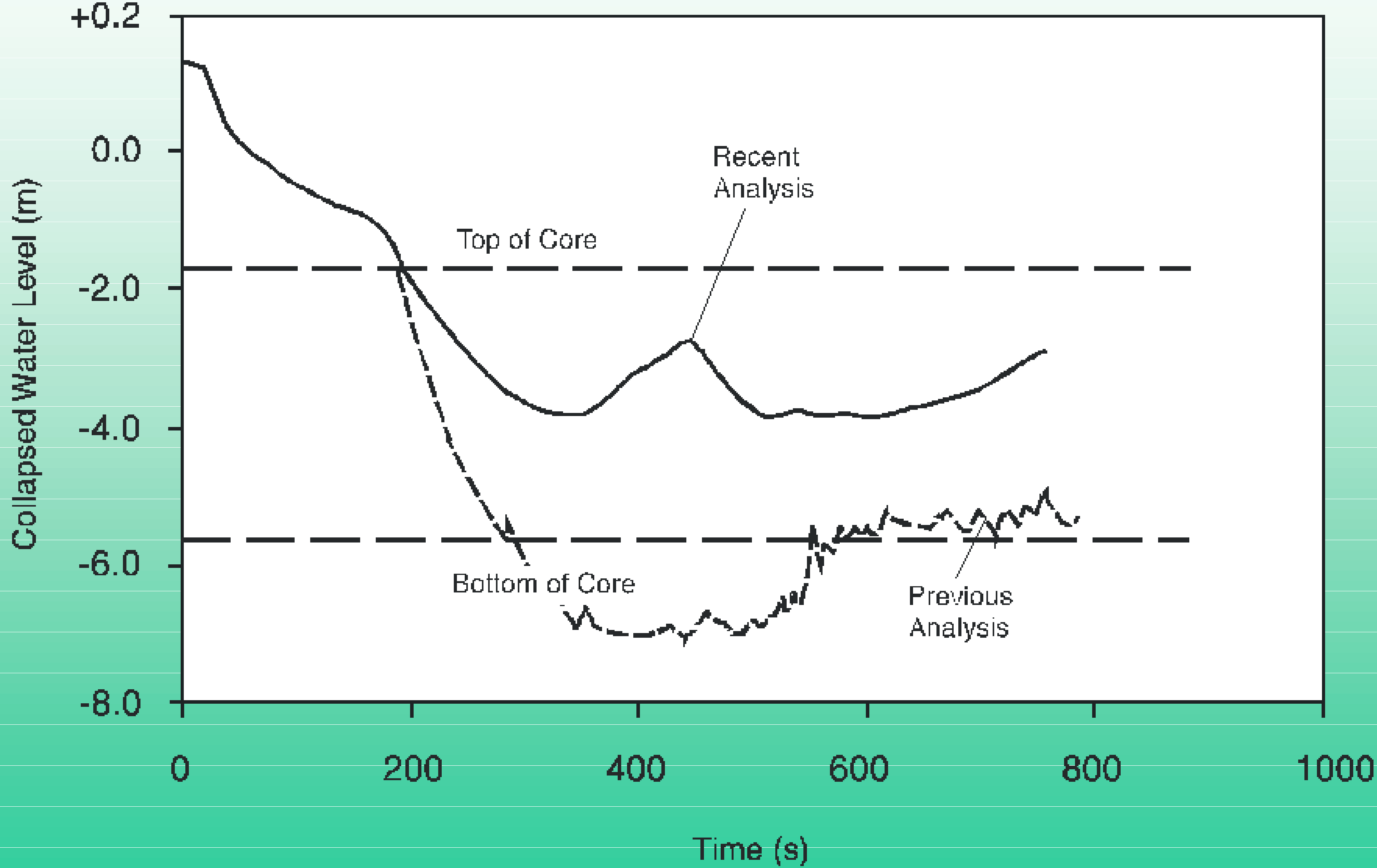


UPTF 11 test results and Drift-flux model of ATHLET

- The model improvements were implemented in the codes after the performance of the licensing calculations for the KONVOI plants. **In order to find out the degree of conservatism of the licensing analyses, some of them were repeated with the improved ATHLET version.** A medium leak size of 160 cm² in the cold leg of the KONVOI plants was selected by the assessor for demonstration.

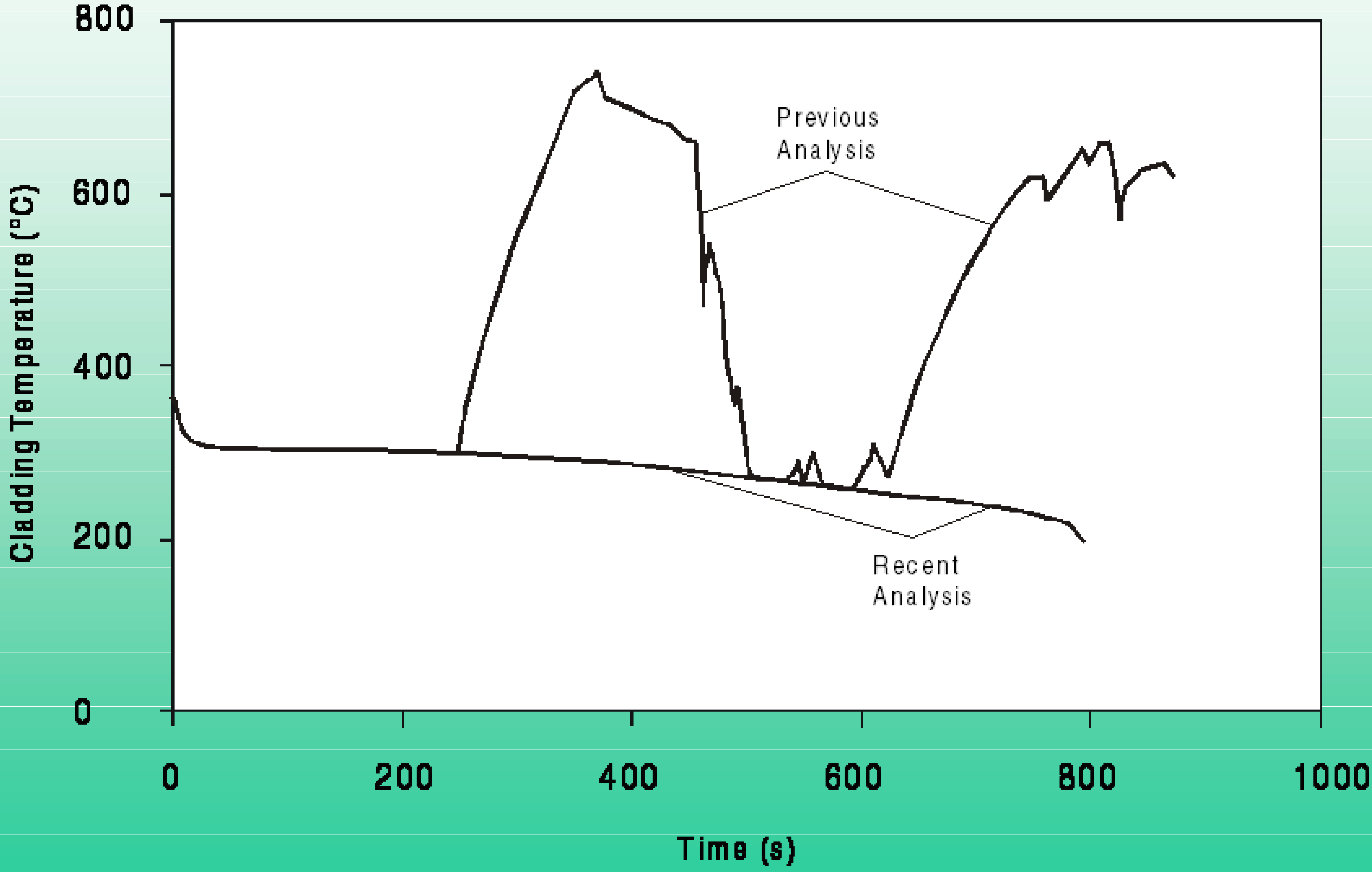
- The time sequence shown in the figures is restricted to the time period between the incident occurrence and the beginning of the injection by the pressure accumulators. Their high injection rate terminated the depletion of the core already in the previous calculations. In the recent calculation, however, the core remains continuously covered with a two-phase mixture even during the high-pressure injection phase. As a consequence, the cladding temperatures do not rise significantly above the saturation temperature of the coolant. The time period during which the coolant is retained in the steam generators tubes and plenums at inlet and exit was reduced compared to the former calculation. **With realistic simulation of the counter-current flow limitation the coolant entered the RPV earlier and caused an increase of the core water level. An uncovery of the core did not occur anymore in the recent analysis.**

Counter Current Flow in Horizontal Flow Channels



Application on SBLOCA (160 cm² in CL of PWR 1300), Core Level

Counter Current Flow in Horizontal Flow Channels



Application on SBLOCA (160 cm² in CL of PWR 1300), PCT in Core

- With the new correlation in the ATHLET code further analyses were performed by one TÜVs in 1992 covering an intermediate break spectrum between 50 cm^2 and 250 cm^2 during which core uncovery did not occur either. This was found also in 1991/92 when the vendor repeated the analyses for the same leak spectrum with the code RELAP. In summary, it was proven that in the SBLOCA analyses, performed independently by the applicant and by the assessors in the frame of the licensing procedure for the KONVOI plants, the core heat-up had been overestimated due to a too strong coupling of the phases in the hot leg.

Examples of Deviations from Recommendations in RSK Guidelines

Recommendations from chapter 22.1.3 of RSK guidelines	Assessors assumptions in the latest licensing calculations (KONVOI) and trends thereafter
Bernoulli/Moody model (sub-cooled/two-phase) for the blow-down mass flow rate if experimentally confirmed correlations are not available	Discharge models validated against a variety of experiments, e.g. Super-Moby-Dick, Marviken. It is not sure that the use of so-called conservative models really leads to conservative results in view of core cooling, therefore parametric variations are very useful, e.g. variation of CD-factor
Estimation of conservative burnout time for the blowdown and/or hot-rod calculations	Multi-channel core representations with cross connections (mainly validated against LOFT, CCTF, SCTF) with consideration of fuel rods with different power and heat flux profile replaced artificial hot-rod calculations. (realistic 3D distributions).For the realistic determination of critical heat flux for the hot channel the Groeneveld “look-up” table is replacing conservative assumptions.
Modified Dougal-Rohsenow heat transfer correlation during blowdown and prior to the drying of the reactor core if experimentally confirmed correlations are not available. The same recommendation also for the heat transfer during quenching.	Since experimentally confirmed realistic models are not available for the complete range of parameters the Groeneveld 5.9 correlation was used as a conservative model (experiments: RS 37 bundle tests), the recommended Dougal-Rohsenow correlation was not confirmed as a conservative correlation
Adiabatic core heat-up prior to flooding or experimental HTC figures	Experimentally verified correlations (selected by quality and temperature difference)

Recommendations from chapter 22.1.3 of RSK guidelines	Assessors assumptions in the latest licensing calculations (KONVOI) and trends thereafter
ECC water supplied directly to the break shall not be taken into consideration for core cooling purposes	In unambiguous cases, e.g. the rupture of the injection line, this recommendation is indisputable, in other cases, e.g. leaks in the neighbourhood of the point of injection, the interactions between sub-cooled water and vapour may influence the distribution of coolant in the entire circuit. Therefore performance of parametric variations was conducted. Today the advanced codes are able to calculate these interactions satisfactorily.
No residual water left in the RPV after the depressurisation phase of LBLOCA	Numerous experiments, e.g. UPTF, have shown that there is overlapping between the phases of depressurisation, refill, and reflood and that residual water exists at the end of depressurisation, the codes can predict now a realistic value of the residual water
Drift correlations: not specifically mentioned in the RSK guidelines	Simple conservative correlations applied during the licensing process for KONVOI plants, later development of realistic correlations after UPTF experimental results became available (see chapter 3 of this report)

Essential Differences between “best-estimate” and “conservative” IBC for LBLOCA

	“best estimate“ IBC	“conservative“ IBC
Break opening time	15 ms	
Reactor power	100 %	106 %
Decay heat	DIN simplified	DIN simplified + 2 σ
Power shape (16x16)	3rd cycle, mid	1st cycle, 4 d
Hot spot factor	2.57	
Gap HTC	axially varying	axially uniform
- average rod	depending on local power	5500 W/m²/K
- hot rod	about 17000 W/m²/K	7500 W/m²/K
Containment pressure	consideration of pressure reducing effects	

Coupling of TH codes with 3D reactor physics codes (examples of BE calculations)

- OECD/NRC Main Steam Line Break Benchmark – code-to-code comparison
- OECD/NRC BWR Turbine Trip Benchmark – comparison with measurements

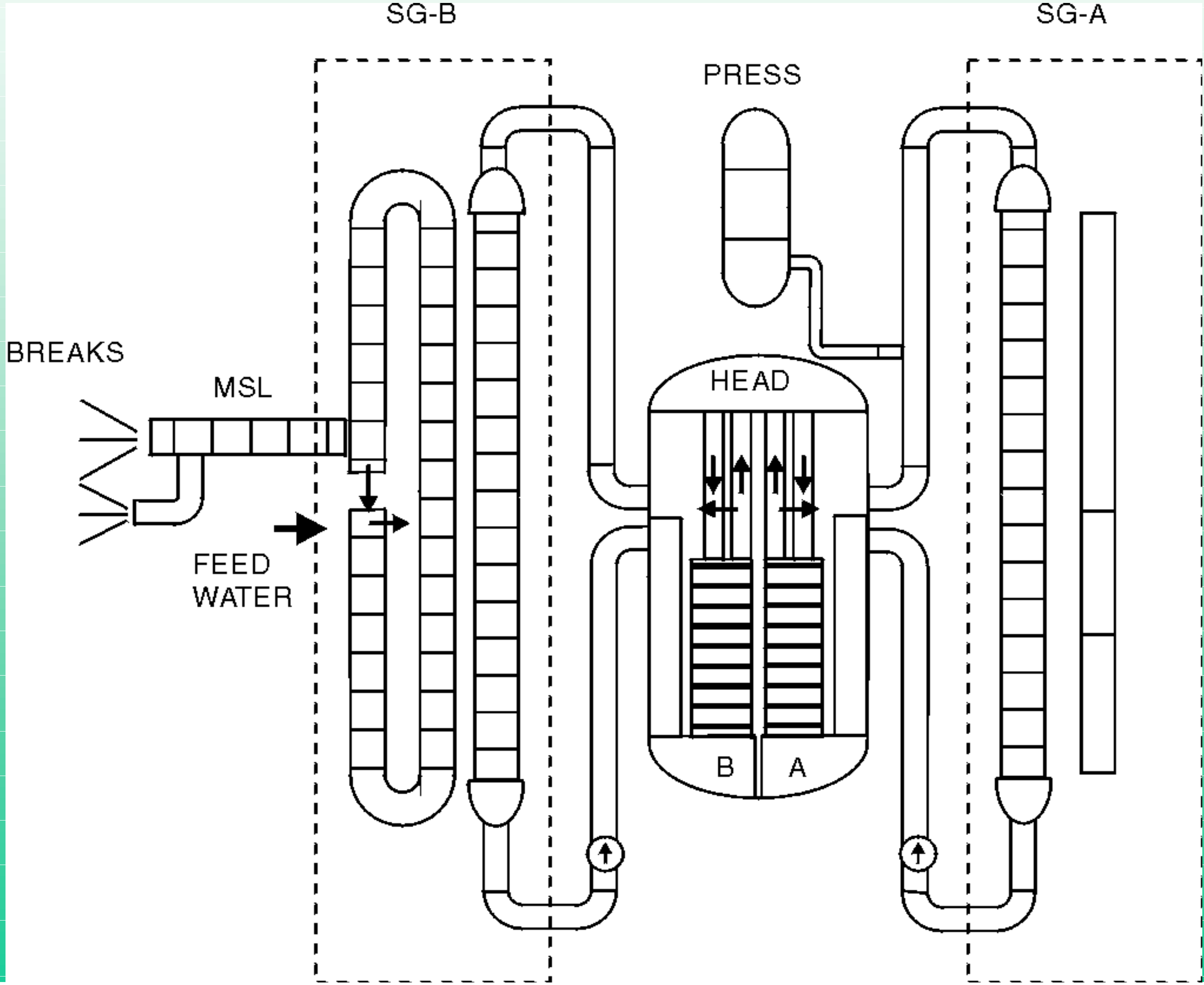
ANALYSIS OF THE PWR MAIN STEAM LINE BREAK (MSLB) BENCHMARK BY THE COUPLED CODE SYSTEM ATHLET - QUABOX/CUBBOX

THREE EXERCISES for the MSLB BENCHMARK

AIMS OF THE BENCHMARK EXERCISES :

- COUPLED CODE TO CODE COMPARISON (a part of the code validation procedure)
- PERFORMANCE OF REALISTIC ANALYSIS
- COMPARISON BETWEEN POINT-KINETICS AND 3D NEUTRONICS RESULTS

PLANT CONFIGURATION



SCENARIO AND SYSTEMS AVAILABILITY FOR THE PWR MSLB TRANSIENT

- Hot full power
- Double-ended rupture of MSL upstream of the isolation valve

  (opening at 0.1s, 24 inch and 8 inch)
- Initiation of rector trip at 114% of nominal power
- Stuck rod of the most effective rod (N12)
- Closing of turbine steam valves
- Recirculation pumps in operation
- High pressure injection pumps in operation
- Feedwater given as time-function

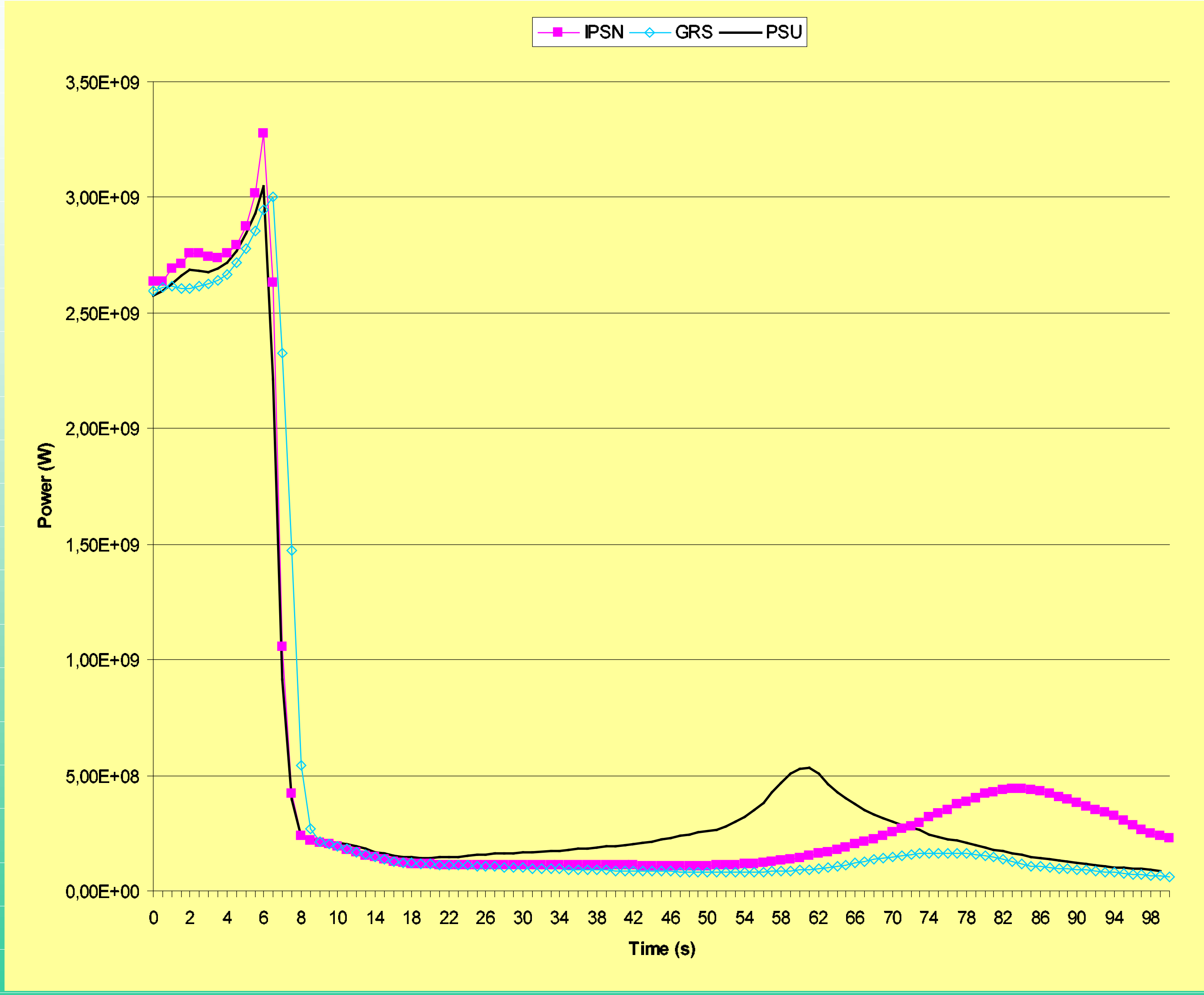
MAIN PHYSICAL PHENOMENA:

- The break causes a pressure decrease in the SG secondary side. This is affected by **break mass flow** (dependent on the ratio of water and steam discharge) and the **feedwater supply**

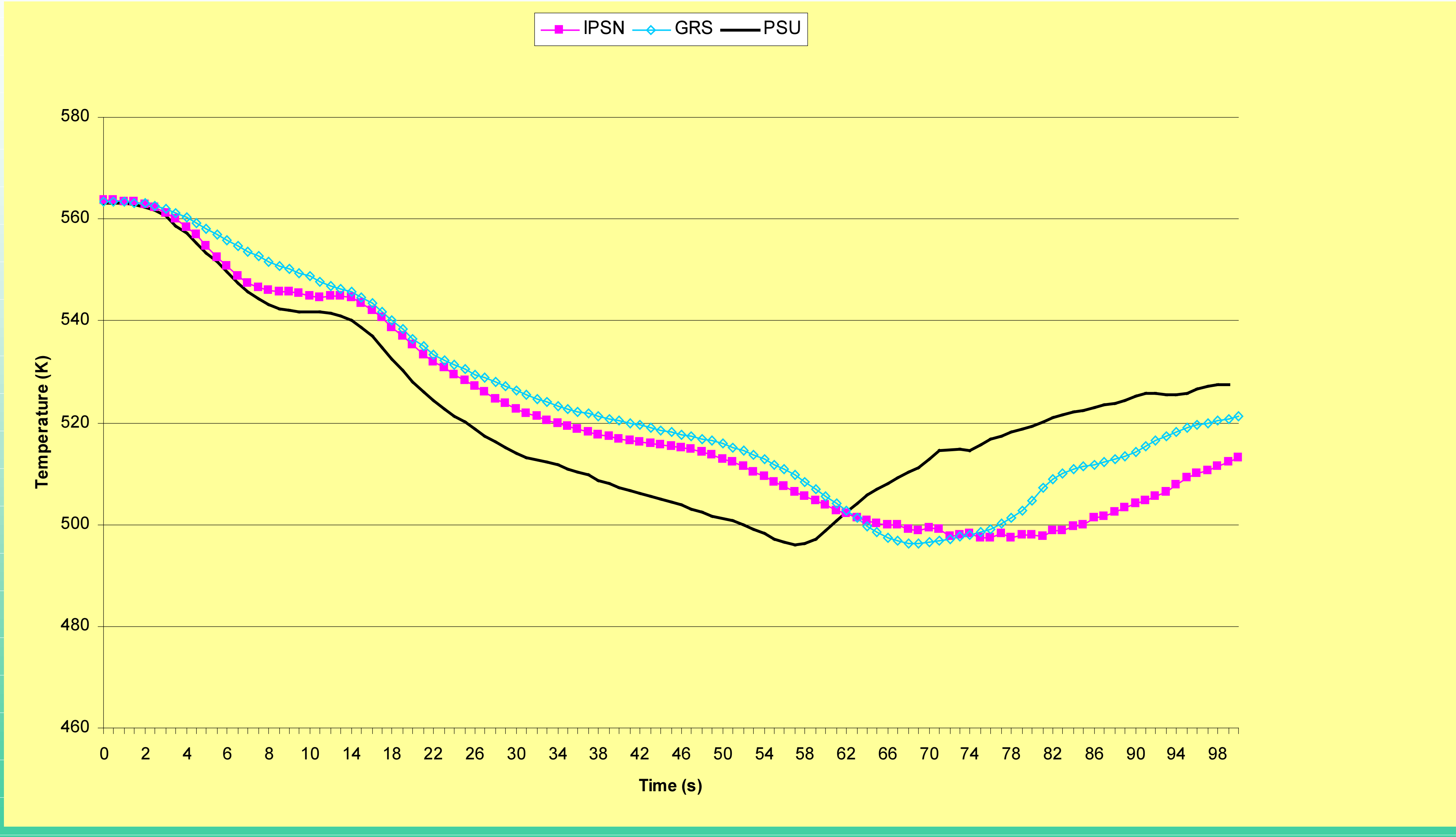
- The pressure in the secondary side determines the **saturation temperature** of the coolant in the SG and consequently the heat removal from the primary side. An efficient heat-transfer exist as long as the SG contains sufficient liquid. Therefor, the **temperature in the cold leg follows directly the decrease of the saturation temperature on the secondary side**.

- The cool-down of primary circuit is terminated when the steam generator falls dry

- **The coolant temperature** in the reactor core, which determines the criticality conditions after reactor trip, is dependent on the **mixing** phenomena between the coolant flows from the affected and intact primary loop as well as the mixing in the core region



REACTOR FISSION POWER (point kinetics)



BROKEN COLD LEG TEMPERATURE