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SCWR Safety

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Objectives

- Safety requirements
- Safety systems design
- Safety analysis tools and challenges
- Sample safety analysis results

Safety of SCWRs

- The basis of the following viewgraphs are results of the EU funded project High Performance Light Water Reactor Phase 2 (HPLWR2) which was performed from 2006 until 2010 (Ref. 1,2)
- In this lecture, HPLWR is taken as an example, where necessary, to discuss safety of SCWR´s

- Safety requirements

Basic requirements for the safety concept

- Since HPLWR is a long term development the requirements which have to be applied in the future can not easily be foreseen for the time of implementation
- General guide are the requirements from the Gen IV initiative
- Additionally the European Utility Requirements (EUR) are taken into account since more practical for design purposes

Generation IV technology goals in the safety and reliability area

- Safety and Reliability –1. Generation IV nuclear energy systems operations will excel in safety and reliability.
- Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
- Safety and Reliability–3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response

Summary of essential safety goals

- Promote the highest levels of safety and reliability by adopting established principles and best practices developed by the industry and regulators
- Probabilistic Risk Assessment (PRA) identifies and helps prevent accident sequences that could result in core damage and off-site radiation releases and reduces the uncertainties associated with them
- Passive features to provide cooling of the fuel and reducing the need for uninterrupted electrical power
- Evaluation of passive safety should be continued and passive safety features incorporated into Gen IV nuclear energy systems whenever appropriate
- For Gen IV systems a design effort focused on elimination of the need for offsite emergency response is warranted

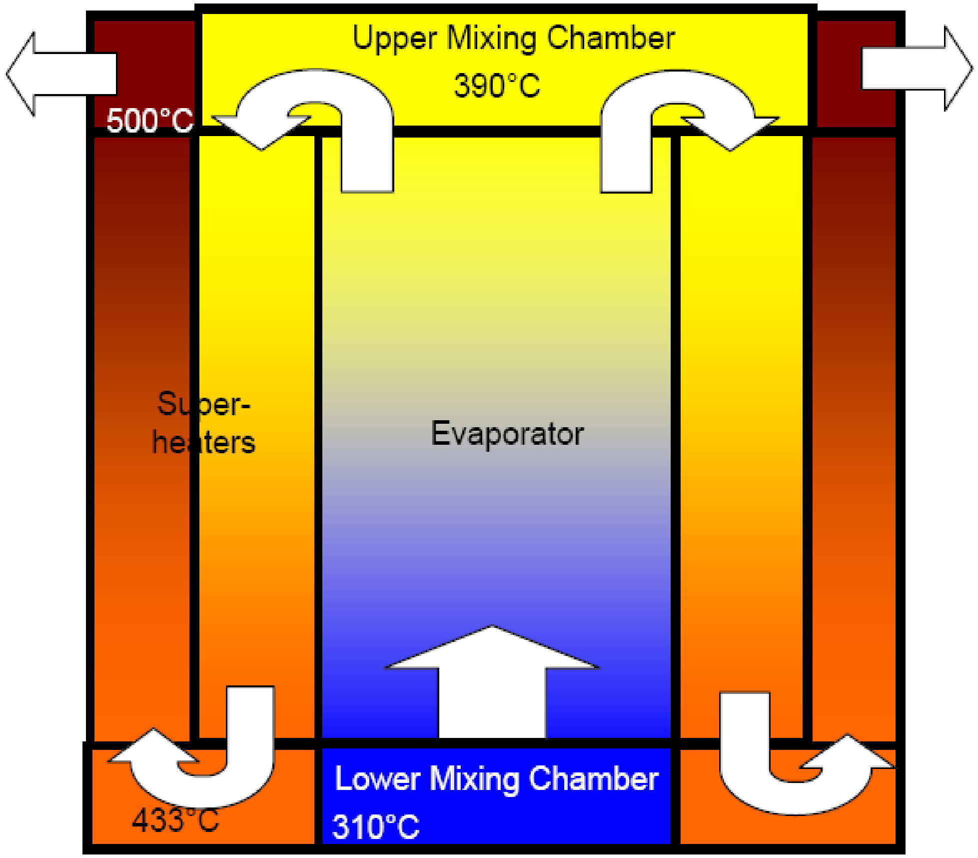
HPLWR Defense-in-depth, design basis conditions

- Normal operation (DBC1) — Operating systems
  - Conservative design, reliability, availability
  - Proven technology, quality assurance
- Operational occurrences (DBC2, $>10^{-2}$/year) — Control, limitation features
  - Surveillance, diagnostics
  - Inherent safety, nuclear stability
- Design basis accidents (DBC3/4, $<10^{-5}$/year) — Safety systems
  - Redundancy, train separation
  - Protection against internal and external hazards
  - Qualification against accident conditions
  - Automatic actions within the first 30 min after accident begin
  - Autarchy

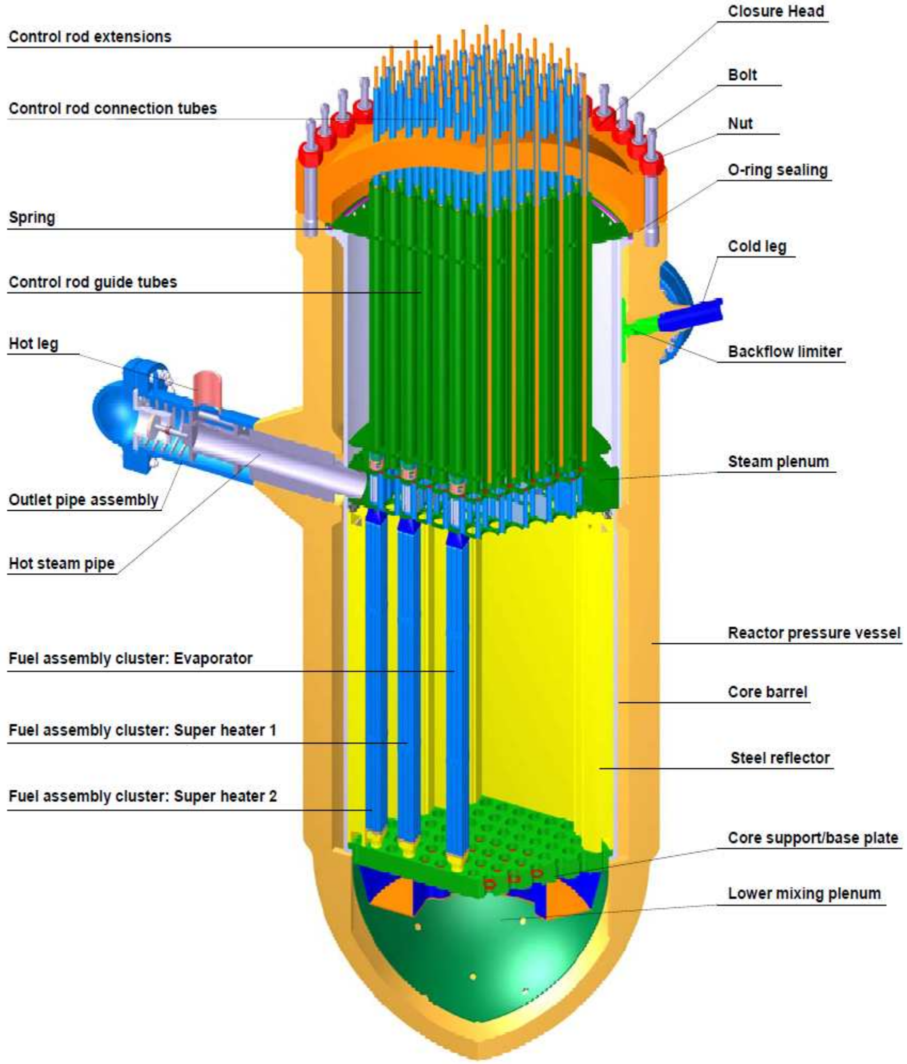
HPLWR Defense-in-depth, design extension conditions

- Multiple failure scenarios (e.g. station blackout, total loss of feedwater, SB-LOCA), severe external events (e.g. military or large commercial airplane crash)
  - Diversified systems
  - Design against external event loads
- Severe accidents
  - Mitigative features
  - Prevention of energetic consequences which could lead to large early containment failure (e.g. steam explosion, direct containment heating, global H_2 detonation)

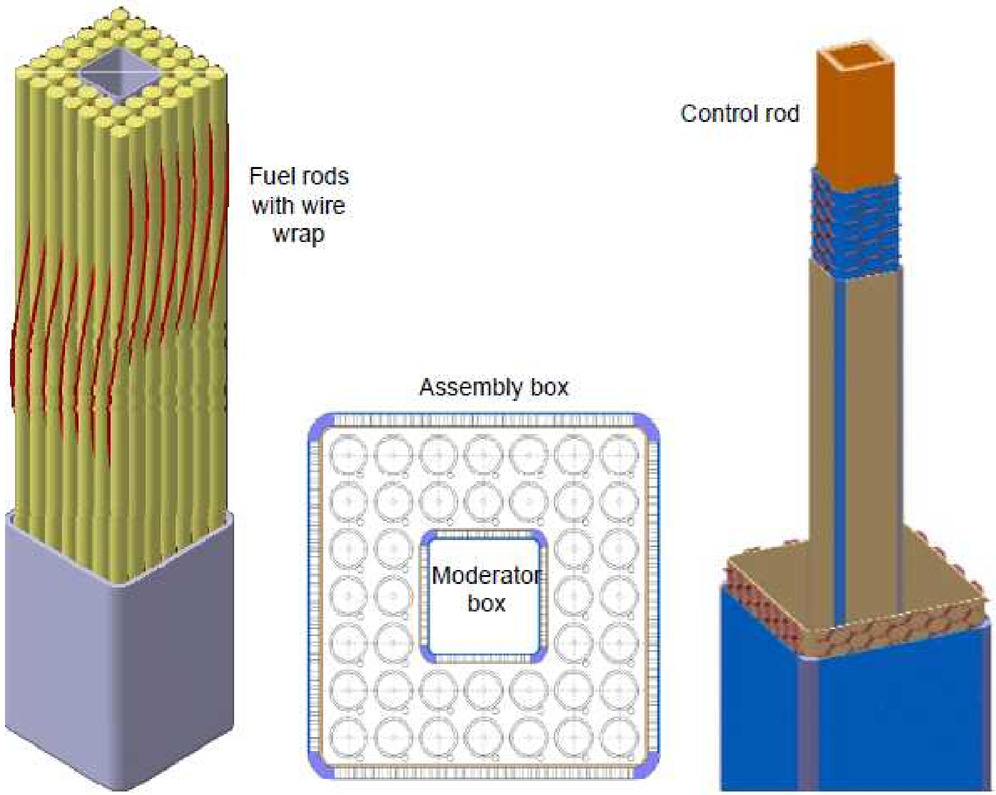
Three pass core concept of HPLWR



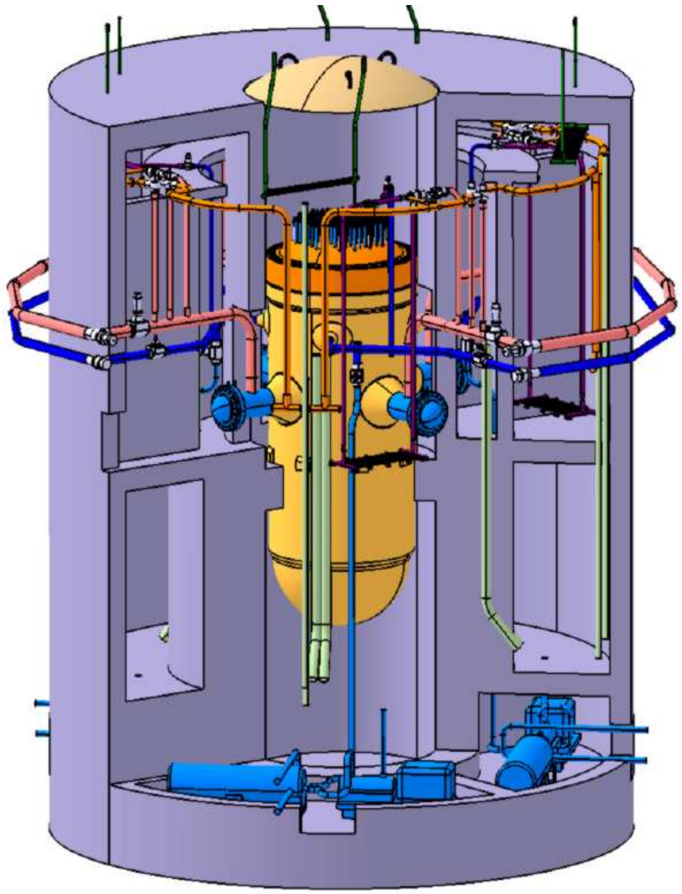
HPLWR assembly of the RPV for the three pass core concept



HPLWR fuel assembly design



Containment overview



HPLWR essential data

Reactor thermal power	2300 MW
Net electrical output	1000 MW
Plant efficiency	43,5%
Pressure	25 MPa
Vessel inlet temperature	280°C
Vessel outlet temperature	500°C
Total flow rate	1179 kg/s
RPV inner diameter	ca. 4500 mm
Wall thickness cylindrical shell	ca 450 mm
Vessel material	20Mn Mo Ni 55
Number of fuel assemblies	1404
Active length	4200 mm
Enrichment	Up to 9%
Cladding outer diameter	8 mm
Cladding material	1.4970 SS
Max. discharge burnup	60 MWd/kg
Doppler coefficient	-3,88 10^{-5} 1/K
Moderator density coefficient	1,86 10^{-4} m^3/kg

- Safety system design

HPLWR characteristics to be considered for safety concept

- Water inventory within primary system is about 1/8 compared to BWR or PWR
- Core is not covered with water after scram if coolant flow is not continuously maintained
- Natural circulation within primary system is not possible for HPLWR design
- Consequences for safety concept:
  - Maintaining the coolant flow through the core is mandatory for all sequences

HPLWR characteristics to be considered for safety concept development

- Water mass within the primary circuit related to the reactor thermal power
  - Gives indication about energy storage capacity and pressure increase velocity
  - Typical values for
    - PWR: 0,1 t/MW
    - BWR: 0,12 t/MW
    - HPLWR: 0,07 t/MW
- Consequences:
  - Heat storage capacity within the coolant is less compared to PWR and BWR which is an indication for potential of faster pressure transients

HPLWR characteristics to be considered for safety concept development

- Heat capacity of the core related to core thermal power
  - Determines the heat fluxes to be coped with immediately after scram
  - Typical values for
    - PWR 0,04 t/MW
    - BWR 0,044 t/MW
    - HPLWR 0,046 t/MW
- Consequences
  - Challenges to be expected with high heat fluxes after scram are comparable to that of PWR and BWR plants

HPLWR characteristics to be considered for safety concept development

- Heat transport capacity in case of loss of off-site power influenced by inertia of the pumps
  - PWR: flywheel provided for main coolant pumps
  - BWR: provision of sufficient inertia of main coolant pump and motor
  - HPLWR: measure to be decided
- Consequence
  - Feedwater-pump-motor system of HPLWR differs from main coolant pump systems

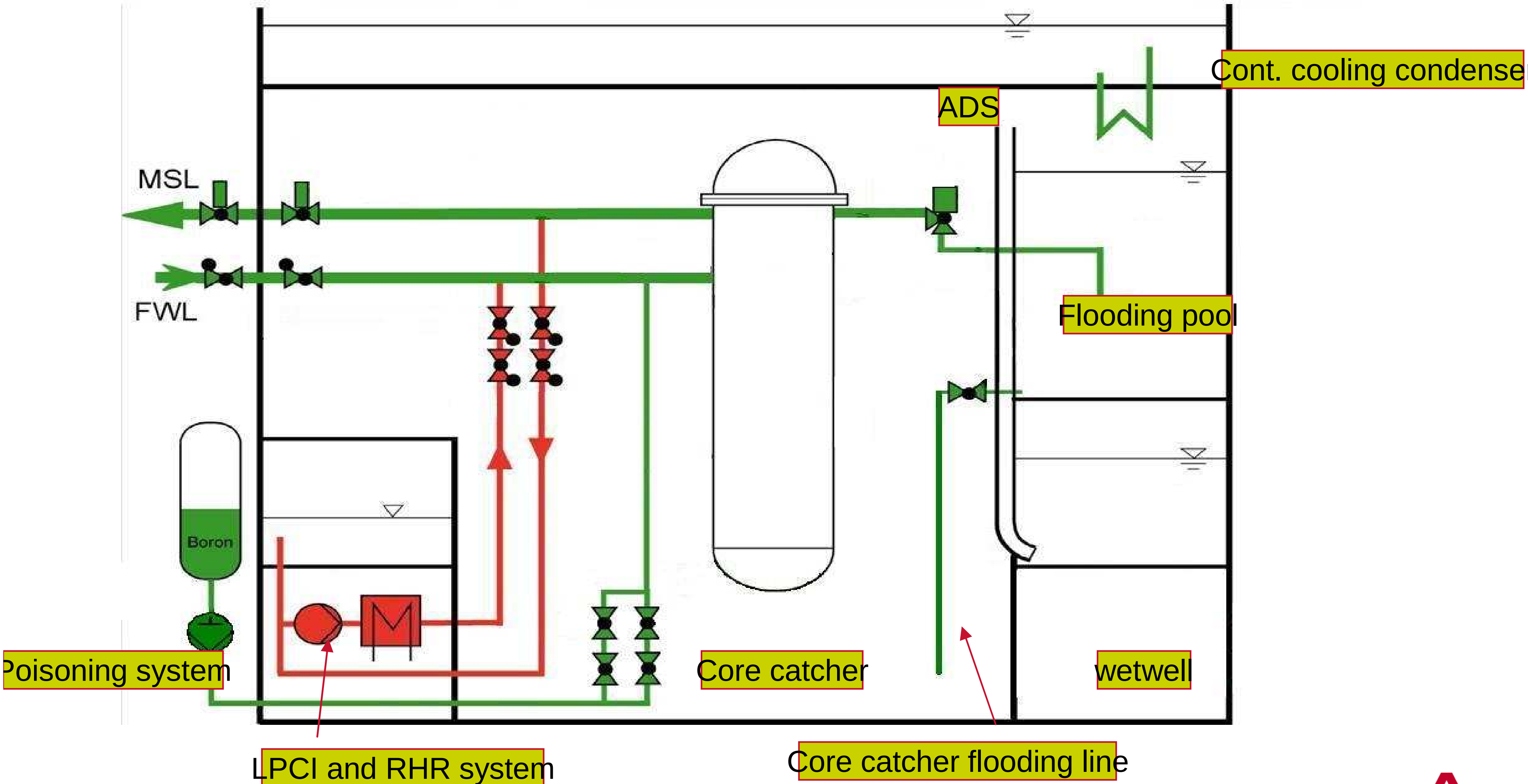
HPLWR characteristics to be considered for safety concept development

- Water level within the core after scram in non LOCA cases
  - PWR and BWR : core is always covered with water with potential of natural circulation or reflux condensing mode
  - HPLWR: due to three pass core configuration none of these cooling modes are possible
- Consequences
  - In all cases water has to be immediately supplied into the primary circuit to provide core cooling

Safety concept

Safety functions	Systems provided
Reactivity control	Two independent scram systems
Containment isolation	2 main steam isolation valves per train
Reactor pressure Control and reactor depressurization	8 safety relief valves;
Core flooding and cooling	4 LPCI systems;
RHR from RPV	4 RHR and LPCI systems;
RHR from containment	4 RHR systems; 4 containment condensers

HPLWR safety systems configuration schematic



- Safety analysis tools and challenges
  - Sample safety analysis results

Safety analyses

- From the full list of 104 events, 21 have been selected to assess the feasibility of the HPLWR. These range from operational conditions (DBC1) to severe accidents (DBC4)
- Two categories:
  1. **Transients controlled by T/H**
  2. **RIAs and ATWS**

- System codes and coupled system -3-D neutronics codes
- All codes (RELAP5, CATHARE, APROS, KIKO3D-ATHLET, SMABRE/TRAB3D)
  have been improved/extended to match the SCWR (HPLWR) conditions
- Some code-by-code comparison for selected transients/conditions permitted to build confidence in the codes

Safety analyses

- Safety analyses have to be considered as very prelimary and not consistent
- Capabilities of the codes used are different
  - ATHLET-KIKO3D (coupling of thermal-hydraulic model with 3D neutronics)
  - RELAP5 (point kinetics and thermal hydraulics, runs at supercritical conditions only)
  - CATHARE (point kinetics and thermal hydraulics, can run both in supercritical and subcritical conditions)
  - SMABRE/TRAB3D (coupling of thermal-hydraulic model with 3D neutronics at supercritical conditions)
  - APROS (thermal-hydraulic code without nuclear recoupling)

Main challenges for the codes to be used for analyses of SCWR´s

- Implementation of appropriate properties for supercritical water
- Change from supercritical fluid conditions to subcritical conditions
- Implementation of adequate heat transfer correlations and its validation
- Coupling of thermal-hydraulics and neutronics
- Modeling of the three-pass core configuration

Transients analysed

Event description (number)	Design Basis Condition (DBC)	Codes used for the analysis
Manual or inadvertent scram (1)	2	CATHARE
Loss-of-offsite power < 2 hours (17)	2	APROS(FZK)
Partial loss of feedwater – various sequences	2	CATHARE RELAP SMABRE
Loss of one feedwater pump with stand-by pump starting (23)	2	CATHARE RELAP
Loss of one feedwater pump with stand-by pump not starting (24)	2	RELAP SMABRE
Spurious closure of one MSIV (46)	2	RELAP
Inadvertent isolation of all steam lines (Closure of all MSIVs) (47)	3	APROS(FZK) RELAP
Total loss of feedwater (ATWS) - no automatic scram, stand-by pump available	4	RELAP SMABRE
Total loss of feedwater - scram, stand-by pump not available	2	APROS(FZK)
Small leak of a steam line (approaching intermediate LOCA) (67 or 69)	4	SMABRE
Main steam line break (2A) inside containment (76)	4	CATHARE APROS(VTT)
Feedwater line break (2A) inside containment (77)	4	CATHARE

Twelve transients have been analysed, including parametric variation of key variables. Sequences entail:

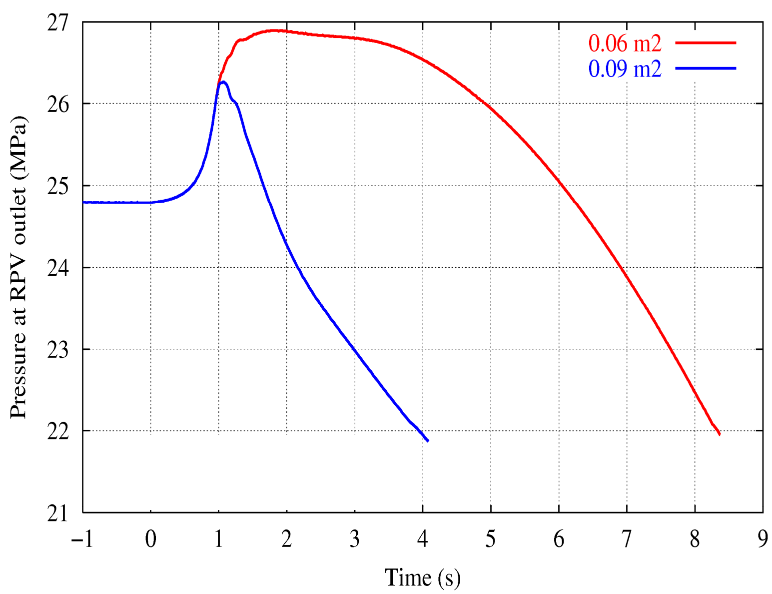
- loss of flow
- depressurisation
- pressurisation
- loss of coolant
- power excursions

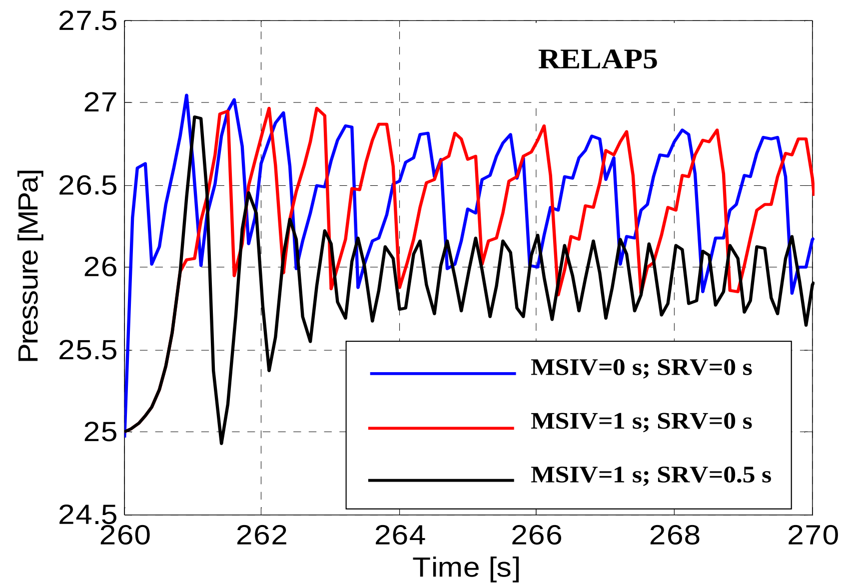
List of RIAs and ATWS

Event description (number)	Design Basis Condition (DBC)	Codes used for the analysis
Uncontrolled absorber group withdrawal from the bottom position (6)	2	ATHLET-KIKO3D
Uncontrolled withdrawal of absorbers from the middle position (6)	2	ATHLET-KIKO3D
Uncontrolled withdrawal of one group of absorbers in asymmetric position (6)	2	ATHLET-KIKO3D
Loss of feedwater heating (32)	2	ATHLET-KIKO3D
Control rod malfunction: stuck control rod in a lower or upper position	2	ATHLET-KIKO3D
Control rod ejection (9)	3	ATHLET-KIKO3D SMABRE/ TRAB-3D
Turbine trip with bypass failure, without scram, aux. power available (82)	4	SMABRE/ TRAB-3D
Loss of HP FW preheaters without scram (88)	4	ATHLET-KIKO3D
Inadvertent withdrawal of control rod groups without scram (89)	4	ATHLET-KIKO3D

Nine events have been analysed, which require the use of the coupled codes

Pressure transient following a turbine trip calculated with the SMABRE/TRAB-3D (left) and RELAP5 (right) codes



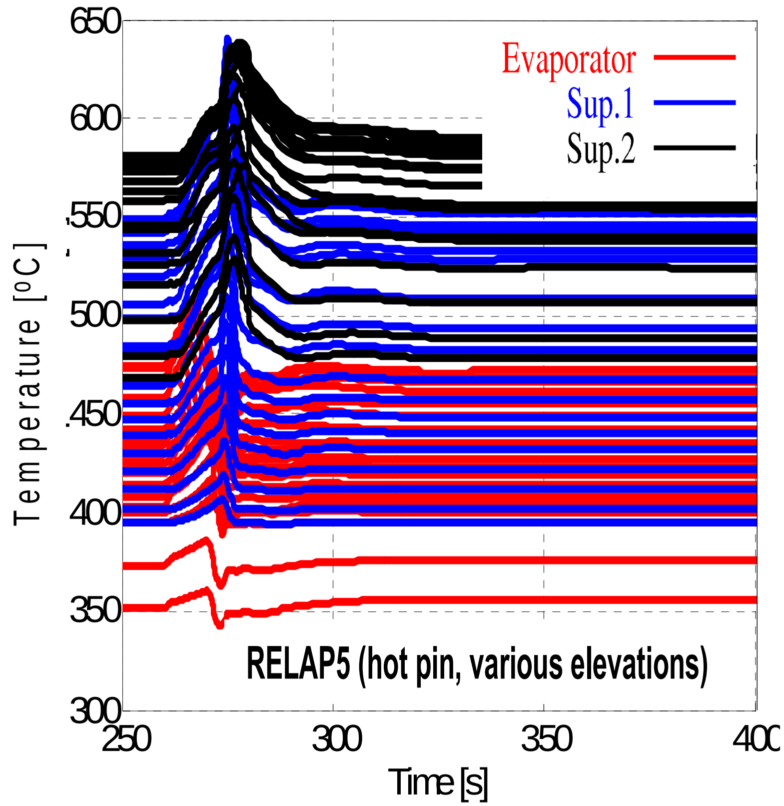


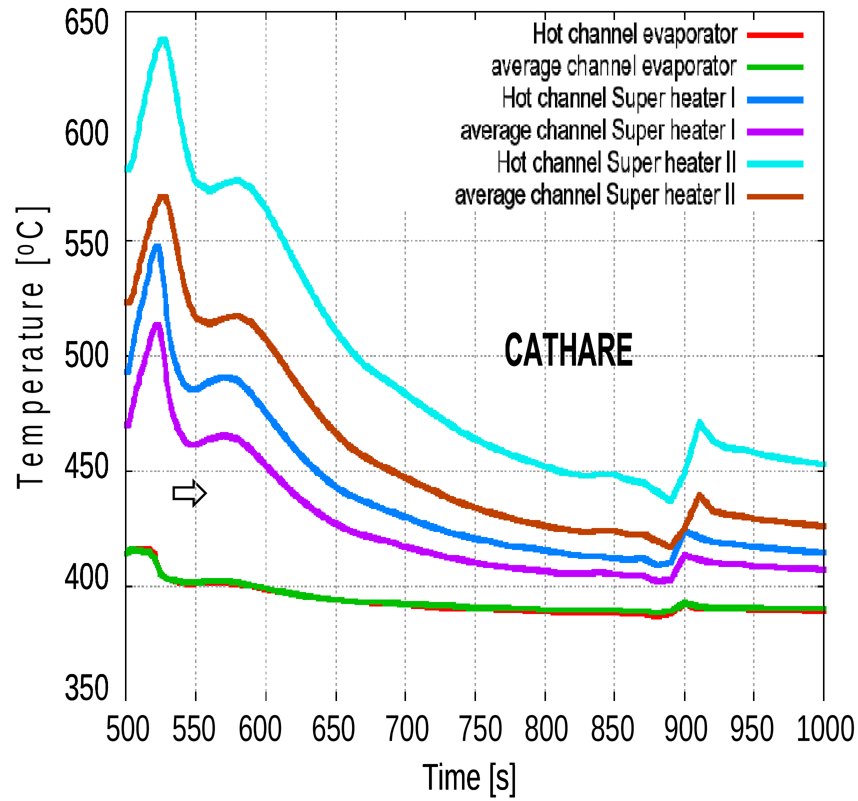
SMABRE/TRAB-3D with ADS actuation at 26 MPa

RELAP5 with SRV setpoint at 26 MPa

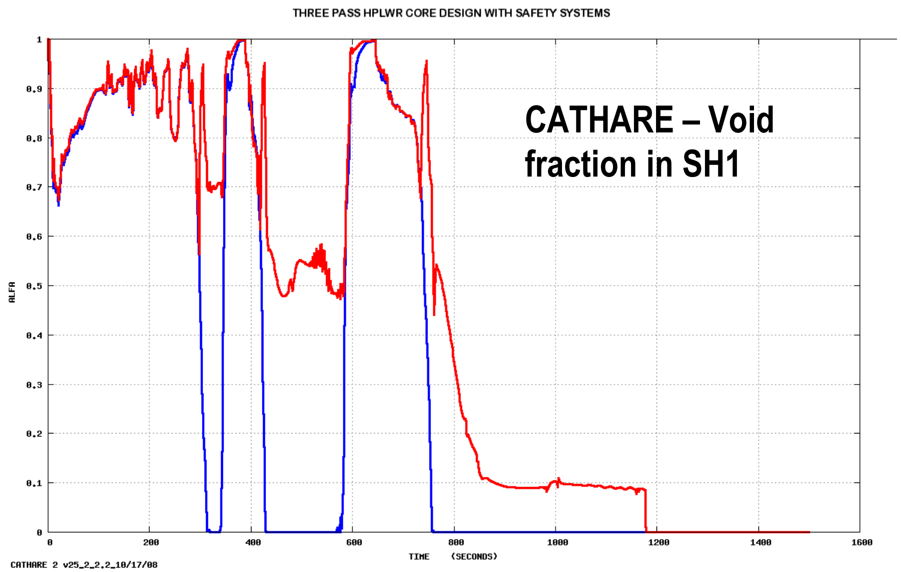
⇨ **Total valve area of 0.09 m^2 and set point at 26 MPa can limit the pressure rise**

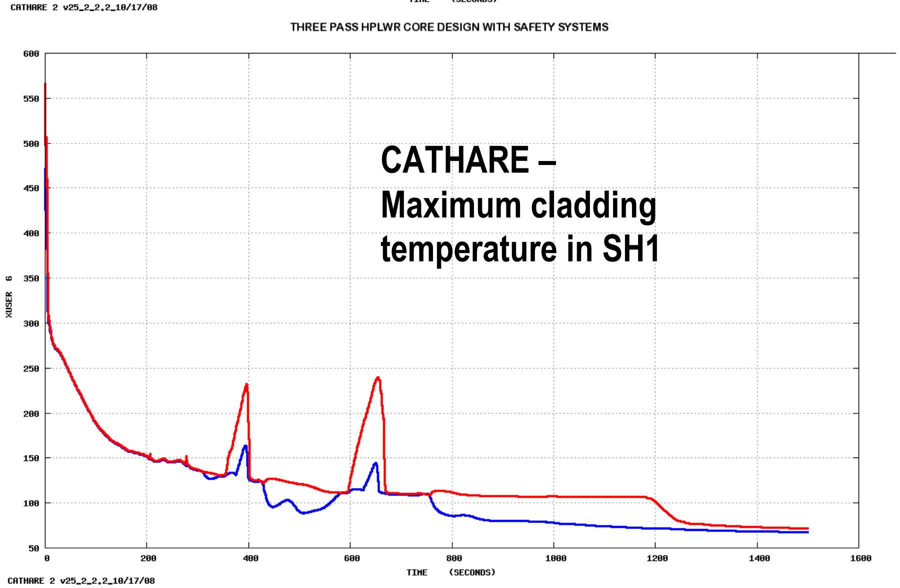
Peak cladding temperature calculated with the RELAP5 and CATHARE codes for a LOFW event, with failure of two pumps and delayed start-up of one stand-by pump.





Average void fraction and maximum cladding temperature in superheater 1 calculated by CATHARE for a main steam line break (MSLB)





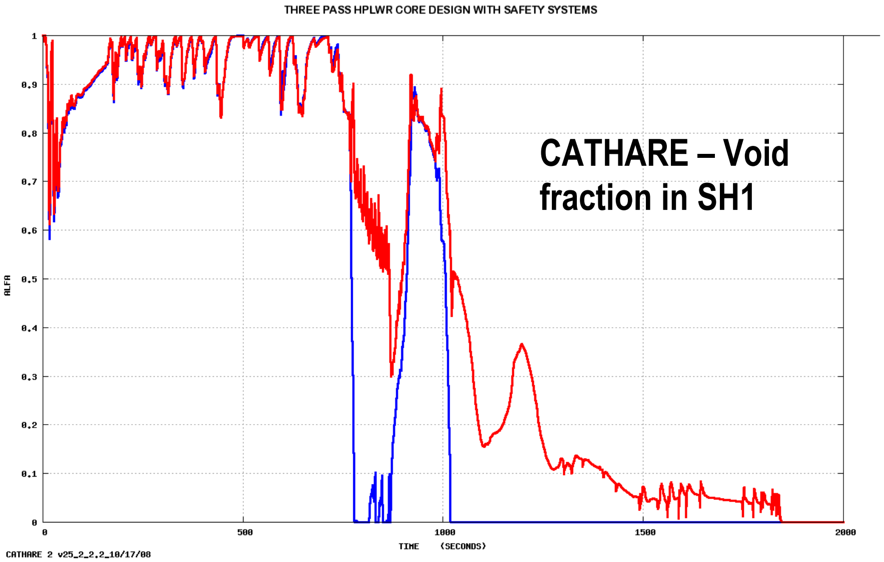
- With a LPCI flow rate of 250 kg/s the core can be refilled within 1200 s (SH1 is the last to refill)

•Blue lines delineate average

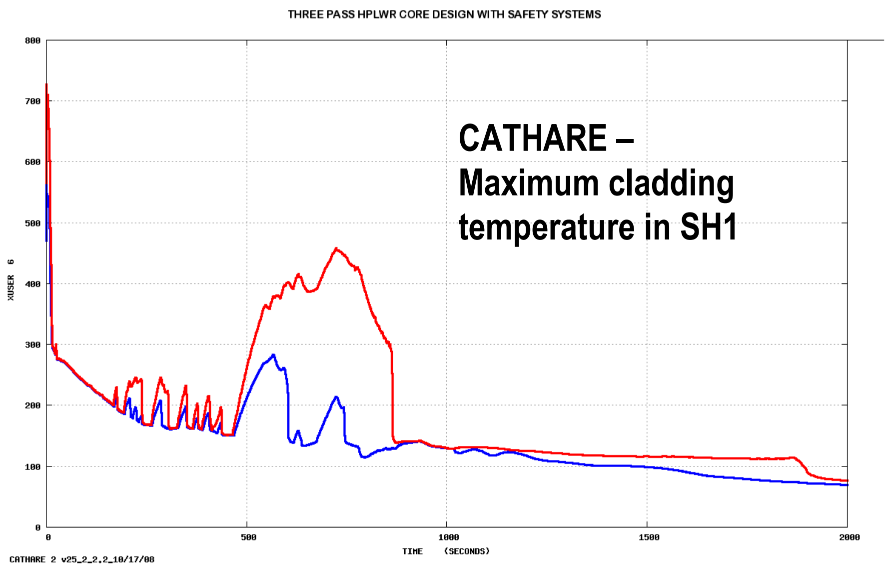
•Red lines delineate hot channel

- The maximum temperature in the core decreases nearly monotonically, with the exception of a low peak during the first seconds and few low peaks of relatively short duration

Average void fraction and peak cladding temperature in SH 1 calculated by CATHARE for a large break in the feedwater line

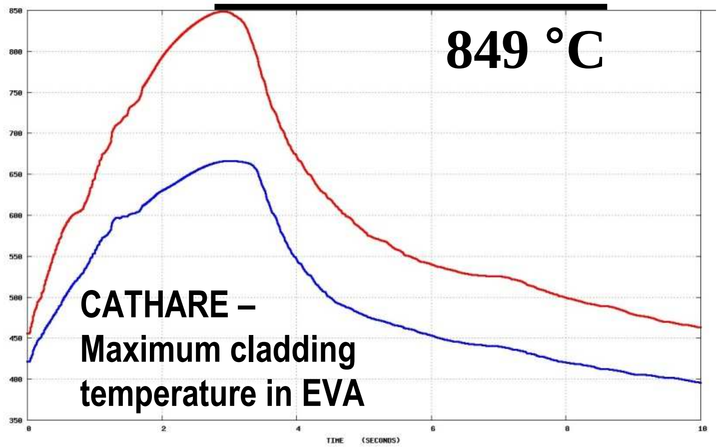


- With a LPCI flow rate of 250 kg/s the core can be refilled within 2000 s



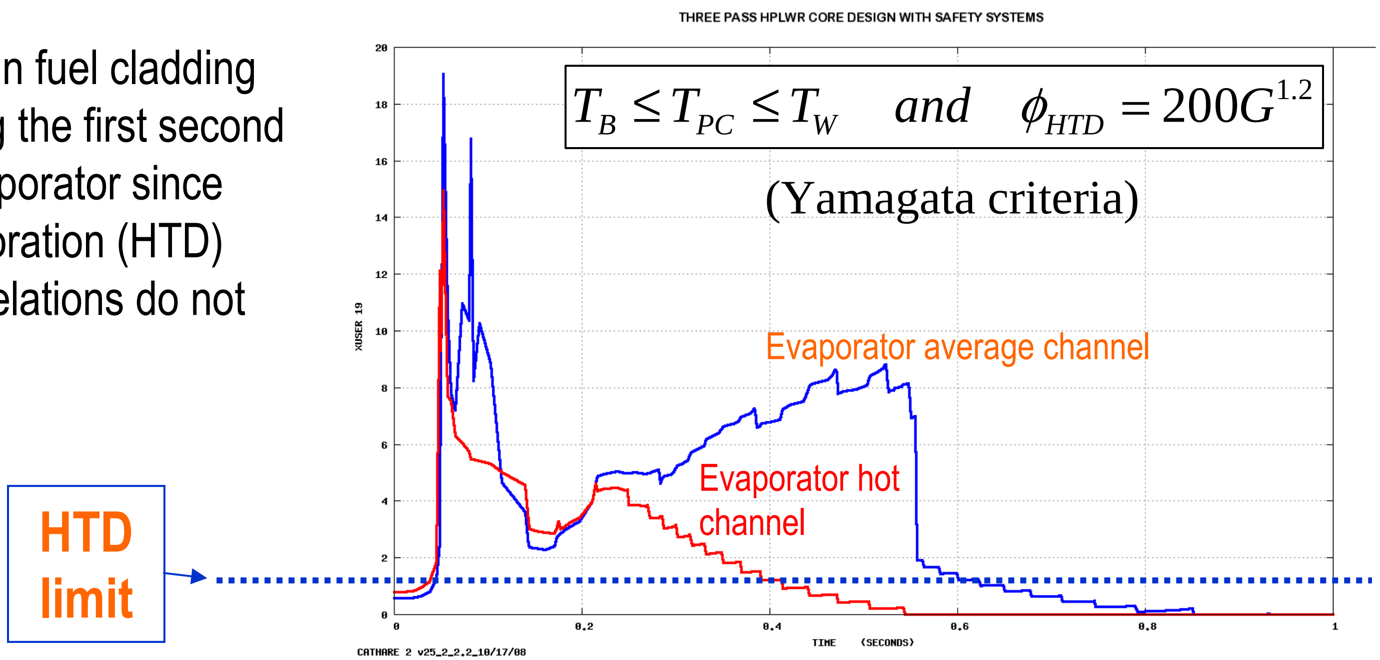
- The maximum temperature in the core **after the pressure has dropped below the critical value** first decreases, then increases and finally decreases again, with a peak value well acceptable

Peak cladding temperature in the evaporator calculated by CATHARE for a large LOCA in the feedwater line during the first seconds

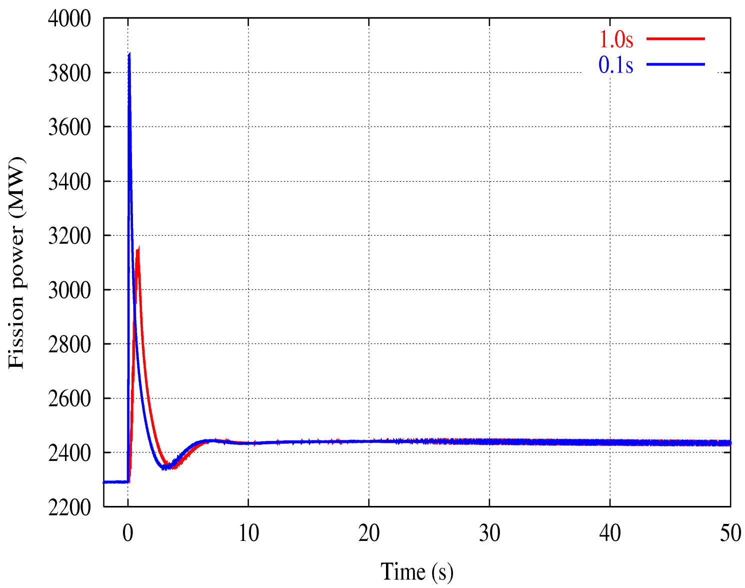


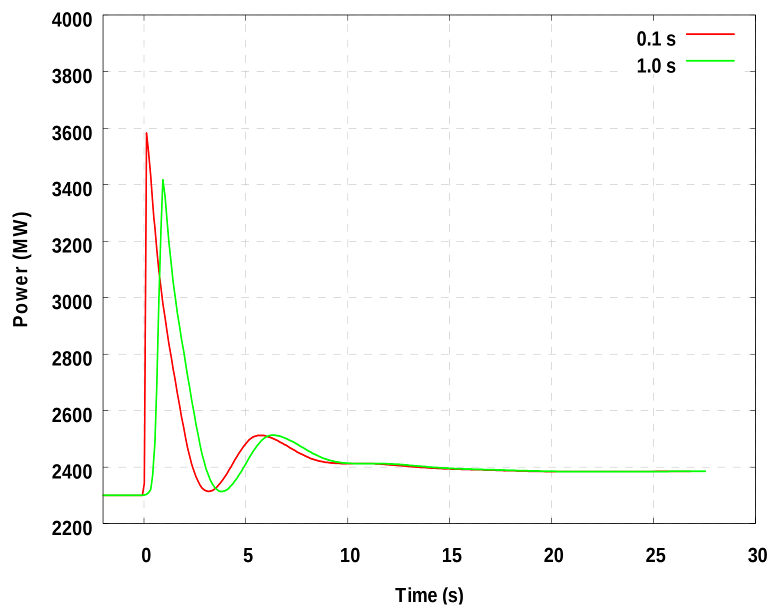
The maximum temperature in the core **before the pressure has dropped below the critical value** is rather high, though far from acceptability limits. HTC calculated with Jackson x 0.9

No real confidence in fuel cladding temperatures during the first second of simulation in Evaporator since heat transfer deterioration (HTD) occurs and HT correlations do not take it into account

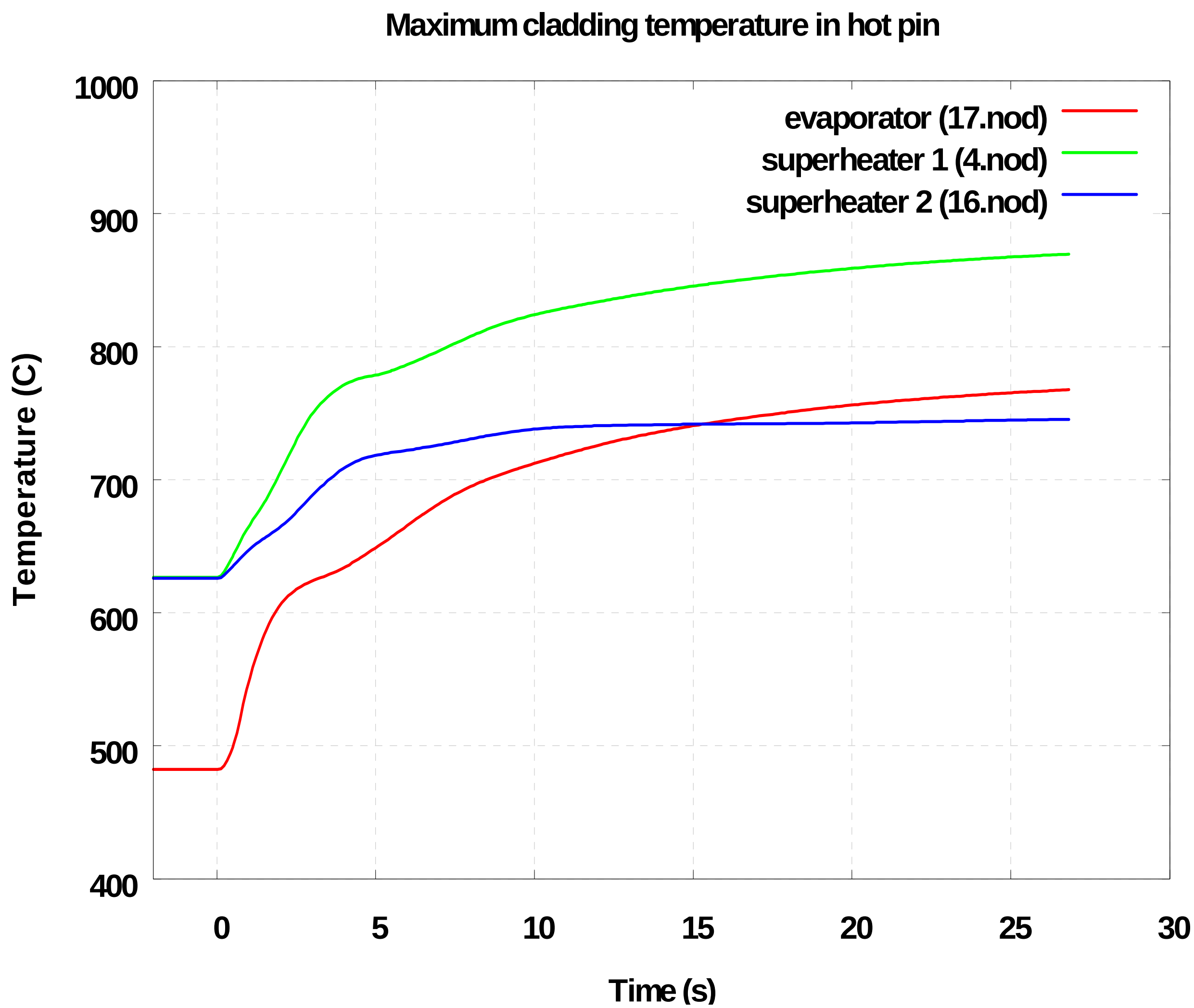


Comparison of the power calculated for a control rod ejection event with SMABRE/TRAB-3D (left) and ATHLET-KIKO3D (right) for the two cases with 0.1 s and 1 s ejection time, respectively





Peak cladding temperatures in the three regions of the core calculated by ATHLET-KIKO3D for a CRE with 0.1 s ejection time



Conclusions

- Code capabilities have been upgraded to a certain extend in order to be able to simulate all selected transients. **Asset for all future SCWR projects.**

- The general safety concept of the HPLWR could be assessed to a large extent. Further issues related to specific transients must be addressed in future projects. **Increased use of coupled code is required**.

- The results of the analysis indicate that the safety criteria could be fulfilled

- The need of a high pressure coolant injection system, preferably passive, should be assessed in a future project.

- Some further study is needed to introduce appropriate measures to prevent core melt in the case of uncontrolled withdrawal of an absorber from the bottom position without SCRAM. Limitations on the allowed control rod worth and positions should be investigated.

Open issues

- Experimental data are now required to assess the capabilities of the codes
- Model for heat transfer deterioration is necessary for evaluating the cladding temperature excursion during the first seconds of transients
- First assessment should be confirmed for an optimised core using a methodology which accounts for a more realistic cladding and fuel temperature distribution
- Coupled 3-D analysis is mandatory in order to reflect the specific features of the core
- Safety analyses of containment response to be performed in future phase of the project
- Second scram system should be investigated

References

1. M. Andreani et al, Evaluation of a preliminary safety concept for the HPLWR, to be published in Nuclear Energy
2. D. Bittermann, The safety concept of the HPLWR, 4th International Symposium on SCWR´s, Heidelberg, March 8-11, 2009