System Codes for Reactor Licensing – Part 1: Code Applications

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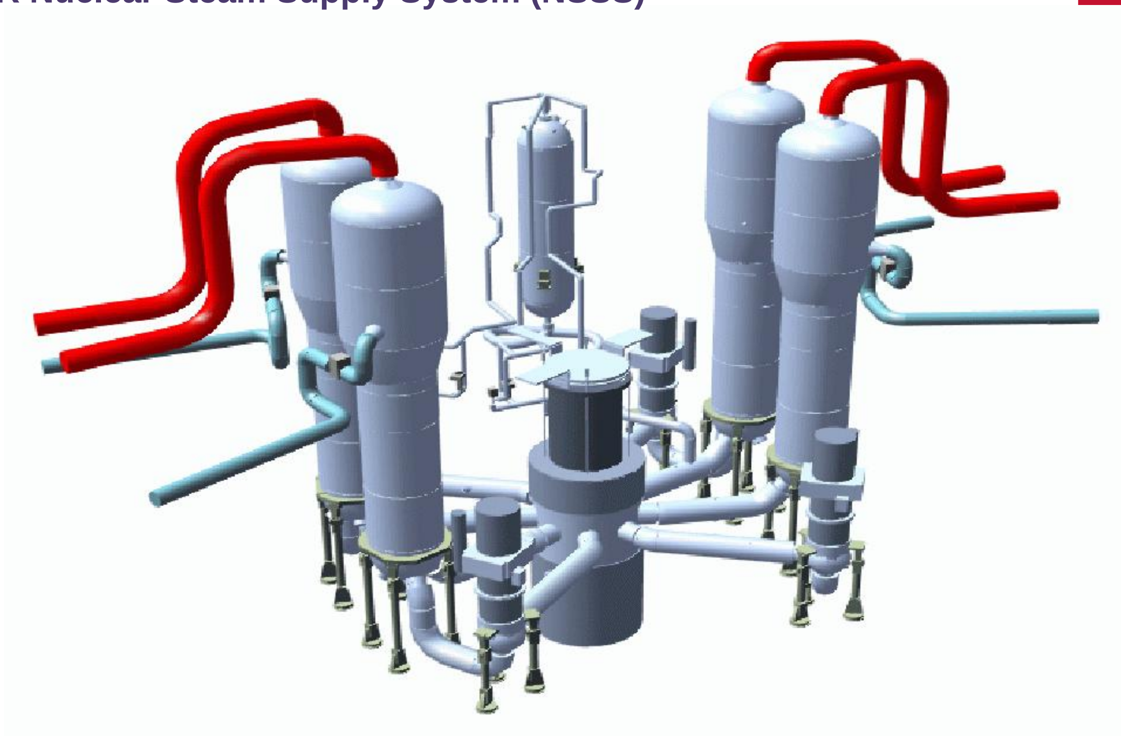
Contents

- Definition of steady state conditions and transients modelled by system codes in EPR safety analysis
- Typical System Codes used for EPR and their validation
- Analysis results for Design Basis Accidents: Illustration of Thermal-hydraulic Phenomena modelled

EPR Nuclear Steam Supply System (NSSS)



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POSTULATED INITIATING EVENTS

- **Multiple Initiating Events (IEs) are analysed in the Reactor Safety Report to show that the following basic safety functions can be achieved:**
  - Core reactivity control
  - Residual heat removal
  - Control of Radioactivity releases
- **The IEs analysed are grouped in categories:**
  - Design Basis Conditions (DBC1 to DBC4)
  - Design Extension Conditions (DECs)
  - Severe Accidents (Core Melt Accidents)
  - Internal and External Hazards

DBCs DEFINED FOR

- **DBC 1 : Normal operational transients – Routine events**
- **DBC 2 : Anticipated operational transients and occurrences – events that might be expected to occur during the life of a unit (1E-2<f<1/yr)**
- **DBC 3 : Incidents/infrequent accidents – events that might expected to occur during the lifetime of a fleet of similar units (1E-4<f<1E-2/yr)**
- **DBC 4 : Limiting Accidents – Events that would not be expected to occur during the lifetime of a fleet of similar units (1E-6<f<1E-4/yr)**

In defining the DBCs, all reactor operating states must be considered: (at power, hot shutdown, cold shutdown with closed circuit, cold shutdown with open circuit, cold shutdown with fuel removed)

DBC 2 Events : $f > 10^{-2}$/yr

- Feedwater malfunction reduction/increase in feedwater temperature
- Excessive increase in secondary steam flow
- Turbine trip
- Loss of condenser vacuum
- Short term loss of offsite power (≤ 2 hours)
- Loss of normal feedwater flow
- Partial loss of core coolant flow (Loss of one reactor coolant pump)
- Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power & hot zero power conditions
- RCCA rod drop
- Start-up of an inactive reactor coolant loop at an incorrect temperature
- RCV [CVCS] malfunction resulting in boron dilution or increase/ decrease in reactor coolant inventory
- Primary side pressure transient (spurious operation of pressuriser spray, heater)
- Uncontrolled level drop in primary circuit in shutdown
- Loss of one Residual Heat Removal System Train during shutdown
- Spurious reactor trip at power

DBC 3 Events : $10^{-2} > f > 10^{-4}$/yr

- Small steam or feedwater system piping failure
- Long term loss of offsite power (> 2 hours)
- Inadvertent opening of a pressuriser safety valve
- Inadvertent opening of a SG relief train or of a safety valve (state A)
- Small break LOCA at power (not greater than DN 50mm)
- Steam generator tube rupture (1 tube)
- Inadvertent closure of one/all main steam isolation valves
- Inadvertent loading and operation of a fuel assembly in an improper position
- Forced decrease of reactor coolant flow (4 pumps)
- Leak in the gaseous or liquid waste processing systems
- Loss of primary coolant outside the containment
- Uncontrolled RCCA bank withdrawal in shutdown
- Uncontrolled single control rod withdrawal
- Long term loss of offsite power (> 2 hours), fuel pool cooling aspect
- Loss of one train of the fuel pool cooling system or of a supporting system
- Isolable piping failure on system connected to the fuel pond

DBC 4 Events : $10^{-4}>f>10^{-6}$/yr

- Long term loss of offsite power in shutdown
- Major Steam system piping break
- Major Feedwater system piping break
- Inadvertent opening of a SG relief train or safety valve – hot shutdown
- RCCA ejection accident
- Intermediate and large break LOCA at power
- Small break LOCA <50 mm during shutdown
- Reactor Coolant Pump seizure (locked rotor)/ shaft break
- Multiple Steam Generator tube rupture (2 tubes in 1 SG)
- Fuel handling accident
- Boron dilution due to a non-isolable rupture of heat exchanger tube
- Rupture of systems containing radioactivity in the Nuclear Auxiliary Building
- Isolable break in safety injection system in residual heat removal mode during shutdown

Design Basis Analysis – Acceptance Criteria

- **More conservative limits applied to more frequent event classes**
  - **Offsite radiological consequences of DBC2 events must be within limits for normal operation**
  - **Offsite radiological consequences of DBC3/4 events must not require off-site countermeasures (10mSv max dose to person at site boundary)**
  - **No fuel clad failures permitted in DBC2 events and DBC3/4 Steam/Feed Line Break Events (no DNB)**
  - **Number fuel rods experiencing DNB for other DBC 3/4 events must be < 10%.**
  - **In LOCAs: peak clad temperature must be< 1200°C, max clad oxidation must be<17% of the clad thickness, max hydrogen generation must be < 1% of maximum from oxidation of active core fuel clad, core geometry must remain coolable etc**

Design Basis Analysis – Analysis Assumptions

- **Conservative assumptions applied for initial and boundary conditions and system modelling (aim is >95% confidence that analysis will be bounding). E.g.**
  - **Initial plant conditions (power, pressure etc) assumed to be at limits allowed by operating rules. (Initial steady state operation assumed).**
  - **Parameters for dominant phenomena set conservatively to allow for modelling uncertainties (e.g. decay heat, reactivity feedback coefficients etc)**
  - **Single failure & maintenance principles applied**
  - **No operator actions from control room claimed within 30 minutes of first indication: no local to plant actions claimed within 60 minutes**
  - **Loss of offsite power assumed in DBC3/4 events (when pessimistic)**

Definition and examples of DECs & Severe Accidents

- **DECs: these are fault sequences involving IE combined with failure of a major safety system, where core melt is averted by use of back-up systems e.g.**
  - **Station Blackout (Loss of offsite power combined with failure of all 4 Emergency Diesel Generators)**
  - **Main feedwater failure combined with failure of the 4 Emergency Feed trains,**
  - **SB-LOCA combined with failure of 4 Medium Head Injection trains**
  - **SGTR combined with stuck open SG relief valve**
- **Severe Accidents : these are core melt accident in which a large release of radioactivity to environment is prevented e.g.**
  - **LOCA with total failure of all Safety Injection Systems (both Medium & Low Head Injection)**
  - **SBO with failure of all 6 diesel generators (Emergency & Back-up)**

DEC Analysis – Acceptance Criteria & Analysis Assumptions

- **Assumptions for DEC more realistic than those applied for design basis event analysis**
  - Standard conditions assumed for initial plant operating state (e.g. nominal rated thermal power)
  - Parameters for phenomena modelled defined more realistically
  - Single failure principle not normally applied. Maintenance principle applied on case-by-case basis
  - No operator actions from control room within 30 minutes: no local to plant actions within 60 minutes – same as DBCs
  - No coincident loss of offsite power assumed
  - Required offsite radiological consequences of DEC events same as DBC3/4 (no off-site countermeasures must be needed)

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Main codes & use for EPR Licensing in UK

Plant thermal hydraulics

6 equations required – separated two-phase-flow

-CATHARE (S-RELAP)

5 equations + drift flux modelling (NON-LOCA)

-MANTA (THEMIS, NLOOP)

Radiological consequences

- ORIGEN, ACARE, PRODOS-B, ALICE, CORA, ASTRAL, COSAC

Core thermal hydraulics & DNBR

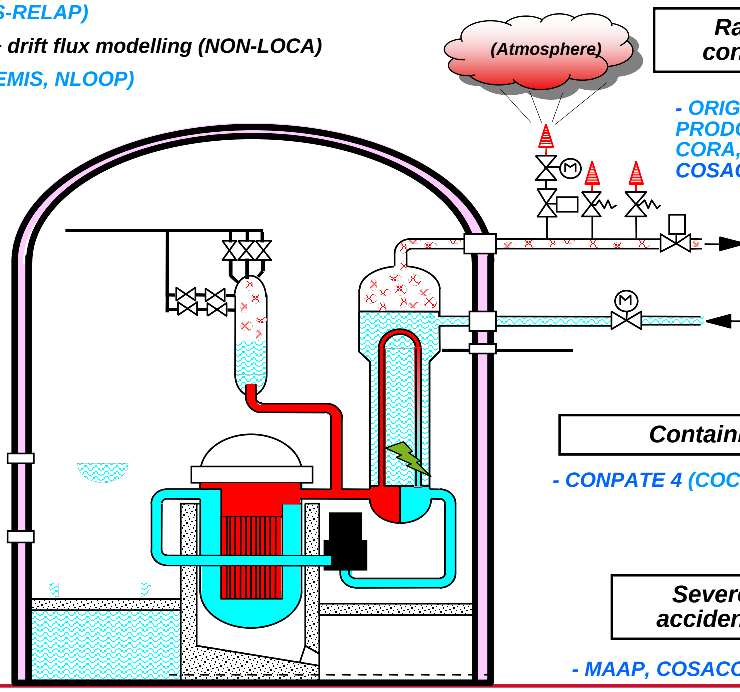
-FLICA (COBRA)

Neutronics

-SMART (PANBOX)

Fuel thermal behaviour

- COMBAT (COPERNIC)



Containment

- CONPATE 4 (COCO, PAREO)

Severe accidents

- MAAP, COSACO, WALTER,…

CATHARE MODEL

- **CATHARE code development launched in 1979 by CEA, EDF, FRAMATOME-ANP. Aim was to develop a state-of-the-art best-estimate thermal-hydraulic code for realistic calculations of accident scenarios in LWRs.**
- **Supported by a comprehensive experimental validation programme including Separate Effects Tests and Integral Effects Tests**
- **Transients addressed involve limited core degradation (fuel cladding deformation and bursting - core melt events excluded).**
- **Main Reactor transient applications :**
  - **LOCAs up to the Double-Ended Guillotine Break of main primary loop pipework**
  - **All accidents leading to "significant 2-phase conditions" in the RCS – characterised by flow stratification in horizontal pipework in main loops**
  - **Transients involving degraded heat transfer in SG secondary system, due to steam/feed pipe ruptures or system malfunctions (LOFW, SLB, FWLB, SGTR, …)**
  - **Modelling of Containment pressure/temperature response due to Mass and Energy Release from the RCS**

CATHARE MODEL

Basic assumptions and models :

- 2 fluid / 6 equation model
- 4 non-condensable gas fields
- 32 radiochemical elements
- Fortran 77 (5000 routines, 720 000 lines)
- Finite difference solution scheme
  - First order, staggered mesh space discretization
  - Fully implicit (0D, 1D) or semi-implicit (3D) time discretization
- Hyperbolic system of equations
- Newton-Raphson method for non-linear equation solution

CATHARE MODEL – 6 Equation Model used for 1D Module

- MASS BALANCE EQUATION FOR PHASE K

$$A\frac{\partial}{\partial t}\alpha_K\rho_K + \frac{\partial}{\partial z}A\alpha_K\rho_K V_K = \Gamma_{iK}$$

- TRANSPORT EQUATION FOR NON CONDENSABLE GAS

$$A\frac{\partial \alpha\rho_G X_i}{\partial t} + \frac{\partial A\alpha\rho_G X_i V_G}{\partial z} = S_i$$

- MOMENTUM BALANCE EQUATION OF PHASE K

$$A\frac{\partial \alpha_K\rho_K V_K}{\partial t} + \frac{\partial A\alpha_K\rho_K V_K^2}{\partial z} + A\alpha_K\frac{\partial P}{\partial z} = AI_{iK} + \chi_F\tau_{WK} + A\alpha_K\rho_K g_z$$

- ENERGY BALANCE EQUATION OF PHASE K

$$A\frac{\partial \alpha_K\rho_K (H_K + \frac{V_K^2}{2})}{\partial t} - A\alpha_K\frac{\partial P}{\partial t} + \frac{\partial A\alpha_K\rho_K (H_K + \frac{V_K^2}{2})}{\partial z} = AQ_{iK} + \chi_F Q_{WK} + A\alpha_K\rho_K V_K g_z$$

- INTERFACE RELATIONSHIP

$$\sum_K \tau_{iK} = 0 \qquad \sum_K I_{iK} = 0 \qquad \sum_K Q_{iK} = 0$$

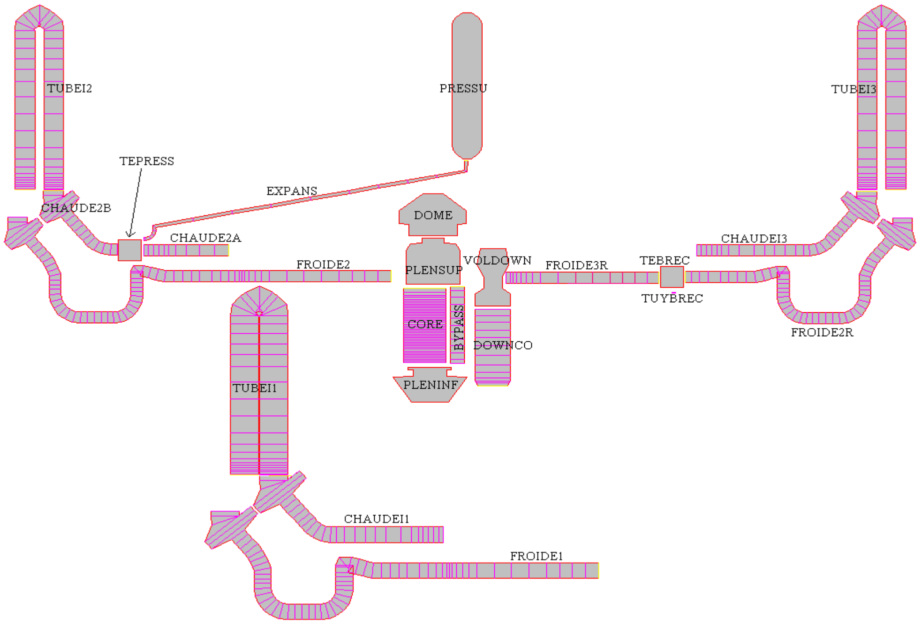
- INTERFACE ENERGY TRANSFER

$$Q_{ik} = q_{iK} + \Gamma_{iK}(H_K + \frac{V_i^2}{2})$$

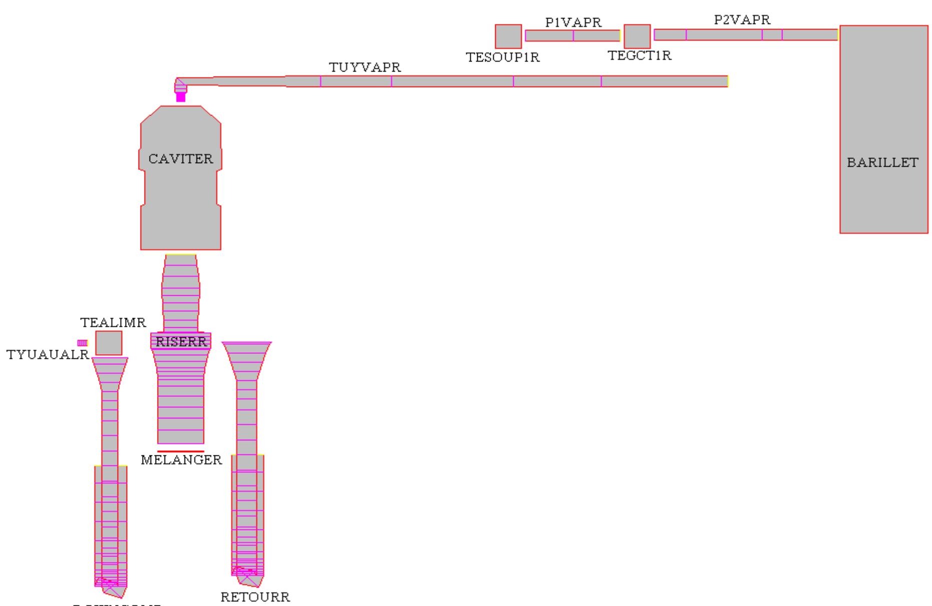
q_{iK} is the interface to phase K heat flux

$\Gamma_{iK}(H_K + \frac{V_i^2}{2})$ is the energy transfer due to mass transfer

CATHARE MODEL – Primary System Nodalisation



CATHARE MODEL – Secondary System Nodalisation



CATHARE MODEL – Validation against system tests

LOOP	VERT. SCALE	VOLUME SCALE	POWER	PRESSURE MPa	LOOP NB	CORE
LOFT	1/2	1/48	100%	16	2	Nucl
LSTF	1/1	1/48	14%	16	2	Elect
BETHSY	1/1	1/100	10%	16	3	Elect
PKL	1/1	1/134	5%	4	3	Elect
LOBI	1/1	1/700	100%	16	3	Elect
SPES	1/1	1/427	100%	16	3	Elect
PACTEL	1/1	1/305		8	3	Elect
PMK	1/1	1/2070	100%	16	1	Elect

MANTA CODE

- **MANTA is an AREVA code used to simulate the transient behavior of a multiple-loop PWRs (non-LOCA) used for:**
  - **Safety analysis report**
  - **Equipment design**
- **Secondary side modelling:**
  - **Steam line break, excessive increase in steam flow, spurious opening of a valve.**
  - **Loss of feed water, feedwater system malfunction**
- **Primary side modelling:**
  - **Natural circulation, loss of reactor coolant flow, startup of a RCP, locked rotor of a RCP,**
  - **Spurious opening of a pressuriser relief valve, spurious startup of safety injection,**
  - **Control rod withdrawal, rod drop, spurious boron dilution,**
  - **ATWS**

MANTA Models

- Core model
  - Fuel to coolant heat transfer model: multiple axial nodes, one radial node per loop, one heat transfer coefficient.
  - Neutron kinetics model: Point kinetics (6 groups of delayed neutrons). Is coupled with 3-D neutronics code SMART if neutron power distribution in core is required.
  - DNBR calculation using simple model function of core power, reactor coolant flow rate and pressurizer pressure.
- Reactor upper head vessel model:
  - Multi-nodal modelling with pressure gradient & heat losses.
- Pressurizer model:
  - Multi-nodal possible with heat losses and mass transfer.
- Steam generator model
  - Multi-nodal modelling for tube bundle and secondary side (boiler, economiser, separator)
- Control and Protection System Modelled in Detail

MANTA – Thermal Hydraulic Modelling

- Control volume method used
- 5 equation model of two-phase flow
  - Mixture mass conservation
  - Vapour mass conservation
  - Mixture momentum conservation
  - Vapour energy conservation
  - Liquid energy conservation
- 4 radial regions in core corresponding to each coolant loop. Thermal and boron mixing between regions simulated using mixing coefficients
- Algebraic drift flux correlations used to represent the velocity difference between liquid and vapor phases. (Code not used for transients with significant two-phase conditions in primary system)
- Zaloudek/Homogeneous Equilibrium Models used two-phase critical flow though orifices/pipes.

MANTA - Validation

Transients on PWRs in France

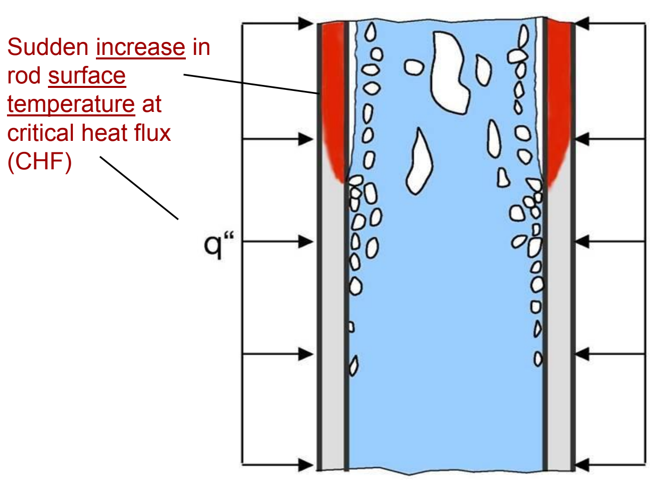
- ► **Reactor steady state operations : Bugey 4, Paluel 1**
- ► **Reactor trip at 50% NP Bugey 4 and 100% NP Paluel 1**
- ► **Primary overpressure transient - Bugey 4**
- ► **Steam generator valves opening transient - Paluel 3**
- ► **RCS natural circulation and void formation under vessel head - Gravelines 1**
- ► **House load operation - Gravelines 6**
- ► **Power transients and feed water injection Chooz B1**

Transients on Large Scale Mock-ups of Steam Generators

- ► **MB2: Steady state, loss of feedwater, steam line break**
- ► **MEGEVE: steady state, reactor trip**

Modelling of Departure from Nucleate Boiling Phenomena

One of the most important tasks in core thermal-hydraulics is the prediction of thermal margin (margin to boiling crisis).



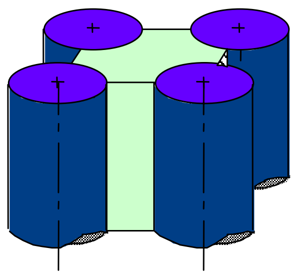
Departure from Nucleate Boiling (DNB, Film Boiling)

DNB limit

- To avoid damage to the cladding due to an excessive increase in the temperature, the heat flux Q must not exceed the critical heat flux Q_c. The **DNBR (Departure from Nucleate Boiling Ratio)** is defined as the ratio of the critical flux to the actual heat flux at any time

$$DNBR = \frac{\text{Critical Heat Flux}}{\text{Local Heat Flux}}$$

- The critical heat flux is determined experimentally. A correlation (or predictor) is established that allows the critical flux Qc to be calculated as a function of the flow and the geometrical characteristics of the channel



Typical cell

DNB risk : rupture of the first barrier

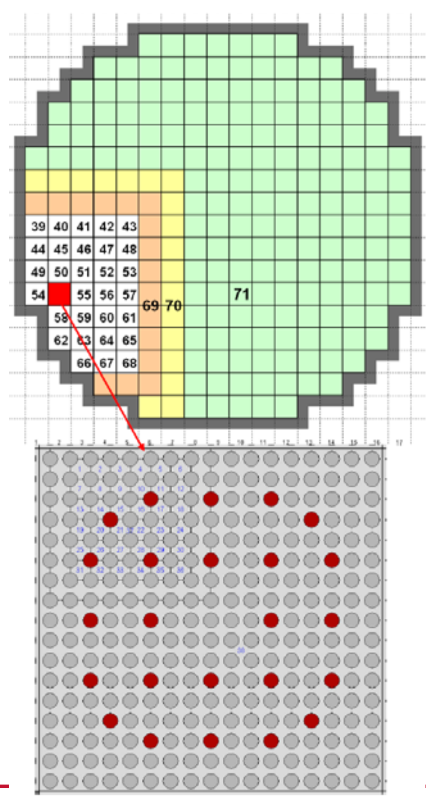
FLICA III-F core thermal-hydraulic model

- FLICA III-F is sub-channel code that calculates two-phase flow and heat transfer in the core of a PWR, in steady and transient states:
  - thermal-hydraulic variables: pressure, enthalpy, temperature, quality, mass flowrate
  - critical heat flux
- FLICA applications:
  - thermal-hydraulic design of reactors: determination of core operating limits in regard to DNB phenomenon
  - modelling of accidents such as steam line break, uncontrolled control rod withdrawal,
  - hydraulic design of core e.g. determination of hydrodynamic lift forces on fuel assemblies

FLICA III-F core thermal-hydraulic model assumptions (1/2)

- Core divided radially into channels and sub channels representing individual subchannels or multiple subchannels or one or several fuel assemblies
- Code assumes vertical uplow flow with mass and energy exchange between adjacent channels
- Single and two-phase flow modelled up to CHF location
- Incompressible flow assumed
- Counter-current flow and flow reversals not modelled
- 4 equation model of two-phase flow used with slip ratio correlation:
  - Mixture mass conservation equation
  - Mixture momentum conservation equation
  - Mixture energy conservation equation
  - Liquid phase energy conservation equation

FLICA – Radial Mesh used for Steam Line Break Fault Analysis



FLICA III-F core thermal-hydraulic model assumptions (2/2)

Two-phase flow models

- Slip ratio model used for calculating the difference in velocity between the two phases – HTFS correlation
- Two phase flow friction factor for axial flow – HTFS correlation used that takes account void fraction, mass velocity and heat flux
- Condensation coefficient for inter-phase heat transfer – correlation from CEA tests on subcooled boiling
- Wall heat transfer coefficients in saturated boiling from Jens-Lottes/Forster-Greif correlations
- Turbulent viscosity and turbulent thermal diffusion modelled for transverse two-phase exchange of heat and mass between subchannels. Mixing coefficients from test data
- Axial thermal conduction and axial turbulent diffusion neglected
- Transverse flow friction factor used in the lateral momentum balance equation

FLICA III-F Code - Validation

- Void fraction measurements in sub-cooled boiling – validation of slip ratio correlation and condensation (inter-phase heat transfer) coefficient
- Mass velocity and steam quality measurements in boiling channels and rod bundle geometries – validation of inter-channel mixing model for single and two-phase flow
- Single phase mixing test in rod bundle geometries: validation of mixing coefficients
- Velocity measurements upstream and downstream of spacer grids
- Pressure drop measurements in two-phase flow – validation of two-phase pressure drop model
- Critical heat flux experiments : validation of CHF correlations
- Benchmarking against previous THINC IV code used for CHF modelling. 3-loop and 4-loop calculations performed for :
  - nominal operating conditions
  - reduced flow
  - overpower operating conditions

Contents

Loss of coolant accident (LOCA)

Several LOCA transients considered in EPR design basis:

- DBC-2: Very small LOCA: No requirement for safety injection function
  - Leakage flow is compensated by normal make-up from CVCS
- DBC-3: Small LOCAs Φ < DN50mm
  - Core uncovery avoided in EPR
  - Safety injection from high head (MHSI) injection system critically important
- DBC-4: Intermediate/Large LOCA
  - ⇨ Cold Leg Breaks up to double ended break of largest connected line (Safety Injection Line Rupture – 225mm ND)
  - ⇨ Hot Leg Break up to double ended break of largest connected line (Pressuriser Surge Line Rupture – 335mm ND)
    - **Limited core uncovery permitted**
    - **Low head, medium head system injection and accumulators injection important**

LOCA – Protection Requirements

Automatic Protection

- Reactor trip on Low Pressuriser pressure signal

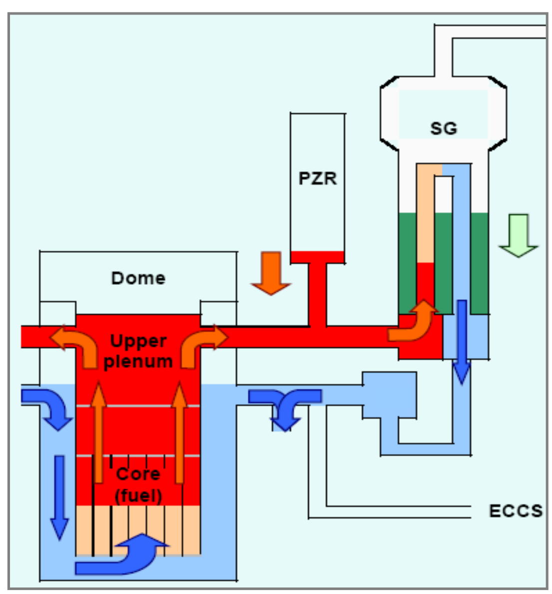
Core cooling

- Safety Injection System signal required to initiate safety injection systems
  - Low pressuriser pressure/ Low Subcooling margin (ΔPsat)/ Low loop level
- Secondary side cooling is a key requirement for EPR
  - Automatic Partial Cooldown system automatically reduces Steam Generator pressure to 60 bars using MSRT (atmospheric steam dump systems – linear temperature decrease). Necessary in EPR due to reduced head of MHSI
  - Steam Generator feed by EFWS

LOCA – Typical sequence of events

Phase 1: Single-phase depressurisation

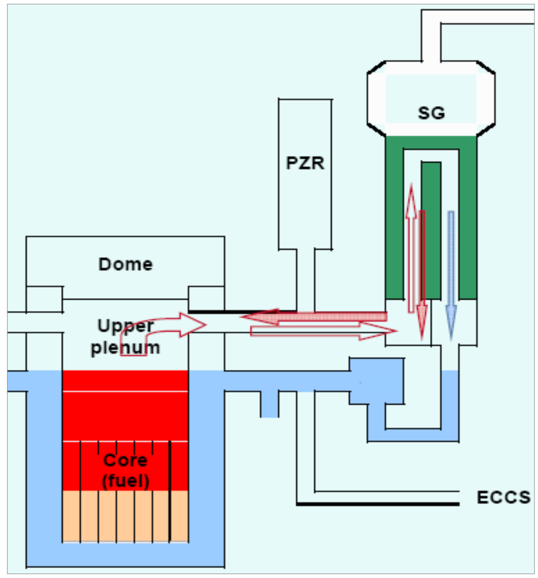
- **Break opens**
- **Pressuriser empties**
- **Primary vessel empties**
  - **PZR Pressure = MIN2 [135 bar]**
    - Reactor Trip
    - Turbine Trip
  - **PZR Pressure = MIN3 [115 bar]**
    - Automatic Partial Cooldown begins
    - Safety injection signal generated
    - EFWS Startup (in case of LOOP)
- **Natural circulation cooling**



LOCA – Typical sequence of events

Phase 2: Vaporisation and stratification

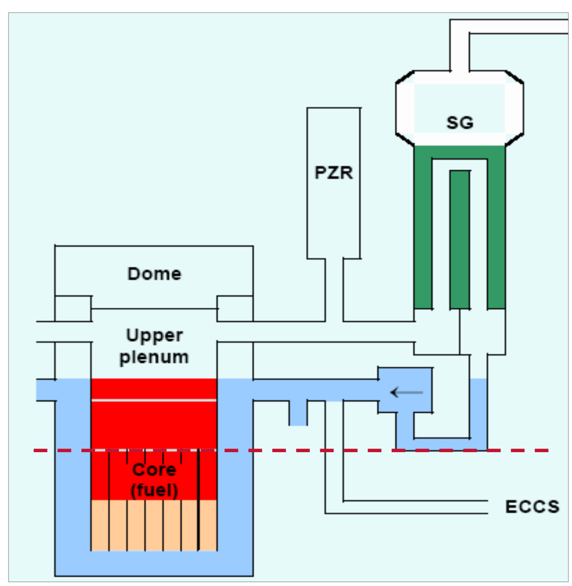
- End of natural circulation
  - SG tubes empty
- Steam condensation in SG tubes
  - Counter-current two phase flow in SG Tubes (riser section)
  - Energy removal by SGs dominates in Small LOCAs
  - Energy removal via break dominates in Large LOCAs



LOCA – Typical sequence of events

Phase 3: Manometric phase

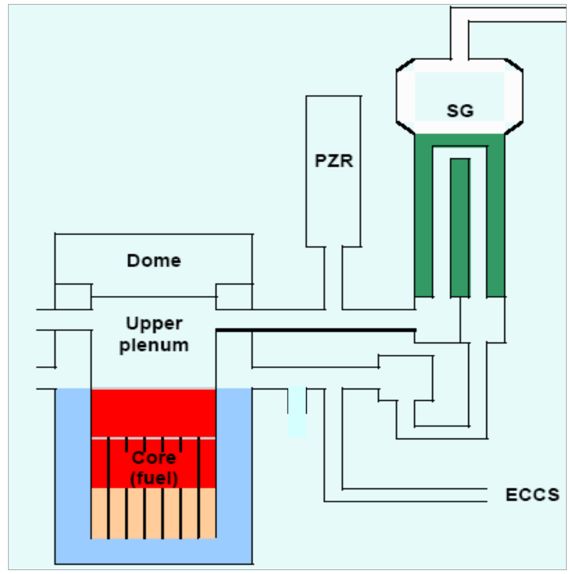
- Liquid flow through break
- Liquid trapped in the U-Legs
- Manometric balance between water level in Core and U-Leg
- Water level lower in core than downcomer
- Water level remains above top of heated core in EPR design



LOCA – Typical sequence of events

Phase 3: End of Manometric phase

- U-Leg clears of liquid
- Water level same in core and downcomer
- Steam flow through break
- Core water inventory decreases
- Primary depressurisation rate increases due to transition to steam discharge



LOCA – Typical sequence of events

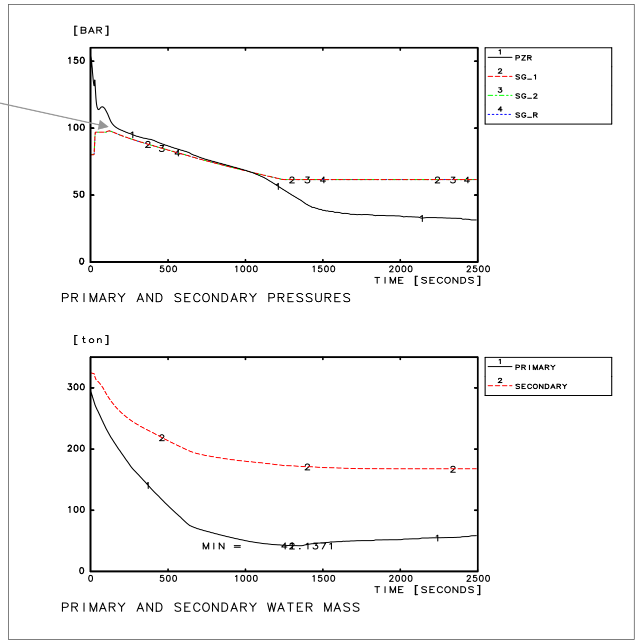
Phase 4 & 5: Core uncovery and reflood

- Core level initially decreases: break flowrate exceeds SIS injection rate. Possible core uncovery.
- Accumulator injection occurs when primary pressure falls to accumulator tank pressure
  - Core reflooding
  - Cladding temperature recovers to saturation temperature
- Long term stable cooling established using Low Head Injection system in recirculation mode (suction water drawn from In-containment Refuelling Water Storage Tank).
- In case of cold leg break, steam continues to be vented into containment. Switch to Hot Leg Injection needed to condense steam from core and prevent over-pressurisation of conatinmnet building

LOCA – Typical sequence of events

- *EPR: worst case break size = 80 cm² (DN100, 4'', 4500 MW)*

Automatic Partial Cooldown

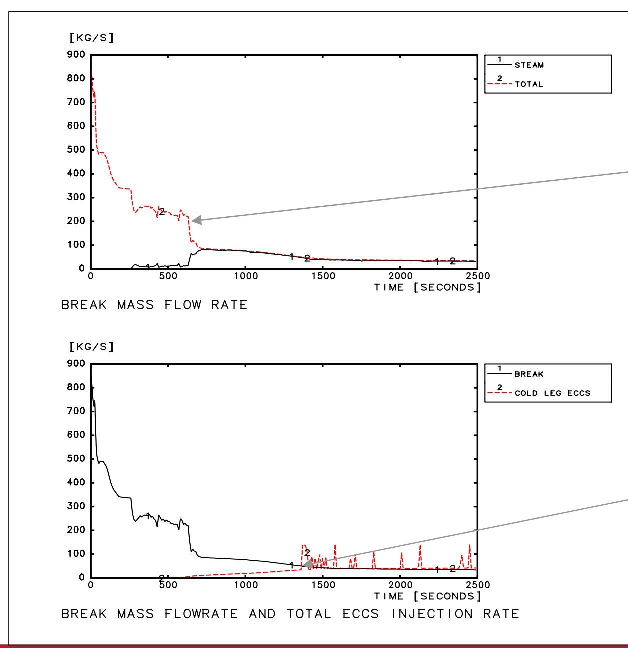


LOCA – Typical sequence of events

- *EPR: worst case break size = 80 cm² (DN100, 4'', 4500 MW)*

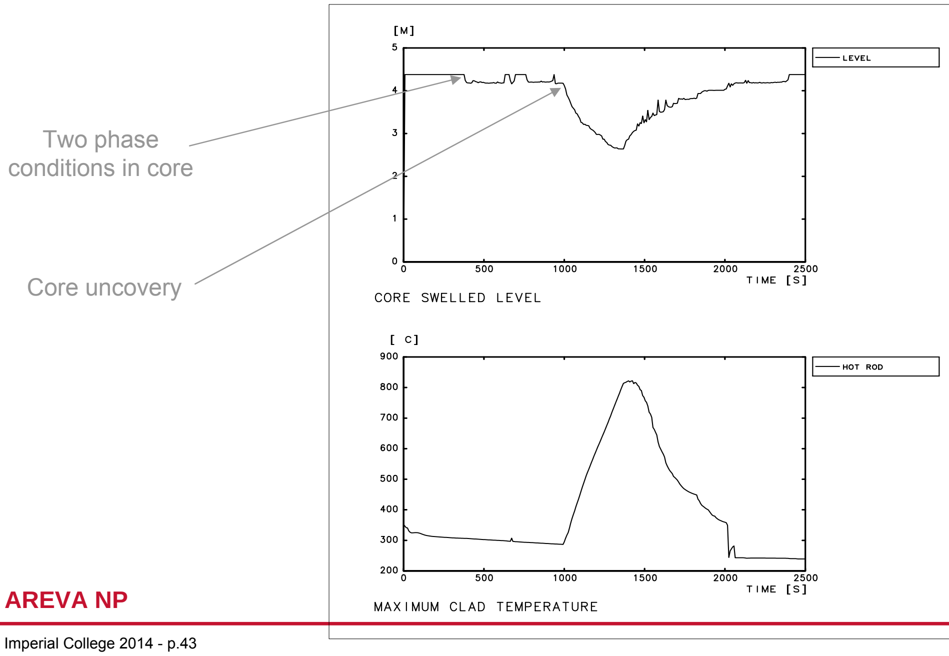
Loop seal clears

Accumulator injection



LOCA – Typical sequence of events

- ***EPR: worst case break size = 80 cm² (DN100, 4'', 4500 MW)***



LOCA – Typical sequence of events

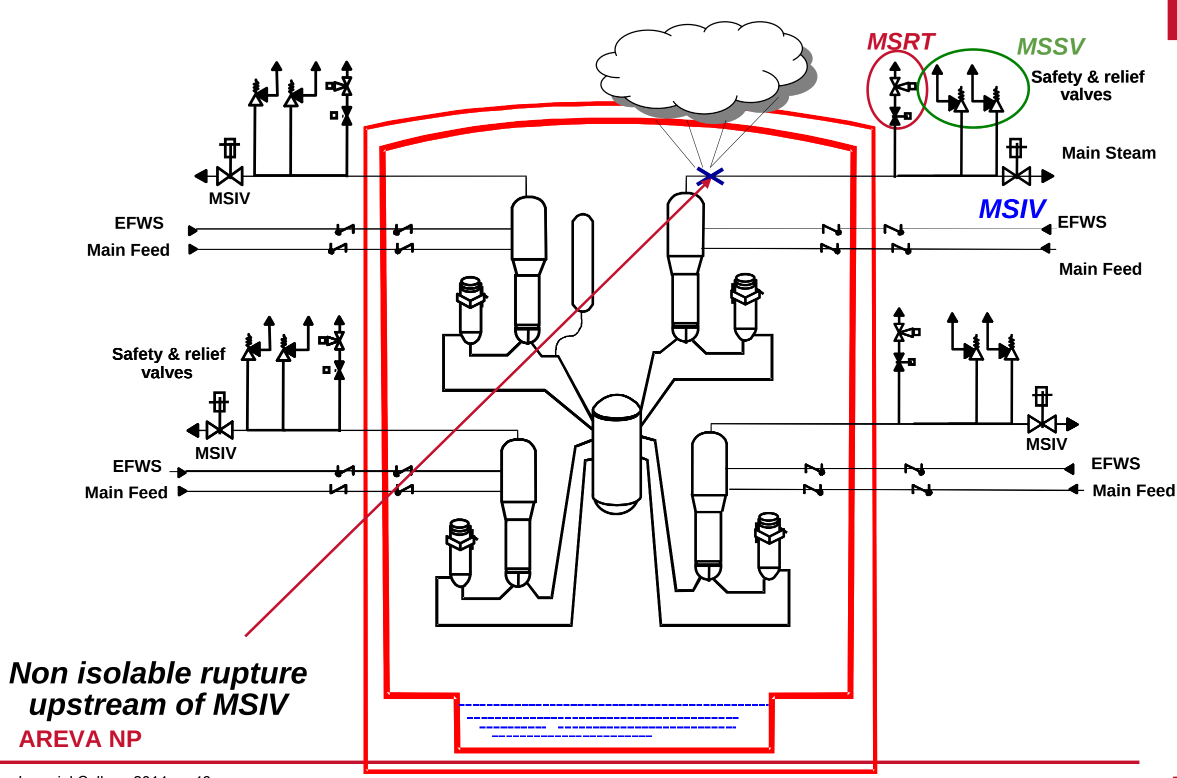
- *EPR: worst case break size = 80 cm² (DN100, 4'', 4500 MW)*

TIME (s)	EVENT
0.0	Break opening
22	PZR pressure < MIN2 (132 bar)
23	RT signal
23.3	RT (beginning rod drop), TT, RCP trip, loss of MFW flow
104	PZR pressure < MIN3 (112 bar)
105	SI and PC signal
110	Pressuriser emptying
145	Starting MHSI, LHSI pumps
543	Beginning MHSI injection in loop 2 (RCP [RCS] pressure < 85 bar)
≈ 1000	Beginning core heat-up
1033	Secondary side no more needed (RCP [RCS] pressure < SG pressure)
1366	Accumulator injection in loops 1, 2, 3 (RCP [RCS] pressure < 45 bar)
≈ 2000	End core heat-up
2500	End of calculation

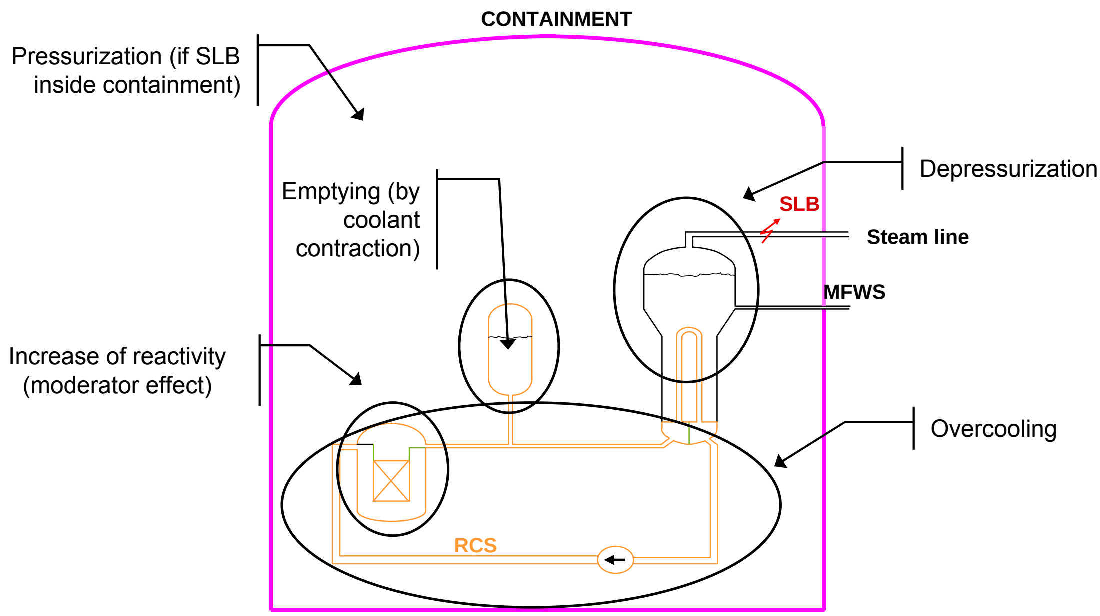
Steam Line Break (SLB) – Introduction

- **Type of accident**
  - Excessive heat removal via the steam generators (SG)
- **Initiating event**
  - Limiting case assumed - double-ended steam system line break (2A break) located upstream the main isolation valve (although high integrity argument made)
- **Limiting event treated as DBC 4: bounds the other overcooling accidents considered for EPR**
  - excessive increase in steam flow (inadvertent opening of a isolable MSB or MSRT (steam dump) valve)
  - main feedwater malfunction (MFWS), leading to a MFWS flow rate increase or a MFWS temperature decrease
  - inadvertent opening of a non-isolable MSRT (steam dump) valve or a main SG safety valve

SLB – Introduction



SLB – Key phenomena in accident



SLB – Consequences & limits challenged

- **Fuel cladding integrity**
  - **Reactivity increase in core due to moderator density increase.**
  - **Worst case single failure applied is stuck control rod in faulted core quadrant**
  - **Because of the asymmetry of the accident, high flux distortion might occur, leading to localized DNB risk.**
  - **Risk of departure from nuclear boiling in core (DNB) & fuel clad damage**

SLB – Consequences
Departure From Nucleate Boiling

DNBR = Critical heat flux / Actual flux

DNBR < 1 → Heat transfer crisis

Core inlet temperature

RCS Pressure

RCS loop flow rate

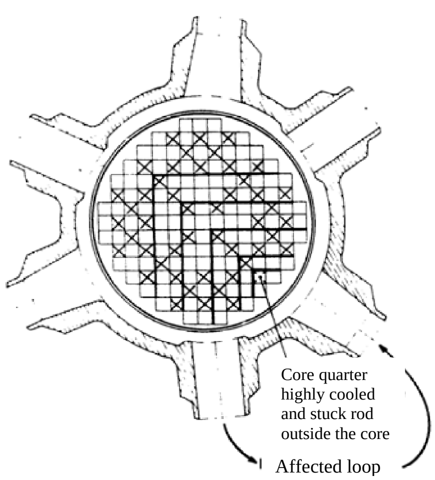
Core power

Departure from Nucleate Boiling Ratio (DNBR)

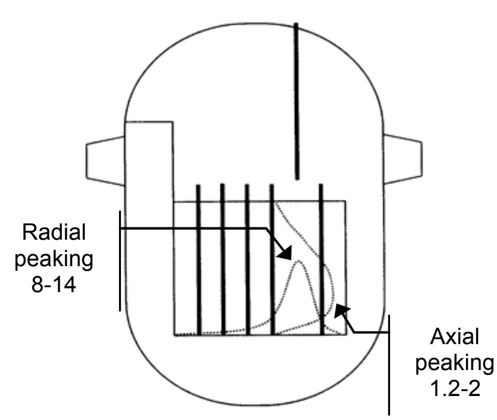
Risk of heat transfer crisis

SLB – Consequences
Flux distortion phenomenon

One loop much cooler than others



One stuck rod assumed in overcooled core quadrant



SLB – Acceptance criteria for accident analysis

- **Safety criteria for accident study**
  - **No core damage : no departure from nucleate boiling (departure from nucleate boiling ratio DNBR > 1.12)**
  - **Demonstration of the capability to reach a long term safe shutdown state**

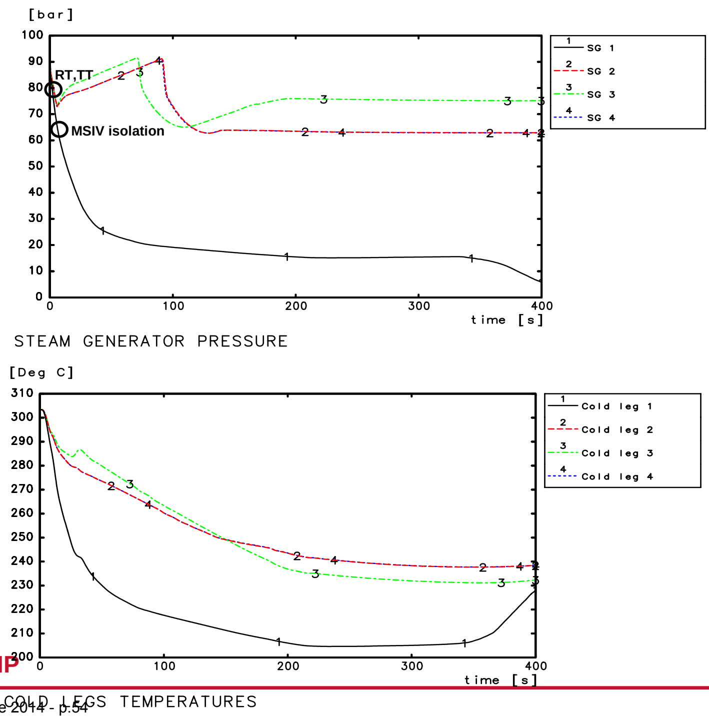
SLB – Selection of bounding assumptions (1/2)

- **Assumptions selected to maximise RCS over-cooling & reactivity increase**
  - **assume double ended guillotine (2A) break upstream the main steam isolation valves**
  - **heat removal via affected SG maximised**
    - Maximum initial SG pressure assumed (hot shutdown conditions)
    - Maximum Main Feedwater flow rate & minimum feedwater temperature assumed
    - Reactor coolant pumps assumed to continue running to maximise heat transfer to the SG

SLB – Selection of bounding assumptions (2/2)

- **Reactivity effects maximised**
  - One rod stuck in its full withdraw position located in faulted quadrant
  - Minimum initial power (10^{-9}), no decay heating
  - Minimum shutdown margin (end of life core)
  - Maximum moderator coefficient (absolute value)
  - Maximum temperature Doppler coefficient (absolute value)
  - Minimum safety injection flow rate and minimum boron concentration (assumed to be zero for short term analysis)

SLB – Typical sequence of events

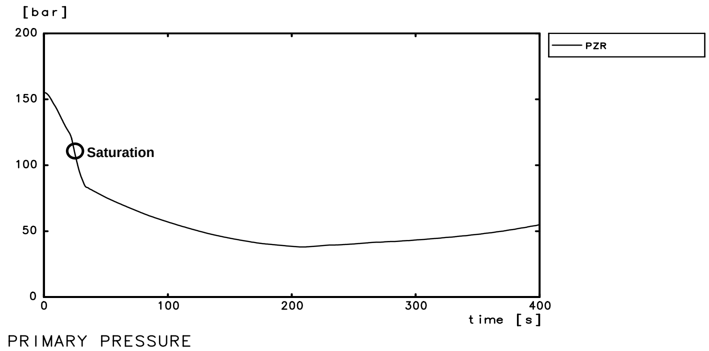


Non-isolable 2A SLB

SG depressurisation

Overcooling at core inlet

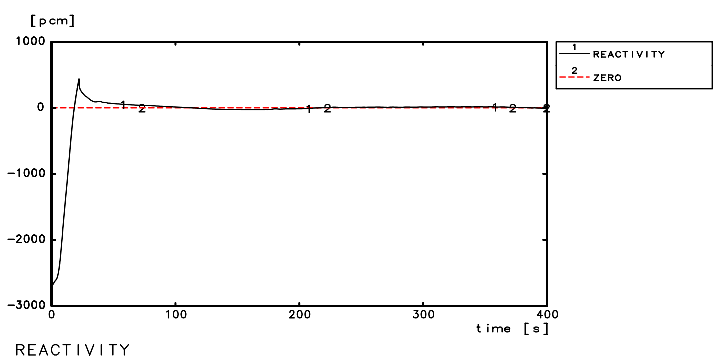
SLB – Typical sequence of events



RCS overcooling

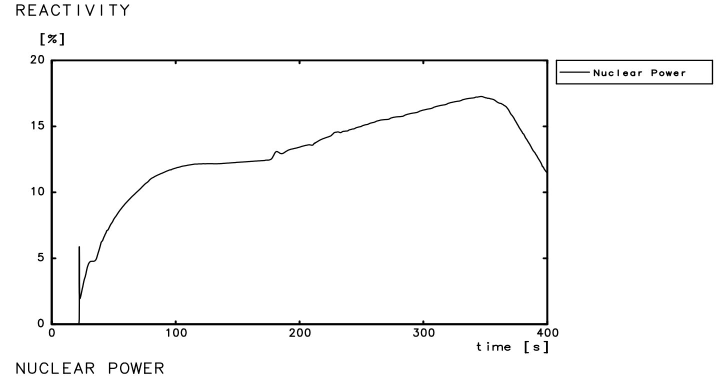
Primary pressure decrease

SLB – Typical sequence of events



Overcooling at core inlet

Reactivity increase (moderator effect)



Nuclear power generation

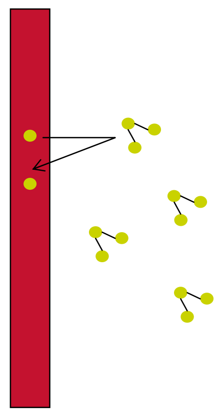
Limited by Doppler feedback effect

SLB – Increase in reactivity (1/2)

- **t = pre-criticality**

$$\Delta K = \Delta K_{mod} + \Delta K_{bore} + \Delta K_{Döppler} + \Delta K_{grappes}$$

ΔK_{mod}	ΔK_{bore}	$\Delta K_{Döppler}$		$\Delta K_{grappes}$
$\alpha_{\rho}\Delta\rho$	$\alpha_{Cb}\Delta Cb$	$\alpha_{\Delta T}\Delta T$ +	$\alpha_{\Delta Q}\Delta Q$	$\Delta K_{grappes}$
⇩	⇩	⇩	⇩	
>0*>0	=0	<0*<0	=0	
>0		**>0**		



Reactor coolant temperature decreases

> Moderation is more efficient (increase of moderator density)

Leads to the cooldown of the fuel

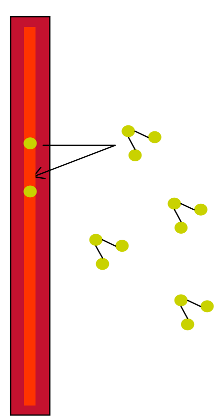
> Doppler temperature effect increases reactivity

SLB – Increase in reactivity (2/2)

- **t = post criticality**

$$\Delta K = \Delta K_{mod} + \Delta K_{bore} + \Delta K_{Döppler} + \Delta K_{grappes}$$

ΔK_{mod}	ΔK_{bore}	$\Delta K_{Döppler}$		$\Delta K_{grappes}$
$\alpha_{\rho}\Delta\rho$	$\alpha_{Cb}\Delta Cb$	$\alpha_{\Delta T}\Delta T$ +	$\alpha_{\Delta Q}\Delta Q$	$\Delta K_{grappes}$
⇩	⇩	⇩	⇩	
>0*>0	=0	<0*<0	<0*>0	
>0		>0	**<0**	

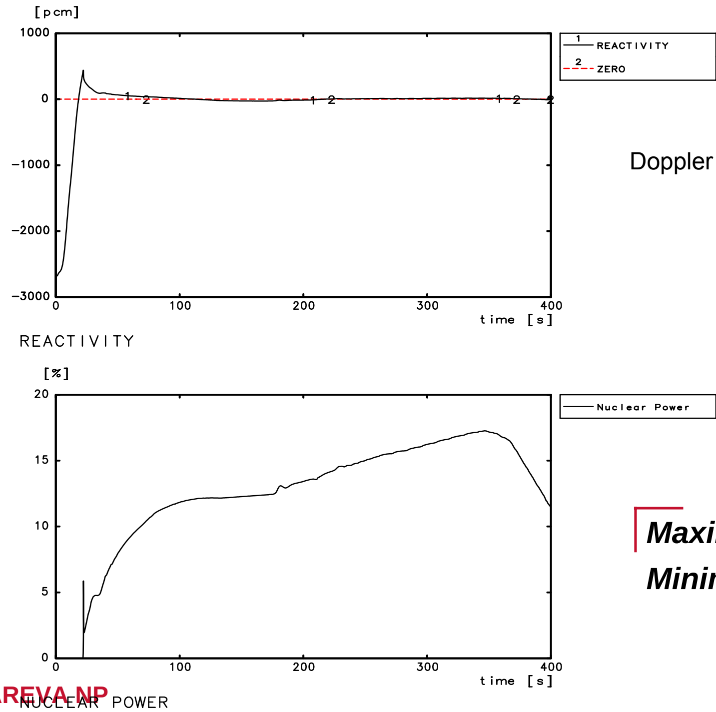


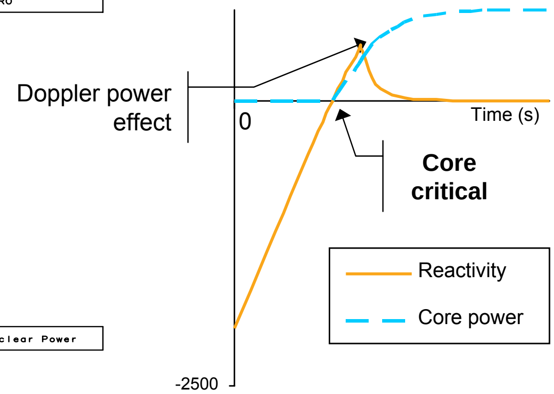
Reactor coolant temperature keeps decreasing

Fuel begins to heat up due to the core power generation

>Doppler power effect reduces reactivity

SLB – Summary of Short-term results





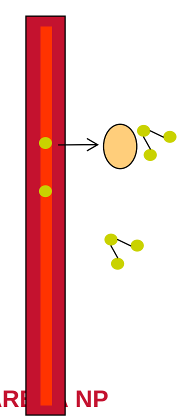
Maximum power level 17% NP

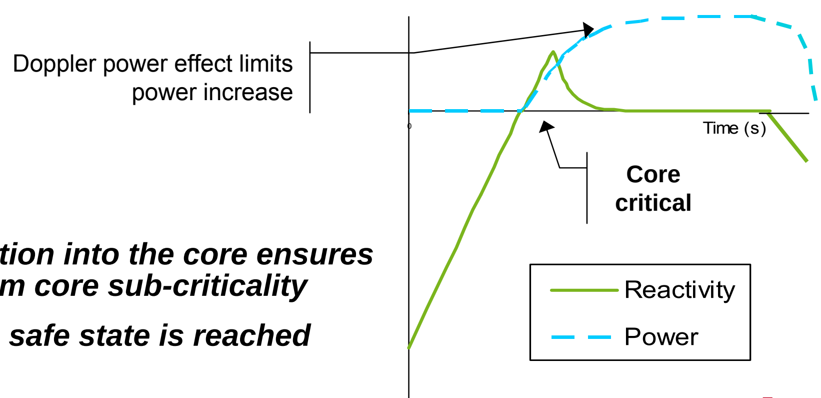
Minimum DNBR: 1.42 > criterion 1.12

SLB – Long-term results

- t = boron injection in the core (manual EBS actuation)

ΔK =	ΔK_{mod} +	ΔK_{bore} +	$\Delta K_{Döppler}$		+ $\Delta K_{grappes}$
	$\alpha_{\rho}\Delta\rho$	$\alpha_{Cb}\Delta Cb$	$\alpha_{\Delta T}\Delta T$ +	$\alpha_{\Delta Q}\Delta Q$	$\Delta K_{grappes}$
	⇩	⇩	⇩	⇩	
	>0*>0	<0*>0	<0*<0	<0*>0	
	>0	<0	>0	<0	





Boron injection into the core ensures the long-term core sub-criticality

A long term safe state is reached

Steam Generator Tube Rupture (STGR) – Introduction

Defining feature

- STGR is a Small break LOCA with bypass of the 3rd barrier (containment)

Initiating event

- Leak or complete severance of one or several SG tubes

Categorization of the transient for EPR

- DBC-3 : 2A-SGTR
- DBC-4 : 4A-SGTR

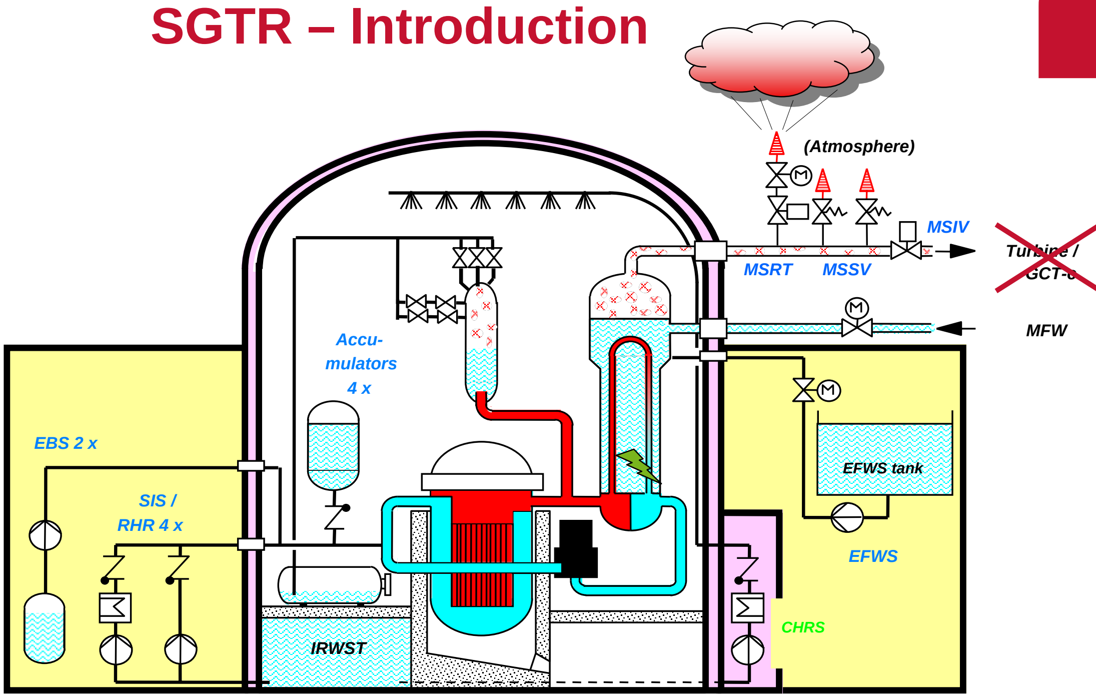
Possible causes

- Vibrations, stress corrosion cracking, foreign objects in SG

Codes used

- CATHARE & S-RELAP (coupled with NLOOP)

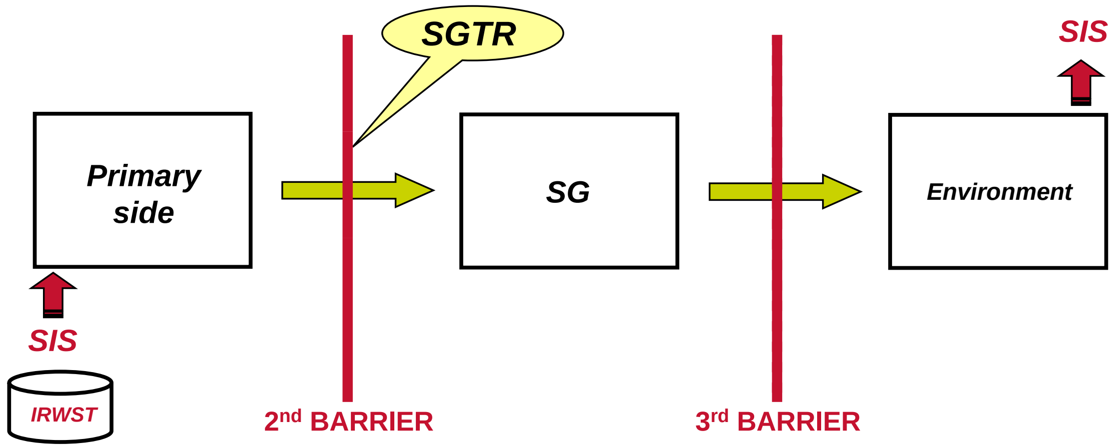
SGTR – Introduction



Risk of direct release of radioactivity to the atmosphere

SGTR – Introduction

Main risk from STGR = Fuel damage/Core melt with containment bypass



Examples : SGTR + MSRT stuck open + Primary pressure > 1 bar
⇨ IRWST drains to the atmosphere
⇨ Possible core damage with containment bypass

SGTR – Acceptance criteria in accident analysis

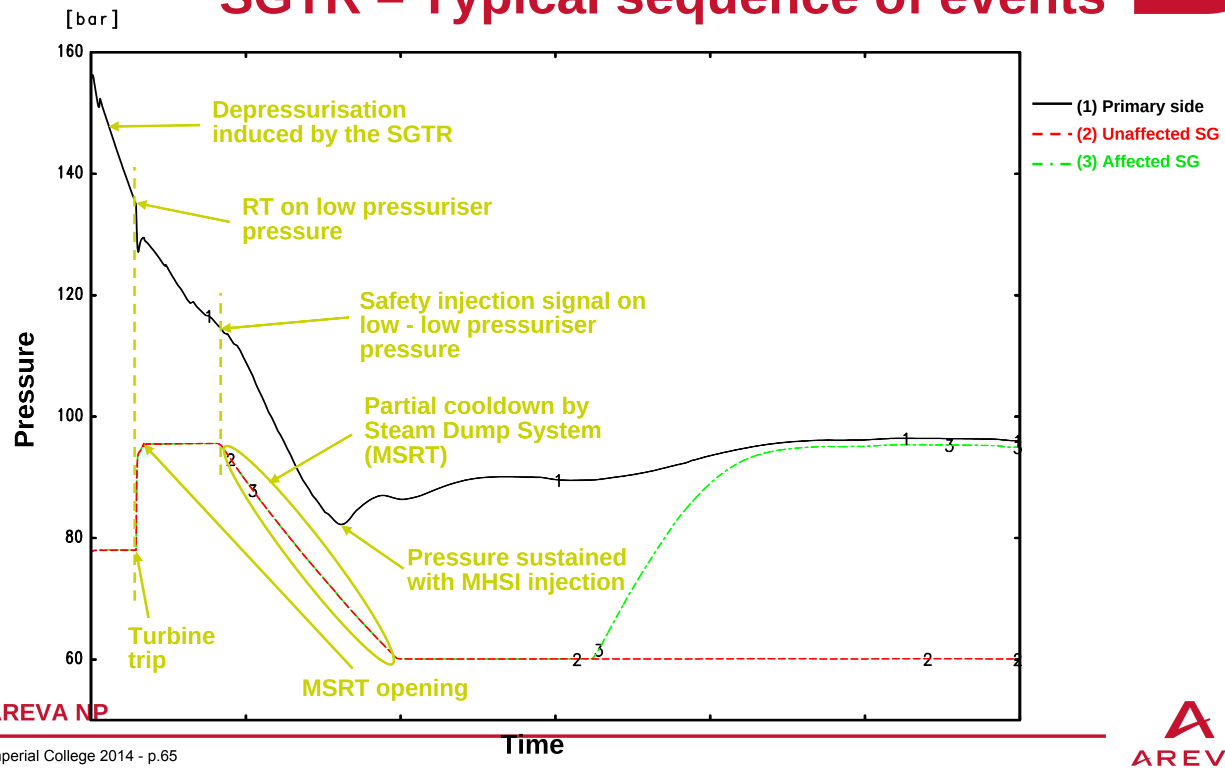
▶ Basic goals

- **no core damage (fuel cladding integrity to be preserved),**
- **no opening of SG safety valves (MSSVs) – as cannot be isolated,**
- **leak to be terminated by automatic actions before SG overfilling – avoids liquid water discharge to environment**

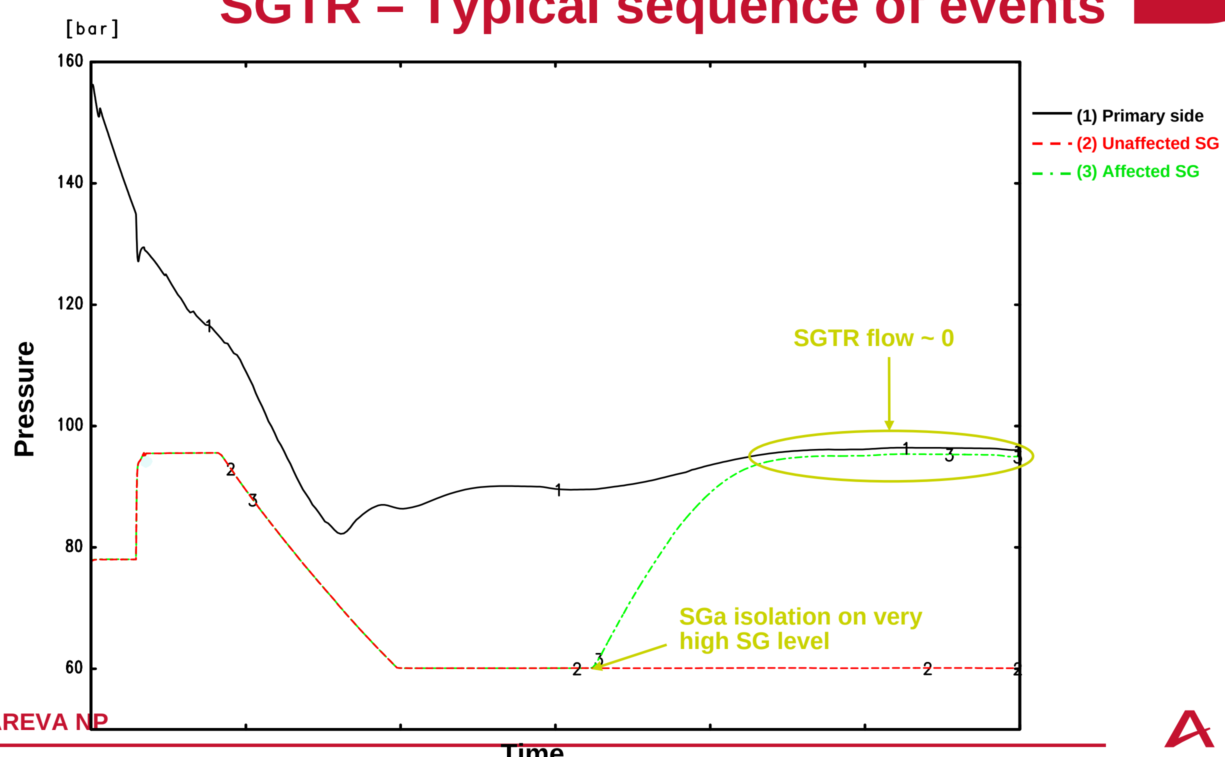
▶ EPR design deeply impacted by SGTR safety goals

- **MHSI pumps: → Delivery head pressure reduced to 85/97 bar (below MSSV set pressure)**
- **Automatic Partial cooldown of SGs:→ SG pressure 95.5 to 60 bar (T_{sat} ~ 260°C)**
- **MSSV → Opening pressure setpoint increased 105 bar abs**
  - **⇨ Shutdown margin → sub-critical core at 260°C (N-1 rods)**
  - **⇨ SG design pressure → 100 bar abs**

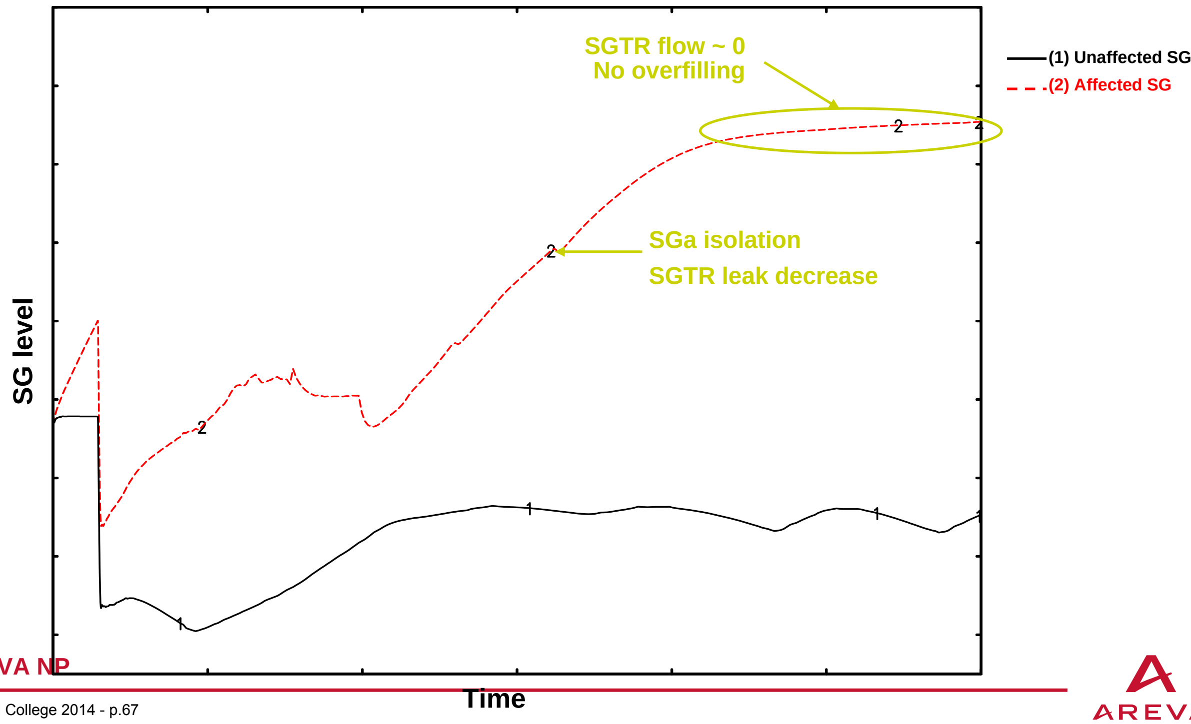
SGTR – Typical sequence of events



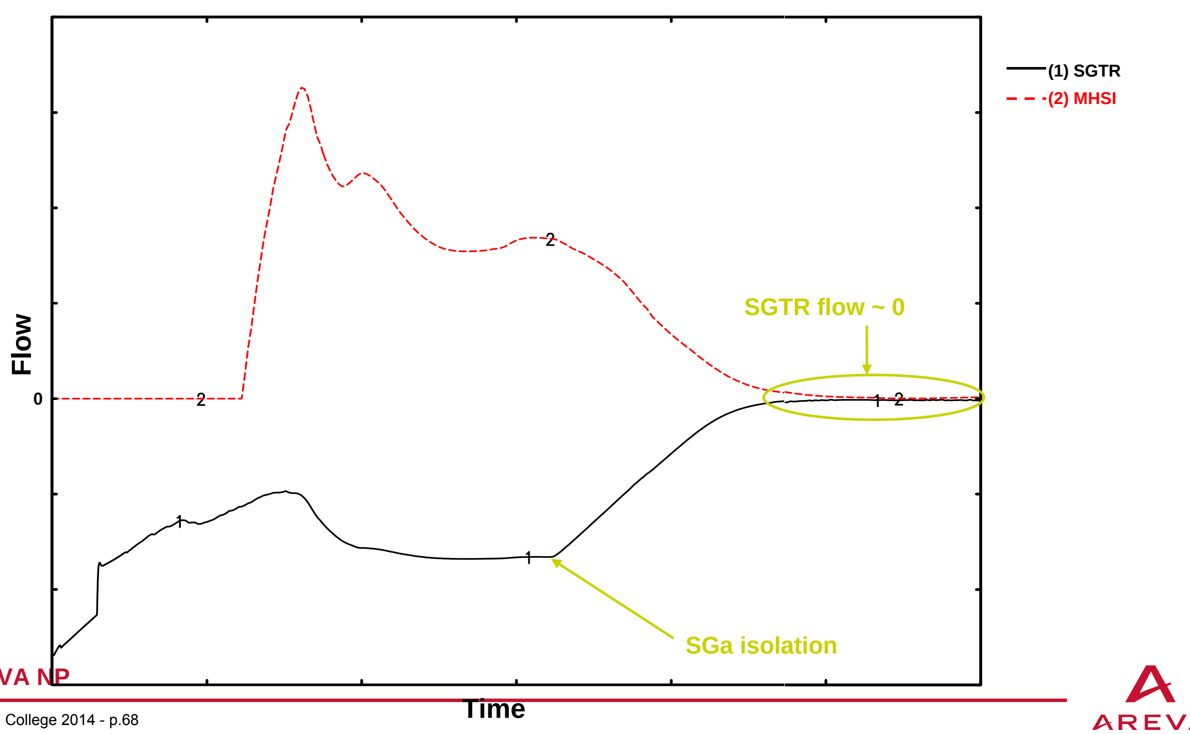
SGTR – Typical sequence of events



SGTR – Typical sequence of events



SGTR – Typical sequence of events



SGTR – Selection of the worst case

- **EPR transient (Single Tube Rupture)– MAIN RESULTS**

Summary of Results – Case 1

Parameter	Case 1
Leak termination	10070 s
Approximate contaminated steam release	118.6 tons
Total SGa VDA [MSRT] steam released	159.2 tons
Primary coolant liquid transferred to SGa	188.5 tons
Primary coolant liquid transferred to SGa prior to Turbine Trip	66.7 tons
Minimum SGa overfill margin	1.8 m

10 thermalhyraulic phenomena seen in PWR accident modelling

Flash boiling of 2-phase mixture (Steam line Break) (8)

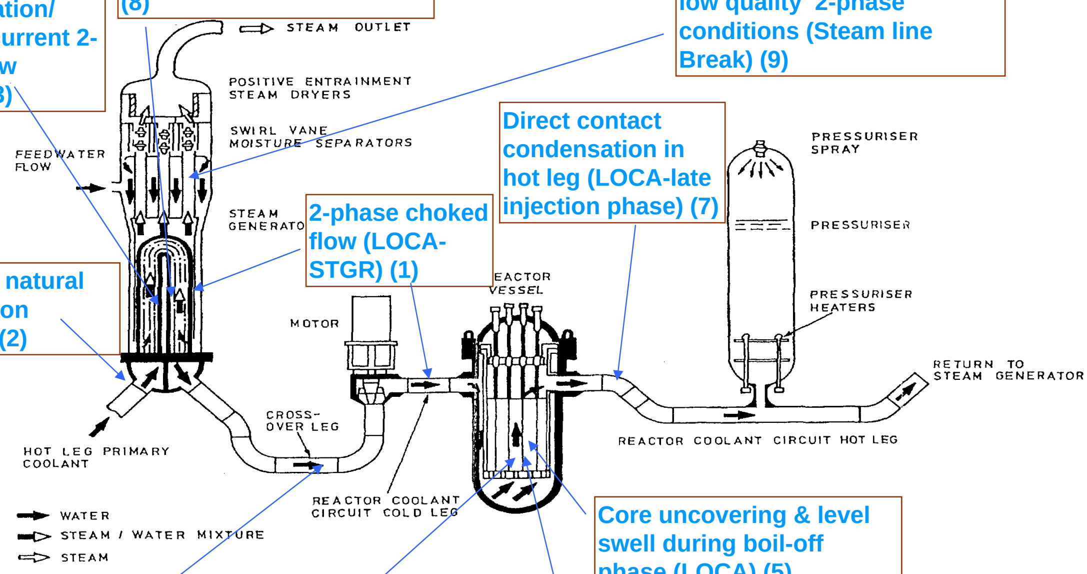
Film condensation/ counter current 2-phase flow (LOCA) (3)

Separator behaviour under low quality 2-phase conditions (Steam line Break) (9)

Direct contact condensation in hot leg (LOCA-late injection phase) (7)

2-phase choked flow (LOCA-STGR) (1)

2-phase natural circulation (LOCA) (2)



Core uncovering & level swell during boil-off phase (LOCA) (5)

Loop seal clearance/ level imbalance between core and downcomer (LOCA) (4)

Quenching and dispersed flow film boiling during reflooding (LOCA) (6)

Critical Heat Flux – (Pressurised Faults) (10)