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APPLICATION OF THE CONSERVATIVE AND BEST ESTIMATE PLUS UNCERTAINTY APPROACH TO ANALYSIS OF LB-LOCA ACCIDENT FOR VVER-1200/491 REACTOR USING RELAP 5

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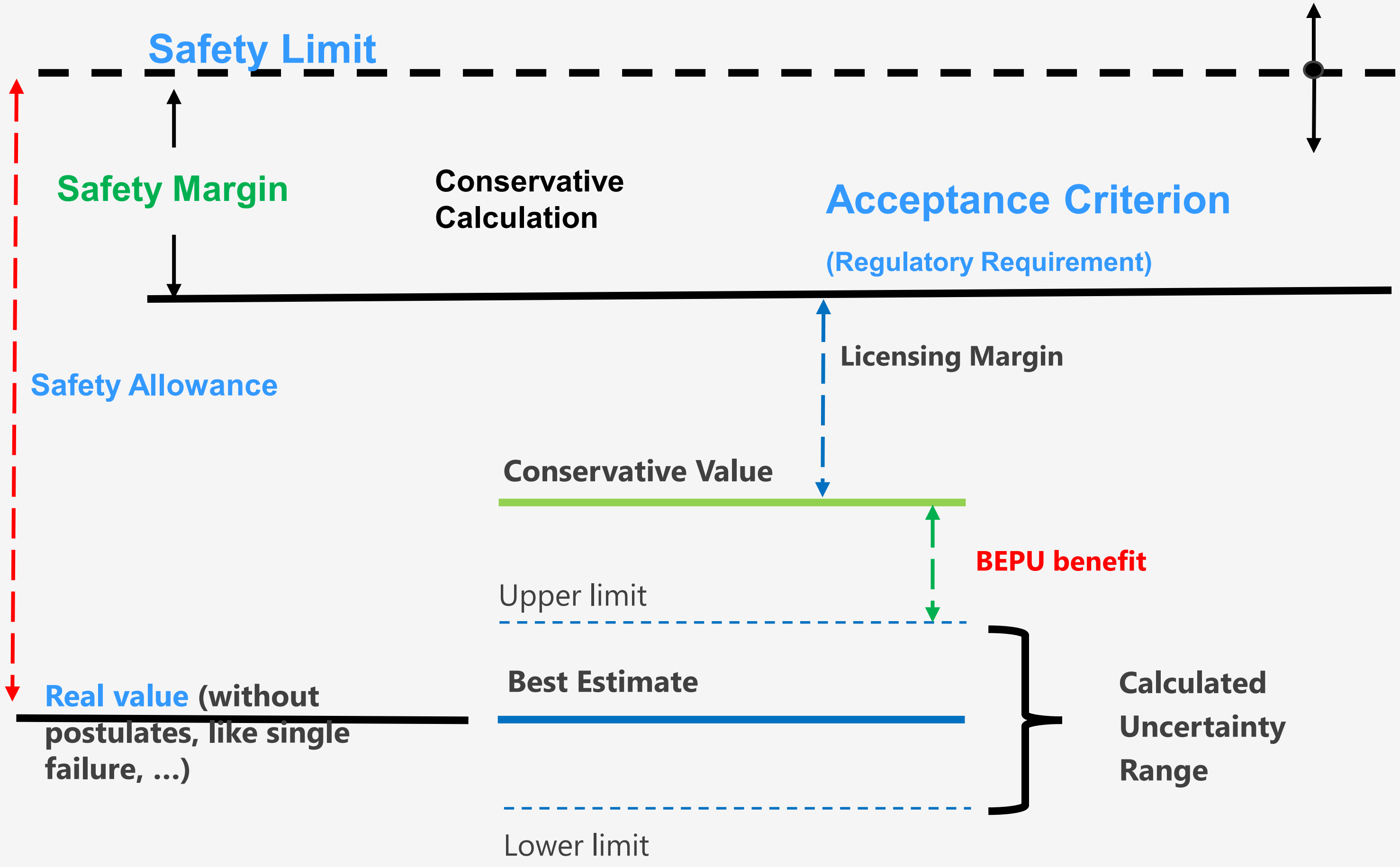
I. Introduction

Illustration of Safety Margins



Safety Limit

Safety Margin

Conservative Calculation

Acceptance Criterion

(Regulatory Requirement)

Safety Allowance

Licensing Margin

Conservative Value

BEPU benefit

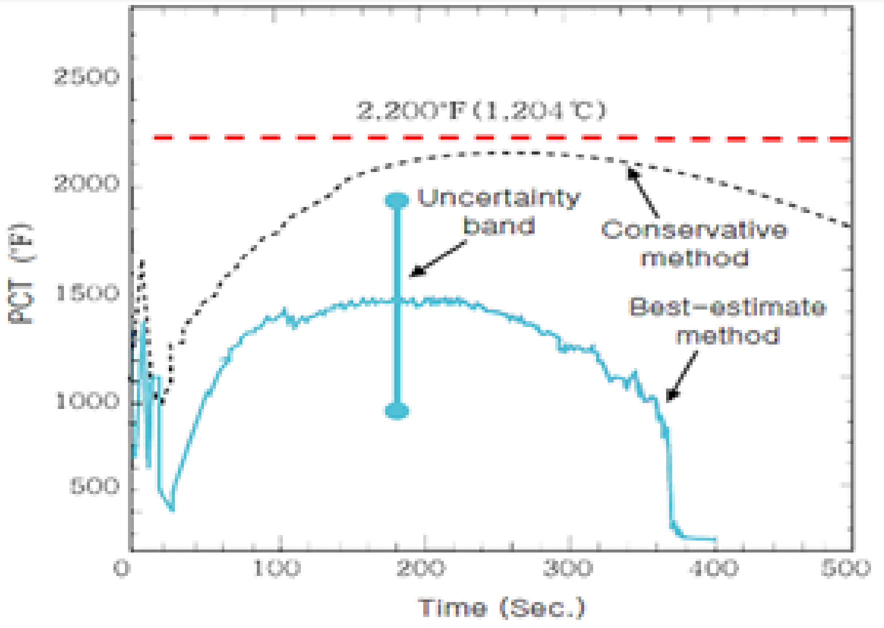
Upper limit

Best Estimate

Calculated Uncertainty Range

Real value (without postulates, like single failure, ...)

Lower limit



I. Introduction

Options for performing deterministic safety analyses (IAEA)

	Option	Computer code	Availability of systems	Initial and boundary conditions
1	Conservative	Conservative	Conservative assumptions	Conservative input data
2	Combined	Best estimate	Conservative assumptions	Conservative input data
3	Best estimate	Best estimate	Conservative assumptions	Realistic plus uncertainty; partly most unfavourable conditions
4	Risk informed	Best estimate	Conservative assumptions	Realistic input data with uncertainties

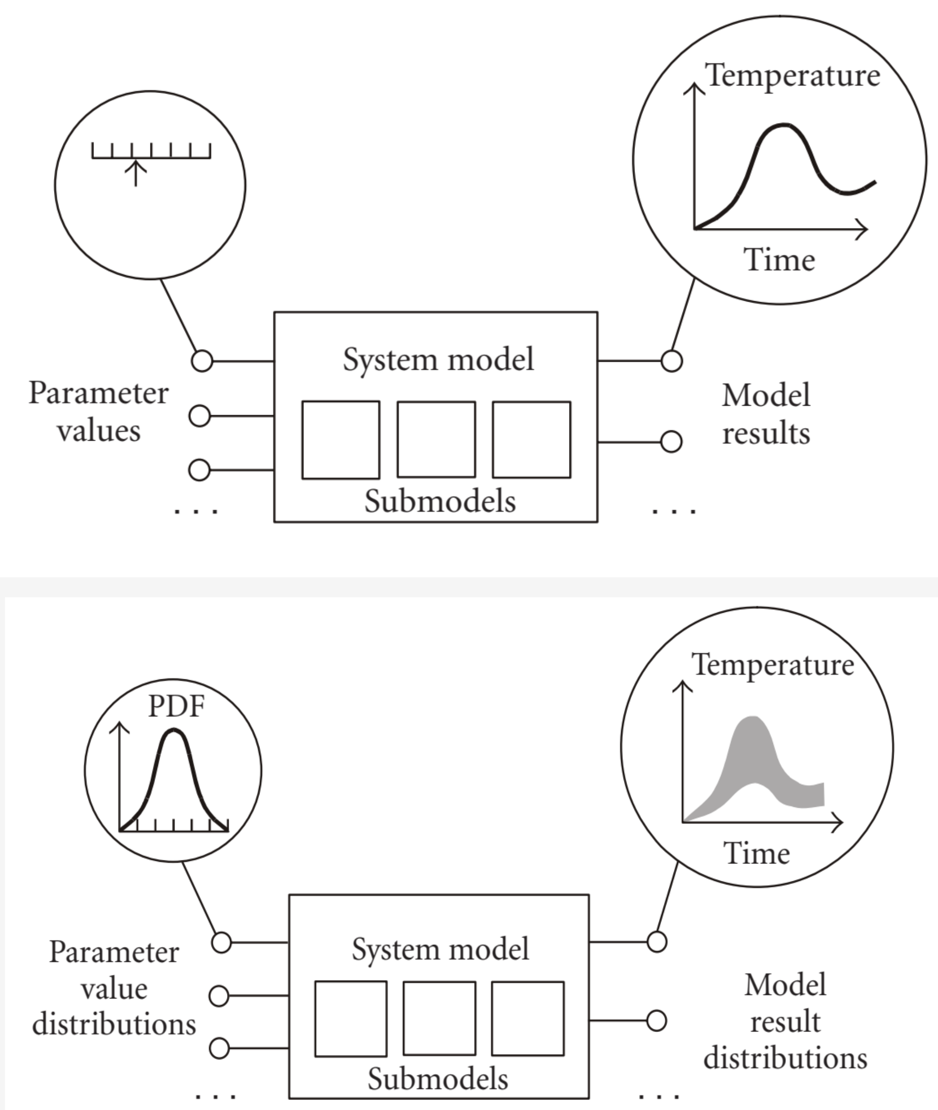
Option 1: Early days, simplify, limited capability of modelling+ knowledge of physical phenomena.
Option 2: Being used for safety analyses in many States including SAR of VVER reactor (Vietnam NPP project).
Option 3: BEPU benefit, consider in some countries (more realistic initial and boundary conditions, uncertainties should be identified).
Option 4: Not yet widely used, development of risk informed decision making.

This study perform an independent thermal-hydraulic safety analysis :
- Option 2 to compare with SAR;
- Option 3 to review whether the identification and quantification of uncertainties for initial and boundary conditions cover all uncertainty or not.

I. Introduction

GRS Method for Uncertainty and Sensitivity Evaluation

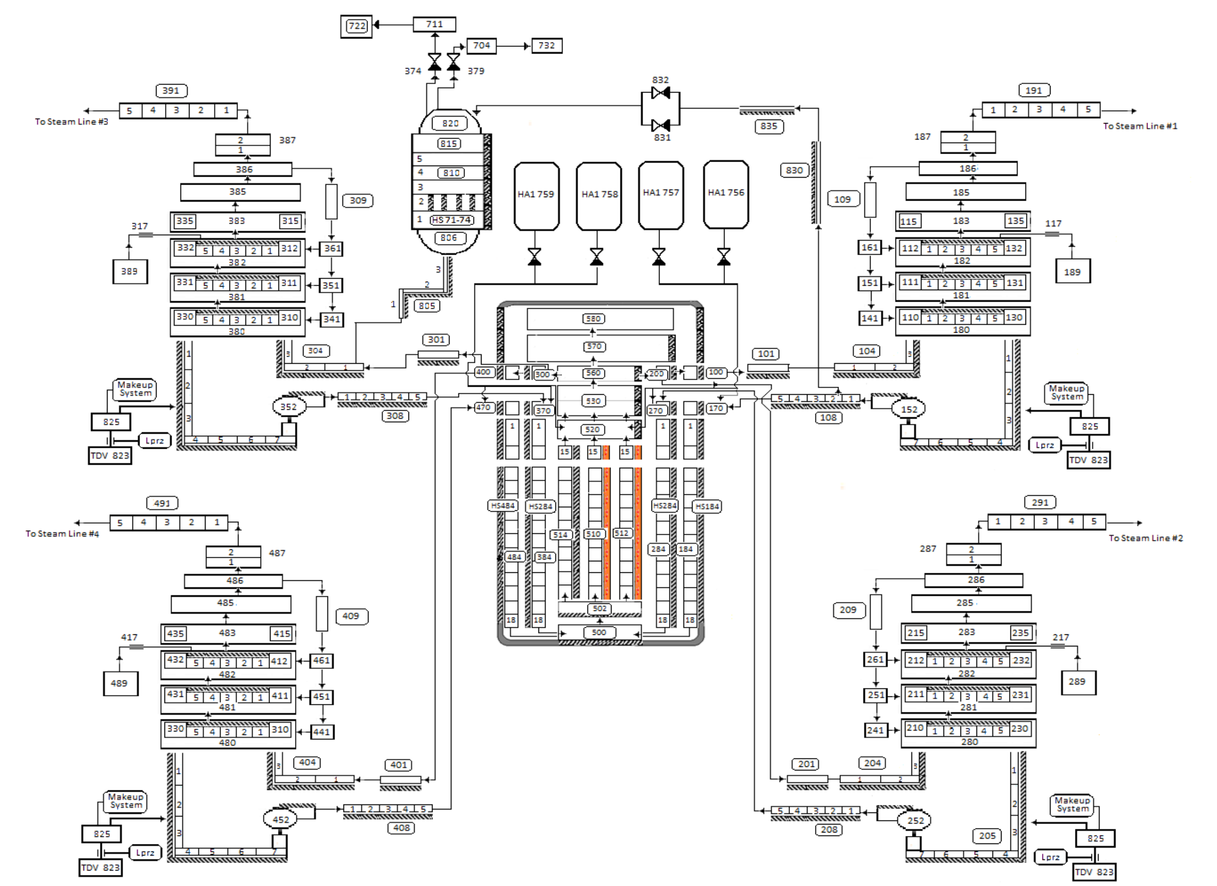
- **Identify and quantify** all potentially important parameters
  - ✓ Validation results are essential basis to quantify input uncertainties
- **Code calculations** with variation of parameter values
- Number of code calculations given by **Wilks' formula**
  - ✓ **Independent of number of uncertain parameters**
  - ✓ Dependent on tolerance limits (or -intervals) for the uncertainty statement of the code results (e.g. 95% probability content, 95% confidence limit require 59 calculations)



Consideration of input parameter value ranges instead of discrete values in the GRS method

II. Simulation Model and Assumptions

The steady-state simulation of the VVER-1200/V491 reactor was carried out as a verification of the model accuracy before transient calculations by comparison with the main design parameters in steady-state mentioned in SAR for VVER-V491



Nodalization of primary system of VVER-1200/V491

Characteristics	Design values [4,9,6]	Calculation Values	Deviation (%)
Reactor power, MW	3200	3200	0.00%
Coolant pressure at the reactor core outlet, MPa	16.2	16.22	0.12%
Coolant temperature at the reactor inlet, ° K	571	571.06	0.01%
Coolant temperature at the reactor outlet, ° K	601	602.3	0.22%
Differential pressure across the core, MPa	0.147	0.152	3.40%
Average speed of coolant in core, m/s	5.70	5.71	0.18%
Coolant flow through the reactor, m^3/h	88000	86168	2.08%
SGs water level, m	2.7 ± 0.05	2.52	6.67%
PRZ level, m	8.17	8.25	0.98%
Pressure at SG outlet, MPa	7.0	7.1	1.43%

The accuracy of the simulation of VVER-V491 given by the input deck is acceptable.

II. Simulation Model and Assumptions

2.1 Assumptions for the base case with conservative approach

A diameter of 850 mm rupture at the reactor inlet.

Availability of systems

- **First scram signal is ignored.**
- **Control protection system starts with 1.9 seconds delay.**
- **That one diesel generator (DG) is being in repair.**
- **A single failure is applied for another DG.**
- **A delay of 40.0 seconds for the boric acid delivery from ECCS pumps into the primary circuit**

Initial and boundary conditions

Table 3. The main input data for conservative and best estimate analysis

Parameter Value	Data for conservative	Design values
Reactor thermal power, MW	3328	3200
Reactor coolant flow, m^3/h	85000	88000
Coolant pressure at core outlet, MPa	16.5	16.2
Coolant temperature at reactor inlet, ^{o}C	300.2	298.2
Steam pressure in SG steam collector, MPa	7.22	7.0

II. Simulation Model and Assumptions

2.2 Assumptions for the case with best estimate plus uncertainty approach

- The uncertainty analysis based of realistic input plus uncertainty evaluation using the GRS method
- 18 important parameters for VVER-1200/V491 referring from BEMUSE program for the Zion reactor
- 100 calculation runs are required ((95%/95%) two-sided tolerance limit)

No	Parameter	Interval	Distribution
1	Containment pressure (P)	(0.85, 1.15)	Uniform
2	Initial core power	(0.98; 1.02)	Normal
3	Peaking factor (power of the hot rod)	(0.95; 1.05)	Normal
4	Hot gap size (whole core except hot rod)	(0.8; 1.2)	Normal
5	Hot gap size (hot rod)	(0.8; 1.2)	Normal
6	Power after scram	(0.92; 1.080	Normal
7	UO2 conductivity	(0.9, 1.1) if T_{fuel} <2000 ^{o}K (0.8,1.2) if T_{fuel} >2000 ^{o}K	Normal
8	UO2 specific heat	(0.98, 1.02) if T_{fuel} <1800^{o}K (0.87,1.13) if T_{fuel} >1800 ^{o}K	Normal
9	Pump - Rotation speed after break for intact loops	(0.98; 1.02)	Normal
10	Pump- Rotation speed after break for broken loop	(0.9; 1.1)	Normal
11	Initial accumulator pressure	(-0.2; +0.2) MPa	Normal
12	Friction form loss in the accumulator line	(0.5; 2)	Log-normal
13	Accumulators initial liquid temperature	(-10; +10) °C	Normal
14	Flow characteristic of LPIS	(0.95 ; 1.05)	Normal
15	Initial level	(-10; +10) cm	Normal
16	Initial pressure	(-0.1; +0.1) MPa	Normal
17	Friction form loss in the surge line	(0.5; 2)	Log-normal
18	Initial intact loop mass flow rate (primary side)	(0.96; 1.04)	Normal

III. Results and Discussion

3.1. Accident progression during LOCA accident in the base case

- The SCRAM signal is recorded at 0.036 s but it is ignored
- CPS absorber rods start falling (the second signal) at 1.9s
- ECCS pumps start delivering boric acid delayed operation until 40 s since the plant was disc.onnected

Event (time in second)	Base case	SAR [2]
Initiation of LOCA	0.000	0.000
The SCRAM signal starts generating	0.040	0.036
Set point to enable the ECCS pumps	0.040	0.040
Main generator stop valve closes	0.700	0.600
The SCRAM signal starts generating	1.605	1.536
CPS absorber rods start falling (the second signal)	2.000	1.900
ECCS reservoirs start delivering boric acid	7.200	6.600
ECCS pumps start delivering boric acid.	40.201	40.000
ECCS reservoirs finish delivering boric acid	65.600	64.800
The end of the calculation	500.000	500.000

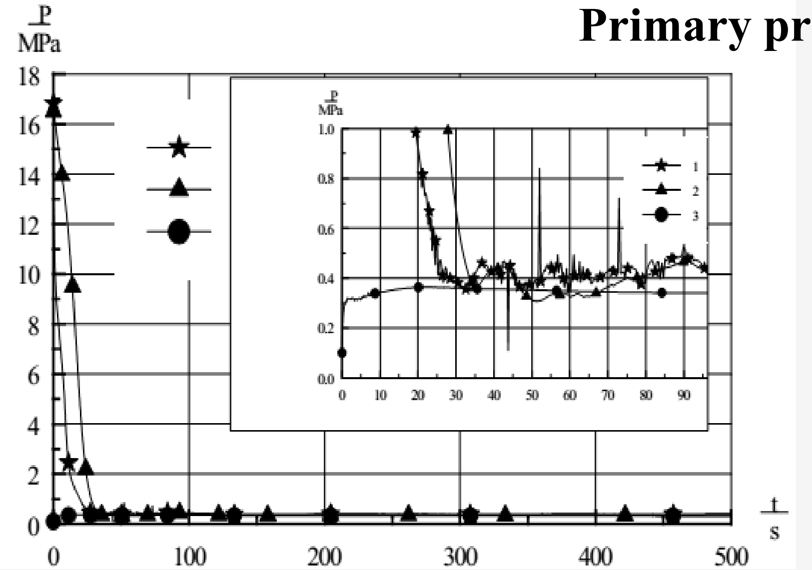
III. Results and Discussion

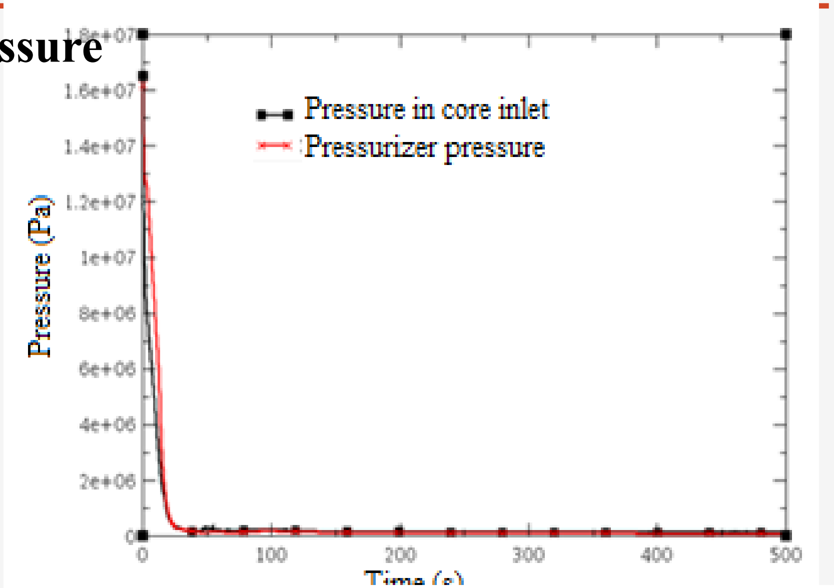
SAR

Calculated

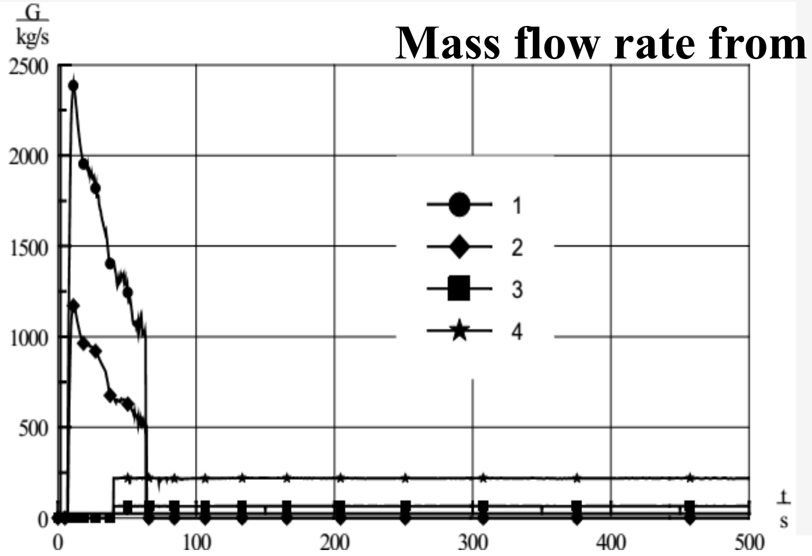
3.1. Accident progression during LOCA accident in the base case

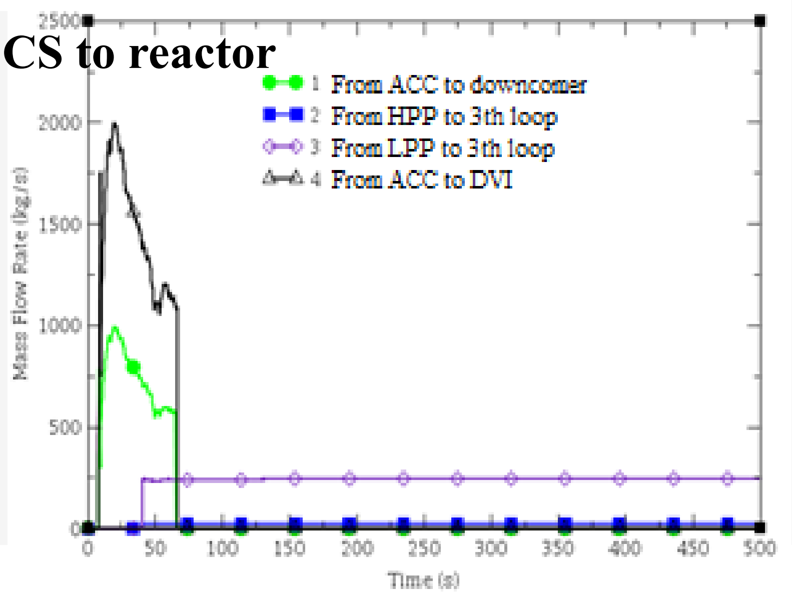
Primary pressure





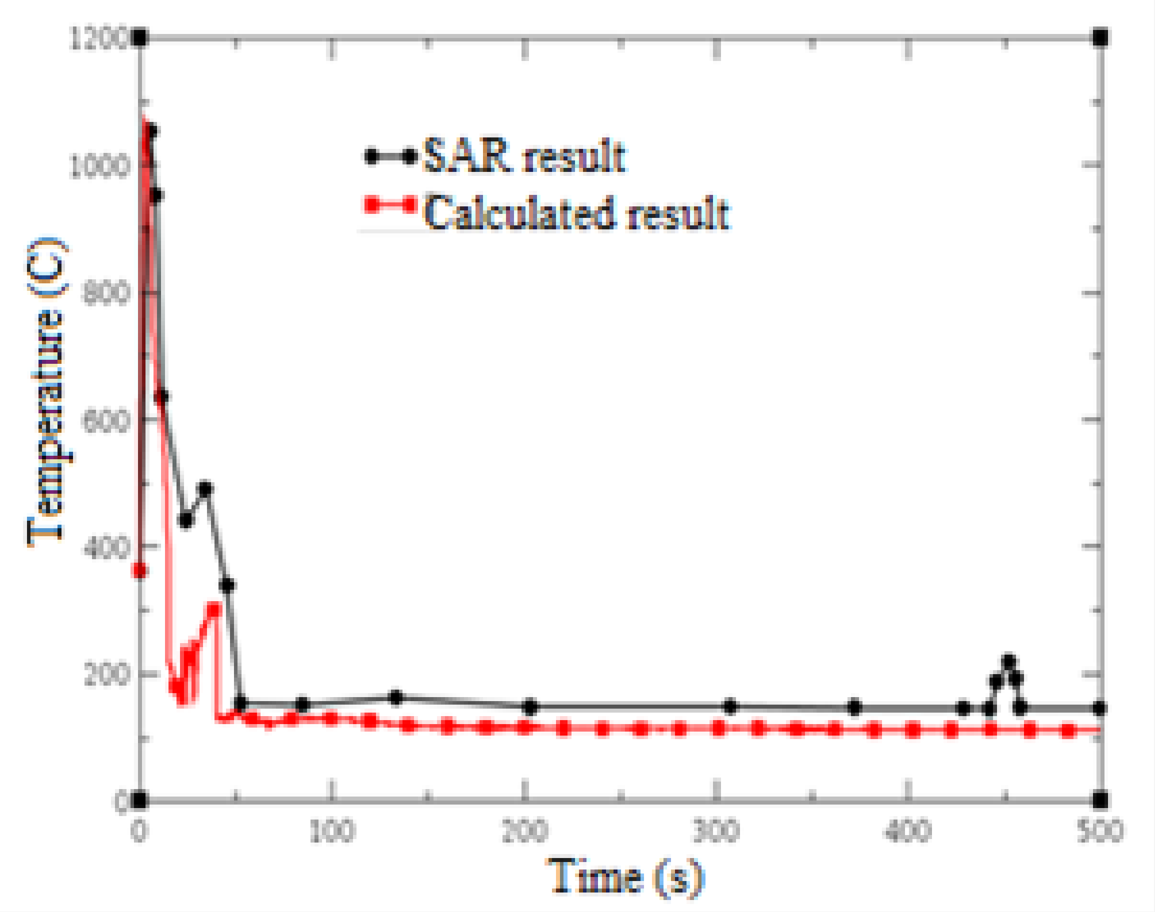
Mass flow rate from ECCS to reactor



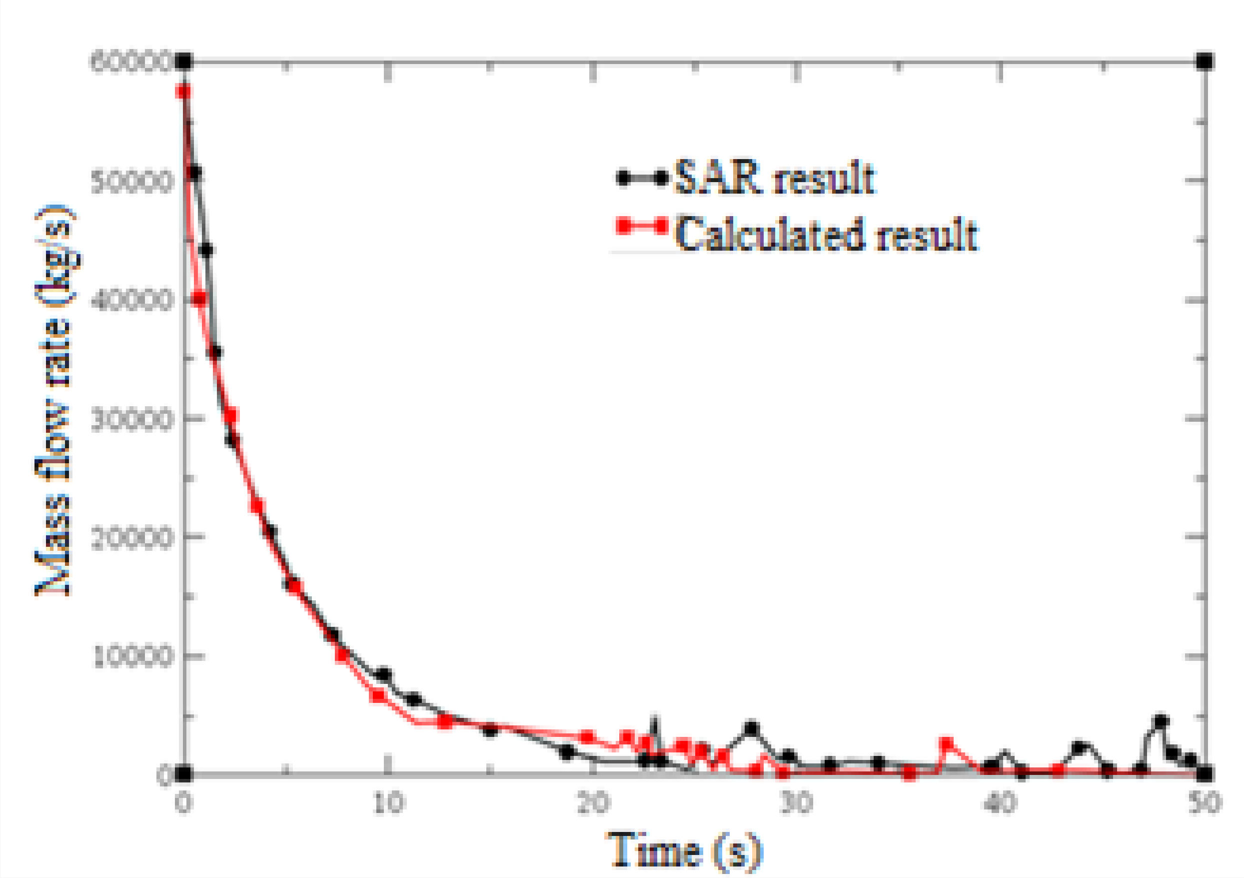


III. Results and Discussion

3.1. Accident progression during LOCA accident in the base case



Peak Cladding Temperature



Coolant mass flow rate break out

III. Results and Discussion

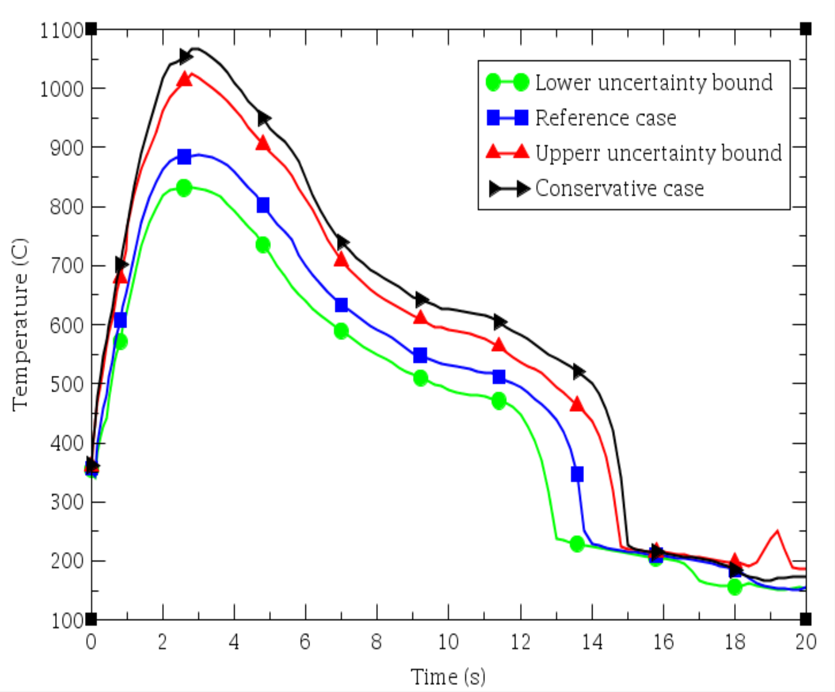
3.2. Peak cladding temperature analysis during LB-LOCA in case with Best estimate plus uncertainty approach

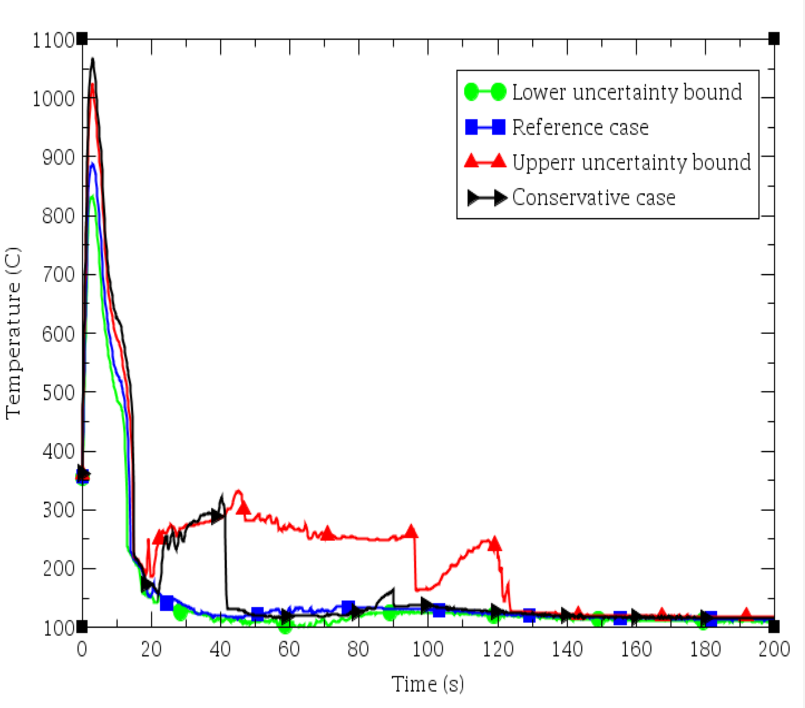
- Using conservative assumptions for the availability of the system

- Realistic input data for initial and boundary conditions

- 100 calculation runs to obtain (95%/95%) two-sided tolerance limit from best estimate case

Conservative PCT calculation vs Best estimate PCT





- The conservative PCT calculation covers all curves in 14 seconds
- The conservative PCT calculation from SAR is not covering of upper boundary of uncertainty in the whole 500 seconds of the transient

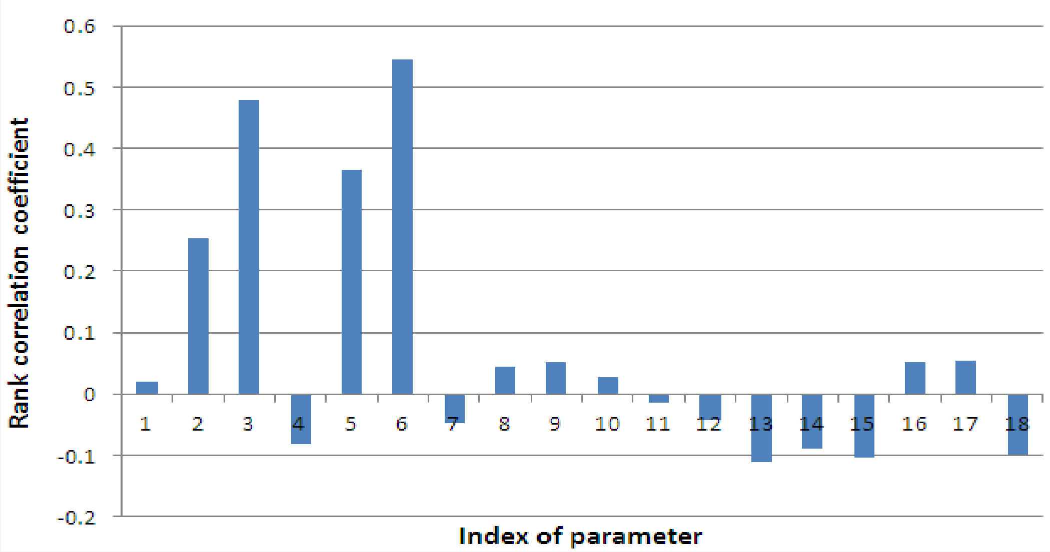
III. Results and Discussion

The most sensitive parameters

3.2. Sensitivity analysis

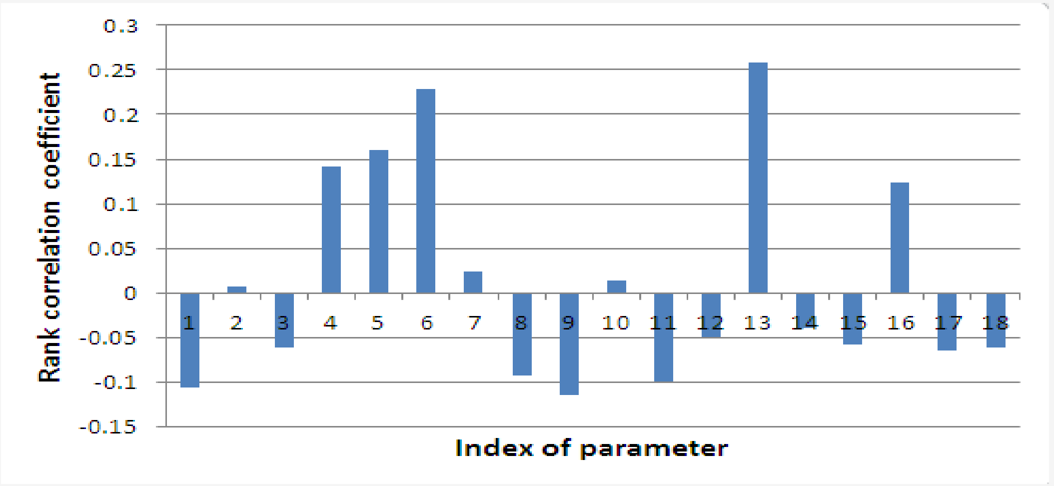
- The Spearmen rank correlation coefficient [10] with a range of values varying from -1 to +1 is used to measure the sensitivity of 18 input parameters

- The sensitivity study is performed with maximum cladding temperature in two stages of studied LB-LOCA: **Blowdown** and **reflood**



Blowdown stage

6. Power After Scram;
3. Peaking Factor;
5. Hot Gap Size;
2. Initial Core Power.



Reflood stage

13. Accumulators Initial Liquid Temperature;
6. Power After Scram;
5. Hot Gap Size.

IV. Conclusions

- There are four different options of deterministic safety analyses for a nuclear power plant predict the response to postulated initiating events : (1) Conservative, (2) Combined, (3) Best estimate and (4) Risk informed.
- This study perform an independent thermal-hydraulic safety analysis : option 2 to compare with SAR; option 3 with GRS method to review whether the identification and quantification of uncertainties for initial and boundary conditions cover all uncertainty or not.
- The fuel cladding temperature behavior between the calculated result and SAR’s results is similar.
- The upper boundary curve of uncertainty for PCT generated from best estimate calculation plus uncertainty evaluation using RELAP5 is not covered fully by conservative calculation from both codes RELAP5 and DINAMIKA-97. This issue leads to discuss whether the conservative assumption given by SAR is compliant with IAEA Specific Safety Guide on Deterministic Safety Analysis for nuclear power plants or not.
- The sensitivity analysis results show that the power after scram, the peaking factor, the accumulator's initial liquid temperature, the hot gap size and the initial core power are the most sensitive parameter affecting the PCT.

Thank you for your attention!