TYPICAL STEAM GENERATOR TUBE RUPTURE (SGTR) EFFECT ON THERMO-HYDRAULIC PARAMETERS OF VVER-1000 PRIMARY LOOP

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Objectives and content of the presentation

- Introduction
- VVER 1000 Nuclear Power Plant Description
- Description of the SGTR scenario
- RELAP5/MOD3.2 Model
- Initial Conditions
- Transient scenario
- Results and discussion
- Conclusions

Investigation of a steam generator tube rupture accident

Introduction

In the operation of nuclear power plants, it is very important to evaluate different accident scenarios in actual plant conditions. One of them is Steam Generator Tube Rupture (SGTR) in the field of nuclear safety.

The purpose of this research is to estimate the variation of thermo-hydraulics parameters in a primary loop under SGTR accidents in a VVER-1000 nuclear power plant.

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It means that the operational data from Bushehr NPP have been used for the purpose of assessing how RELAP5 model compares against the plant data.

VVER 1000 Nuclear Power Plant Description

the reference power plant is Bushehr NPP site.

Characteristics :

- VVER-1000 / Model V320
- Thermal power, 3000 MW
- Four primary loops
- One turbine generator
- 1000 MW electric power
- 850 mm Inlet/outlet nozzles
- each loop includes: one main MCP and SG

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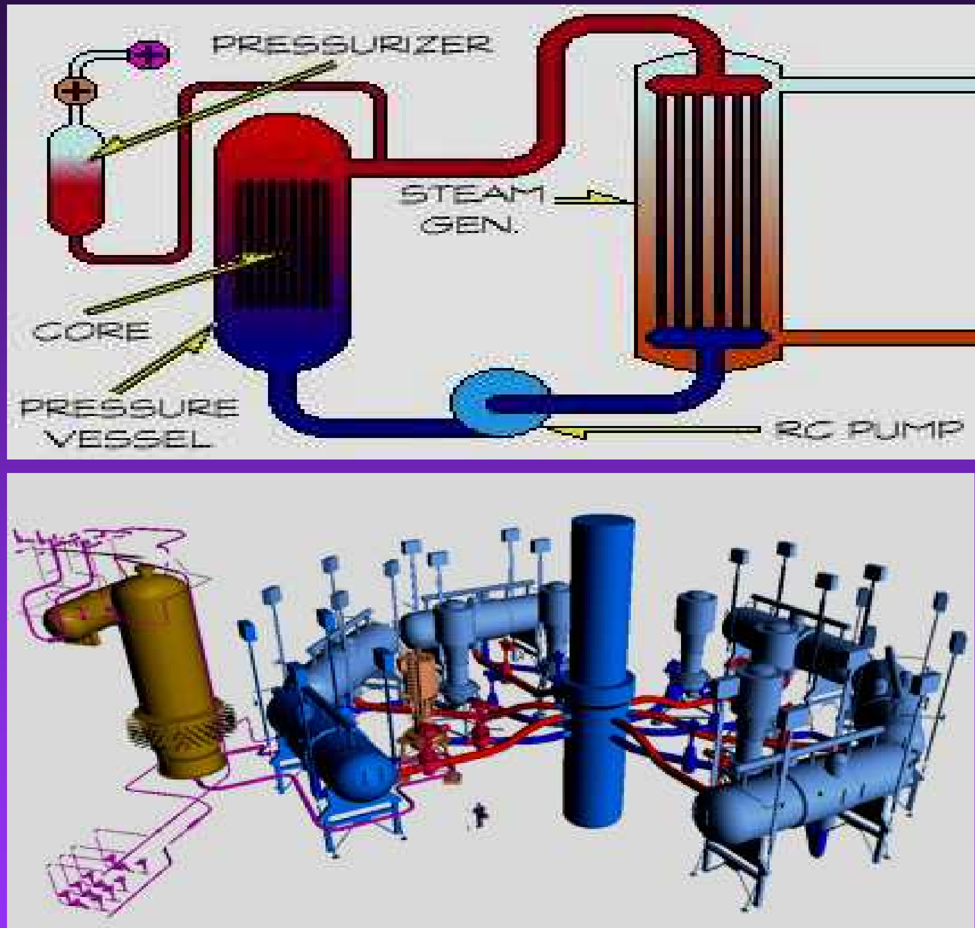

Fig A. General view of VVER-1000 primary loop

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Description of the SGTR scenario

Accident Scenario:

- possible failures of the safety systems
- loss of power to the power unit auxiliaries
- failure of two diesel generators
  - emergency feed-water system
  - additional boron injection system
  - high-pressure emergency boron injection

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The most conservative version of damage is supposed as an instantaneous break with an equivalent diameter of **100 mm** of steam generator cold collector in the area of lower row of heat exchanging tubes is created.

RELAP5/MOD3.2 Model

The Relap5 Model was developed for analysis of:

- operational occurrences
- abnormal events
- design basis scenarios

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The systems were modeled by Relap5:

- reactor core
- reactor vessel
- MCPs
- SGs
- PRZ
- steam lines
- main steam header
- emergency protection system
- make up system
- safety injection system
- steam dumping devices
- main feed water system

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The reactor vessel model includes:

- downcomer
- lower plenum
- outlet plenum

The pressurizer (PRZ) system includes:

- heaters
- spray
- pressurizer relief capability

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The safety system includes:

- accumulators
- high and low pressure injection systems
- reactor scram system

The steam generators system include:

- heat exchanger tubes
- steam dryer
- steam line

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RELAP5 heat structure components are:

- fuel rods
- vessel structural internals
  - core barrel
  - core baffle
  - lower and upper plates
  - protective tube block
  - the reactor vessel

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- Steam generator heat structure
  - hot header wall
  - cold header wall
  - heat transfer tubes

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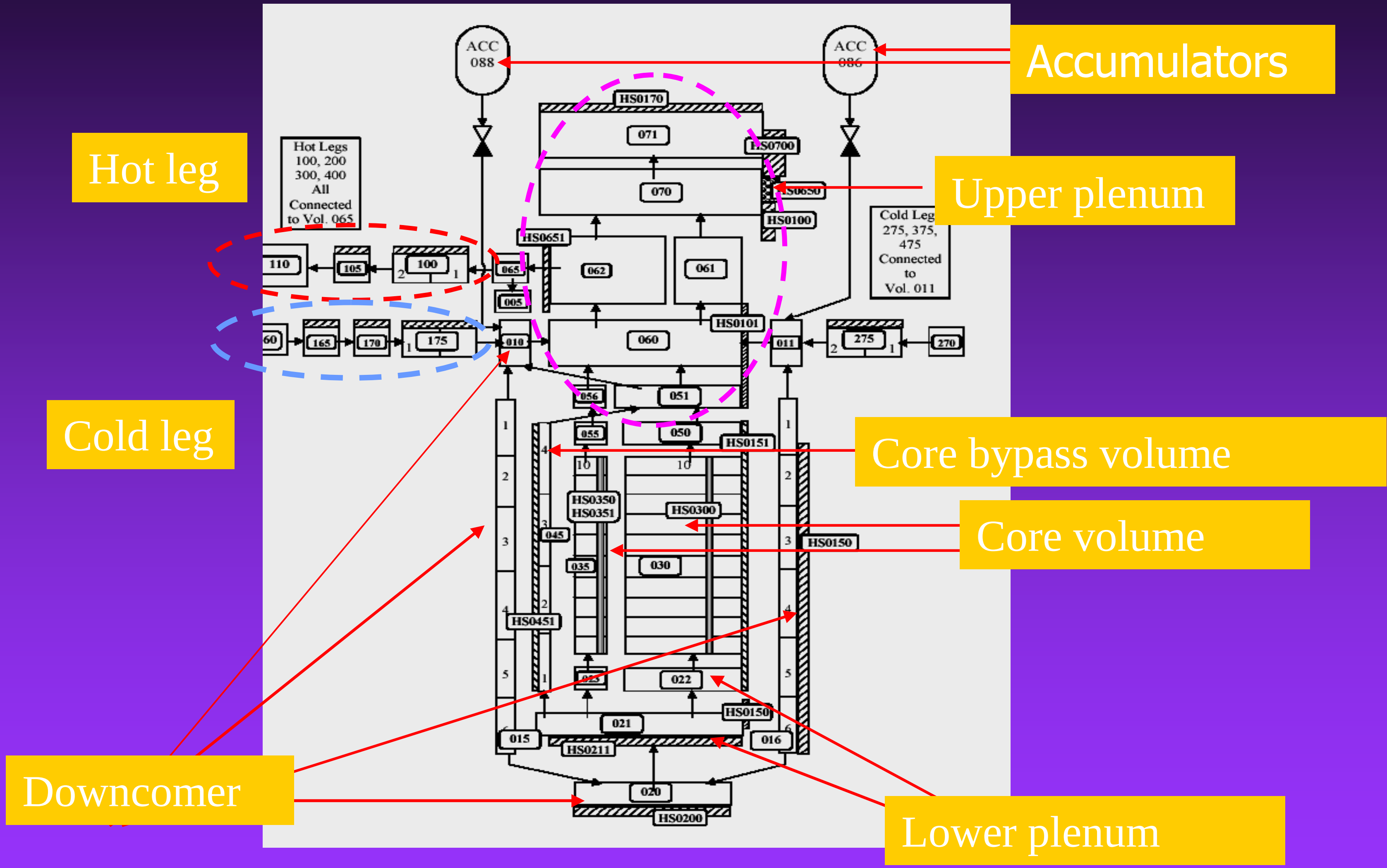

Fig B. Reactor Core Relap5 Model

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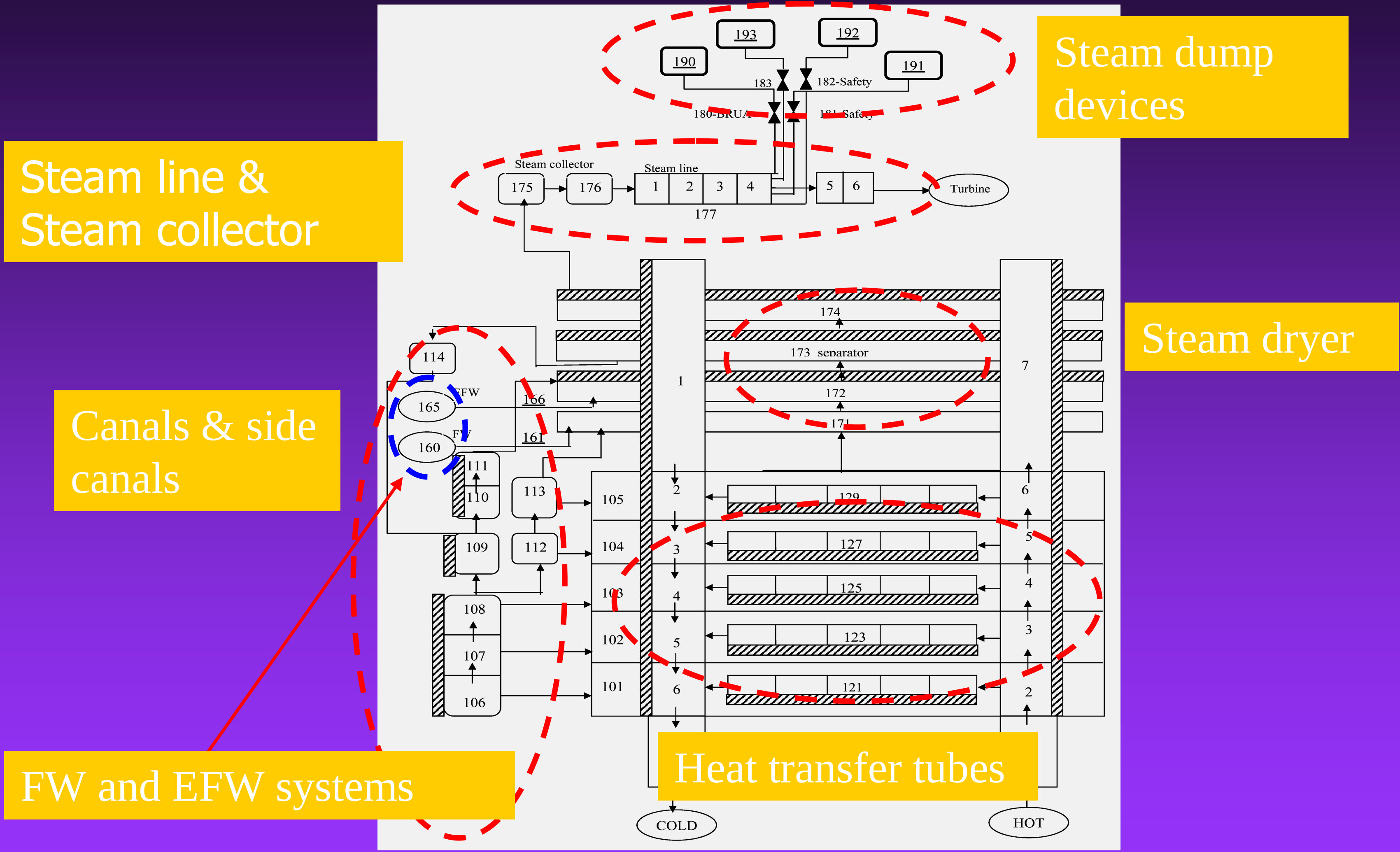

Fig C. Steam Generator Relap5 Model

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Initial Conditions:

Table 1. Comparison of initial conditions between measured plant parameters and RELAP5 calculations

Parameters	*BNPP Value*	*Relap5 calculation*
Core power, MW	3120	3120
Primary side pressure, MPa	15.7±.3	16.0
Cold leg #1 temperature, K	565.0	567.0
Cold leg #2 temperature, K	565.0	567.0
Cold leg #3 temperature, K	565.0	567.0
Cold leg #4 temperature, K	565.0	567.0
Hot leg #1 temperature, K	598.0	598.0
Hot leg #2 temperature, K	598.0	598.0
Hot leg #3 temperature, K	598.0	600.0
Hot leg #4 temperature, K	598.0	600.0
Core flow rate, kg/s	16400.0	16600.0
Loop #1 flow rate, kg/s	4100.0	4150.0

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Parameters	*BNPP Value*	*Relap5 calculation*
Loop #2 flow rate, kg/s	4100.0	4150.0
Loop #3 flow rate, kg/s	4100.0	4150.0
Loop #4 flow rate, kg/s	4100.0	4150.0
Pressurizer water level, m	7.8	7.8
Water level in SG#1, m	1.87	2.0
Water level in SG#2, m	1.87	2.0
Water level in SG#3, m	1.87	2.0
Water level in SG#4, m	1.87	2.0
Secondary side pressure, MPa	6.37	6.3

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Transient scenario:

The accident of a primary-to-secondary leak could be identified by the following symptoms:

- increase in activity in the affected SG steamline (10^{-3} msv/h) ;
- decrease in the primary coolant pressure;
- increase in level in the affected SG (+300 mm);
- PRZ level lowering (less than 4 m).

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Table 2. Chronological sequence of transient events

No. Description of event	*PSAR value, sec*	*RELAP5 calculated results, sec*
1. SG-2 cold collector breaks (100.0 mm equivalent diameter)	0.0	0.0 + 200.0
2. RCP set trip of affected loop	9.75	9.750 + 200.0
3. Generation of radiation signal	10.0	No data
5. Reactor SCRAM	11.20	11.20 + 200.0
6. Turbine stop valve closure	11.40	11.40 + 200.0
7. diesel-generators start-up	12.80	12.80 + 200.0
8. SG-2 BRU-A opening	13.50	15.0 + 200.0
9. Additional boron injection start to PRZ	41.75	45.0 + 200.0
10. End of RCP set coastdown of operable loops	87.0	87.0 + 200.0
11. Closing of BRU-A of affected SG-2	130.0	130.0 + 200.0
12. HPIS-(loop 3,4) start injection	140.0	150.0 + 200.0
13. EFW (SG 3,4) start injection	1000.0	1050.0 + 200.0
14. Filling of operable SGs 3,4 to the nominal water level	2200.0	2250.0 + 200.0
15. End of calculation	5000.0	5000.0 + 200.0

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Results and discussion

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$$\Delta P = P_p - P_S \approx 9.5\,Mpa$$

$$\dot{m}_{(leak)} \uparrow$$

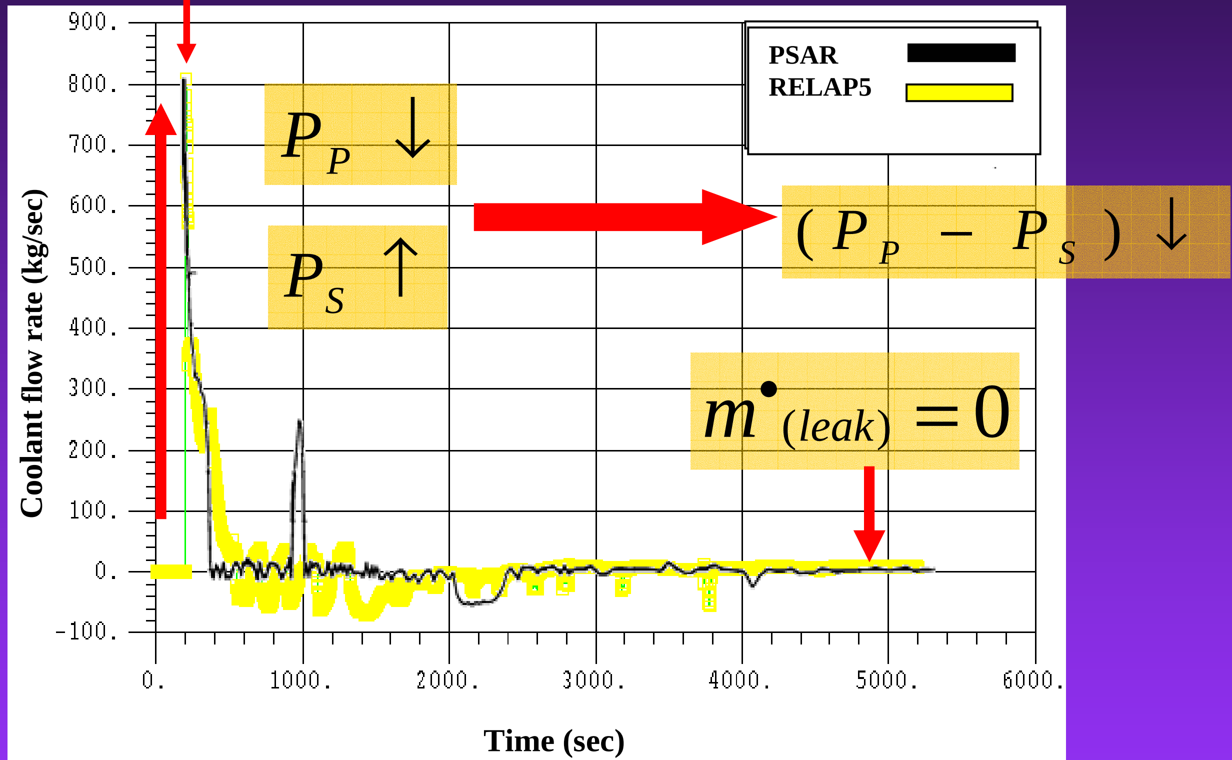

Fig 1. Primary to secondary leak

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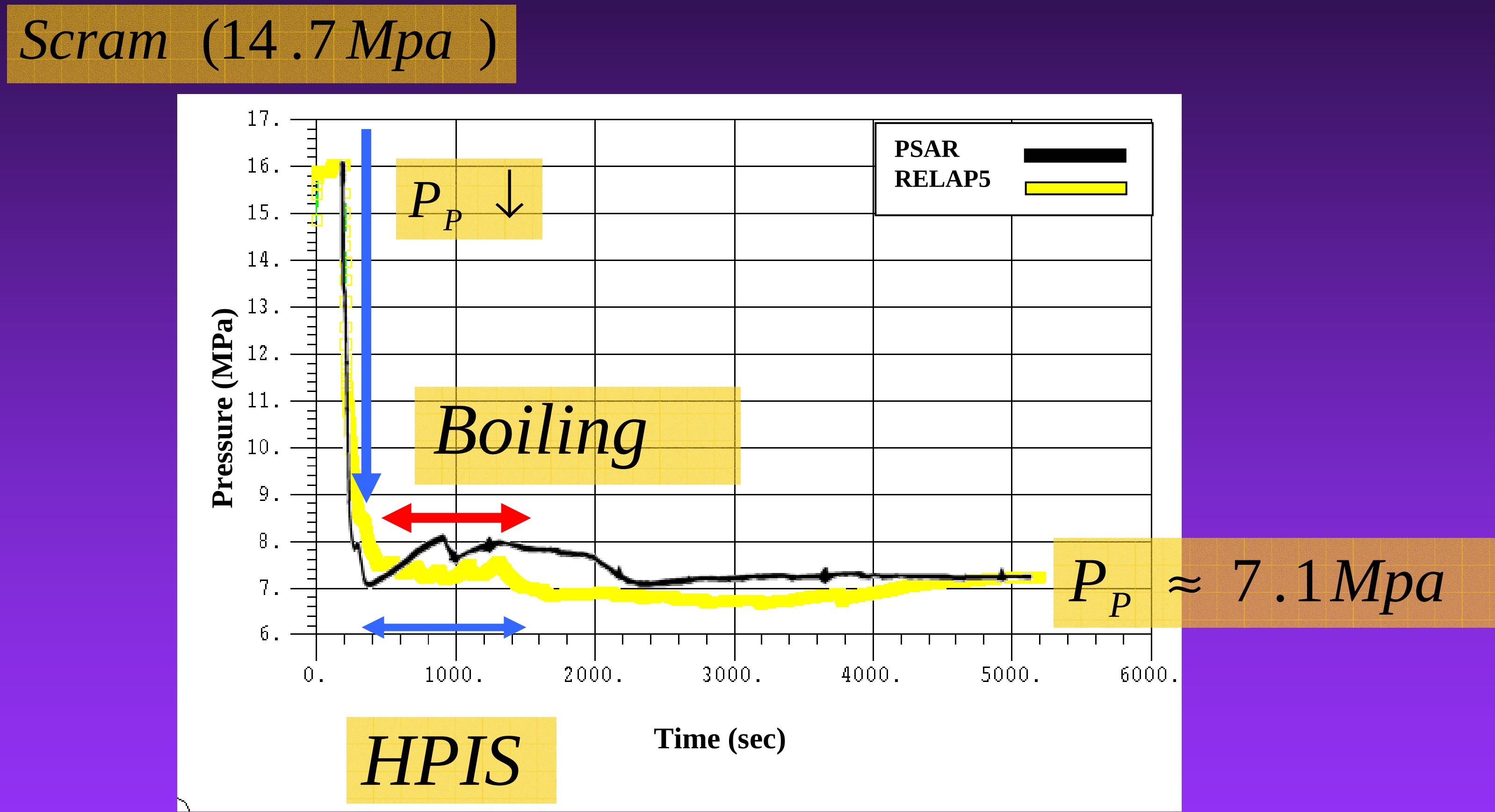

Fig 2. Pressure at the core outlet

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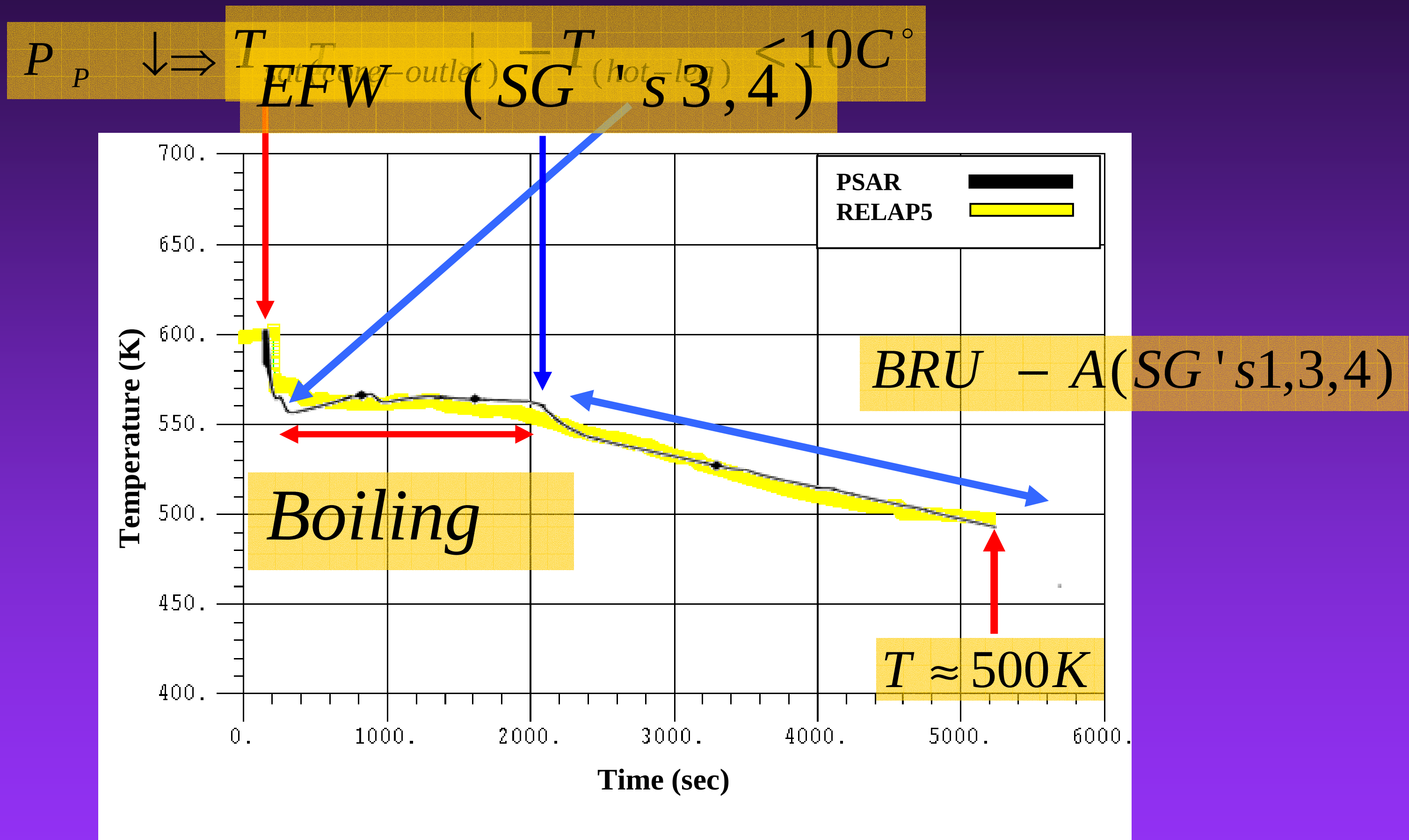

Fig 3. Coolant temperature at the core outlet

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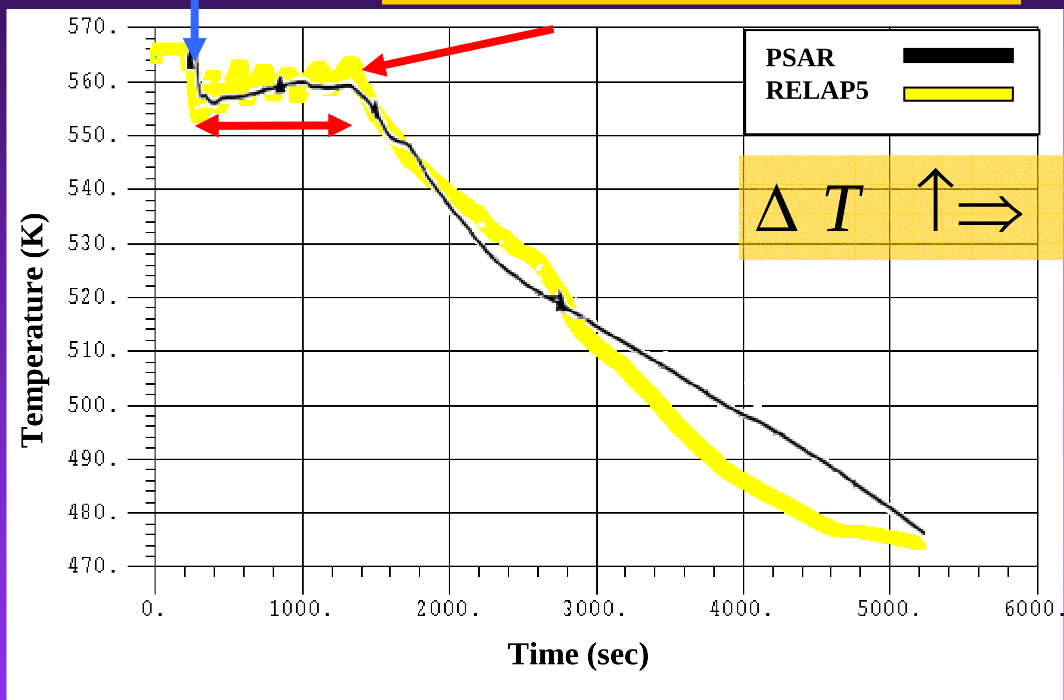
$P_S \uparrow BRU - A(SG's1,3,4)$ $\downarrow$ $(SG's\,3,4)$



$\Delta T \uparrow \Rightarrow q'' \uparrow$

Fig 4. Coolant temperature at the core inlet

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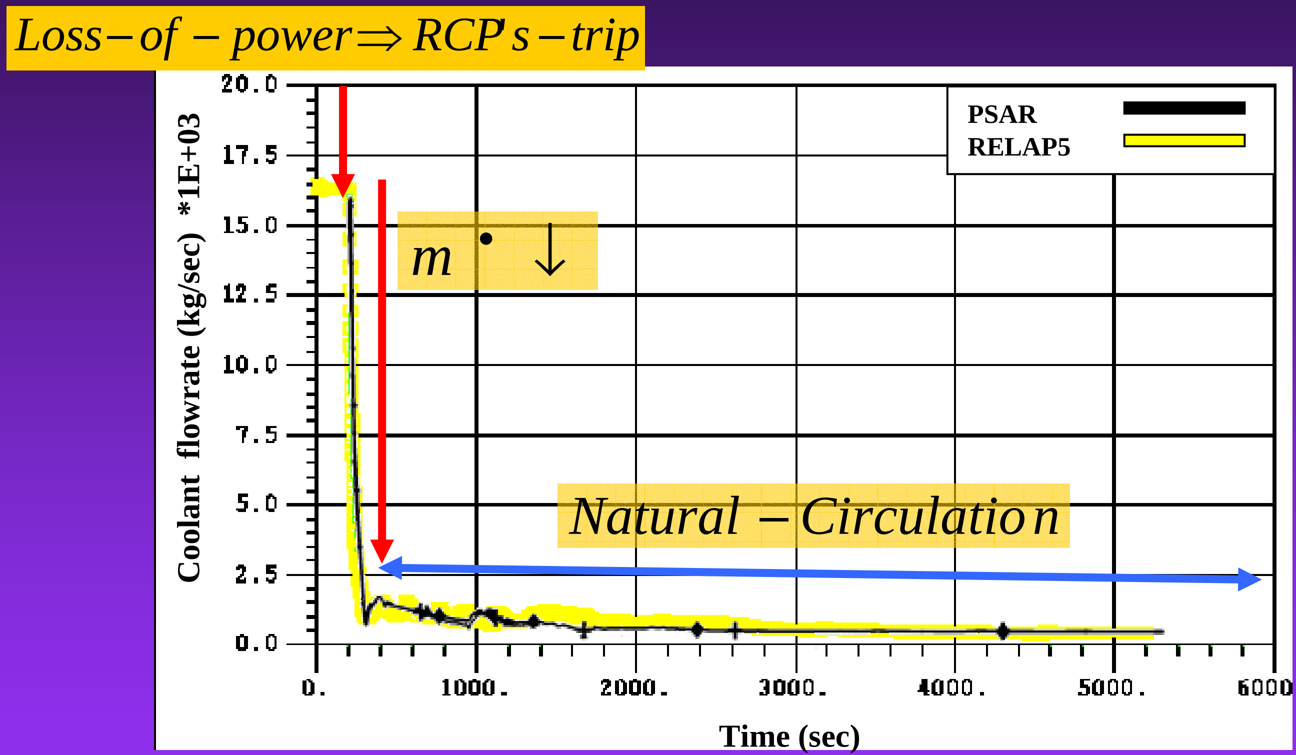

Fig 5. Coolant flow rate at the core inlet

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Conclusions:

- Although the calculation dealing with the consideration of the SGTR accident is a very conservative version of the damage, according to the present results acceptance design criteria is met.

- In the PSAR of the Bushehr NPP, DINAMIKA-97 code was used to evaluate the SGTR accident. According to the results obtained in this report, RELAP5 correctly predicts the behavior of the main plant parameters in comparison with the reported PSAR data.

Thank u !!!