Application of Best Estimate Thermalhydraulic Codes for the Safety Analysis of Research Reactors

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Outlines

- Need for the application of Best Estimate method for RR safety analysis
  - Reduction of conservative safety margins
  - Mitigation of constraining limits
- Reference plants: **10 MW MTR POOL TYPE RR**
  23 MW HW TANK TYPE RR
- Application of BE termalhydraulic codes for the analysis of RR transients (RELAP5/MOD3.3 & CATHENA)
- Comparison of the results
- Consequences of the transients
- Conclusions

OUTLINES

SAFETY TECHNOLOGY OF NPP
established international expertise in relation to:
computational tools
procedures for their application
(BE methods + uncertainty evaluation)
comprehensive experimental database

IAEA 10 MW MTR RR problem is re-considered through the use of a BE computational thermal - hydraulic code: **RELAP5/Mod 3.2**

FRJ 2 23 MW MTR RR comparison **RELAP5/CATHENA**

Importance of transferring NPP safety technology tools and methods to RR safety technology

- different ranges
- large variety
- bounding and generalized lists of events are available from IAEA documents

System codes are mature for application to transient analysis in RR

Computational Tool - RELAP5

✓The thermal-hydraulic system code RELAP5 was developed to simulate transient scenarios in Power reactors such as PWR, BWR, VVER…

✓Limited work was performed to access the applicability of the code to RR operating conditions:

- ✓Low pressure
- ✓Plate type fuel element (MTR)
- ✓Top-down flow (nom. condition), very low flow & flow reversal
- ✓Pool model (natural convection)

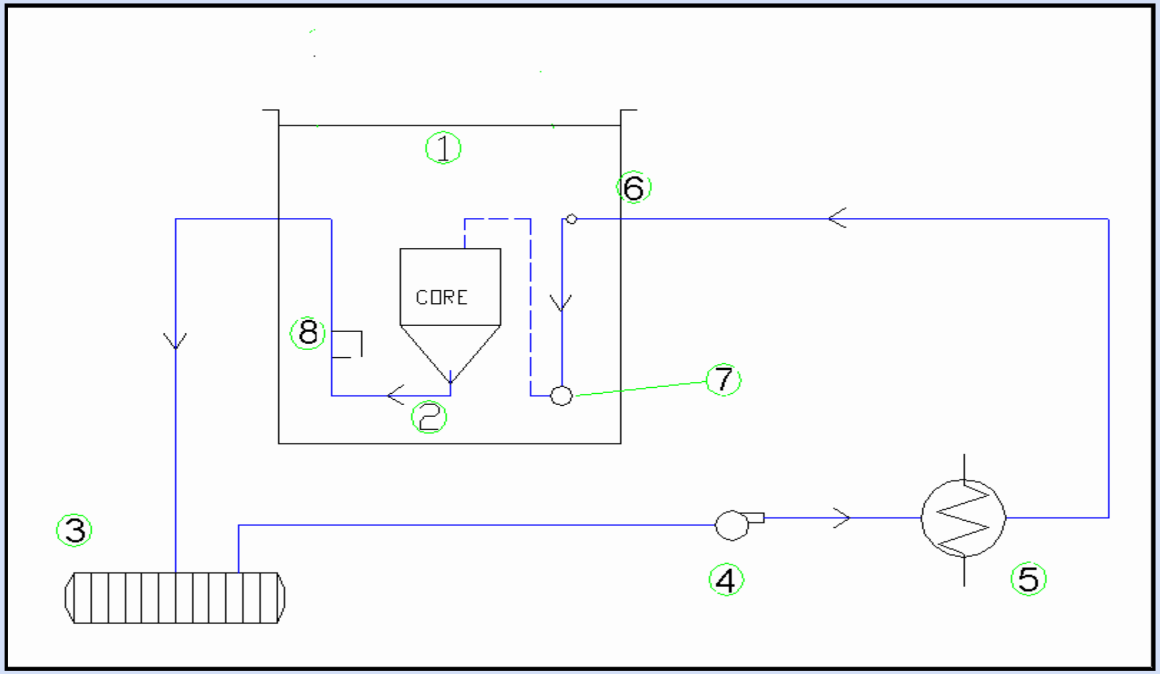
REFERENCE PLANT

IAEA 10 MW MTR pool type

RESEARCH REACTOR

- ✓ An attempt to perform standardized safety analyses for RR was proposed by the IAEA in the framework of core conversion from the use of highly enriched uranium fuel to the use of low enriched uranium fuel.
- ✓ A safety-related benchmark problem for an idealized generic 10 MW MTR light-water pool-type reactor was specified in order to compare calculational methods used in various research centers and institutions.

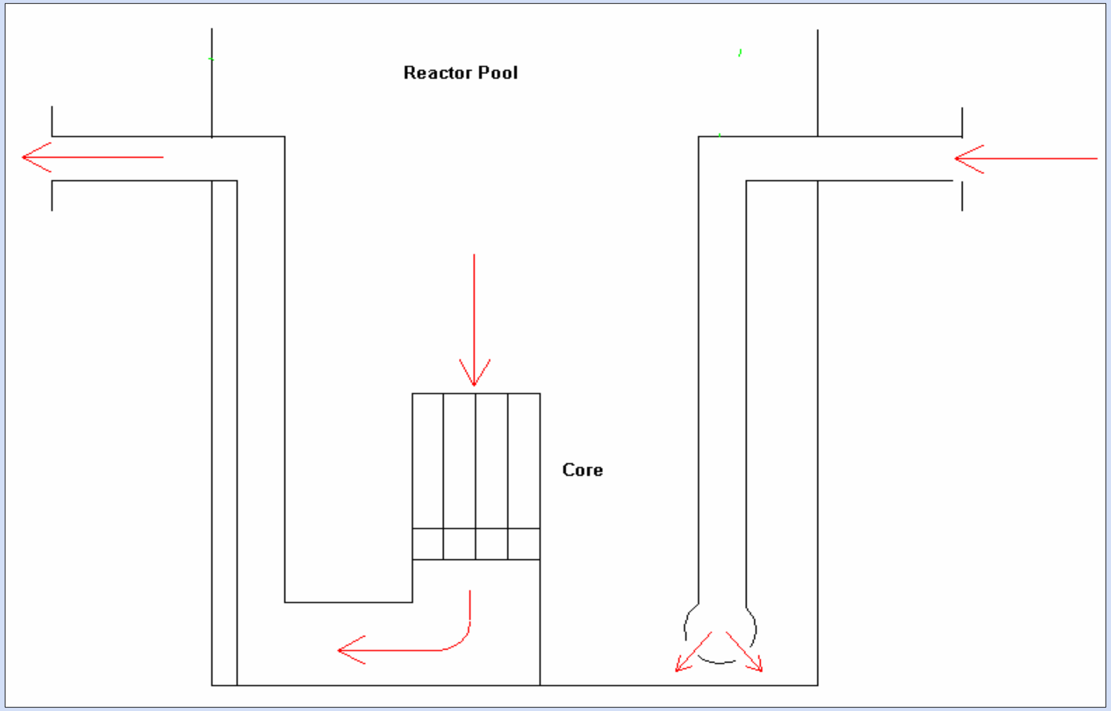
PRIMARY SYSTEM REACTOR



1. Reactor slot;
2. Outlet tubing;
3. Holdup tank;
4. Pump;
5. Heat exchanger;
6. Inlet tubing;
7. Distribution Ring;
8. Natural convection valve

Typical pool-type research reactor with an open water surface

CORE COOLING MODES



10 MW power cooled by downward flow
(steady–state)

REACTOR CORE GRID

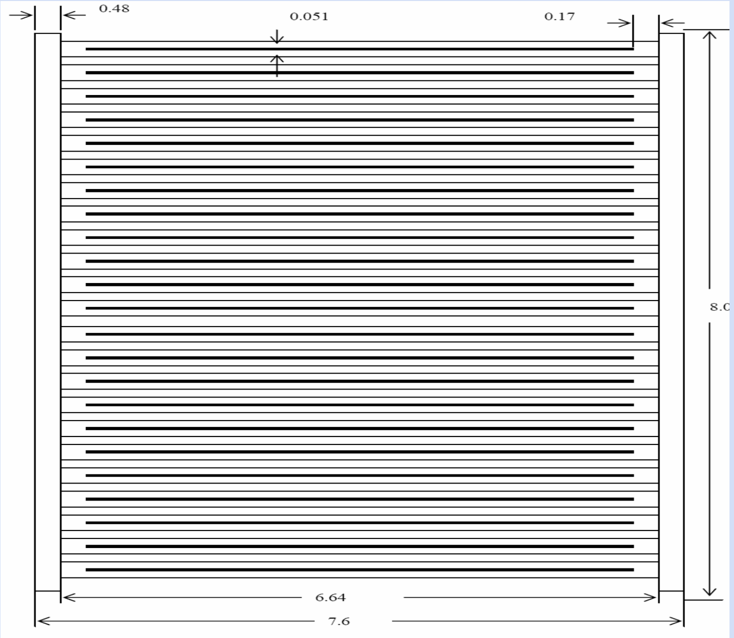
W	G	G	G	G	G
W	SFE 5%	SFE 25%	SFE 25%	SFE 5%	W
SFE 5%	CFE 25%	SFE 45%	SFE 45%	CFE 25%	SFE 5%
SFE 25%	SFE 45%	SFE 45%	H_2O + AL SFE 45%	SFE 45%	SFE 25%
SFE 5%	CFE 25%	SFE 45%	SFE 45%	CFE [illegible]	SFE 5%
W	SFE 5%	SFE 25%	SFE 25%	SFE 25%	W
W	G	G	G	G	W

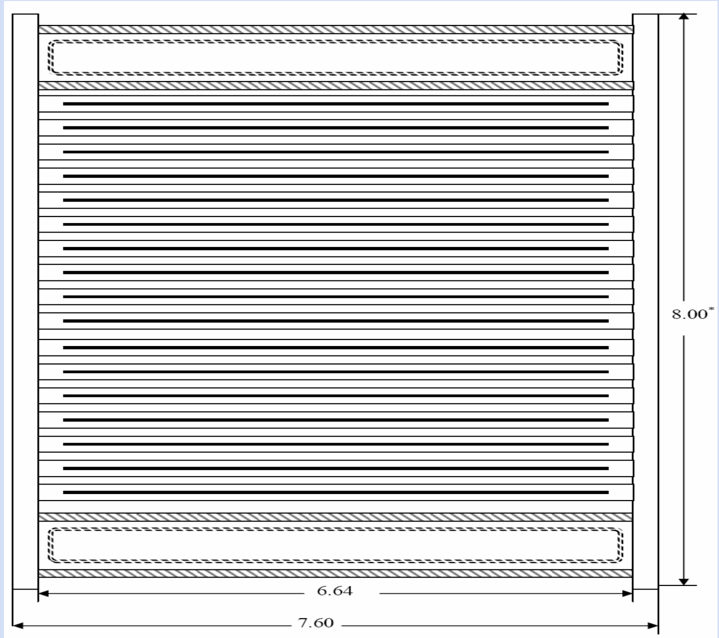
W	WATER	SFE	STANDARD FUEL ELEMENT
G	GRAPHITE	CFE	CONTROL FUEL ELEMENT
%	% URANIUM CONSUMED		

5x6 grid composed by:
-**21** standard MTR fuel element (**SFE**)
-**4** control fuel element (**CFE**)

The core is surrounded by graphite in two sides and in the other two sides is surrounded by water.

FUEL ELEMENT

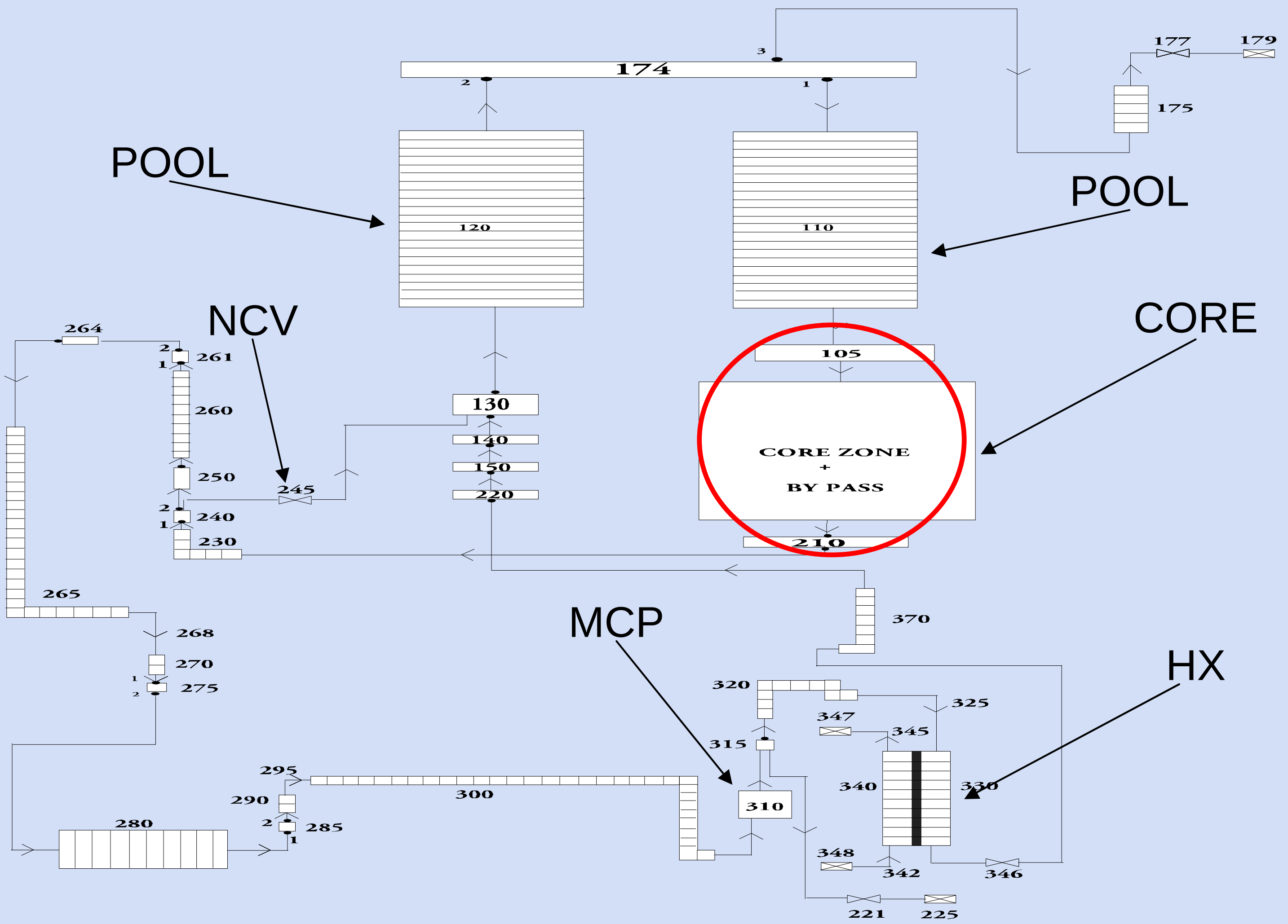




SFE
23 fuel plates located in a box

CFE
17 fuel plates and 2 control plates located in a box

Nodalization



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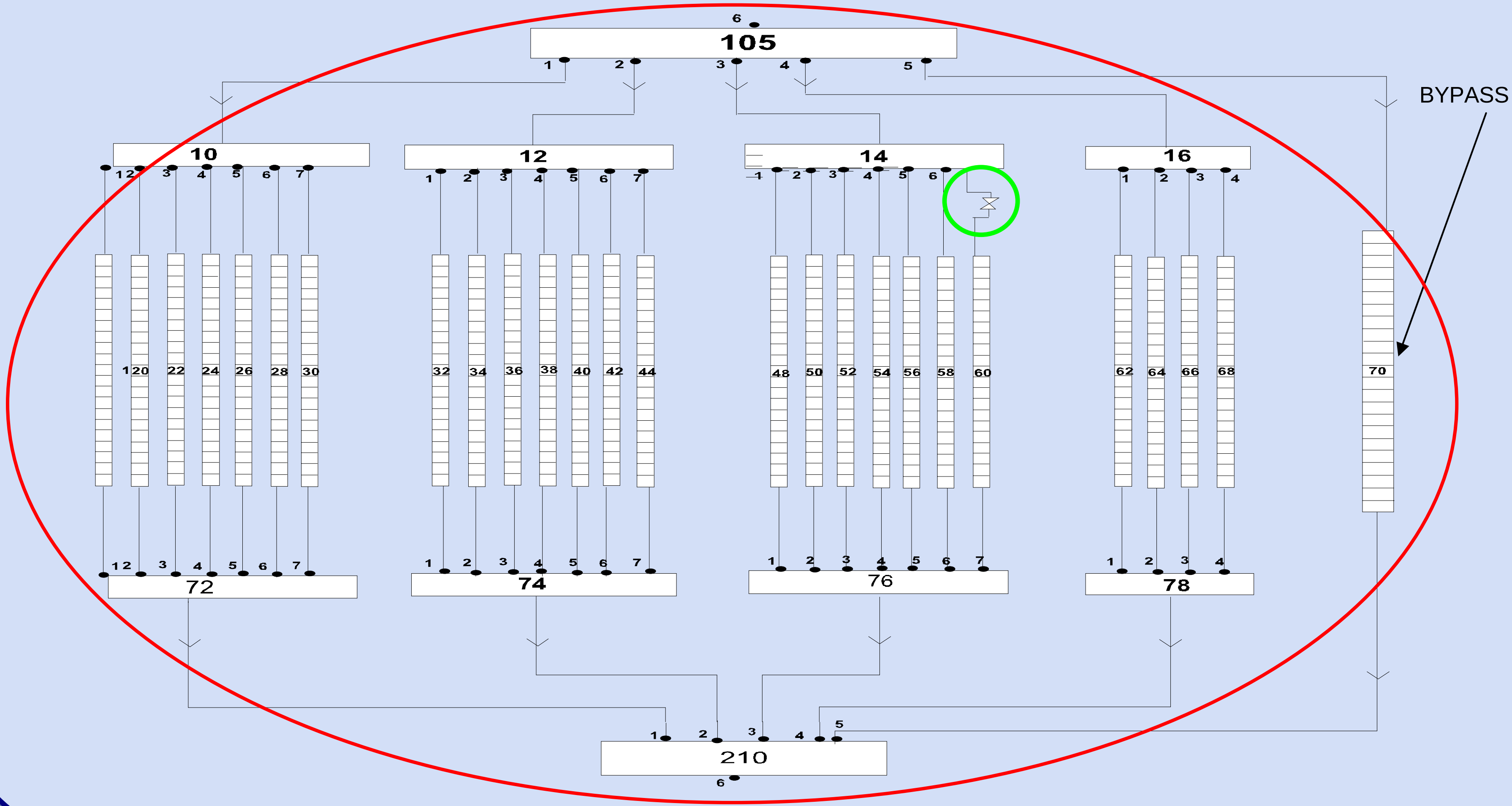
CORE NODALISATION



Computational Tool - RELAP5

✓RELAP5 stand alone uses the 0D (Point Kinetic) model to derive the core power.

✓ The next Step of this framework is to perform a Best Estimate simulation of the transients using coupled RELAP5/3D Neutron Kinetic code.

Transients

- **RIA** (Overpower Transient)

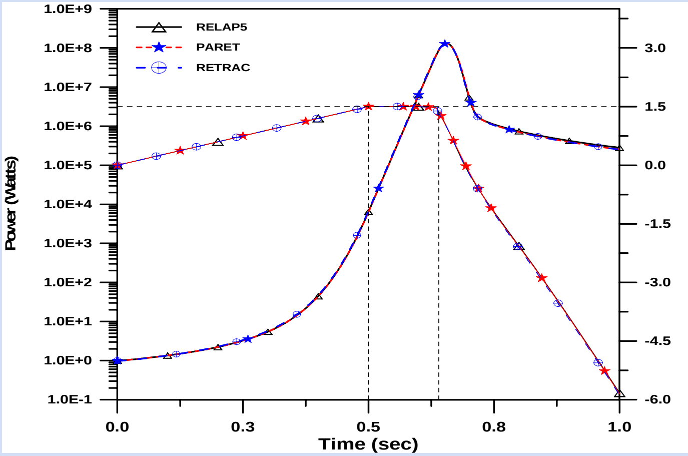
TRANSIENT INITIATED BY RUPTURE OF THE SUPPORT MECHANISM OF CCA DURING NOMINAL AND START UP CORE POWER CONDITIONS

- **LOFA** (Core Cooling Failure)

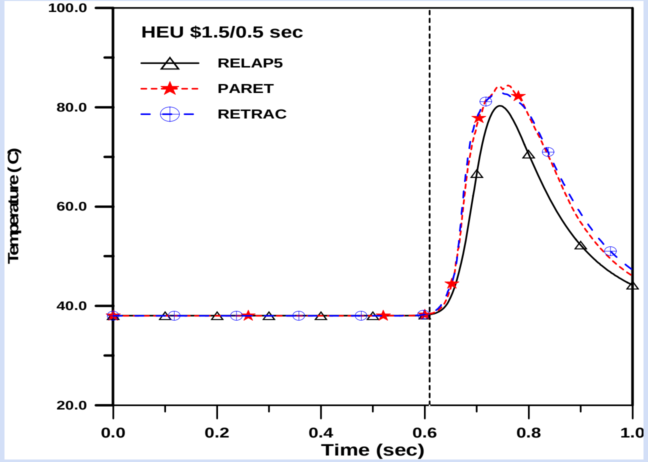
EFFECT OF THE TOTAL LOSS OF ACTIVE HEAT REMOVAL SYSTEM ON THE CORE BEHAVIOUR

- **FB** (95%, 100%)

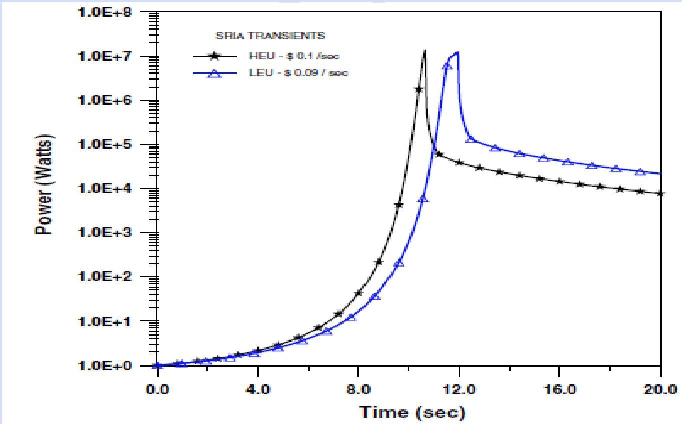
RIA



Core power course during FRIA



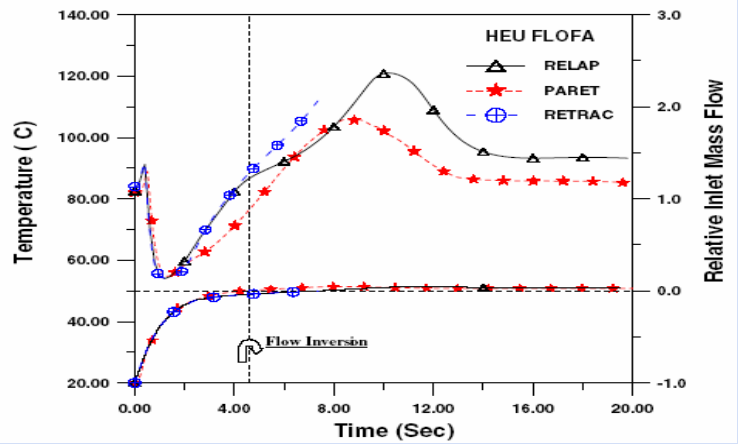
Core power during SRIA transients



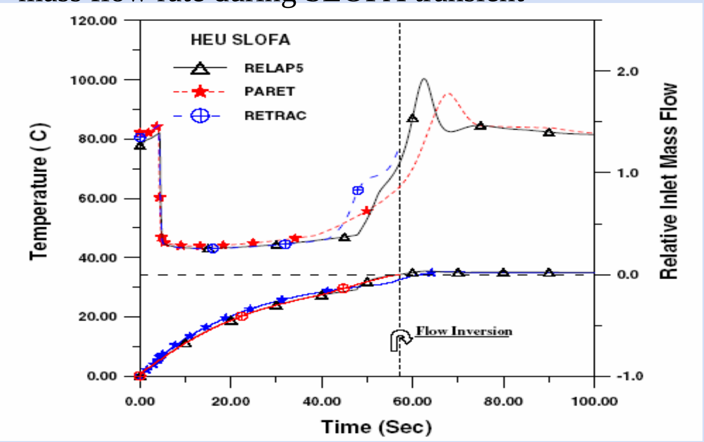
Outlet coolant temperature during FRIA

Good agreement with results obtained by channel codes specially developed for RRs

LOFA



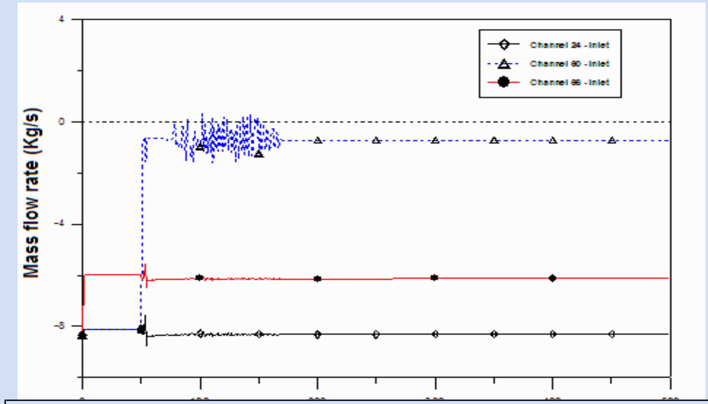
Clad surface temperature and relative
mass flow rate during SLOFA transient

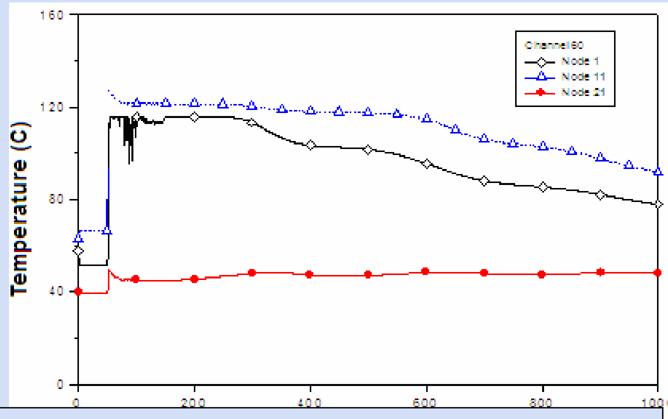


Clad surface temperature and relative inlet
mass flow rate during FLOFA transient

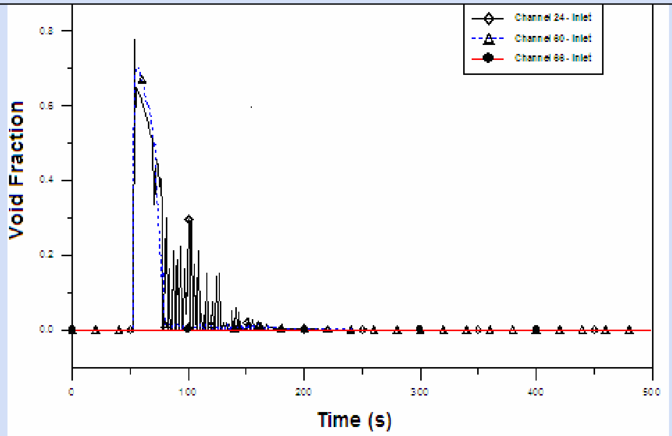
- In general, RELAP5 predictions for the LOFA transients are better than those predicted by channel codes since interactions between the core and the coolant loop are taken into account.

Partial Blockage

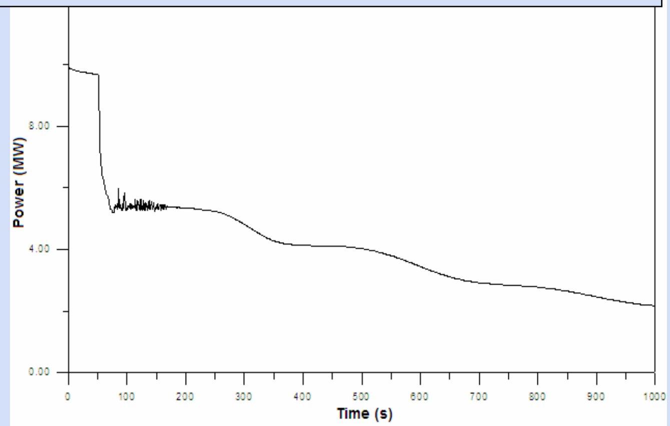




No FA damage has been observed during this transient



Void fraction in the obstructed channel



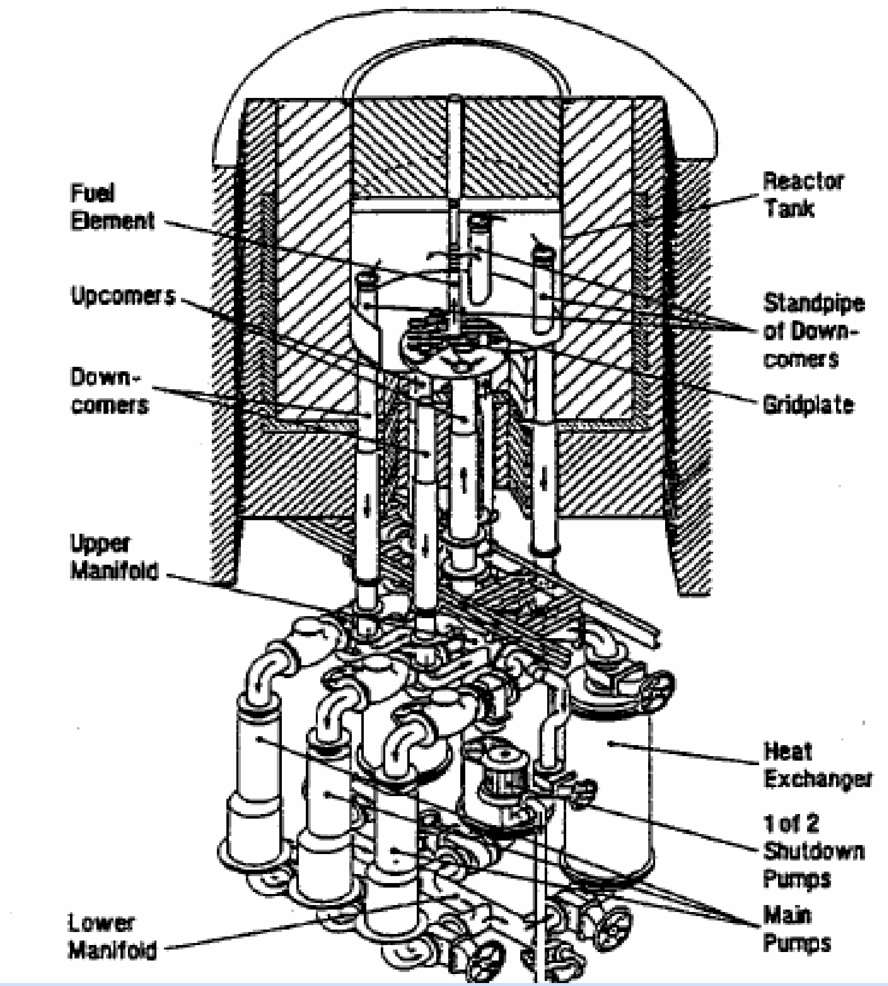
Reactor Power

REFERENCE PLANT

FRJ 2 23 MW heavy water

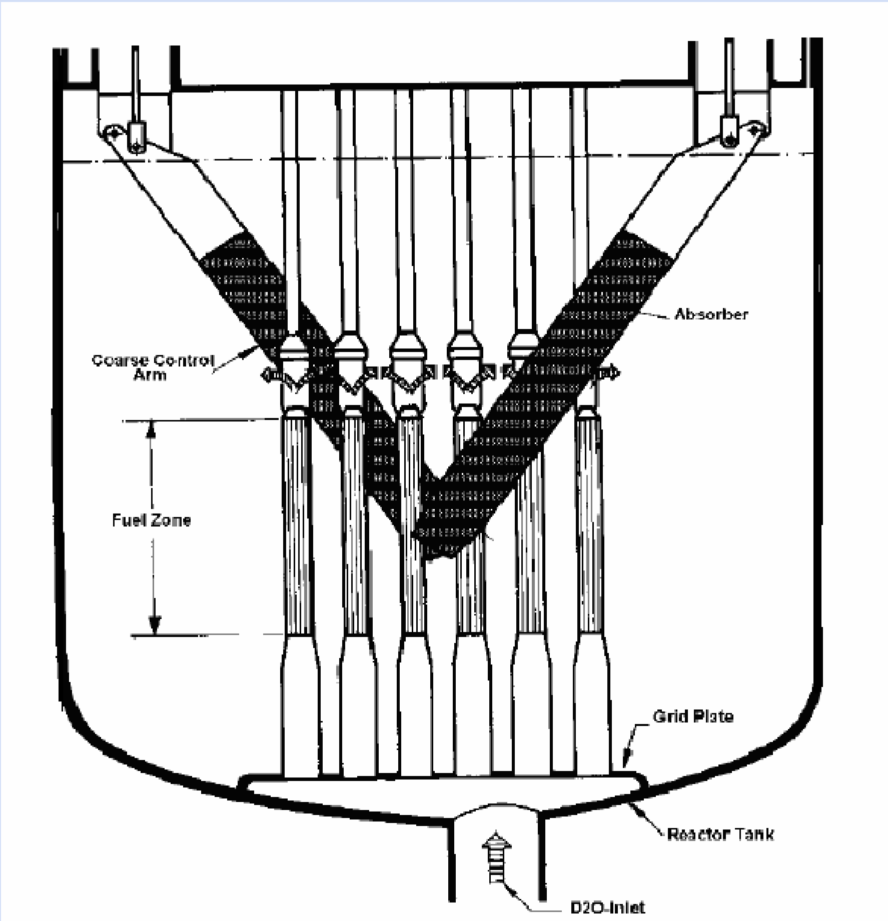
RESEARCH REACTOR

Reactor Block and Primary Components

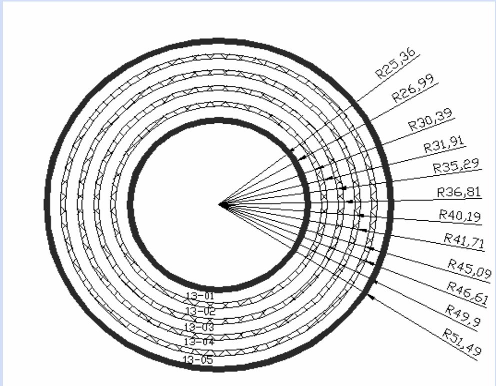


- Tank type RR
- **23** MW
- Graphite reflector
- Moderator and coolant: heavy water at atmospheric pressure
- Use:
  - material research
  - irradiation purposes

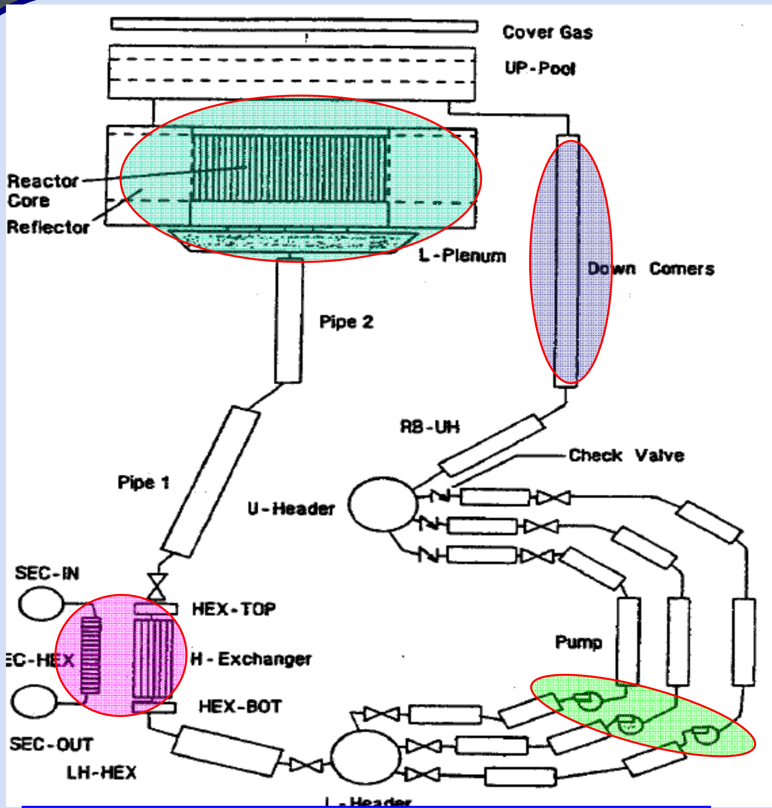
Vertical cross section of Al Tank



- **25** tubular MTR HEU FEs cooled by upward flow of D2O
- **6** CCAs (operation and shut down)
- **3** Rapid Shut Down Rods



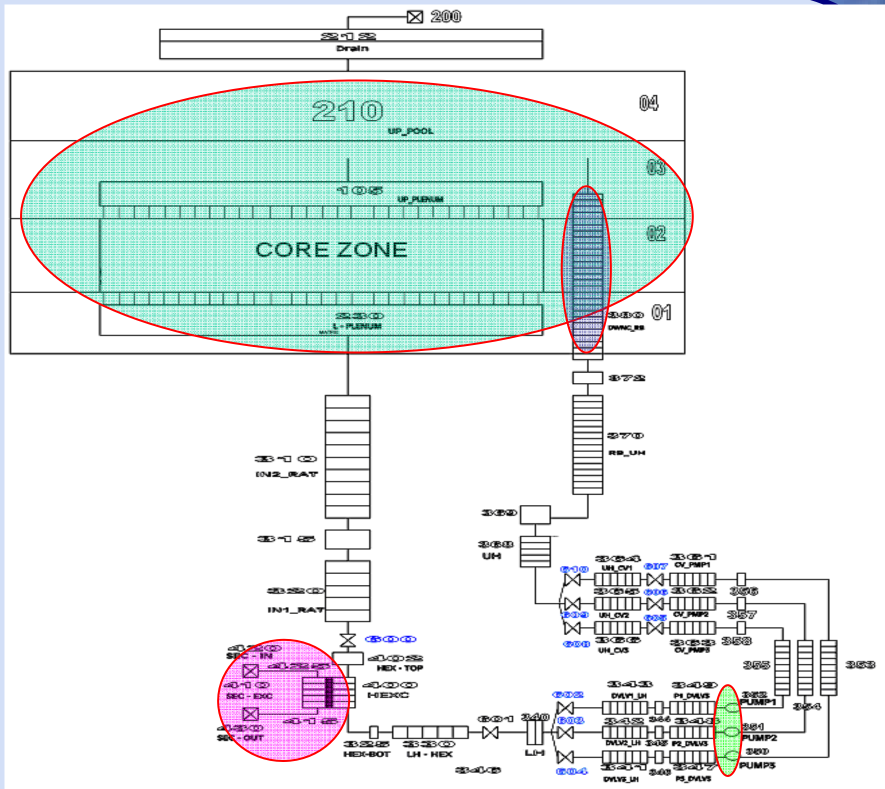
CATHENA nodalization



FRJ 2 Research Reactor

Cathena input deck

General scheme

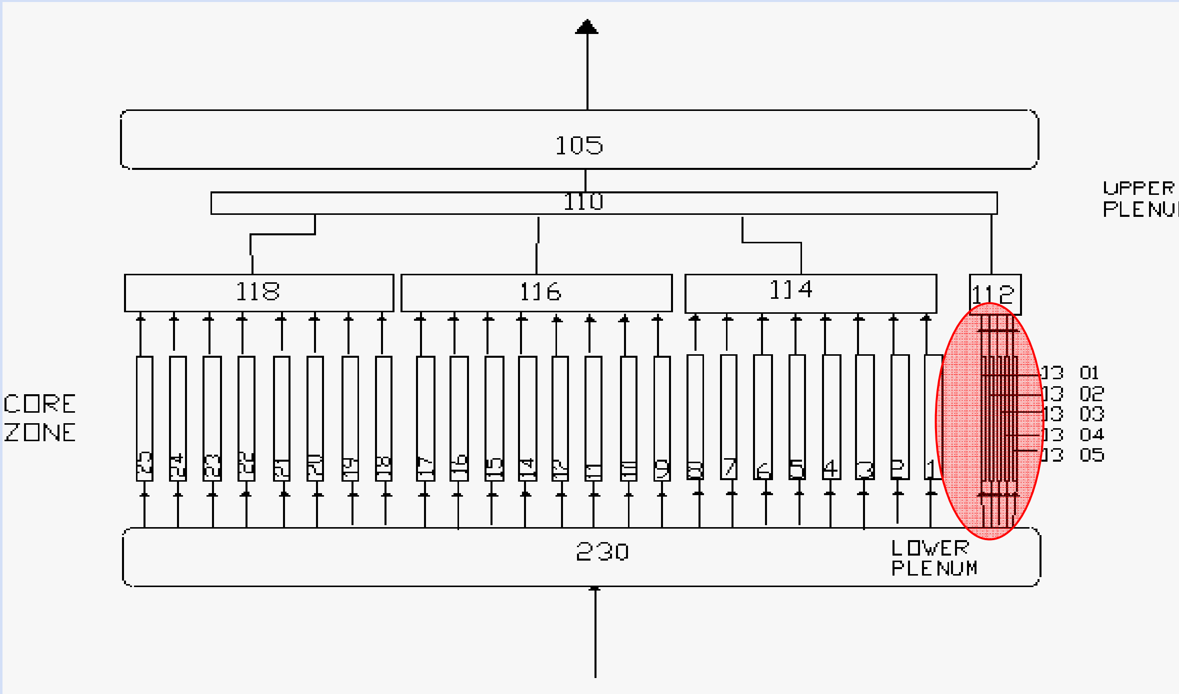


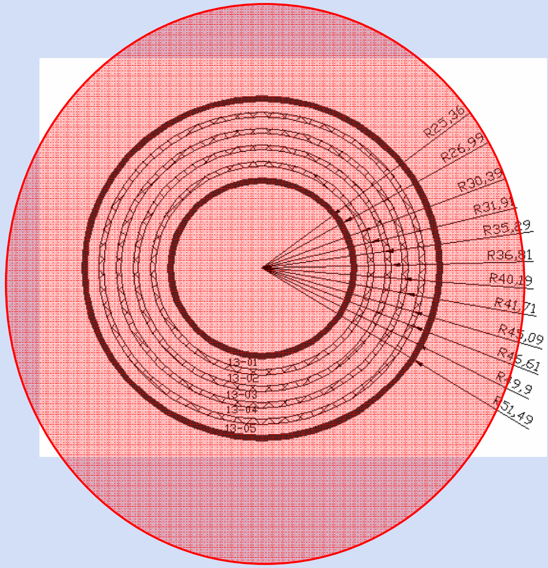
FRJ 2 Research Reactor

Relap5 mod3.3 input deck

General scheme

RELAP5 Core Zone nodalization



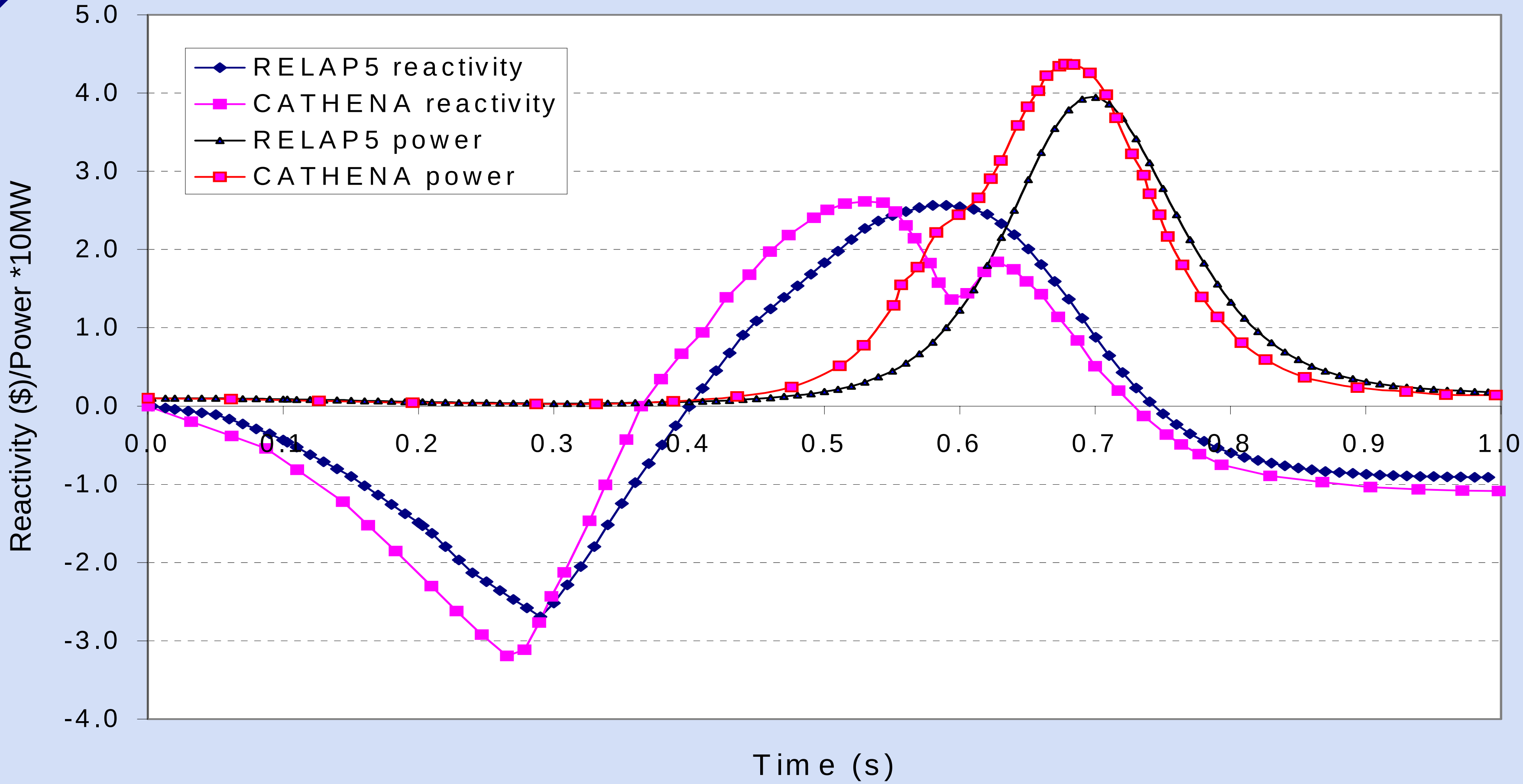


Description of the transient

Hypothetical fast reactivity transient

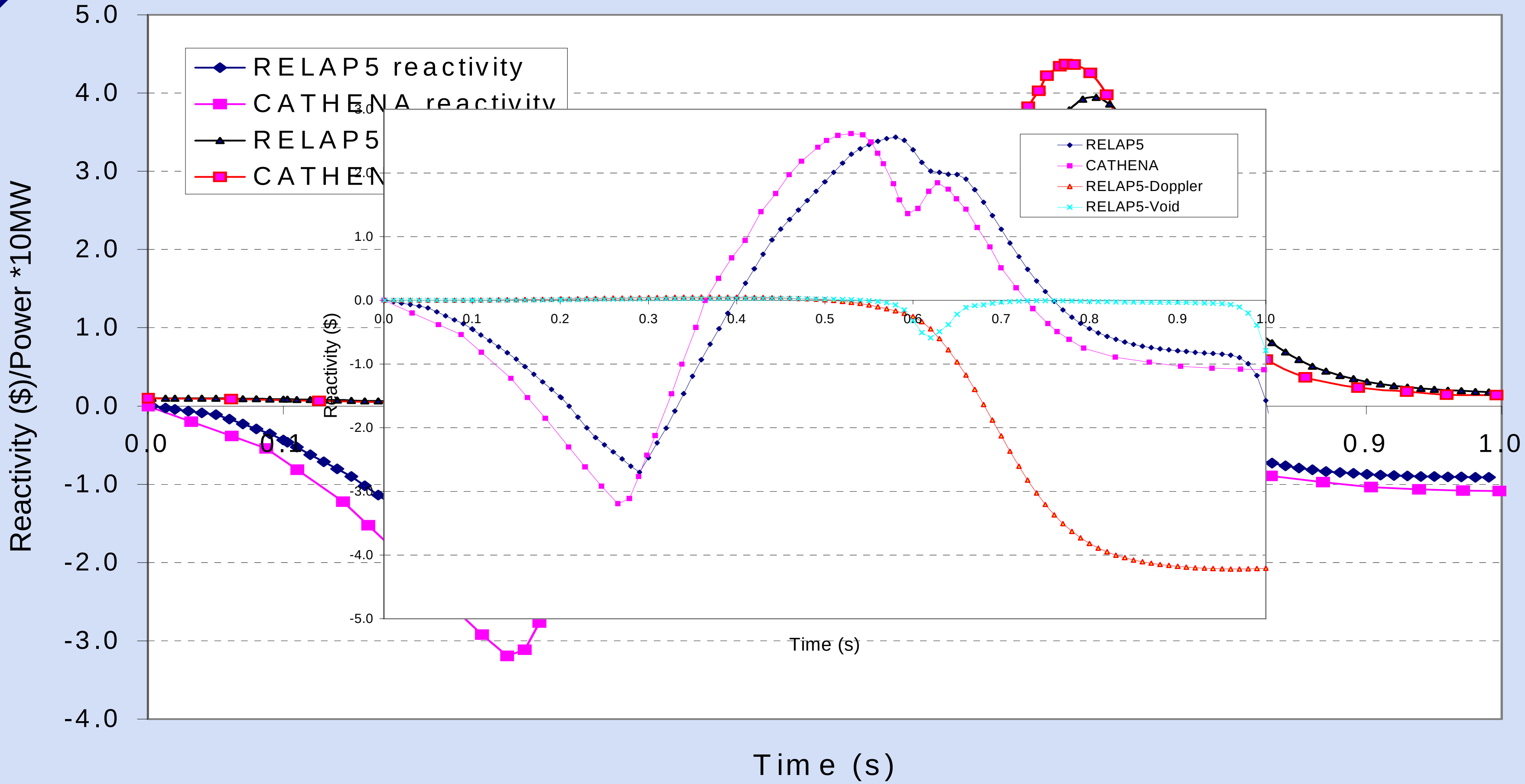
Transient key Parameters	**RIA**
Initial power (MW)	Nominal 23 MW
Scram setting point	Disabled
Maximum reactivity absorption	$ 2.8 (1.9%dk/k)
Maximum reactivity insertion	$ 3.3 (2.27%dk/k)

Results



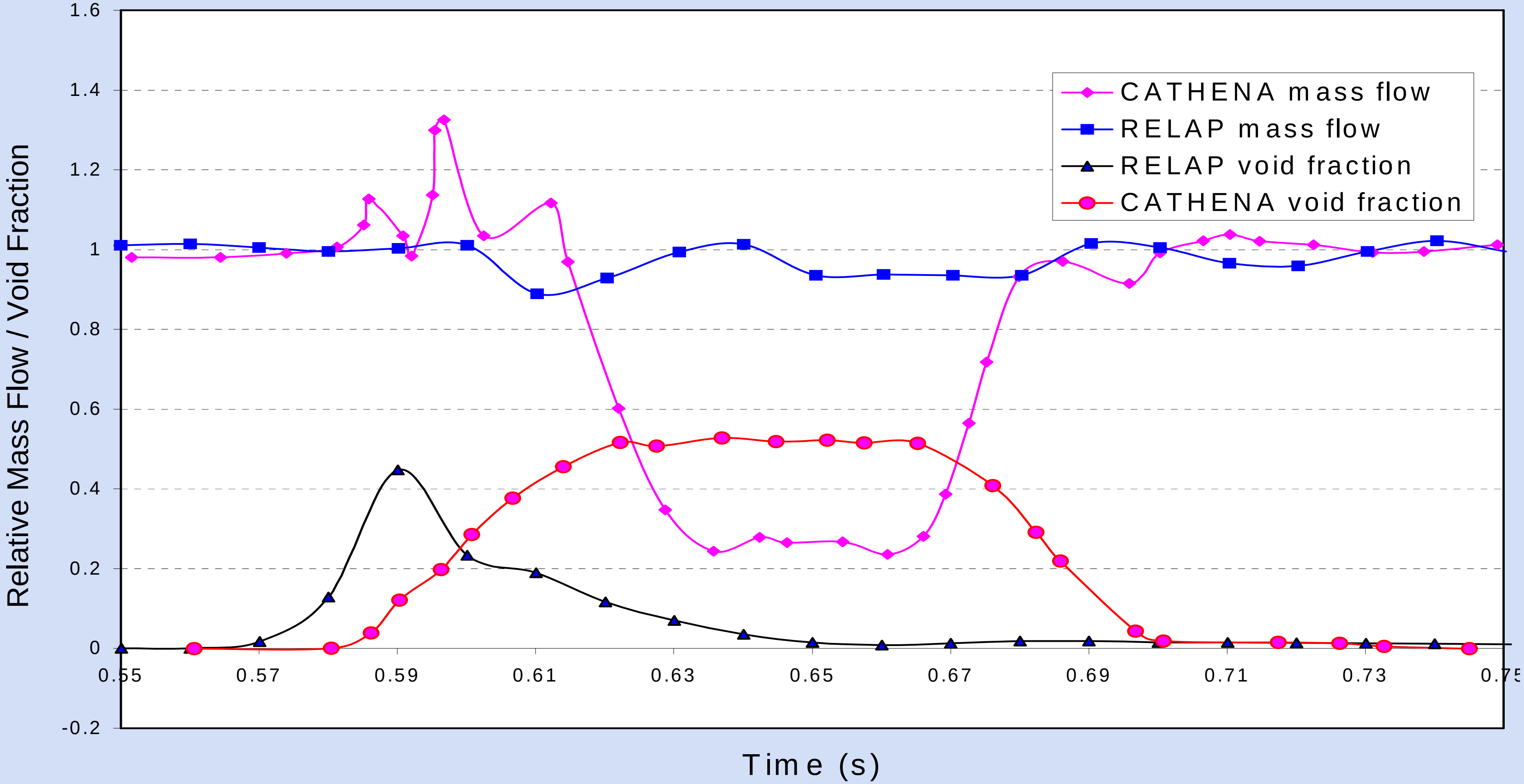
Reactivity/Core Power

Results



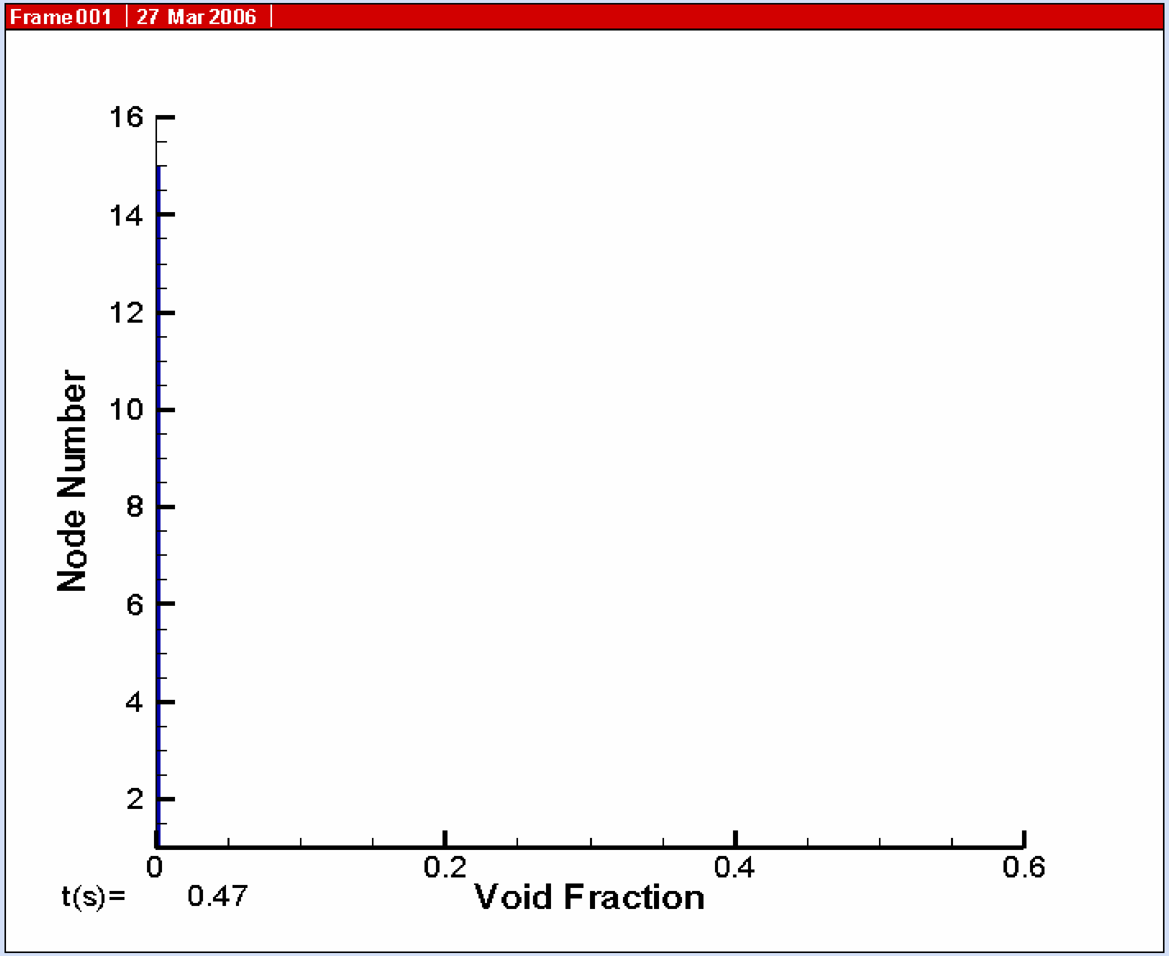
Reactivity/Core Power

Results



Mass Flow / Void Fraction

Results



Frame 001 | 27 Mar 2006
Node Number
16
14
12
10
8
6
4
2
0
0.2
0.4
0.6
Void Fraction
t(s)= 0.47

CONCLUSIONS

- The current work constitutes an attempt to apply this technique to the Research Reactors operating conditions.
- In general, for all the considered transients, the obtained results show similar trends with some specified channel codes results
- RELAP5 simulation seems to be more realistic since it take into account the interaction between the coolant loop and the core dynamic, especially, during fast power excursion and loss of flow transients.

CONCLUSIONS

- ✓ The **demonstration of applicability** of qualified BE system codes to RR accident analysis constitutes the key message from this paper: a proper accident analysis should be developed for RR that could benefit of the experience available from NPP.