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3D Analysis Methods - Study and Seminar

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Abstract

The first part of the report results from a study that was performed as a Nordic co-operation activity with active participation from Studsvik Scandpower and Westinghouse Atom in Sweden, and VTT in Finland. The purpose of the study was to identify and investigate the effects rising from using the 3D transient computer codes in BWR safety analysis, and their influence on the transient analysis methodology. One of the main questions involves the critical power ratio (CPR) calculation methodology. The present way, where the CPR calculation is performed with a separate hot channel calculation, can be artificially conservative.

In the investigated cases, no dramatic minimum CPR effect coming from the 3D calculation is apparent. Some cases show some decrease in the transient change of minimum CPR with the 3D calculation, which confirms the general thinking that the 1D calculation is conservative. On the other hand, the observed effect on neutron flux behaviour is quite large. In a slower transient the 3D effect might be stronger.

The second part of the report is a summary of a related seminar that was held on the 3D analysis methods. The seminar was sponsored by the Reactor Safety part (NKS-R) of the Nordic Nuclear Safety Research Programme (NKS).

Key words

3D transient computer codes, critical power ratio, BWR safety analysis

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PREFACE

The first part of this report results from a study that was performed as a Nordic co-operation activity with active participation from Studsvik Scandpower and Westinghouse Atom in Sweden, and VTT in Finland. The activity was jointly funded by the NKS-R research progamme and the participants.

The second part of this report is a summary of a related seminar that was held on the 3D analysis methods. The seminar was sponsored by the Reactor Safety part (NKS-R) of the Nordic Nuclear Safety Research Programme (NKS).

Part I

1 INTRODUCTION

Until now the safety analyses for boiling water reactors (BWR) have been carried out with axially one-dimensional models for the reactor core. Because of limited knowledge and the need to cover a great number of similar cases with one analysis, the phenomena affecting the analyzed accident are exaggerated to ensure the conservatism of the result.

Recently several best-estimate computer codes, using three-dimensional models for the core, have been developed in the Nordic countries (TRAB-3D, POLCA-T, SIMULATE-3K) and elsewhere in the world. The need of three-dimensional neutronics calculation is largest in such cases, where fission power development is important and its spatial distribution changes during the transient. These cases include almost all reactivity initiated accidents (RIA) and anticipated transients without scram (ATWS). The stability considerations of BWRs always need a three-dimensional core model.

The purpose of this preliminary study is to identify and investigate the effects rising from using the 3D transient computer codes in BWR safety analysis, and their influence on the transient analysis methodology.

Two different types of transient calculations can be identified. Safety analysis report (SAR) should cover a large number of cycles and cores (i.e., not frequently updated), whereas in the cycle-specific analyses the core is well known. These two cases have different level of conservatism, which should be taken into account also in the case of 3D methodology.

One of the main questions involve the critical power ratio (CPR) calculation methodology. The present way, where the CPR calculation is performed with a separate hot channel calculation, can be artificially conservative. In reality, the location of the most limiting fuel bundle or coolant channel can change during the transient. A 3D calculation makes a whole core 3D CPR calculation at least an alternative option to the hot channel calculation.

The more realistic 3D model of the reactor core removes the need for some conservatism rising from the reactor data condensation to one dimension. This condensation is laborious and leads inevitably to inaccuracy, especially with mixed cores of several different fuel bundle types. In addition, partly inserted control rods and control rod movements e.g. in partial scram can be more easily modelled realistically with a 3D core.

One trend in the plant modernisation projects has been the elimination of the typical fast transients as the limiting cases in licensing. Slower 10-15 second transients that have become decisive include a much stronger coolant mass flow redistribution between the fuel bundles. This is not considered in a typical 1D calculation.

This preliminary study was focused on the cycle-specific CPR calculations of operational transients in BWRs. The following investigations were decided to be carried out, in order to address the 1D to 3D methodology transfer:

1. Demonstration of the typical differences between 3D and 1D approach for evaluation of partial scram and full scram with partly inserted control rods.

2. Demonstration of possible conservatism for a typical fast flow decrease transient in a 1D method compared to a 3D calculation.

3. Demonstration of possible conservatism for a typical fast pressure increase transient in a 1D method compared to a 3D calculation.

The first item covers obvious 3D problems, which have so far been addressed by 1D methods. The second and third cover both the possible 3D influence on global results and the difficult transfer of information from the 1D average channel calculation to the 1D hot channel calculation.

2 PARTIAL SCRAM AND PARTLY INSERTED CONTROL RODS

In transient analyses of Nordic BWRs the treatment of initially partly inserted control rods in a transient with a reactor scram is nearly always a problem that has had to be addressed in a 1D calculation. In a small number of Nordic BWRs also the partial scram is used and has had to be modelled with 1D methods.

The partial scram can be conservatively treated with regard to the scram effectiveness in a 1D calculation. However, the 3D effects are not obviously treated in that case. The BOC and MOC situation of partly inserted control rods and the way the full scram should be modelled has also been treated mostly conservatively in Nordic BWR transient safety analysis. The most direct method is to assume that the partly inserted control rods are not to be inserted during the full scram. The 3D effects rising from both of these 'problems' was investigated using the STUDSVIK codes SIMULATE3 and SIMULATE3K (abbreviated S3K) in this work. SIMULATE3 is used to find the steady-state core conditions. S3K is used to evaluate the 1D reactor loop and the 3D core transient.

The demonstration cases for partial scram, an internal pump reactor pump trip with partial scram and an isolated partial scram, resulted in the following preliminary conclusions:

- The 3D influence on the generated core power is significant on the local bundles surrounding the partial scram rods.

- For typical fast transients the local (3D) power changes do not seem to influence the minimum CPR during the transient because of the filtering effect of the fuel pin time constant.

- Based on the isolated partial scram investigation a fairly low impact of around 0.03 in minimum CPR can be observed.

The overall power reduction of a partial scram, i.e. the average power decrease, has been demonstrated in real events to be highly dependent on rod positioning in the radial power map. This effect is of obvious reasons not covered in a 1D methodology.

The demonstration case for the initially partly inserted control rods, a turbine valve closure at off rated operating conditions, resulted in the following preliminary conclusions:

- In the typical fast pressurisation event the impact of partly inserted control rods is relatively small. This is due to the conservatively late initiation of scram.

- Locally both positive and negative 3D power (and CPR) effects are demonstrated.

- In the example the minimum CPR during the transient was exactly the same between the 3D situation and a simulated 1D situation.

- In a situation where the limiting fuel bundle is adjacent to a partly inserted control rod a more pronounced impact is expected.

The combined information from the Studsvik and the VTT turbine trip analysis indicate that the influence of initially partly inserted rods could be more important in a case with early, (best-estimate, non-conservative) scram timing.

3 FAST FLOW DECREASE TRANSIENT

A flow decrease transient was selected as one of the typical transients to start calculating with a coupled 3D neutron kinetic and thermal hydraulic code. The selected flow decrease transient was a total pump trip; i.e., all pumps are tripped at the same time point and the coast down behaviour of the pumps is identical. The transient was calculated with Westinghouse Atom's POLCA-T code on a model of Olkiluoto 2 with a fictitious core. The aim of the simulation was to find out if any 3D effects can be detected during the event. One of the 3D effects looked for was to see, if the location of the minimum critical power ratio, CPR, moves from one fuel bundle to another during the flow decrease or not.

The model of TVO 2 for POLCA-T is a quarter part of the core for nuclear kinetics loaded with SVEA 64 fuel. The thermal hydraulic model covers parallel bundles, by-pass, inlet and outlet plenum, pressure vessel internals and main recirculation pumps. The energy stored in the vessel, bundles and internals are taken into account. The evaluation of the critical power ratio is in this simulation, done with the AA 74 correlation, which is applied for each fuel bundle. Each fuel bundle is divided in 25 axial cells in both the hydraulics and the fuel rods and boxes as well as the common by-pass channel for the core.

The result from this particular simulation shows that the location of the minimum critical power ratio is locked to one fuel bundle over the entire simulated timed. The behaviour of the minimum CPR, however, is dependent on the radial power distribution and the way the actual core is loaded. Control rod movements during the transient, i.e. a partial scram can cause a shift of the minimum CPR to another assembly.

Axial behaviour of the CPR for fuel rods during the transient is dependent on the axial power distribution. If the power distribution is bottom peaked, the CPR at the top of fuel rods, which is usually the location of the overall minimum CPR, will increase in the beginning of the transient. This is, in addition to the axial power distribution, due to the fact that the flow decrease is slower at the top of the fuel bundle compared to the inlet flow decrease.

4 FAST PRESSURE INCREASE TRANSIENT

To see the difference between 1D and 3D calculation in a fast pressure increase transient, a real pressurisation transient that happened in Olkiluoto 1 plant in 1985 was calculated with VTT’s 1D code TRAB and 3D code TRAB-3D. The comparison is meaningful, because the BWR circuit models are identical in the codes. Nevertheless, getting the codes to calculate the same situation is somewhat difficult. In this investigation the TRAB core model was tuned to get the same axial power distribution and feedback coefficients as TRAB-3D in steady state.

The transient occurred because of erroneous functioning of the reactor pressure controller leading to the closing of the turbine valves in approximately 0.5 seconds and a maximum measured pressure of 78.5 bar. The incident was safely terminated by the normal operation of the reactor safety systems, including the reactor scram and the relief valves.

The TRAB-3D results are closer to the measurement data as can be seen in Figure 1. This is, mostly, explained, by the remaining conservatism in the TRAB calculation caused by not taking into account the initially partly inserted control rods in the TRAB calculation. A second TRAB-3D calculation with the partly inserted rods not being inserted during the scram shows even greater deviation from the measurements. In this respect the results are similar as the Studsvik results for the partly inserted rods.

From the preliminary investigation it appears that the transient change in CPR is 0.1 smaller with the best-estimate 3D calculation compared to the TRAB hot channel calculation with the same hot channel factor as the maximum relative bundle power in steady state in the 3D calculation. With the assumption of partly inserted rods not taking part in the scram, this difference between the 1D and the 3D calculation is reduced to

0.03. The minimum CPR behaviour is shown in Figure 2. The remaining difference can be due to mass flow redistribution during the transient.

It must be noted, however, that his case is not a safety analysis pressurisation transient with a conservative late initiation of the scram but a real transient with as realistic a model as possible. In this study the CPR calculation was made for 10 channels selected from the TRAB-3D calculation using the multiple hot channel methodology earlier applied at VTT for VVER analyses. In future, a full core CPR calculation approach may be more appealing.

Earlier calculations with both TRAB and TRAB-3D show that this transient is extremely sensitive to the choice of some uncertain parameters. Especially gas gap conductance has a much larger effect on the results than the choice of using 1D or 3D neutronics.

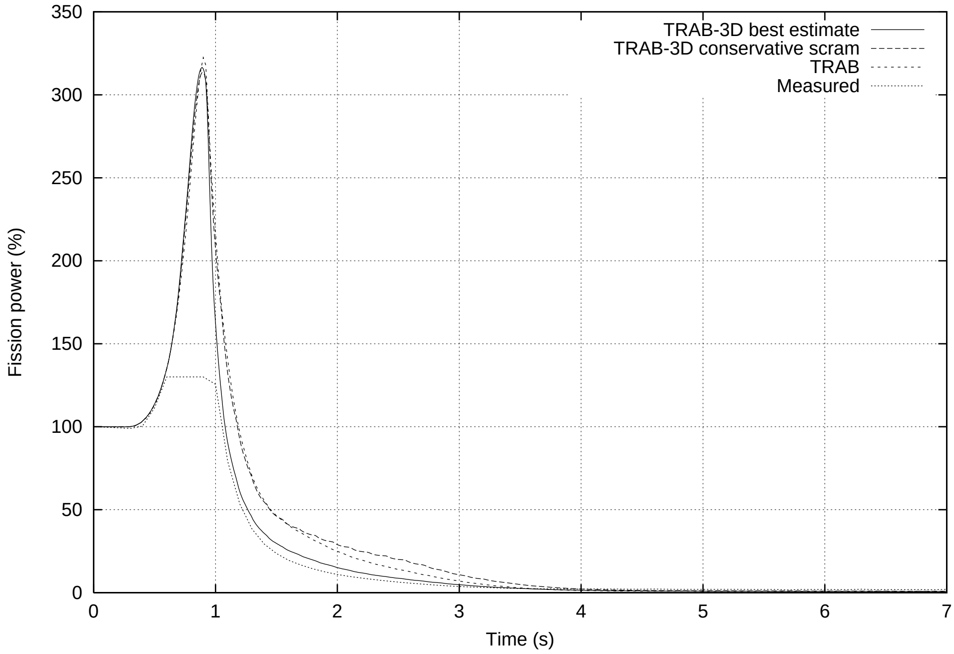


Figure 1. Calculated fission power behaviour against measurements. Note: the measurement system did not record power levels over 130%.

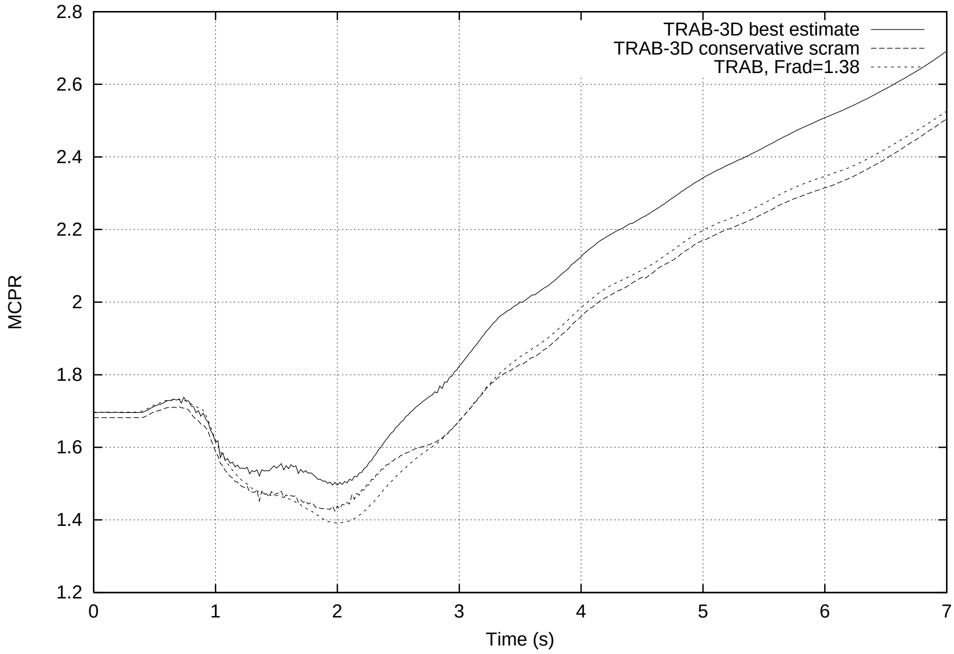


Figure 2. The minimum CPR during the transient calculated with TRAB and TRAB-3D.

5 DISCUSSION

In the investigated cases, no dramatic minimum CPR effect coming from the 3D calculation is apparent. Some cases show some decrease in the transient change of minimum CPR with the 3D calculation, which confirms the general thinking that the 1D calculation is conservative. On the other hand, the observed effect on neutron flux behaviour is quite large, especially in the partial scram case. But because all the investigated cases were fast, the transient taking only few seconds, this momentary effect is filtered out by the relatively long time constant of the fuel. In a slower transient the 3D effect might be stronger.

3D methods are not necessary in all transient analyses. Safety analysis report (SAR) type calculations that should cover a broad range of possible core loadings will probably be done with 1D methods also in near future, unless some transient is very close to a licensing limit and a more accurate analysis is needed. In this case a special licensing core could possibly be used, because the exact core composition is not known. For cycle-specific calculations, in which the core is well known, a 3D analysis could be more beneficial. It should also be noted, that the present acceptance criteria and operating limits are based on 1D analyses.

In the present 1D methodology, conservatism is applied on several different levels. Conservatism is included in physical parameters in the core itself, such as the void reactivity feedback coefficient, axial power profile and heat transfer parameters. In some cases the conservativeness of the parameter choices is difficult to assess, particularly in complicated BWR transients that include typically simultaneous power increase and coolant flow reduction. In 3D an additional complication comes from the fact that changing e.g. the heat transfer parameters immediately affects the mass flow distribution between the fuel bundles and the radial power distribution. Thus, being sure that the parameter choice is indeed conservative becomes even more difficult.

Conservatism applied to the transient scenario and the behaviour of the plant systems, with time delays between various actions, can be treated similarly in 1D and 3D analyses. Conservatism can be also included in the acceptance criteria. The control rod drop transient, which is always calculated with a 3D neutronics model is an example of a case where the acceptance criteria have been chosen rather conservatively.

One possibility coming from transferring to 3D methodology is to move from the traditional hot channel CPR calculation to a full core CPR calculation, where the critical power ratio is evaluated for each calculational node for every time step during the transient. This leads to a vast amount of data, the handling of which can be a problem in itself, but allowing also statistical treatment of the fuel bundles. The best-estimate 3D calculation with uncertainty analysis approach in general is presently the subject of several international research activities, such as the CRISSUE and VALCO EU projects.

One clear advantage of a 3D core model is to avoid the condensation of true 3D fuel bundle data into 1D approximate data. An experienced analyst can make the 1D model behave similarly to the 3D model, but correct behaviour in new types of situations is difficult to guarantee.

The results and findings of the investigations performed in connection of this activity were presented in a seminar in 2003, where the questions of 3D methodology were discussed with the representatives of the Nordic authorities and utilities. The summary of this seminar is included as Part II of this report.

Part II

6 SEMINAR SUMMARY

The NKS 3D BWR transient methodology seminar was held at VTT in April 8, 2003. In addition to the 3D transient code developers that participated in the first part of the activity, representatives of Finnish and Swedish utilities, safety authorities and a German fuel vendor contributed to the seminar. The seminar program and the list of participants can be seen in Appendices I and II, respectively. The presentations in the seminar are included as Appendix III.

The seminar was started with a general presentation of the NKS 3D BWR transient methodology activity by A. Daavittila. The presentation dealt mainly with the issues discussed and the questions identified during the meetings of Part I of the activity. This was followed by the presentations on the transient calculations of Part I. C. Jönsson (Studsvik Scandpower) discussed the calculations demonstrating the 3D effect rising from a partial scram or initially partially inserted control rods (Section 2 of this report). U. Bredolt (Westinghouse Atom) described the results of a 3D calculation of a fast flow reduction transient (Section 3) and A. Daavittila (VTT) presented the fast pressure increase transient calculations (Section 4).

The overall conclusion regarding these typical BWR transients was, that the pure 1D vs. 3D effect is not really significant, most of the conservatism in actual licensing analyses of this kind comes from the conservative assumptions made in the transient definitions (i.e. assumptions of plant behaviour and parameters). If, however, different transients (e.g. ATWS) become the limiting ones, the 3D effect could be more significant.

K. Valtonen began his presentation on STUK's view on the subject by describing the current practice of using 1D codes with conservatism. The need for 3D analysis comes from the heterogeneous cores with mixed loadings of several different highly optimised fuel assembly types, and the increasing fuel burnup. On the other hand, the developed 3D kinetics codes and modelling enhancements in e.g. fuel behaviour make the 3D methodology development currently possible. K. Valtonen emphasized the adequate validation of the 3D models, as well as the importance of moving in the direction of best-estimate calculation with uncertainty analysis.

N. Garis from SKI was not able to be at the seminar, but his presentation was distributed to the participants. The presentation continued with the theme of adequate validation with an emphasis on analysing the events that have occurred in real nuclear power plants. This requires, that the plant measurement data is saved in a proper way.

As work done recently at VTT outside of this activity, H. Räty presented the calculation of the Olkiluoto 1 load drop test perfomed in 1998, which was used as a validation case for the TRAB-3D code. The case is interesting from a 3D point of view because the test included an asymmetric partial scram and the measurement data from several local power range monitors was available for comparison.

A. Hämäläinen from VTT gave a presentation on the EU VALCO project, where best-estimate calculation methods were combined with an uncertainty methodology developed by GRS in Germany. The VALCO project deals with the VVER-440 and VVER-1000 type reactor transients.

Roger Velten from Framatome ANP described their 3D code system and its validation with experiments, plant measurements and international benchmark calculations. In Germany the authority (TÜV Nord) has already accepted a 3D calculation, which gave a reduction of 0.07 in MCPR for a pressurisation transient.

The presentations were followed by a general discussion. The overall consensus was that the tools for 3D analyses exist and there is a desire to use them, which would at least eliminate the uncertainties associated with the data conversion into one dimension. The biggest issue is the development of common rules and common methodology.

The utilities expressed their wish to move in small steps and keeping the methodology as simple as possible. There could be, naturally, cases were using a 3D methodology would lead to financial gain, but this depends greatly on the type of transient that is limiting in each case.

One problem for the validation of 3D codes is that not all plant data is freely available and publishable. This problem exists also for the proprietary fuel assembly correlations resulting from the vendors’ test facilities.

In general the participants of the seminar expressed a wish to have some kind of a Nordic forum for transient methodology discussion also in future, preferably regularly arranged meetings e.g. once a year.

APPENDIX I:

SCHEDULE FOR THE NKS 3D BWR TRANSIENT METHODOLOGY SEMINAR, APRIL 8, 2003

10.00 Opening of the seminar

10.10 General presentation of the activity (A. Daavittila/VTT)

10.20 Partial scram and partly inserted control rods (Christian Jönsson/Studsvik Scandpower)

10.50 Fast flow reduction transient (Ulf Bredolt/Westinghouse Atom)

11.20 Fast pressure increase transient (A. Daavittila/VTT)

11.50 STUK view on the subject (K. Valtonen/STUK)

12.20 Lunch

13.20 Calculation of the 1998 load rejection test with partial scram (H.Räty/VTT)

13.40 3D methodology in VVER transients with uncertainty analyses(A. Hämäläinen/VTT)

14.10 SKI view (N. Garis/SKI)

14.40 Coffee

15.00 Advanced methods for BWR transient analysis (R. Velten/Framatome ANP)

15.30 Comments from utilities and discussion

17.00 Adjourn

APPENDIX II:

NKS-R 3D BWR TRANSIENT METHODOLOGY SEMINAR 8 APR 2003

List of Participants:

Forsmark:

Pär Lansåker

Elisabeth Rudbäck

Framatome ANP:

Dieter Kreuter

Roger Velten

OKG:

Per Claesson

Marcus Johansson

Christer Netterbrandt

Göran Wiksell

SKI:

Ninos Garis

Studsvik:

Malte Edenius

Christian Jönsson

Lars Moberg

STUK:

Keijo Valtonen

Nina Lahtinen

Vattenfall:

Torbjörn Espefält

Eric Ramenblad

Irina Sitnikova

Westinghouse Atom:

Ulf Bredolt

Per Jerfsten

Henrik Nerman

Lars Paulsson

VTT:

Antti Daavittila

Anitta Hämäläinen

Randolph Höglund

Hanna Räty

TVO:

Kim Dahlbacka

Saku Latokartano

General overview of the activity

A. Daavittila

NKS 3D BWR Transient methodolgy seminar, April 8, 2003

General overview

- Participants: VTT, Studsvik Scandpower, Westinghouse Atom
- The activity has been partly funded by the NKS-R research programme
- Two meetings organized: June 4th, 2002 in Västerås and October 24th in Espoo
- Some new 1D vs. 3D calculations made, scope limited to cycle-specific CPR calculations of operational transients
- Second part: this seminar

Discussed points

- Why 3D?
  - advantages (reduction of excessive margin?)
  - more accurate knowledge of true transient behaviour (best-estimate)
  - needed in cases where fission power development and its distribution changes during the transient (RIA, ATWS, stability)
- Cycle-specific vs. SAR calculations, different levels of conservatism
- Conservativism: physical parameters (void feedback, heat transfer), plant behaviour, acceptance criteria
- CPR methodology: traditional hot channel, full core (MCPR can move), or selected channels from 3D calculation
- Higher burnup: fresh fuel bundle with high power not necessarily limiting

Discussed points

- Best-estimate + uncertainty analyses vs. traditional type of conservatism
- Difficulty of conservatism with 3D, change in e.g. heat transfer parameters leads to power and mass flow redistribution
- 3D input usually easier to make correctly, difficulty of data condensation to 1D
- Acceptance criteria and operating limits based on 1D methods

Select problem to address

1D to 3D transient methodology – how does it influence the result?

- average core to hot channel calculation mapping is avoided
- void reactivity evaluation is avoided
- local effects during partial scram is avoided
- local effects from partly inserted control rods during scram is avoided
- coolant flow redistribution effects in the power is accounted for

We are probably reducing conservatism by being more detailed when going from a 1D to a 3D method BUT DON'T FORGET:

Transients like the traditional

- 'load rejection without bypass' or

- 'turbine trip without bypass'

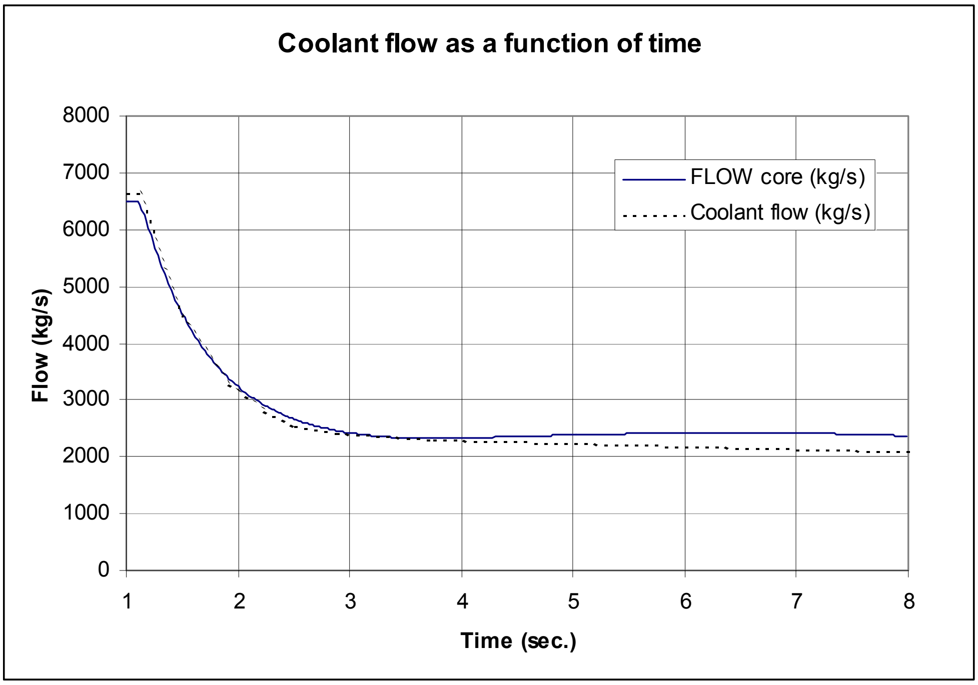
are defined with a substantial conservatism in the event description!

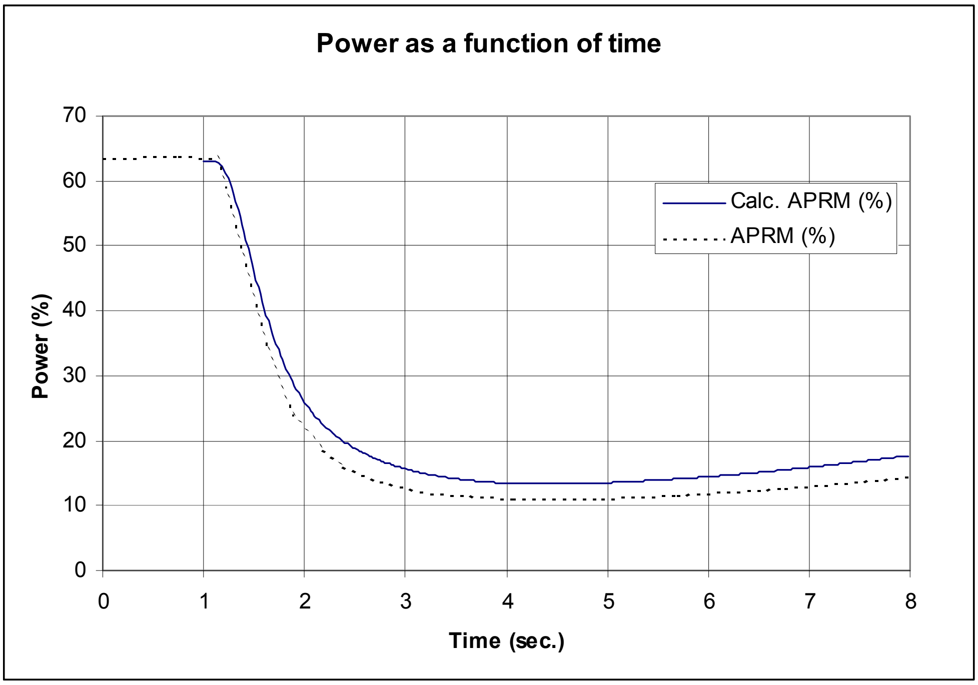
These items include:

- **valve closure time**
- **timing of valve closure in different steam lines**
- **steam lines have different lengths**
- **control rod insertion data**

Typical transient result – OL 1 c1 measurement

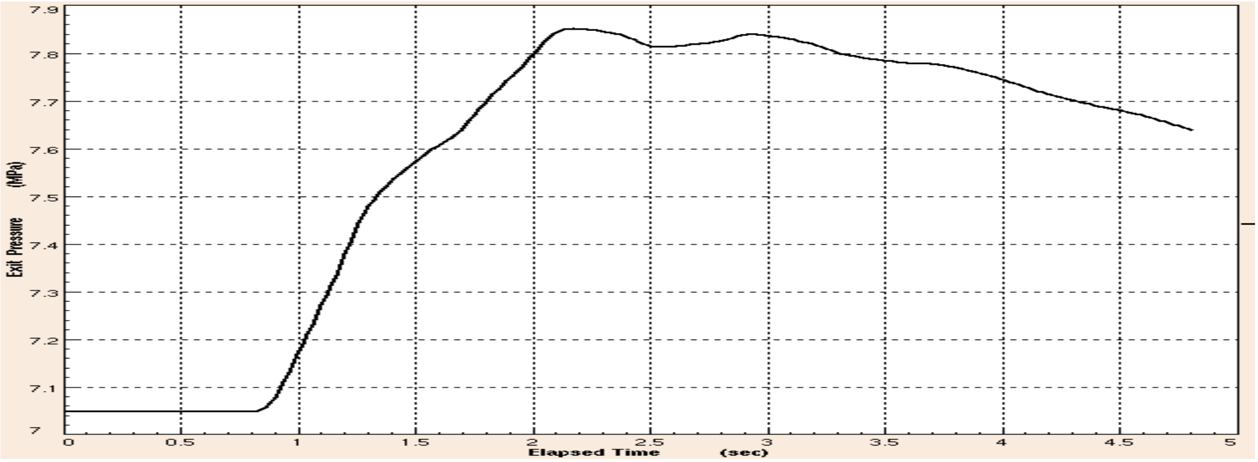
Below is an S3K calculation of the OL 1 cycle 1 total pump trip

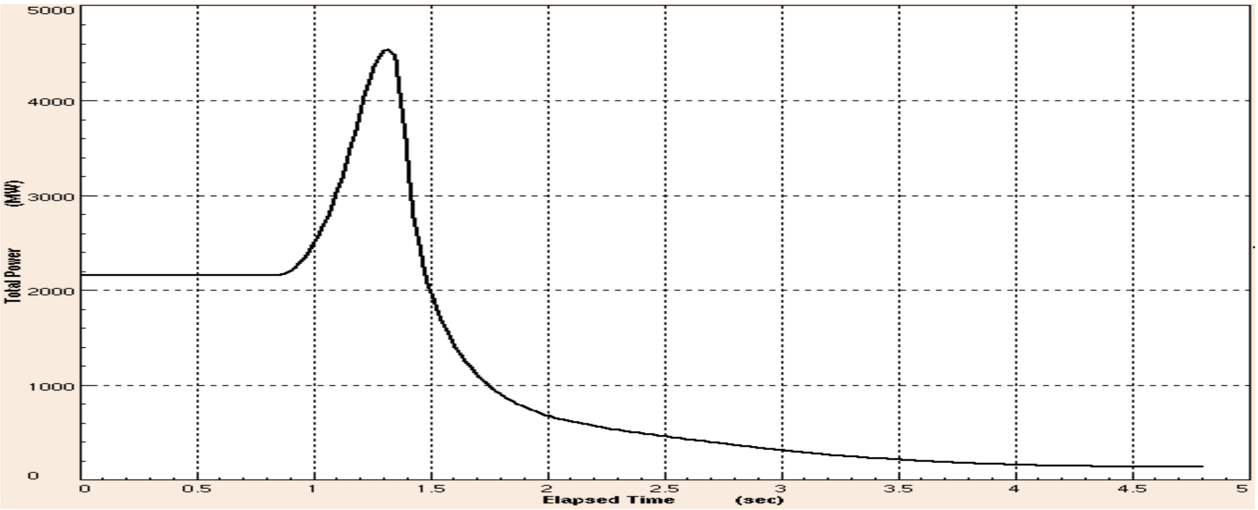


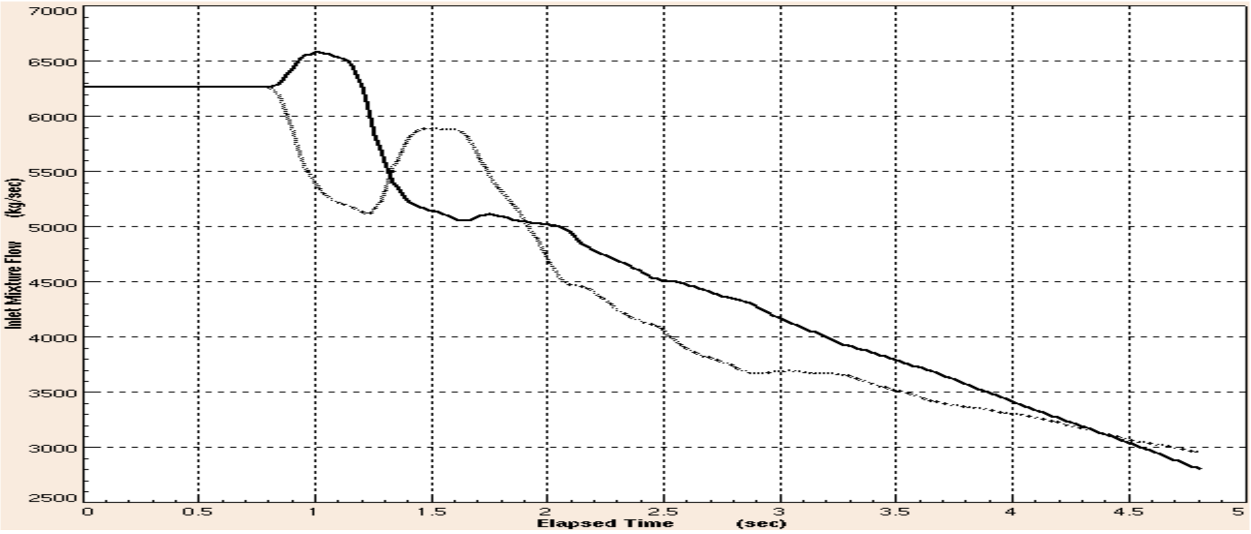


Typical transient result – OL 1 c7 event

Below is an S3K calculation of the OL 1 cycle 7 pressurization event







Improvement scenario in safety analysis

Scenario

- analysis are made with an old tool – validated against reasonable events

- point-kinetic or 1D methods seem typical for 'global' core events

- limitations of the methods put restriction on the extrapolation of the result

Example of improvement

- stability has, historically, been evaluated with different approximations of the core

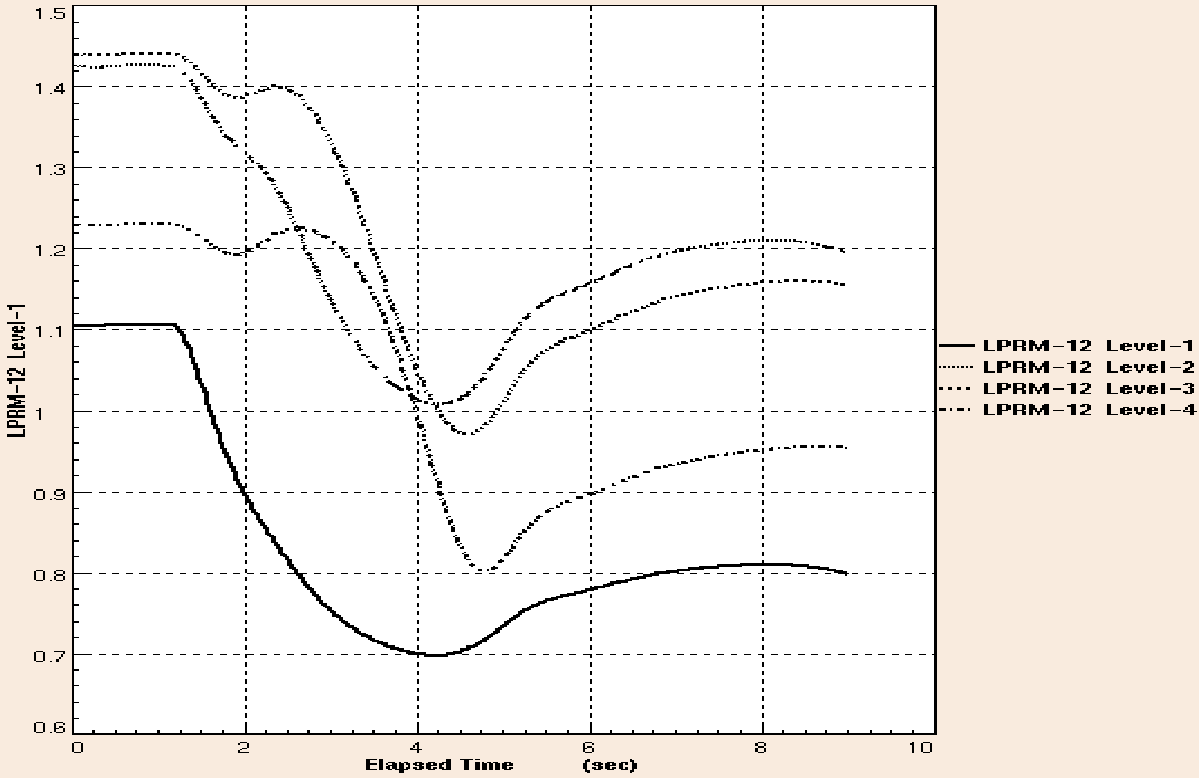
- most organizations have started to use 3D codes
  IF NOT
  the analysis can be made with large conservative measures

  why?

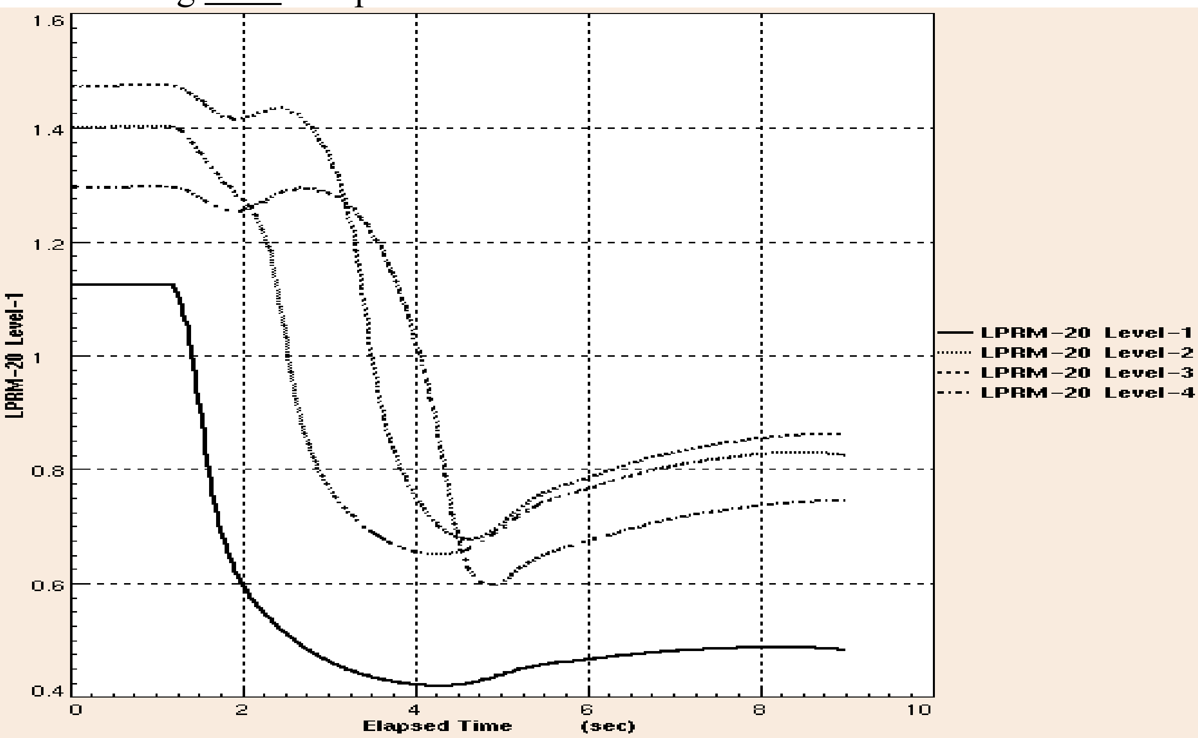
The local effects are or have been more and more obvious. The approximation because of limited core model is reduced.

Partial control rod insertion

LPRM string far away from any partial scram rods



LPRM string close to a partial scram rod



3D effects in typical limiting transients

Partial scram/selected rod insertion/PULK Einfahren

- reduces power without getting a shut down
- influences the vicinity of the inserted rods
- the influence is dependent of the radial core power distribution

Movement of partially inserted control rods during (full) scram

- gives an efficient initial power reduction
- the influence is dependent on control rod pattern

⇨ how big is the influence?

Partial scram example

(time = 0.0 sec.)

MCPR - partial scram - time zero

1.84	1.31	1.76	1.50	1.48	**1.20**	1.51	1.63	1.64	1.33	1.83	1.47	1.99	4.05	7.97
1.81	1.78	1.72	1.24	1.66	1.35	1.68	1.25	1.79	1.45	1.86	1.50	2.62	4.23	8.11
1.26	1.75	1.40	1.72	1.25	1.69	1.53	1.48	1.24	1.79	1.55	1.72	2.27	4.79	10.00
1.47	1.56	1.26	1.75	1.57	1.55	1.23	1.49	1.52	1.40	1.41	1.87	3.57	5.90	10.00
1.46	1.22	1.59	1.62	1.83	1.27	1.69	1.50	1.52	1.30	2.03	2.96	4.43		
1.66	1.37	1.73	1.28	1.76	1.58	1.41	1.72	1.27	1.88	1.78	3.28	5.52		
1.26	1.74	1.59	1.57	1.26	1.60	1.76	1.29	1.80	1.42	1.88	3.55	6.16		
1.79	1.57	1.26	1.54	1.56	1.57	1.30	1.90	1.59	1.72	2.05	4.11	8.09		
1.70	1.30	1.55	1.56	1.56	1.30	1.71	1.58	2.15	1.71	3.12	5.10	10.00		
1.81	1.48	1.87	1.47	1.36	1.76	1.68	1.80	1.73	2.24	4.35	7.00			
1.35	1.89	1.57	1.46	2.16	1.97	2.75	2.72	2.49	3.97	6.01				
1.63	1.51	1.79	1.94	2.90	3.67	4.10	4.55	5.25	6.93					
2.06	2.74	2.29	3.67	4.65	6.04	7.39	8.71	10.00						
4.22	4.35	4.71	6.08											
8.13	8.64	10.00	10.00											

(time = min CPR)

MCPR - partial scram - at minimum MCPR

1.82	1.30	1.74	1.48	1.45	**1.17**	**1.46**	1.60	1.62	1.32	1.82	1.47	1.98	4.05	7.97
1.80	1.76	1.71	1.23	1.64	1.33	1.65	1.24	1.77	1.44	1.86	1.50	2.61	4.24	8.11
1.25	1.73	1.39	1.70	1.24	1.67	1.51	1.47	1.23	1.79	1.54	1.72	2.27	4.79	10.00
1.45	1.54	1.25	1.74	1.56	1.54	1.22	1.48	1.51	1.40	1.40	1.87	3.57	5.90	10.00
1.44	1.20	1.58	1.60	1.81	1.26	1.68	1.49	1.51	1.30	2.02	2.96	4.43		
1.61	1.35	1.71	1.27	1.75	1.57	1.41	1.71	1.26	1.87	1.77	3.27	5.53		
1.23	1.71	1.58	1.56	1.25	1.58	1.75	1.29	1.79	1.41	1.87	3.54	6.17		
1.77	1.55	1.25	1.53	1.55	1.56	1.29	1.88	1.57	1.70	2.04	4.10	8.09		
1.69	1.29	1.54	1.55	1.56	1.29	1.70	1.56	2.13	1.69	3.11	5.10	10.00		
1.81	1.47	1.86	1.46	1.35	1.75	1.67	**1.76**	**1.69**	2.22	4.34	6.99			
1.35	1.89	1.56	1.45	2.15	1.96	2.73	**2.69**	**2.46**	3.95	6.00				
1.63	1.51	1.78	1.93	2.90	3.66	4.09	4.54	5.23	6.91					
2.05	2.73	2.29	3.67	4.65	6.04	7.38	8.70	10.00						
4.22	4.35	4.71	6.08											
8.13	8.64	10.00	10.00											

Partial scram example

Local analysis concept (ΔCPR)

partial scram delta CPR

0.02	0.01	0.02	0.02	0.03	**0.03**	**0.05**	0.02	0.01	0.01	0.01	0.00	0.01	0.00	-0.01
0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00
0.01	0.02	0.01	0.02	0.01	0.02	0.02	0.01	0.01	0.01	0.01	0.00	0.01	0.00	0.00
0.02	0.02	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.00	0.00	0.00
0.02	0.01	0.02	0.01	0.01	0.01	0.01	0.01	0.01	0.00	0.01	0.00	0.00		
0.05	0.02	0.02	0.01	0.01	0.01	0.01	0.01	0.00	0.01	0.01	0.00	0.00		
0.03	0.02	0.02	0.01	0.01	0.01	0.01	0.00	0.01	0.01	0.01	0.00	0.00		
0.02	0.02	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.01	0.01	0.00	0.00		
0.01	0.01	0.01	0.01	0.01	0.01	0.02	0.02	0.02	0.02	0.01	0.00	0.00		
0.01	0.01	0.01	0.01	0.01	0.01	0.02	**0.04**	**0.04**	0.02	0.01	0.00			
0.00	0.01	0.01	0.01	0.01	0.01	0.02	**0.03**	**0.03**	0.02	0.01				
0.00	0.01	0.01	0.01	0.01	0.00	0.01	0.01	0.01	0.01					
0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00						
0.00	0.00	0.00	0.00											
-0.01	0.00	0.00	0.00											

Average analysis concept (ΔCPR)

partial delta CPR - averaged pss

0.03	0.01	0.02	0.02	0.02	**0.01**	**0.02**	0.02	0.02	0.01	0.02	0.01	0.01	0.00	0.00
0.03	0.03	0.03	0.01	0.02	0.01	0.02	0.01	0.02	0.01	0.02	0.01	0.01	0.00	0.00
0.01	0.03	0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.02	0.01	0.01	0.00	0.00
0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.02	0.01	0.01	0.02	0.01	0.00	0.00
0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.01	0.00		
0.02	0.01	0.02	0.01	0.02	0.02	0.01	0.02	0.01	0.02	0.01	0.01	0.00		
0.01	0.02	0.02	0.02	0.01	0.02	0.02	0.01	0.02	0.01	0.02	0.01	0.00		
0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.01	0.01	0.00		
0.02	0.01	0.02	0.02	0.02	0.01	0.02	0.01	0.02	0.01	0.01	0.00	0.00		
0.02	0.01	0.02	0.01	0.01	0.02	0.01	**0.02**	**0.01**	0.01	0.00	0.00			
0.01	0.01	0.01	0.01	0.01	0.01	0.01	**0.01**	**0.01**	0.01	0.00				
0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00					
0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00						
0.00	0.00	0.00	0.00											
0.00	0.00	0.00	0.00											

Partial scram example

Local analysis concept (Δsteam quality)

diff - steam qualities (step97-step1)*1000.

0.2	1.2	0.3	0.5	-0.3	**-1.9**	**-2.8**	-0.2	0.7	1.5	0.8	1.4	0.7	0.1	0.0
0.3	0.3	0.4	1.7	-0.4	0.5	0.1	1.2	0.4	1.7	0.7	1.3	0.4	0.1	0.0
1.9	0.3	1.4	0.6	1.4	0.3	0.3	1.1	2.2	0.8	1.4	1.3	0.3	0.1	0.0
0.8	0.5	1.6	0.4	0.5	0.6	2.1	1.3	1.4	2.5	1.7	0.8	0.2	0.0	0.0
0.3	1.5	0.3	0.5	0.4	1.9	0.6	1.6	1.6	2.2	0.5	0.1	0.1		
-2.3	0.6	0.3	1.6	0.6	0.9	2.0	0.9	2.3	0.8	0.9	0.2	0.1		
-1.9	-0.2	0.3	0.8	2.0	0.8	0.8	1.9	0.7	1.4	0.7	0.1	0.0		
-0.1	0.2	1.6	1.2	1.1	1.0	1.8	0.4	0.8	0.5	0.3	0.1	0.0		
0.4	1.5	0.8	1.1	1.2	2.1	0.6	0.3	0.1	0.4	0.2	0.1	0.0		
0.6	1.4	0.5	1.7	1.6	0.8	0.5	**-1.3**	**-1.0**	-0.2	0.1	0.0			
1.8	0.8	1.2	1.3	0.5	0.3	0.3	**0.1**	**-0.3**	0.1	0.0				
1.6	1.1	1.0	0.6	0.2	0.2	0.1	0.1	0.1	0.0					
0.7	0.4	0.4	0.1	0.0	0.1	0.0	0.0	0.0						
0.1	0.1	0.1	0.0											
0.0	0.0	0.0	0.0											

Partial scram example

Summary

- **The 3D impact of the PSS/SRI/PULK is pronounced from a neutron flux standpoint BUT <u>small from a ΔCPR standpoint</u>**

- **The average power reduction is obvious**

- **One licensing transient – pump trip with a partial scram in an internal pump reactor was analysed: PSS too late to influence the ΔCPR**

⇨ **the 1D or 3D core model assumptions is less important**
⇨ **the 1D model assumption does, however, demand more of the user in order to master the problem of partial scram reactivity**

Partly inserted control rods + scram example

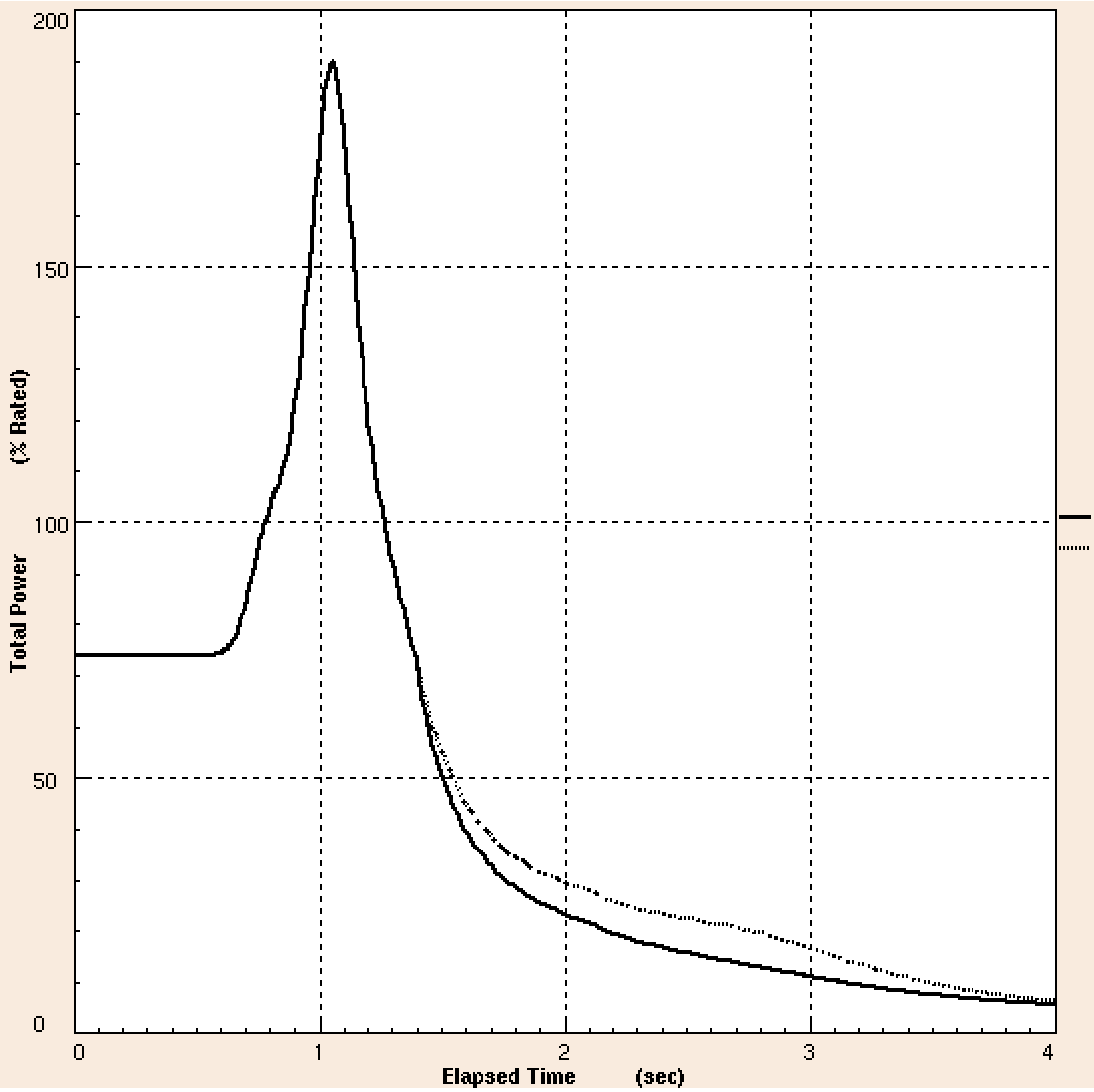
- Fast pressure increase transient
- Internal pump reactor
- Limiting case is typically in the middle of the cycle
- Control rod pattern is (withdrawn %):

```
            --   --   --
       -- -- --   --   -- -- --
    -- -- 87 --   4   -- 93 -- --
  -- -- -- -- --   --   -- -- -- -- --
  -- 93 --  4 --   43   --  4 -- 87 --
-- -- -- -- -- --   --   -- -- -- -- -- --

-- --  4 -- 43 --   4   -- 43 --  4 -- --

-- -- -- -- -- --   --   -- -- -- -- -- --
  -- 87 --  4 --   43   --  4 -- 93 --
  -- -- -- -- --   --   -- -- -- -- --
    -- -- 93 --   4   -- 87 -- --
       -- -- --   --   -- -- --
            --   --   --
```

Partly inserted control rods + scram example

Calculated APRM as a function of time – (low power and flow).

(solid line is scram using all rods,
dashed line is scram using all totally withdrawn rods)



Partly inserted control rods + scram example

(time = 0.0 sec.)

Initial CPR

2.47	1.65	1.31	**1.90**	**1.94**	1.61	1.38	**2.36**	**2.48**	1.54	1.88	2.64	4.80
1.52	1.45	1.48	1.28	1.42	1.26	1.59	1.36	1.51	1.59	1.75	2.96	5.17
1.31	1.53	1.26	1.44	1.31	1.52	1.25	1.36	1.29	1.45	1.96	3.60	5.78
1.92	1.28	1.43	**2.32**	**2.34**	1.30	1.34	**1.39**	**1.63**	1.74	2.58	4.18	
1.92	1.40	1.34	**2.31**	**2.35**	1.65	1.22	**1.57**	**1.60**	1.87	3.18		
1.44	1.24	1.49	1.30	1.43	1.25	1.36	1.33	1.71	2.35	4.02		
1.38	1.54	1.23	1.47	1.24	1.48	1.29	1.60	2.25	3.22	5.11		
2.39	1.35	1.52	**1.45**	**1.59**	1.34	1.60	1.86	2.87	4.24			
2.52	1.53	1.33	**1.67**	**1.62**	1.72	1.98	2.82	4.15				
1.55	1.59	1.47	1.69	1.80	2.29	3.11	4.05					
1.86	1.76	2.01	2.43	3.10	3.91	4.94						
2.84	2.96	3.61	4.10									
4.49	5.28	5.75										

(time = min CPR)

all banks - mcpr minimum

2.38	1.35	0.97	**1.74**	**1.79**	1.25	1.04	**2.16**	**2.31**	1.16	1.46	2.55	5.53
1.17	1.08	1.12	0.92	1.01	0.86	1.21	1.00	1.04	1.12	1.41	2.98	6.15
0.98	1.19	0.89	1.02	0.93	1.12	0.87	0.91	0.87	1.06	1.59	3.78	10.00
1.77	0.91	1.01	**2.15**	**2.16**	0.91	0.89	**0.94**	**1.25**	1.32	2.49	4.64	
1.75	1.00	1.00	**2.12**	**2.17**	1.29	0.80	**1.19**	**1.18**	1.54	3.21		
1.03	0.85	1.10	0.92	1.00	0.84	0.92	0.94	1.30	2.19	4.38		
1.04	1.15	0.83	1.08	0.85	1.08	0.89	1.16	2.00	3.32	6.15		
2.21	0.96	1.12	**1.06**	**1.29**	0.97	1.19	1.51	2.85	4.79			
2.35	1.08	0.94	**1.32**	**1.23**	1.35	1.64	2.78	4.54				
1.18	1.13	1.09	1.29	1.49	2.11	3.15	4.47					
1.43	1.45	1.63	2.29	3.10	4.19	5.87						
2.82	2.98	3.77	4.50									
5.15	6.30	10.00										

Partly inserted control rods + scram example

Local analysis concept (ΔCPR)

delta CPR - all banks

0.08	0.30	0.34	**0.16**	**0.16**	0.35	0.34	**0.20**	**0.18**	0.38	0.42	0.09	-0.73
0.36	0.37	0.36	0.36	0.40	0.39	0.38	0.36	0.47	0.47	0.33	-0.02	-0.98
0.34	0.35	0.37	0.41	0.38	0.40	0.38	0.45	0.42	0.40	0.38	-0.18	-4.22
0.16	0.36	0.42	**0.17**	**0.18**	0.39	0.45	**0.45**	**0.39**	0.42	0.09	-0.46	
0.16	0.41	0.34	**0.19**	**0.18**	0.36	0.42	**0.39**	**0.42**	0.32	-0.04		
0.40	0.39	0.39	0.38	0.43	0.41	0.44	0.39	0.41	0.15	-0.35		
0.34	0.39	0.40	0.39	0.39	0.40	0.40	0.43	0.24	-0.10	-1.04		
0.19	0.38	0.40	**0.38**	**0.30**	0.37	0.41	0.35	0.02	-0.55			
0.17	0.45	0.39	**0.35**	**0.39**	0.37	0.34	0.04	-0.40				
0.37	0.46	0.37	0.40	0.30	0.18	-0.04	-0.42					
0.43	0.31	0.38	0.14	0.00	-0.28	-0.94						
0.02	-0.01	-0.16	-0.40									
-0.66	-1.02	-4.25										

Average analysis concept – partial rods don’t move (ΔCPR)

delta cpr - totally withdrawn banks

0.11	0.32	0.35	**0.20**	**0.20**	0.37	0.35	**0.22**	**0.19**	0.38	0.42	0.10	-0.69
0.38	0.39	0.38	0.37	0.42	0.40	0.39	0.36	0.47	0.47	0.33	-0.01	-0.93
0.35	0.36	0.38	0.42	0.38	0.41	0.38	0.46	0.42	0.39	0.38	-0.16	-4.22
0.19	0.37	0.43	**0.19**	**0.20**	0.39	0.45	**0.44**	**0.38**	0.42	0.09	-0.43	
0.20	0.42	0.34	**0.21**	**0.19**	0.37	0.42	**0.38**	**0.41**	0.32	-0.02		
0.41	0.40	0.40	0.38	0.43	0.41	0.44	0.39	0.40	0.16	-0.34		
0.35	0.40	0.40	0.38	0.39	0.40	0.40	0.43	0.25	-0.09	-0.99		
0.20	0.39	0.39	**0.36**	**0.33**	0.37	0.41	0.35	0.02	-0.54			
0.18	0.45	0.39	**0.33**	**0.37**	0.36	0.33	0.04	-0.37				
0.37	0.46	0.37	0.40	0.29	0.17	-0.04	-0.41					
0.43	0.31	0.38	0.14	0.01	-0.27	-0.89						
0.02	-0.01	-0.14	-0.38									
-0.63	-0.96	-4.25										

Partly inserted control rods + scram example

Difference between assumptions (ΔCPR*100)

diff between min cpr maps (*100)

3	2	1	**4**	**4**	1	1	**2**	**1**	0	0	1	4
2	2	2	1	1	1	1	0	0	0	0	1	6
1	2	1	1	1	1	0	0	0	0	0	2	0
4	1	1	**2**	**2**	0	0	**-1**	**-1**	0	1	2	
3	1	1	**2**	**1**	1	0	**-1**	**-1**	0	1		
1	1	1	0	0	0	0	0	0	0	2		
0	1	0	0	0	0	0	0	0	1	5		
1	0	0	**-2**	**3**	-1	0	0	0	2			
1	0	0	**-2**	**-3**	-1	-1	0	2				
0	0	0	-1	-1	-1	0	1					
0	0	0	0	1	1	4						
1	1	2	2									
3	6	0										

(Grey boxes represent min CPR bundles)

Partly inserted control rods + scram example

Summary

- The 3D impact of the PSS/SRI/PULK is pronounced from a neutron flux standpoint
- The impact on ΔCPR is important for a small number of bundles

⇨ the 1D or 3D core model assumptions is important for bundles close to the partially inserted control rods

Summary

- there are significant 3D effects in the typical fast transient licensing evaluation

- the influence on the important ΔCPR is limited to a few bundles

⇨ the 1D or 3D core model assumptions important when the limiting bundle appear close to partial control rod or a partly inserted control rod

NKS 3D Methodology

Fast flow reduction transient

presented by

Ulf Bredolt

Westinghouse Atom

April 8, 2003

NKS 3D Methodology

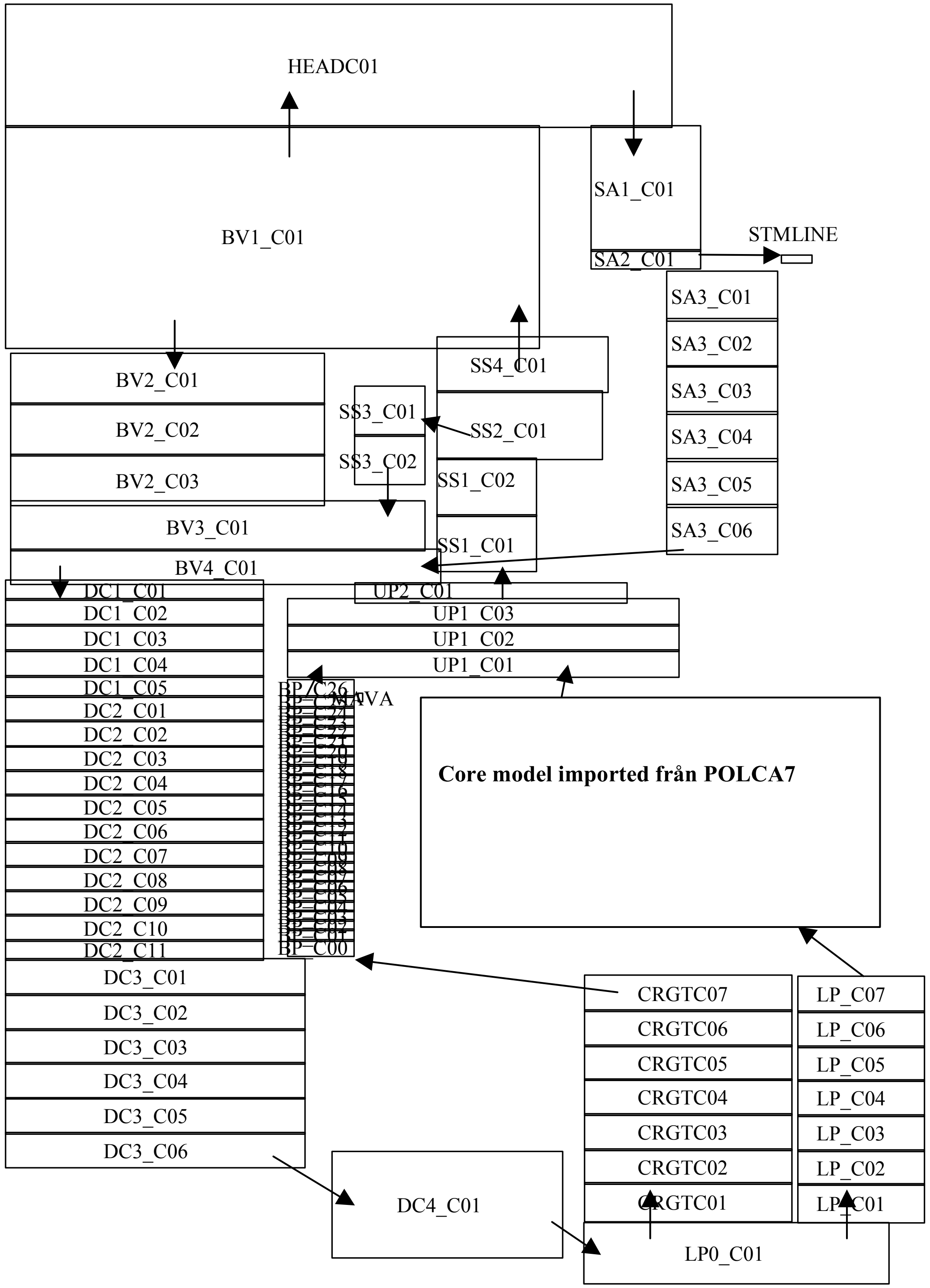
- Flow decrease transient
  - Pumptrip
- Possible 3D effects
  - Moving min CPR location
  - Flow redistribution

NKS 3D Methodology

- Simulation tool
  - POLCA-T
    - A coupled 3D neutronic-thermal hydraulic code
    - Version 1.0.0
- Simulation model
  - Plant
    - TVO 2
    - RPV + all internals + steam lines
    - Quarter symmetric core (fictious cycle)

NKS 3D Methodology

- Simulation model of TVO2 for POLCA-T

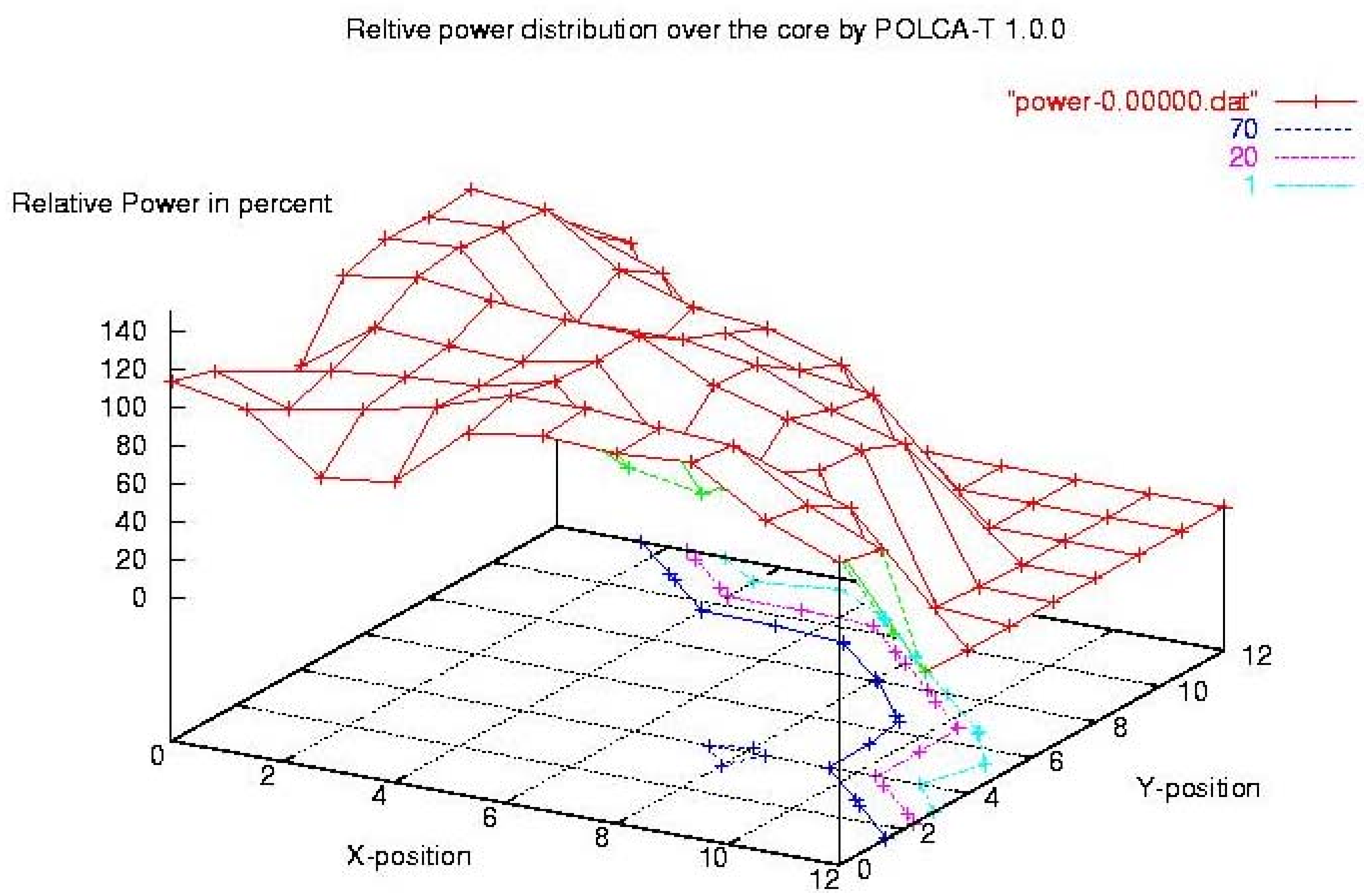


NKS 3D Methodology

- Computation model
  - TVO 2
    - Fission power 2160 MW
    - SVEA 64 fuel core
    - Reactor Pressure Vessel with internals
    - CPR correlation AA 74

NKS 3D Methodology

- Power distribution at full power



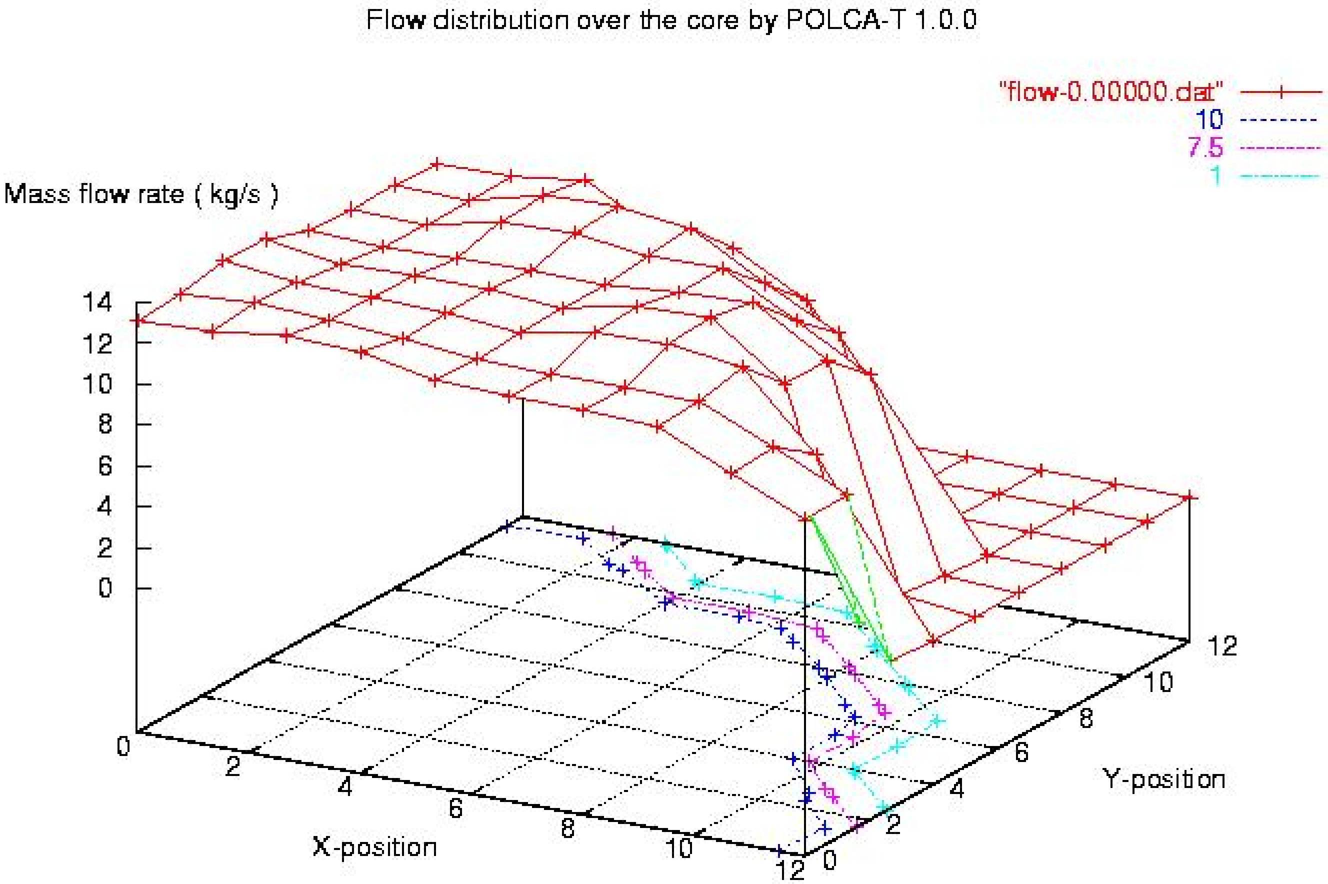
NKS 3D Methodology

- Control rod pattern

```
100 100 100
100 100 100 100 100 100 100
100 100   2 100  73 100  21 100 100
100 100  93 100 100 100 100 100  93 100 100
100  21 100  51 100  11 100  51 100   2 100
100 100 100 100 100  92 100  36 100 100 100 100 100
100 100  73 100  11 100 100 100  11 100  73 100 100
100 100 100 100 100  36 100  92 100 100 100 100 100
100   2 100  51 100  11 100  51 100  21 100
100 100  93 100 100 100 100 100  93 100 100
100 100  21 100  73 100   2 100 100
100 100 100 100 100 100 100
100 100 100
```

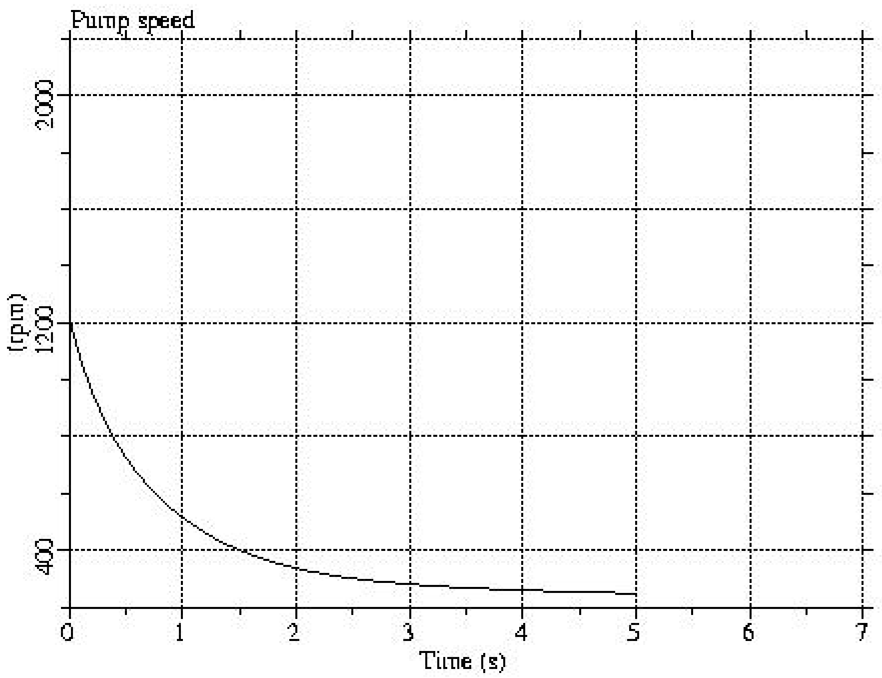
NKS 3D Methodology

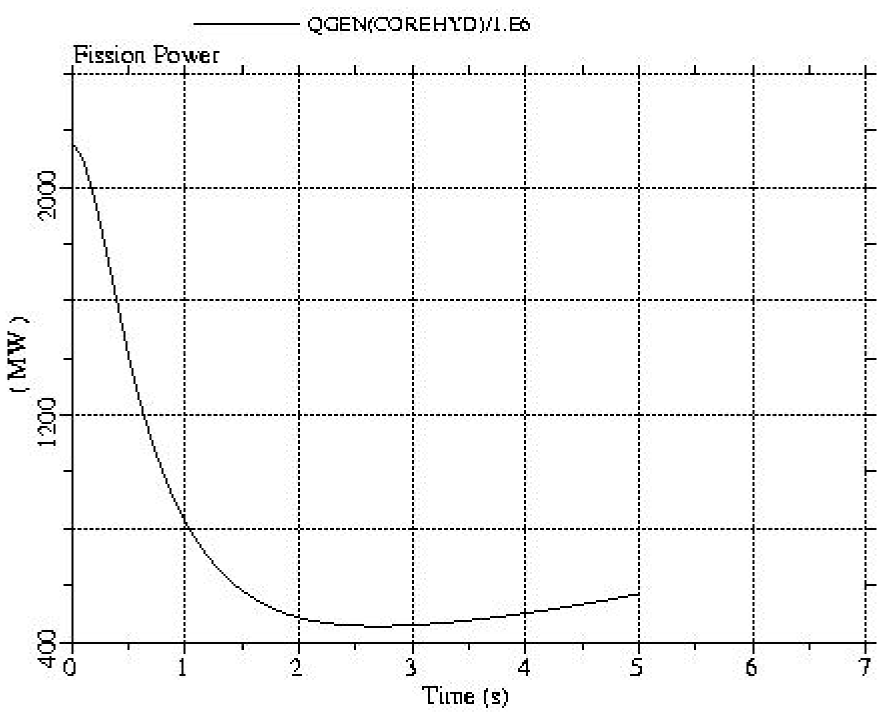
- Bundle flow distribution at time 0.0



NKS 3D Methodology

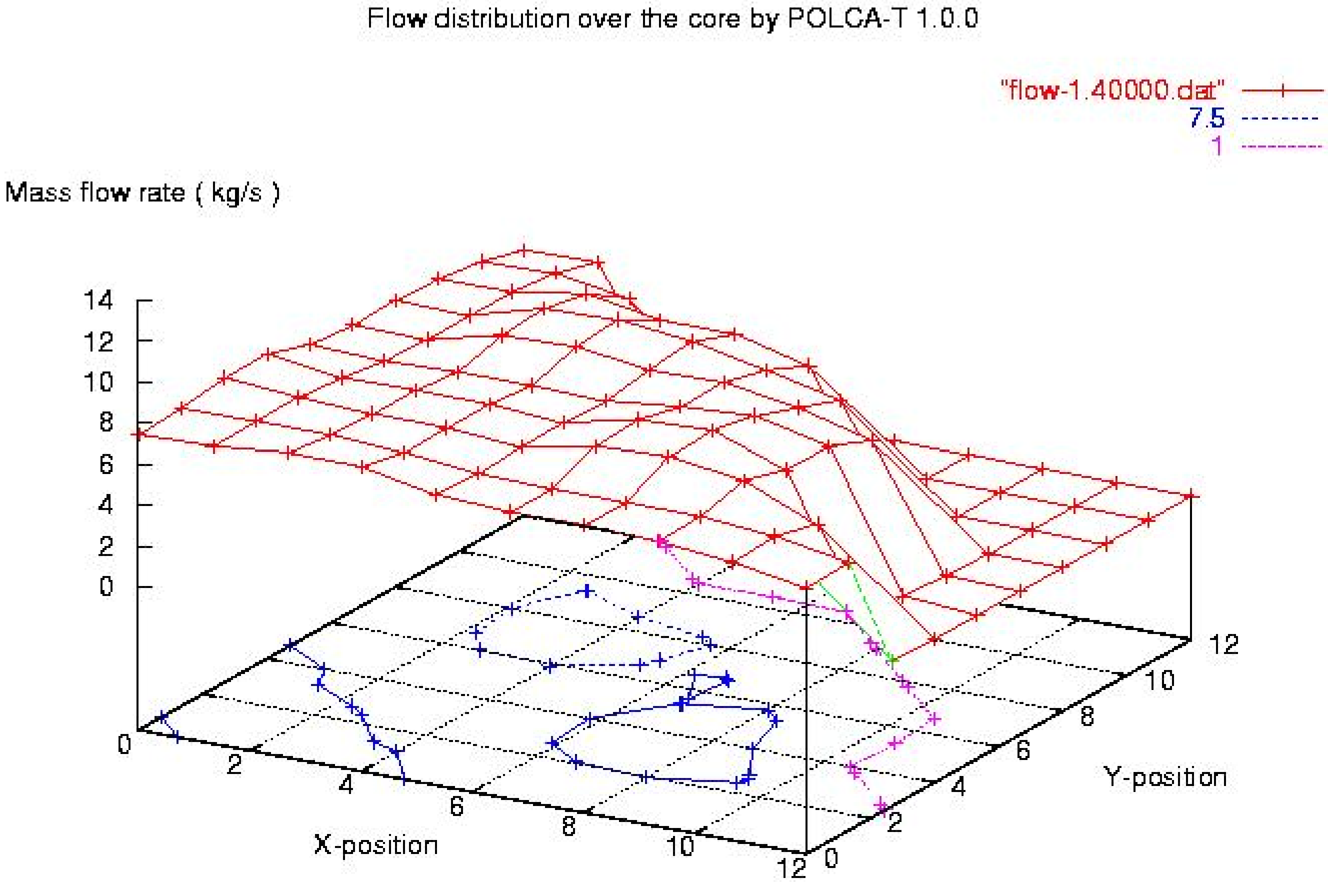
- Pump coast down & Fission power





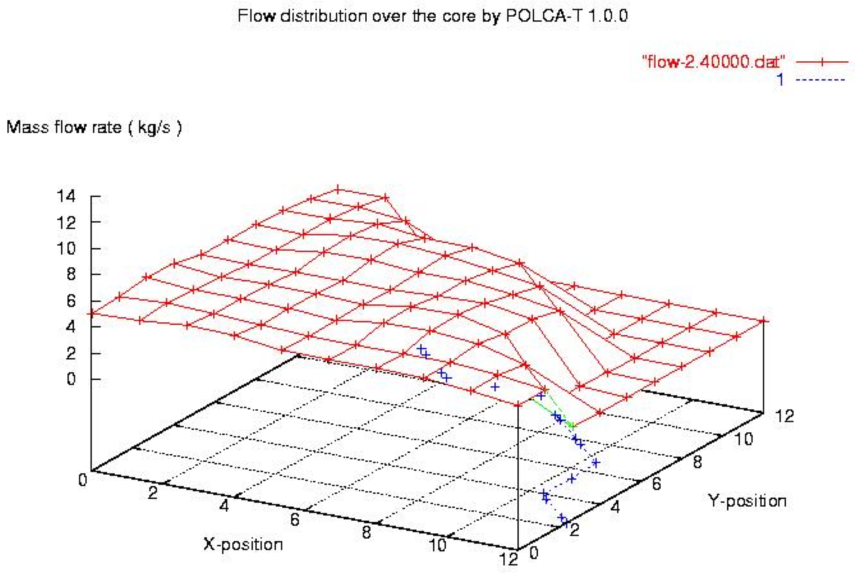
NKS 3D Methodology

- Bundle flow distribution at time 1,4 seconds



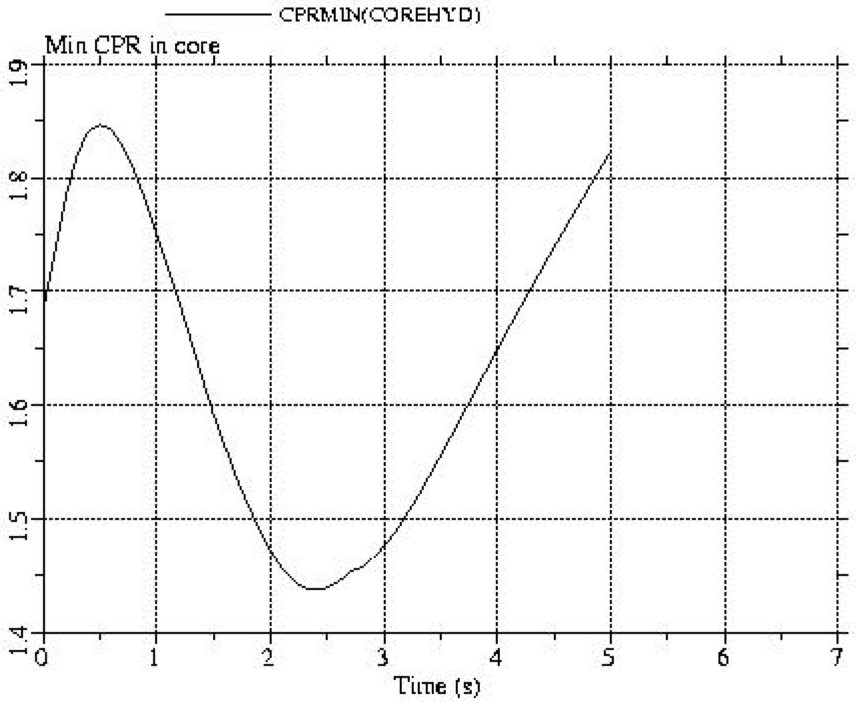
NKS 3D Methodology

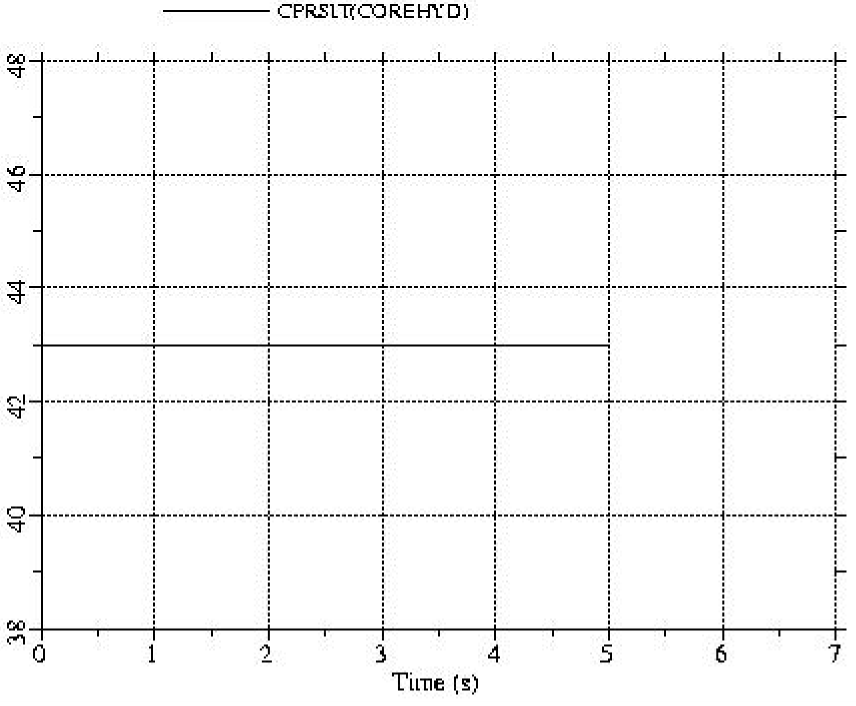
- Bundle flow distribution at time 2,4 seconds



NKS 3D Methodology

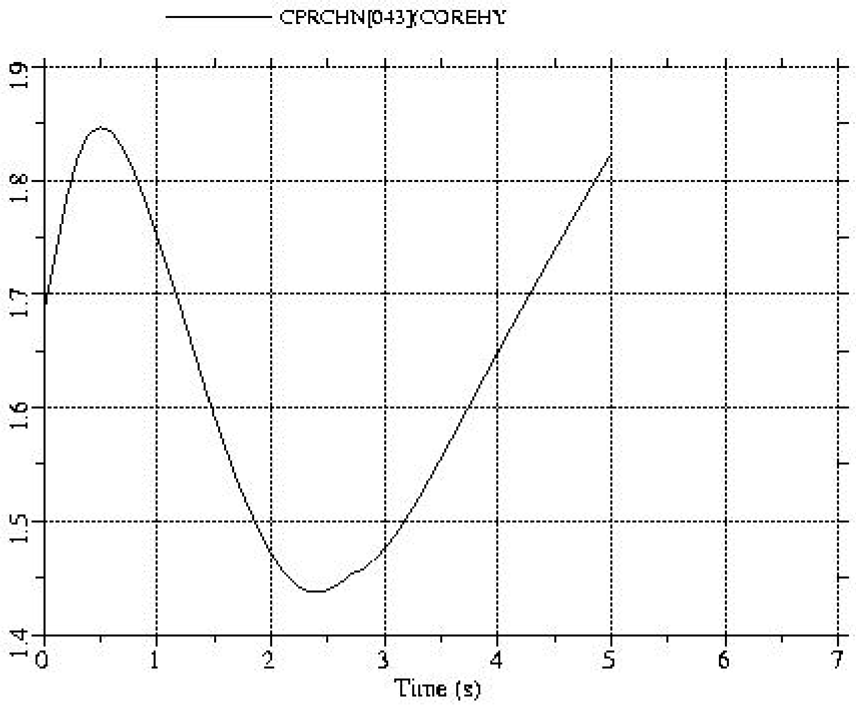
- Minimal CPR and its Bundle

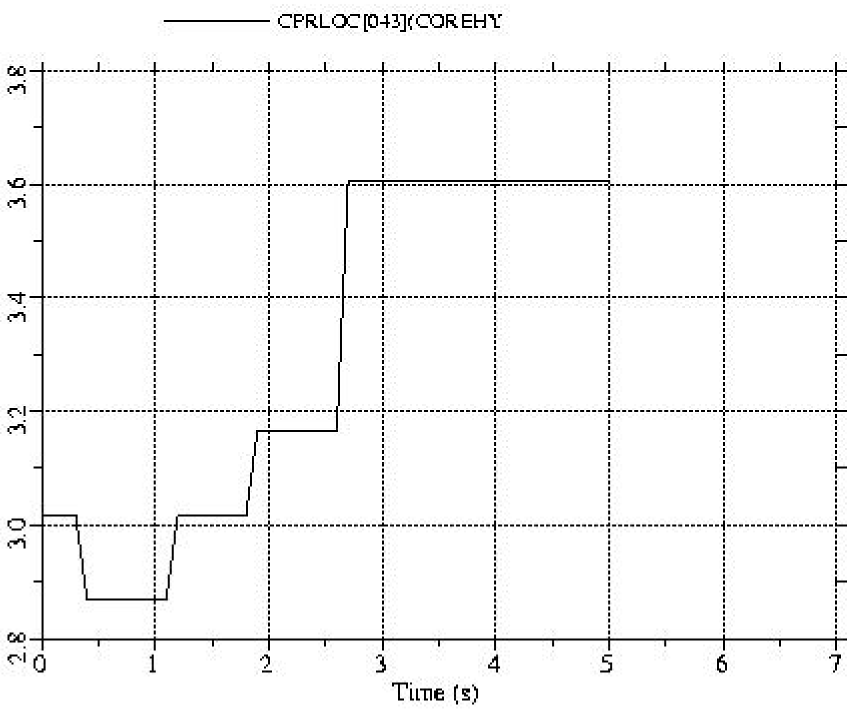




NKS 3D Methodology

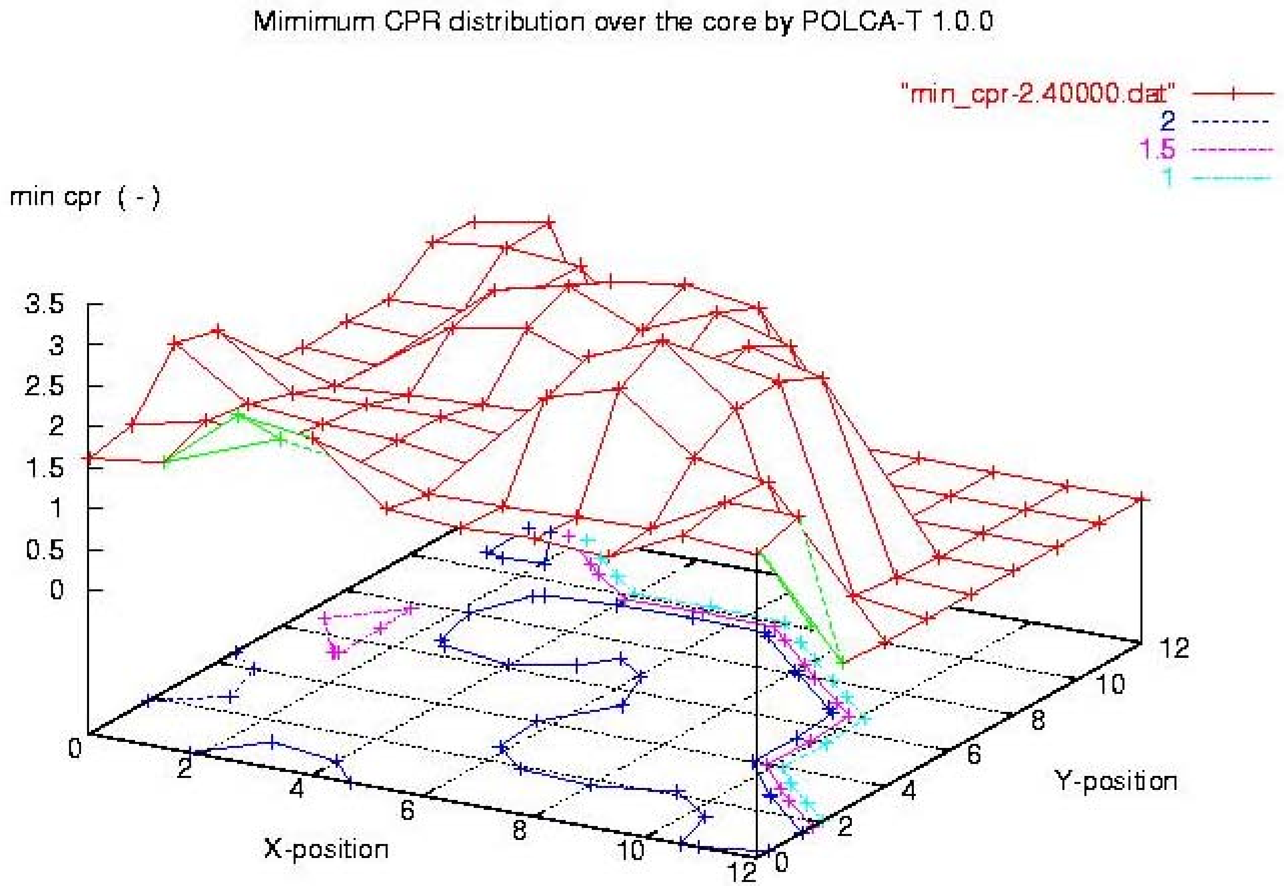
- Minimal CPR and its location in Bundle 43





NKS 3D Methodology

- Minimal CPR distribution over the core at time 2.4 seconds

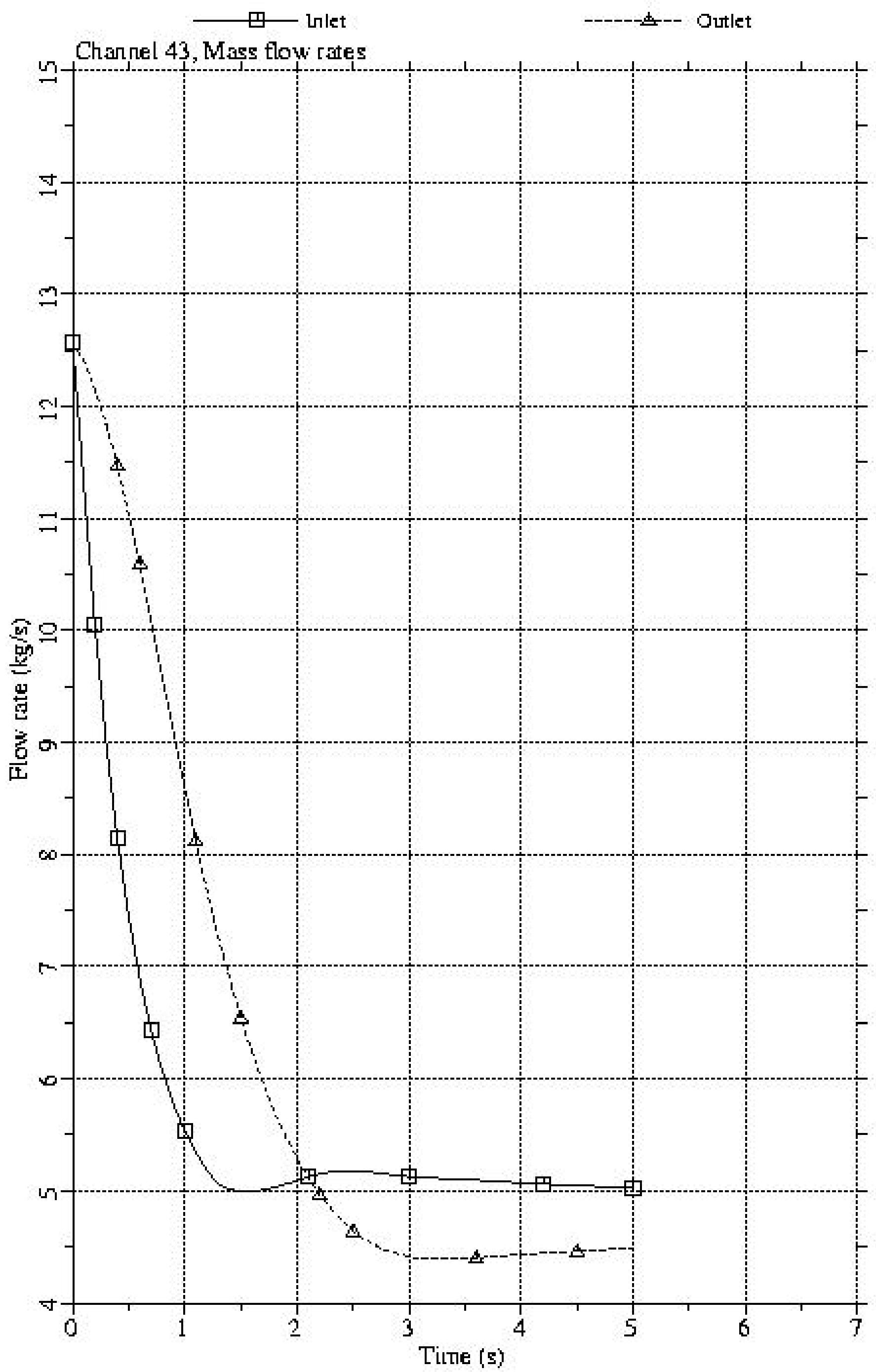


BNFL

Westinghouse

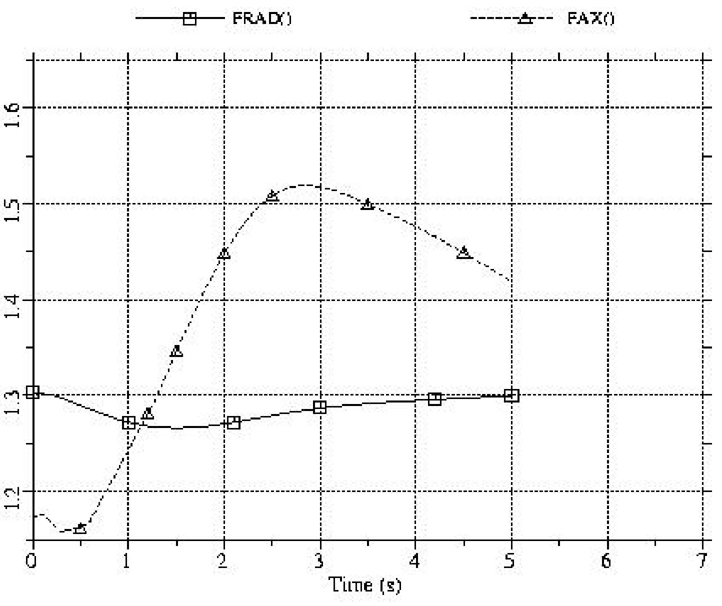
NKS 3D Methodology

- Bundle 43 mass flow rates inlet/outlet



NKS 3D Methodology

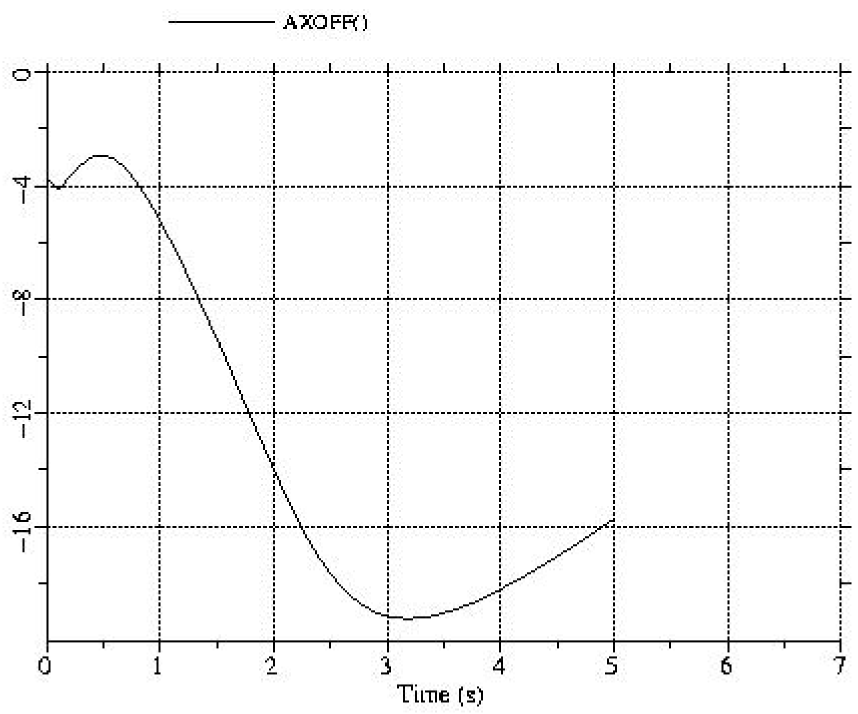
- Nuclear overall Shape factors F_{rad} and F_{ax}



- The location is fixed over the simulated time

NKS 3D Methodology

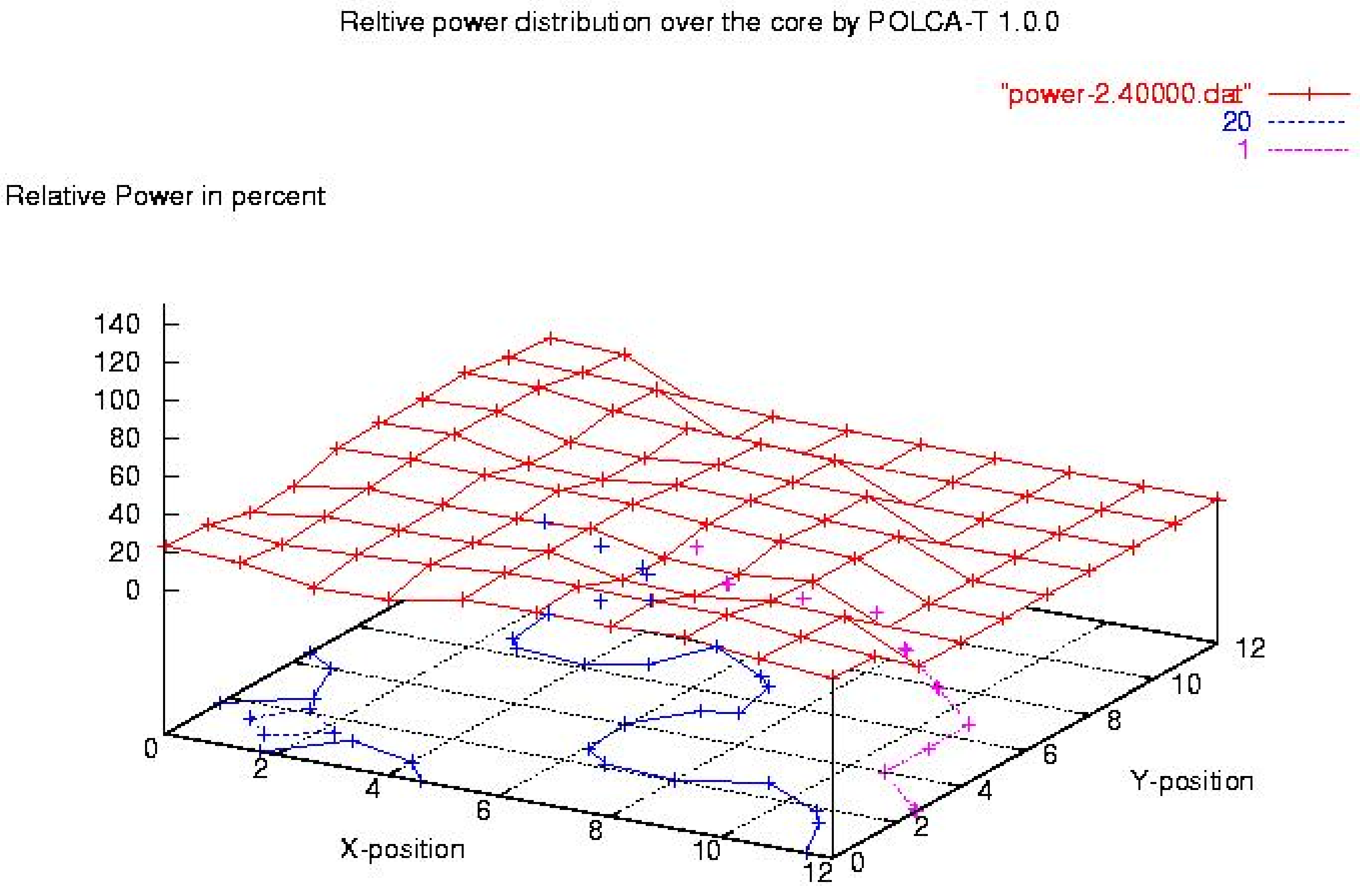
- Axial power off set



- Shows a power redistribution, the power is pushed downward

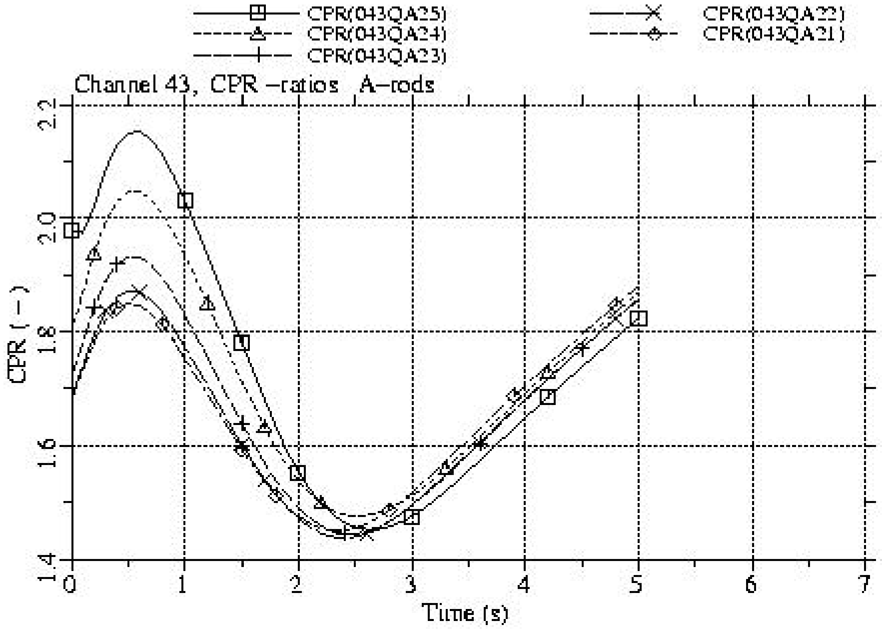
NKS 3D Methodology

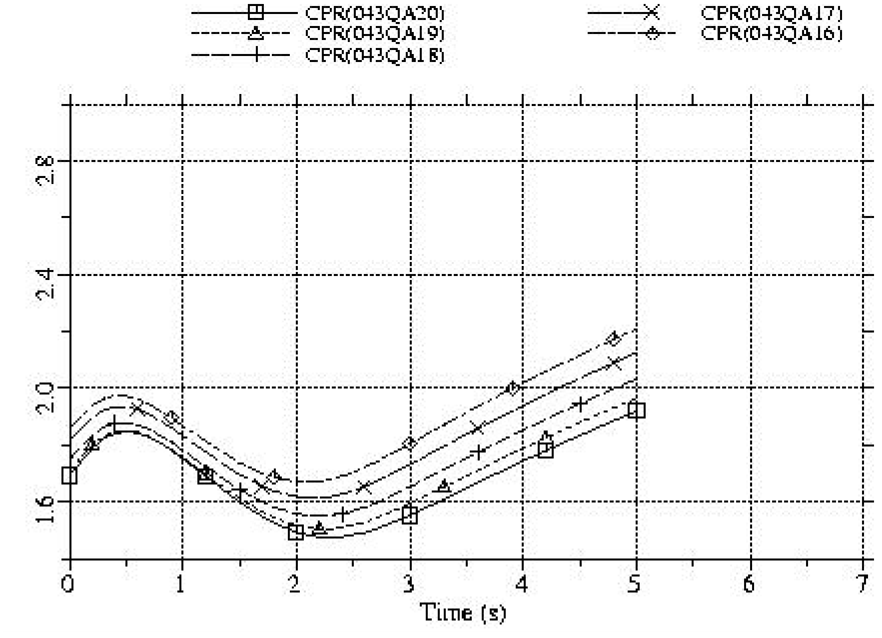
- Power distribution at time 2,4 seconds



NKS 3D Methodology

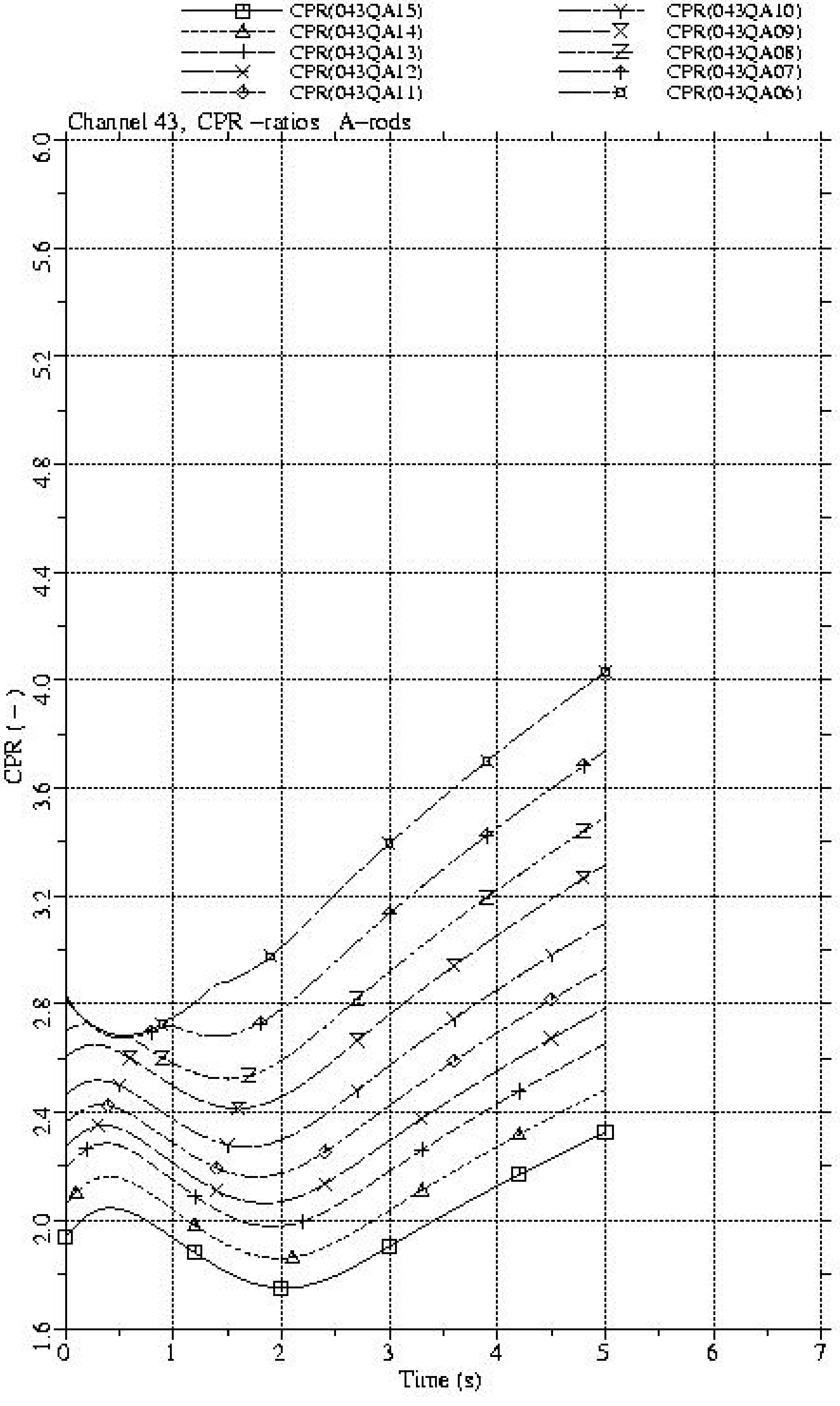
- Bundle 43 CPR-ratios, Axial level 16-25





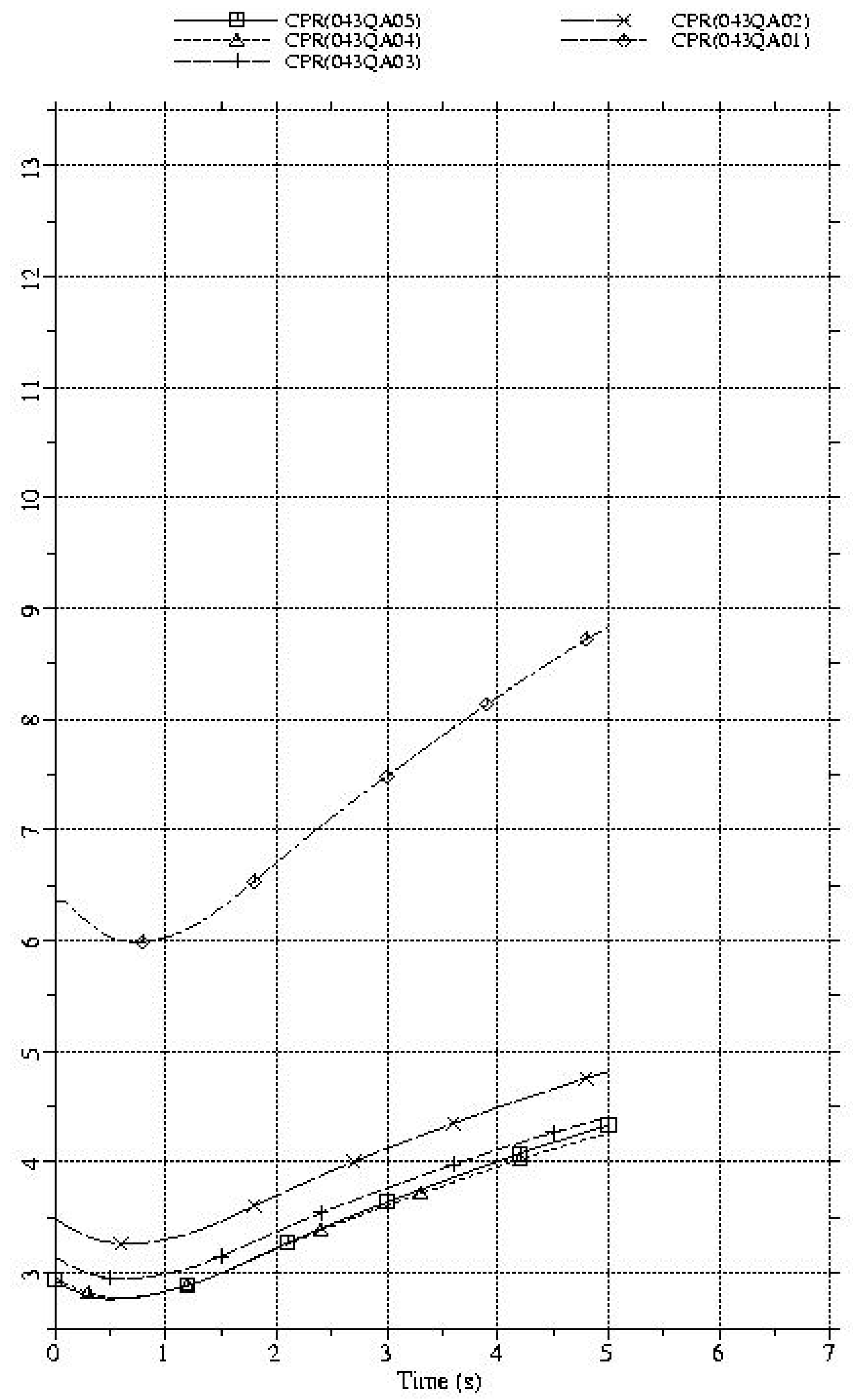
NKS 3D Methodology

- Bundle 43 CPR-ratios, Axial level 6-15



NKS 3D Methodology

- Bundle 43 CPR-ratios, Axial level 1-5



NKS 3D Methodology

- Conclusion
  - Location of min CPR don’t move Probably it is dependent on power distribution !
  - Axial CPR distribution is power shape dependent
  - Minor redistribution of flow
- Discussion/ Continued work
  - How to treat all the data ?
  - New measure, statistical approach ?

TRAB-3D/TRAB pressure transient calculation

A. Daavittila

NKS 3D BWR Transient methodolgy seminar, April 8, 2003

TRAB-3D Overview

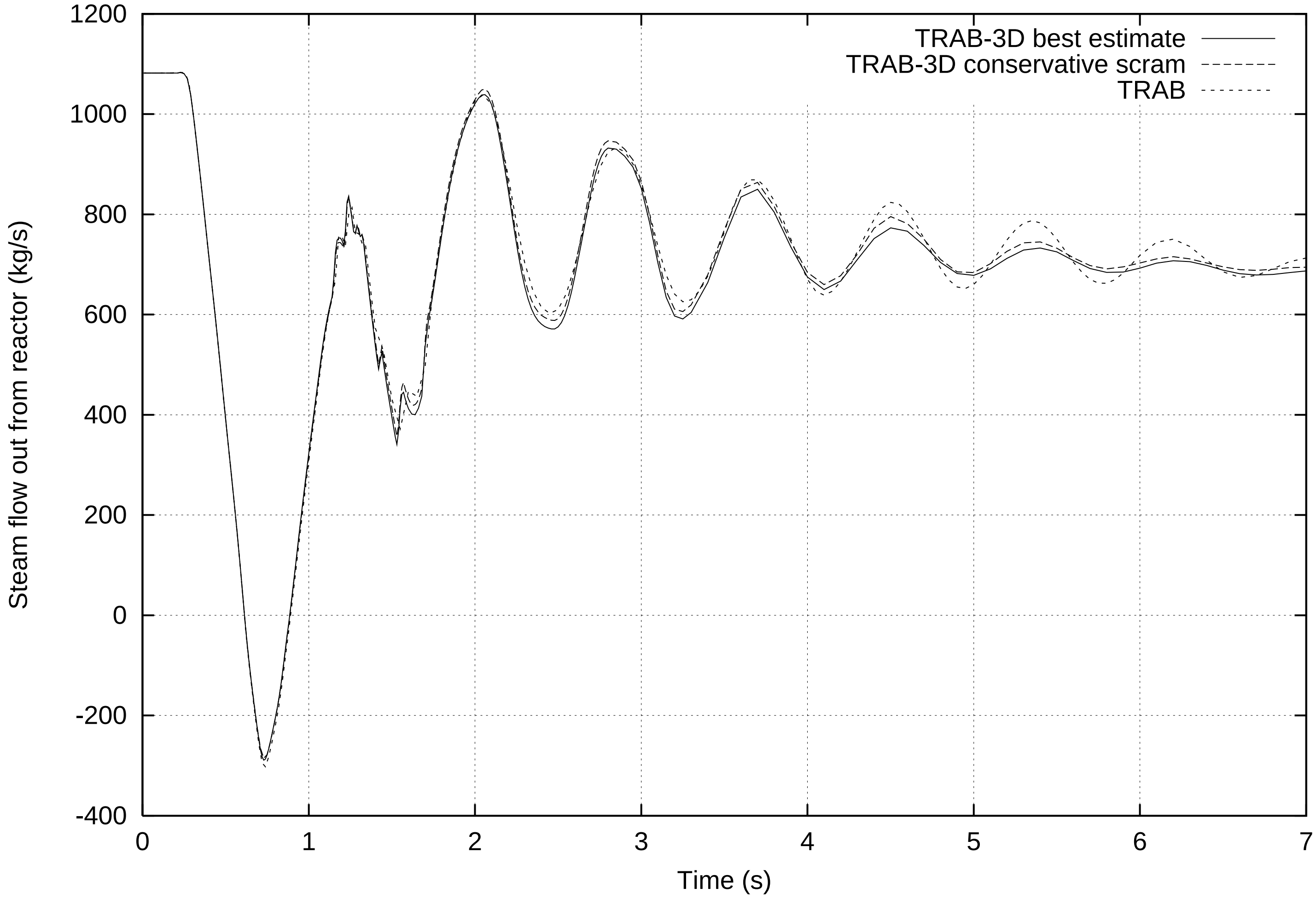
- Based on earlier codes TRAB (1D BWR) and HEXTRAN (3D hexagonal core)
- 3D neutronics BWR dynamics code with rectangular core geometry
- 1D parallel channel hydraulics for the core
- Includes: the main circulation system inside the pressure vessel, steam lines, pumps and control systems
- Core and circuit TH iterated together with neutronics during each time step
- Separate core model can be coupled to the fast-running SMABRE TH-code for PWR calculations

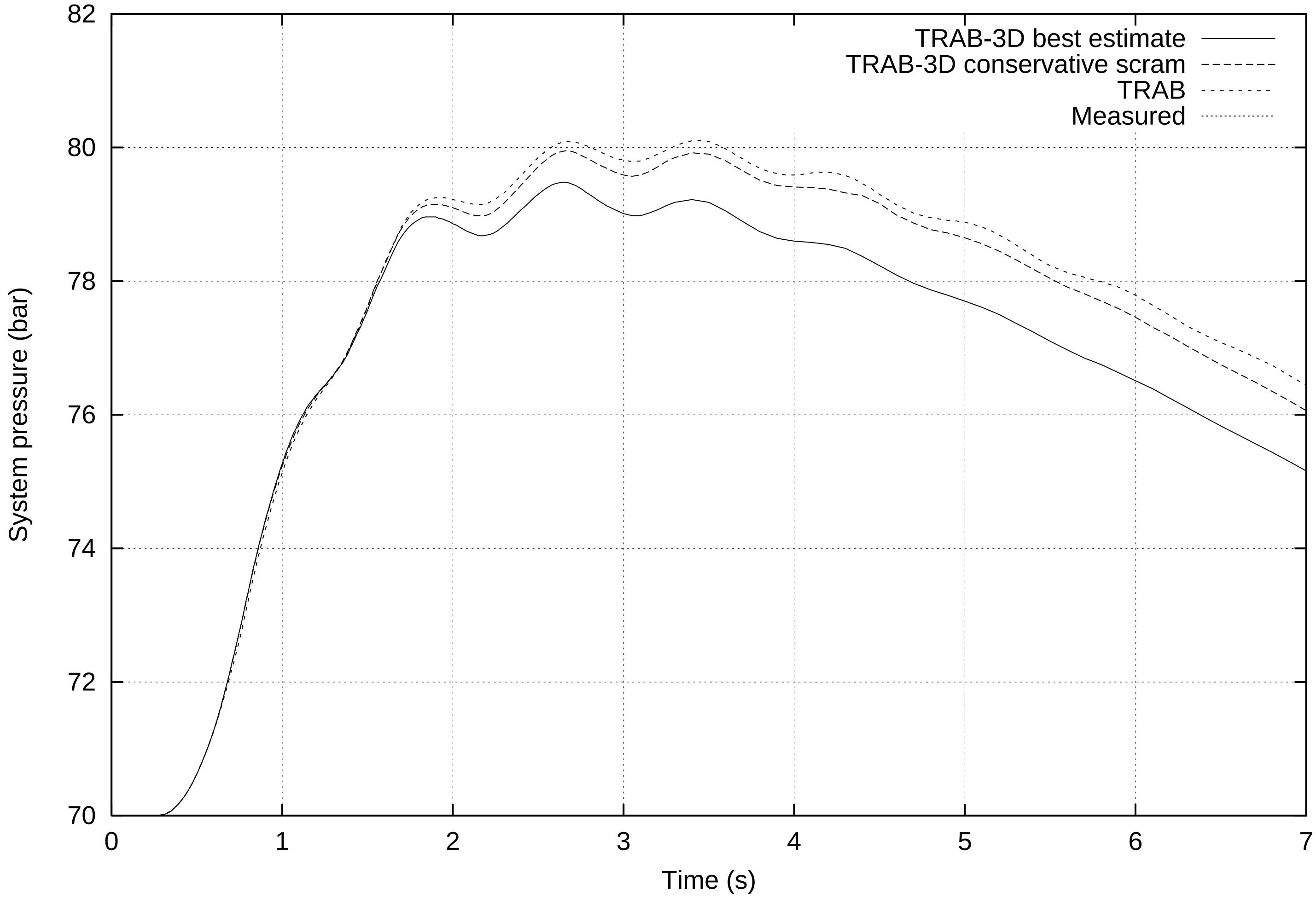
1D vs. 3D calculation

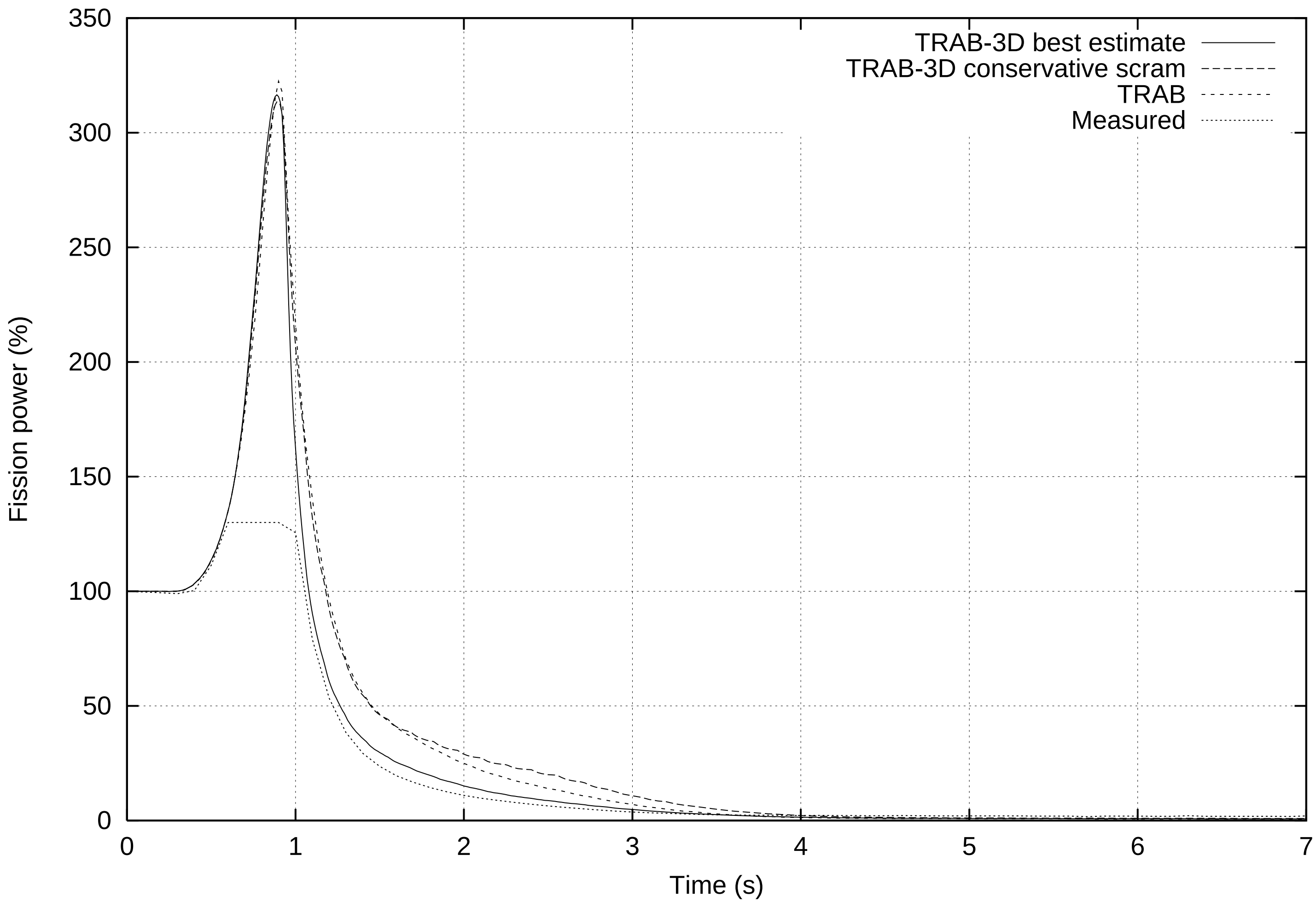
- Same case calculated with TRAB and TRAB-3D
- Identical transient boundary conditions and circuit models used, only difference in core description
- Goal is to see the difference between 1D and 3D using, as far as possible, same assumptions
- The initial state of 1D calculation tuned close to the TRAB-3D initial state (feedback coefficients, axial power distribution), similar approach (with conservatism added) used in 1D licensing calculations
- TRAB-3D calculation repeated with a conservative scram (partially inserted rods not moving) corresponding to the TRAB calculation

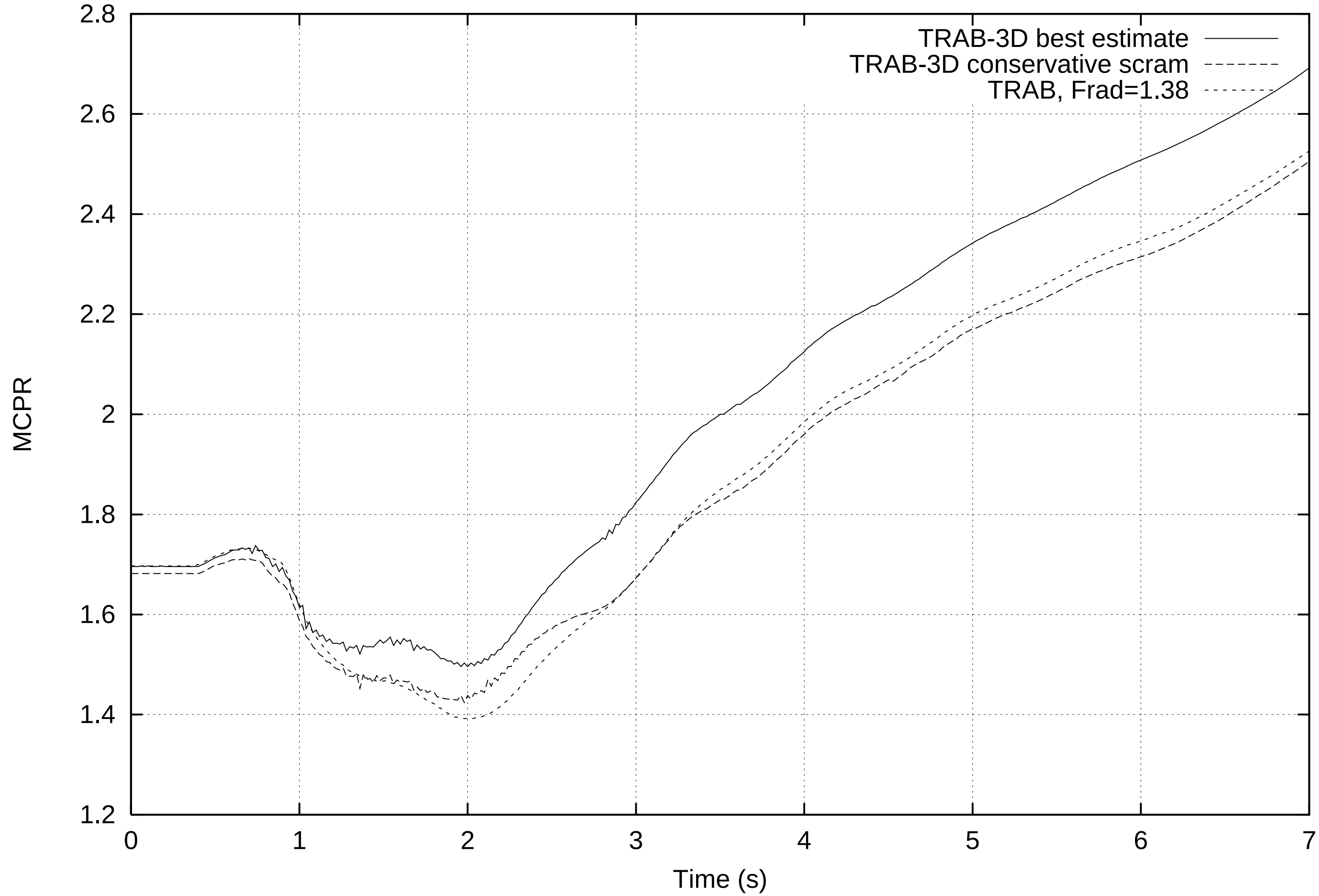
Olkiluoto 1 pressure transient of 1985

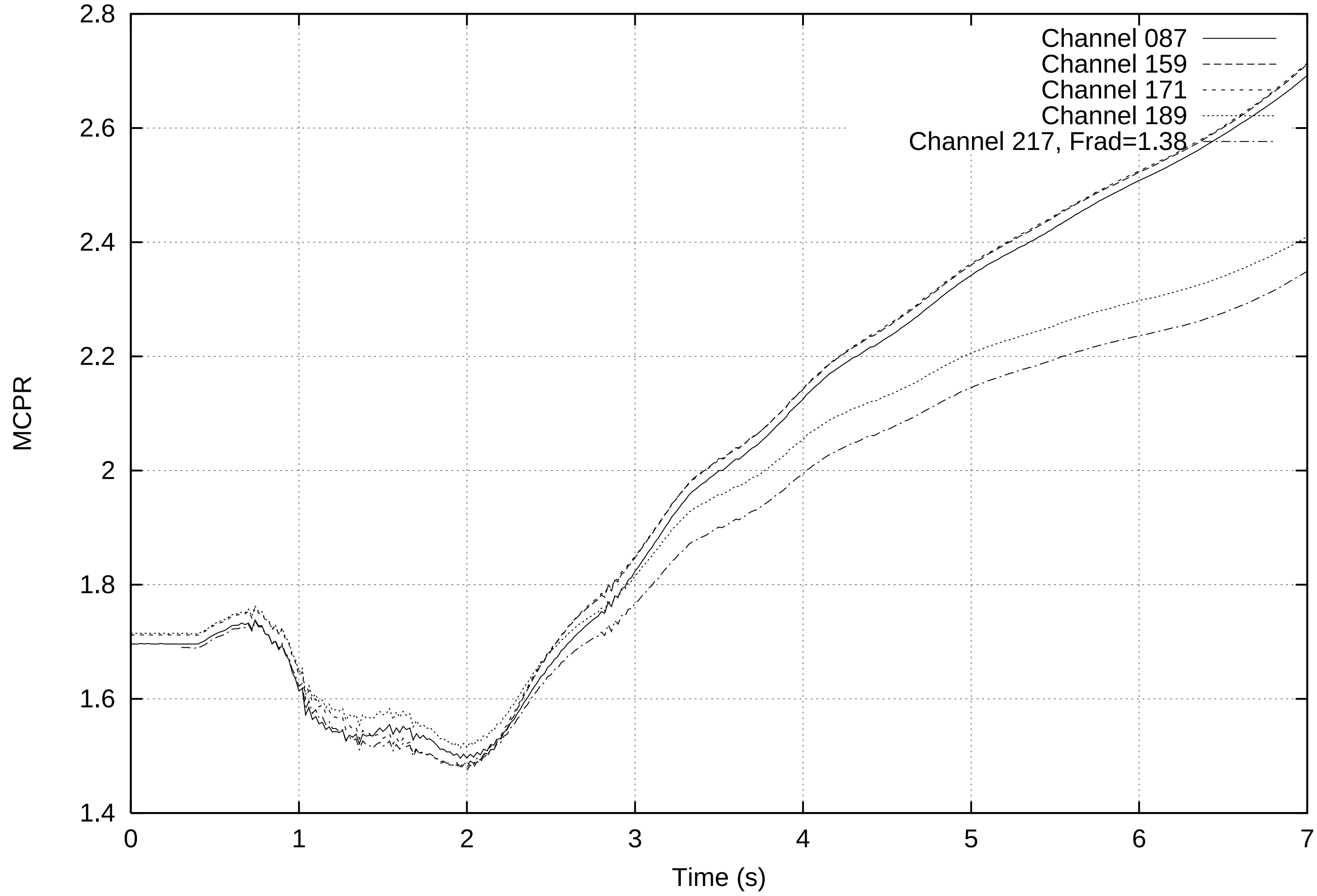
- September 10, 1985
- Erroneous functioning of the pressure controller led to the closing of the turbine valves in 0.5 seconds at full power
- Maximum measured pressure 78.5 bar
- Transient terminated by normal operation of safety systems (reactor scram, relief valves)
- Previously used for the validation of both TRAB and TRAB-3D
- Some measurement data of the real transient available


TRAB-3D best estimate
TRAB-3D conservative scram
TRAB
Steam flow out from reactor (kg/s)
Time (s)
1200
1000
800
600
400
200
0
-200
-400
0
1
2
3
4
5
6
7


TRAB-3D best estimate
TRAB-3D conservative scram
TRAB
Measured
82
80
78
76
74
72
70
0
1
2
3
4
5
6
7
System pressure (bar)
Time (s)


350
300
250
200
150
100
50
0
Fission power (%)
0
1
2
3
4
5
6
7
Time (s)
TRAB-3D best estimate
TRAB-3D conservative scram
TRAB
Measured

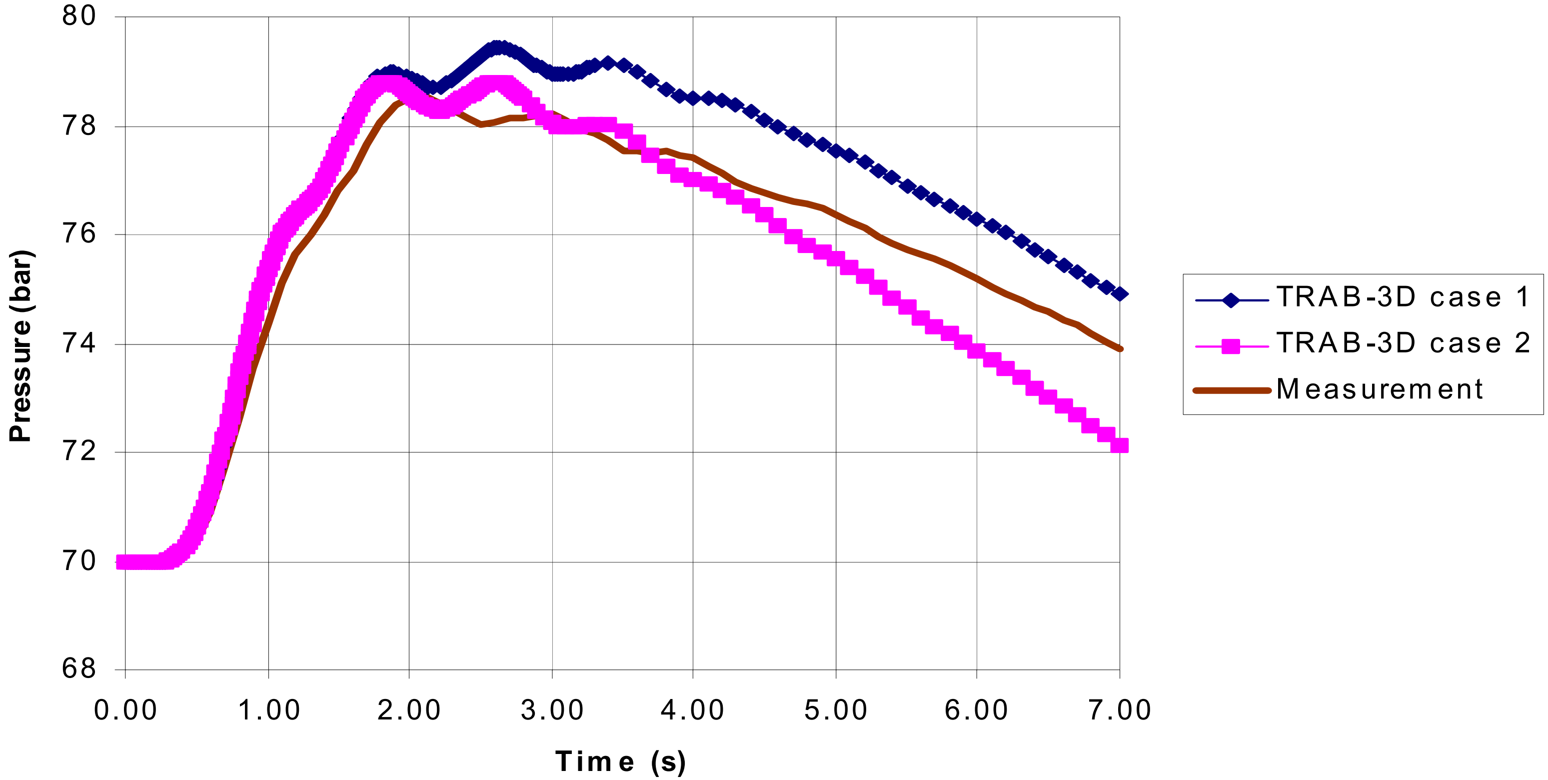

TRAB-3D best estimate
TRAB-3D conservative scram
TRAB, Frad=1.38
MCPR
2.8
2.6
2.4
2.2
2
1.8
1.6
1.4
1.2
0
1
2
3
4
5
6
7
Time (s)


2.8
2.6
2.4
2.2
2
1.8
1.6
1.4
MCPR
0
1
2
3
4
5
6
7
Time (s)
Channel 087
Channel 159
Channel 171
Channel 189
Channel 217, Frad=1.38

Conclusions

- Difference between 1D and 3D transient MCPR quite small
- Difference between straight-forward 1D and 3D calculation mostly explained by conservatism in 1D scram modeling
- Larger effects could be anticipated in longer transients (ATWS) with more dramatic flow redistribution
- The example case is not a safety analysis case, but a real plant transient!
- Results extremely sensitive to uncertain parameters, such as gas gap conductance

Sensitivity to gas gap conductance



Conclusions (cont'd)

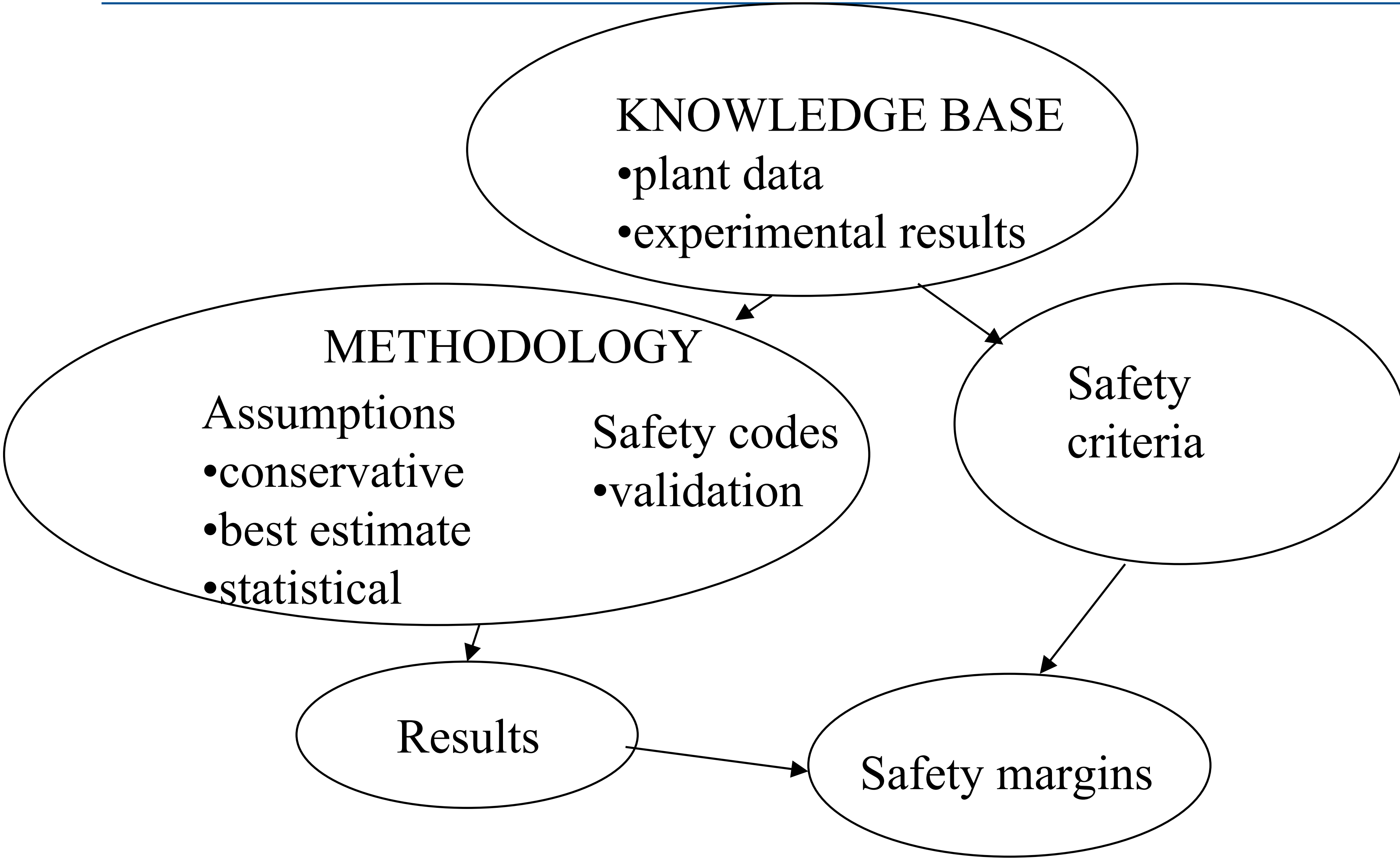
- Core geometry, burnup distribution, heat transfer parameters etc. much more realistic in 3D, especially in a mixed loading
- Improving core models (e.g. gas gap dependence on burnup) easier
- In this case multiple hot channels used (this approach has been used at VTT for VVERs also), in future probably automatic CPR calculation of every channel
- Traditional conservatism in 3D leads to wrong flow distribution, and can be difficult to assess, conservatism easy to apply in hot channel calculation

3D-METHODOLOGY AND LICENSING OF CORE DESIGNS

Keijo Valtonen

8.4.2003

NKS 3D BWR


KNOWLEDGE BASE
•plant data
•experimental results
METHODOLOGY
Assumptions
•conservative
•best estimate
•statistical
Safety codes
•validation
Safety criteria
Results
Safety margins

CURRENT LICENSING PRACTICES

- LICENSING OF CORE DESIGN
  - BASED ON 1D METHODOLOGY -AVERAGE CORE BEHAVIOUR AND SEPARATE HOT CHANNEL ANALYSIS
  - 1D CODES ”INHERENTLY” CONSERVATIVE
  - VALIDATION BASED ON
    - SEPARATE PHENOMENA TESTS
    - SOME PLANT TRANSIENTS

CURRENT LICENSING PRACTICES

- 3D CODES
  - TRADITIONALLY BEEN USED FOR SPECIAL NEEDS
    - CONTROL ROD DROP
    - OSCILLATION TRANSIENTS
  - VALIDATION BASED MAINLY ON PLANT TRANSIENTS, BENCHMARKING AND SEPARATE EFFECT TESTS
  - MODELLING NOT ADEQUATE IN SOME AREAS (ESPECIALLY FUEL) - TYPICALLY ONLY ONE FUEL TYPE

CURRENT LICENSING PRACTICES

- ADEQUATE SAFETY MARGINS ARE ASSURED BY CONSERVATIVE MODELS AND INPUT VALUES.

DEVELOPMENT OF 3D-METHODOLOGY

- GROUNDS
  - HETEROGENEOUS CORE DESIGNS
    - MIXED CORE CONFIGURATIONS
    - HIGHER FUEL BURNUP
- MODELLING ENHANCEMENTS
  - 3D KINETICS
  - FUEL
    - HIGH BURNUP PHENOMENA

DEVELOPMENT OF 3D-METHODOLOGY

- THERMAL HYDRAULICS
  - NEW FUEL DESIGNS
    - COMPLEX FUEL LATTICE DESIGN
    - SPACER DESIGNS
    - CORRELATIONS
  - LOWER PLENUM
- VALIDATION
  - LIMITED VALIDATION DATA AVAILABLE
    - PLANT DATA (TRANSIENSTS) - QUALITY OF DATA IS PROBLEM
    - BENCHMARKING
    - SEPARATE EFFECT TESTS

CONCLUSIONS

- REQUIREMENTS FOR 3D METHODOLOGY
  - ADEQUATE MODELLING AND VALIDATION
  - SOME CONSERVATISM IN ORDER TO TAKE INTO ACCOUNT MODELLING AND VALIDATION SHORTCOMINGS
  - BEST ESTIMATE MEHODOLOGY WITH SENSITIVITY ANALYSIS

SKI Point of View
on
3D BWR Transient Methodology

NKS Seminar
Otaniemi, Finland
April 8, 2003

Ninos Garis

SKI

Analysis of events which have occurred

- Analysis of events which have occurred is an important source of information for evaluation of the behaviour of various safety systems during transients.
- This is also important for validation of advanced codes
- However, this requires that relevant transient information (input & measurement data) is saved in a proper way
- This will in turn improve the understanding of events which have occurred (regulatory goal).

SKI´s regulations on events which have occurred

In SKI´s regulation SKIFS 1998:1, there are regulations on

- investigation of events which have occurred (Chapter 5. 6§)
- reporting of events which have occurred (Chapter 7. 1§)
- documentation of process and parameter data (Chapter 8. 2§)

SKi

Regulation on investigation of events

- The events shall be investigated in a systematic manner in order to determine sequences and causes as well as in order to establish the measures required to restore the safety margins and to prevent recurrence.

SKI´s general recommendations

All events and conditions should be systematically investigated so that

- the entire event sequence is clarified including the circumstances which could have prevented and stopped the sequence,
- the consequences are determined,
- the root causes are established with a high degree of probability,
- well-founded measures are specified to prevent similar events or conditions from recurring.

Regulation on reporting events

- The events shall be reported without delay to the Swedish Nuclear Power Inspectorate i a certain manner.

Regulation on document retention

- The documentation of activities which are important for safety (process and parameter data) shall be retained for the necessary length of time in order to be able to investigate events which have occurred at the facility and to analyse the causes of these events

Signaler som bör sparas vid en inträffad händelse

- **Neutronflöde (APRM och LPRM)**
- **Vattennivå (fin och grov)**
- **Matarvattenflöde**
- **Matarvattentemperatur**
- **Inloppsunderkylning (temp. i nedre plenum)**
- **Ångdomstryck (fin och grov)**
- **Styrstavsläge**
- **Ångflöde**
- **HC-flöde**
- **Turbinventil- och dumpventillägen**
- **Interna regulatorsignaler för tryckregulatorn, mavaregulatorn och HC-flödesregulatorn**
- **Tider för säkerhetskedjor som löses ut**
- **Pumpvarvtalen för RC**

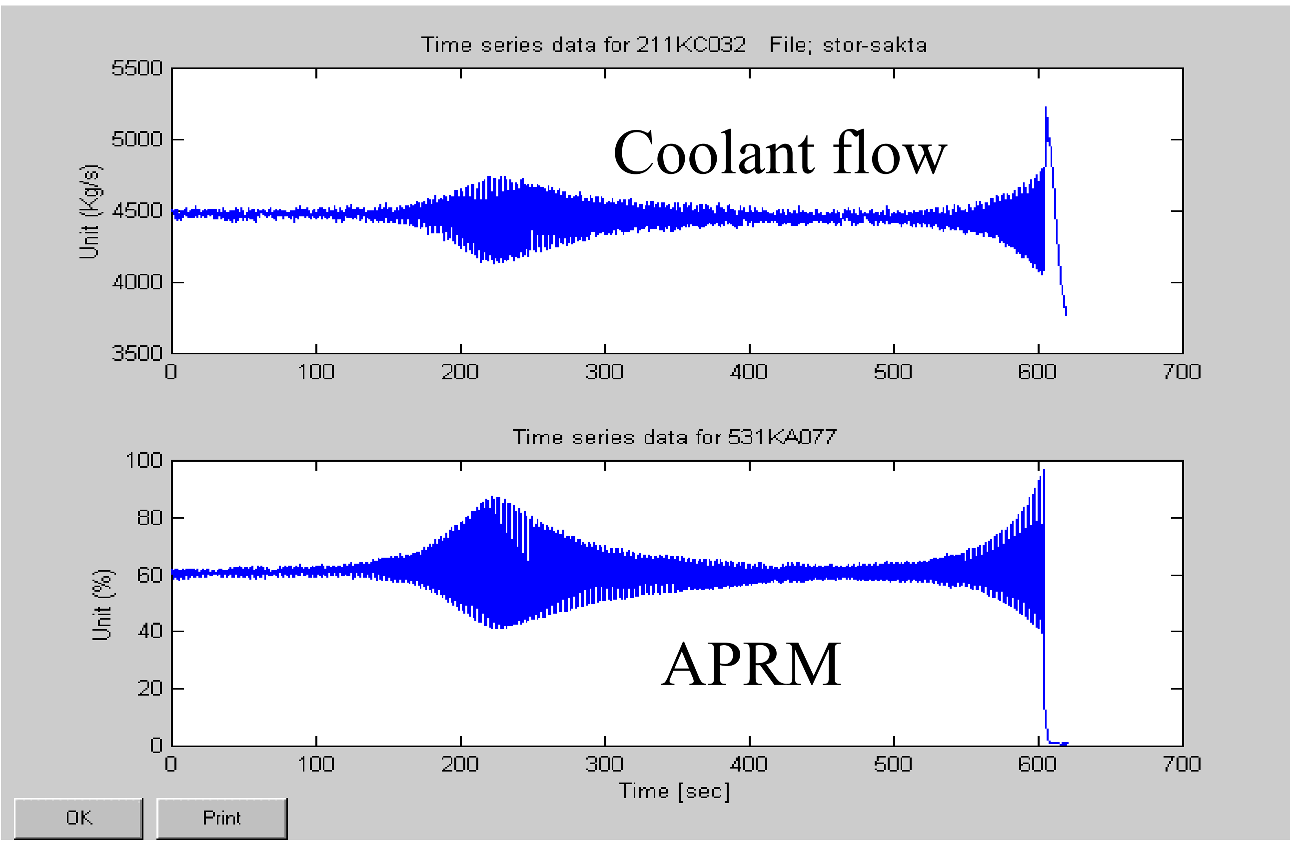
Samplingsfrekvensen > 25 Hz och upplösningen hos A/D-omvandlaren >12 bitars. Registreringen bör vara 15 minuter lång inklusive 10 minuters förhistoria.

SKi

Some events which have occurred in Swedish Reactors

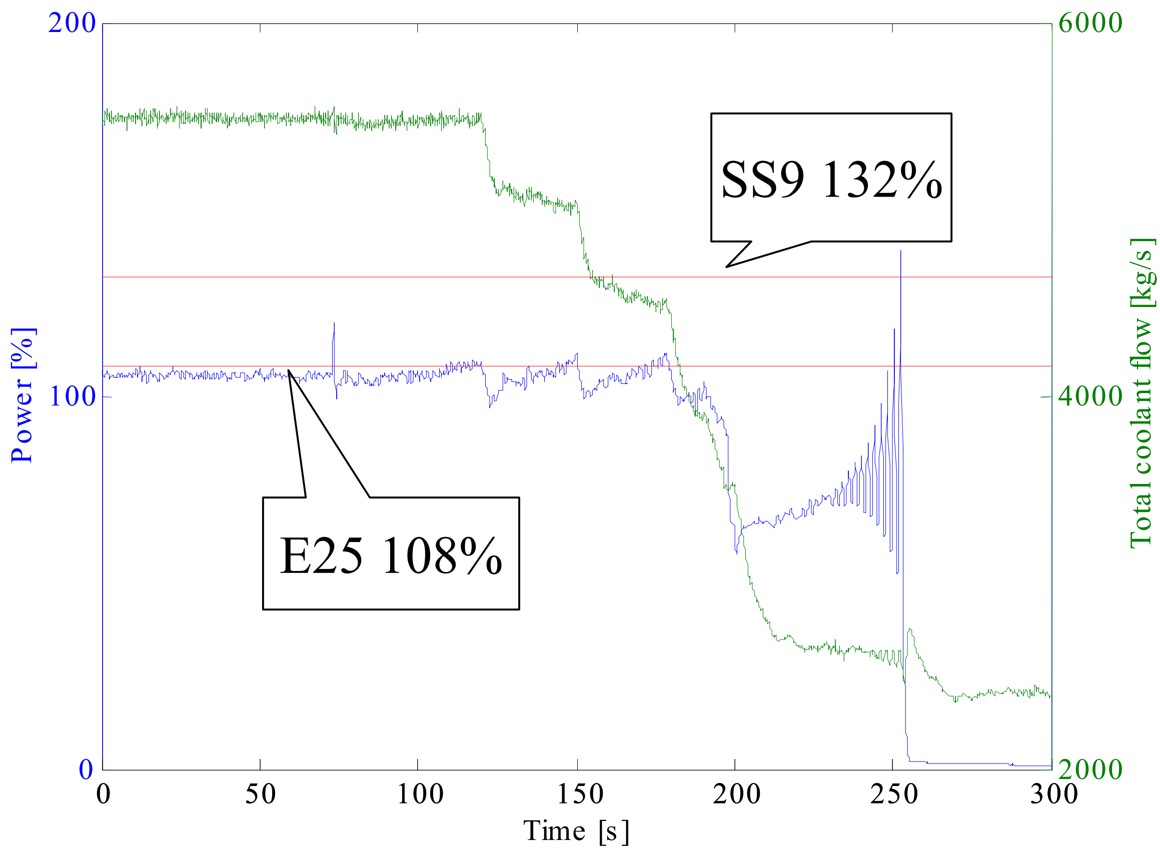
- BWR instabilities (Oskarshamn 3, 1998)
- Feed water transients (Oskarshamn 2, 1999)
- Local instability induced by unseated fuel asssemblies (Forsmark, 1996)
- Box-bowing and stationary dryout in BWR (Oskarshamn 2, 1988)
- Fuel failures in Ringhals 1, possibly due to power history/control rod movement or RIA
- Spurious motion of single control rods (Forsmark)
- Pressure increase transients (Barsebäck)
- S-shaped fuel in Ringhals (PWR)

The instability event at Oskarshamn 3, February 8, 1998



Coolant flow and APRM as a function of time. The reactor power oscillates with high amplitude during two periods, between 200 and 300 s, and just before scram.

The instability event at Oskarshamn 2, February 25, 1999



Reactor power and coolant flow as a function of time during the transient

Conclusions

SKI is of the opinion that

- validated BE methods give a more accurate perception of actual safety
- BE methods can be used to reduce unnecessary conservatism
- BE methods can be used to improve the understanding of events that have occurred

Advanced Method for BWR Transient Analysis

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P.O. BOX 3220
91050 Erlangen Germany

FRAMATOME ANP

Advanced Method for BWR Transient Analysis

- Overview of Framatome ANP's BWR Methodology
- Application of the Advanced Transient Analysis
- OECD/NRC Boiling Water Reactor Turbine Trip Benchmark

Challenges for Design and Operation of Modern BWR Fuel Assemblies and Cores

> ***Customer***

Optimal fuel utilization for safe, reliable and highly flexible reactor operation

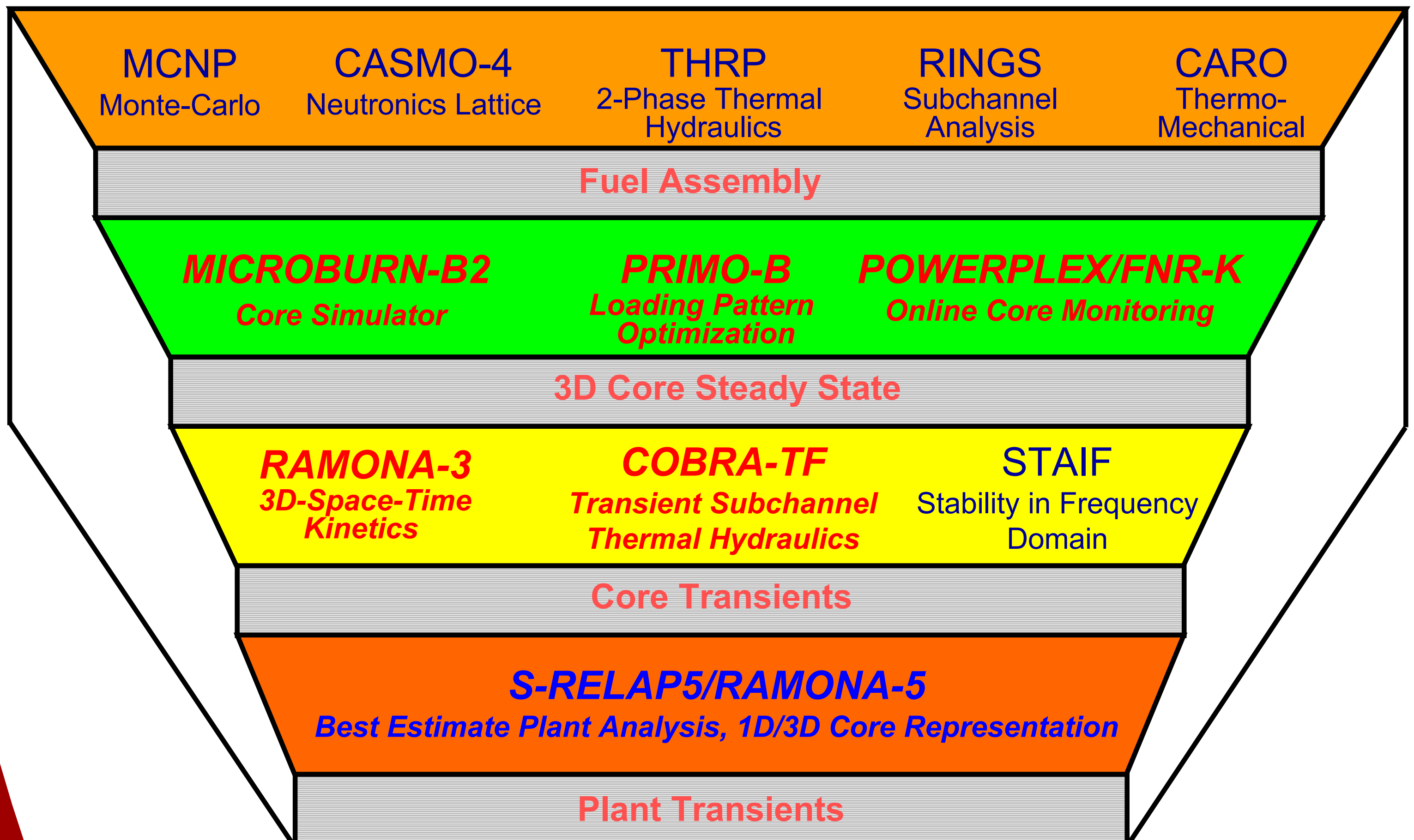
> ***Framatome ANP***

- Advanced fuel design
- Modern core loading concepts
- High operational flexibility

> ***Design Tools***

- Comprehensive physical modelling
- Qualified single codes and code systems

Overview of Framatome ANP’s BWR Methodology



Scheme for Advanced Transient Analysis

	Prior Practice	Current Practice	
			Advanced Method
Type of Analysis	All Transients	All Transients	Limiting Transient
Plant and Core Behaviour	VERENA/ COSBWR(1D)	S-RELAP5/ RAMONA5(1D)	RAMONA3(3D)
Hot Channel	FRANCESCA Multi-Channel		FRANCESCA Single-Channel
Validation	Well validated design methodology		Good agreement with S-RELAP5

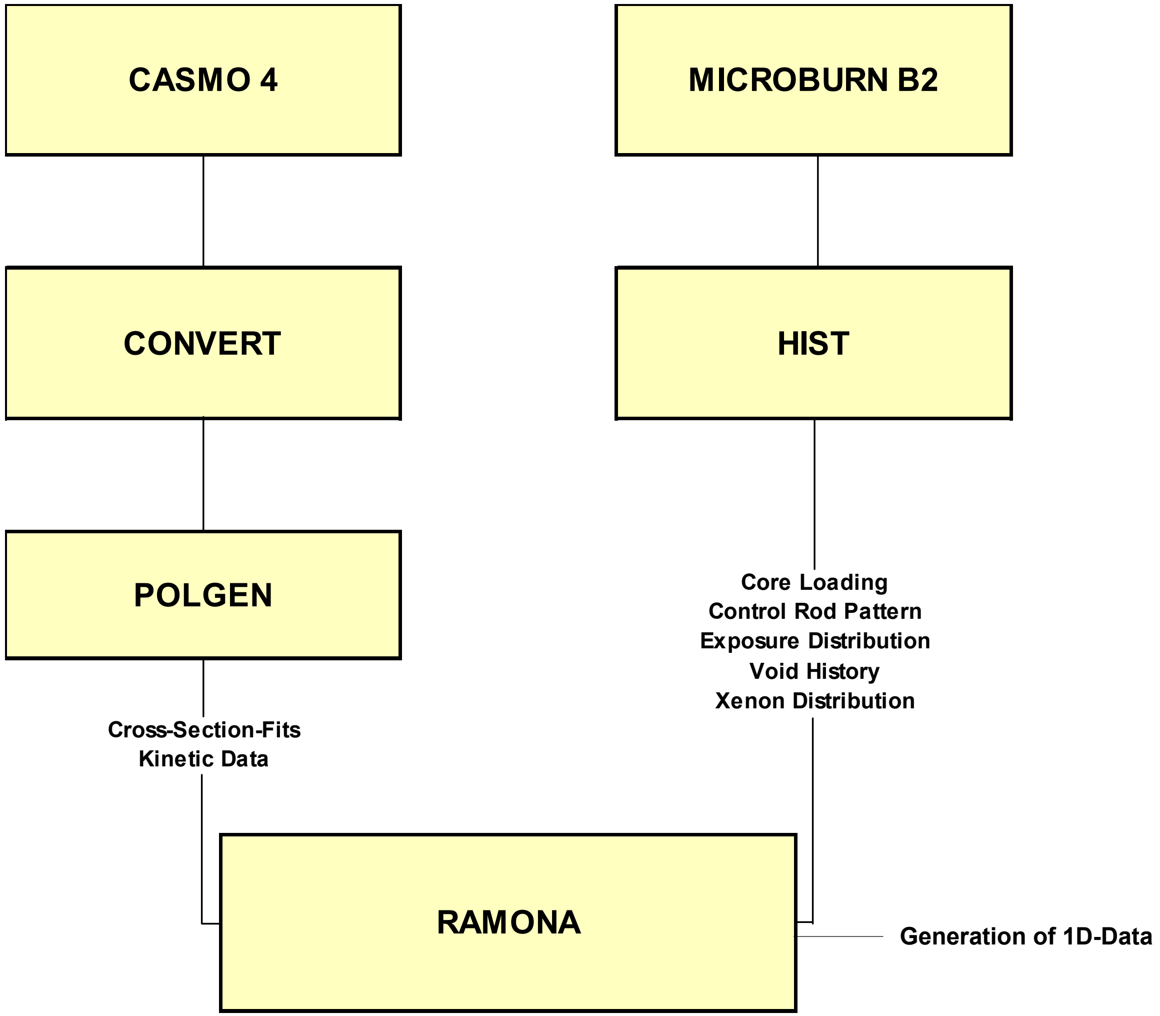
3D Transient Code RAMONA 3

- **Neutronics:** **$1^1/_2$-Group Diffusion Model**
- **Thermal Hydraulics:** **4-Equation Drift Flux Model**
- **BWR System Components:** **Pumps, Separators, Steam Line, Feedwater and Pressure Controller, Reactor Protection System**

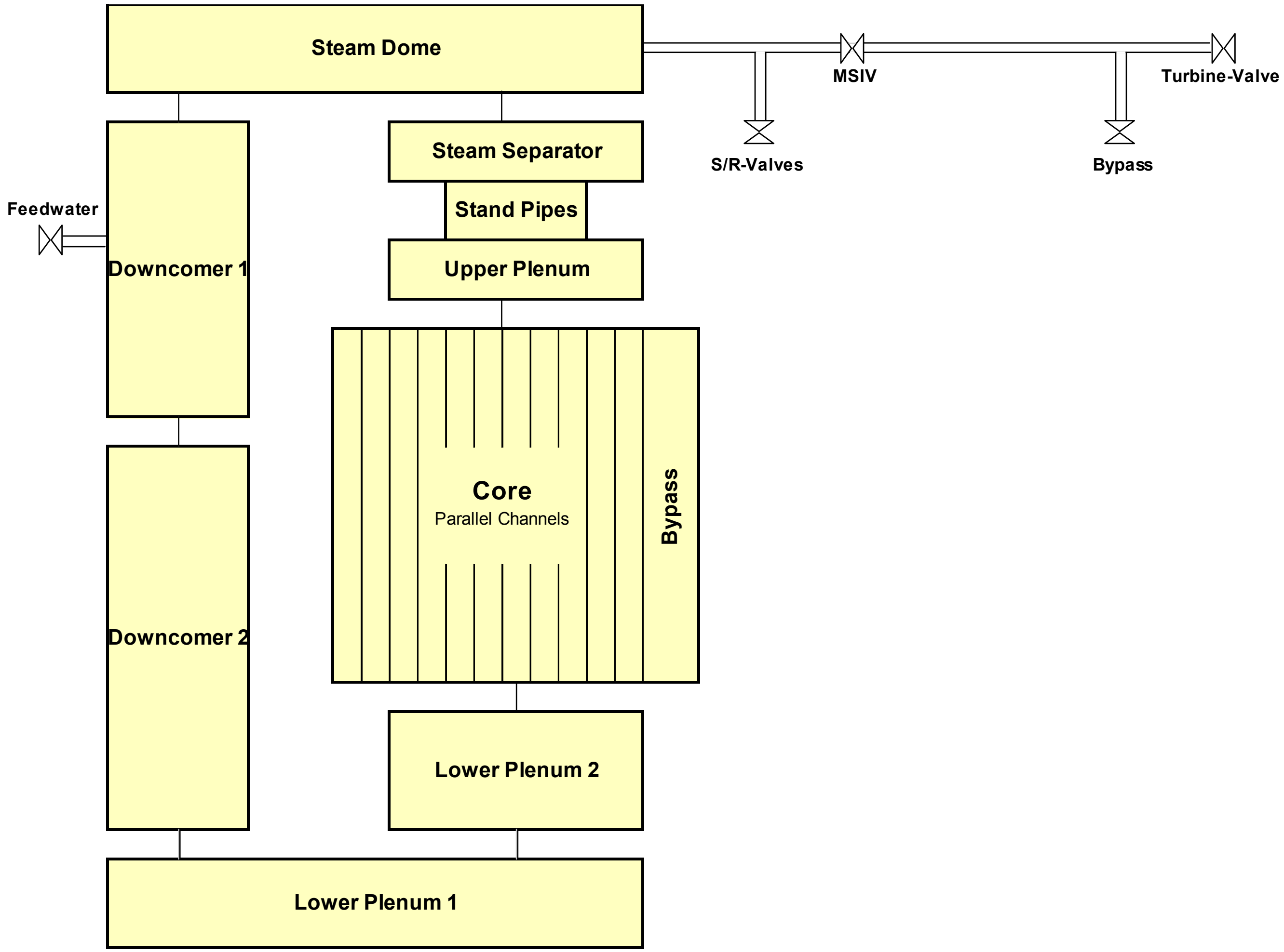
Validation:

- **Peach Bottom Turbine Trip**
- **Spert Reactivity Insertion Experiments**
- **Ringhals Stability Benchmark**
- **Validation against Plant Stability Measurements and Operational Transients**
- **GUN C Cycle 12 and 13 (global and regional instabilities)**

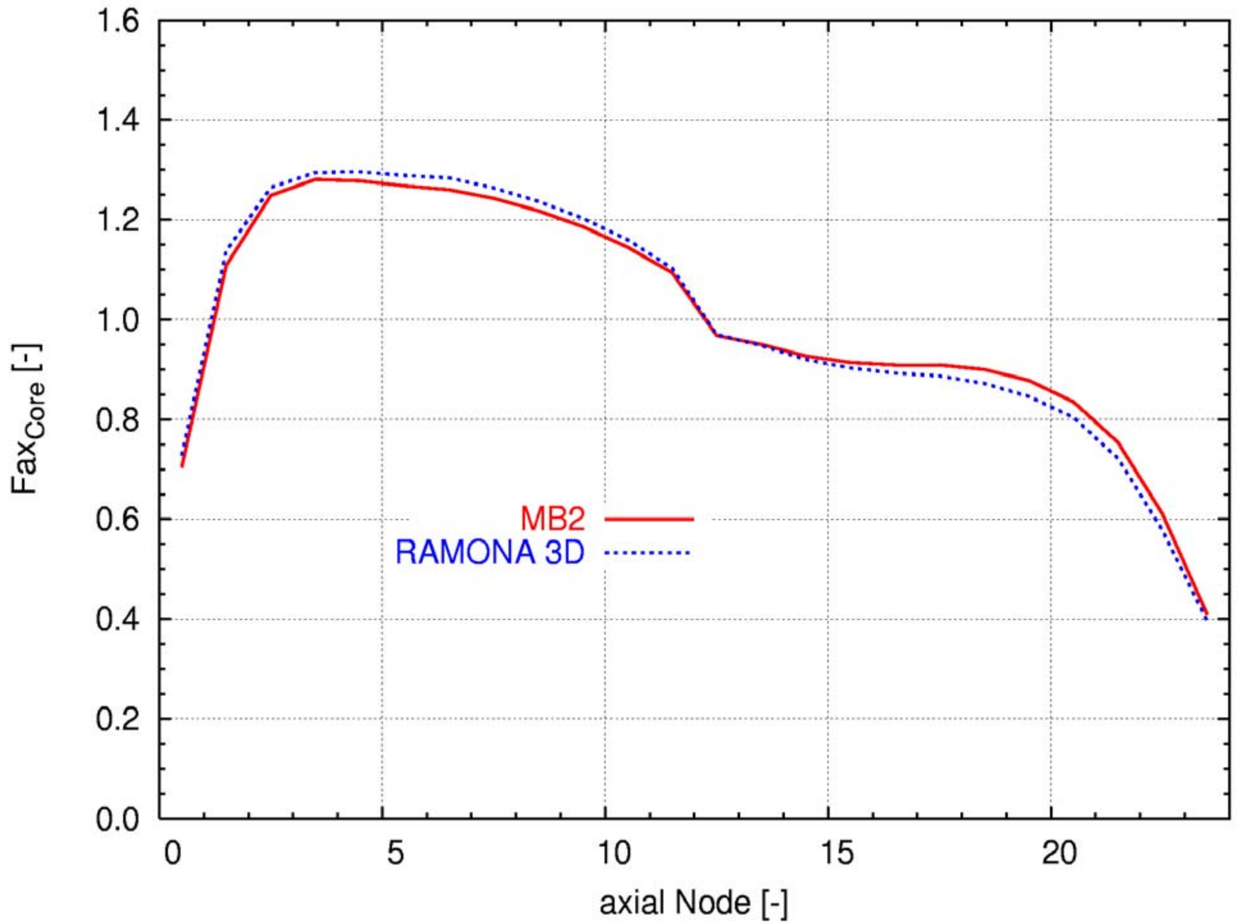
RAMONA3 – Neutronic Data Generation



RAMONA3 – Hydraulic Model Components



BOC - Core Averaged Axial Power



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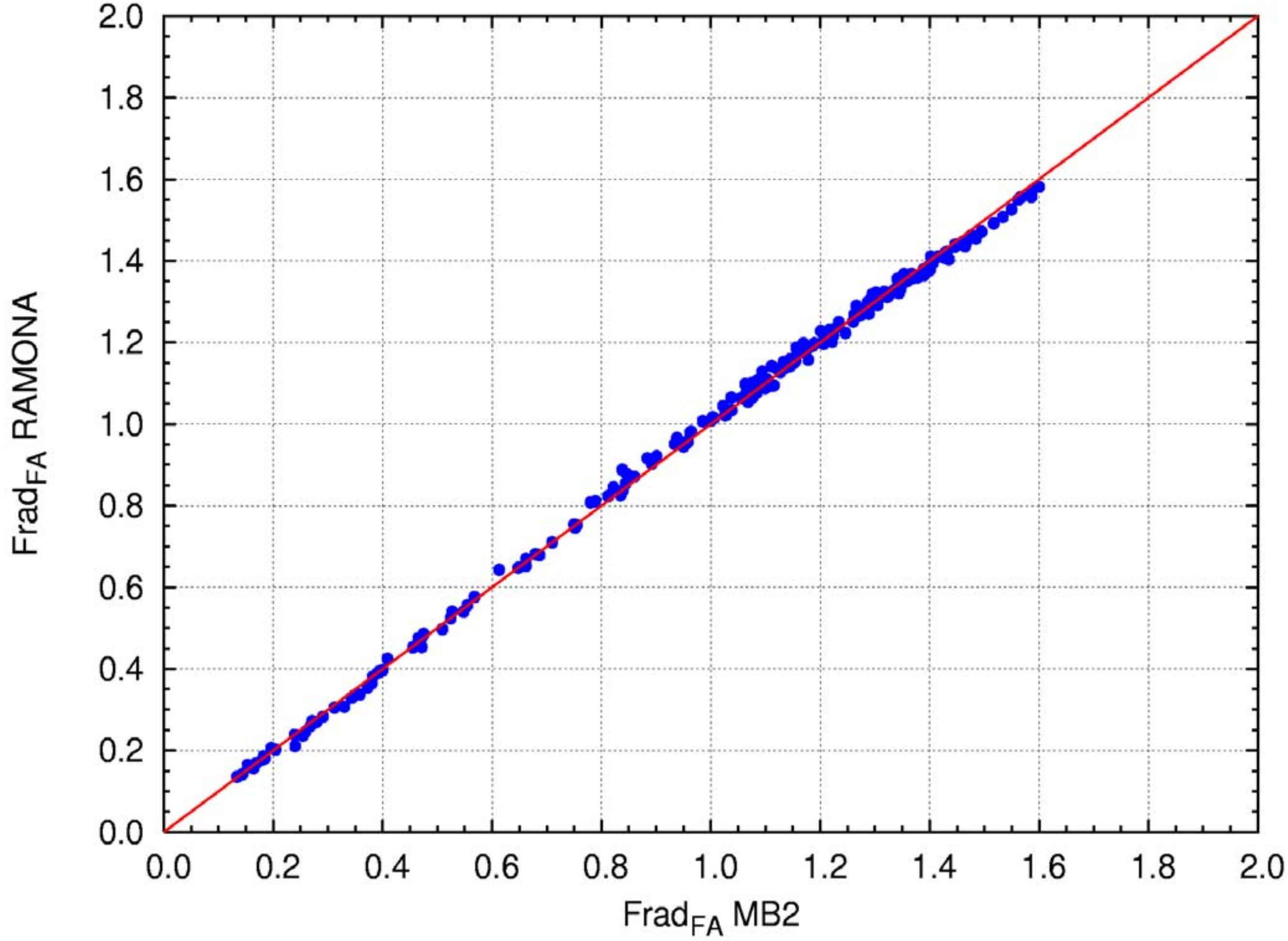
BOC – Radial Power Factors



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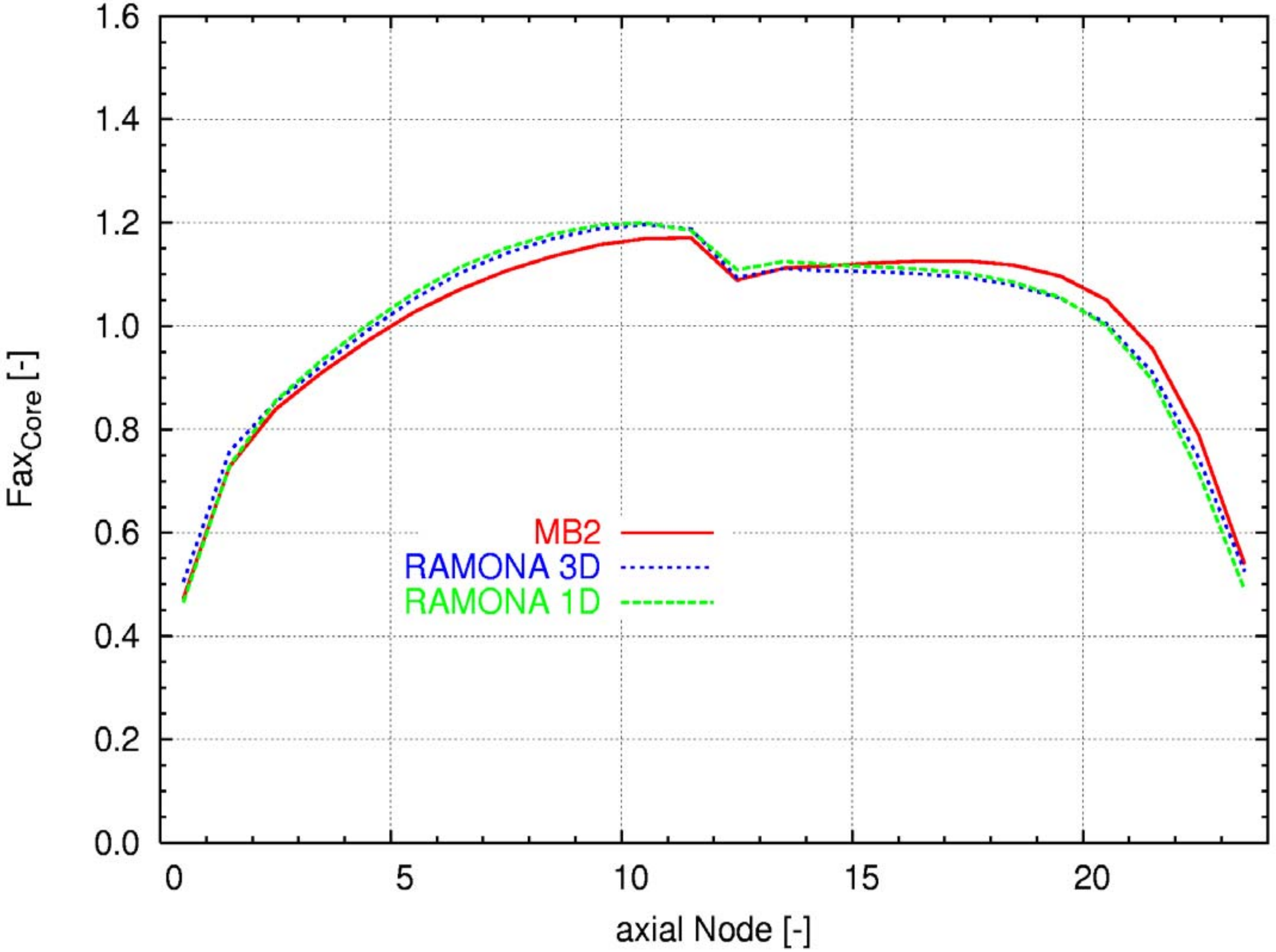
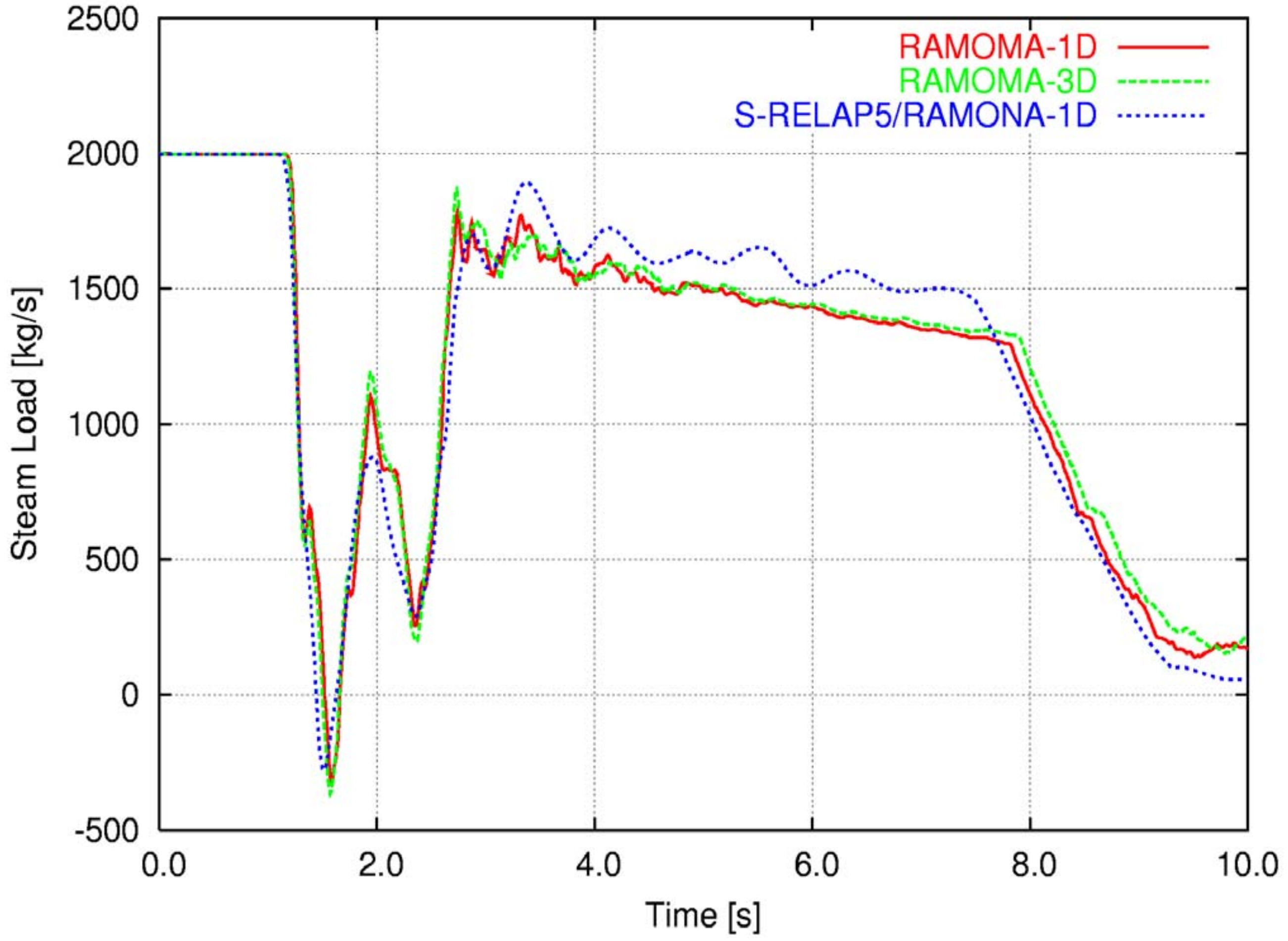
EOC - Core Averaged Axial Power



Comparison of the Steam Loads



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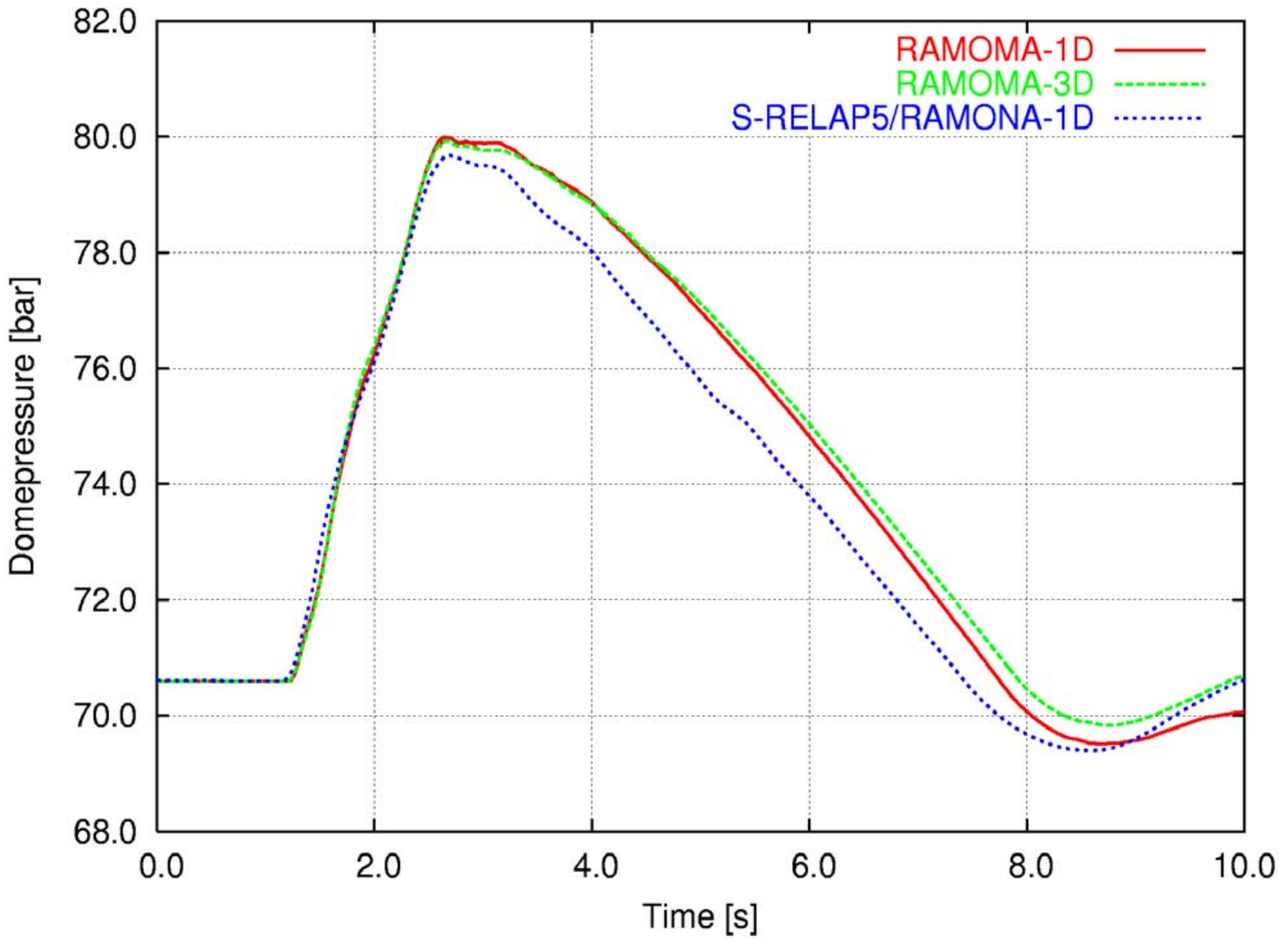
Comparison of the Dome Pressures



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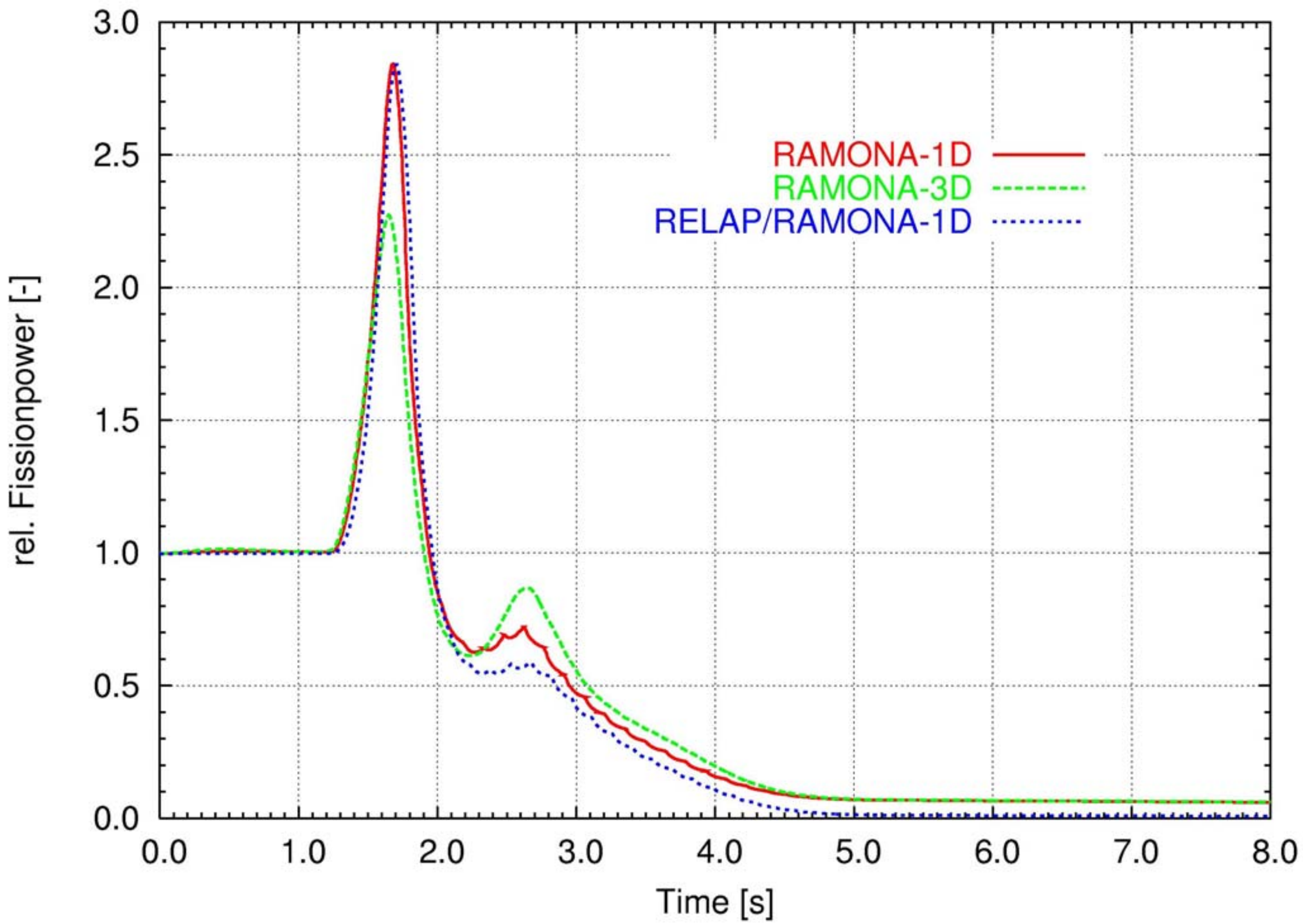
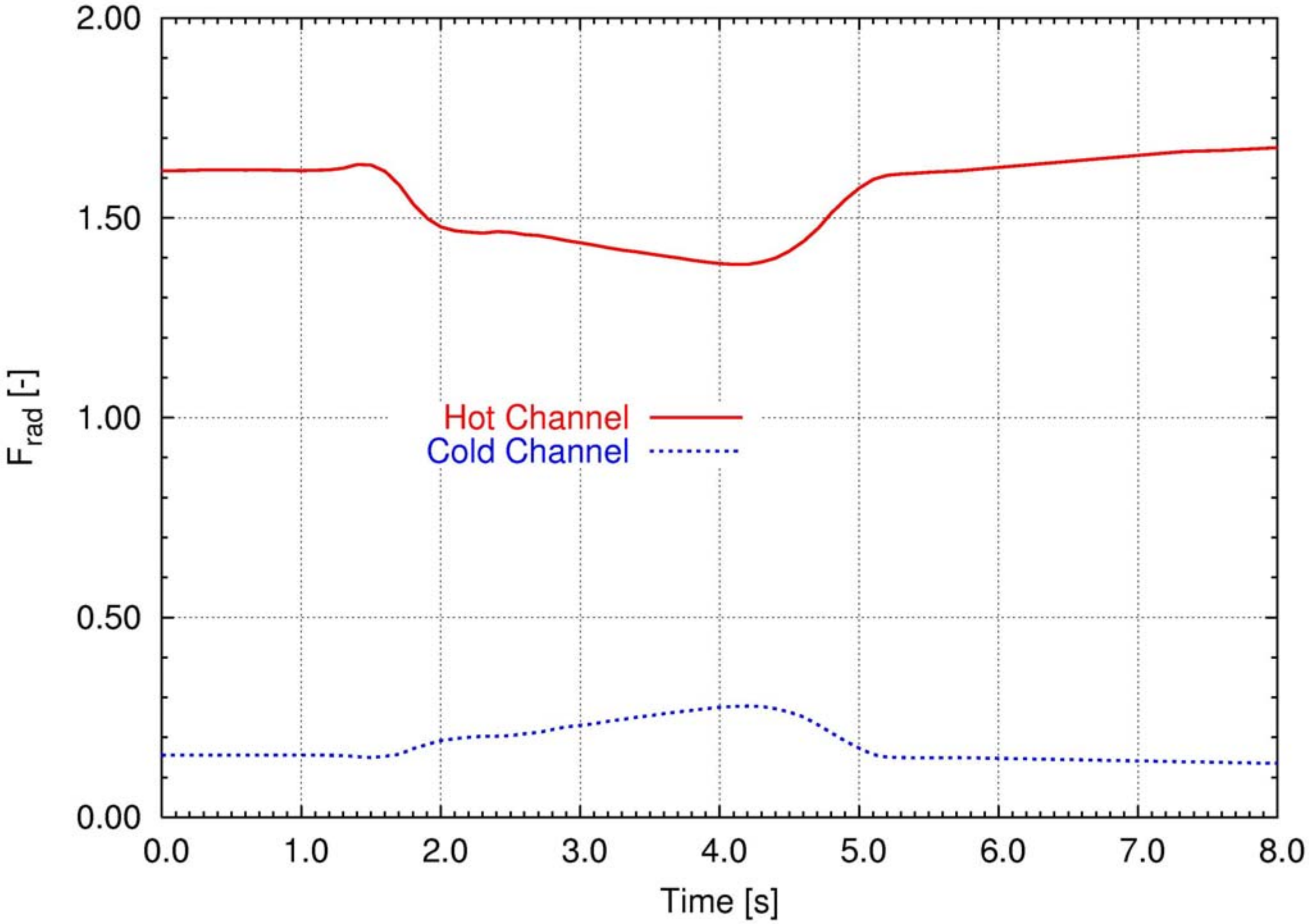
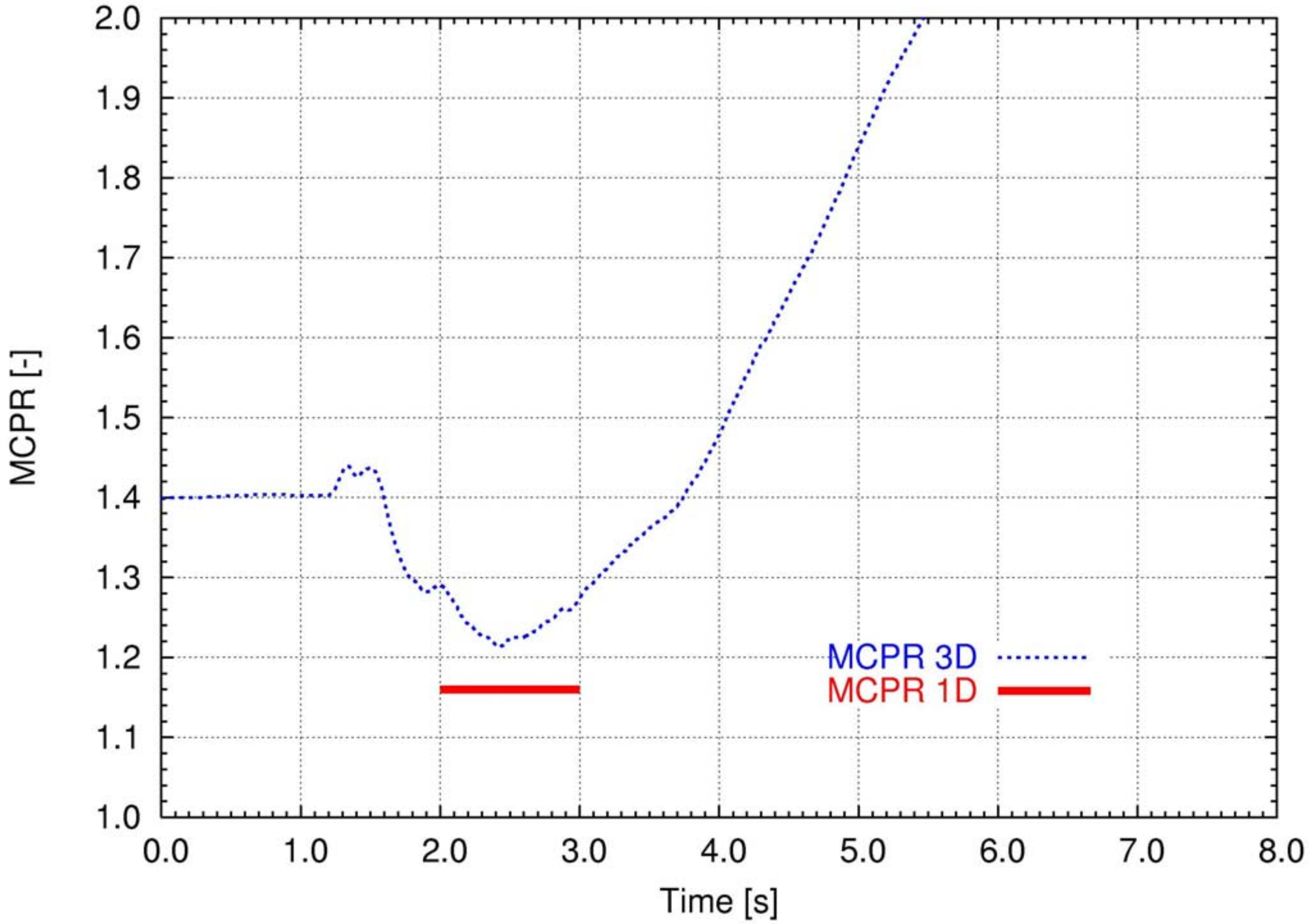
Comparison of the Fission Powers



Limiting Case – Radial Factor of Hot & Cold Channel



Limiting Case – MCPR of the Hot Channel



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OECD/NRC BWR Turbine Trip Benchmark

> *Recent progress in computer technology allows the calculation of plant transients with system transient codes coupled with 3D core models.*

> *A turbine trip transient in a BWR is a pressurisation event in which the coupling between core phenomena and system dynamics plays an important role.*

> *NEA, OECD and US NRC have approved a BWR TT benchmark based on the Peach Bottom Tests for the purpose of validating advanced system best-estimate analysis codes.*

BWR Turbine Trip Benchmark

> Exercise 1

- Power vs. Time plant system simulation with fixed axial power profile table is given => thermal-hydraulic system response

> Exercise 2

- Coupled 3D and/or 1D kinetics/core thermal-hydraulic BC model
  - Hot zero power.
  - Hot full power and transient using the provided core BC.

> Exercise 3

- Best-estimate coupled 3D core/thermal-hydraulic system modeling

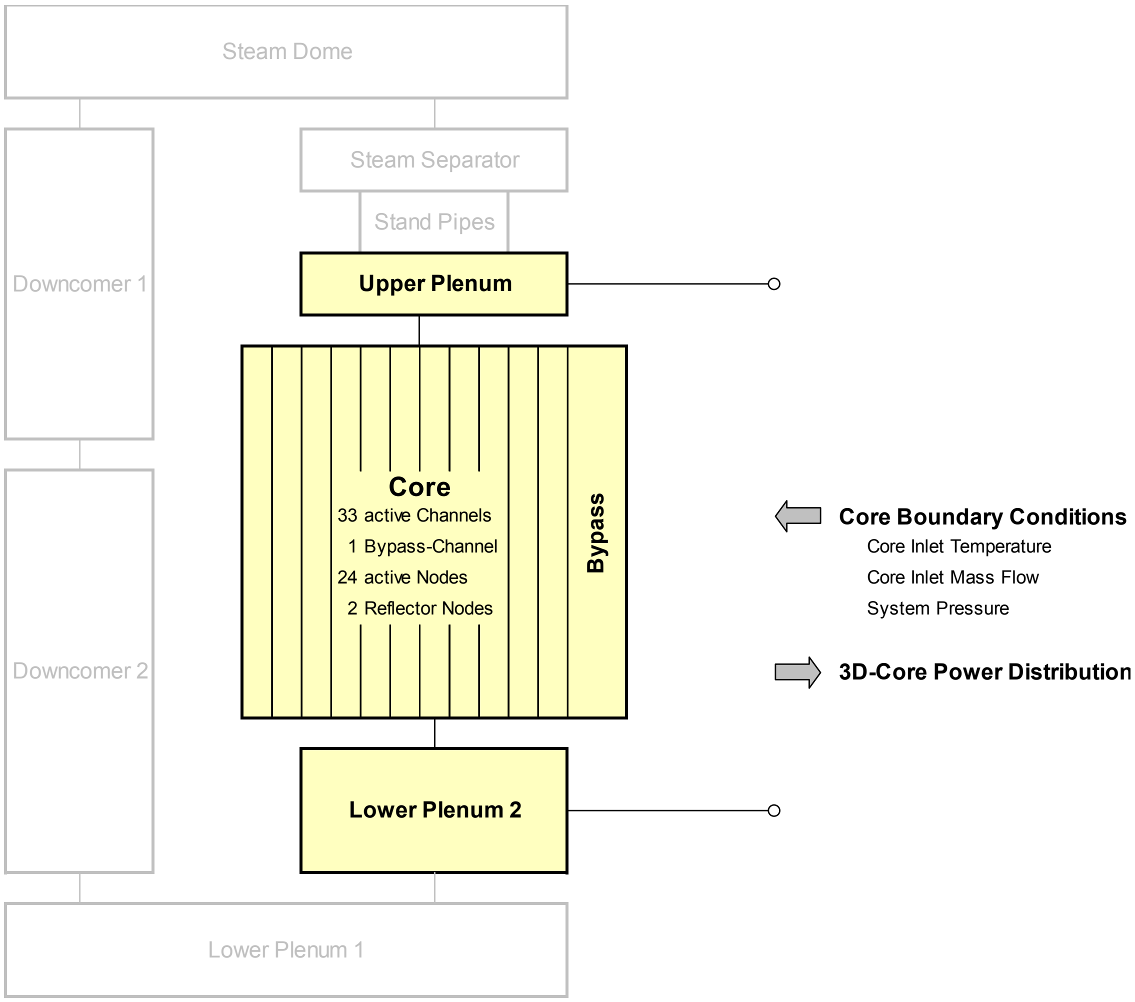
Plant Code S-RELAP5

> S-RELAP5 is based on RELAP5/MOD2 and incorporates elements of RELAP5/MOD3 and RELAP5-3D

> Special Features:

- 2-dimensional component model
- improved formulations for energy and momentum equations
- modified heat transfer and hydrodynamic constitutive models
- special fuel modeling
- suited for best-estimate licensing methods

S-RELAP5/RAMONA5 – Hydraulic Nodalisation



RAMONA5 – Core Model Components

POLGEN
cross section model

THORGEN
cross section model

Benchmark
cross section model

PRESTO-1 nodal method
1.5 group model (thermal diffusion simplified)

PRESTO-2 advanced nodal method
2 group model

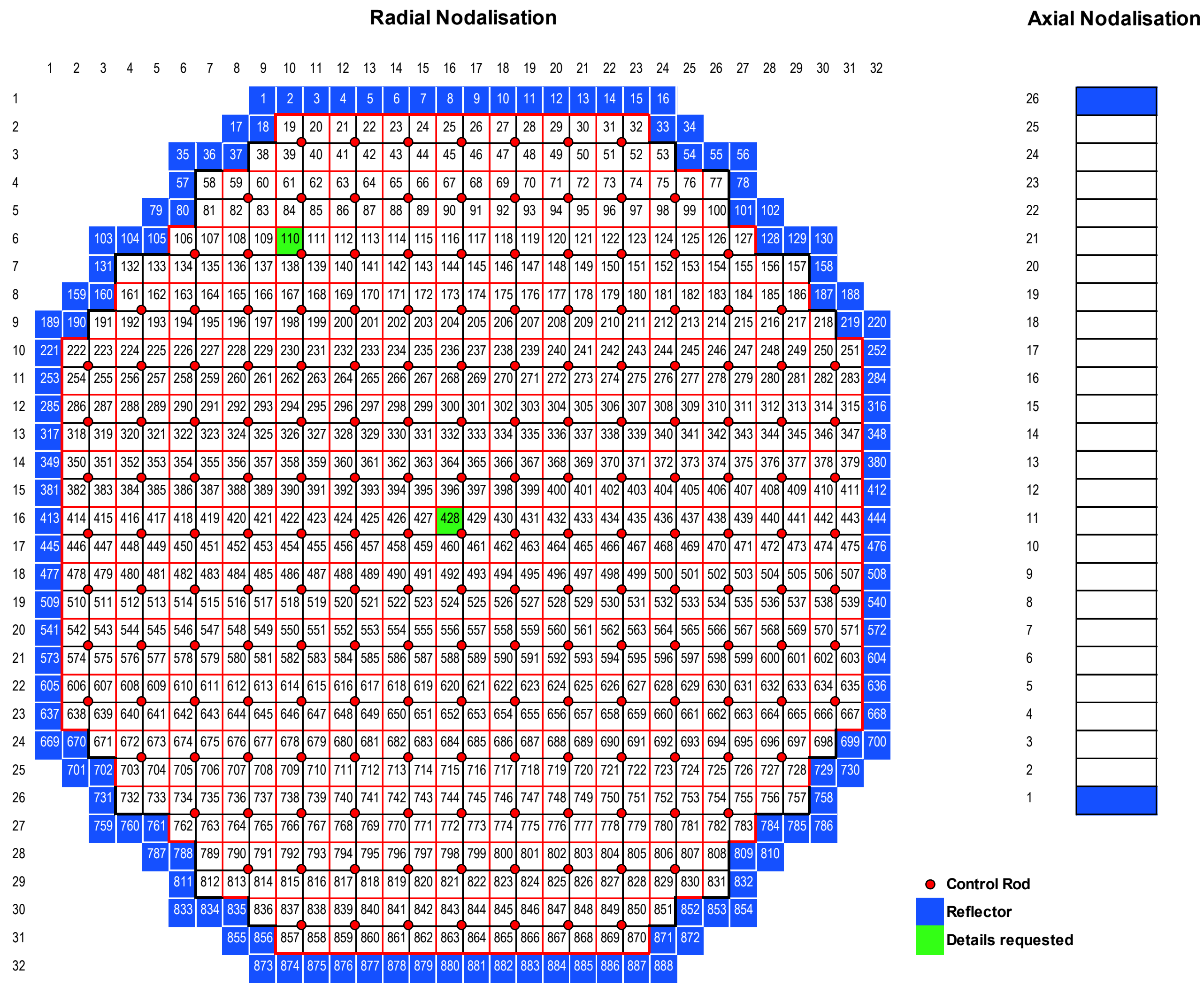
Power: prompt fission and decay heat (ANSI/ANS-5.1-1979), direct heating

dormant options: quarter assembly subdivision, free axial meshing and reflector nodes

Fuel Rod Model
for every hydraulic channel

Thermal Hydraulics Model
four equation, non-homogeneous, non-equilibrium, one-dim. two-phase flow model.

RAMONA5 – Benchmark Neutronic Nodalisation



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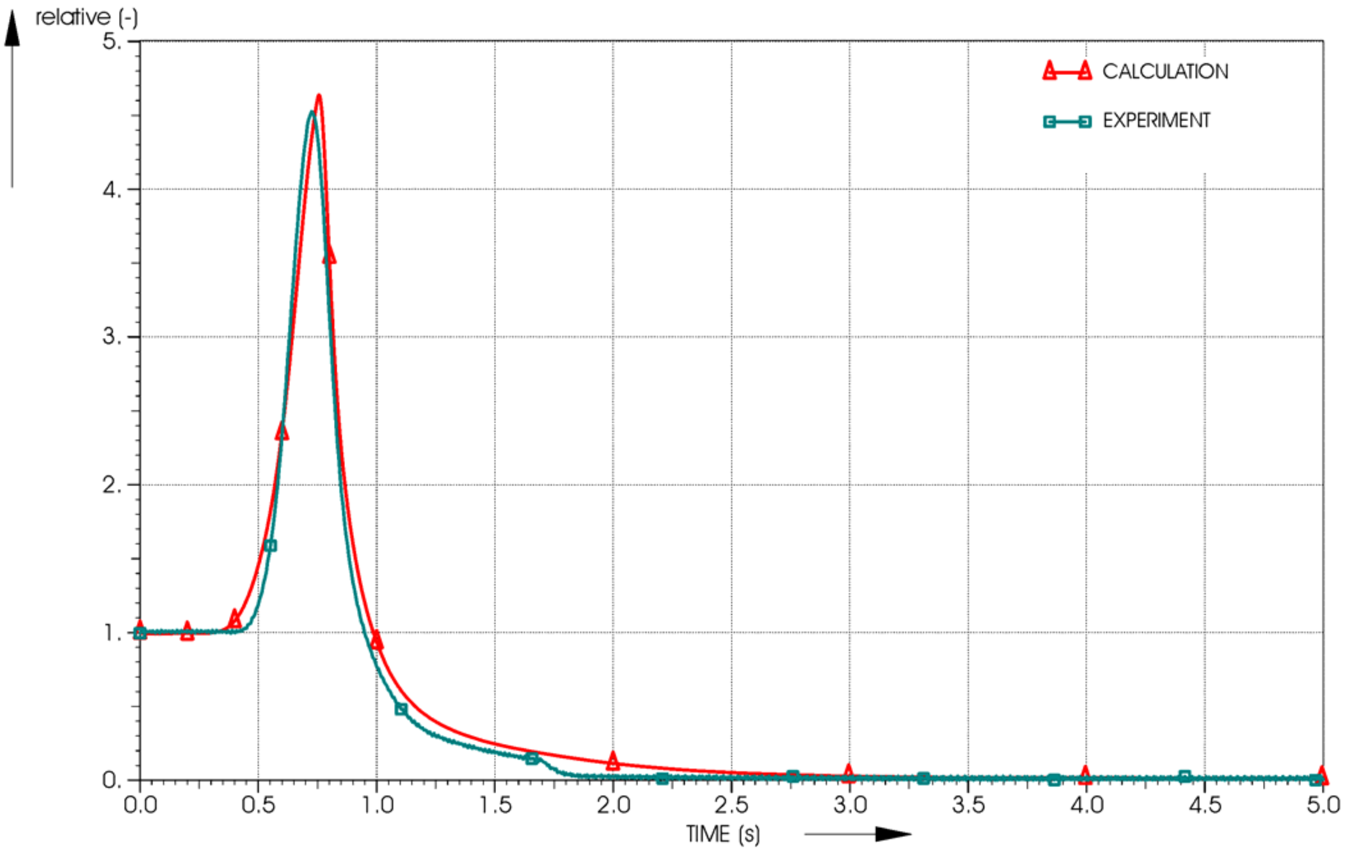
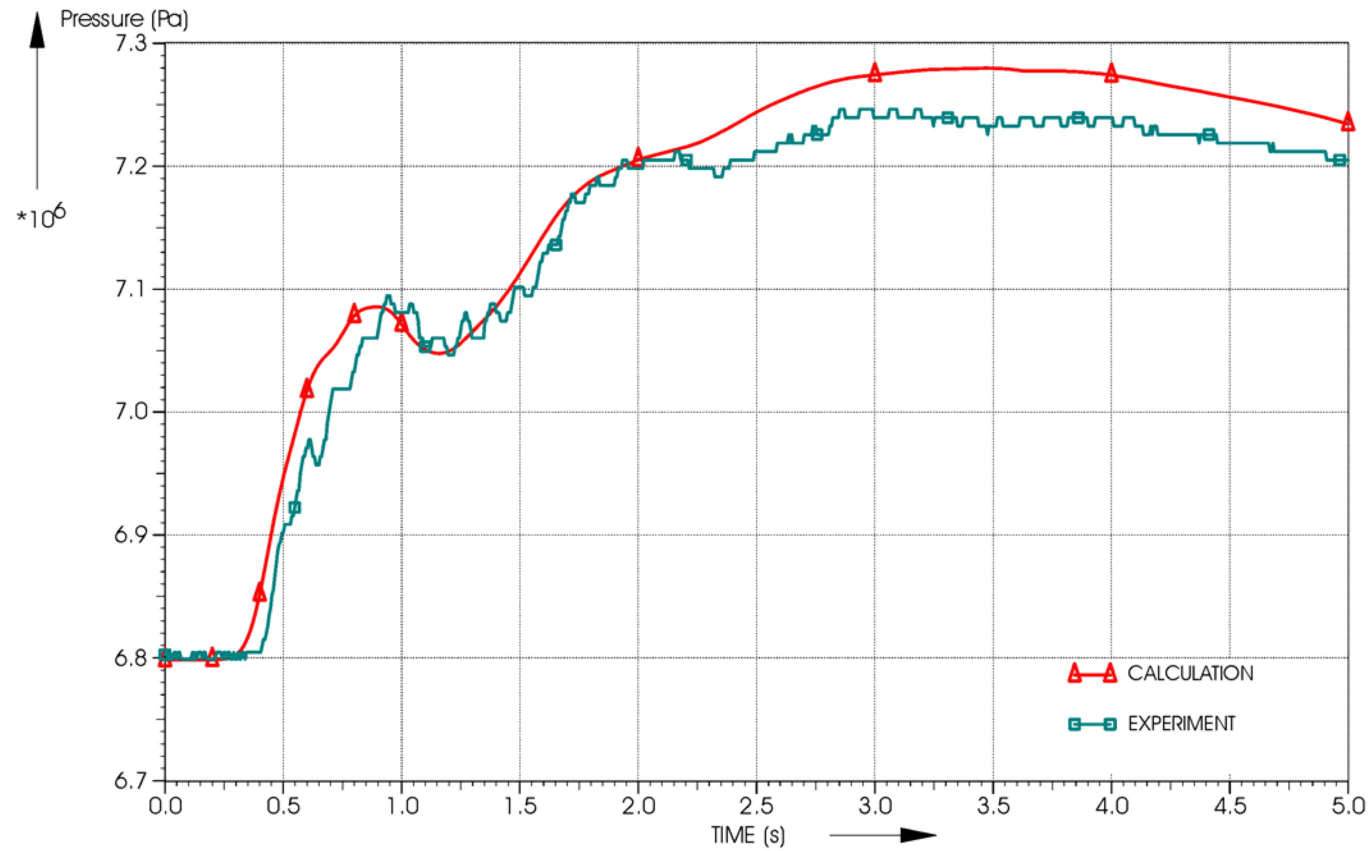
Exercise 3: Fission Rate



PEACH BOTTOM 2 : Turbine Trip Test TT 2
Exercise 3: Post Test Calculation with S-RELAP5/RAMONA5 (3d)
Fisson Rate

Exercise 3: Dome Pressure



PEACH BOTTOM 2 : Turbine Trip Test TT2
Exercise 3: Post Test Calculation with S-RELAP5/RAMONA5 (3d)
Pressure in Vessel

Advanced Transient Methodology - Summary

> Advanced method reduces OLMCPR by app. 0.07 compared to the conservative 1D methodology.

> The method allows more operational flexibility.

> The advanced transient method has already been approved by TÜV NORD.

> The code system S-RELAP5/RAMONA5 has been successfully applied to the BWR Turbine Trip Benchmark.

Bibliographic Data Sheet **NKS-89**

Title	3D Analysis Methods - Study and Seminar
Author(s)	Antti Daavittila
Affiliation(s)	VTT, Finland
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No. of illustrations	2
No. of references	0
Abstract	The first part of the report results from a study that was performed as a Nordic co-operation activity with active participation from Studsvik Scandpower and Westinghouse Atom in Sweden, and VTT in Finland. The purpose of the study was to identify and investigate the effects rising from using the 3D transient computer codes in BWR safety analysis, and their influence on the transient analysis methodology. One of the main questions involves the critical power ratio (CPR) calculation methodology. The present way, where the CPR calculation is performed with a separate hot channel calculation, can be artificially conservative. In the investigated cases, no dramatic minimum CPR effect coming from the 3D calculation is apparent. Some cases show some decrease in the transient change of minimum CPR with the 3D calculation, which confirms the general thinking that the 1D calculation is conservative. On the other hand, the observed effect on neutron flux behaviour is quite large. In a slower transient the 3D effect might be stronger. The second part of the report is a summary of a related seminar that was held on the 3D analysis methods. The seminar was sponsored by the Reactor Safety part (NKS-R) of the Nordic Nuclear Safety Research Programme (NKS).
Key words	3D transient computer codes, critical power ratio, BWR safety analysis

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