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Analysis of the OECD/NEA PWR Main Steam Line Break (MSLB) Benchmark Exercise 3 with the Coupled Code System RELAP5/PANBOX

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Abstract

The main purpose of the computational OECD/NEA PWR MSLB-Benchmark is the evaluation of the prediction capability of advanced code systems by means of a code-to-code comparison.

The postulated MSLB-transient is characterized by a strong non-symmetrical core thermal behaviour due to the feedback between neutron kinetics and plant thermal hydraulics. The analysis of such transients with pronounced spatial power distortion represents a considerable challenge for advanced code systems.

It is initiated by a break of one main steam line when the reactor TMI-1 is operated at nominal power. High heat removal through the break leads to a strong cooldown rate of the broken loop compared to the intact one. Under such conditions a power increase and a re-criticality of the core despite scram can not be excluded due to the negative reactivity coefficients.

The MSLB-Benchmark enfolds three exercises as follows: Exercise 1: integral plant simulation with best-estimate codes using the point kinetics, Exercise 2: multidimensional simulation of the core for given initial and boundary conditions, and Exercise 3: integral plant simulation with coupled, best-estimate codes using 3D-neutron kinetics models.

Forschungszentrum Karlsruhe (FZK) and Framatome Advanced Nuclear Power (ANP) Erlangen participated on the MSLB-Benchmark with the code system RELAP5/PANBOX for the Exercise 3: Based on the plant and core models elaborated for Exercise 1 and 2, an integral TMI-1 plant model was elaborated for Exercise 3. Special emphasis was put on the development of a multidimensional core model for the space-time kinetics. Two scenarios, the best-estimate (BE) and the return-to-power (RP) scenario, were investigated. Additional investigations aimed to investigate the influence of the coolant mixing on re-criticality and power increase.

Results of these investigations are presented and discussed in this report. It has been demonstrated that RELAP5/PANBOX is capable to simulate complex transient in a reliable way compared to the point kinetics approach.

Untersuchung der Phase 3 des OECD/NEA DWR Benchmarks zum Frischdampfleitungsbruch mit dem gekoppelten Programmsystem RELAP5/PANBOX

Zusammenfassung

Das Ziel des theoretischen OECD/NEA MSLB PWR Benchmarks ist die Vorhersagbarkeit fortgeschrittener Codesysteme durch einen Code-zu-Code-Vergleich zu bewerten.

Die postulierte MSLB-Transiente ist durch ein stark unsymmetrisches thermisches Kernverhalten gekennzeichnet, welches sich aus der Rückkopplung zwischen der Neutronenkinetik und der Kreislauf-Thermohydraulik ergibt. Die Analyse solcher Transienten mit erheblicher Verzerrung der räumlichen Leistungsverteilung stellt hohe Anforderungen an fortgeschrittene Codesysteme.

Es wird angenommen, dass die Transiente durch einen doppelendigen Bruch der Frischdampfleitung bei nominaler Leistung am Zyklusende (EOC) ausgelöst wird. Der Bruchöffnung folgt ein starker Wärmeaustrag über die Leckstelle, was zu erheblicher Abkühlung des defekten Kreislaufs im Vergleich zum intakten Kreislauf führt. Unter solchen Bedingungen ist eine Leistungssteigerung und Rekritikalität des Reaktors trotz Reaktorschnellabschaltung (RESA) nicht auszuschließen.

Der MSLB-Benchmark umfasst drei Phase wie folgt: Phase 1:Simulation der gesamten Anlage mit der Punktkinetik, Phase 2:dreidimensionale Kernsimulation für festgelegte Anfangs- und Randbedingungen und Phase 3: Integrale Anlagensimulation mit gekoppelten Programmsystemen unter Verwendung von 3D-Neutronenkinetik-Modellen.

Das Forschungszentrum Karlsruhe (FZK) und Framatome ANP Erlangen (früher SIEMENS/KWU) beteiligten sich gemeinsam an dem Benchmark. Die Phase 3 wurde mit dem Code RELAP5/PANBOX untersucht. Auf der Basis der für die Phase 1 und 2 entwickelte Anlagen- und Kernmodells wurde ein integrales Anlagemodell für die Phase 3 entwickelt. Besondere Aufmerksamkeit galt der Entwicklung eines mehrdimensionalen Kernmodells für die 3D-Neutronenkinetik. Zwei Szenarien Best-Estimate (BE) und Return-to-Power (RP) wurden für die Phase 3 untersucht. Zusätzliche Untersuchungen zielten darauf ab, den Einfluss der Kühlmittelvermischung auf die globale Reaktivität und Reaktorleistung zu bestimmen.

Ergebnisse dieser Untersuchungen werden vorgestellt und diskutiert. Es wird gezeigt, dass RELAP5/PANBOX gut geeignet ist, komplexe Transiente mit geringerer Unsicherheiten als die Punktkinetik zu simulieren.

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1 Introduction

As a logical continuation of former benchmarks devoted to 3D-reactor physics [Finn92a] and [Fraik93a] the OECD/NEA Nuclear Science Committee started the PWR Main Steam Line Break (MSLB) to verify the prediction capability of best-estimate coupled codes in analyzing complex transients. These code systems are characterized by a direct coupling between the one dimensional thermal hydraulics plant models with three-dimensional neutron kinetics models. To this class of codes belong e.g. ATHLET-QUABOX/CUBBOX [Lange99], ATHLET-DYN3D-R [Grund95], RELAP5/ PANBOX [Knoll98], RELAP5/PARCS [Miller98], TRAC-M/PARCS [Miller99], TRAC/PF1/NEM [Ivan99b], and CATHARE/CRONOS/FLICA4 [Royer98].

The postulated MSLB is initiated by the double-ended rupture upstream of a cross connect in a main steam line of the TMI-1 plant. Due to the intensive flashing on the secondary side of the broken SG after break opening, overcooling of the primary coolant system is initiated. The cooldown rate of the broken loop is much more pronounced than the one of the intact loop. This leads to an asymmetrical thermal core behavior characterized by significant spatial distortion of power within the core.

As a consequence of the overcooling of the primary circuit, large amounts of positive reactivity are added to the core so that a power increase and re-criticality of the core despite scram can not be excluded during the course of the transient.

The benchmark consists of three separate exercises as follows:

- **Exercise 1**: Integral plant simulation using point reactor kinetics,
- **Exercise 2**: Spatial reactor kinetics simulation with given time-dependent core boundary conditions,
- **Exercise 3**: Integral plant simulation with spatial reactor kinetics.

It is assumed that the MSLB occurs when the reactor is operated at EOC with nominal power. Moreover several assumptions were made according to the US-licensing criteria and aiming to maximize the heat removal over the broken leg SG. It was also assumed that the control rod with the highest worth is stuck out of the core when scram takes place. This assumption together with the asymmetrical core cooldown reinforce the three-dimensional effects within the core during course of the transient. Thus a multi-dimensional analysis is imperative to simulate the MSLB-transient realistically.

This report is devoted to the investigations performed with the coupled code RELAP5/PANBOX to analyze the Exercise 3. For this purpose the integral plant model developed for the Exercise 1 [Sanc00a] was extended with the multidimensional core model elaborated for the Exercise 2 [Böer99a].

In the frame of Exercise 3 two scenarios were investigated with RELAP5/PANBOX, namely (1) best-estimate scenario (BE) and (2) return-to-power scenario (RP).

All data needed for the development of the spatial reactor kinetics model as well as the problem assumptions were taken from final specifications [Ivan99].

The results obtained for the two scenarios are presented and discussed in this report. In addition a comparison between the results obtained for phase 1 and phase 3 are briefly presented. A sensitivity study was also performed to investigate the influence of the coolant mixing within RPV on e.g. parameters like core power and reactivity using multidimensional reactor kinetics (PANBOX).

The code-to-code comparison for Exercise 3 is being performed by PSU taking into account the results of all participants. A comparison report will be published by the PSU, where a detailed evaluation of the participant results will be summarized. Thus few graphs are presented in this report showing the RELAP5/PANBOX-results in comparison with other results.

2 The coupled code system RELAP5/PANBOX

The coupling of RELAP5 with the spatial kinetics code PANBOX extends its application to a wide range of postulated PWR plant transients. The RELAP5/PANBOX system performs best-estimate analysis covering core thermal hydraulics, reactor physics and plant dynamics.

2.1 The system code RELAP5/MOD3.2

The RELAP5 code is a widely used transient analysis code, developed by the USNRC [R5TH99] to simulate the LWR plant response for different postulated accidents including loss of coolant and reactivity insertion accidents as well as operational transients. The modeling scope covers most relevant primary and secondary plant circuits, control system, and core neutron kinetics.

2.2 The core code PANBOX

The code system PANBOX is a three dimensional neutron kinetics code coupled with multidimensional core thermal hydraulics module, developed by Siemens/KWU [Böer92] to perform PWR safety analysis and all kinds of transient in which the power distribution is significantly affected.

In PANBOX, the time-dependent few-group diffusion equation is solved in cartesian geometry using a semi-analytical nodal expansion method (NEM). It takes advantage of the transverse integration procedure to derive auxiliary one-dimensional diffusion equations which are then solved by various integration techniques.

In combination with an accurate and efficient pin power reconstruction methodology, the advanced thermal hydraulics code COBRA3-CP is capable of perform not only global neutron kinetics and thermal hydraulics calculations but also to facilitate the evaluation of important safety-related parameters. COBRA3-CP makes use of the parallel and open-channel methods.

2.3 Coupling strategy

The coupling between RELAP5 and PANBOX was realized via the general interface package EUMOD. It consists of a set of subroutines that allows external codes to be explicitly linked to RELAP5 [Rothe92]. The main functions of EUMOD are the transfer of data from RELAP5 to the external code, the call for execution of that code and the appropriate transfer of its results back to RELAP5. This process is repeated after each RELAP5-time step. Since no iteration is carried out between the neutron kinetics solution and RELAP5, the EUMOD interface affects neither the integration algorithm nor the physical models of RELAP5. Reduction of the time step is performed by the RELAP5 stability criteria. But the RELAP5-time step can be also reduced by PANBOX.

As can be seen in Figure 2-1, after each RELAP5-time step thermal hydraulic data are passed to PANBOX, where the 3-D power distribution is calculated and if necessary the safety-related parameters are evaluated. Then PANOX-data is adapted and transferred back to RELAP5, where the next time step is determined based on new power distribution.

The coupling options developed for RELAP5/PANBOX will be briefly described next [Knoll98], [Jack97].

2.3.1 Feedback from COBRA thermal hydraulics (external integration)

The thermal hydraulic boundary conditions at core inlet and outlet are passed from RELAP5 to COBRA. Then COBRA recalculates the core thermal hydraulics conditions using one channel per FA-grid or a coarser thermal hydraulic grid and the PANBOX power profile. The COBRA thermal hydraulics data are used to update the nuclear cross sections. Then a neutron kinetics time step is advanced using the updated cross sections. The so predicted new power distribution is collapsed to the RELAP5 core nodalization and passed back to RELAP5. This option allows the calculation of the thermal safety margins.

2.3.2 Feedback from RELAP5 thermal hydraulics (internal integration)

The core thermal hydraulic data is predicted by RELAP5 and passed to PANBOX, where it is mapped and interpolated onto the 3D-core nodalization. The resulting node-wise thermal hydraulic data are used to update the nuclear cross sections. Then a neutron kinetics time step is performed in PANBOX obtaining a new power distribution. This is collapsed to the RELAP5-nodalization and passed back to RELAP5. This option is useful for transients in which the RELAP5 flow model is more adequate than the COBRA one.

2.3.3 Feedback from RELAP5 thermal hydraulics and parallel COBRA calculation

This option is identical to the previous one, except that a COBRA core thermal hydraulic calculation is made in parallel to the RELAP5 one. The COBRA boundary conditions are taken from RELAP5 but the nuclear cross sections are updated using the RELAP5-data.

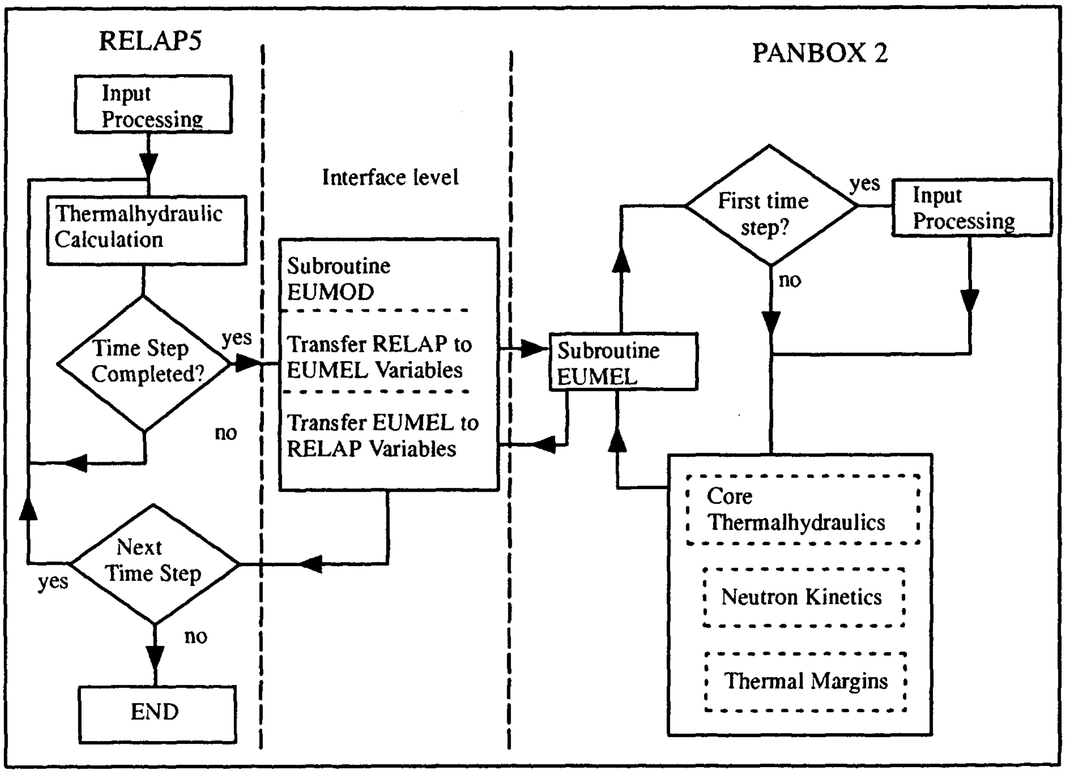


Figure 2-1 Flow logic for coupling of RELAP5 with PANBOX using EUMOD

3 Integral TMI-1 plant model

The integral plant model developed for the Exercise 1 [Sanc00a] is also used for Exercise 3 except the core modelling. It was completely replaced by a 3D-core model not only from the thermal hydraulics but also from the neutronic point of view. In Figure 3-1 the integral TMI-1 plant model developed for the MSLB-benchmark is given. The multi-dimensional core model is described in some detail in the following subsections.

3.1 RELAP5-model for core thermal hydraulics

The developed core model consist of 19 parallel coarse coolant channels. Each coarse channel represents the flow conditions of a certain group of fuel assemblies located in both the intact and affected core halves as recommended in the specifications, except the channel 19. This channel is formed by all fuel assemblies situated on the central assembly row which belongs to both intact and defect core half.

In Figure 3-2 the 2D-fuel assembly distribution of the TMI-1 core is shown. The fuel assemblies with the same number correspond to the respective coarse coolant channel as indicated in Figure 3-3. It can be seen that the channels with the number 1, 2, 3, 7, 8, 9, 13, 14, and 15 belong to the intact core half and channel number 4, 5, 6, 10, 11, 12, 16, 17, and 18 belong to the defect core half.

Each coarse coolant channel is represented by a pipe component with eleven axial nodes of the same height (0.3246 m) as shown in Figure 3-3. No coolant mixing between these core channels are considered, as the mass flow rate is very high during the transient progression.

3.2 RELAP5-model for the fuel assemblies

The 177 fuel assemblies are modeled as 19 representative fuel rods (RELAP heat structure), whereby the FA with the same number forms a representative fuel rod. This is subdivided in eleven axial nodes of the same height as the respective coolant channel, see Figure 3-4. In radial direction a representative fuel rod consists of three material zones: fuel pellet, gap and cladding and a total of 10 mesh points for temperature calculation. The “Doppler” temperature definition, the fuel rod gap conductance as well as the material properties for UO_2 and Zircaloy were taken from the final specification.

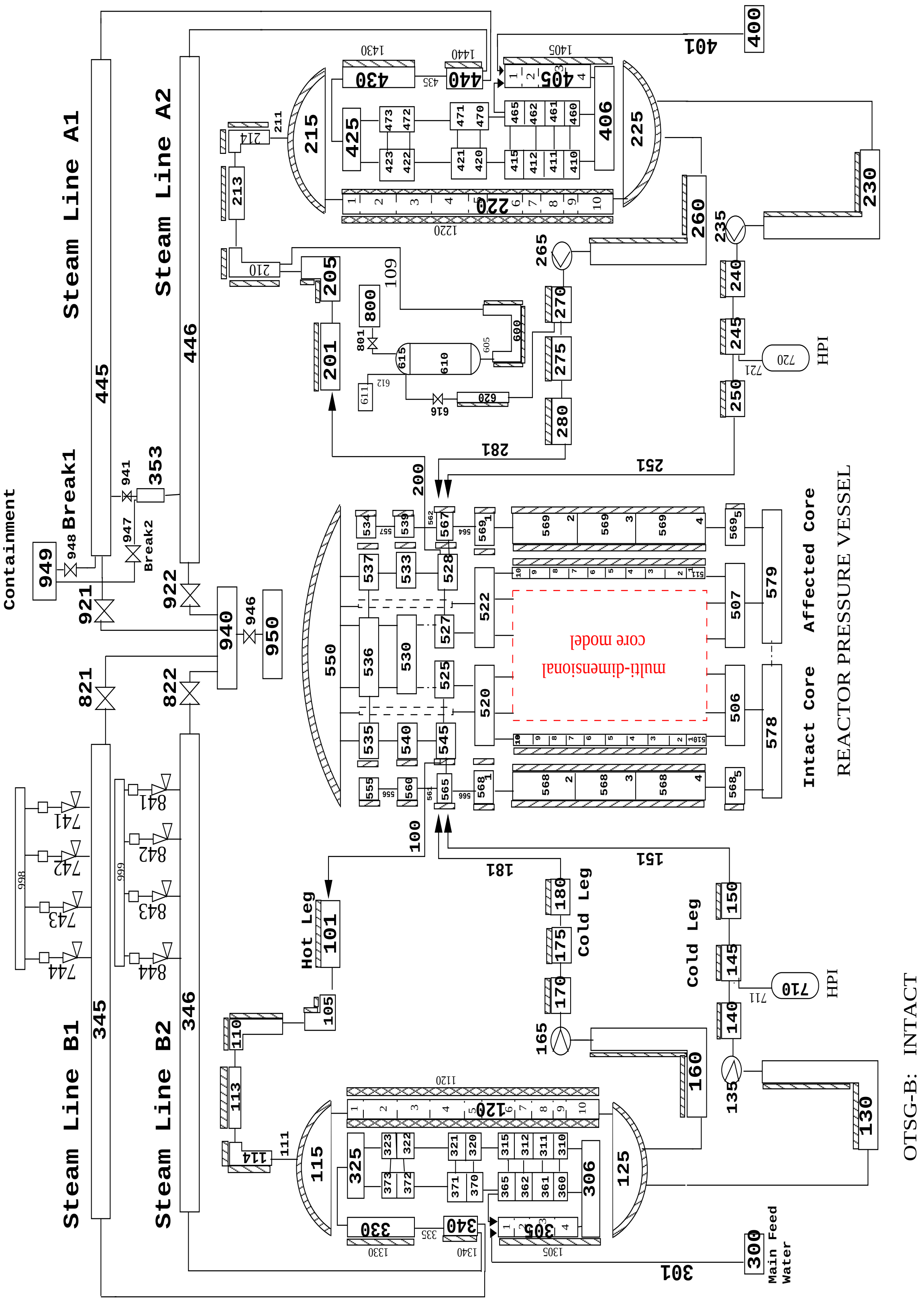


Figure 3-1 Integral plant model for the Exercise 3

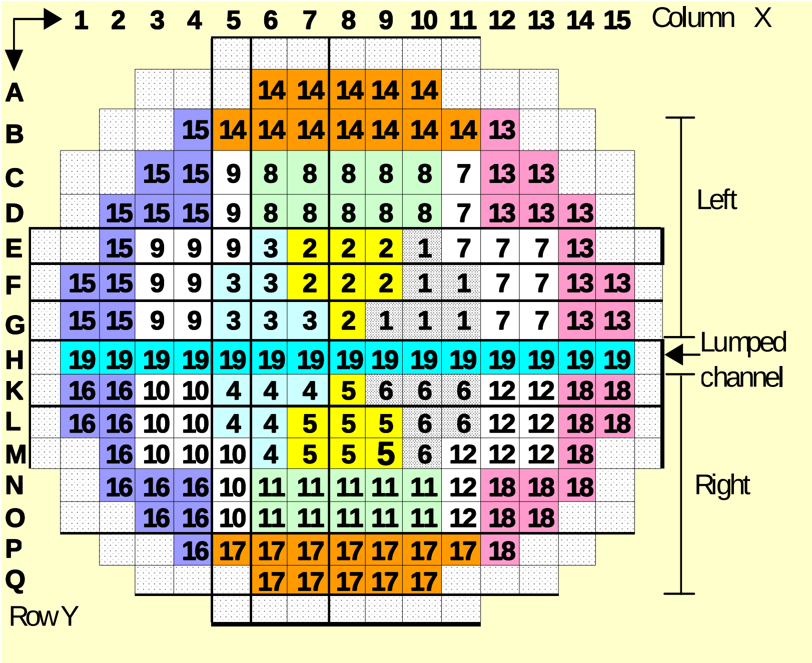


Figure 3-2 2D-distribution of fuel assemblies for mapping with coolant channels

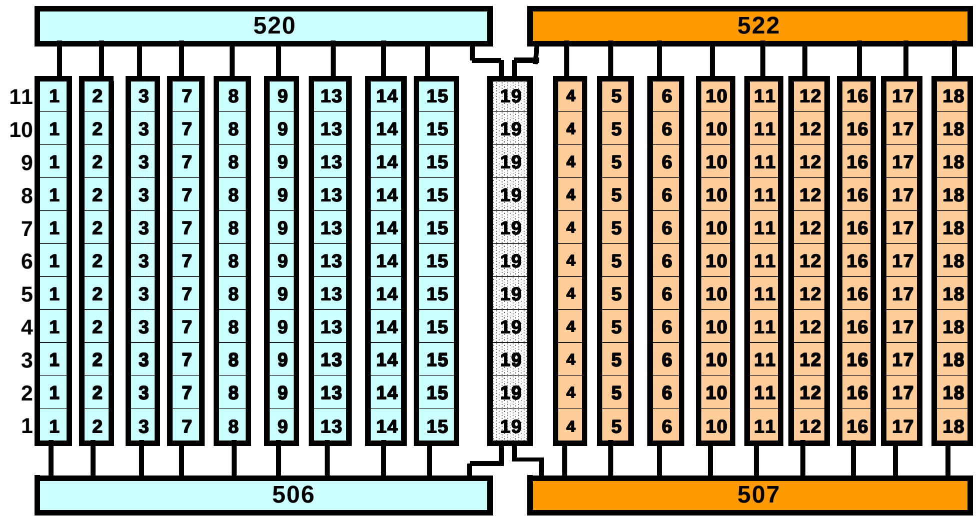


Figure 3-3 RELAP5 thermal hydraulic core model for mapping with fuel assemblies

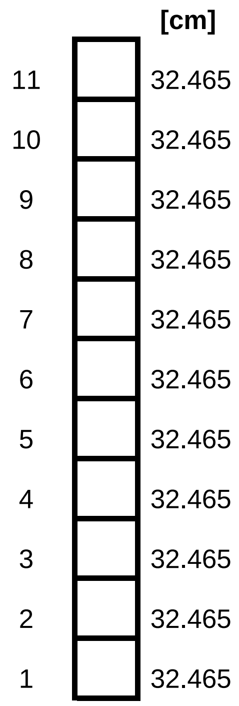


Figure 3-4 RELAP5 representative fuel rod model with axial nodalization

In Tab. 3-1 the number of fuel assemblies (and fuel rods) per representative fuel rod (RELAP5 heat structure) for each core half is listed.

Tab. 3-1 RELAP5 representative fuel rod compositon

Broken Loop-A				**Intact Loop-B**		
Channel number	**Number of fuel assemblies**	**Number of fuel rods**		**Channel number**	**Number of fuel assemblies**	**Number of fuel rods**
1	6	1248		4	6	1248
2	7	1456		5	7	1456
3	6	1248		6	6	1248
7	9	1872		10	9	1872
8	10	2080		11	10	2080
9	9	1872		12	9	1872
13	11	2288		16	11	2288
14	12	2496		17	12	2496
15	11	2288		18	11	2288
Coarse coolant channel						
19 (A+B)	15	3120				

3.3 3D-PANBOX reactor model

3.3.1 Core characteristics

In [Ivan99] the characteristics of the fuel assemblies for the TMI-1 core are extensively described. It consists of 30 FA-types, each one with an unique axial material composition map as indicated in the Tab. 3-2. The radial distribution of the FA-types within the core is shown in Figure 3-5 and in Figure 3-6 one-eight core symmetry sector with the average burn-up is given.

Tab. 3-2 FA- types chracteristics

FA-type	**Fuel assembly type characteristics**		
	Enrichment	**Burnable poison**	**Gadolinium pins**
1	4.00 w/o	No BP	No Gd pins
2	4.95 w/o	3.5% BP	4 Gd pins
3	5.00 w/o	3.5% BP pulled	4 Gd pins
4	4.95 w/o	3.5% BP	4 Gd pins
5	4.40 w/o	No BP	No Gd pins
6	5.00 w/o	3.5% BP	4 Gd pins
7	4.85 w/o	No BP	4 Gd pins
8	4.85 w/o	No BP	4 Gd pins
9	4.95 w/o	3.5% BP pulled	4 Gd pins
10	4.95 w/o	3.5% BP	4 Gd pins
11	4.85 w/o	3.5% BP pulled	4 Gd pins
12	4.95 w/o	3.5% BP	4 Gd pins
13	5.00 w/o	3.5% BP	4 Gd pins
14	5.00 w/o	No BP	8 Gd pins
15	5.00 w/o	No BP	8 Gd pins
16	4.95 w/o	3.5% BP pulled	4 Gd pins
17	4.95 w/o	3.5% BP	4 Gd pins
18	4.95 w/o	3.5% BP pulled	4 Gd pins
19	5.00 w/o	3.5% BP	4 Gd pins
20	4.40 w/o	No BP	No Gd pins
21	4.85 w/o	3.5% BP pulled	4 Gd pins
22	4.40 w/o	No BP	No Gd pins
23	4.95 w/o	3.5% BP	No Gd pins
24	4.95 w/o	3.5% BP pulled	4 Gd pins
25	5.00 w/o	No BP	8 Gd pins
26	5.00 w/o	No BP	4 Gd pins
27	5.00 w/o	No BP	No Gd pins
28	4.95 w/o	3.5% pulled	4 Gd pins
29	5.00 w/o	No BP	4 Gd pins
30	Radial reflector		

	x=1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
y=1	0	0	0	0	0	30	30	30	30	30	30	30	0	0	0	0	0
2	0	0	0	30	30	30	21	15	8	15	21	30	30	30	0	0	0
3	0	0	30	30	28	25	20	14	7	14	20	25	28	30	30	0	0
4	0	30	30	29	27	24	19	13	6	13	19	24	27	29	30	30	0
5	0	30	28	27	26	23	18	12	5	12	18	23	26	27	28	30	0
6	30	30	25	24	23	22	17	11	4	11	17	22	23	24	25	30	30
7	30	21	20	19	18	17	16	10	3	10	16	17	18	19	20	21	30
8	30	15	14	13	12	11	10	9	2	9	10	11	12	13	14	15	30
9	30	8	7	6	5	4	3	2	1	2	3	4	5	6	7	8	30
10	30	15	14	13	12	11	10	9	2	9	10	11	12	13	14	15	30
11	30	21	20	19	18	17	16	10	3	10	16	17	18	19	20	21	30
12	30	30	25	24	23	22	17	11	4	11	17	22	23	24	25	30	30
13	0	30	28	27	26	23	18	12	5	12	18	23	26	27	28	30	0
14	0	30	30	29	27	24	19	13	6	13	19	24	27	29	30	30	0
15	0	0	30	30	28	25	20	14	7	14	20	25	28	30	30	0	0
16	0	0	0	30	30	30	21	15	8	15	21	30	30	30	0	0	0
17	0	0	0	0	0	30	30	30	30	30	30	30	0	0	0	0	0

Figure 3-5 2D-fuel assembly type map within the whole core (17x17)

	8	9	10	11	12	13	14	15
H	1 52.863	2 30.192	3 56.246	4 30.852	5 49.532	6 28.115	7 53.861	8 55.787
K		9 57.945	10 30.798	11 55.427	12 29.834	13 53.954	14 25.555	15 49.166
L			16 57.569	17 30.218	18 54.398	19 27.862	20 23.297	21 47.300
M				22 49.712	23 28.848	24 52.846	25 40.937	
N					26 48.746	27 23.857	28 41.453	
O						29 37.343	A B	
P								
R								

Figure 3-6 2D-fuel assembly type mapping in the core (A: FA-type, B: average burn-up)

A total of 633 compositions (438 unrodded and 195 rodded compositions) are defined for the whole core with respective sets of cross-section, see Figure 3-7. The composition number at each axial node (z-coordinate) is indicated for each FA-type. For each composition, two-group constant data are provided by the specifications (data files **nemtab**, **nemtabr.rp** and **nemtabr**).

In these data files the cross section is given as function of the fuel temperature and moderator density.

The data file **nemtab** is for unrodded compositions, i.e. numerical nodes without control rods, for both scenarios (BE and RP). The files **nemtabr** and **nemtabr.rp** are for rodded compositions, i.e. numerical nodes with control rods, of the BE and RP scenarios, respectively.

z\fa	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
1	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24
2	1	16	34	49	64	79	94	109	124	139	154	169	184	199	214	229	244	259	274	289	304	319	334	349	364	379	394	409	424	25
3	2	17	35	50	65	80	95	110	125	140	155	170	185	200	215	230	245	260	275	290	305	320	335	350	365	380	395	410	425	25
4	3	18	36	51	66	81	96	111	126	141	156	171	186	201	216	231	246	261	276	291	306	321	336	351	366	381	396	411	426	25
5	4	19	37	52	67	82	97	112	127	142	157	172	187	202	217	232	247	262	277	292	307	322	337	352	367	382	397	412	427	25
6	5	20	38	53	68	83	98	113	128	143	158	173	188	203	218	233	248	263	278	293	308	323	338	353	368	383	398	413	428	25
7	6	21	39	54	69	84	99	114	129	144	159	174	189	204	219	234	249	264	279	294	309	324	339	354	369	384	399	414	429	25
8	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
9	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
10	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
11	7	22	40	55	70	85	100	115	130	145	160	175	190	205	220	235	250	265	280	295	310	325	340	355	370	385	400	415	430	25
12	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
13	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
14	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
15	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
16	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
17	8	23	41	56	71	86	101	116	131	146	161	176	191	206	221	236	251	266	281	296	311	326	341	356	371	386	401	416	431	25
18	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
19	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
20	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
21	9	27	42	57	72	87	102	117	132	147	162	177	192	207	222	237	252	267	282	297	312	327	342	357	372	387	402	417	432	25
22	10	28	43	58	73	88	103	118	133	148	163	178	193	208	223	238	253	268	283	298	313	328	343	358	373	388	403	418	433	25
23	11	29	44	59	74	89	104	119	134	149	164	179	194	209	224	239	254	269	284	299	314	329	344	359	374	389	404	419	434	25
24	12	30	45	60	75	90	105	120	135	150	165	180	195	210	225	240	255	270	285	300	315	330	345	360	375	390	405	420	435	25
25	13	31	46	61	76	91	106	121	136	151	166	181	196	211	226	241	256	271	286	301	316	331	346	361	376	391	406	421	436	25
26	14	32	47	62	77	92	107	122	137	152	167	182	197	212	227	242	257	272	287	302	317	332	347	362	377	392	407	422	437	25
27	15	33	48	63	78	93	108	123	138	153	168	183	198	213	228	243	258	273	288	303	318	333	348	363	378	393	408	423	438	25
28	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26	26

Figure 3-7 Fuel assembly composition matrix for the whole core

3.3.2 Core model

For the PANBOX-model a FA represents a radial node. In axial direction, a FA is subdivided into 26 nodes for the active core height and two additional nodes for the axial reflector, see Figure 3-8. The two additional nodes compared to the 24 specified in the specification originate from the subdivision of the two middle nodes with a height of 29.76 cm in the original nodalization. The fuel rod itself is subdivided in 19 radial intervals, i.e. 16 intervals for the UO_2-pellet, 1 interval for the gap, and 3 intervals for the cladding.

Summarizing, the 3D-neutron kinetics problem consists of a total number of 6748 nodes. The two-energy group neutron diffusion equation is solved for this computational grid using the semi-analytical NEM with the quadratic transverse leakage option.

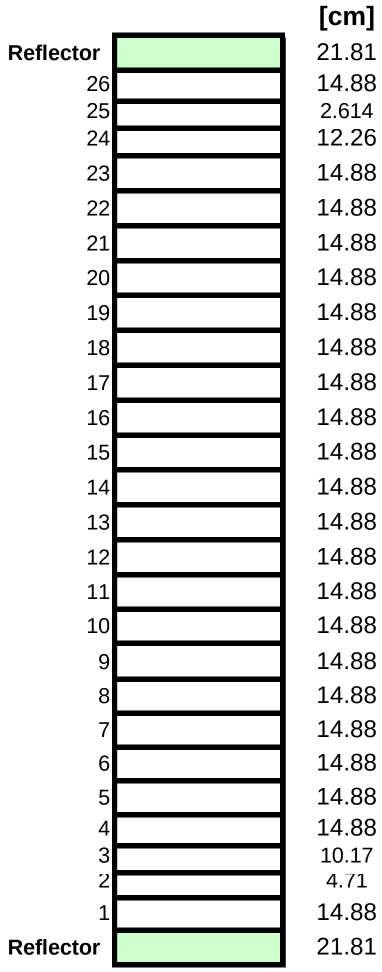


Figure 3-8 PANBOX axial nodalization of the fuel rod

3.3.3 Decay heat model

The decay heat tables for both scenarios (BE and RP) were taken from the specifications. The 3D-distribution of the decay heat in PANBOX is based on weighted fission rates.

3.3.4 Core cross section sets

In [Ivan99] the respective cross-sections data set for each material composition are provided. Neutronic data like diffusion coefficients, macroscopic cross-sections for scattering, absorption, and fission for two-energy groups are contained in the two-dimensional look-up tables (files nemtab, nemtabr) as function of moderator density and fuel temperature. The group inverse neutron velocities, decay constants and fraction of delay neutrons are included in the look-up tables, too.

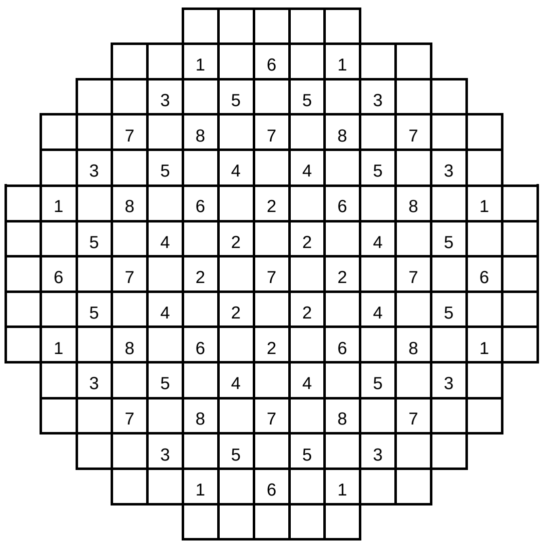
Based on actual reactor conditions during the transient, the appropriate total cross-sections are obtained by PANBOX from the look-up tables using a linear interpolation scheme.

To predict the actual cross section of the bottom reflector a fuel temperature equal to the inlet coolant temperature and coolant density equal to the inlet coolant density are used. The same procedure is used for the reflector on the top. Thus the influence of the plant thermal hydraulics on the neutron balance in the core can be directed calculated without the necessity of using reactivity feedback coefficients.

3.3.5 Control rod banks

There are eight control rod banks in the core. Seven of them are full length control rods (342.7055 cm). Their radial arrangement within the core is given in Figure 3-9. The group eight consist of part-length control rods and it has no contribution in this problem. At steady-state conditions the sixth CA-groups are assumed to be totally withdrawn. Only the CA-group 7 is 90 % withdrawn. But the stuck-rod is always 100 % withdrawn.

The introduction of the CA-groups after scram is modelled in PANBOX according to Specifications, i.e. all control rods are inserted into the core except the stuck rod. The stuck-rod is in the position N12, see Figure 3-2.



Bank	No. Rods	Purpose
1	4	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Regulating
6	12	Regulating
7	9	Regulating
8	8	APSR

Figure 3-9 Arrangement of the control rod groups within the core

4. Definition of the MSLB-scenarios

For the multi-dimensional analysis of the core behaviour in interaction with the plant thermal hydraulics two scenarios were specified within the Exercise 3 as follows:

- Best-Estimate scenario (BE): It is based on current licensing practice.
- Return-to-power scenario (RP): It was specified upon request of participants to better test the coupling of 3D kinetics models with the plant thermal hydraulics models.

To increase the likelihood of return-to-power the values of tripped rod worth, stuck-rod and decay heat were reduced compared to the BE-scenario, see Tab. 4-1.

Tab. 4-1 Values of TRW and stuck-rod

Rod type	Best-estimate (BE)	Return-to-power (RP)
Tripped rod worth (%K/K)	4.5698 $ (100 %)	3.0575 (67 %)
Stuck-rod worth (%K/K)	0.6857 (100 %)	0.3925 (57 %)

In general the same boundary conditions and assumptions are valid for BE and RP. They are identical with the ones prescribed in the specifications for the Exercise 1.

5 Steady-state calculation

The steady-state calculation was carried out with RELAP5/PANBOX using the same plant model of Exercise 1 but with a multidimensional core model. In Tab. 5-1 the predicted parameters are compared to those given in the Specifications. In this context an eigenvalue calculation is performed to get a critical reactor conditions at transient begin. It can be seen that prediction and specified data are in good agreement. Based on this results a transient calculation can be performed with the coupled system.

Tab. 5-1 Nominal plant parameters predicted with RELAP5/PANBOX versus data

Parameters	**Specifications**	**RELAP5/ PANBOX**	**Deviation (%)**
Core power, MW_{th}	2772	2772	0
Coolant temperature cold leg (K)	563.76	563.25	0.09
Coolant temperature of hot leg	591.43	591.31	0.02
Lower plenum pressure (MPa)	15.36	15.38	- 0.13
Upper plenum pressure (MPa)	15.17	15.22	- 0.33
Primary system pressure (MPa)	14.96	14.97	- 0.06
Total primary circuit flow rate	17692	17613	0.45
Core flow rate (kg/s)	16052	15934	0.73
Bypass flow rate (kg/s)	1549	1560	- 0.71
SG outlet pressure (MPa)	6.41	6.39	0.31
SG outlet superheating (K)	19.67	18.45	6.2
Initial SG-fluid inventory (kg)	26000	26546	- 2.1

6 Transient calculation

The coupled calculation with RELAP5/PANBOX was performed on a HP-UX.10.20 Workstation. Typical CPU-time for 100 s problem time amounts about 8 h with a minimum time step of 2.5 ms. It has to be noted that most of CPU-time was consumed by PANBOX.

6.1 Predicted sequence of events

In Tab. 6-1 a list of events predicted with RELAP5/PANBOX for both scenarios is given. Initially there are no differences between BE and RP until scram is reached. Later on differences arise between both scenarios as a consequence of assumed values for the TRW, stuck-rod, and decay heat power. As expected, RELAP5/PANBOX predicts a re-criticality and a considerable power increase despite scram for the RP-scenario (35.1 %) than for the BE-scenario (6.5 %). The time for achieving the power peak differs a little between the RP (65.8 s) and BE (67.2 s) scenario.

Tab. 6-1 Main events for both scenarios a) best-estimate and b) return-to-power

Events	Best-estimate (BE)	Return-to-power (RP)
Overpower trip (P>114%)	5.68 s	5.68 s
Control rods insertion	6.08 s	6.08 s
Turbine trip	6.08 s	6.08 s
Turbine valve close (intact loop)	6.58 s	6.58 s
High pressure injection trip (P<11.34	11.69 s	12.19 s
High pressure injection (25 s delay)	36.70 s	37.20 s
Re-criticality	No (- 1.33 $)	Yes at 60.9 s up to 67.6 s
Maximal power peak after scram	180 MW (6.5 %) at 67.2 s	973 MW (35.1 %) at 65.8 s

In the subsequent section the time history of selected thermal hydraulic and neutronic parameters are presented and discussed for both scenarios.

6.2 Time history of selected parameters

6.2.1 Global plant behaviour

It is assumed that the break opens when the reactor is being operated with full power at EOC. Immediately after break opening, a sudden depressurization of the broken SG-secondary side takes place, see Figure 6-1. A considerable amount fluid is dumped through both breaks into the containment, see Figure 6-2 and Figure 6-3.

As a consequence of the improved heat transfer from the primary to the secondary side, especially over the broken SG, the coolant temperature of the primary system undergoes a strong overcooling, see Figure 6-4. Note that the cooldown rate of the broken loop-A is much larger than that of the intact loop-B. Its coolant temperature decreases mainly due to the coolant mixing in the RPV.

As consequence of the overcooling of the primary circuit the fluid contraction and thus pressure reduction is significant, see Figure 6-5. The corresponding core average coolant temperature and Doppler temperature for BE and RP are shown in Figure 6-6 and Figure 6-7. Due to the cross-section dependency on Doppler and moderator temperature, a steady insertion of positive reactivity into the core occurs after scram until around 70 s, see Figure 6-8. At the time the broken SG is almost empty, so that no more cooldown of the primary circuit is possible. The differences in the TRW between RP and BE can be seen in Figure 6-8 just after scram. Since the power increase after scram depends on the rate of positive reactivity insertion, the RP-scenario experiences a considerable power increase compared to the 6.5 % power peak of the BE-scenario, see Figure 6-9.

The fluid inventory of the broken and intact SG shows different trends during the transient, see Figure 6-10 and Figure 6-11. After scram the turbine connection of the intact loop is isolated by closing of the MSIV. Then SG-inventory increases a little until the feed-water injection reaches zero within 10 s after scram. At the same time the pressure in intact SG-secondary side increases until the set point for the actuation of the MSSV is reached. Then it decreases continuously throughout the transient, see Figure 6-12.

The power removed over both SGs differs from each other a lot as shown in Figure 6-13 and Figure 6-14. In case of the intact loop the heat removal falls down very much after scram due to the pressure reduction of the primary circuit and the concurrent secondary pressure increase. After about 30 s in the transient, a low reverse heat transfer, i.e. from the secondary to the primary circuit, takes place, Figure 6-14. This contributes to a continuous decrease of the secondary pressure of intact loop too.

The large amount of steam and liquid dumped through the breaks is given for both scenarios (RP and BE) in Figure 6-15 and Figure 6-16.

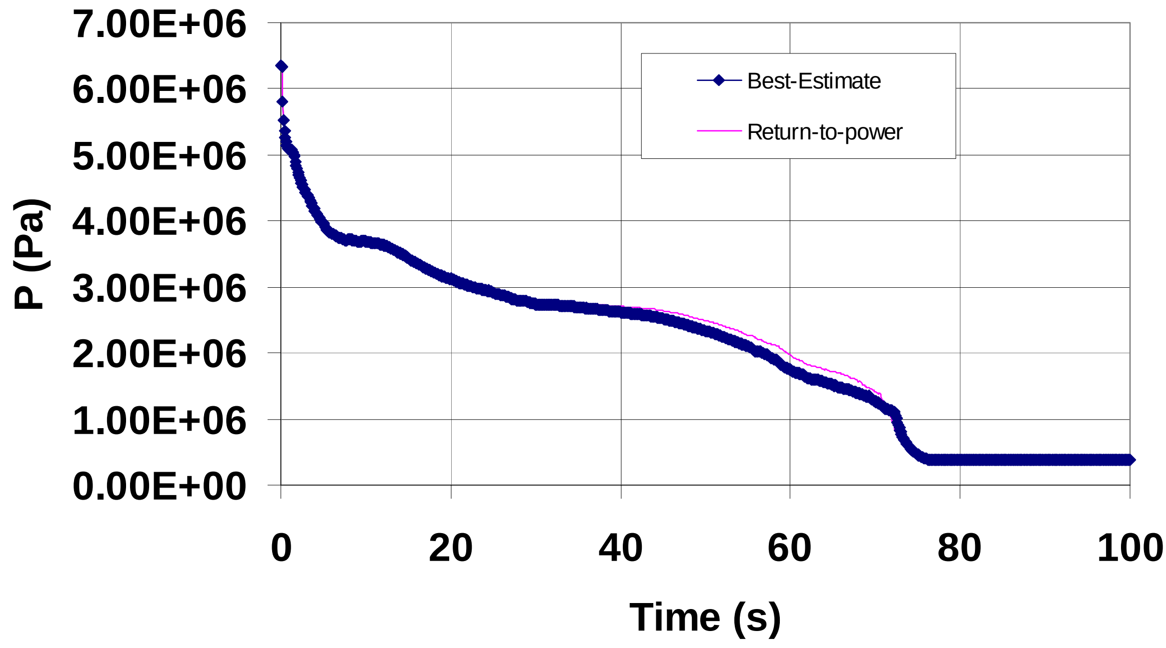


Figure 6-1 Secondary system pressure of broken loop-A

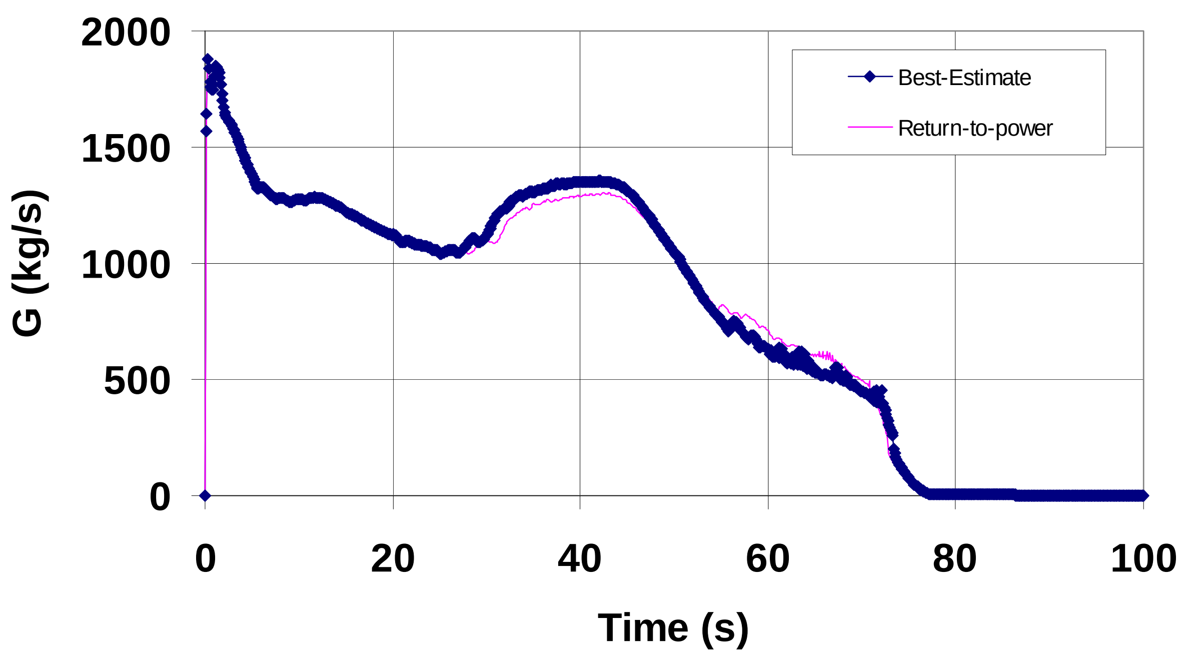


Figure 6-2 Outflow rate through large break

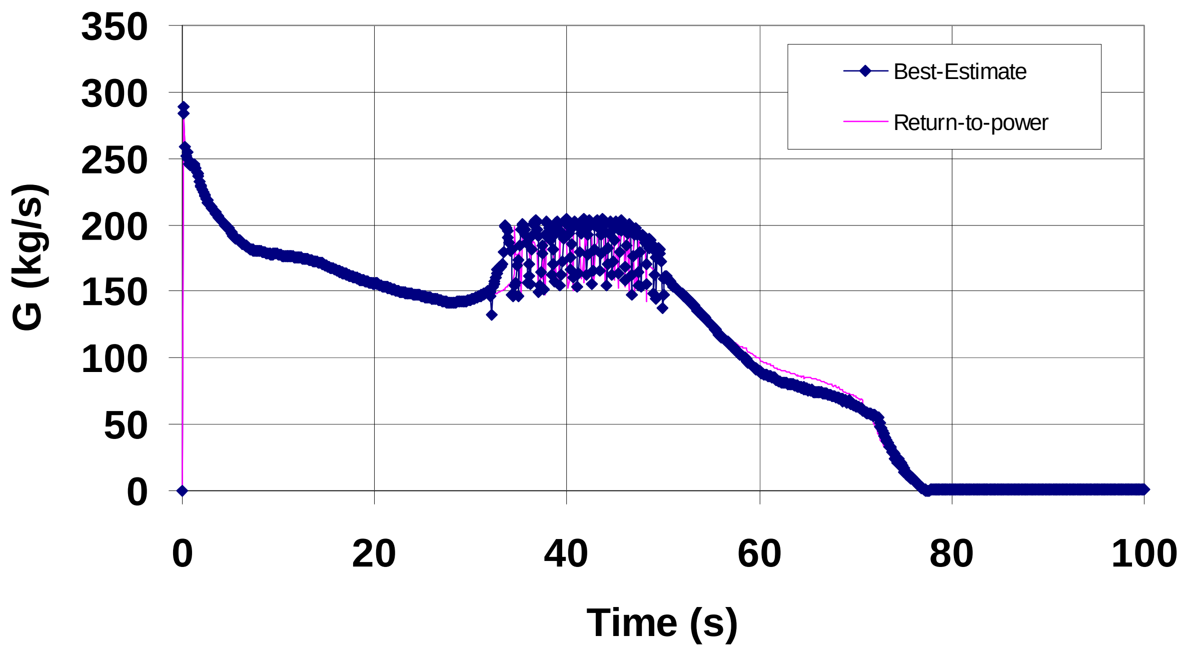


Figure 6-3 Outflow through small break

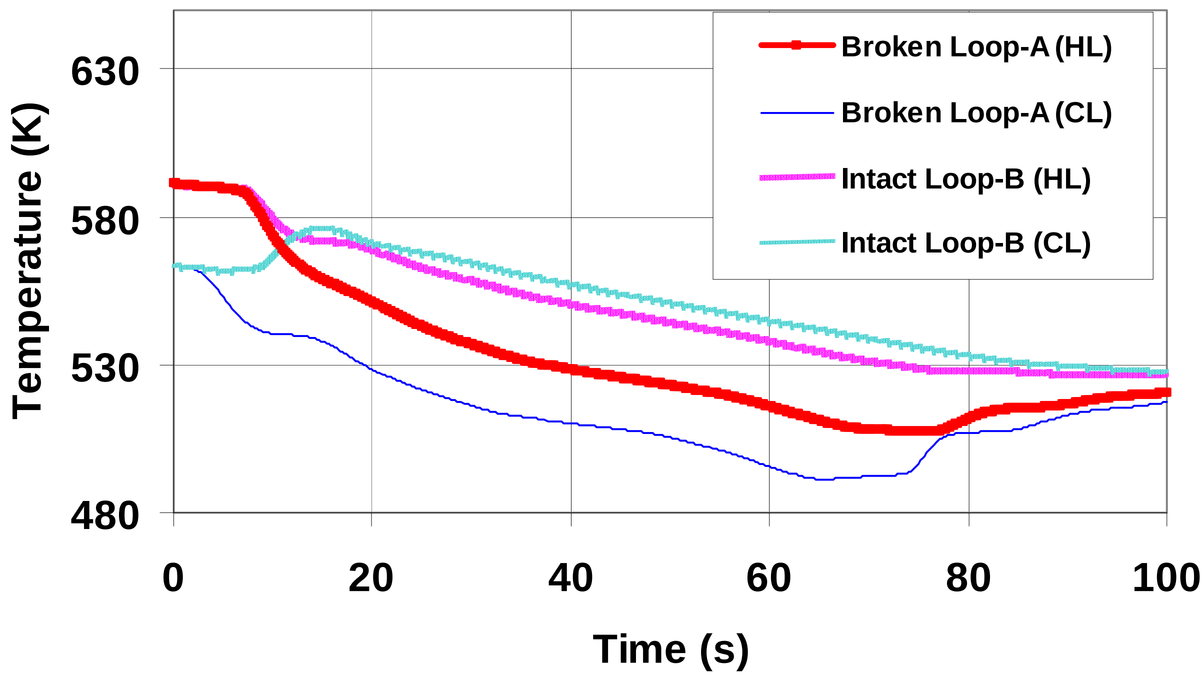


Figure 6-4 Cold/hot leg coolant temperature of both loops (BE case)

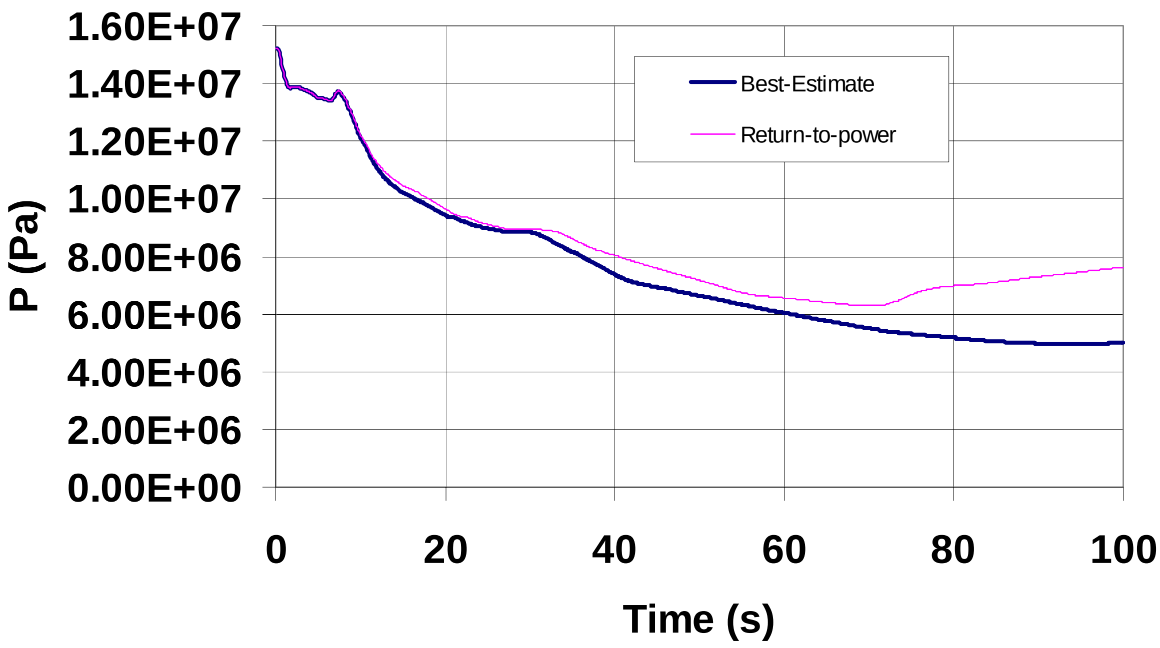


Figure 6-5 Primary system pressure of hot leg (BE and RP case)

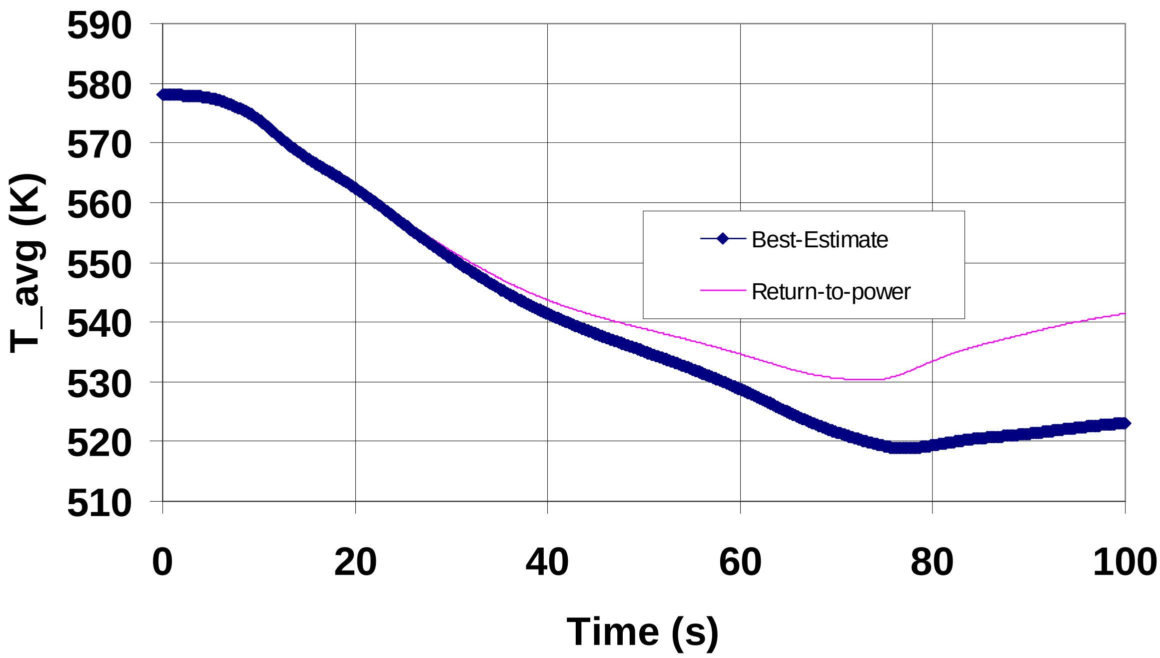


Figure 6-6 Core average moderator temperature (BE and RP case)

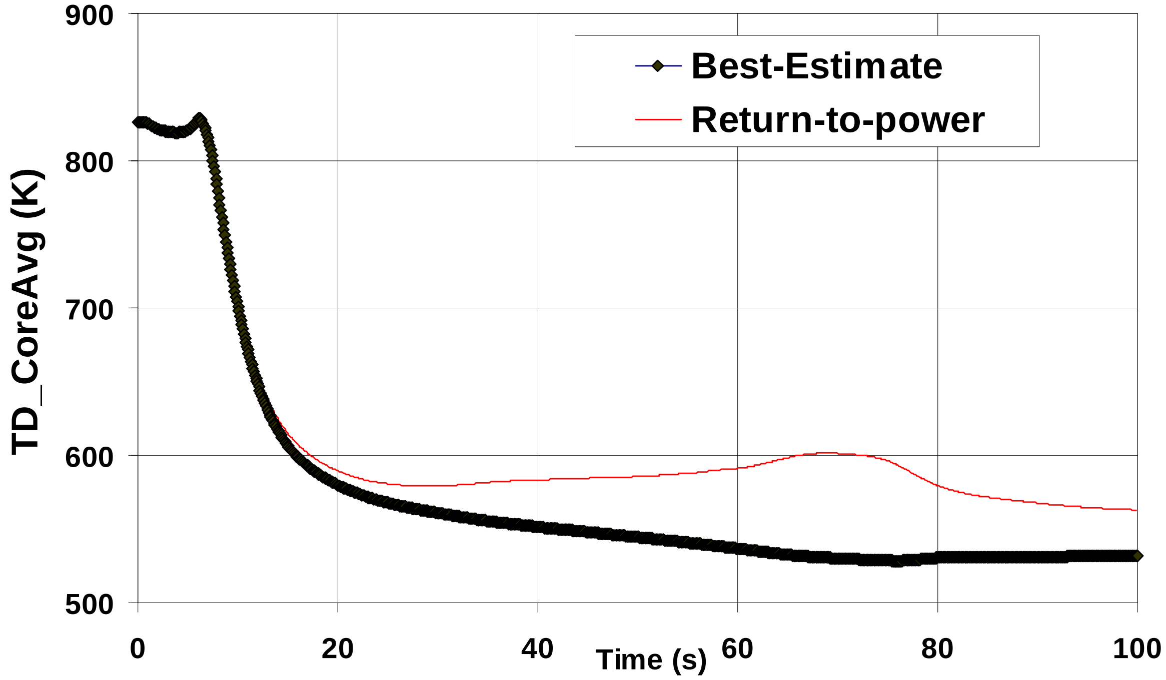


Figure 6-7 Core average Doppler temperature (BE and RP case)

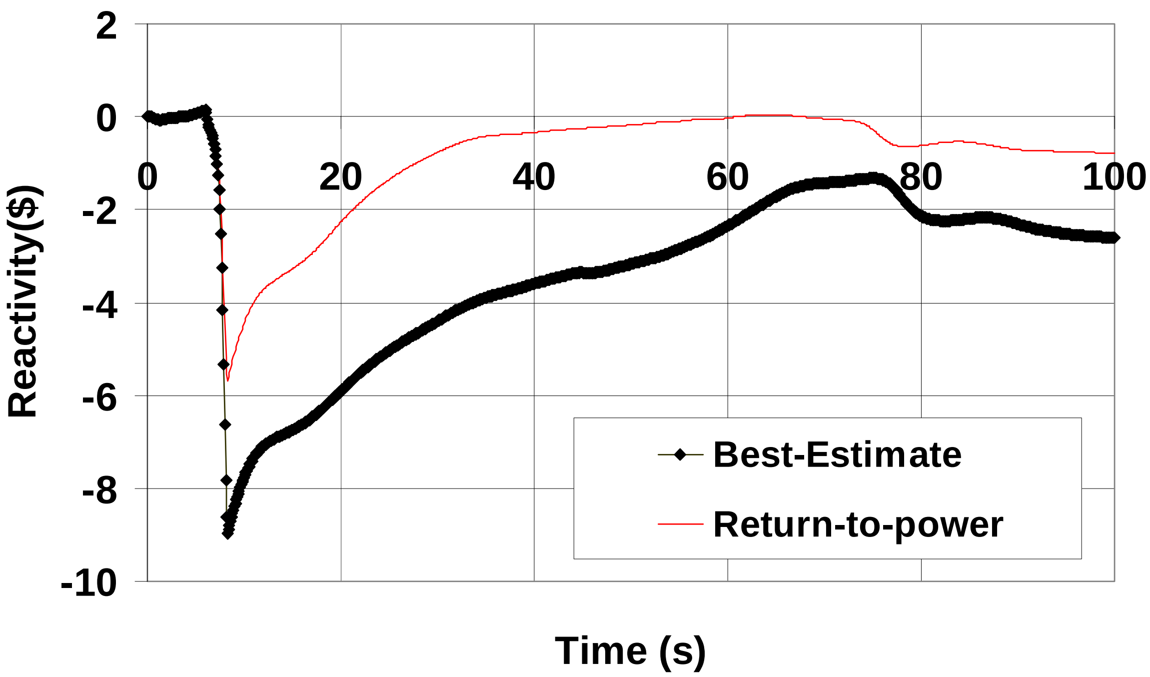


Figure 6-8 Total core reactivity (BE and RP case)

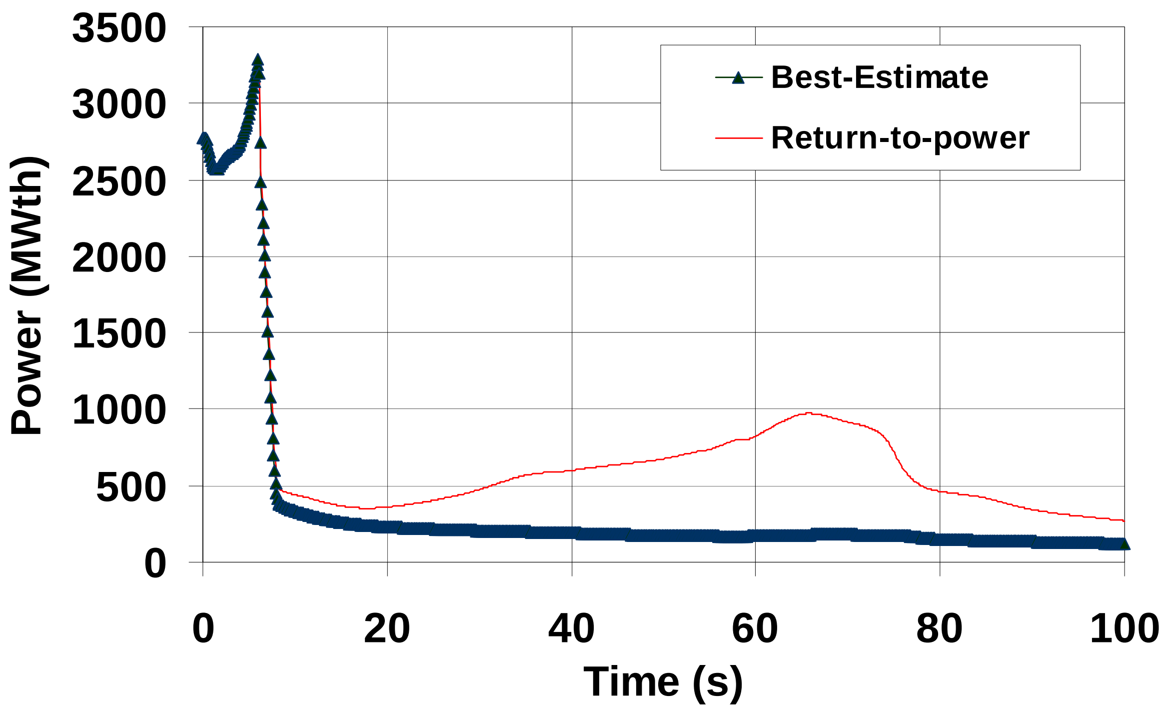


Figure 6-9 Predicted total core power (BE and RP case)

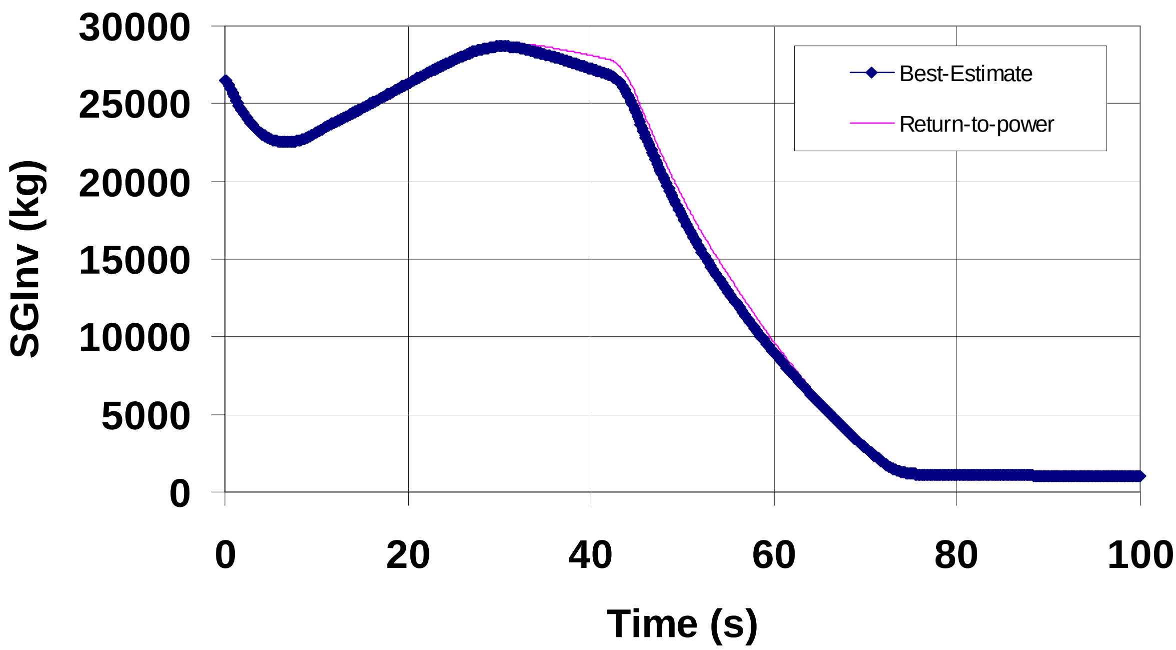


Figure 6-10 Fluid inventory of broken steam generator (BE and RP case)

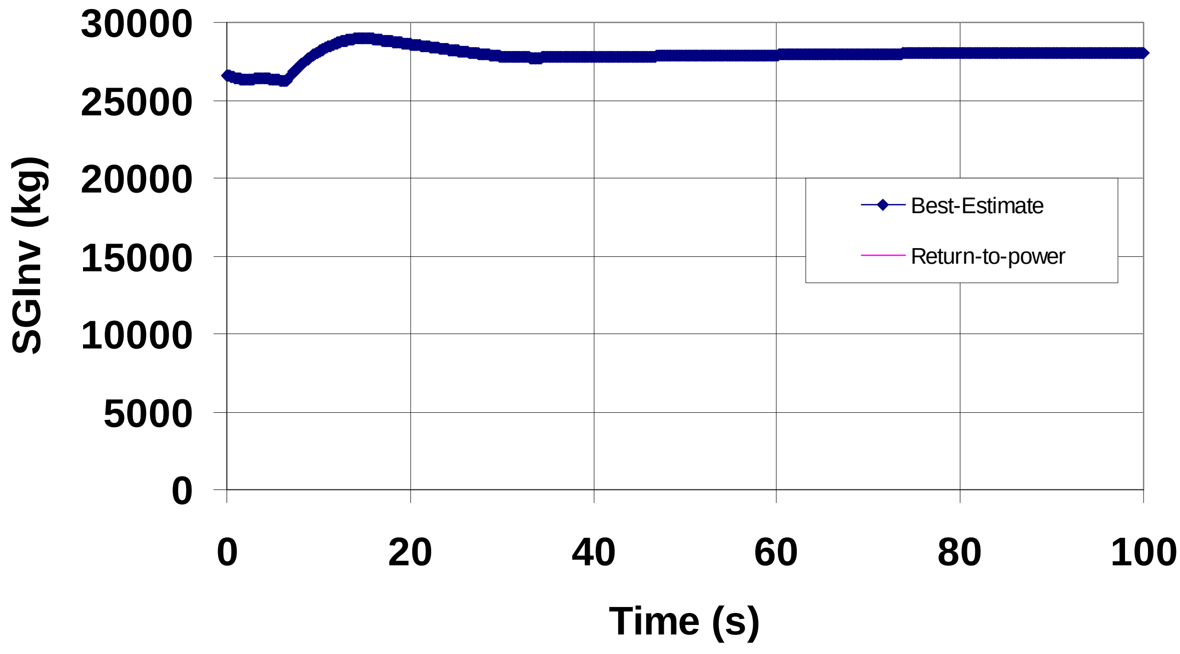


Figure 6-11 Fluid inventory of intact steam generator (BE and RP case)

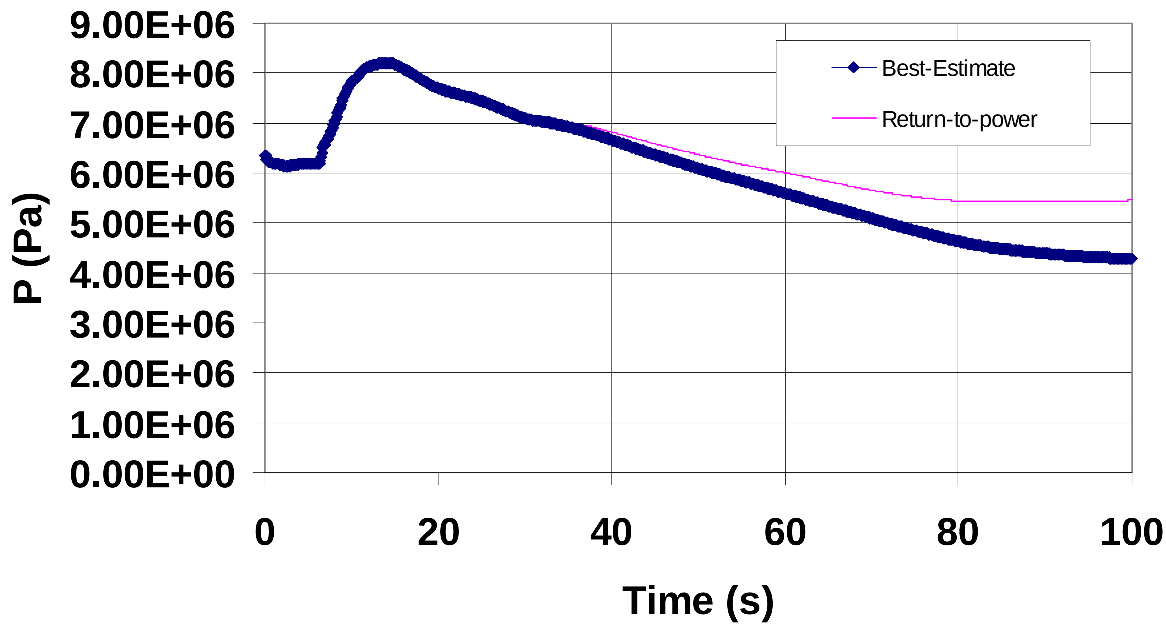


Figure 6-12 Pressure of intact SG-secondary side (BE and RP case)

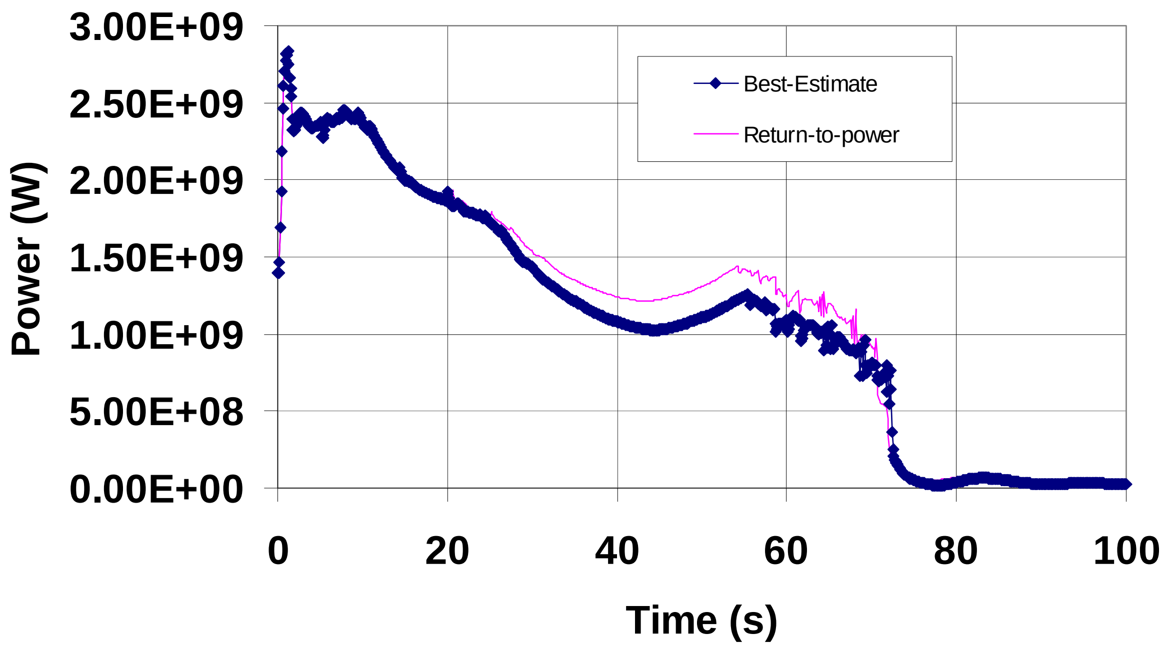


Figure 6-13 Transferred power over broken SG (BE and RP case)

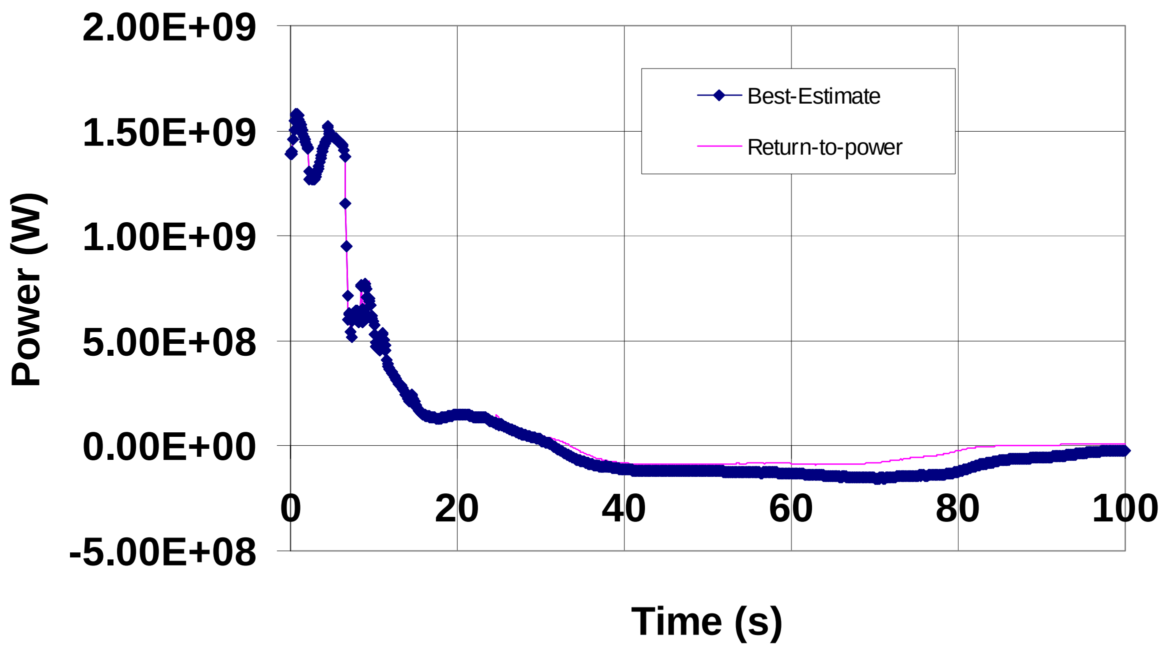


Figure 6-14 Transferred power over intact SG (BE and RP case)

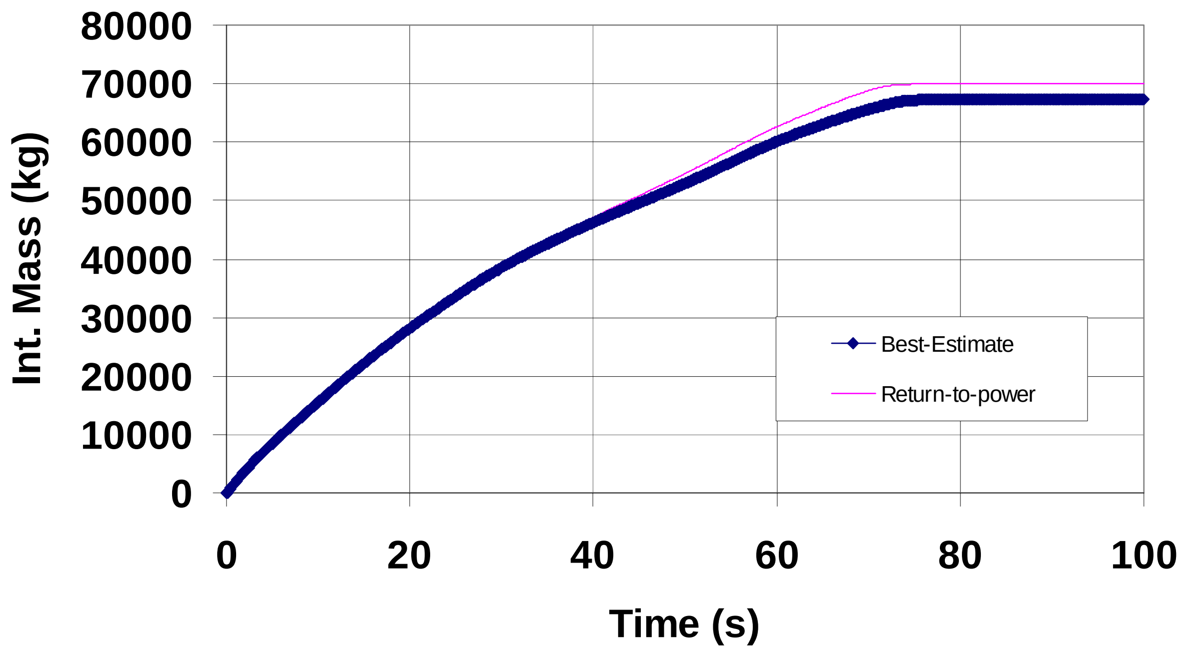


Figure 6-15 Integral liquid mass loss through both breaks (BE and RP case)

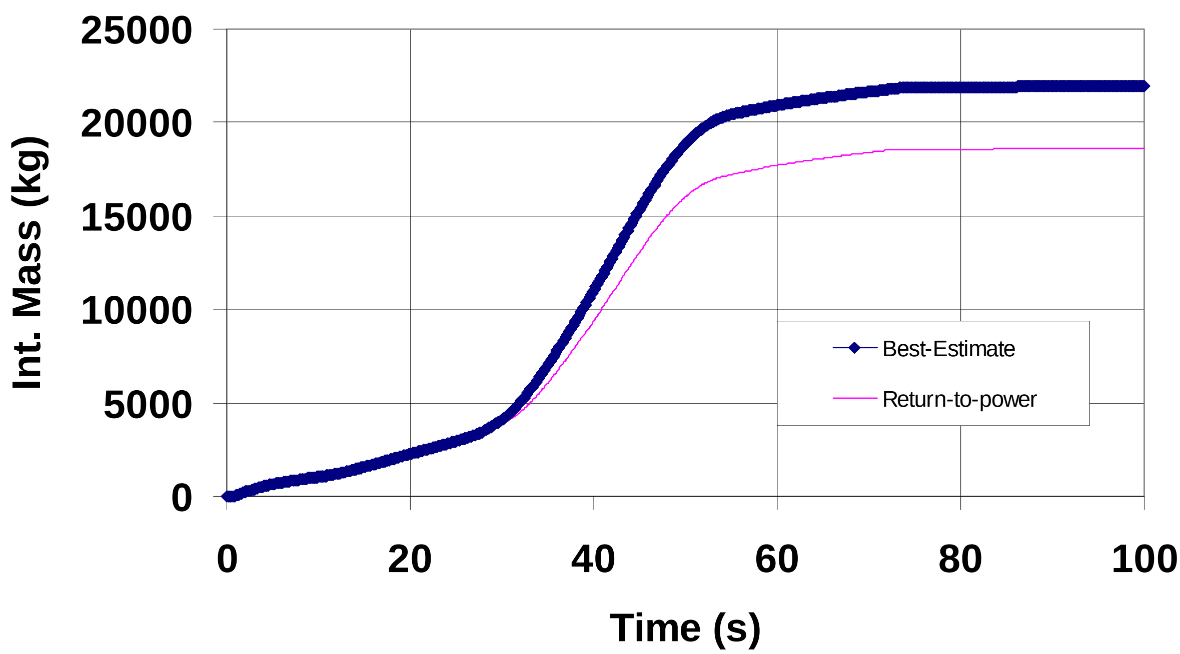


Figure 6-16 Integral steam mass loss through the breaks (BE and RP case)

6.2.2 Spatial distribution of selected parameters

RELAP5/PANBOX is able to predict the spatial behaviour of pertinent core parameters with detailed spatial resolution. At transient begin the coolant inlet and outlet temperature are uniformly distributed within the core. But this changes dramatically in the course of the MSLB-transient affecting the spatial power distribution. The radial coolant temperature distribution predicted by RELAP/PANBOX at the core height of 3.14 m is shown in Figure 6-17. At problem time 6.1 s (just after scram) the asymmetrical cooldown of the core is already noticeable. But at the time of 66 s the nonsymmetrical coolant temperature distribution is even worse (see Figure 6-17, lower part).

The corresponding radial power profile for the same time windows and at the same z-plane is exhibited in Figure 6-18. The relative radial power profile is quite uniform at 6.1 s. Later on, it undergoes a strong variation and at 66 s it is highly asymmetric with a maximum around the stuck rod (position N12 in Figure 3-5), located in the affected core half. On the contrary the power distribution of the intact core half shows a roughly uniform power profile with low values.

Similar spatial distribution of the coolant temperature and of the power were calculated for the return-to-power scenario, see Figure 6-19 and Figure 6-20. It confirms the very asymmetrical core behaviour, typical of the MSLB-transient. In contrast to the results obtained for the BE-scenario, the RP-results showed a more pronounced power distortion.

This power distortion is not only in radial but also in axial direction. In Figure 6-21 and Figure 6-22 predicted core averaged axial power profile calculated for both scenarios is plotted for different times in the course of the transient. In Figure 6-21 the power profile at three times i.e. just after scram (6.1 s), at highest power peak (66 s) and at transient end (100 s) is depicted for the BE-scenario. It can be seen that the axial power shape undergoes large changes in time. The highest relative value (1.43) is found at the upper core height for the time 66 s. The axial power shape evolution for the RP-scenario is different than that of the BE-scenario, see Figure 6-22. Its variation in time is very strong leading to very low values in the lower core part (0.4) and high values in the upper core height (1.91). These results are contradictory to the assumed constant axial power profile used for point kinetics in Exercise 1.

The results demonstrate the RELAP5/PANBOX-capability to predict the local hot spots during the course of complex transients taking into account a direct coupling between the thermal hydraulics and neutron kinetics via the cross-section update.

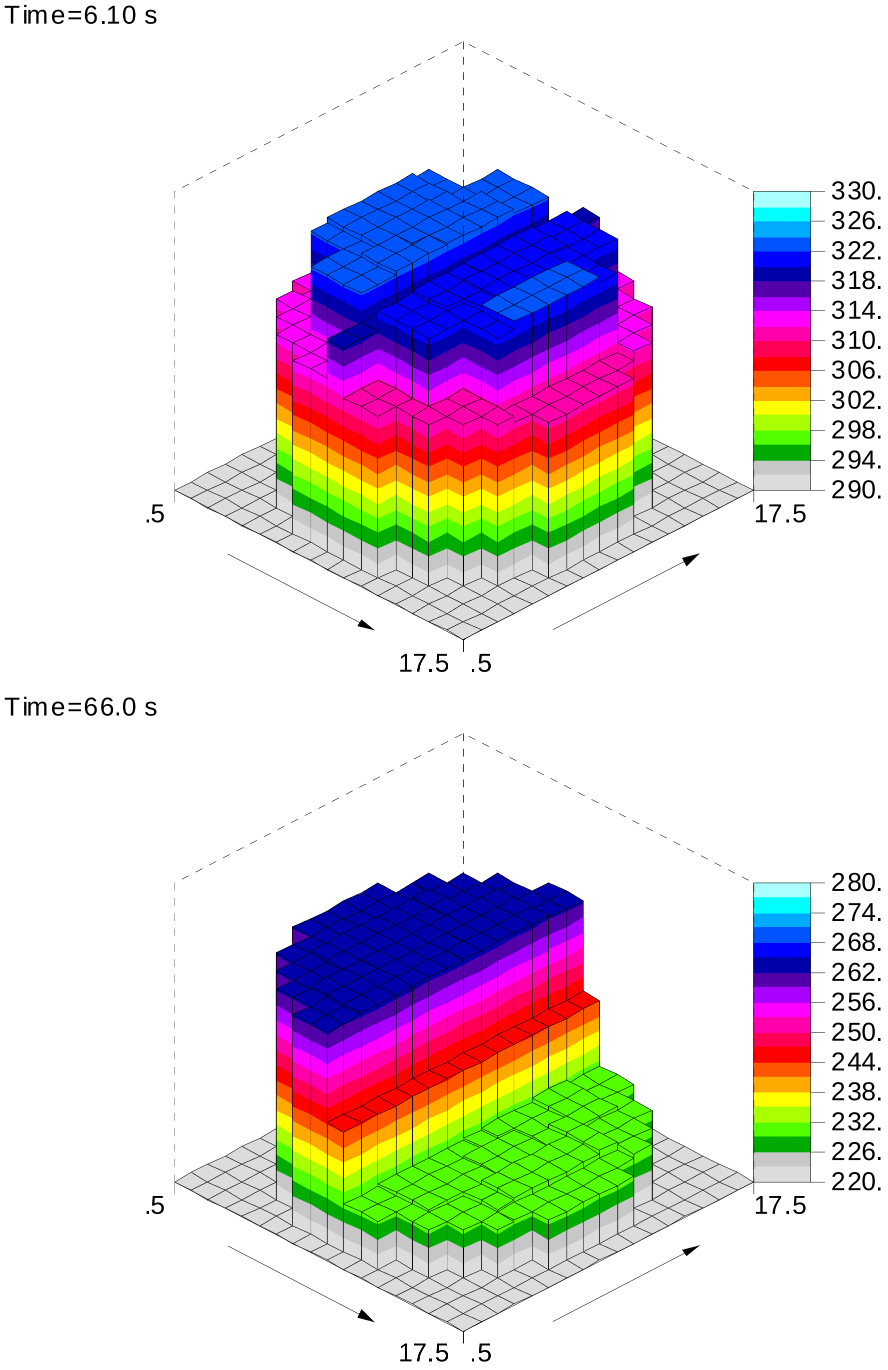


Figure 6-17 2D coolant temperature distribution at plane z=3.14 m for BE-case

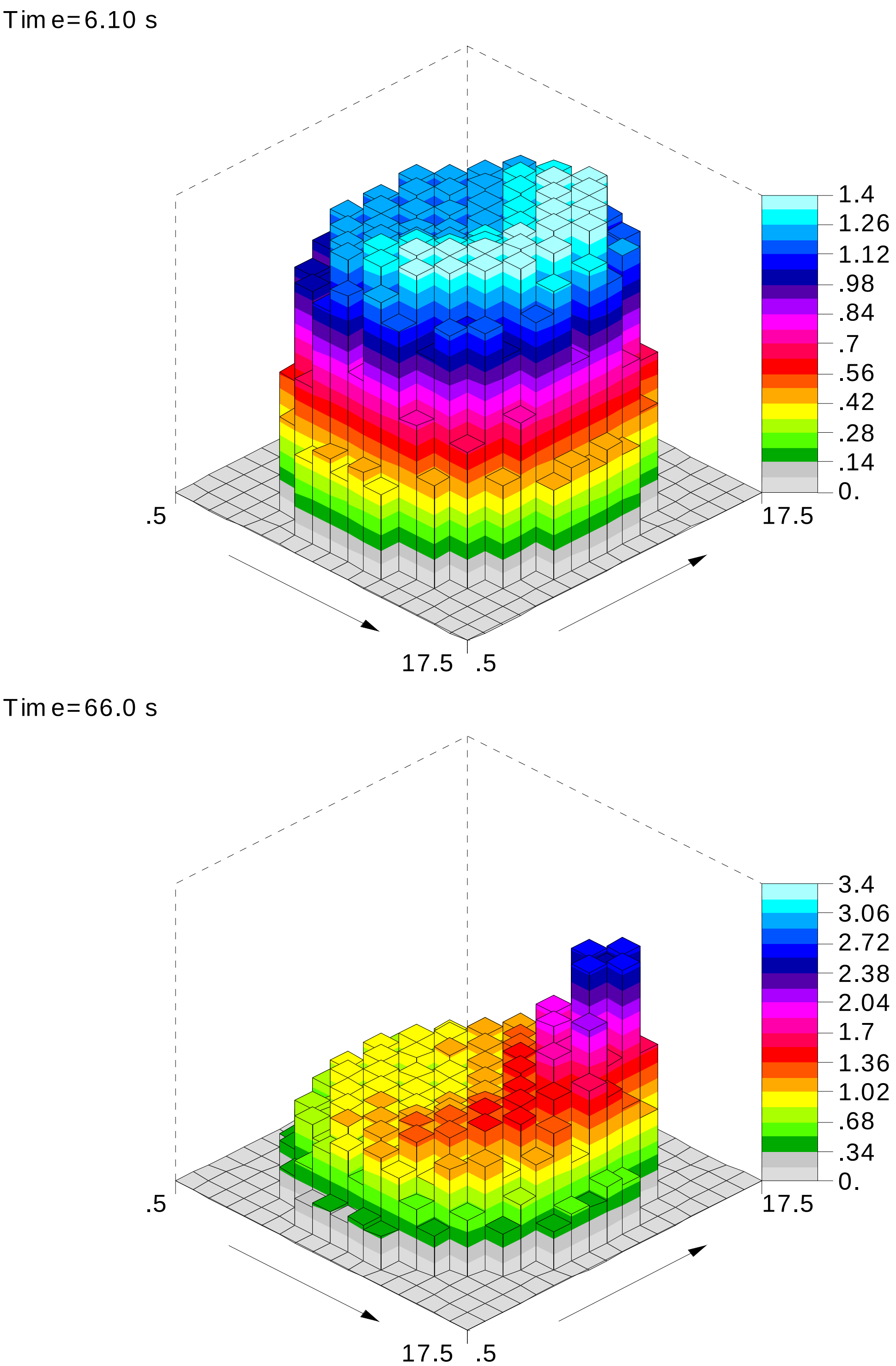
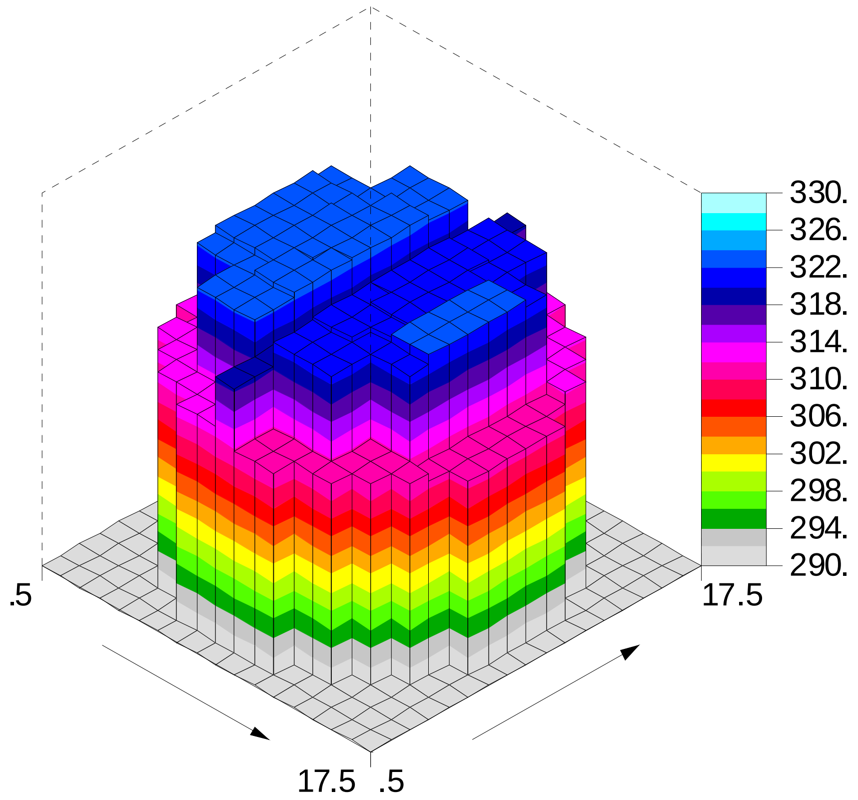


Figure 6-18 2D relative radial power distribution at plane z=3.14 m for BE-case



Time=66.0 s

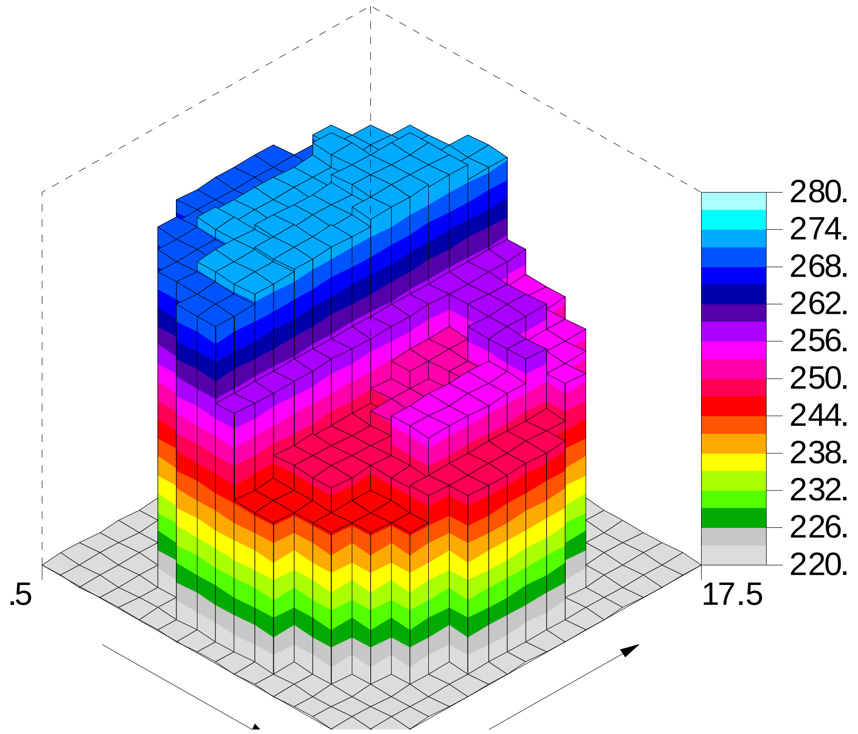
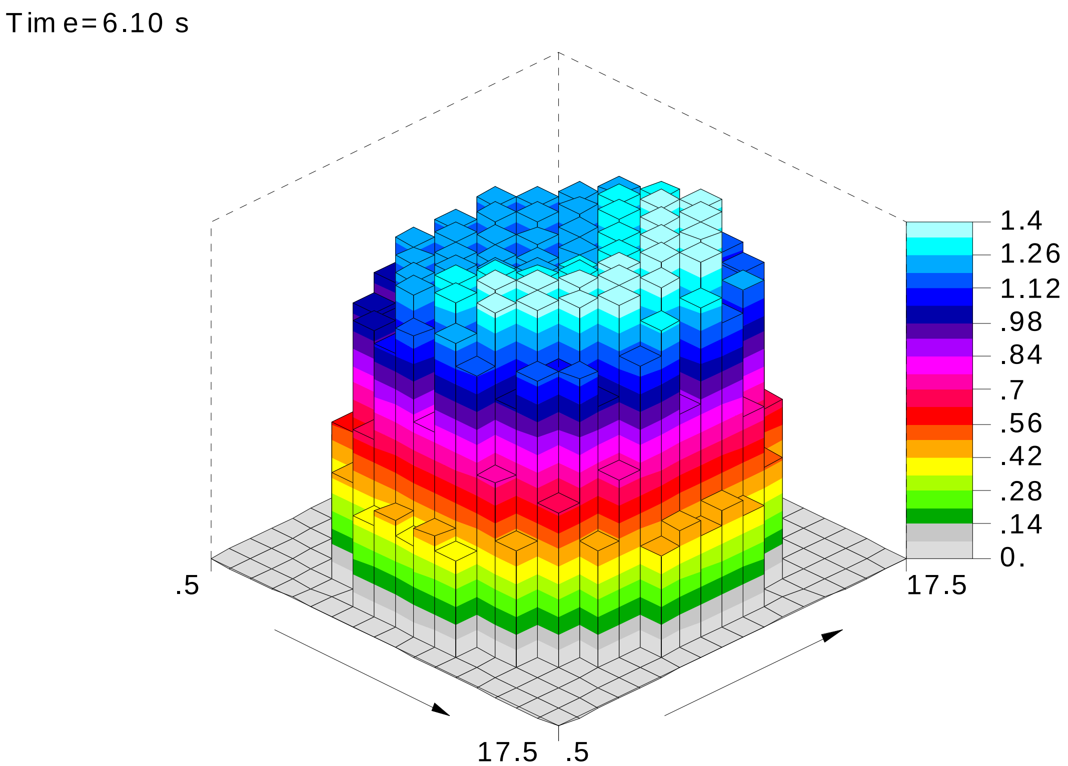


Figure 6-19 2D coolant temperature distribution at plane z=3.14 m (node=27) for RP-case



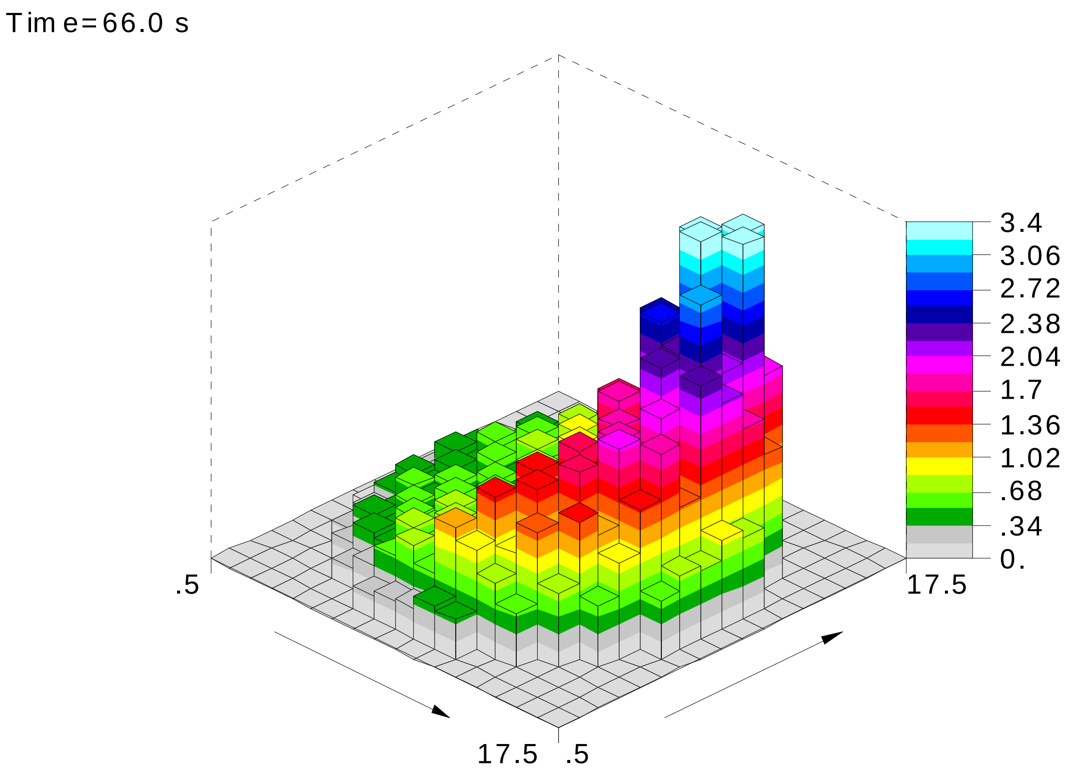


Figure 6-20 2D relative radial power distribution at plane z=3.14 m for RP-case

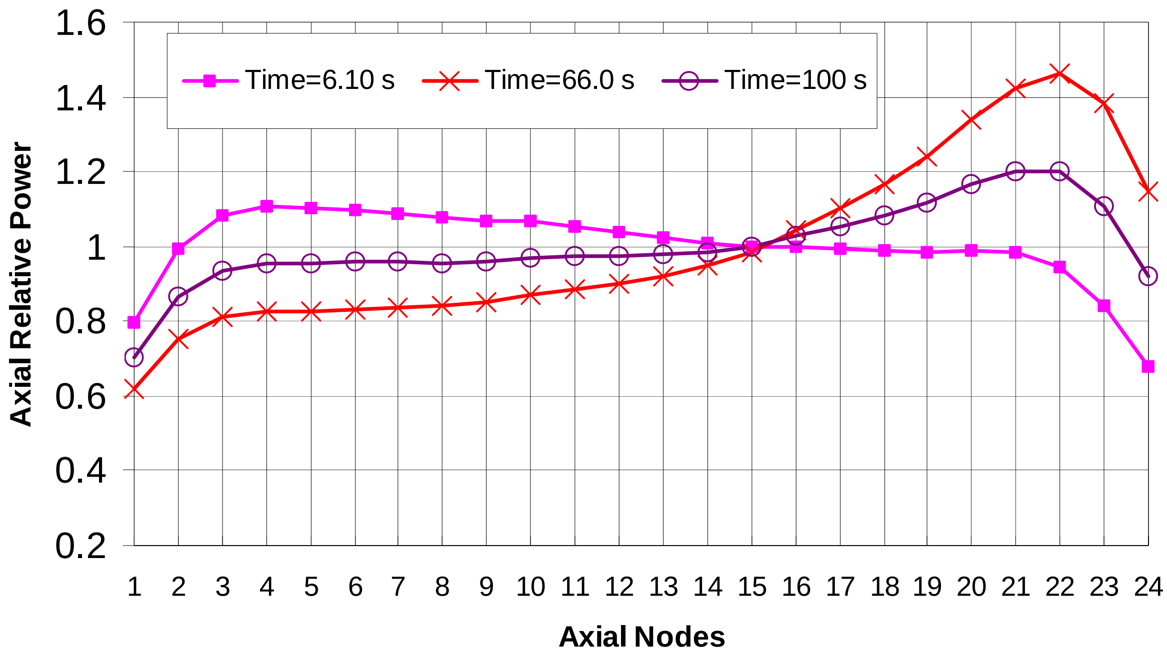


Figure 6-21 Predicted core average axial power profile for BE-scenario

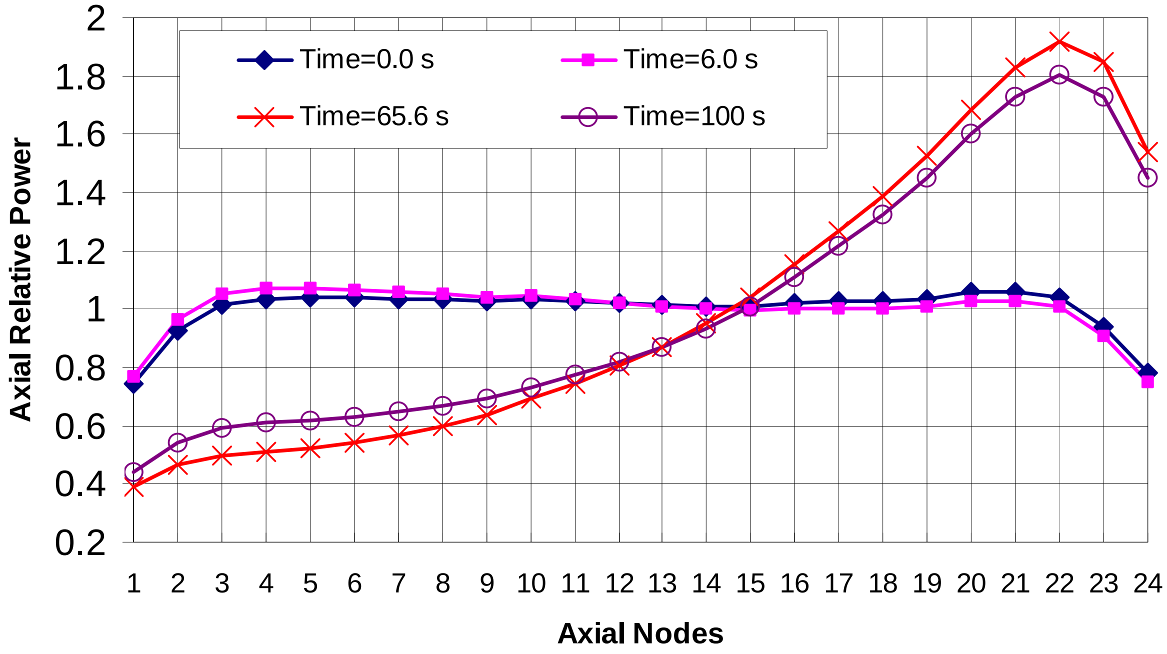


Figure 6-22 Predicted core average axial power profile for RP-scenario

6.2.3 Influence of coolant mixing within RPV

The influence of coolant mixing in the RPV on core reactor parameters was investigated for the BE-scenario with RELAP5/PANBOX. In addition to the coolant mixing prescribed in [Ivan99], the following cases were considered:

- No coolant mixing at all within the RPV (No Mixing),

- Full coolant mixing in the lower plenum (LP-Mixing).

The coolant temperature of the cold leg predicted for these cases is compared with the one predicted for the BE-scenario. In Figure 6-23 the broken cold leg temperature is shown. It can be seen that for the case “No Mixing” the largest temperature reduction is predicted. Consequently the intact coolant temperature predicted for the “No Mixing“ case undergoes the lowest reduction as shown in Figure 6-24. As a result of this thermal hydraulic behaviour the predicted trends of the reactivity and of the power differs from each other for the investigated cases. In Figure 6-25 the predicted total reactivity curves for these three cases are compared. The respective total power is shown in Figure 6-26. It can be concluded that the largest reactivity insertion and thus the highest power peak is predicted for the “No Mixing” case.

But this results obtained with RELAP5/PANBOX are contrary to those predicted by RELAP5 using a point kinetics approach [Sanc00a].

It has to be pointed out that RELAP5 predicted the highest power peak for the “LP Mixing” case even though the mixing assumptions used for RELAP5/PANBOX and RELAP5 were the same.

This results underlines the necessity for using multidimensional neutronic codes to simulate transients dominated by 3D-effects like the MSLB. On the other hand, it demonstrates that the use of point kinetics even with very conservative assumptions may fail to predict the maximal power peak.

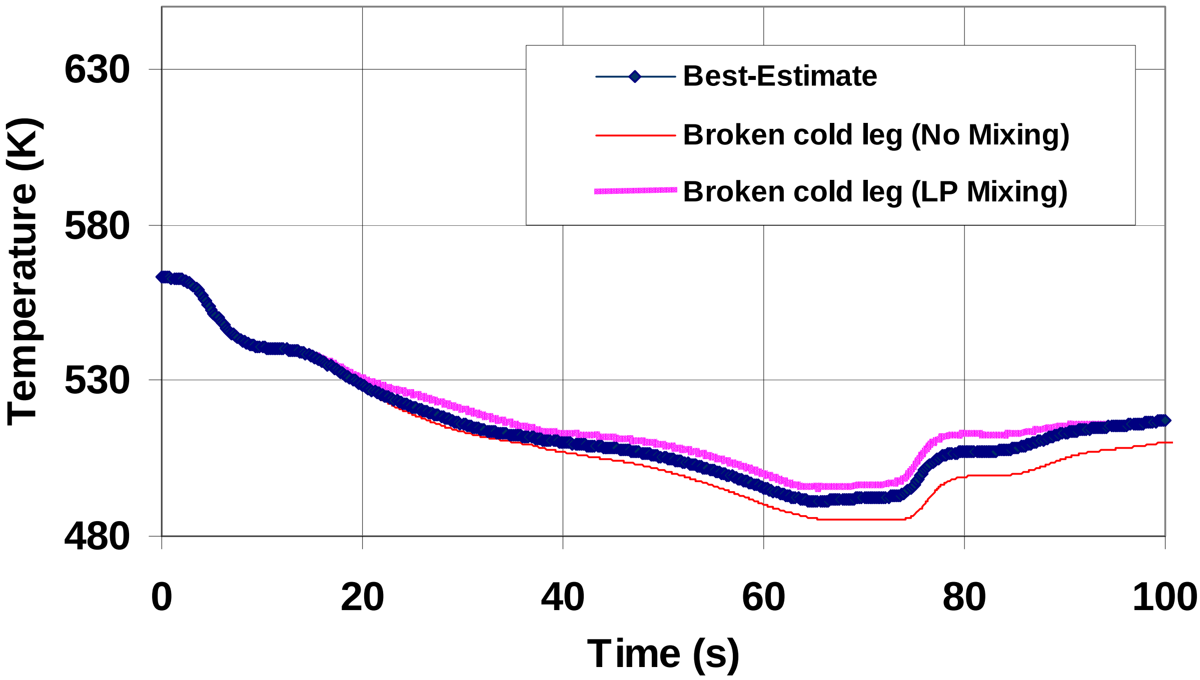


Figure 6-23 Coolant temperature of intact/broken cold legs

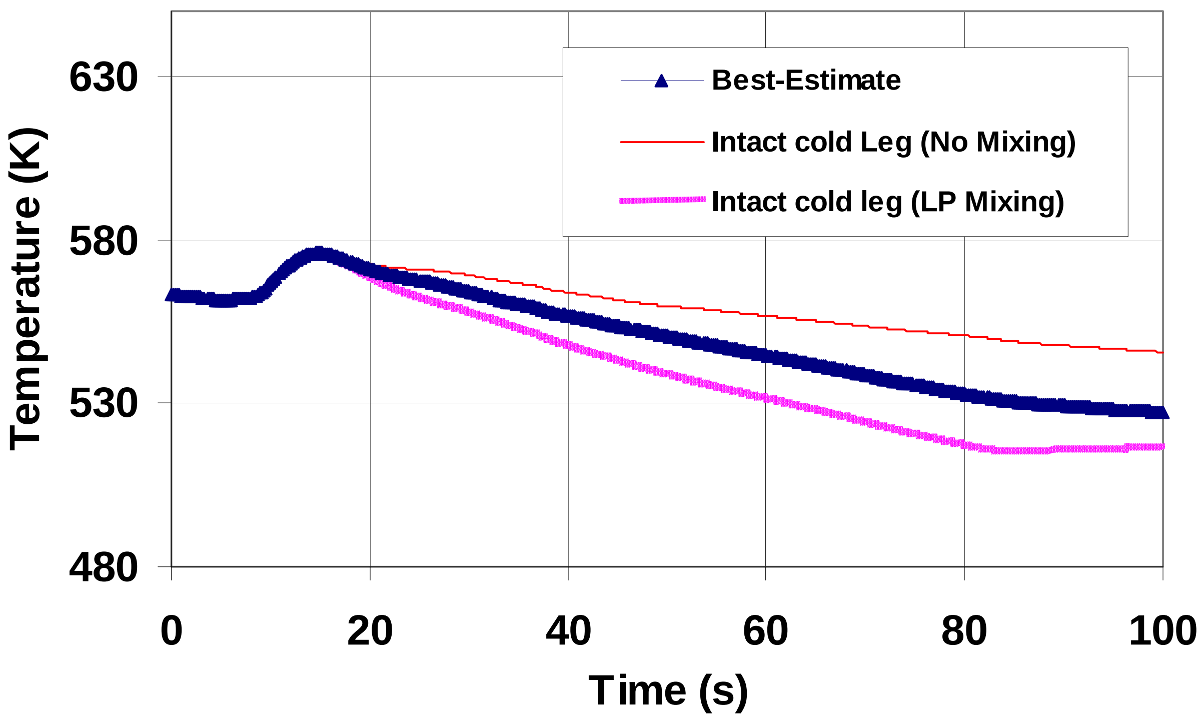


Figure 6-24 Coolant temperature of intact/broken cold legs

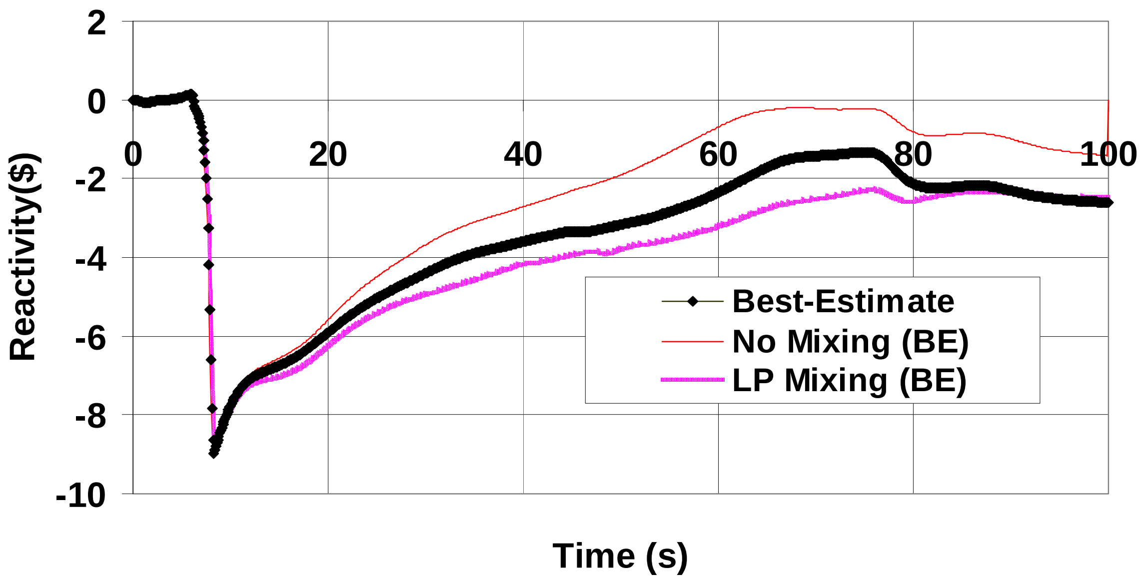


Figure 6-25 Total core reactivity predicted for different mixing conditions

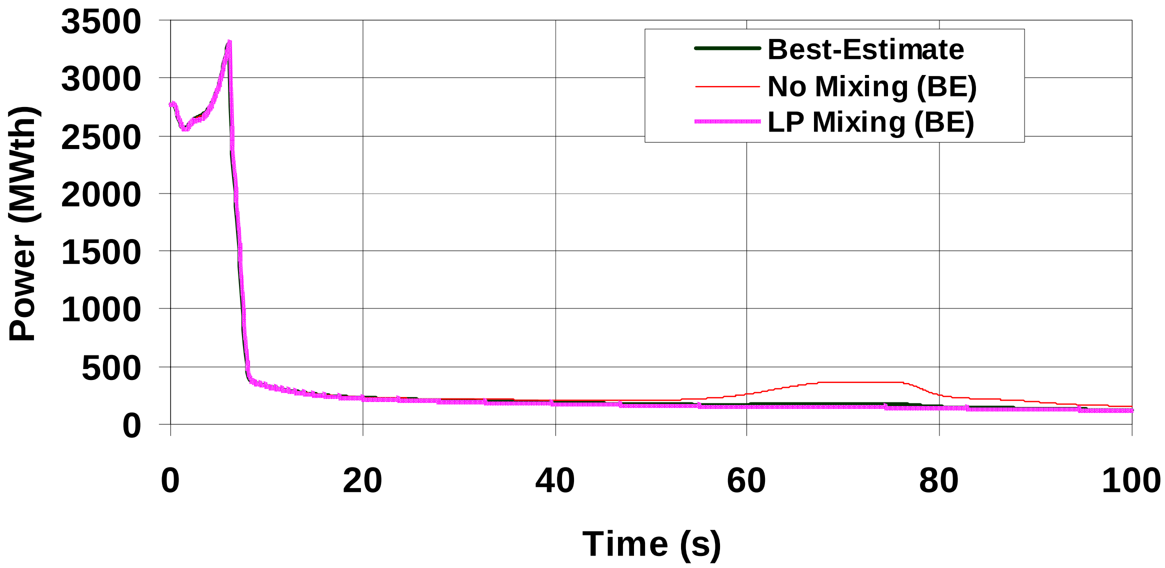


Figure 6-26 Total core power predicted for different mixing conditions

6.3 Comparison of Exercise 1 versus Exercise 3

A brief comparison of selected results obtained with the point kinetics approach (Exercise 1) and the spatial kinetics approach (Exercise 3) is presented. The BE-scenario of Exercise 3 is defined in the Benchmark specifications consistent with Exercise 1. Thus selected results of both scenarios will be compared to each other.

In Figure 6-27 the total reactivity predicted by RELAP5 and RELAP5/PANBOX is shown. It can be seen that both approaches calculated qualitatively the same of inserted positive reactivity. But quantitatively there is a constant difference of about 2 $ throughout the whole transient. A reason for this is the assumption of a fixed axial power profile for the whole transient in the point kinetics calculation. Moreover the assumed reactivity feedback coefficients for Doppler and moderator lead to an overestimation of the total reactivity as they are applied to the whole core only using an axial weighting. At the end of the transient RELAP5/PANBOX predicts a larger shutdown margin than the point kinetics model.

In Figure 6-28 the total power predicted with point and spatial kinetics are exhibited. At transient begin, the total power is the same for both calculations. Then the power of the point kinetics calculation increases while that of the spatial kinetics decreases first due to the different treatment of the feedbacks mechanisms. Around scram time both calculations predicts the same power up to about 20 s transient time. Later on the power predicted by the point kinetics starts to continuously increase faster with a larger gradient than the one of the spatial kinetics. The discrepancies become larger at the time when the cold leg coolant temperature reaches the minimum value. The point kinetics calculates a considerable power increase of about 20.3 % while REAP5/PANBOX predicts only a small power peak of 6.5 %.

This is a direct consequence of the amount of positive reactivity insertion predicted by both procedures, see Figure 6-27.

The main reason for overestimation of the power and reactivity trends by the point kinetics is the fact that the PK-calculation relies on both constant reactivity feedback coefficient for moderator and Doppler and with fixed axial power profile during the whole transient, which is far from reality, especially for transients like the MSLB.

In conclusion the RELAP5 stand-alone version predicts much higher thermal hot spots for the MSLB than the multidimensional best-estimate RELAP5/PANBOX based on the same integral plant model and assumptions.

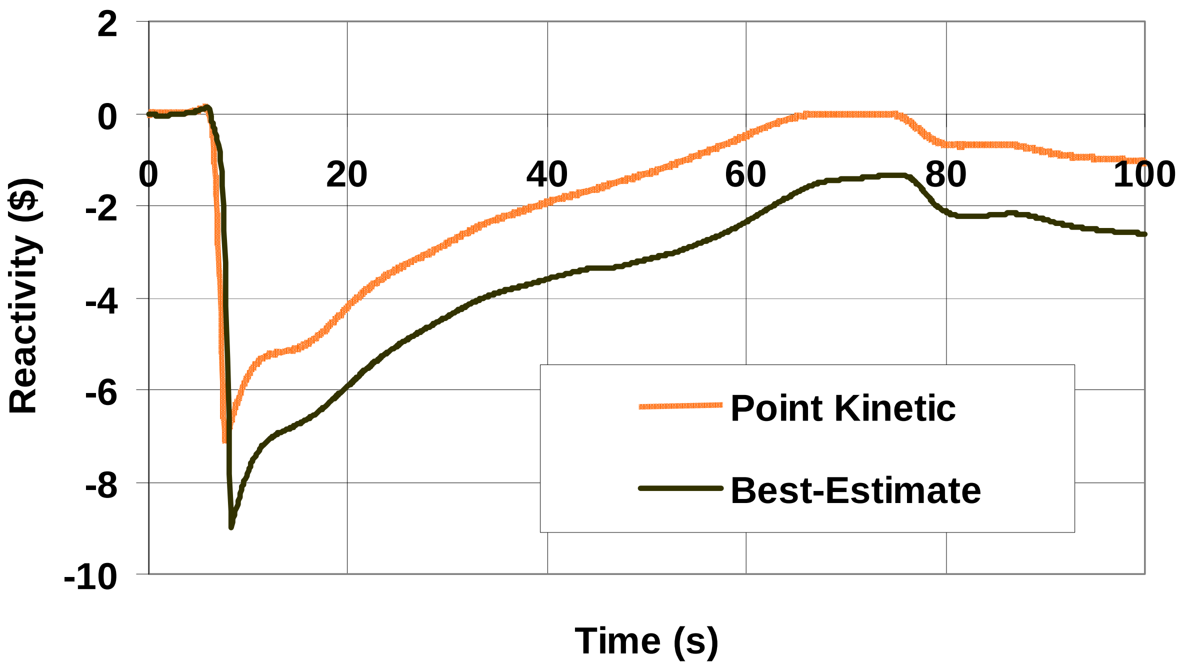


Figure 6-27 Comparison of total reactivity for Exercise 1 and 3

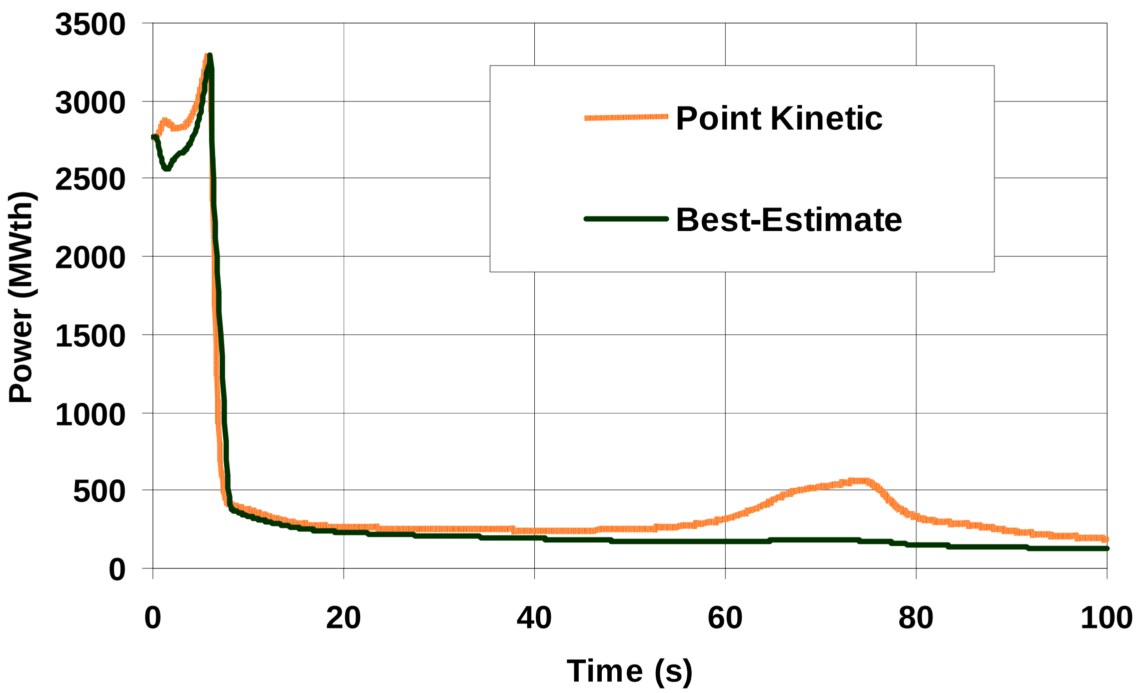


Figure 6-28 Comparison of total power for Exercise 1 and 3

6.4 Code-to-code comparison

The results obtained by Benchmark`s participants for Exercise 3 are being extensively evaluated by the PSU [Ivan00b]. Hence a coarse comparison of selected global parameters, i.e. core power and reactivity, predicted by different version of the RELAP5-code family coupled with different spatial kinetics codes like PANBOX, PARCS and MASTER, will be given here. In Figure 6-29 and Figure 6-30, the total power predicted by these codes for BE and RP scenarios are given. The agreement is fairly good almost for the whole course of the transient. But small discrepancies exist between predictions around the time of maximal power after scram. RELAP5/PARCS predicts the highest power peak, while the power calculated with RELAP5/PANBOX and MARS/MASTER are much closer to each other.

In addition, the total reactivity predicted by the mentioned codes for both scenarios is in excellent until 40 s in the transient, see Figure 6-31 and Figure 6-32. Afterwards the discrepancies increase a little but all predictions show qualitatively a similar trend. The reactivity insertion rate is mainly determined by the thermal hydraulic plant behaviour, especially the cooldown rate of the primary circuit and break outflow rate.

In Figure 6-33, the core-averaged axial power profile predicted by the mentioned codes is compared to each other at the end of the transient. Only small discrepancies exist, especially at the lower and upper part of the core.

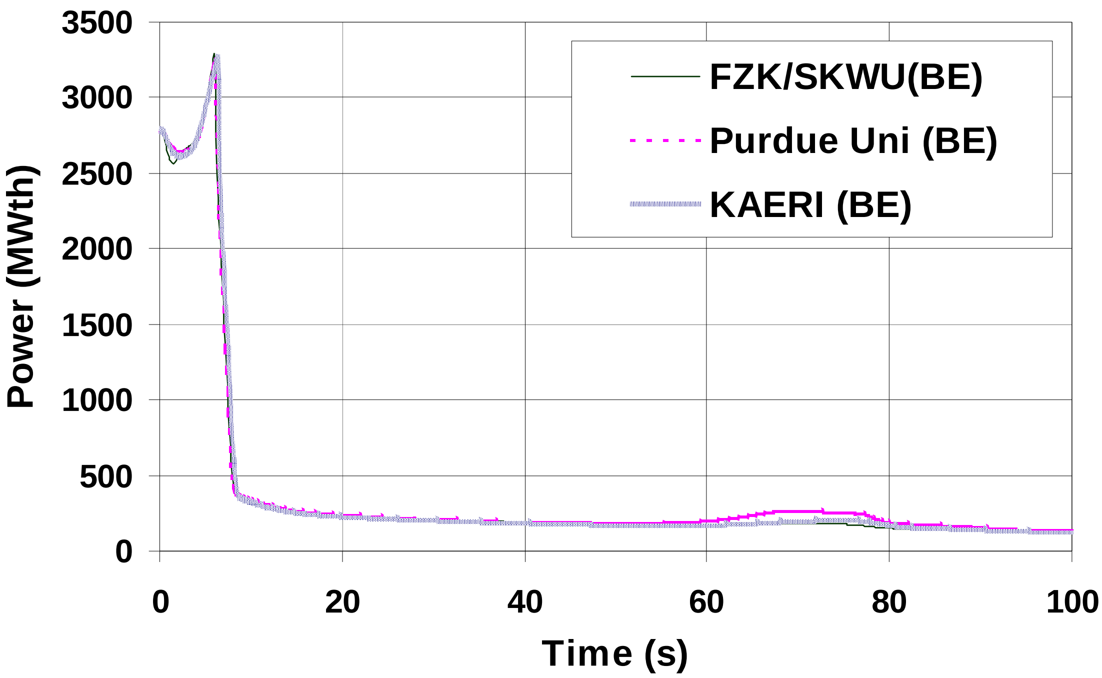


Figure 6-29 Total core power (Bst-Estimate case)

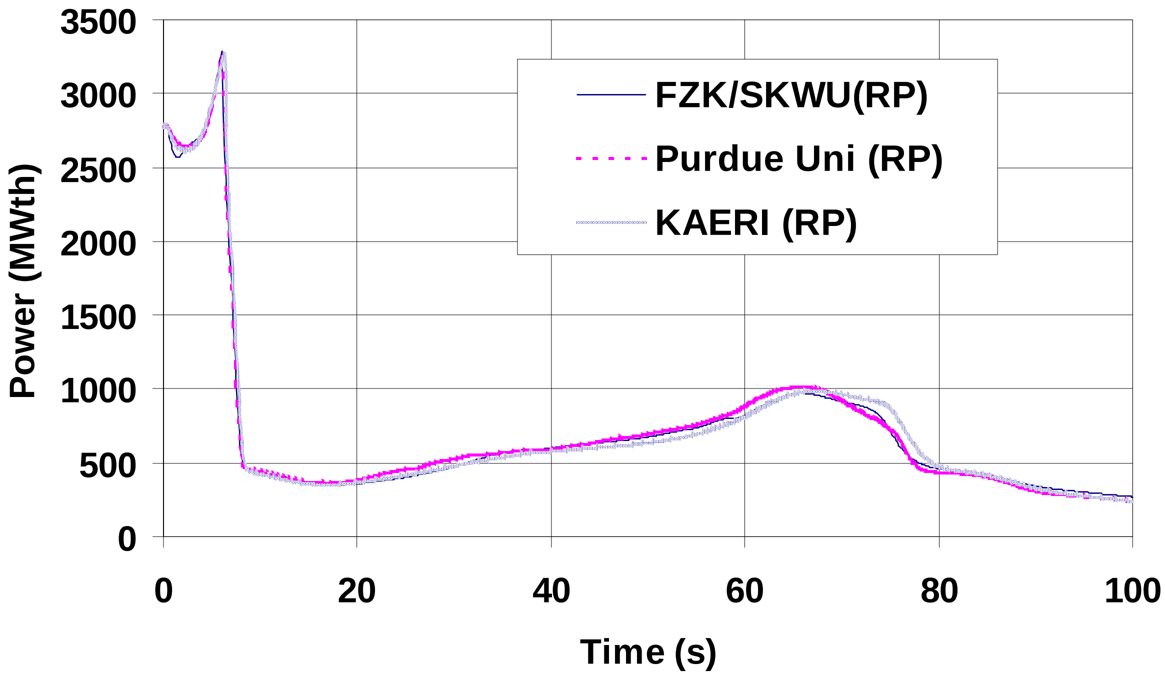


Figure 6-30 Total core power (Return-to-power case)

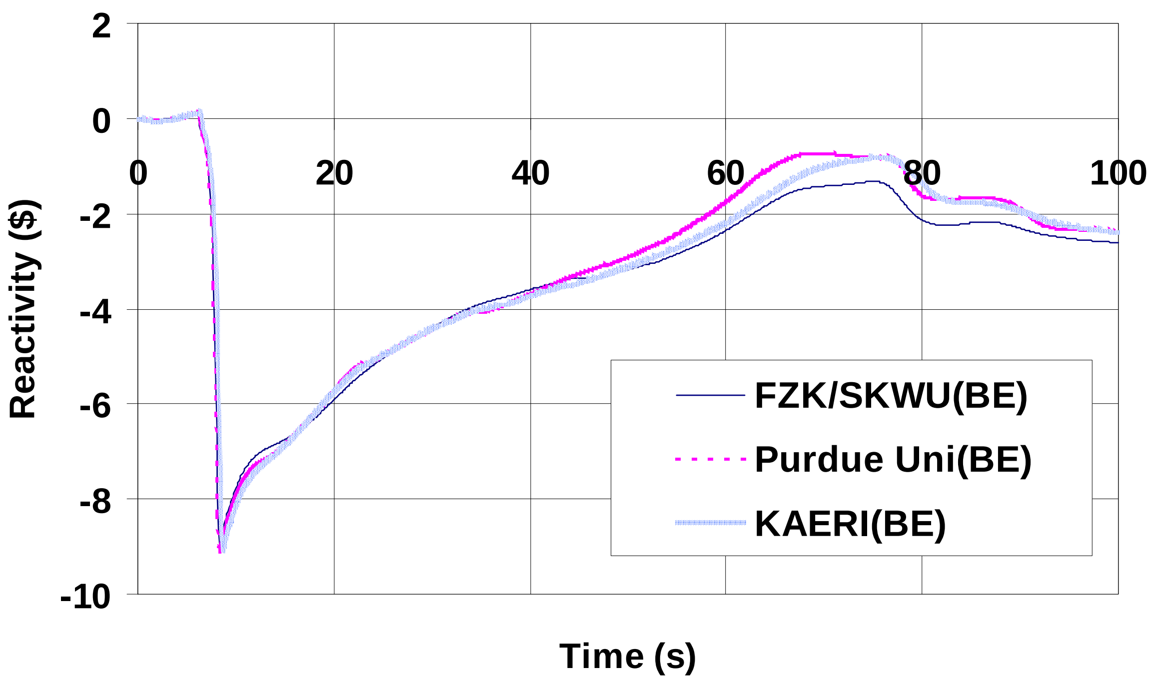


Figure 6-31 Total core reactivity (Best-Estimate case)

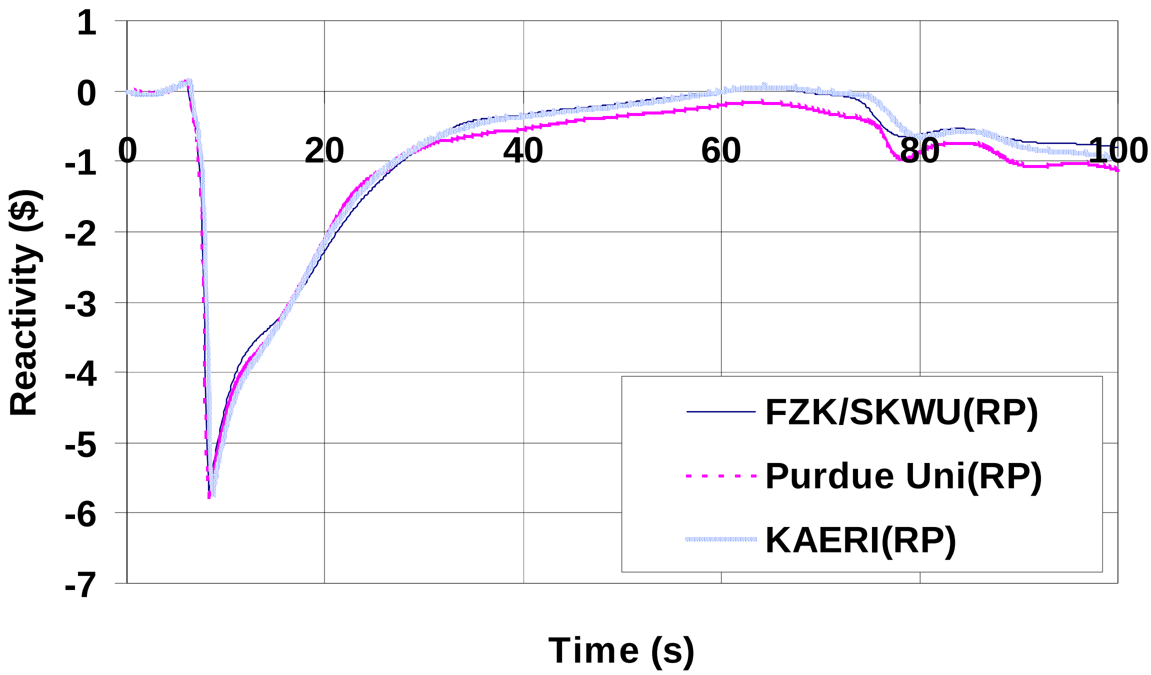


Figure 6-32 Code-to-code comparison: total core reactivity for RP-scenario

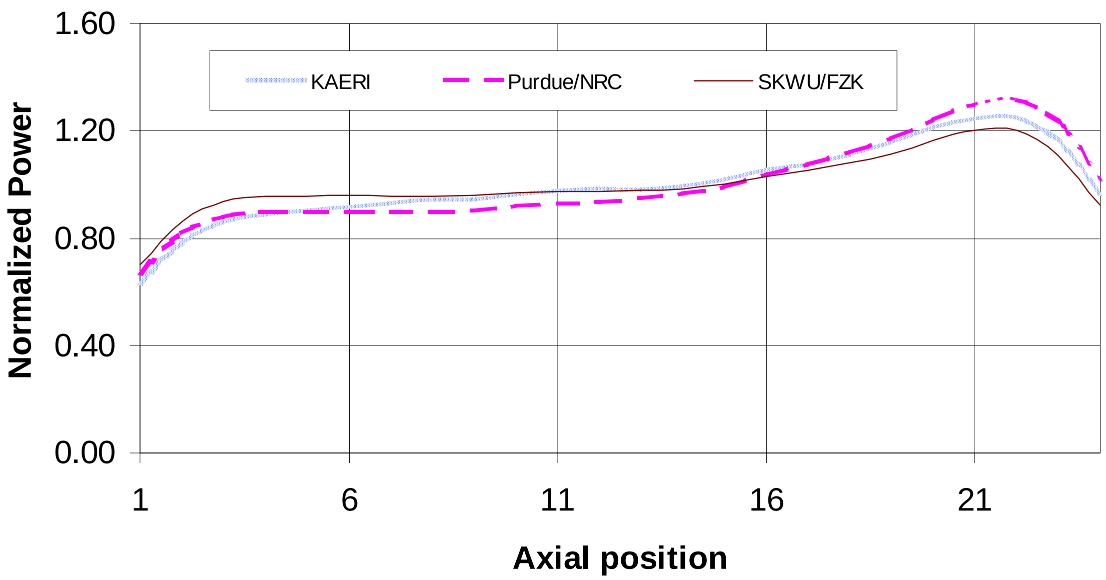


Figure 6-33 Code-to-code comparison: core average axial power distribution for BE

7 Summary and Conclusions

The MSLB-Benchmark Exercise 3 was analyzed with RELAP5/PANBOX based on an integral TMI-1 plant model with a multidimensional core representation developed for Exercise 1 and 2 respectively.

Two scenarios, the best-estimate (BE) and the return-to-power (RP), were successfully predicted. A considerable power increase of 35 % and re-criticality for about 7 s were predicted for the RP-scenario, while no re-criticality and solely a very small power increase (6,5 % of nominal power) were calculated for the BE-scenario. The predicted radial and axial core power distributions showed large distortions and asymmetrical shape, especially for the RP-scenario. These results demonstrate the capability of coupled code systems with space-time kinetics to accurately predict local hot spots with detailed spatial resolution and thereby to evaluate safety thermal margins.

A comparison of the results obtained for the MSLB-transient with point kinetics and spatial kinetics have clearly shown that the point kinetics overestimates the power increase and the reactivity insertion during the MSLB-progression. It predicts a maximal power increase of about 20 % compared to the 6.5 % predicted by the space-time kinetics (BE-scenario).

Moreover the sensitivity study performed to assess the influence of the coolant mixing on global parameters (reactivity and thermal power) indicates that the highest power increase is predicted for the “No Mixing” case. This is fully contrary to the results obtained with the point kinetics, where the highest power peak was calculated for the “LP Mixing” case. This comparison highlights the shortcomings and limitations of the point kinetics approach to simulate complex transient like the MSLB.

The code-to-code comparison revealed that the RELAP5/PANBOX-results are in good agreement with most of the trends predicted by similar codes.

In general it can be concluded that the results obtained by RELAP5/PANBOX for Exercise 3 emphasize the benefits of using space-time neutron kinetics instead of the point kinetics approaches.

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Appendix A: List of coupled code systems

Tab. 8-1 List of participants in the third Exercise of the PWR MSLB Benchmark

Number	Company Name	Country	Code
1	VTT	Finland	TRAC-3D/SMABRE
2	GRS	Germany	ATHLET-QUABOX/CUBBOX
3	FZR	Germany	ATHLET-DYN3D/R
4	GPUN/CSA/EPRI	USA	RETRAN-3D/MOD2
5a 5b	University of Pisa/Zagreb (UP1) University of Pisa/Zagreb (UP2)	Italy / Croatia	RELAP5-QUABOX (UP1) RELAP5/PARCS(UP2)
6	British Energy/Tractabel	UK/Belgium	RELAP5/PANTHER
7a 7b	IPSN/CEA (Set 1) IPSN/CEA (Set 2)	France	CATHARE/CRONOS2/FLICA4 CATHARE/CRONOS2/FLICA4
8	FZK/SKWU	Germany	RELAP5/PANBOX
9	Universitat Politecnica de Catalunya (UPC)	Spain	RELAP5/PARCS
10	Universidad Politecnica de Valencia (UPV)	Spain	TRAC-PF1/NEM
11	Universidad Politecnica de Madrid (UPM)	Spain	RELAP/SIMTRAN
12	JAERI	Japan	THYDE-NEU
13	Purdue/NRC	USA	RELAP5/PARCS
14	KAERI	Korea	MARS/MASTER
15	PSU	USA	TRAC –PF1/NEM