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Risø-M-2209

THE THREE-DIMENSIONAL PWR TRANSIENT CODE ANTI; ROD EJECTION TEST CALCULATION

A.M. Hvidtfeldt Larsen

Abstract. ANTI is a computer program being developed for three-dimensional coupled neutronics and thermal-hydraulics description of a PWR core under transient conditions. In this report a test example calculated by the program is described. The test example is a simulation of a control rod ejection from a very small reactor core (to save computing time). In order to show the influence of cross flow between adjacent fuel elements the same calculation was performed both with the cross flow option and with closed hydraulic channels.

INIS descriptors. A CODES, CONTROL ELEMENTS, HYDRAULICS, PWR TYPE REACTORS, REACTOR KINETICS, ROD DROP ACCIDENTS, THREE-DIMENSIONAL CALCULATIONS, TRANSIENTS.

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1. INTRODUCTION

The ANTI computer program is being developed for three-dimensional coupled neutronics/thermal-hydraulics transient calculations for the PWR core. It combines the three-dimensional nodal theory neutron kinetics part of the BWR program ANDYCAP [1-5] with the transient subchannel hydraulics program TINA [6-11]. The program is intended mainly for transients where the spatial distribution of power and coolant flow in the core is important, especially in case of a local power increase. It is hoped that ANTI, when completed, will be a useful tool for the analysis of transients ranging from normal and abnormal operational transients to postulated accident conditions.

This report describes a test calculation, simulating a control rod ejection accident. In order to keep the computer costs at a reasonable level the testcase was made as a small reactor core from which 1/8 is represented by means of 112 nodes in the neutronics part and 4 subchannels in the hydraulics. A control rod is removed from the central fuel element at a constant speed which takes the rod from a fully inserted position at time zero to fully out at 0.3 s. The example is intended as nothing more than an initial test of the program, and the results of the calculation should not be expected to give information about the severity of a control rod ejection accident, since the neutron cross sections and the hydraulics input data were chosen more or less by change. The calculation is therefore not representative of any existing reactor.

To give an impression of the influence of the cross flow on this particular transient, the testcase was repeated without cross flow between the subchannels. The steady state calculation and the power history during the transient are nearly the same as in the cross flow case, but the maximum void content of the hot channel goes up by a factor of two when cross flow is not allowed.

2. BRIEF DESCRIPTION OF THE ANTI PROGRAM

As mentioned in the introduction the ANTI program is a combination of the neutron physics part of the BWR dynamics program ANDYCAP [1-5] and the transient subchannel hydraulics program TINA [6-11]. The present status of ANTI is that the program is under development; a preliminary version is running, but needs testing and quite a few modifications before the program development is completed. A program description has not yet been issued, and therefore the main features of the program will be briefly described in the following chapter.

The ANTI program is dealing only with the reactor core, unlike ANDYCAP where the upper and lower plena and the downcomer are included. Therefore boundary conditions at core in- and outlet must be supplied. These two-dimensional time-dependent boundary conditions are calculated (or estimated) by means of dynamic plant models, or for some applications a qualified guess of for example the flow and temperature distributions at the inlet may be sufficient.

The steady state for a given reactor core is calculated first, and the results are written on a disk file. The steady state can then be used as the starting point for a number of different transient calculations.

2.1. Neutron kinetics

The neutron physics part of ANTI was taken over from the ANDYCAP program described in refs. 1-5. The power distribution in the core is calculated by means of three-dimensional nodal theory, where the average one-group neutron flux is calculated in box-shaped coarse-mesh regions (nodes). The node division is typically obtained by subdividing a fuel element axially, giving node dimensions of about 20 x 20 x 20 cm^3. The neutron diffusion between the nodes is accounted for by the so-called coupling coefficients which may be adjusted to the problem in question by means of four input dials, or g-factors.

The boundary to the reflector is treated by constant albedo values given as input. A maximum of 1680 nodes may be used in the present program version. The size of a problem may be reduced by utilizing one of a number of symmetry options.

Two-group nuclear constants are provided as input in the form of functions of fuel temperature, moderator density and moderator temperature. Before the flux calculation, the cross sections are collapsed to one energy group, assuming that the ratio for each node between fast and thermal flux can be expressed as

$$\frac{\phi^1}{\phi^2} = \frac{\Sigma_a^2}{\Sigma_s^{12}} \qquad (1)$$

where ϕ^1 and ϕ^2 are the group fluxes for the fast and thermal neutron group, respectively, Σ_a^2 is the macroscopic thermal absorption cross section and Σ_s^{12} is the macroscopic down-scatter cross section from the fast to the thermal group.

For the kinetics calculation it is possible to have a maximum of six delayed neutron groups. The integration is carried out by the first order backward formula

$$y_{s+1} = y_s + h\dot{y}_{s+1} \qquad (2)$$

where y denotes a time dependent function, s is a time step number and h is the time step length. For the solution a predictor-corrector method is employed; first the flux and power distributions at the next time step are predicted by linear extrapolation, then (2) is solved, and the results are compared to the predicted values. If the difference exceeds a specified limit, the time step is reduced by an input factor between 1 and 2 and the process repeated with the shorter time step. After a number of "good" steps, the time step is increased by the same factor.

2.2. Hydraulics and fuel rod conduction model

The hydraulics part of ANTI is an almost unchanged version of the program TINA [6-11]. It was developed primarily for PWR

blowdown calculations and has been used for simulation of the Semiscale tests [11] and for investigation of three-dimensional effects during the blowdown phase of a hypothetical loss-of-coolant accident in the three-looped Westinghouse reactor (Ringhals 3) [12].

TINA solves the two phase drift flux model equations for a number of parallel subchannels with the possibility of cross flow and turbulent mixing between the channels. For the ANTI calculations a subchannel in principle is defined as the water volume of one fuel element; however, since the number of subchannels is limited for practical reasons, several fuel elements will normally have to be put together in one subchannel, giving quite large flow areas per subchannel.

The present ANTI version has the following upper limits

number of subchannels: 10
number of boundaries between subchannels: 15
number of axial mesh-points: 51.

The transient solution is implicit and therefore allows large time steps, at least for slowly developing transients.

The fuel rod model is described in ref. 8. The transient heat conduction problem is solved in one-dimensional cylindrical geometry by an implicit numerical method, stable for large time steps. The power produced in the fuel is distributed evenly between a number of concentric rings, all having the same cross section area. In the present version the maximum possible number of rings in the fuel itself is 8; adding one for the cladding and one for the fuel-cladding gap gives the total maximum of fuel rod meshes of 10.

All fuel rods in the reactor are assumed to be identical with respect to geometry and material properties. All the fuel rods belonging to one subchannel are lumped into one representative rod which has the average power generation for the channel in question.

The correlations for heat transfer from fuel rod to coolant used in the present calculations are those given in ref. 11. In more recent versions of TINA these correlations have been changed.

2.3. Combination of neutronics and thermal-hydraulics

The two basic program units, the neutronics part and the thermal-hydraulics part, have been kept separate as far as possible and are mixed only in the main program and a few administrative routines. A data conversion routine, called SHUFL, was programmed to take care of the data transfer from neutronics to hydraulics and vice versa.

In the neutron physics part the power distribution is calculated in the form of the power produced in each node. What is needed for the hydraulics calculation is the power distribution given in the hydraulic meshpoints. To obtain this, first the powers at each level from all the fuel elements belonging to one hydraulic channel are added to give the nodal power distribution per channel. Then, assuming that the nodal power distribution is a pointwise distribution with the power values given in the node midpoints, the power distribution at the hydraulic meshpoints is obtained by linear interpolation. Finally, the power distribution for each channel is normalized to make sure that the total channel power is correct.

Most of the fission power is released as heat in the fuel itself. A small part is released in the coolant and the structural materials of the core. Therefore, the power calculated by the neutron physics must be split up. The larger part goes into the fuel rod model as the heat source, while a smaller part, about 3 per cent, is added to the heat which is transferred from the fuel rod surface to the coolant. This direct power term is important for some transients, because it speeds up the feedback on reactivity from the moderator/coolant.

In the hydraulics the distributions of moderator density ρ_m and temperature T_m are calculated. The fuel rod model contributes the

distribution of the fuel rod average temperature T_f. The three variables T_f, ρ_m and T_m are in the routine SHUFL converted into node average values for use as parameters in the expressions for the calculation of the neutron cross sections. In this way the results of the thermal-hydraulics calculation influence the flux and power calculation of the neutronics.

Fig. 2.1 shows the flow of information during steady state iterations. An initial power distribution may be given as input, otherwise a uniform distribution is assumed. With the initial power distribution at the specified total reactor power a steady state hydraulics calculation is performed (TINA) followed by the fuel rod calculation (FUEL). Then follows a steady state neutronics calculation (POLKA) giving the new power distribution for use in TINA, and so on. Iterations are stopped when the differences between the k_{eff}'s and power distributions of two successive neutronics calculations become lower than a limit given as convergence criterion.

The k_{eff} calculated for the steady state is normally somewhat different from one. Therefore, to start from a stable condition the neutron production terms in the cross sections are divided by the calculated k_{eff} before the dynamic calculation is initiated.

In dynamic calculations the data transfer between neutronics and hydraulics is more or less the same as for stady state. At each time step the predicted power is used for the hydraulics and fuel rod calculations, and the results herof go back for the calculation of corrected flux and power values. If predicted and corrected values are sufficiently close new powers are predicted for the next time step, hydraulic calculations performed, and the process continues as described in section 2.1.

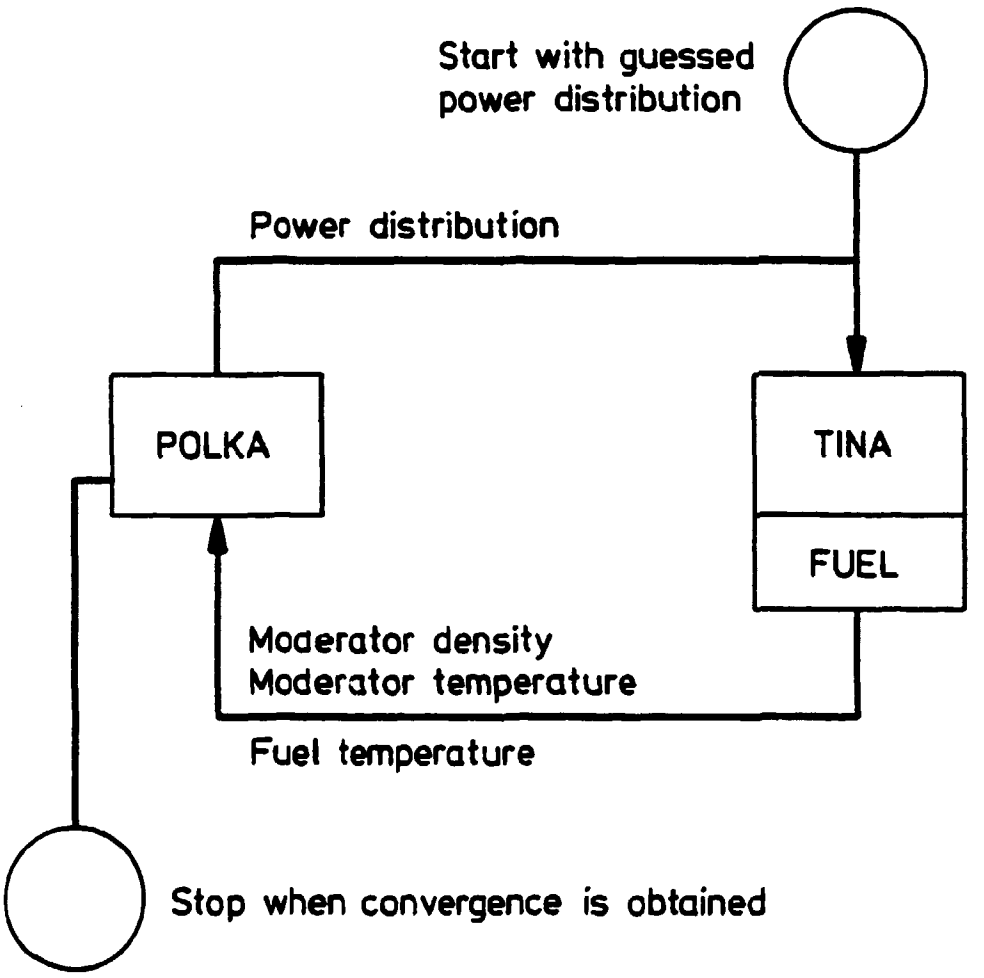


Fig. 2.1. Steady state iteration.

3. DESCRIPTION OF THE TESTCASE

The testcase is a control rod ejection calculation. In order to save computer time only a small reactor core is described. The geometry of the small reactor and the input data for the neutron physics part of the program were taken over almost unchanged from the small reactor calculations of ref. 13, whereas the hydraulic input should simulate typical PWR conditions. The calculation is not meant to be realistic, only to serve as a first test of the program performance.

To show the influence of the cross flow between the hydraulic channels the same calculation was repeated with closed channels, i.e. without boundaries between the channels, corresponding to the BWR situation with the channel boxes separating the fuel elements.

3.1. Geometry

The small reactor core used for the testcase is shown in Fig. 3.1. It consists of 37 fuel elements, initially with five control rods fully inserted. The fuel element width is 21.4 cm, and the core height is 365 cm. To further reduce the computer time, symmetry is assumed to allow the core to be represented by only 1/8 as shown in Fig. 3.2. This makes the core model consist of only eight fuel elements, two of them containing control rods. The rod which is removed to initiate the control rod ejection transient is the one inserted in the central fuel element.

The reactor core is surrounded by reflecting water, represented in the calculation by albedo values.

The fuel element is a 15 x 15 array of rods from which 204 are fuel rods with an outer diameter of 1.072 cm and 21 are control rod guide tubes of outer diameter 1.379 cm. The fuel is UO_2 with circaloy cladding, and the fuel-cladding gap is assumed to be positioned at a distance of 0.4742 cm from the fuel rod centre.

For the calculation the core is divided into four hydraulic subchannels distributed as shown by the numbers 1-4 in Fig. 3.2. The central fuel element with the ejected rod is a separate channel, the other element containing a control rod also has a channel of its own, and the last two channels are shared between the remainder of the elements so that elements adjacent to the reflector are taken as one channel and elements from the inner core as another. For the cross flow calculation the fuel elements are assumed to be completely open, and therefore the boundary gap width between two adjacent elements equals the length of the fuel element side, 21.4 cm.

Flow restrictions are assumed at the core inlet and outlet, and in addition spacers are placed at two levels in the core. These restrictions are identical for all of the four subchannels.

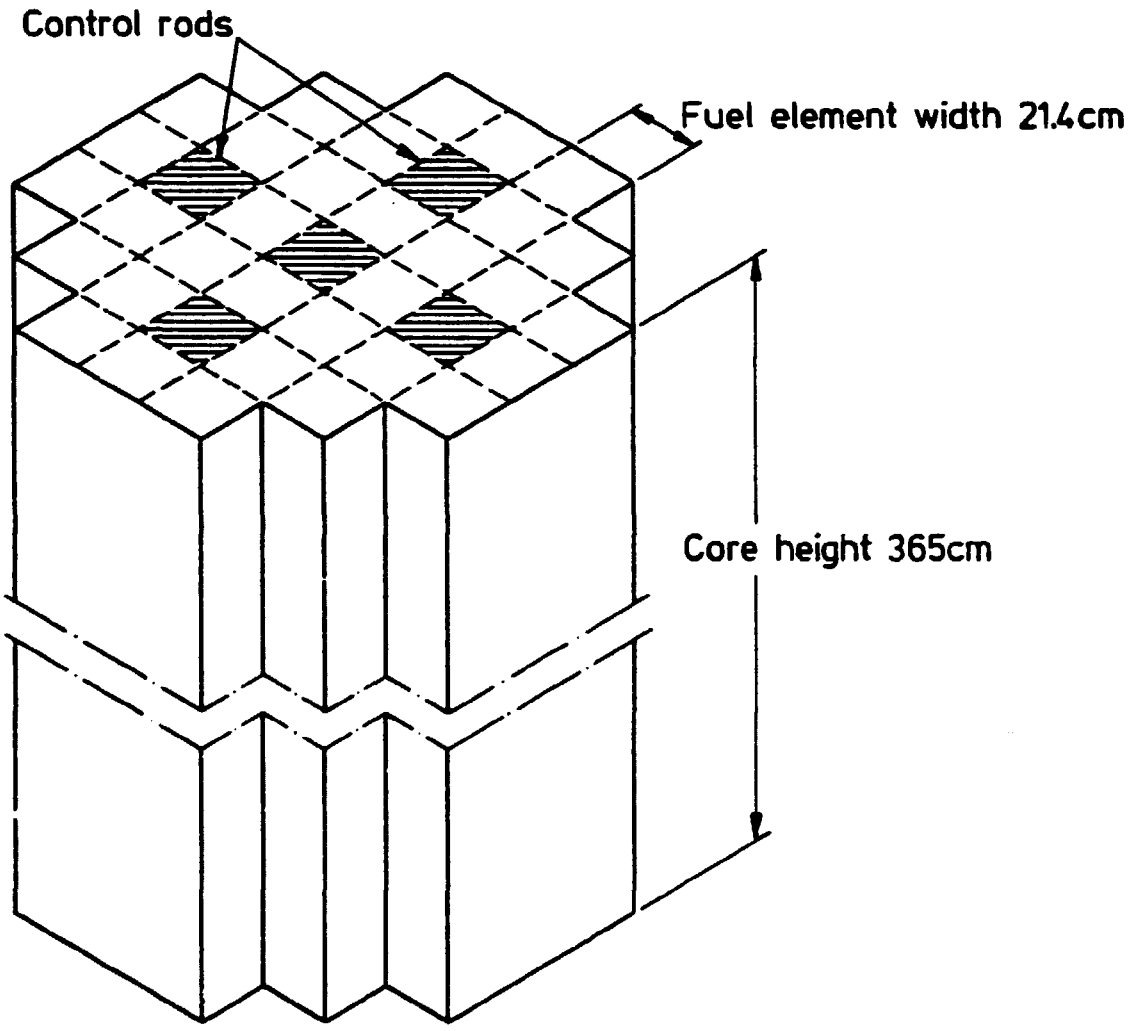


Fig. 3.1. Small reactor core used for the test calculation.

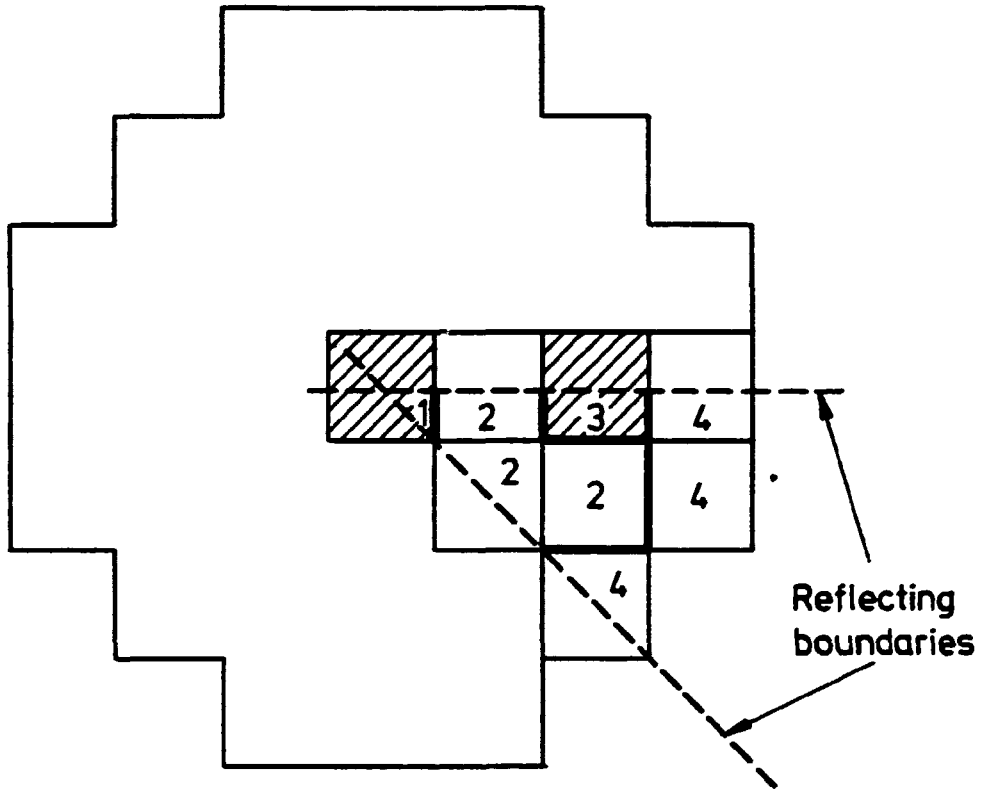


Fig. 3.2. Horizontal section of the small reactor core showing the represented part of the core with the hydraulics channel distribution.

3.2. Neutron physics input, steady state

Most of the input data for the neutron physics part of the program is listed in Table 3.1. The represented 1/8 of the core is described by 112 nodes, the nodes obtained by dividing the fuel elements axially into 14 segments. The reflector is described by means of the constant albedo values given in the table.

The steady state power for the full reactor core was chosen to be 454.24 MW. A constant fraction, about 3 per cent of the fission power, is released directly in the coolant.

The neutron cross sections are expressed as functions of the three parameters fuel temperature, moderator density and moderator temperature. For the present testcase the constants for the cross section calculation were taken over unchanged from the calculations described in ref. 13. The cross sections used were, therefore, intended for BWR calculations, but this was thought to be unimportant for the present purpose of testing the ANTI program. A total of 12 fuel types are used, each of them described by 43 coefficients. The absorption cross section for equilibrium xenon is added to the absorption cross section of each node in the steady state calculation.

The represented part of the core contains fully inserted control rods in two fuel elements. The control rod absorption is calculated by Henry's method[14], assuming that the control rod is cross-shaped; for the present test calculation it was found to be satisfactory, since the effect of the control rod, an increase in the absorption cross section, is obtained no matter how the control rod is described. The absorption of the central control rod was reduced by a factor of approximately 0.8 to make the reactor near critical in the initial state.

Table 3.1

Neutron physics input data for the ANTI small core testcase, steady state

Number of nodes for 1/8 of core		112
	Horizontal nodes	8
	Vertical nodes	14
Node dimensions		
	Horizontally	21.402 cm
	Vertically	26.0714 cm
Albedo values		
	Top reflector	0.3
	Bottom reflector	0.0
	Side reflector	0.45
Power (1/8 of total reactor power)		$5.678.10^{7}$ W
Power fraction released in coolant		0.0312
Energy released in fuel per fission		0.31601×10^{-10} J
Nodal theory model (ref. 1)		TRILUX
g-factors	g_1	1.7
	g_2	0.038
	g_3	3.07
	g_4	4.5
Xe -poisoning		equilibrium
$Iodine^{135}$ yield per fission		0.061
- decay constant		2.87×10^{-5} s^{-1}
$Xenon^{135}$ yield per fission		0.003
- decay constant		2.093×10^{-5} s^{-1}
Microscopic thermal Xe^{135} absorption cross section		3.5×10^{6} barns
Number of different fuel types		12
Cross section coefficients the same as used for the calculations of ref. 13		
Number of control rods inserted in core		2

(Table 3.1 continued)

Control rod representation with the constants (ref. 1)		Henry's method (ref. 14)
	α_1	0.01
	α_2	0.5
	h	13.6 cm
	m	15.38 cm
	a	0.4 cm
Absorption of central control rod multiplied by the factor		0.8383

3.3. Hydraulics and fuel rod model input, steady state

The input for the hydraulic part of the program and for the fuel rod model is given in Table 3.2. The flow areas of the four channels are calculated by subtracting the area taken up by fuel rods and control rod guide tubes from the cross section area of the fuel elements included in each channel (see Fig. 3.2). The heated perimeter is the total perimeter of the fuel rods belonging to the channel, and the wetted perimeter is the sum of the perimeters of fuel rods and guide tubes.

The boundary gaps are calculated simply as the lengths of the boundaries between adjacent subchannels (Fig. 3.2).

The flow restrictions are the same for all four channels, and are placed at the axial positions and with the loss coefficients given in the table.

The inlet and outlet boundary conditions used are inlet enthalpy, inlet flow and outlet pressure as shown in the table. The subchannel pressure drop is specified constant for all subchannels.

The fuel rod dimensions and the radial mesh division for the fuel are also given in the table. Axially the fuel rod mesh division is the same as for the hydraulic calculation, i.e. 16 equidistant axial mesh-points.

In addition to the input values of Table 3.2 a number of correlation constants are input specified, i.e. constants for use in the expressions for slip, single phase friction, flashing and condensation, cross flow mixing and cross flow resistance. The constants for slip were chosen to put the steam drift velocity equal to zero, in other words the calculations were done with a homogeneous two-phase flow model. All other constants were given typical values, and no careful study was performed of what their proper values should be.

Table 3.2.

Thermal-hydraulics input data for the ANTI small core testcase, steady state.

Number of subchannels		4	
Number of axial mesh-points		16	
Flow areas of subchannels	ch 1	0.003032	m^2
	ch 2	0.048511	m^2
	ch 3	0.012128	m^2
	ch 4	0.048511	m^2
Wetted perimeters of subchannels	ch 1	0.9724	m
	ch 2	15.559	m
	ch 3	3.8898	m
	ch 4	15.559	m
Heated perimeters of subchannels	ch 1	0.8587	m
	ch 2	13.739	m
	ch 3	3.4348	m
	ch 4	13.739	m
Number of boundaries		4	
Gap widths of boundaries			
	ch 1 - ch 2	0.1070	m
	ch 2 - ch 3	0.3210	m
	ch 2 - ch 4	0.4280	m
	ch 3 - ch 4	0.1070	m
Number of spacers		4	

	Distance from bottom of core	Loss coefficient
1	0.0037 m	4.0
2	1.217 m	1.5
3	2.433 m	1.5
4	3.645 m	0.5

Pressure at outlet	1.551×10^7	N/m^2
Inlet water enthalpy	1.241×10^6	J/kg
Inlet mass flux	3.445×10^3	kg/m^2s
Fuel rod outer radius	0.5359	cm
Fuel-cladding gap positioned at radius	0.4742	cm
Number of radial meshes in fuel	2	
Number of radial meshes in cladding	1	

3.4. Input for the transient calculation

The transient is initiated by the removal of the control rod from the central fuel element. The control rod is withdrawn upwards at constant speed from its fully inserted position at time zero to fully out at 0.3 s. The hydraulic boundary conditions at core inlet and outlet are kept constant throughout the transient, an assumption which of course is unrealistic.

The delayed neutron constants used for the transient calculation and the splitting up of the energy released per fission are given in Table 3.3. Of the delayed energy only the slowly decaying part is included in the program; it is considered constant in time and is calculated on the basis of the steady state power.

Table 3.3.

Input for rod ejection transient calculation, ANTI small core test case

Number of delayed neutron groups		6
group	delayed neutron fraction	decay constant
1	0.00024	0.0124 s^{-1}
2	0.00123	0.0305 s^{-1}
3	0.00130	0.111 s^{-1}
4	0.00250	0.301 s^{-1}
5	0.00090	1.14 s^{-1}
6	0.00030	3.01 s^{-1}
Distribution of energy per fission (ref. 1)		
prompt energy, fuel		0.29334×10^{-10} J
prompt energy, coolant		0.91320×10^{-12} J
delayed energy, fuel		0.22670×10^{-11} J
delayed energy, coolant		0.10414×10^{-12} J
Control rod movement		
central rod fully inserted at		0.0 s
- - - withdrawn at		0.3 s

4. RESULTS

The transient chosen for the present calculations is not a very dramatic one. This is mainly due to the start condition of the reactor, hot full power. A control rod ejection from zero or low power would have much more serious consequences.

Two calculations were made; one with and one without the possibility of cross flow between the fuel elements. In both cases the first 4.7 s of the transient were calculated in about 1300 time steps. The computing time for the cross flow case was nearly 8.5 hours CPU on the Burroughs B6700 at Risø, and for the case without cross flow a little more than 5 hours was used. These rather long running times can probably be reduced by optimisation of the program.

4.1. Steady state solution

The solution for the steady state required 23 iterations between neutronics and hydraulics to converge. The results, however, did not change much after the first ten iterations, but to be on the safe side rather strict convergence criteria were applied. Very little difference was seen between the cases with and without cross flow in the steady state.

The k_{eff} of the test-case reactor was found to be 1.0043 for the cross flow case and 1.0041 without cross flow (these values are used to correct the neutron cross sections before start of the transient so that the initial state is exactly critical). The normalized steady state power distribution is shown in Fig. 4.1 in the form of fuel element powers and axial power distribution. Since all control rods are either fully inserted or fully withdrawn, the axial distribution is nearly the same for all elements. The steady state power distributions for the cases with and without cross flow are almost identical.

No boiling is observed in the steady state. Maximum cladding temperatures are 324 degrees C for the cross flow case and 327 C for the no cross flow case. The average fuel temperature

is 707 C in both steady state calculations, and the peak values are 960 and 963 C, respectively. As expected the highest temperatures are found in channel no. 2, which consists of the three fuel elements of the inner core without control rods.

4.2. Rod ejection

The transient is initiated by the withdrawal of the control rod inserted in the central fuel element. The rod is withdrawn upwards at a constant speed and is fully out at 0.3 s from the beginning of the transient. This is the only disturbance, and no reactor scram is assumed to take place. The hydraulic boundary conditions, i.e. inlet flow and enthalpy and outlet pressure, are kept constant throughout the transient. Results of the transient calculation are shown in Figs. 4.2-4.6.

The total reactor power, shown in Fig. 4.2, increases rapidly at the beginning of the transient as a result of the reactivity insertion. After approximately 0.2 s the fuel temperature increase seen from Fig. 4.4 causes the reactor core to become subcritical and the power to drop. From about 1 s into the transient the power stabilizes at a power level of about twice the initial power, corresponding to a new critical condition of the core without the central control rod. Fig. 4.3 shows the fraction of the power which is released directly in the coolant; it is simply 3 per cent of the total power shown in Fig. 4.2.

The steady state power form factor is 1.76 (highest node power divided by average node power). During the initial power peak the form factor goes up to about 3.0, then decreases again and levels off around a value of 2.35, the highest rated fuel element now being the central element from which the control rod was removed. This form factor increase is very modest compared to the earlier control rod ejection calculation by Bjørn Thorlaksen[13] where the power form factor rose from 2.5 to 13 in the first 0.2 s of the transient. This difference is explained by the limited size of the core in the present test

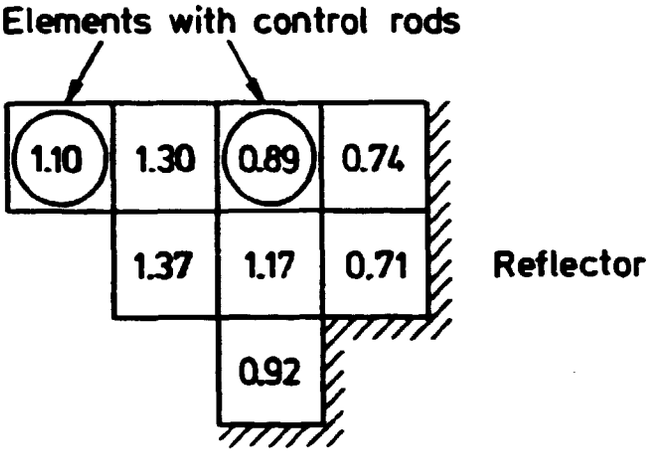
example and by the start condition which is hot full power here and was hot zero power for the calculation reported in ref. 13.

In Fig. 4.4. the core average fuel temperature and the fuel temperature of the hottest node are shown as functions of time. The temperatures start increasing very soon after the beginning of the transient, thereby limiting the power excursion. Core average fuel temperature stabilizes at 1000 degrees C; in the hottest node which by the end of the transient calculation is found in the lower part of the central fuel element the fuel temperature is 1750 C and still slowly increasing when the calculation is terminated at 4.7 s.

The cladding temperature increase starts a little later and stops again at the onset of subcooled boiling. The maximum cladding temperature is shown in Fig. 4.5 as a function of time. After the initial increase and after the boiling has started it stays constant at the value of 350 C.

The maximum void fraction in the core is shown as a function of time in Fig. 4.6. The feed-back from water density has only a limited influence on the course of the transient, since the voiding of the core occurs after the power peak. For the case without cross flow the void content in the hot channel goes up to 30 per cent, whereas the influence from adjacent, cooler channels brings the maximum void down to 15 per cent in the cross flow case. In any case the void fractions never get very high, and no burnout is predicted. The lowest value of the DNB-ratio calculated during the transient is 1.3 in the cross flow case (in the case without cross flow DNB-ratios were not calculated).

Only the void fractions show an important difference between the cross flow and the no cross flow calculations. The power production is a bit lower in the no cross flow case between 0.5 and 3 s (Figs. 4.2 and 4.3) probably because of the higher void content of the moderator. Consequently also the fuel temperatures (Fig. 4.4) are slightly lower, but the cladding temperatures are identical in the two cases.



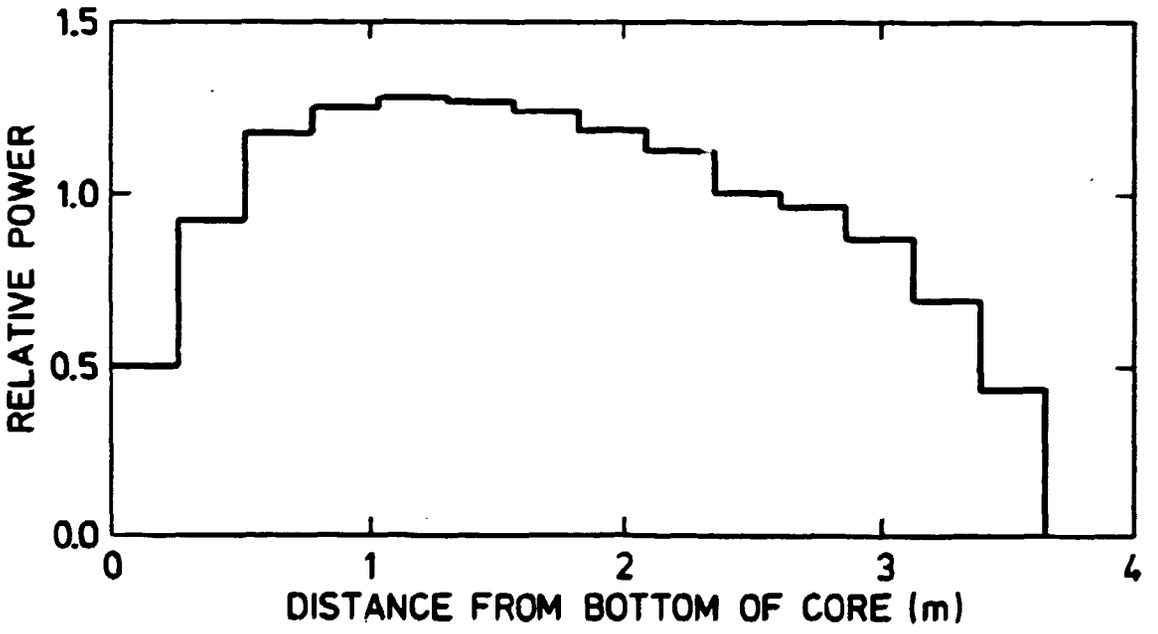


Fig. 4.1. Horizontal and vertical normalized power distribution, steady state.

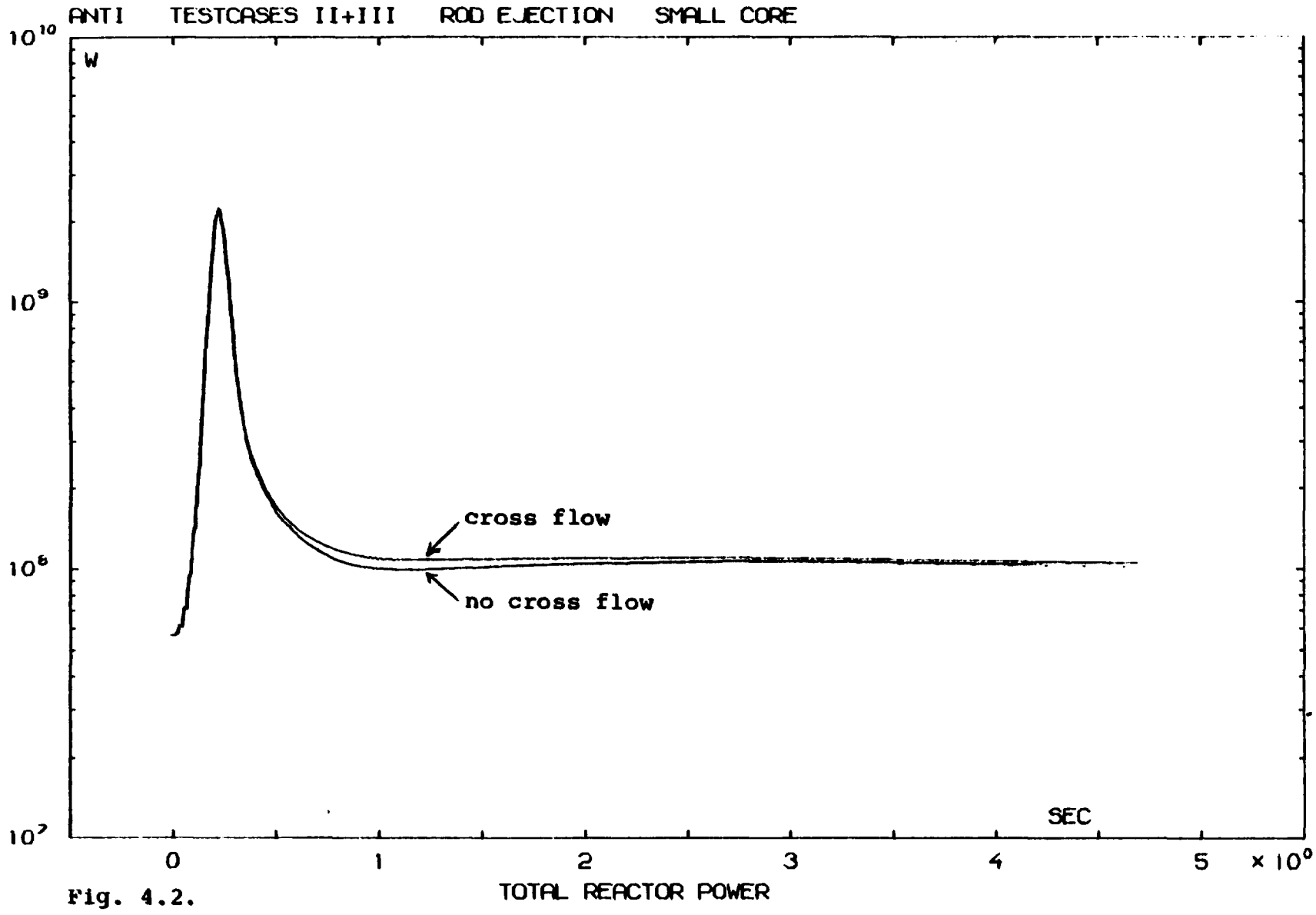

ANTI TESTCASES II+III ROD EJECTION SMALL CORE
W
10^{10}
10^{9}
10^{8}
10^{7}
cross flow
no cross flow
SEC
0
1
2
3
4
5
x 10^{0}
TOTAL REACTOR POWER

Fig. 4.2.

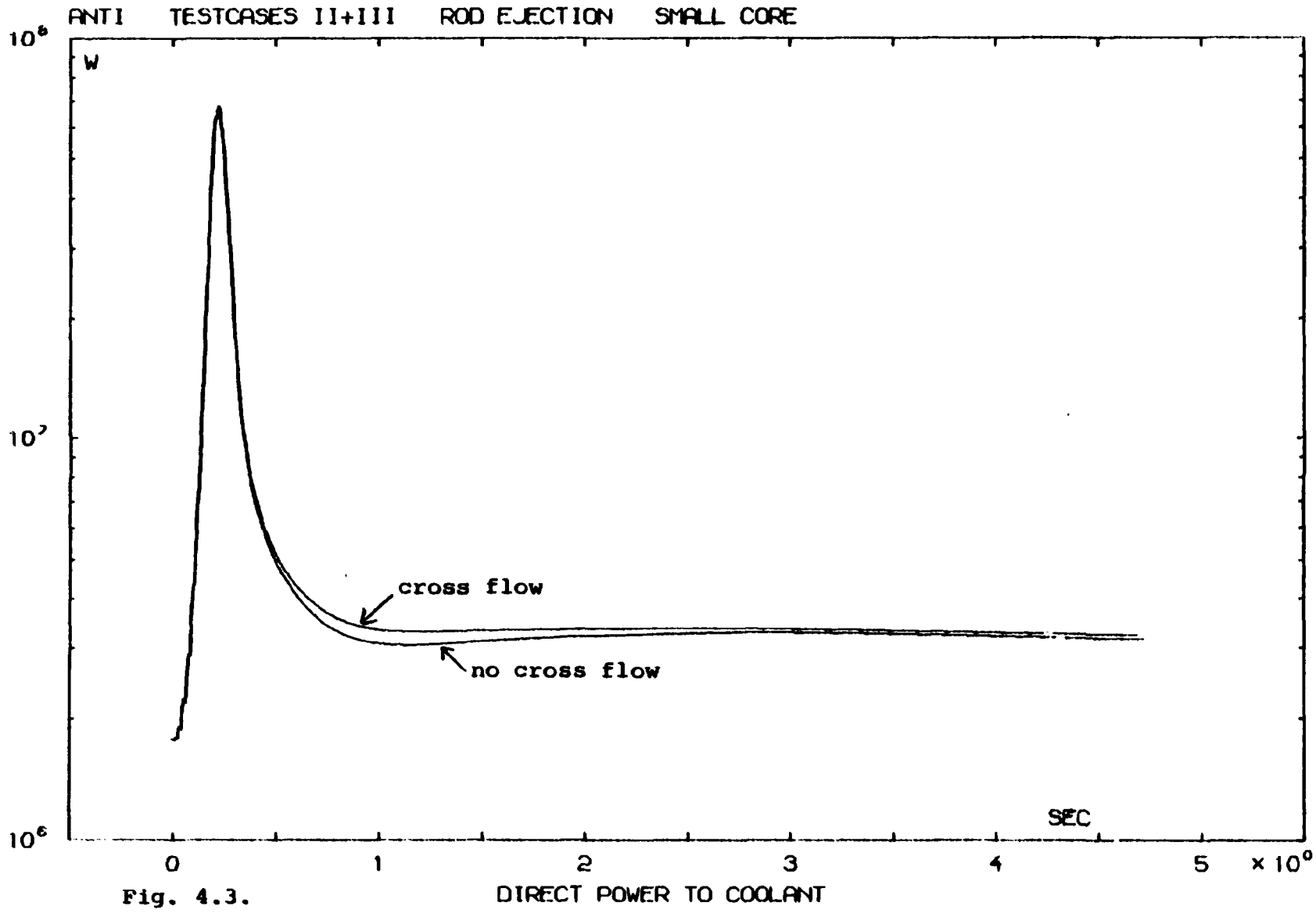

ANTI TESTCASES II+III ROD EJECTION SMALL CORE
10⁸
W
10⁷
10⁶
cross flow
no cross flow
SEC
0
1
2
3
4
5
x 10⁰
DIRECT POWER TO COOLANT

Fig. 4.3.

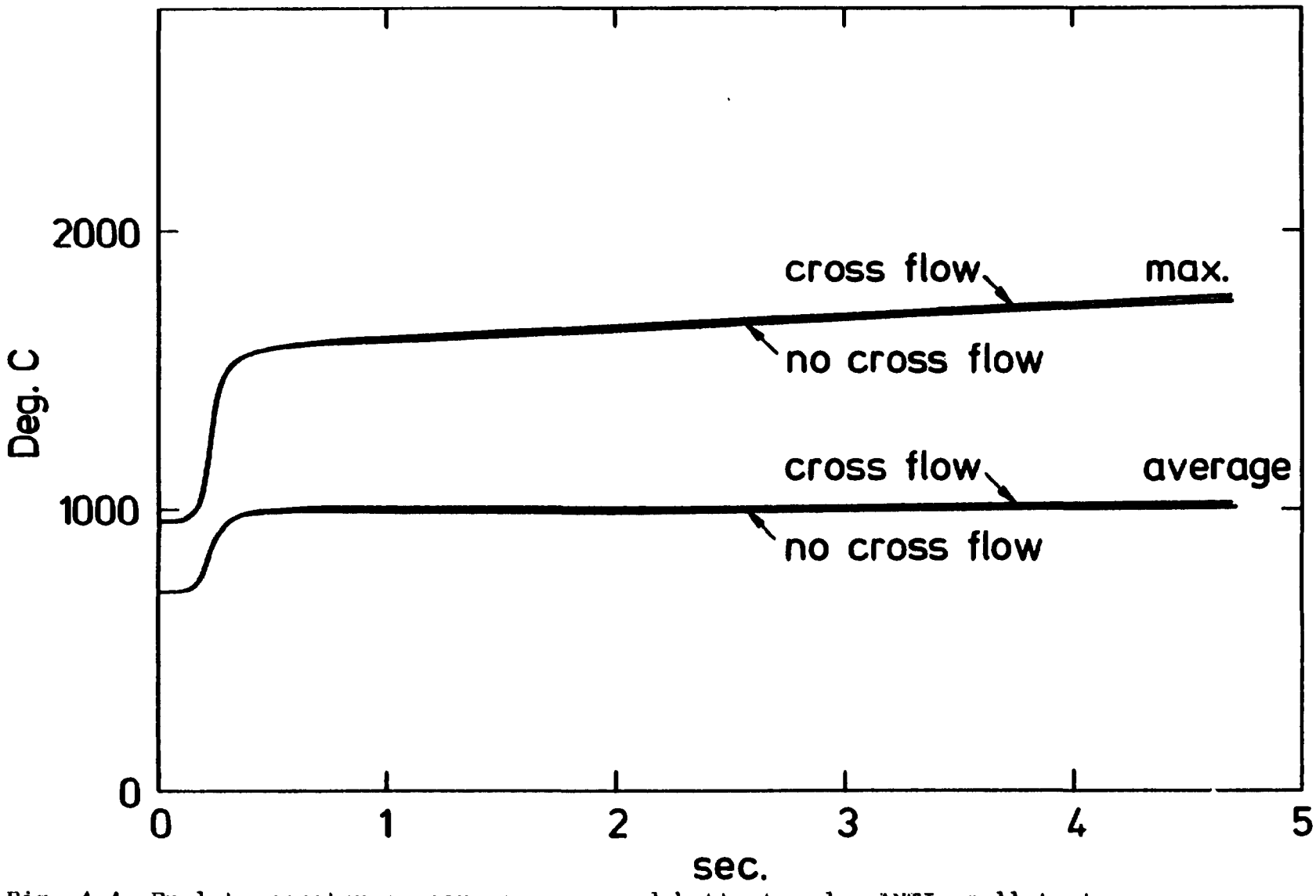


Fig. 4.4. Fuel temperatures, core average and hottest node. ANTI small testcase with and without cross flow.

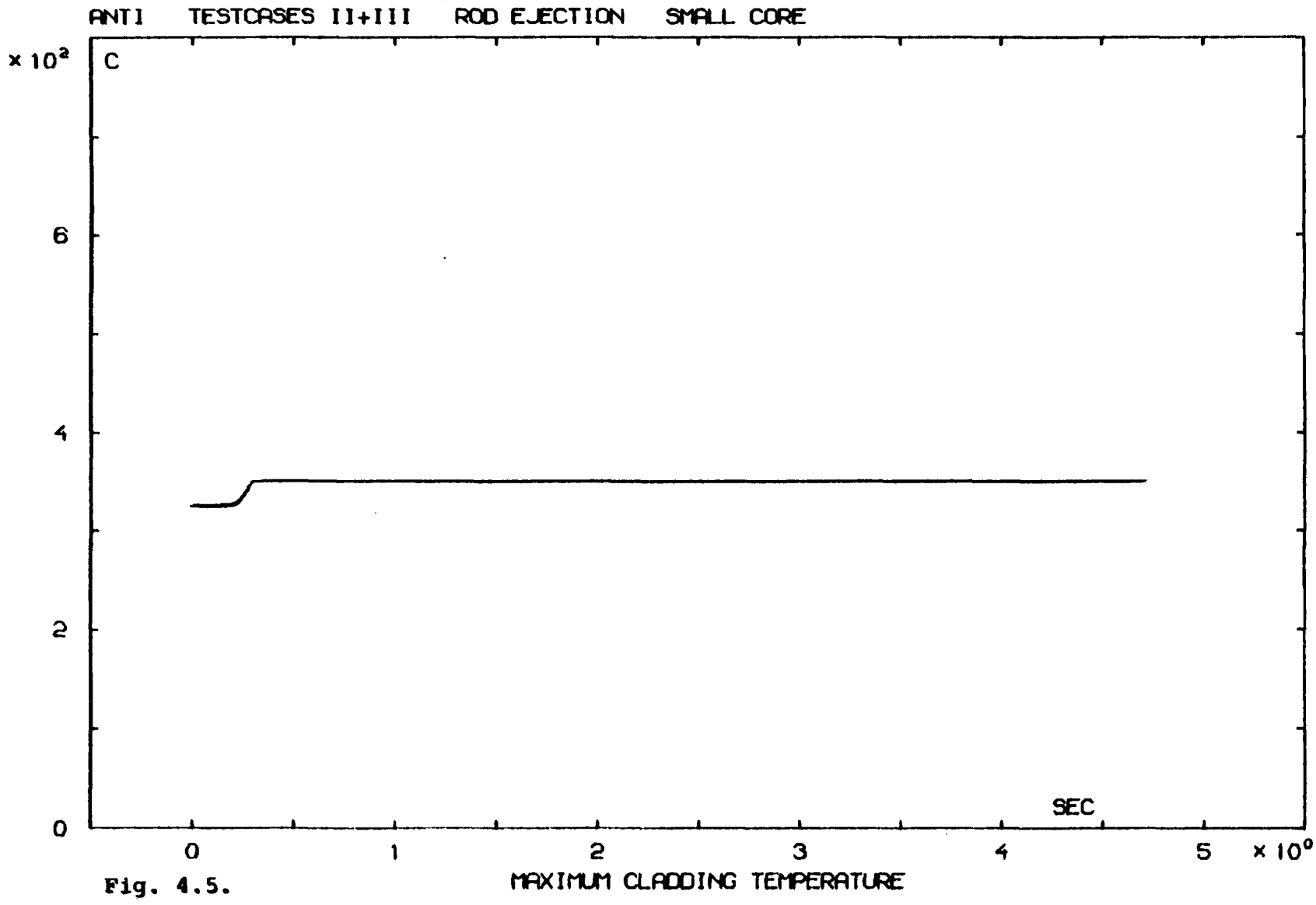

ANTI TESTCASES II+III ROD EJECTION SMALL CORE
x 10²
C
6
4
2
0
SEC
0
1
2
3
4
5
x 10⁰
MAXIMUM CLADDING TEMPERATURE

Fig. 4.5.

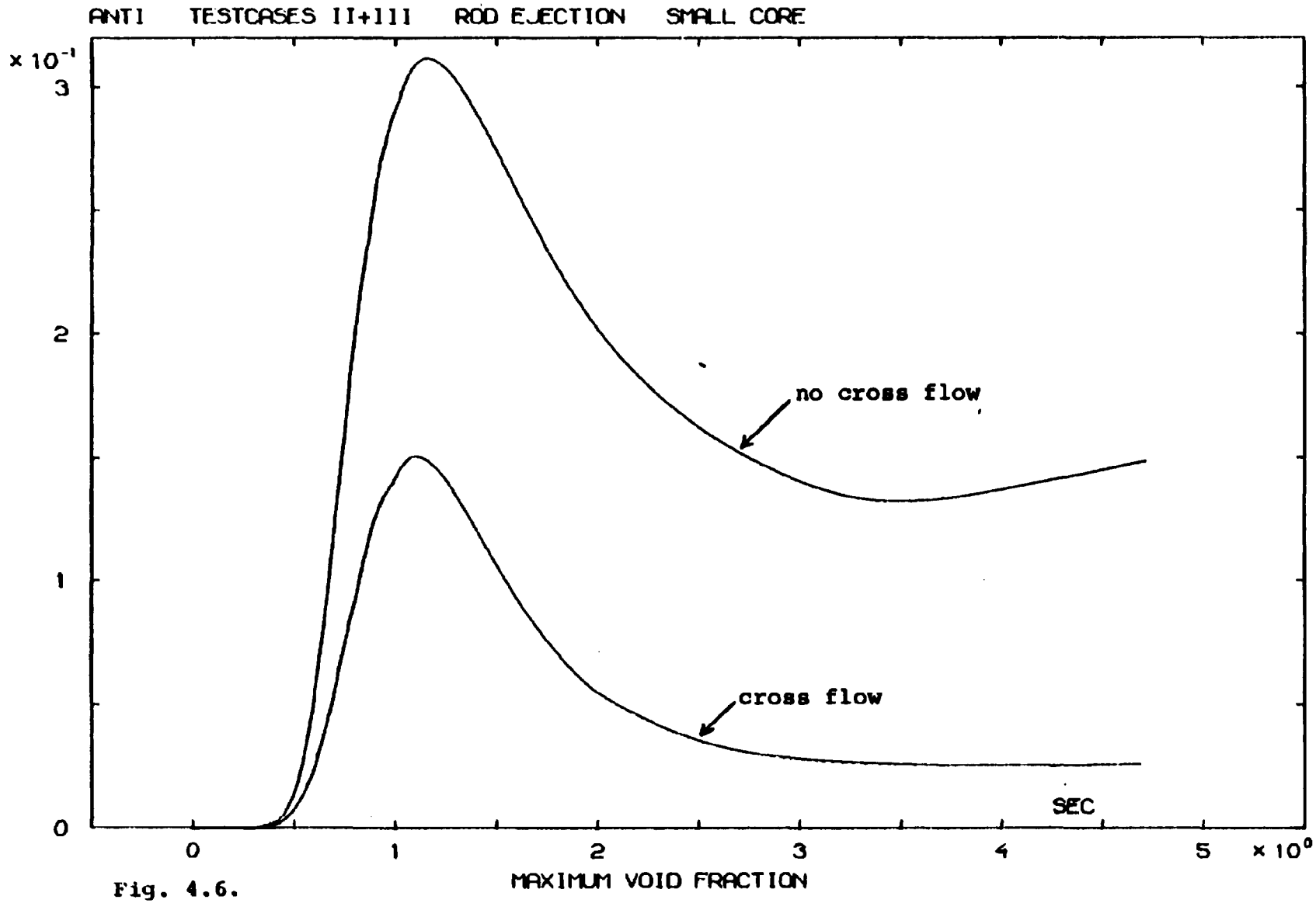

ANTI TESTCASES II+III ROD EJECTION SMALL CORE
x 10⁻¹
3
2
1
0
no cross flow
cross flow
SEC
0
1
2
3
4
5
x 10⁰
MAXIMUM VOID FRACTION

Fig. 4.6.

5. CONCLUSION

The calculation with and without cross flow of the conditions in the core for the first seconds of a PWR control rod ejection accident were described. These calculations are part of the initial testing of the ANTI program intended for three dimensional PWR core transient calculations.

The test cases have demonstrated that the ANTI program is working and that the coupling between the hydraulics and neutronics parts is functioning in the way expected. The calculational results, however, should not be taken as indicative of what would happen in an actual reactor in case of a hypothetical control rod ejection accident. The effect of cross flow was, as expected, a reduction in the difference between the hot channel and adjacent channels. In this particular case of a not very dramatic transient the maximum void was the parameter most affected by the cross flow.

In order to gain confidence in the results produced by ANTI more test calculations are needed. Such test calculations should preferably be on realistic cases with comparison to measured data, or, if this is not possible, at least comparison to the results of other computer codes should be made. Both types of calculations are under preparation; a calculation for the Westinghouse three-loop type reactor core is being set up, and small test-cases for comparison with calculations by the ANDYCAP BWR program and the MIT program MEKIN are planned.

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THE THREE-DIMENSIONAL PWR TRANSIENT CODE ANTI; ROD EJECTION TEST CALCULATION

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Date
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Department or group

Department of Reactor Technology

Group's own registration number(s)

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Abstract

ANTI is a computer program being developed for three-dimensional coupled neutronics and thermal-hydraulics description of a PWR core under transient conditions. In this report a test example calculated by the program is described. The test example is a simulation of a control rod ejection from a very small reactor core (to save computing time). In order to show the influence of cross flow between adjacent fuel elements the same calculation was performed both with the cross flow option and with closed hydraulic channels.

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