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EUROPEAN ATOMIC ENERGY COMMUNITY — EURATOM

HIGH VOID TESTS AT THE GARIGLIANO NUCLEAR POWER STATION

by

F. SANTASILIA
(ENEL)

1968

EURATOM/US Agreement for Cooperation
EURAEC Report No. 1782 prepared by ENEL
Ente Nazionale per l'Energia Elettrica, Rome — Italy

Euratom Contract No. 101-63-11 RDI

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KEYWORDS

SENN — 1
TESTING
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POWER
TEMPERATURE
OSCILLATIONS
CONTROL ELEMENTS
TRANSIENTS
COOLANT LOOPS
NEUTRON FLUX
STEAM
WATER COOLANT
REACTOR CORE

CONTENT

FOREWORD

This report describes a study conducted under the ENEL-EURATOM Research Program for the Garigliano nuclear power station.

The Research Program was set up in contract executed by the European Community for Atomic Energy (EURATOM) and the former Società Elettronucleare Nazionale (SENN), now Ente Nazionale per l'Energia Elettrica (ENEL), and put into effect on November 1, 1966.

The contract comes under the provisions of the Agreement for Cooperation signed by the Community and the United States Government at Brussels on November 18, 1958, and is part of the Research and Development Program of the United States Atomic Energy Commission and EURATOM, published in the Official Gazette of the European Community, No. 2, January 9, 1959.

HIGH VOID TESTS AT THE GARIGLIANO NUCLEAR POWER STATION

1. Introduction (*)

Within the framework of the Research and Development Program on the Garigliano station, a certain number of special tests were performed on the reactor from May 20 to June 1, 1966, to ascertain the stability of the plant under quite different conditions from steady state.

The tests were very delicate, and the definition of the performance limits and the detail programming of the individual tests took considerable technical and organizational effort. It should be borne in mind that a power plant is designed to produce electricity and therefore it is not always possible to meet all the requirements of a research program with the characteristics of the equipment subjected to special testing. While an experimental plant is conceived to operate within a very wide range of operating conditions, a power plant, for its very nature, is not so flexible and does not lend itself just as well to even temporary deviations from the normal operating procedures.

Another point of importance is that a power station is an extremely expensive research tool. Trouble caused by the use of the equipment under off-design conditions could give rise to prolonged plant outages thus imposing very heavy financial burdens on the operator.

Therefore, power stations should be used only for those tests which cannot be performed elsewhere and which allow an improvement in the performance of similar plants and the incorporation of the results obtained in the design of new stations.

In the case in question, the purpose of the tests was to simulate higher power density conditions and therefore closer margins on the thermal limits.

This report will organically illustrate the results of the tests, the processing of the data obtained and a few conclusions on reactor behavior under streched conditions.

The results and analyses of the specific stability tests (control rod oscillation, transients, etc.) will be presented

(*) Manuscript received on March 29, 1968.

in detailed form and elaborated in another special report.

2. Special tests

All the tests performed on the Garigliano plant were amply described elsewhere (*) and therefore this report will merely give a brief summary of the operations and action taken to bring the reactor to each of the preset testing conditions, with an aim at a better understanding of the significance of the information obtained and of the subsequent analyses.

In order to simulate an increased power density without exceeding the license thermal power rating of the reactor (506.3 MW) the coolant flow through the core was properly decreased thus obtaining a higher void content. This condition can also be obtained by leaving the colant flow unaltered and by increasing the reactor power above the rated value. Of course, the higher void content during the tests induced an appreciable negative reactivity which had to be off-set by withdrawing the control rods considerably.

In addition to the aim of increasing the void content in the core, the influence of subcooling on reactor stability was also to be ascertained. Therefore, for the same thermal output in the same coolant flow, the temperature of the water entering the reactor was properly varied by acting on the primary/secondary steam ratio. Indeed, it is evident that having subtracted power from the primary system, to produce a greater amount of secondary steam it is necessary to increase subcooling at the core inlet directly. The test program made provisions for even more appreciable variations in the subcooling by cutting out all the feedwater heaters. This aim, however, was not reached because such an exceptional mode of operation caused excessive vibrations in the piping system and equipment connected to the turbine extraction lines. It should be noted that the operating procedures contemplated the possibility of cutting out, at the most, two of the four heaters.

(*) See Quarterly Report No. 9 of July 1, 1966: "Research Program for the Garigliano Nuclear Power Station", ENEL, Direzione delle Costruzioni Termiche e Nucleari.

3. Methods and calculations adopted for the determination of critical parameters

Before describing the individual tests and giving the results obtained, it is necessary to make a few preliminary considerations on the methods adopted and calculations performed prior to and during the tests to check the reactor safety margins continuously.

The basic thermal limits that were to be assumed during the tests for the Research Program as agreed by ENEL and General Electric and approved by the Italian National Committee for Nuclear Energy (CNEN) were the following (*):

Maximum local heat flux	113.5 W/cm^2
Minimum critical heat flux ratio at overpower (MCHFR)	1.5
Overpower factor (OP)	1.20
Setting of high flux trip	120% of actual power

The determination of the maximum local heat flux for each fuel element depends on the following factors:

a) Average heat flux in the core = $\dfrac{\text{reactor power}}{19.22 \times 10^6 \text{ cm}^2}$

b) Radial power factor of the fuel element

c) Axial power factor

d) Maximum local power factor.

The product of these four factors gives the maximum local heat flux for each of the 208 elements in the core. The determination of the axial power factor is performed both analytically (by means of a digital computer) and experimentally (by means of wire irradiation, movable in-core chambers),

(*) See GEAP 4899, June 1, 1965 "Final Safeguards Report" - Garigliano Development Program whereas the radial power factor is only determined analytically with the digital computer. For the latter factor it is therefore essential that the measurement of the parameters required for the calculation be carried out as carefully as possible. One of the most important parameters is the coolant flow in the core or, better, the coolant flow through the fuel channels (active flow). This in turn depends on the total recirculation flow and on the water flow crossing the core without entering the channels (leakage flow) (*).

The active flow is a basic value in the determination of the MCHFR which is a function both of the specific channel coolant flow and of the steam quality which is also dependent on the coolant flow.

3.1 Determination of coolant flow

The total recirculation flow W_T is normally determined as the sum of the flows W_A and W_B in the two recirculation loops. The partial flows are, in turn, obtained from the relations:

$$W_A = K_A \Delta P_A; \qquad W_B = K_B \Delta P_B$$

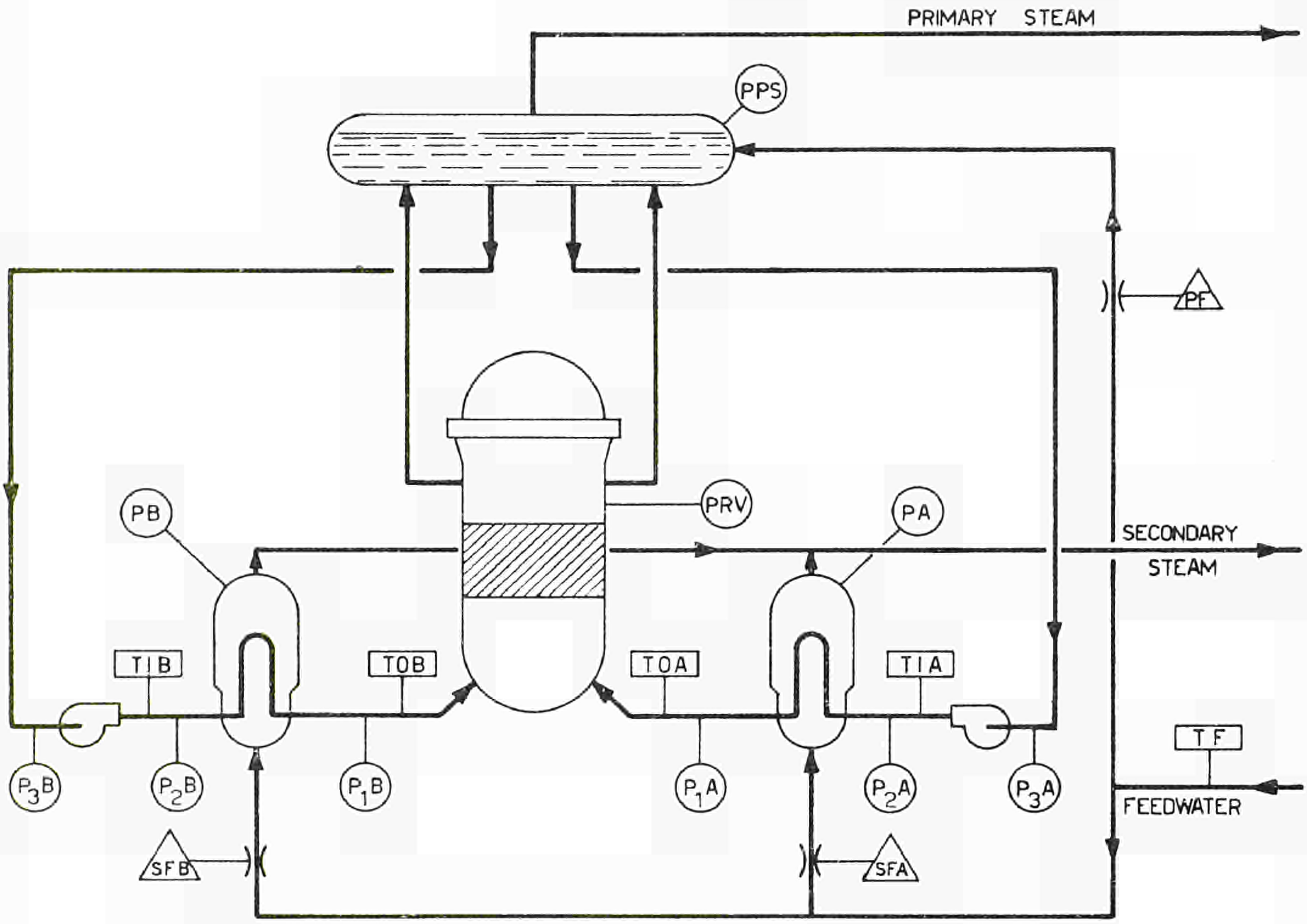
where ΔP_A and ΔP_B are the pressure drops measured across the secondary steam generators, and K_A and K_B are the constants of the generators themselves.

A few days before the tests, a certain number of measurements were taken to find out the hydraulic calibration constants of the generators. These measurements, due to operating requirements, were performed only in cold conditions. Once the pressures P_1A, P_2A, P_3A, and pressures P_1B, P_2B, P_3B were known, (Fig. 1), the constants K_A and K_B were calculated with the recirculation pump characteristics and the following equations were obtained:

(*) See "GEAP 4899, June 1, 1965, "Final Safeguards Report" - Garigliano Development Program, Pages 4/1 - 4/43.

CENTRALE DEL GARIGLIANO

SKETCH OF NUCLEAR STEAM SUPPLY SYSTEM



INSTRUMENT SYMBOLS

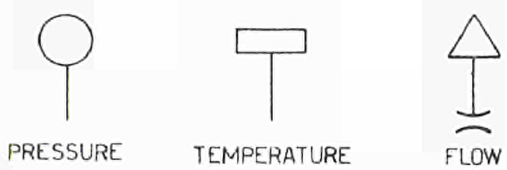


FIG. 1

$$\Delta P_A = 0.3838 \times 10^{-6} W_A^2$$

$$\Delta P_B = 0.03726 \times 10^{-6} W_B^2$$

where ΔP_A and ΔP_B are expressed in kg/cm^2, and W_A and W_B in tons/hr.

Since these equations are valid for a water temperature of 36°C, they were corrected for density so as to extend their validity to the operating temperature (273.5°C). Therefore:

$$\Delta P_A = 0.05001 \times 10^{-6} W_A^2$$

$$\Delta P_B = 0.04855 \times 10^{-6} W_B^2$$

These equations were derived from measurements taken with the full recirculating flow and subsequently with the inlet valves 2/3 open and then 1/3 open. During those special tests of the Research Program, in which the reactor was in natural circulation, the reactor circulation pumps were out; since the characteristics of the pumps were not available, reliance had to be made on the above mentioned equations. However, to maintain the calculations of the reactor safety margins more prudential, higher hydraulic coefficients were adopted in these natural circulation tests, namely:

$$K_A = 0.06300 \times 10^{-6}$$

$$K_B = 0.06305 \times 10^{-6}$$

Therefore, the digital computer always demonstrated lower flow rates and consequently slightly higher steam contents in the channels then the actual ones.

By means of heat balances it was then possible to check the recirculation flow rates very carefully. Indeed, with the secondary steam generators in operation the recirculation flow for each loop was found to be:

$$W_A = \frac{(SFA) \times (H_{PA} - H_{TF})}{H_{TIA} - H_{TOA}}$$

$$W_B = \frac{(SFB) \times (H_{PB} - H_{TF})}{H_{TIB} - H_{TOB}}$$

where (Fig. 1):

W_A and W_B = Recirculation flow rates in the loops A and B (tons/hr)

SFA and SFB = Feed flows to generators A and B (tons/hr)

H_{PA} and H_{PB} = Saturated steam enthalpies determined on the basis of pressures PA and PB (kCal/kg)

H_{TF} = Feedwater enthalpy determined on the basis of temperature TF (kCal/kg)

H_{TIA} and H_{TIB} = Enthalpies of recirculation water entering generators A and B, determined on the basis of temperatures TIA and TIB (kCal/kg)

H_{TOA} and H_{TOB} = Enthalpies of recirculation water leaving generators A and B, determined on the basis of temperatures TOA and TOB (kCal/kg).

When the secondary steam was completely cut out, use was made of the following formula:

$$W_T = PF \; \frac{H_{PPS} - H_{TF}}{H_{PPS} - H_{TI}}$$

where (Fig. 1):

W_T = Total recirculation flow (tons/hr)

PF = Primary feedwater flow (tons/hr)

H_{PPS} = Saturated liquid temperature at steam drum pressure (PPS)

H_{TF} = Feedwater enthalpy

H_{TI} = Recirculation water enthalpy determined on the basis of temperatures TIA and TIB.

The foregoing formulas confirmed the validity of the relationship between recirculation flow and pressure drop in the secondary steam generators, obtained from measurements taken in cold condition before the tests. Therefore, the core relations used by the digital computer during the tests were actually too conservative and this made it necessary to review all calculations during the processing of the results.

3.2 Determination of the leakage flow

Another important parameter for the determination of the active flow is the leakage flow around the fuel elements. In the thermohydraulic computer program the leakage flow is determined by means of the following formulas:

$$W_L = 2.148 \times 10^3 \, (\Delta P_{CSP})^{1/2} + 0.337 \times 10^3 \quad \text{(in forced circ.)}$$

$$W_L = 2.148 \times 10^3 \, (\Delta P_{CSP})^{1/2} \quad \text{(in natural circ.)}$$

where

W_L = leakage flow (tons/hr)

ΔP_{CSP} = pressure drop across the core support plate (kg/cm^2)

Since the computer measures the pressure drop across the whole core directly by means of the system simplified in Fig. 2, in order to derive the ΔP_{CSP} (pressure drop between points A and B) it must correct the measurement according to the following formula:

$$\Delta P_{CPS} = \Delta P_M + L \, (\rho_{amb} - \rho_R)$$

where

ΔP_M = measured pressure drop

L = liquid column between the measurement point and the reactor bottom

ρ_{amb} = water density in the measurement tubes

ρ_R = water density in the reactor.

CENTRALE DEL GARIGLIANO

CORE ΔP SYSTEM

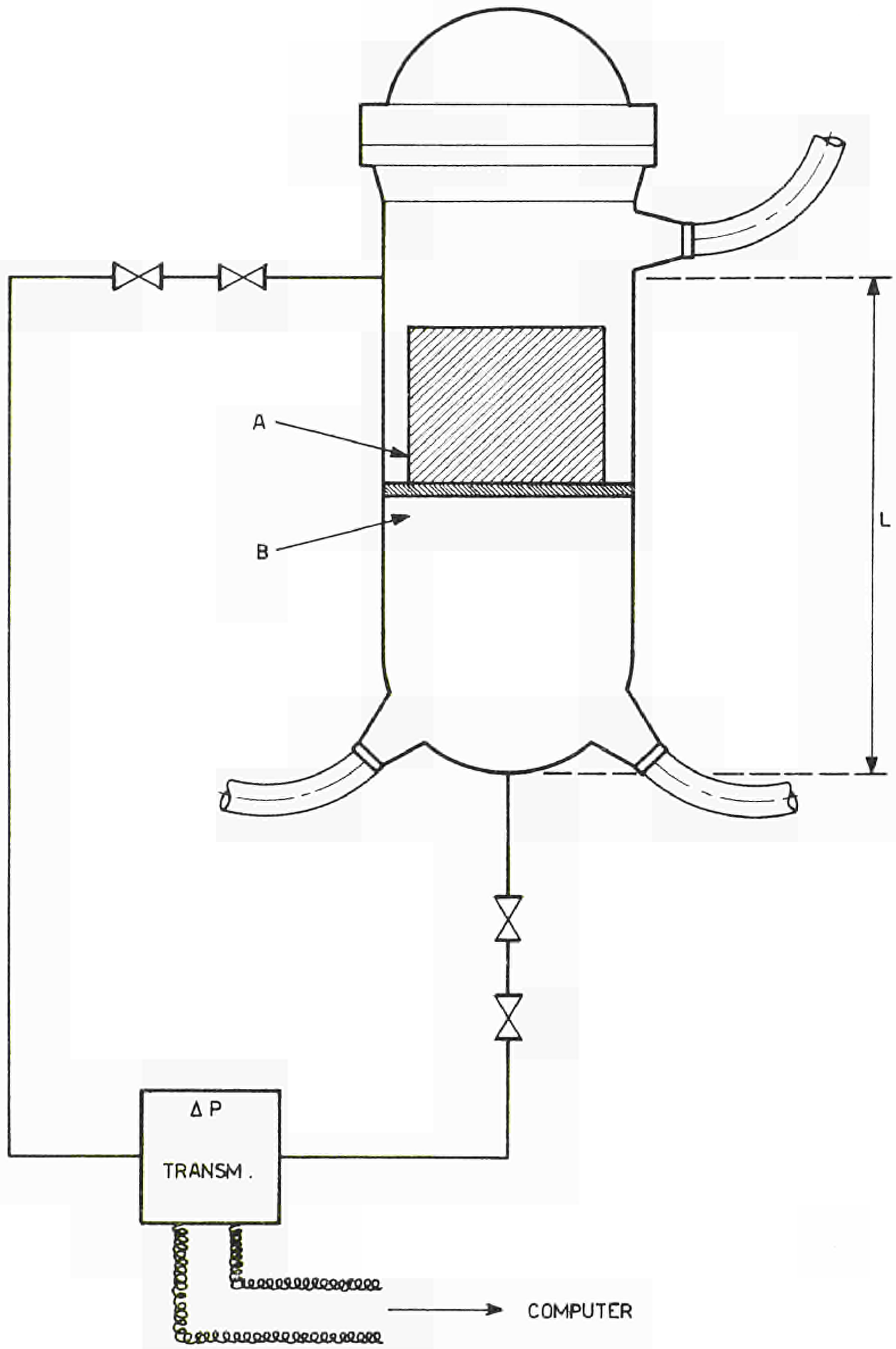


FIG. 2

However, in natural circulation, because of the low flow rates, the term $L(\rho_{amb} - \rho_R)$ prevails over ΔP_{CSP}, and the computer would therefore measure a negative ΔP_{MIS}. This would have complicated the tests, so that it was decided that fixed values should be assumed for the leakage flow, namely, 19% and 10% of the total recirculation flow, respectively for forced-circulation tests and natural-circulation tests. Upon further examination, these values were found to be too conservative. On the basis of theoretical studies performed in the United States (★), it was demonstrated that the above leakage flow percentage decreased proportionately to the total recirculation flow. In addition, a few observations with instrumented fuel assemblies confirmed that the values of the leakage flow should have been smaller. This matter will be dealt with in greater detail later on.

In the subsequent test data processing, the leakage flows were modified as follows:

Test 1-A: from 19% to 13.5% of the recirculation flow

Test 7 : from 10% to 8% of the recirculation flow

Test 8 : from 10% to 7% of the recirculation flow

Test 9 : from 10% to 7% of the recirculation flow.

For the remaining tests (3, 5 and 6) it was not necessary to change the assumed value of 10%.

3.3 Subcooling

For the calculation of the void content in the core, the axial distribution of the steam quality in the channels, and thus the MCHFR and channel pressure drop, it becomes extremely important to know the amount of subcooling at the reactor inlet. The computer determines this parameter by measuring the reactor pressure (PRV) and inlet water temperatures (TOA and TOB) directly (Fig. 1). The amount of sub-

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(★) See, GEAP 4899 June 1, 1965 "Final Safeguards Report" - Garigliano Development Program - Pages 5/12 and 5/20 cooling is therefore given by the differences between the enthalpy of the saturated liquid at the reactor pressure, and the average enthalpy of the two inlet loops weighted over the recirculation flow rates. During data processing it was realized that for some tests the reactor pressure was too low when compared with the steam drum pressure (PPS) (Fig. 1). The following table shows the values given by the computer for the individual tests.

	Reactor pressure PRV	Steam drum pressure PPS	Difference PRV-PPS
Test 1-A	69.6 ata	68.2 ata	+ 1.4
Test 3	69.5 "	68.8 "	+ 0.7
Test 5	69.4 "	68.9 "	+ 0.5
Test 6	69.5 "	68.9 "	+ 0.6
Test 7	69.2 "	68.9 "	+ 0.3
Test 8	67.8 "	68.5 "	- 0.7
Test 9	67.7 "	68.3 "	- 0.6

Evidently, the differences between the two pressures could not be so small and even less negative, specially in the last tests. Since the steam pressure (PPS) measured by the computer remained constant and always in agreement with the readings of the large-scale pressure gage in the control room, it was inferred that the reactor pressure gage had gone out of calibration during the tests. Therefore, the difference between reactor pressure and steam pressure was calculated on the basis of the steam-water mixture flow rates through the risers, consideration being given to the pressure losses due to the water head, friction along the risers and the restrictions at the reactor outlet and steam drum inlet. Neither the slight variation in steam quality along the piping nor the difference in velocity of the two phases were taken into account in the calculations; however, the resulting error is negligible. Table I shows the results of the calculations and the corrections that it was necessary to make in the reactor pressure, specially for the last tests. Though these

TABLE 3-I

	Recirculation flow	Average steam quality at core outlet	Pressure drop between reactor and steam drum	Reactor pressure (corrected)	Difference from measured value
Test 1-A	9720 tons/hr	8.7%	1.37 kg/cm^2	69.57 ata	-0.03 ata
Test 3	4575 tons/hr	13%	0.68 kg/cm^2	69.48 ata	-0.02 ata
Test 5	4625 tons/hr	17%	0.69 kg/cm^2	69.59 ata	+0.19 ata
Test 6	4810 tons/hr	14%	0.73 kg/cm^2	69.63 ata	+0.13 ata
Test 7	2775 tons/hr	18%	0.50 kg/cm^2	69.40 ata	+0.20 ata
Test 8	2875 tons/hr	22%	0.56 kg/cm^2	69.06 ata	+1.26 ata
Test 9	2835 tons/hr	21%	0.55 kg/cm^2	68.85 ata	+1.15 ata

corrections do not appear to be very great, they have a considerable effect on subcooling, because an error of 1% in the saturation enthalpy means an error of 10% in the subcooling of the coolant entering the core.

3.4 Flow distribution in the channels

The digital computer determines the flow distribution in the channels by means of the thermohydraulic program (*) by iterating the calculation till the total pressure drop is the same for all the 208 channels in the core. Four instrumented assemblies provided with inlet flowmeters had been installed for the Research Program, and it was therefore possible to compair the measured flows with those computed by means of the thermohydraulic code. It was immediately observed that the channel flows calculated for the instrumented assemblies were smaller than those recorded by the flowmeters by 23-25%. After having ascertained that the hydraulic coefficients of pressure drops at the inlet of the four instrumented assemblies had been factored in the calculation correctly, we realized that the total pressure drop across the core, as computed with the thermohydraulic code, was smaller than the measured drop (Fig. 2). The difference could be computed both to a too low value assumed for the total recirculation flow, and to changed hydraulic characteristics of the core. The first assumption was immediately discarded as many factors concurred to demonstrate the correct assessment of the coolant flow. Instead, the second assumption was credible because during the first fuel cycle, crud deposits were found on the fuel element base plates with a consequent restriction of the flow passages. During plant shutdown these deposits were removed,though slight percentage remained on the tube plates thus contributing to change the hydraulic characteristics of the system. By trial and error the hydraulic coefficients of the base plates of all the elements were increased except for the instrumented assemblies which had been placed in the reactor recently and presented no deposits.

(*) See, GEAP 4702 "Specification of SENN on-line computer system and thermal-hydraulics calculation functions", by T. Sorlie

The pressure drop localized on the bottom plate of the elements is expressed by the following formula:

$$\Delta P = \frac{K}{A^2} \frac{W^2}{2g\rho}$$

where

ΔP = pressure drop (kg/cm^2)

K = loss factor for change in section

A = area of restricted section (cm^2)

W = coolant flow (kg/sec)

g = gravity acceleration (cm/sec^2)

ρ = density of the liquid (kg/cm^3)

For the normal elements, the design coefficient was $1.38 \times 10^{-3}\ cm^{-4}$. After a few trials, this factor was raised to $3.17 \times 10^{-3}\ cm^{-4}$, whereas the factor related to the instrumented assemblies was not changed. The immediate result was that the total core pressure drop obtained in the calculation was equal to the measured value. A further confirmation of the validity of the modification was that the channel flows in the instrumented assemblies obtained with the thermohydraulic code were almost the same as the measured values. To understand this factor better it is necessary to consider the following table:

	Values before the modification	Values after the modification
Average channel flow in normal elements	40 tons/hr	40 tons/hr
Measured flow in the instrumented assemblies	70 tons/hr	70 tons/hr
Calculated flow in the instrumented assemblies	47 tons/hr	72 tons/hr
Measured core ΔP	0.53 kg/cm^2	0.53 kg/cm^2
Calculated core ΔP	0.37 kg/cm^2	0.53 kg/cm^2

It is evident that the flow in the instrumented assemblies is distinctly greater than in the normal elements. Indeed, the instrumented assemblies were designed to ensure the rated flow with the flowmeter turbines locked. Therefore, the cross section of the opening in the orifice cup has a larger diameter then normal to compensate for the additional pressure drop. Since the turbines were free during the tests, these precaution was excessive with a result that the channel flow was almost doubled. In addition, all the other conditions being equal, the instrumented assemblies, which have a greater flow and, therefore less void, operate at a higher power than the normal symmetrical elements, also because they were fresh assemblies in Zircaloy sheaths. Therefore, the correction in the core hydraulic coefficients allowed the computer to determine the output of the instrumented assemblies, as well as their flow, correctly.

Close observation of the instrumented assembly flows also allowed an approximation check of the core leakage flow. Indeed, a slight increase in the active flow, imposed on the computer by a reduction in the leakage flow, is distinctly observed in the calculation of the flow of the instrumented assembly whose larger passages offer a lower resistance to the coolant. Therefore, in the assumption that the total recirculation flow and the total core pressure drop are correct, the difference between the measured flows and the calculated flows through the instrumented assembly constitute an indirect means to check, to a fair degree of approximation, whether the leakage flow was evaluated correctly.

4. Test results

The Research Program (*) called for performance of special tests with the reactor in eight different conditions but always at full power (506 MW). Consideration was also given to two special conditions, later discarded, with a pronounced radial power distribution at the center of the core (**).

(*) See, GEAP 4899. June 1, 1965 "Final Safeguards Report" - Garigliano Development Program" - Pages 5-28

(**)See, "Garigliano Research and Development Program - Test Procedure No. 4 - Reactor Test Sequence, Rev. 2, 28/3/66"

During the preparation of the procedures, the program was subdivided in twelve tests as it was deemed necessary to go through intermediate stages at lower power levels. Thus, the reactor would have been taken through all the various significant conditions of recirculation, subcooling and flux distribution compatible with plant safety and operability.

However, during the tests, the program had to be varied considerably. Some tests were eliminated, others were performed at reduced power or with a reduced subcooling. At any rate, none of the program variations decreased the importance of the results obtained as it was possible to obtain the desired conditions of streched operation by giving up those of minor interest. The necessity of modifying the tests stemmed mainly from the basic philosophy of complying first and above all with the safety limits and regulations.

As it is known, the purpose of the special tests in the Research Program was to bring the reactor, through subsequent steps, to a high void content. This aim was achieved and an average void content at least of 48% was obtained in the core at a power level below the rated value since the recirculation flow in the test conditions was less than anticipated.

Before starting the tests proper it was necessary to modify the basic control rod configuration because the normal operating pattern ill fitted the subsequent patterns required during the tests. The new configuration was also dictated by the requirement of extracting the control rod F6 to notch 15 so that it would be in the best position for the oscillation tests.

After a first attempt of stability test (Test 1) with the central rod withdrawn to notch 10, the rod was definitely withdrawn to position 15 as the first test gave insignificant results due to the low ratio between the neutron flux signal and background noise.

4.1 Test 1-A - Steady-state operation

The first test, called Test 1-A aimed at establishing the reference conditions for the following tests. Only 98.4% of the full power rating was reached because the 20th-stage wheel of the turbine was out and the turbine could not take the rated steam flow The conditions established during

Test 1-A are summarized in Table 4-I. There are slight differences in respect of rated operating conditions. The primary steam flow is 98.6% of rated, and the secondary steam is 86.5%. The total recirculation flow is 97% of the design value because of the presence of traces of crud residues on the fuel; this was also confirmed by the core pressure drop which was 0.53 kg/cm^2 versus 0.37 kg/cm^2 of the clean elements.

The strongest fuel element, that is the one with the highest radial power factor, was the instrumented assembly 55-16 as can be seen in Fig. 3. For this assembly a radial power factor of 1.346 was calculated versus a factor of approximately 1.16 of the symmetrical elements. Although it was controlled just as the symmetrical elements (one control rod fully withdrawn, the other withdrawn by 75%), this instrumented assembly was expected to give a greater output. Indeed, as the other three instrumented assemblies, it was fresh and had a Zircaloy, instead of stainless steel, sheath; however, what is more important, it was cooled by a greater flow than the average because of the different hydraulic characteristics at its inlet.

The average void fraction in the core, that is, the voids averaged over all the water contained inside the channels, was 16.5%. This reference value is important for the comparison with the values obtained in the severer tests.

The maximum steam quality was observed at the channel outlet of element 68-11: 12% by weight equal to 62% in void content. This element had the highest power factor after the instrumented assemblies and therefore it had the highest steam quality because it was cooled by a much smaller flow of water than the instrumented assemblies.

The maximum local heat flux (68 W/cm^2) was found at the center of the core and specifically in correspondence of the north corner rod of the instrumented assembly 61-12. The minimum critical heat flux ratio reached a value of 2.68 thus deviating appreciably from the usual value for the Garigliano station under the normal conditions and at the beginning of the cycle. This was due to the higher than usual local heat fluxes as a result of the particular rod configuration adopted in the Research Program. The element presenting the MCHFR was 69-08 with a channel flow of 39.3 tons/hr.

For comparison purposes, Table 4-I also shows the percentage of withdrawn rods expressed as a percent ratio of the number of withdrawn notches to the total number of notches.

Figs 4, 5 and 6 also show the normalized axial neutron flux distributions obtained from the traces of the movable in-core chambers 106, 109 and 112 (as can be seen in Fig. 3) and through the physical code of the digital computer. It will be noted that in steady-state conditions the flux has a tendency to shift upwards. This is due to the strong control exerted by the rods in the lower part, since the core is at the beginning of the cycle, and to the low void content in the higher part of the reactor.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - I

Results of Test No. 1-A

Reactor Power (MW)	497.6
Gross Generator Power (MW)	154.5
Primary Steam Flow Rate (tons/hr)	690
Secondary Steam Flow Rate (tons/hr)	187
Feedwater Temperature (°C)	187.6
Steam Drum Pressure (ata)	68.2
Reactor Pressure (ata	69.6
Total Recirculation Flow Rate (tons/hr)	9720
Core Leakage Flow (tons/hr)	1312
Core Inlet Subcooling (kCal/kg)	19.3
Core Average Heat Flux (W/cm^2)	25.8
Peak Heat Flux (W/cm^2)	88
Average Core Exit Quality (%)	8.7
Maximum Channel Exit Quality (%)	12
Average Channel Void Fraction (%)	16.5
Maximum Exit Void Fraction (%)	62
Min. Critical Heat Flux Ratio at Overpower (MCHFR)	2.68
Radial Power Factor for the MCHFR Fuel Element	1.192
Axial Power Factor at the MCHFR Point	1.784
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at MCHFR Point (at Overpower) (%)	3.1
MCHFR Channel Flow Rate (tons/hr)	39.3
Average Core Exposure (MWD/MT)	5100
Percent of Withdrawn Control Rods	76.2

CENTRALE DEL GARIGLIANO

Core map symbols

Fuel element with Zircalloy channel

Instrumented fuel assembly or fresh element with Zircalloy channel

Fuel element with stainless steel channel

Fresh element with stainless steel channel

Control rod completely withdrawn

Control rod at position 15

In-core chamber location

Start-up source location

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

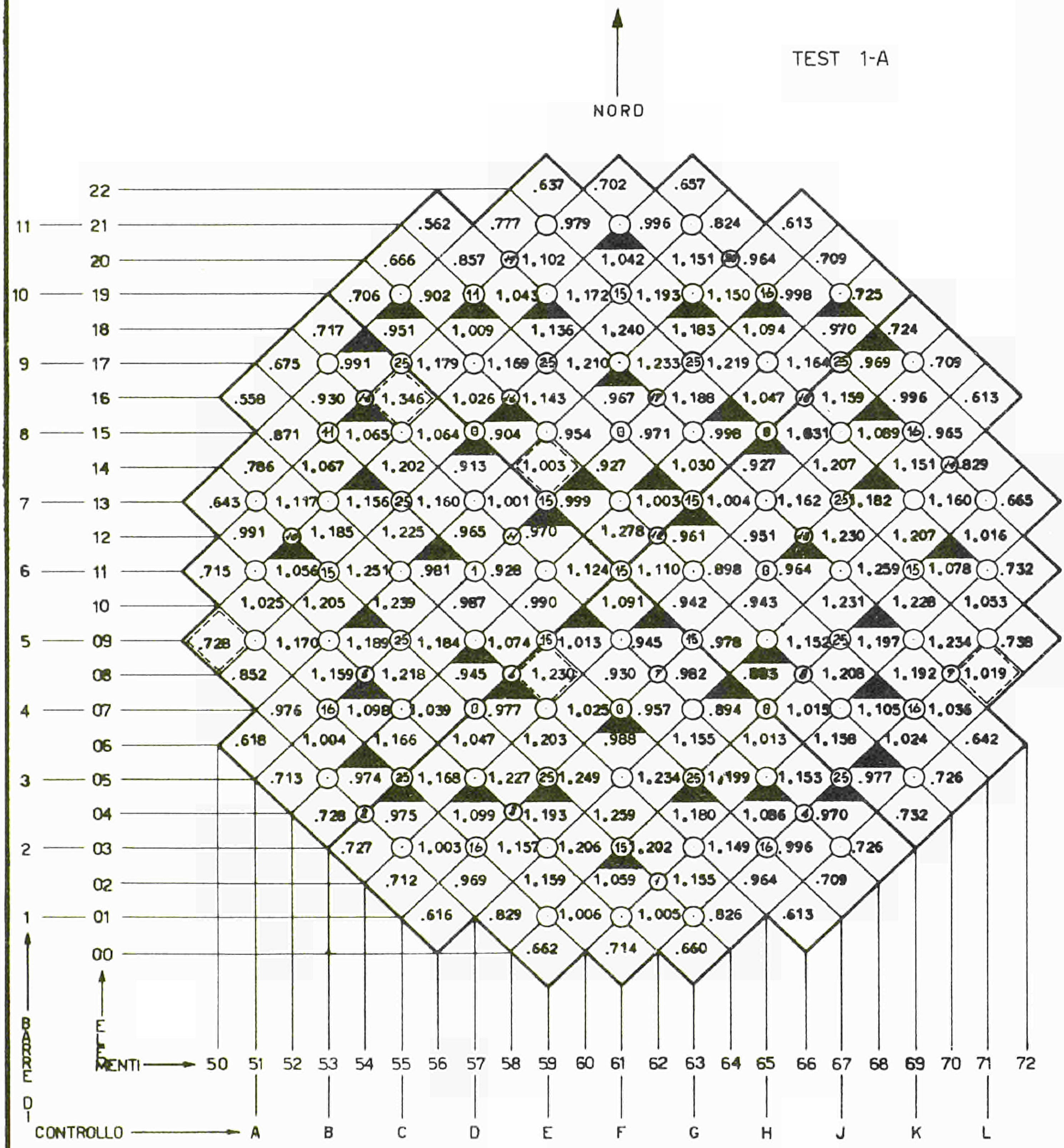

FIG. 3

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

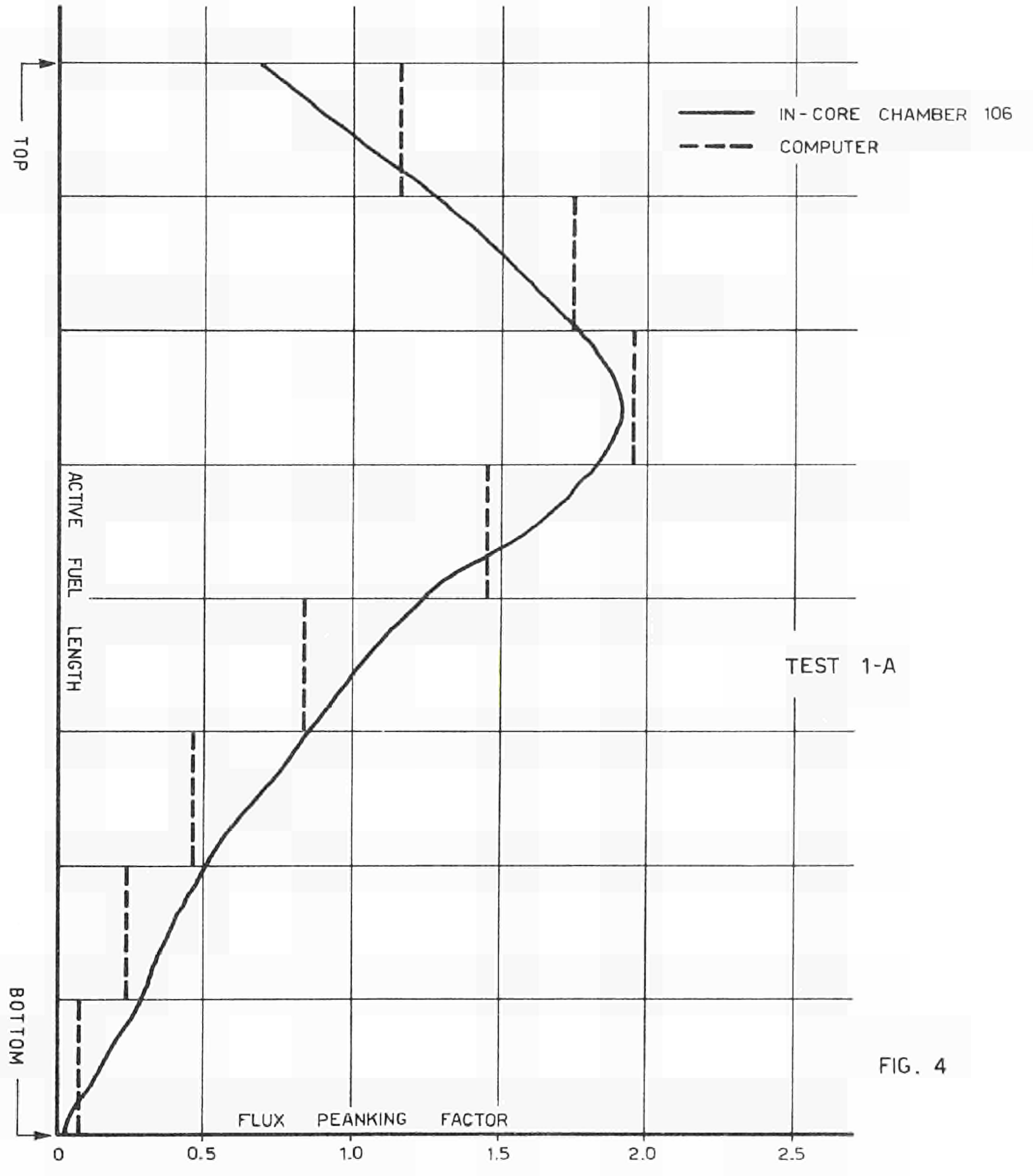

FIG. 4

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

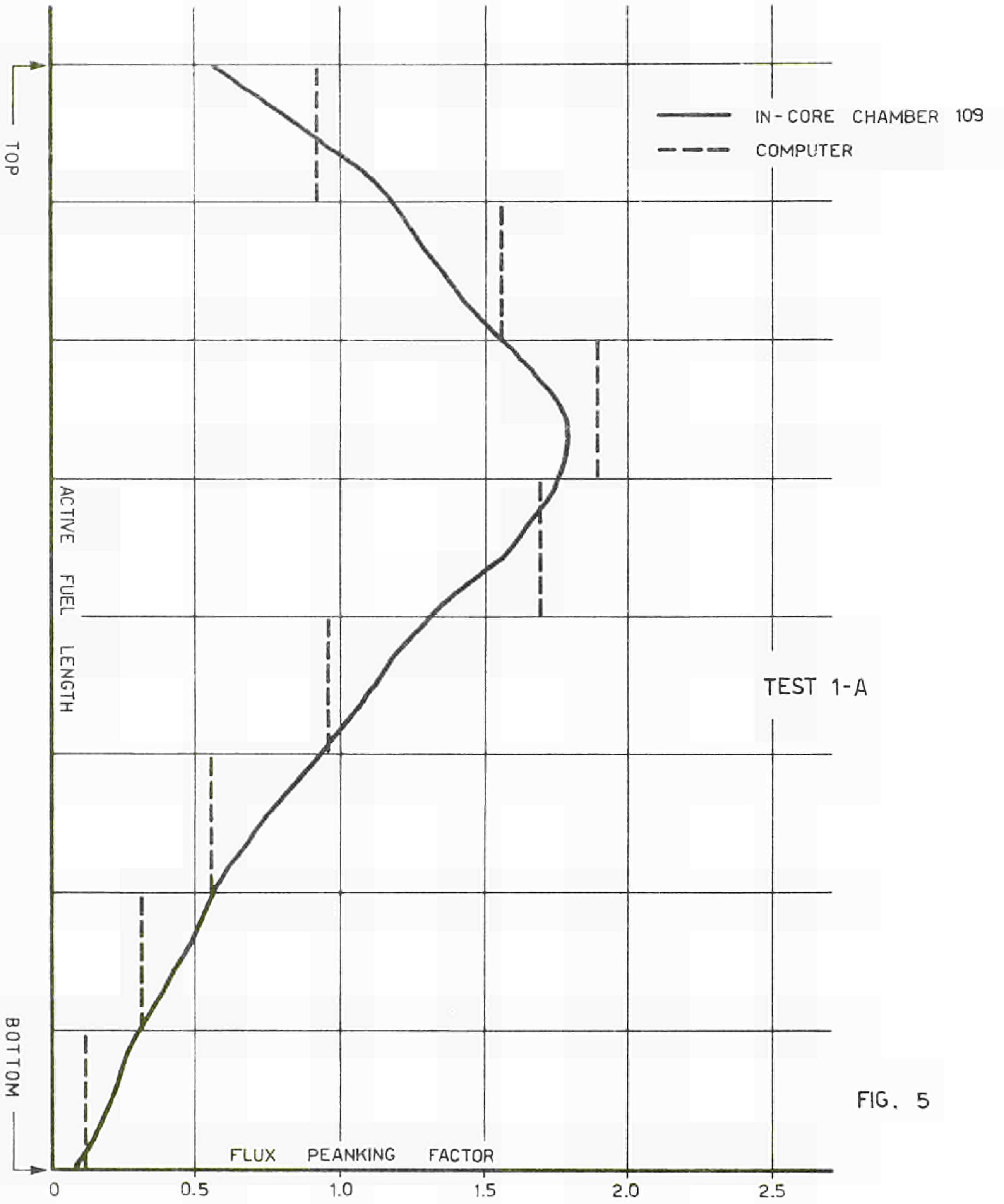

FIG. 5

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

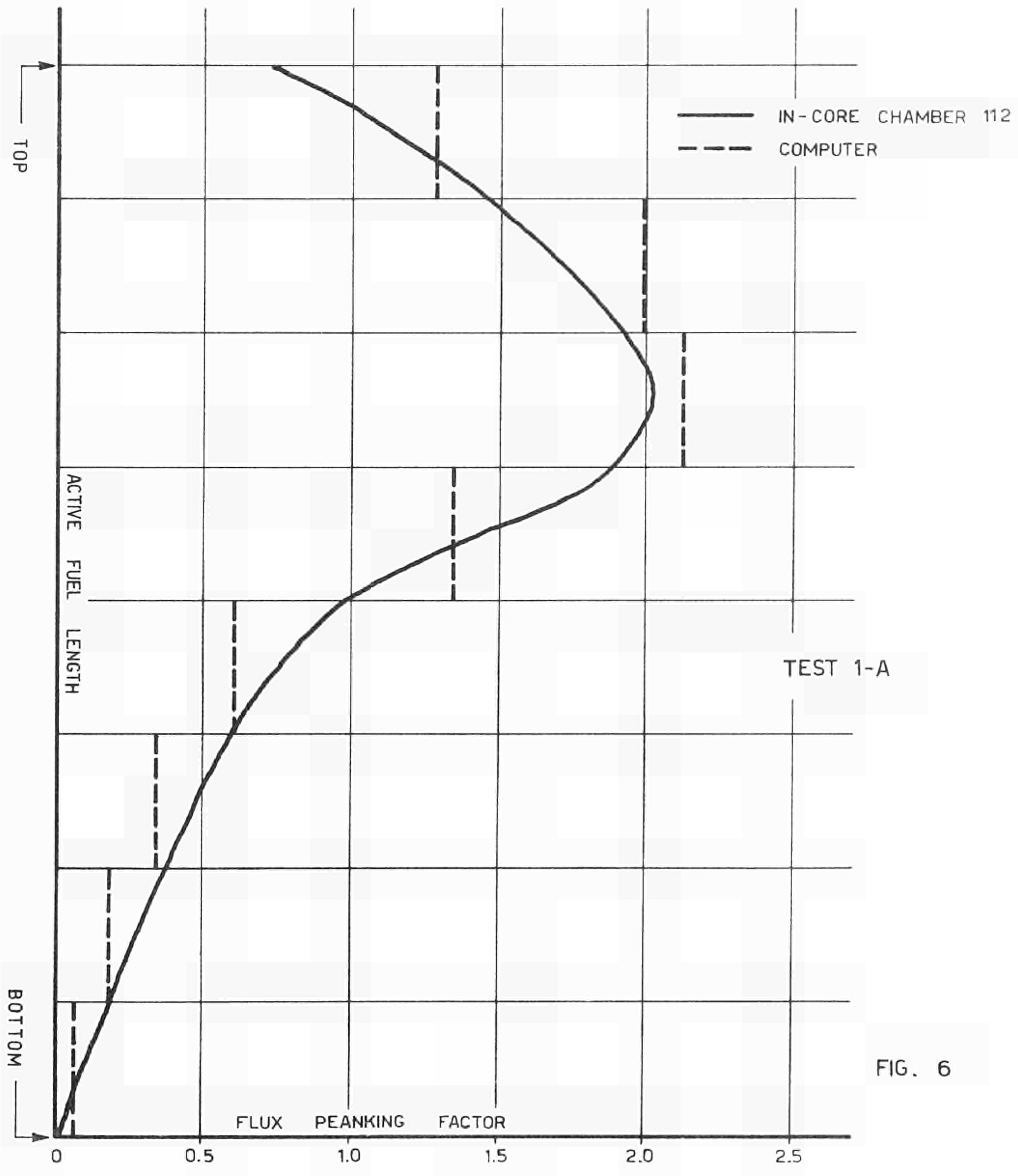

FIG. 6

4.2 Test 3 - 75% reactor power, two loops in natural circulation, maximum subcooling

As described elsewhere (*), test No. 2, which called for the reactor at rated power with one recirculating loop isolated, the other in forced circulation and the four heaters out of operation, could not be performed because of a number of inconveniences. When a power level of 470 MWe was reached under the aforesaid conditions, low-frequency power oscillations occurred as a result of the periodical admission of cold water in the one recirculation loop in operation. This was caused by poor mixing of the feedwater in the steam drum. Besides, the turbine extraction lines were subjected to considerable vibration because, with the heaters out of service, the spills had to be bypassed directly to the main condenser to ensure extraction of the moisture from the turbine. The arrival of an excessive amount of fluid in the flash tank immediately upstream from the condenser caused intolerable vibration.

Thus, we went on to test 3, tripping both recirculation pumps and reaching 75% of rated power with the maximum possible amount of secondary steam. Thus the maximum subcooling required for the test (31.4 kCal/kg) was attained.

Table 4-II summarizes the most important data of test 3 in which the average void content attained was 24%, equal to 145% of the steady-state value.

The total coolant flow in natural circulation reached 42% of the steady-state value, against the 55% anticipated through extrapolation of the flow rate versus power curves obtained from the data of the first plant startup test (**).

As in the previous test, the instrumented assembly 55-16 proved to be the strongest with a radial power factor of 1.342 (Fig. 7).

(*) See "Garigliano Nuclear Power Station Research Program - Quarterly Report No. 9" - ENEL, July 1, 1966, Pages 17-19.

(**) See GECR-4736 "SENN Startup Hydraulic and Thermal Performance" - September 1964, Page 12, Fig. 7.

The maximum thermal flux localized in the west corner rod of element 68-11 reached 55.6 W/cm^2 and the MCHFR at overpower remained above 3.42 (element 68-11).

Figures 8, 9 and 10 show the axial neutron flux distributions in three points of the core (Fig. 7). It will be noted that the power tends to shift downwards owing to the greater void content in the higher part of the reactor and to the lesser control exerted by the rods in the lower part of the reactor. This is more evident in the peripheral area of the core near the movable in-core chamber 109 (Fig. 9).

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - II

Results of Test No. 3

Reactor Power (MW)	386.8
Gross Generator Power (MW)	118.1
Primary Steam Flow Rate (tons/hr)	523
Secondary Steam Flow Rate (tons/hr)	152
Feedwater Temperature (°C)	178
Steam Drum Pressure (ata)	68.8
Reactor Pressure (ata	69.5
Total Recirculation Flow Rate (tons/hr)	4575
Core Leakage Flow (tons/hr)	657
Core Inlet Subcooling (kCal/kg)	31.4
Core Average Heat Flux (W/cm^2)	20.1
Peak Heat Flux (W/cm^2)	59.6
Average Core Exit Quality (%)	13
Maximum Channel Exit Quality (%)	20
Average Channel Void Fraction (%)	24
Maximum Exit Void Fraction (%)	70
Min. Critical Heat Flux Ratio at Overpower (MCHFR)	.42
Radial Power Factor for the MCHFR Fuel Element	1.284
Axial Power Factor at the MCHFR Point	1.472
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower)(%)	4
MCHFR Channel Flow Rate (tons/hr)	19.8
Average Core Exposure (MWD/MT)	5120
Percent of Withdrawn Control Rods	77.8

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

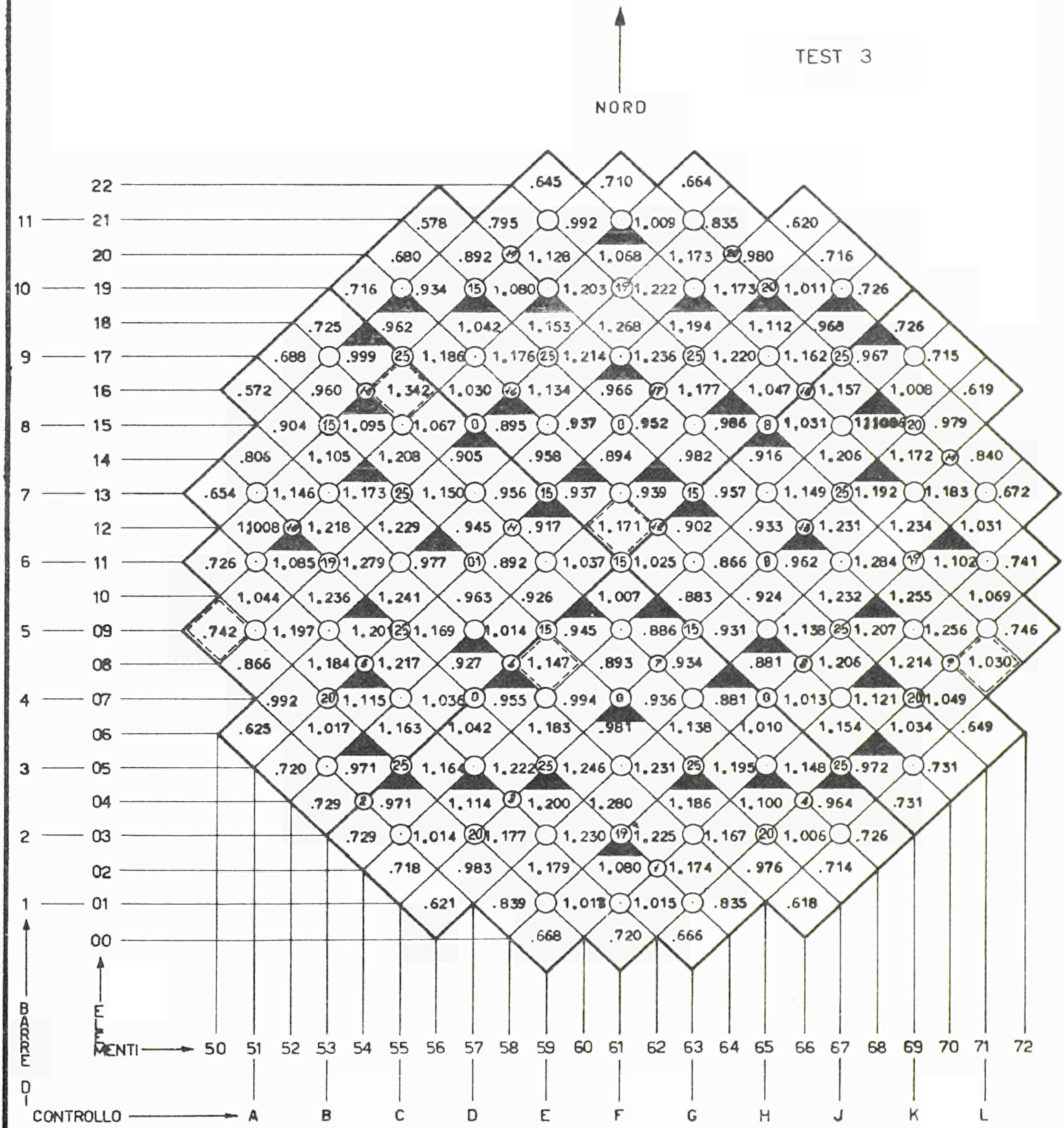

FIG. 7

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

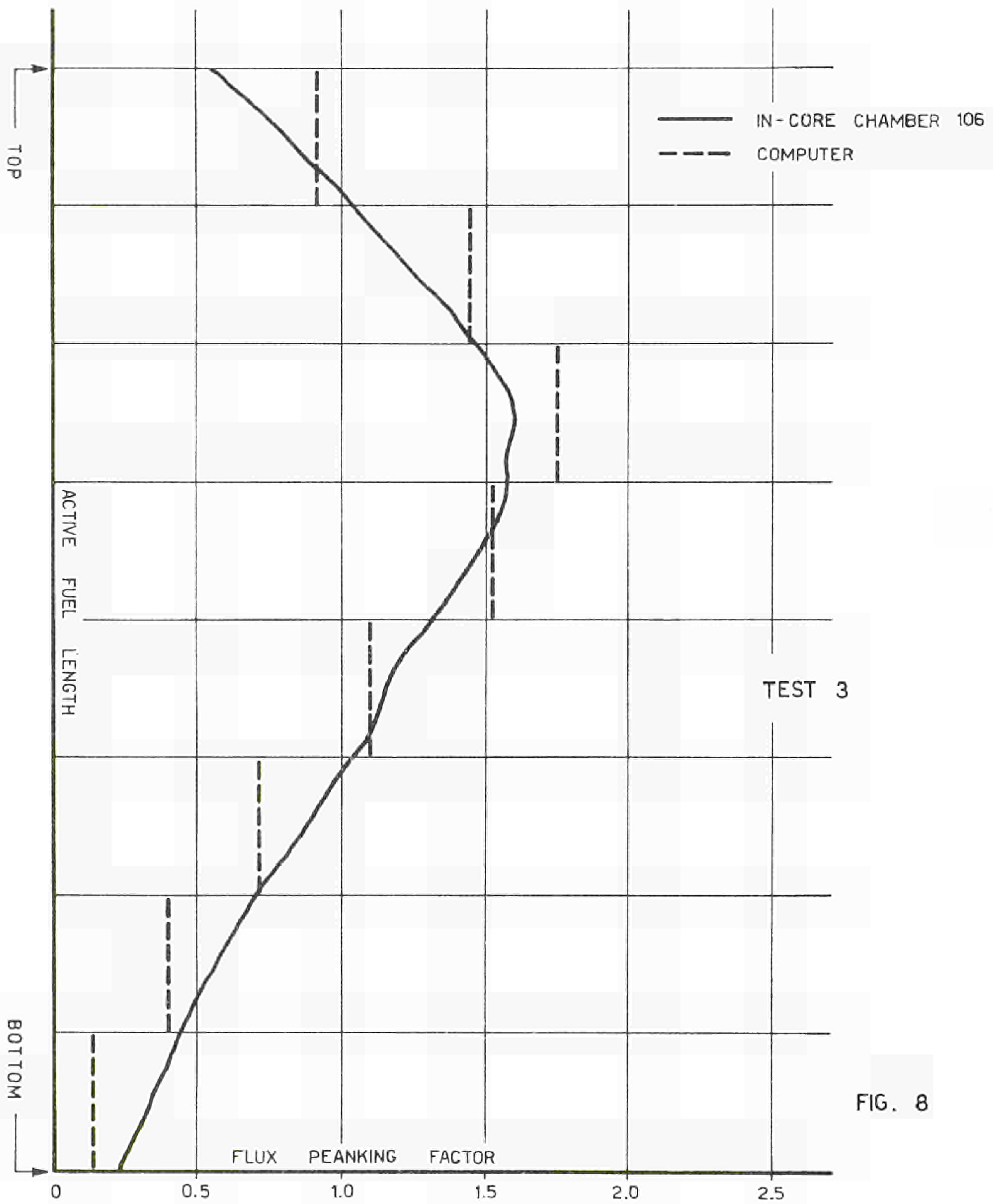

FIG. 8

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

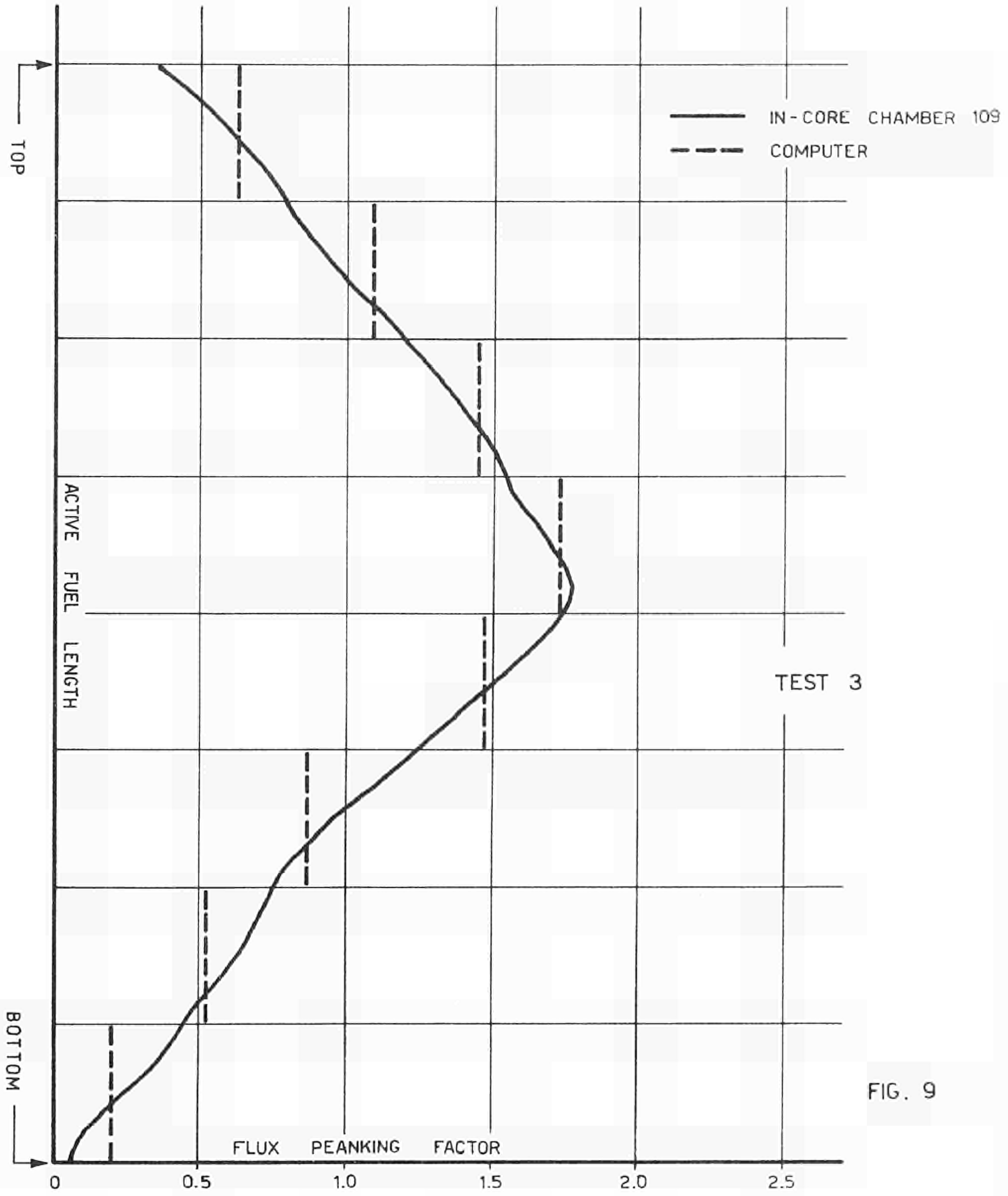

FIG. 9

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

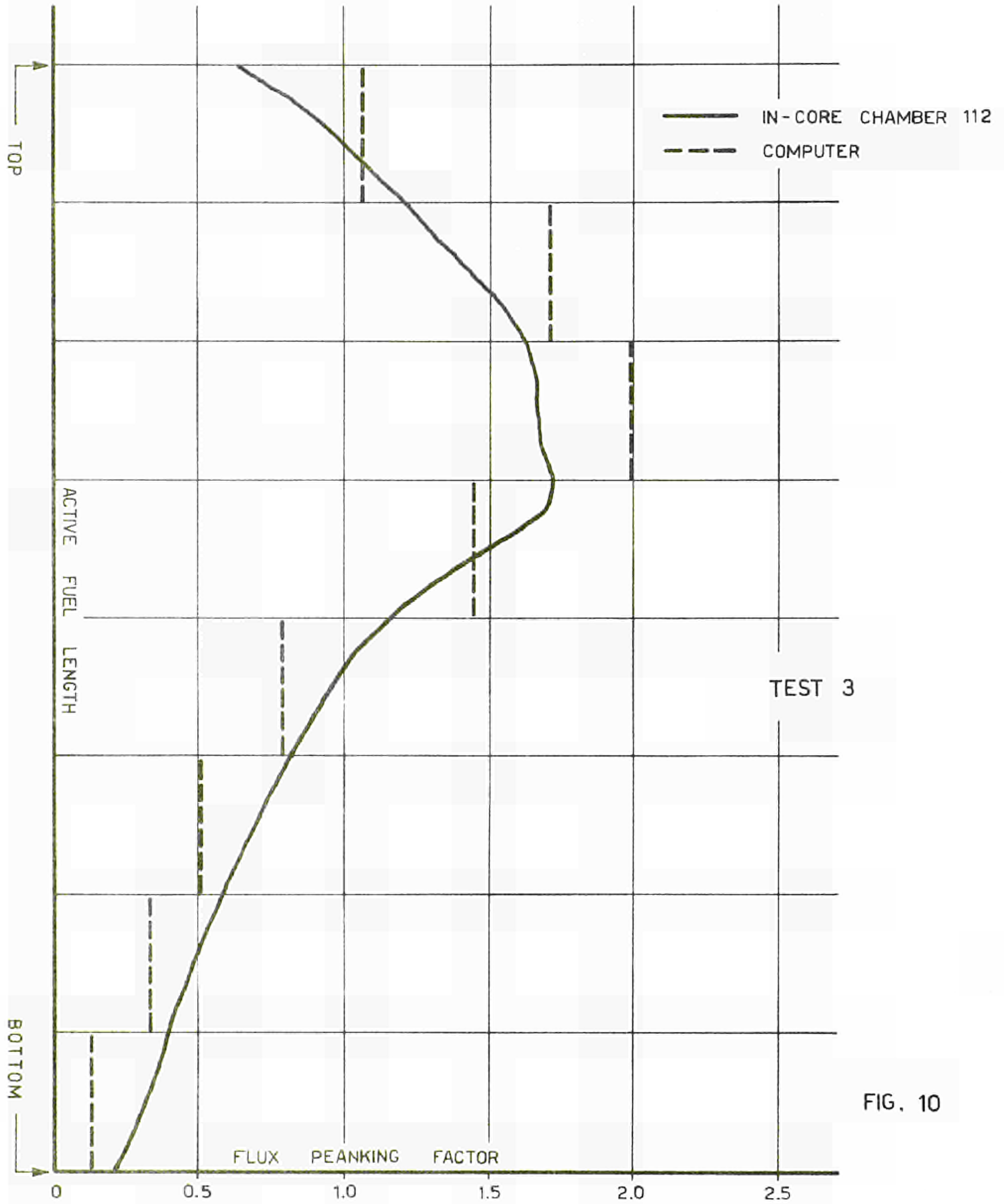

FIG. 10

4.3 Test 5 - Rated power, two loops in natural circulation, maximum subcooling

To go to the test conditions preceding test 5, the reactor power was raised from 75% to 94% by extracting a considerable number of control rods. It will suffice to say that from test 1-A to test 3 the percentage of notches withdrawn increased by 1.6%, whereas to go from test 3 to test 5, the increase was 8.2%.

All the power increase was obtained by means of primary steam which rose from 75% to 99% of the flow rate (700 tons/hr); therefore, the void content in the core increased considerably reaching 236% of the steady-state value.

The total coolant flow in natural circulation was increased over that of the previous test because of the great increase in voids. It reached 47.6% of the value in forced circulation with a 5.6% increment over the flow rate established in test 3. Also in this case, the recirculation flow was smaller than anticipated and thus a higher void content than expected was reached.

The strongest element - the one with the maximum outlet steam quality - was 55-14 (Fig. 11) with a power factor of 1.265.

The maximum heat flux and the MCHFR were localized in the south corner rod of the instrumented assembly 55-16. This demonstrated that also in high void condition the MCHFR remained independent from the steam quality and dependent only on the thermal heat flux peak. In other words, although the steam quality in the higher part of the core was high, the MCHFR was not small in this region because the heat flux peak was located considerably low (at approximately 3/8 of the core height). This can easily be seen in Figs 12 and 14 in which the normalized traces of the movable in-core chambers give a very good idea of how the neutron flux distribution was depressed downwards due to the effect of the high void content in the higher part of the core. In brief, the MCHFR, in going from the forced circulation steady-state condition to the natural circulation steady-state condition at the same power level dropped only from 2.68 to 2.56.

This result is undoubtedly excellent and it is mainly due to the good axial power distribution.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - III

Results of Test No. 5

Reactor Power (MW)	477
Gross Generator Power (MW)	152
Primary Steam Flow Rate (tons/hr)	693
Secondary Steam Flow Rate (tons/hr)	160
Feedwater Temperature (°C)	187.3
Steam Drum Pressure (ata)	68.9
Reactor Pressure (ata	69.3
Total Recirculation Flow Rate (tons/hr)	4625
Core Leakage Flow (tons/hr)	462
Core Inlet Subcooling (kCal/kg)	33.3
Core Average Heat Flux (W/cm^2)	24.8
Peak Heat Flux (W/cm^2)	82.8
Average Core Exit Quality	18
Maximum Channel Exit Quality	25.8
Average Channel Void Fraction	39
Maximum Exit Void Fraction	74.6
Min. Critical Heat Flux Ratio at Overpower (MCHFR)	2.56
Radial Power Factor for the MCHFR Fuel Element	1.288
Axial Power Factor at the MCHFR Point	1.721
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower)(%)	8.4
MCHFR Channel Flow Rate (tons/hr)	28.9
Average Core Exposure (MWD/MT)	5150
Percent of Withdrawn Control Rods	86

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

TEST 5

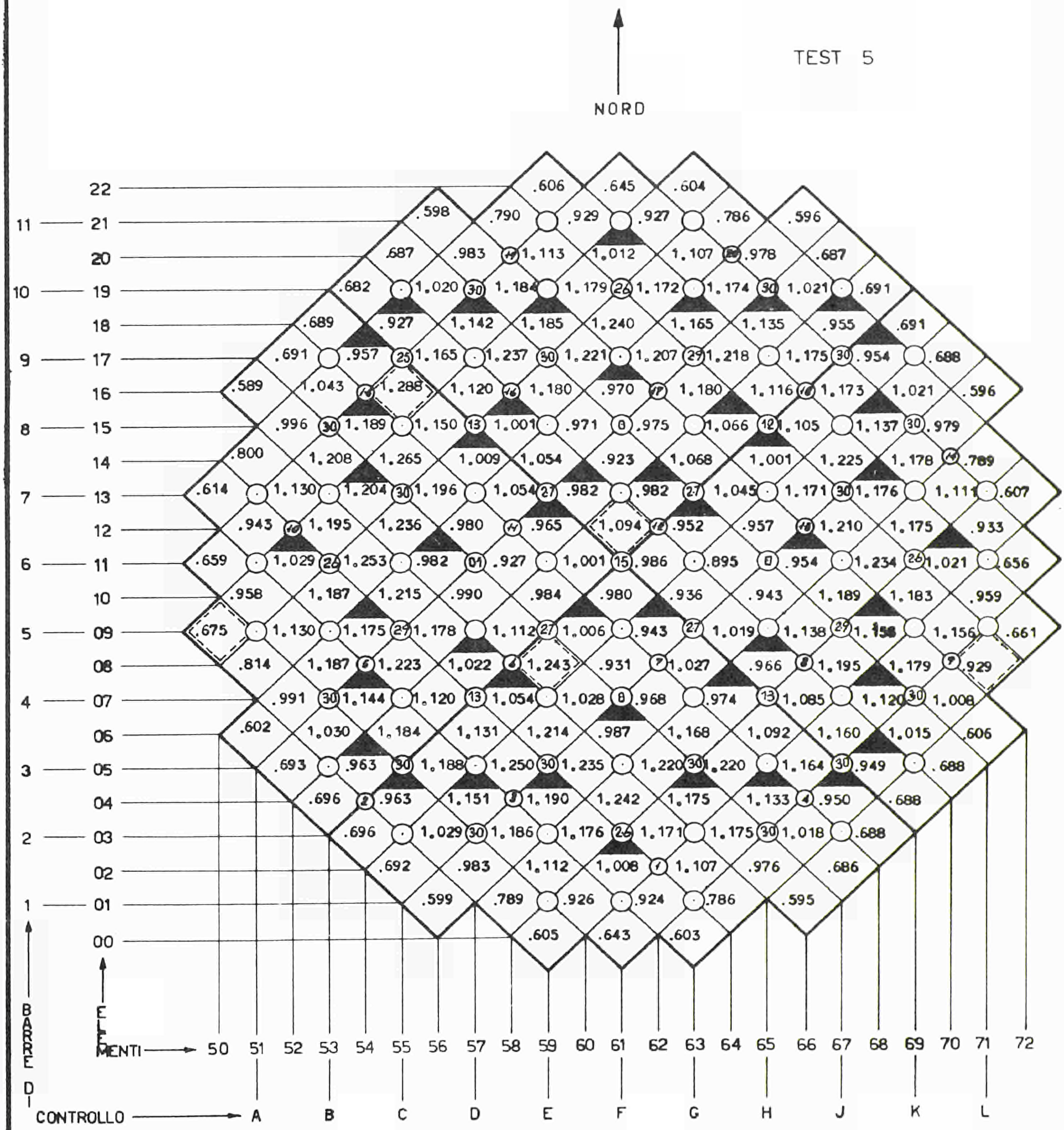

FIG. 11

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

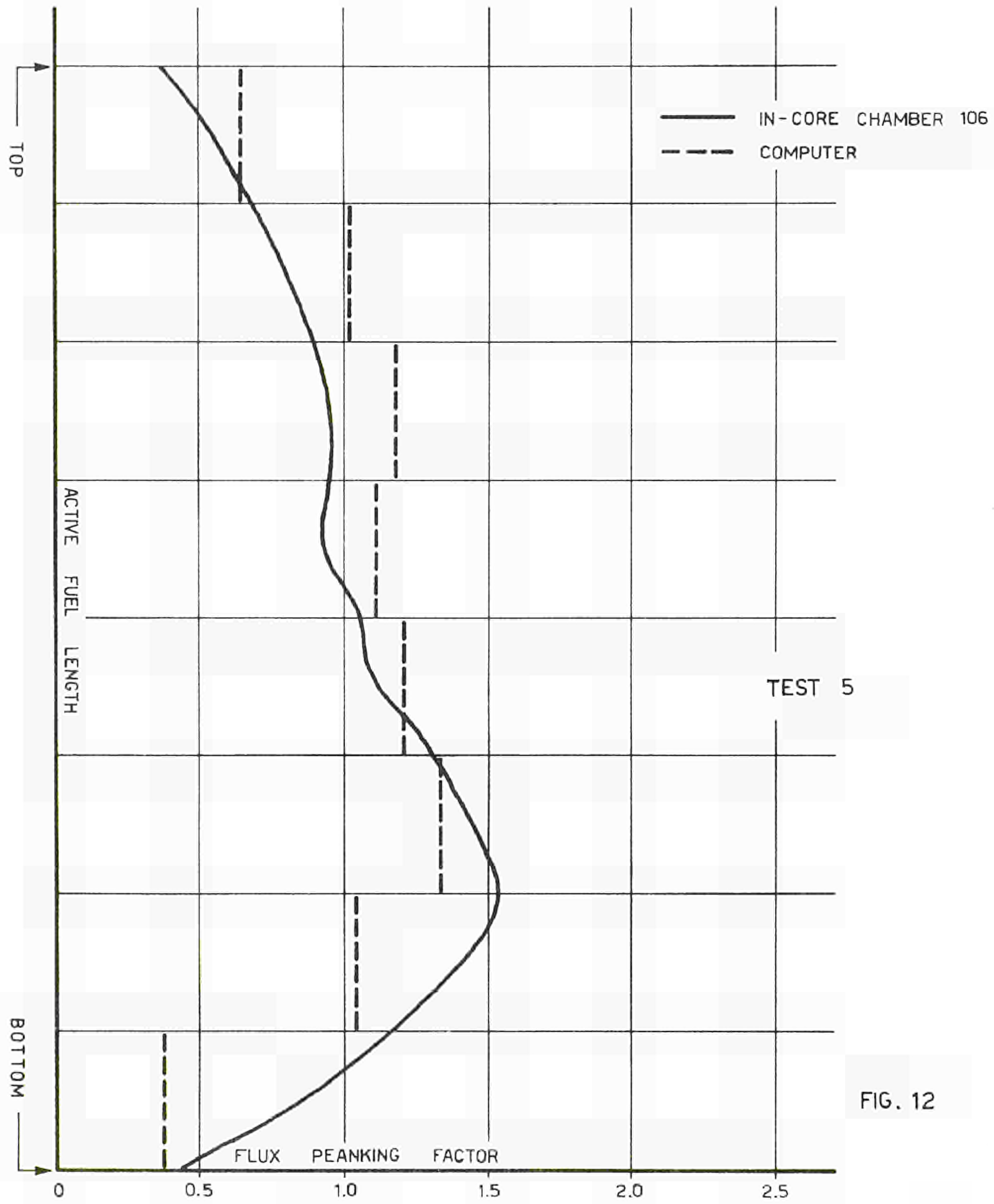

FIG. 12

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

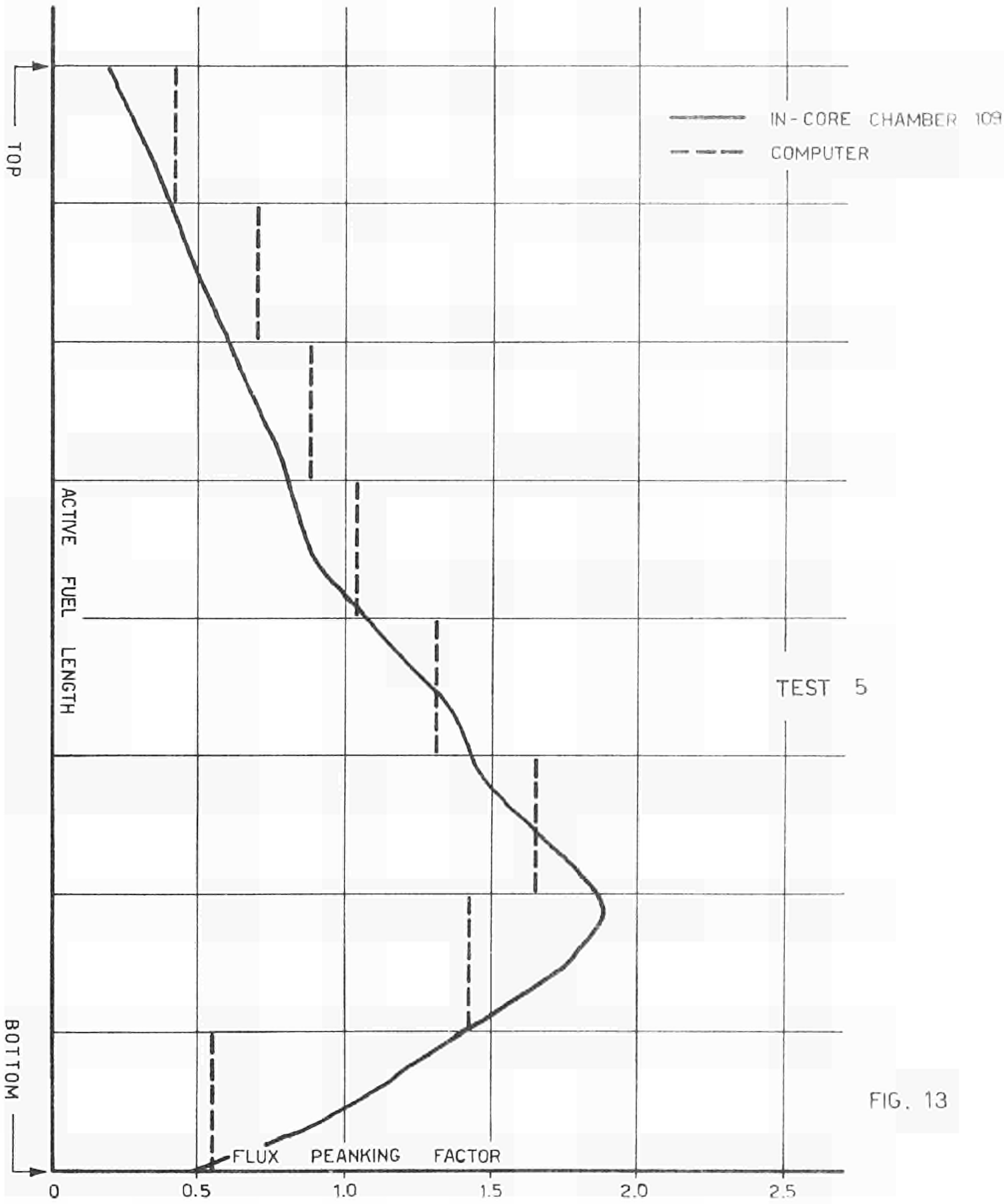

FIG. 13

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

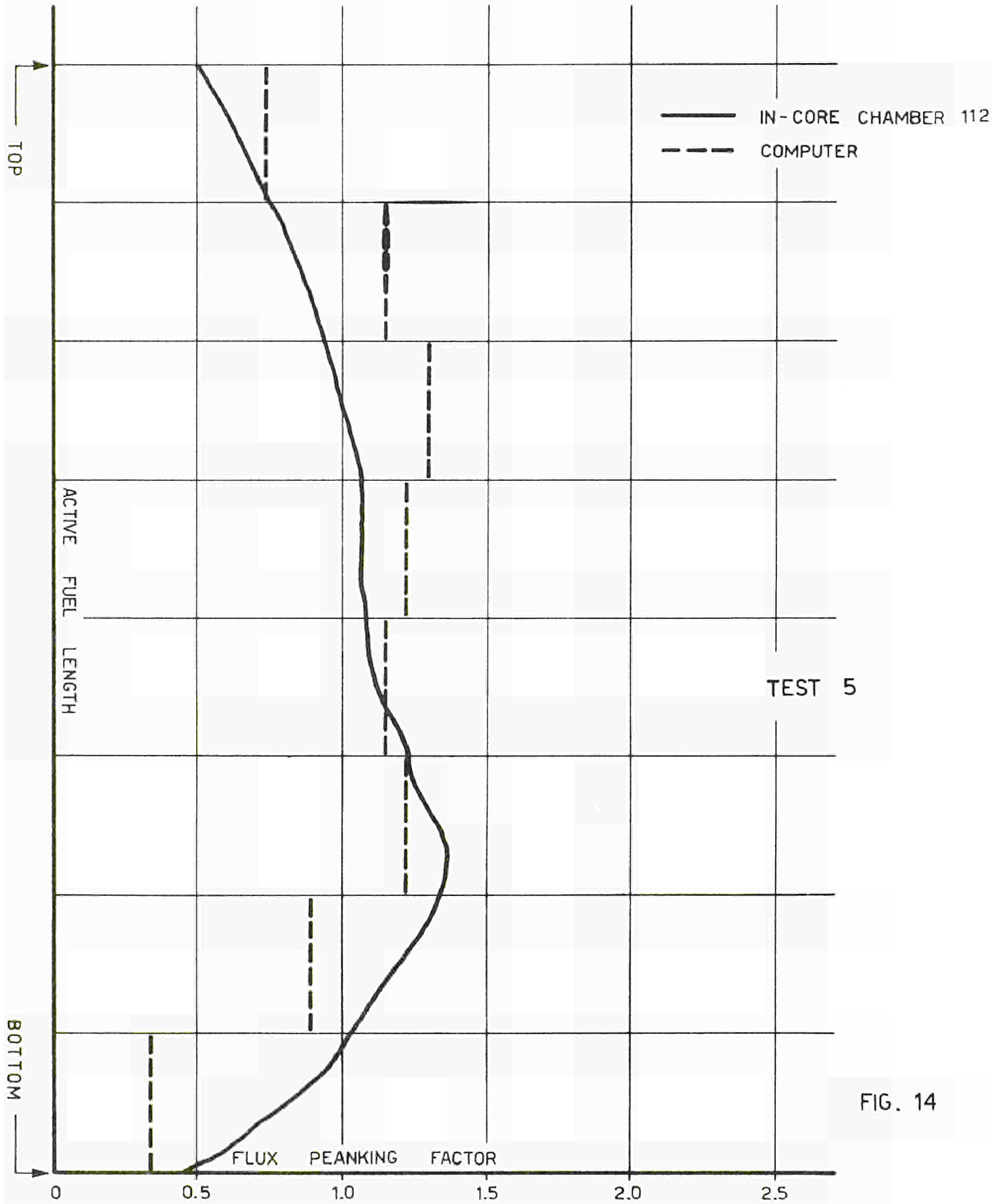

FIG. 14

4.4 Test 6 - Reactor power 75%, two loops in natural circulation, minimum subcooling

This test can be compared directly with test N° 3 as it was performed at the same power level and with both loops in natural circulation; subcooling at the core inlet differentiated the two tests as it was 31.4 kCal/kg in test 3, and 18.2 kCal/kg in test 6. This was obtained by reducing the secondary steam flow to zero and then making up for it with primary steam. Since this test followed the full power test No. 5, closing the secondary loop lowered the power to a level just below 75%; it was therefore necessary to withdraw the control rods slightly to bring back the reactor to the same power level as in test 3.

The average void content in the core reached 42% against 24% of test 3; indeed, the primary steam flow rate, for the same power level, changed from 523 tons/hr in test 3 to 653 tons/hr in test 6.

The maximum heat flux was localized in the north corner rod of element 55-16, whereas the strongest element was element 55-14 (Fig. 15).

The MCHFR, equal to 3.46, was found at the heat flux peak again in element 55-16. Also in this case the axial flux distribution was shifted downwards and actually the MCHFR was found at about 2/8 of the core height. Figures 16, 17 and 18 show very clearly, when compared with Figures 8, 9 and 10 of test 3, how the different subcooling affected the axial neutron flux distribution. Indeed, since approximately 150 tons/hr of secondary steam had to be made up with primary steam, the control rods had to be withdrawn considerably; this caused a significant rise in the void content at the core outlet. Therefore, the lesser control by the rods in the lower part of the reactor and the higher void content at the top of the core contributed to depress the axial flux distribution at the top and enhanced it at the bottom.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - IV

Results of Test No. 6

Reactor Power (MW)	379.4
Gross Generator Power (MW)	120.9
Primary Steam Flow Rate (tons/hr)	653
Secondary Steam Flow Rate (tons/hr)	0
Feedwater Temperature (°C)	175.1
Steam Drum Pressure (ata)	68.9
Reactor Pressure (ata	69.5
Total Recirculation Flow Rate (tons/hr)	1810
Core Leakage Flow (tons/hr)	481
Core Inlet Subcooling (kCal/kg)	18.2
Core Average Heat Flux (W/cm^2)	19.7
Peak Heat Flux (W/cm^2)	69.6
Average Core Exit Quality (%)	15
Maximum Channel Exit Quality (%)	19.6
Average Channel Void Fraction (%)	42
Maximum Exit Void Fraction (%)	70.4
Min. Critical Heat Flux Ratio at Overpower (MCHFR)	3.46
Radial Power Factor for the MCHFR Fuel Element	1.134
Axial Power Factor at the MCHFR Point	1.768
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower) (%)	4.1
MCHFR Channel Flow Rate (tons/hr)	20.6
Average Core Exposure (MWD/MT)	5160
Percent of Withdrawn Control Rods	88.4

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

TEST 6

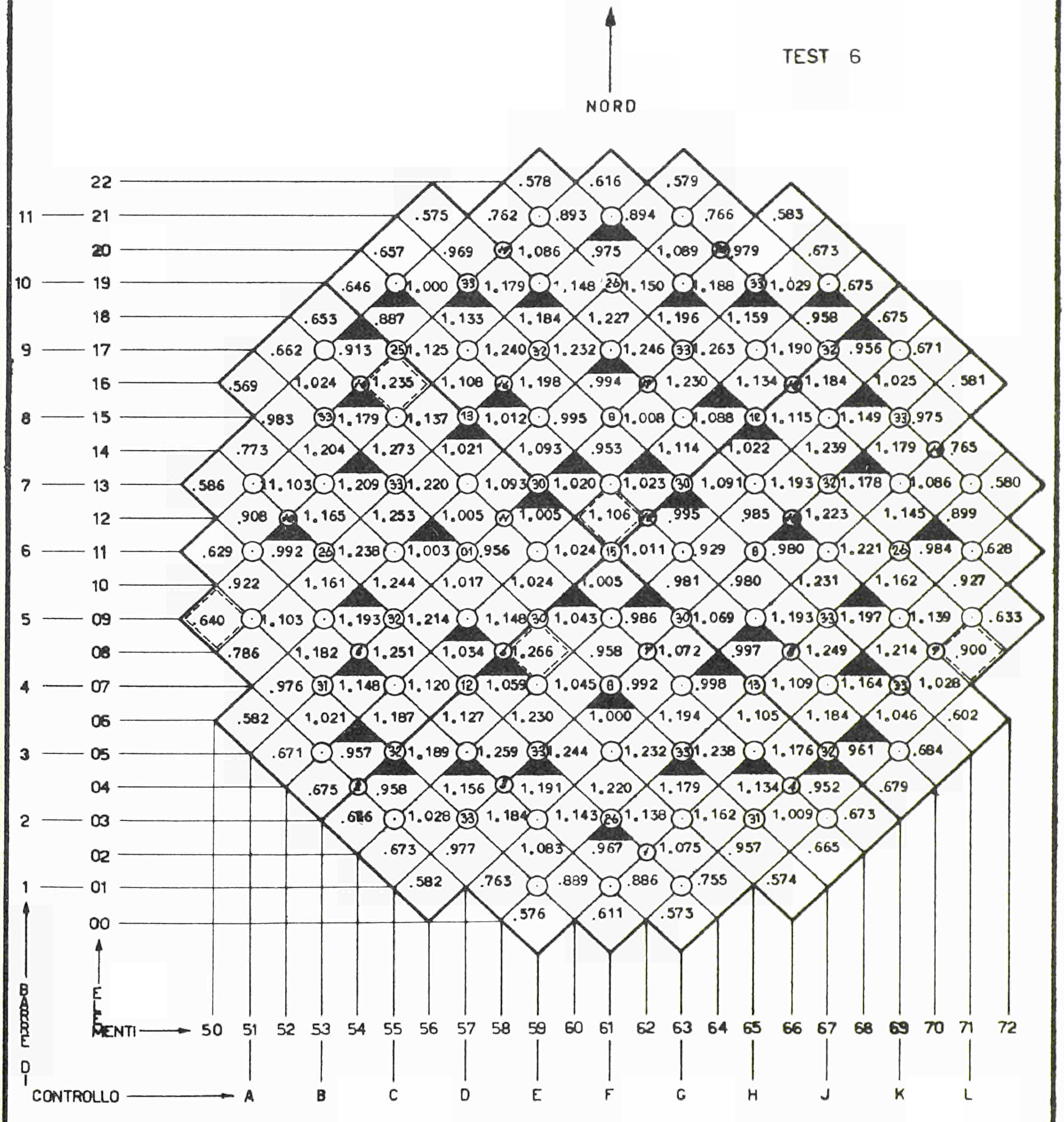

FIG. 15

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

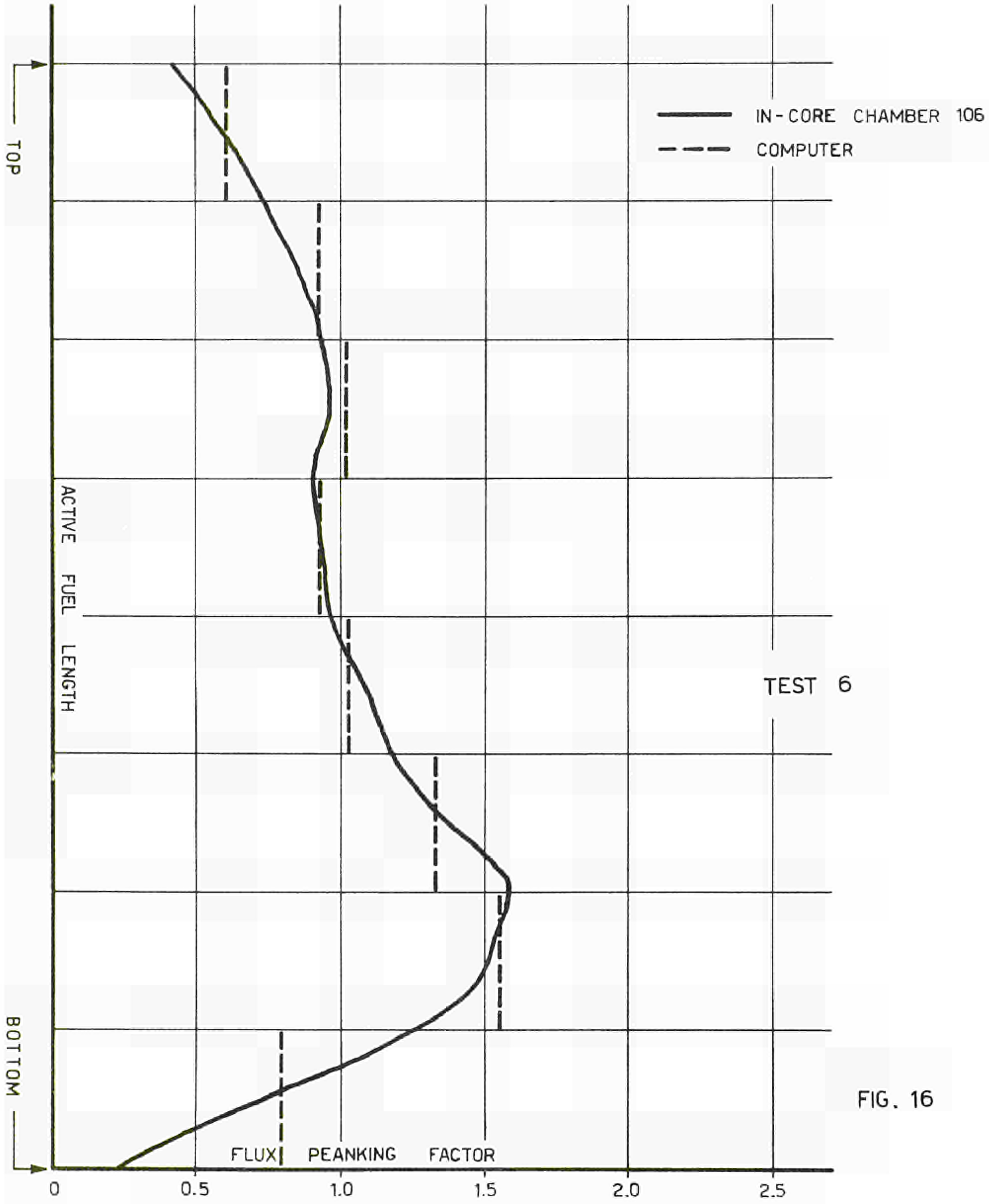

FIG. 16

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

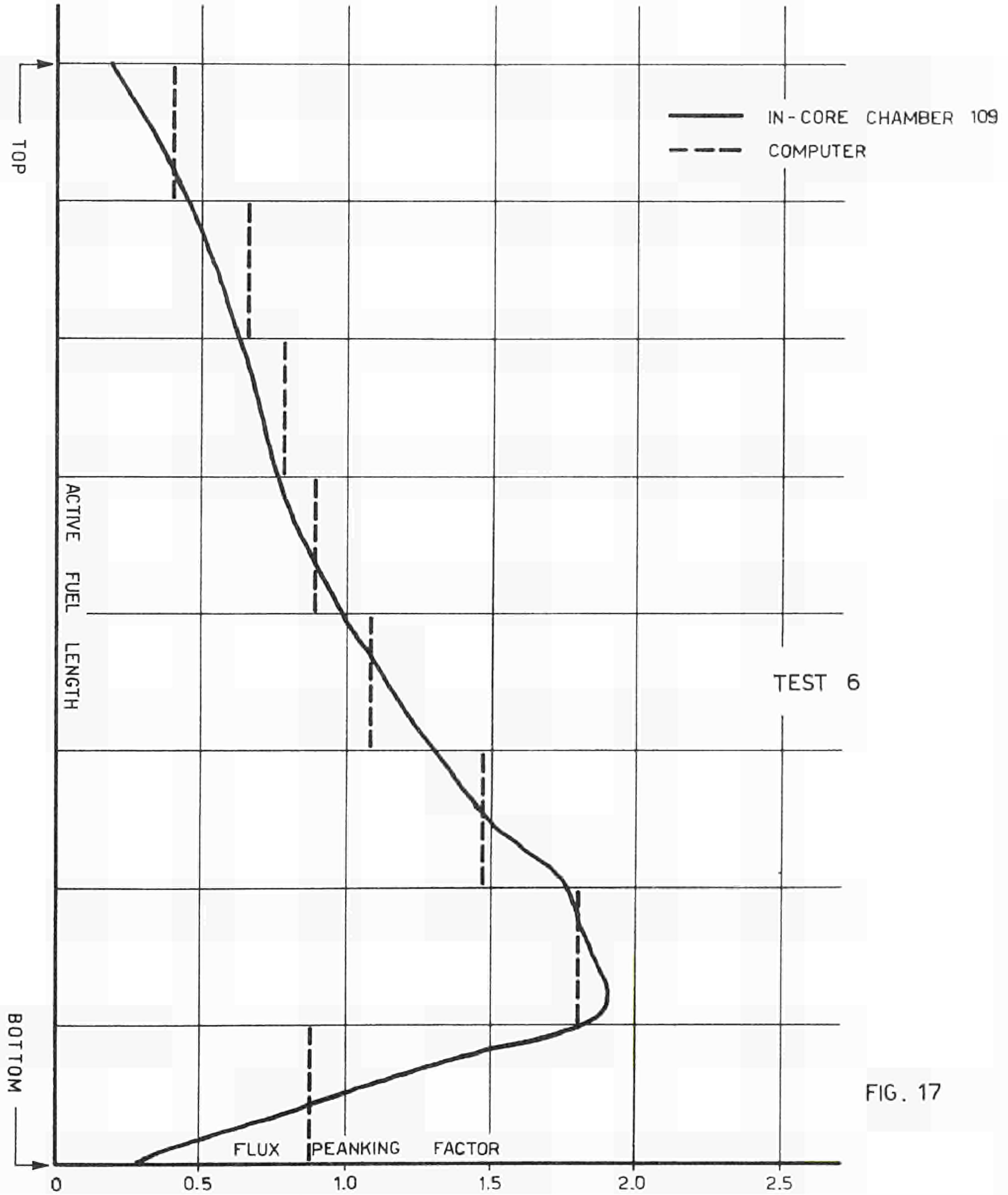

FIG. 17

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

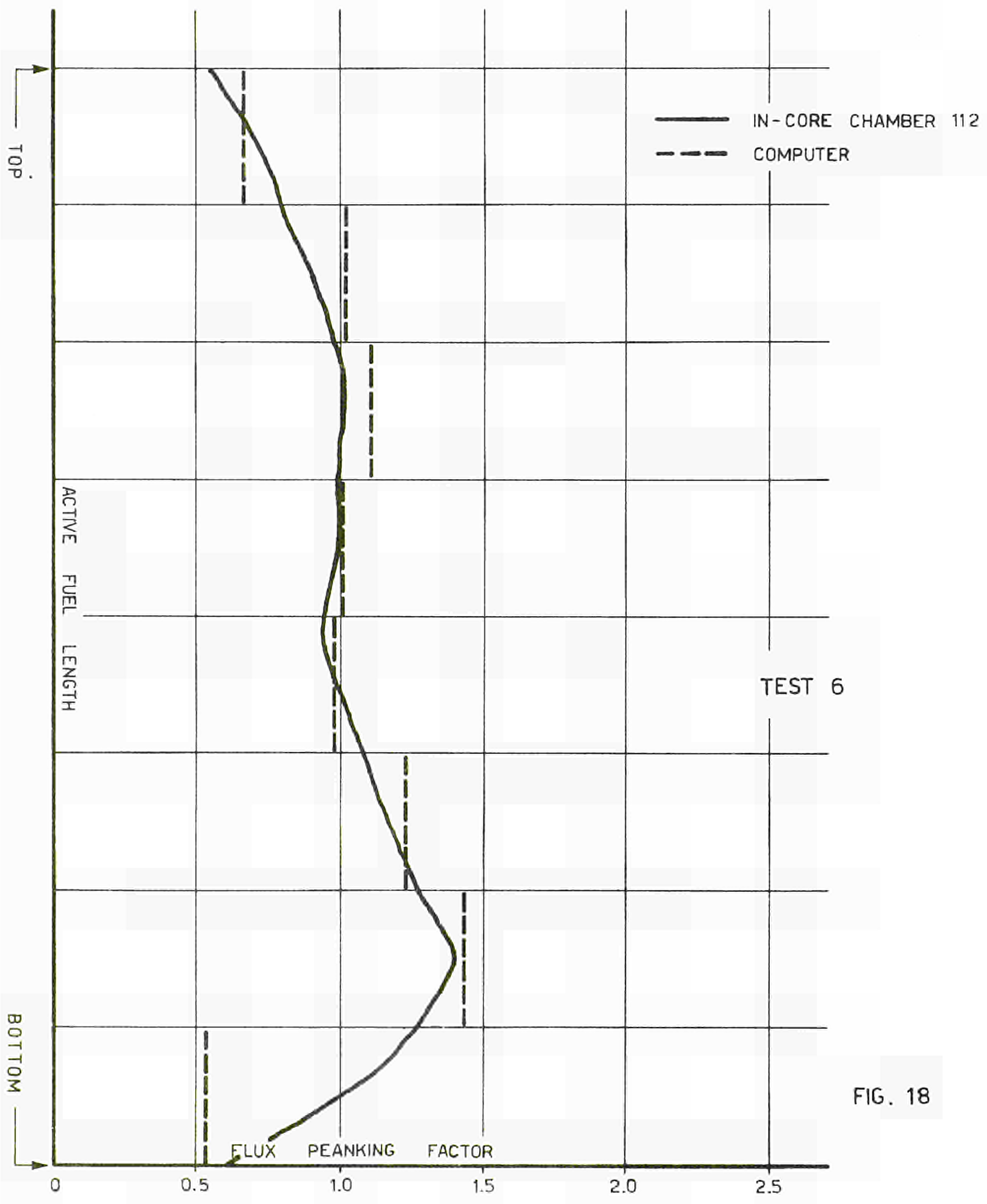

FIG. 18

4.5 Test 7 - Reactor power 50%, one loop in natural circulation, minimum subcooling

Test No. 7, the first of three with only one recirculation loop in operation and the other out of service, represented a low power intermediate stage to determine the feasibility of creating the maximum void content in the subsequent tests. From the previous test we passed on to test No. 7 by simply cutting out one recirculation loop, which reduced the reactor power to 50% of rated. Since no secondary steam was generated, subcooling was the minimum possible (22.2 kCal/kg) even though higher than the value in test No. 6 (18.2 kCal/kg). This difference was due to the fact that in test No. 6 the influence of the feedwater on the recirculating water was less significant than in test No. 7. In the previous case, the feedwater temperature was 175.1°C and its flow represented 13.6% of the recirculating flow; in test No. 7 the temperature dropped to 159.5°C and the feedwater flow was 15.6% of the recirculating flow.

In this test an average void content of 45% was reached in the core, equal to 273% of the value at rated conditions.

In view of the low power level, all the critical parameters remained well below the limits, so that from the standpoint of core thermal performance this test is of limited interest.

The maximum heat flux (43.2 W/cm^2) and the MCHFR (4.5) were localized at approximately 2/8 of the height from the bottom of the north rod in element 65-16 (Fig. 19). The highest output was generated by element 58-05 with a radial factor of 1.265.

Figures 20, 21 and 22 show how the axial flux peak has not shifted significantly along the core height from the previous test. However, in view of the higher void content it is slightly enhanced whereas the flux is depressed at the top of the reactor.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - V

Results of Test No. 7

Reactor Power (MW)	260.5
Gross Generator Power (MW)	79
Primary Steam Flow Rate (tons/hr)	434
Secondary Steam Flow Rate (tons/hr)	0
Feedwater Temperature (°C)	159.5
Steam Drum Pressure (ata)	68.9
Reactor Pressure (ata	69.4
Total Recirculation Flow Rate (tons/hr)	2775
Core Leakage Flow (tons/hr)	222
Core Inlet Subcooling (kCal/kg)	22.2
Core Average Heat Flux (W/cm^2)	13.5
Peak Heat Flux (W/cm^2)	43.2
Average Core Exit Quality (%)	18
Maximum Channel Exit Quality (%)	22
Average Channel Void Fraction (%)	45
Maximum Exit Void Fraction (%)	72
Min.Critical Heat Flux Ratio at Overpower (MCHFR)	4.5
Radial Power Factor for the MCHFR Fuel Element	1.087
Axial Power Factor at the MCHFR Point	1.914
Local Power Factor at the MCHFR Point	1.540
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower)(%)	4.78
MCHFR Channel Flow Rate (tons/hr)	12.56
Average Core Exposure (MWD/MT)	5160
Percent of Withdrawn Control Rods	87.4

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

TEST 7

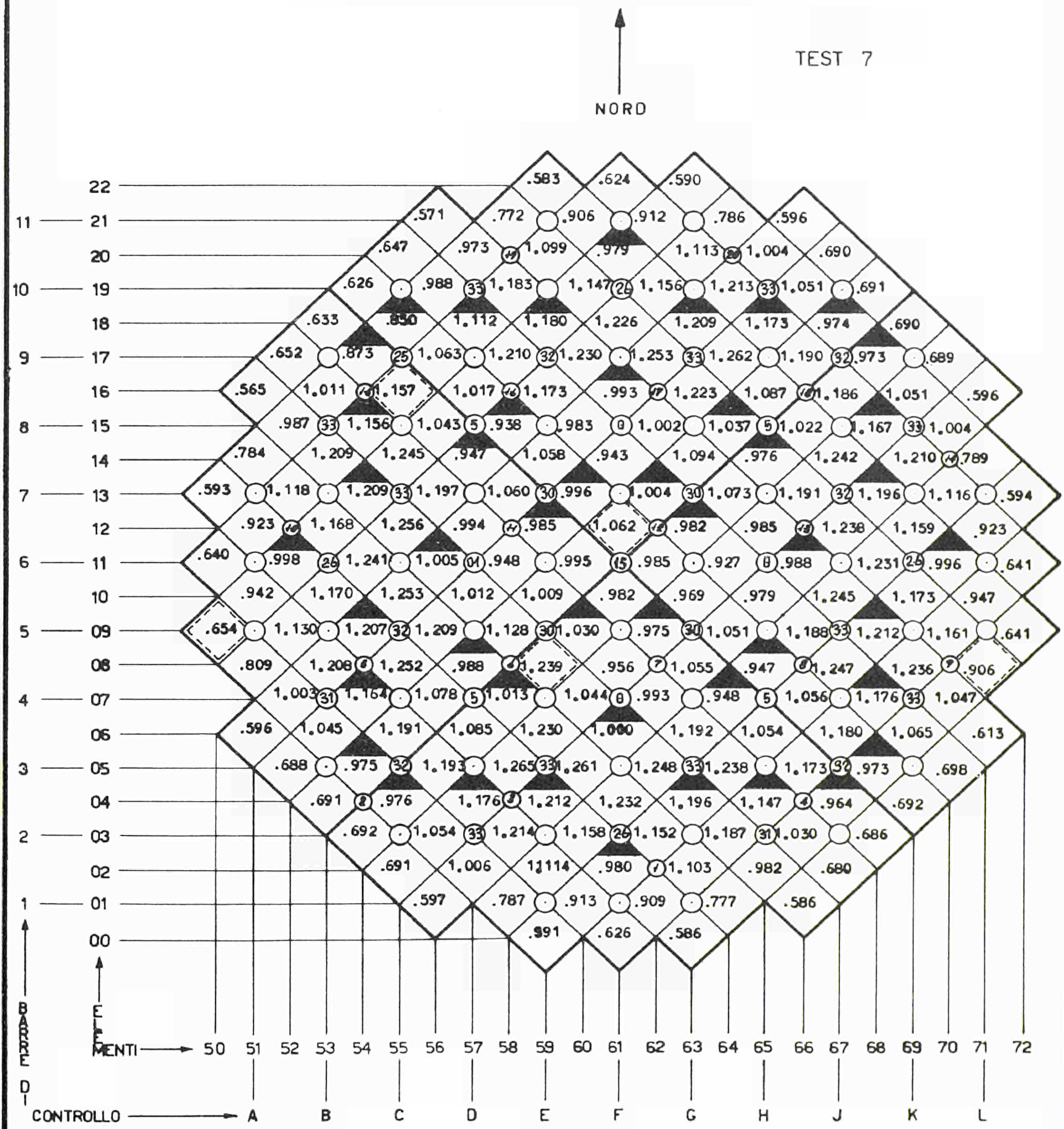

FIG. 19

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

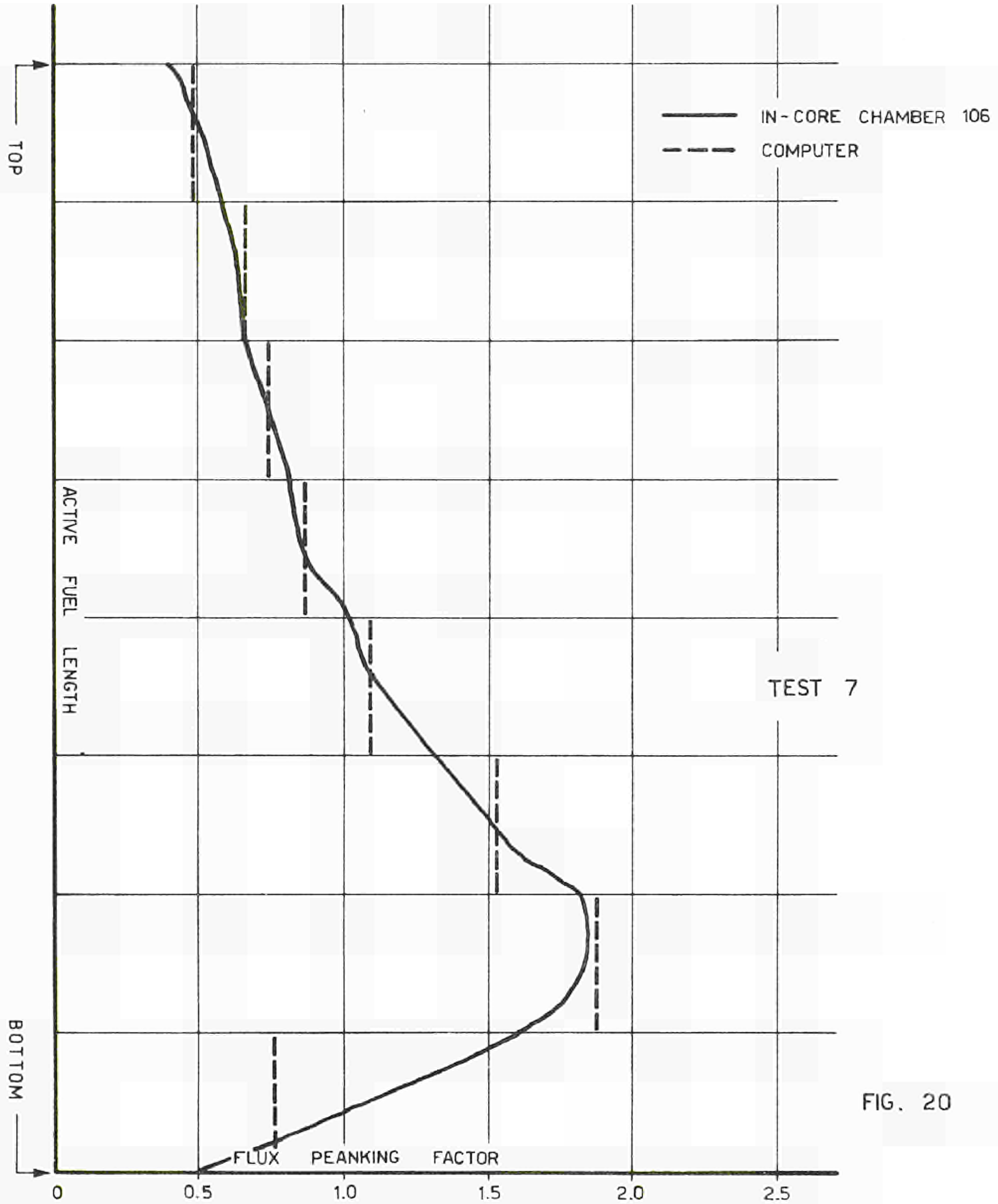

FIG. 20

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

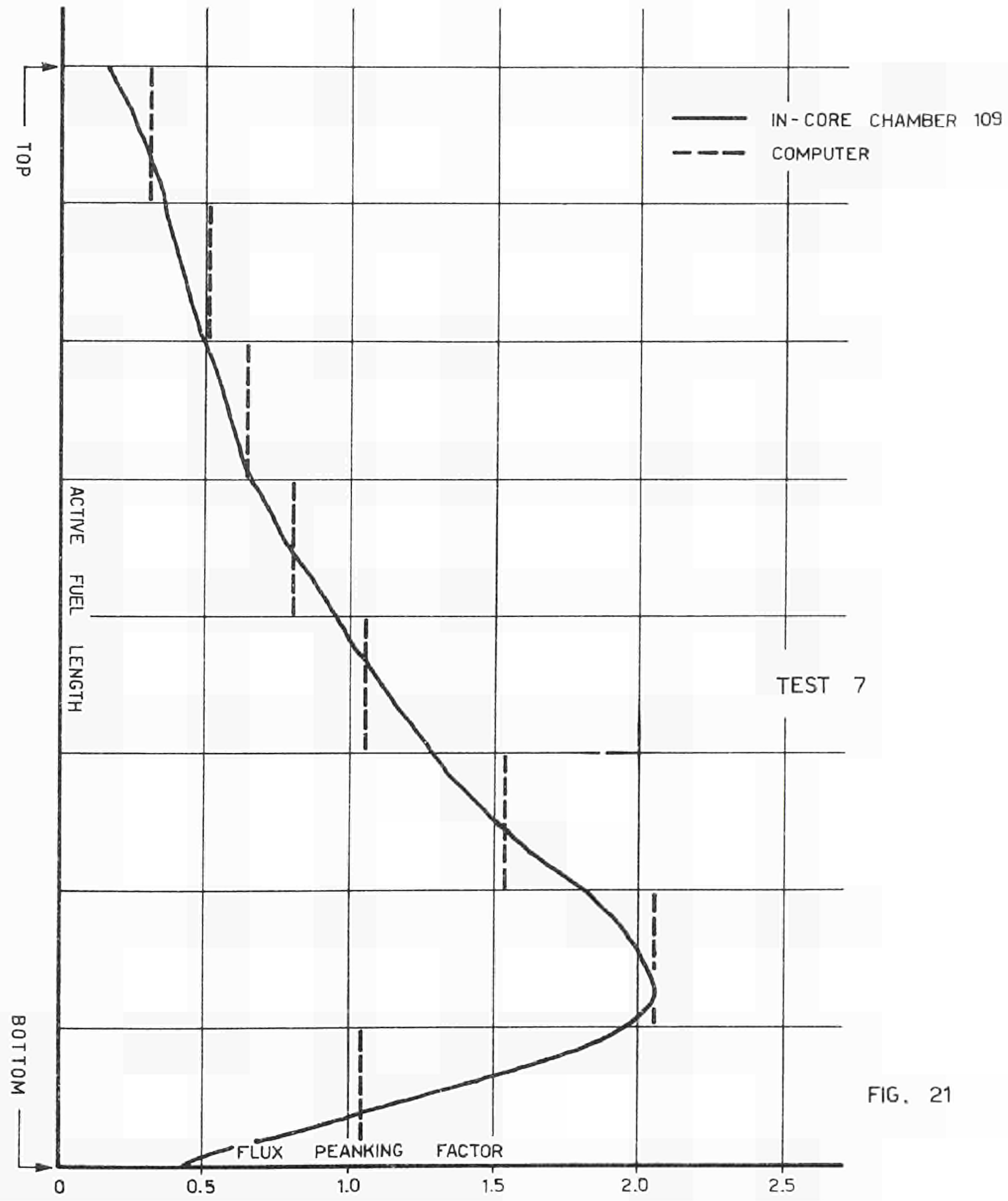

FIG. 21

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

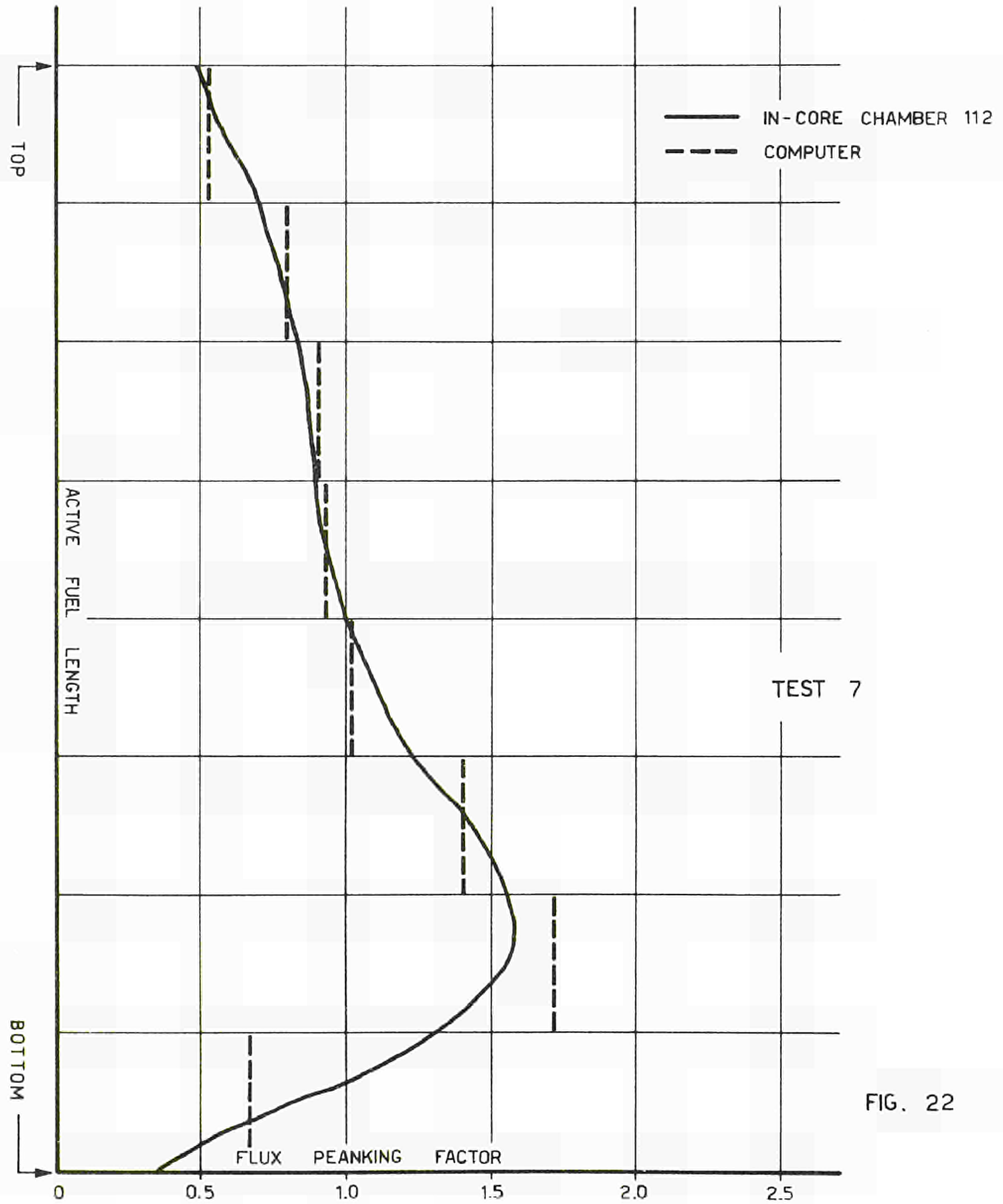

FIG. 22

4.6 Test 8 - Reactor power 65%, one loop in natural circulation, minimum subcooling

In test No. 8 the highest void condition of all the Research Program was reached even though the critical parameters were well below the prescribed limits, owing to the reduced reactor power (331.7 MWt).

It was never possible to reach higher power levels as anticipated because some of the components of the safety system reacted to the higher void content. Indeed, the low reactor level alarms worked erratically when the fluid density deviated too much from the specified conditions (*).

However, it was possible to obtain 48% voids, equal to 291% of the rated content, with an average value at the core outlet of 74%.

The ratio of primary steam (or feedwater) to the recirculation flow was 1:5. The reactor behavior was excellent under these conditions as the heat limits prescribed by the safety regulations were never approached.

The strongest element, with a power factor of 1.279, was element 64-17 (Fig. 23), whereas the maximum local heat flux (61.3 W/cm^2) was given by element 54-11.

The MCHFR, localized in element 68-11, never dropped below 3.22 thanks to the excellent axial power distribution with a peak shifted to 2/8 from the core bottom and therefore in an area where the steam quality was low (7-8%). This is evidenced in Figures 24, 25 and 26 which show the normalized traces of the movable in-core chambers and the flux distributions determined by means of the digital computer.

In this test, to make up for the strongest negative reactivity introduced by the voids, it was necessary to withdraw 90% of the control rods versus 76.2% in the test at rated condition. This figure gives an idea of the definite control exerted by the voids, especially when it is recalled that in test No. 8 rated power was not even reached as was instead the case with test 1-A.

(*) See "Garigliano Nuclear Power Station Research Program - Quarterly Report No. 9" ENEL, July 1, 1966 - Pages 22 and 25.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - VI

Results of Test No. 8

Reactor Power (MW)	331.7
Gross Generator Power (MW)	102.8
Primary Steam Flow Rate (tons/hr)	562
Secondary Steam Flow Rate (tons/hr)	0
Feedwater Temperature (°C)	169.3
Steam Drum Pressure (ata)	68.5
Reactor Pressure (ata	69
Total Recirculation Flow Rate (tons/hr)	2875
Core Leakage Flow (tons/hr)	202
Core Inlet Subcooling (kCal/kg)	25.5
Core Average Heat Flux (W/cm^2)	17.2
Peak Heat Flux (W/cm^2)	61.3
Average Core Exit Quality (%)	22
Maximum Channel Exit Quality (%)	29
Average Channel Void Fraction (%)	48
Maximum Exit Void Fraction (%)	75
Min.Critical Heat Flux Ratio at Overpower (MCHFR)	3.22
Radial Power Factor for the MCHFR Fuel Element	1.244
Axial Power Factor at the MCHFR Point	1.860
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower) (%)	7.95
MCHFR Channel Flow Rate (tons/hr)	13.01
Average Core Exposure (MWD/MT)	5170
Percent of Withdrawn Control Rods	90

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

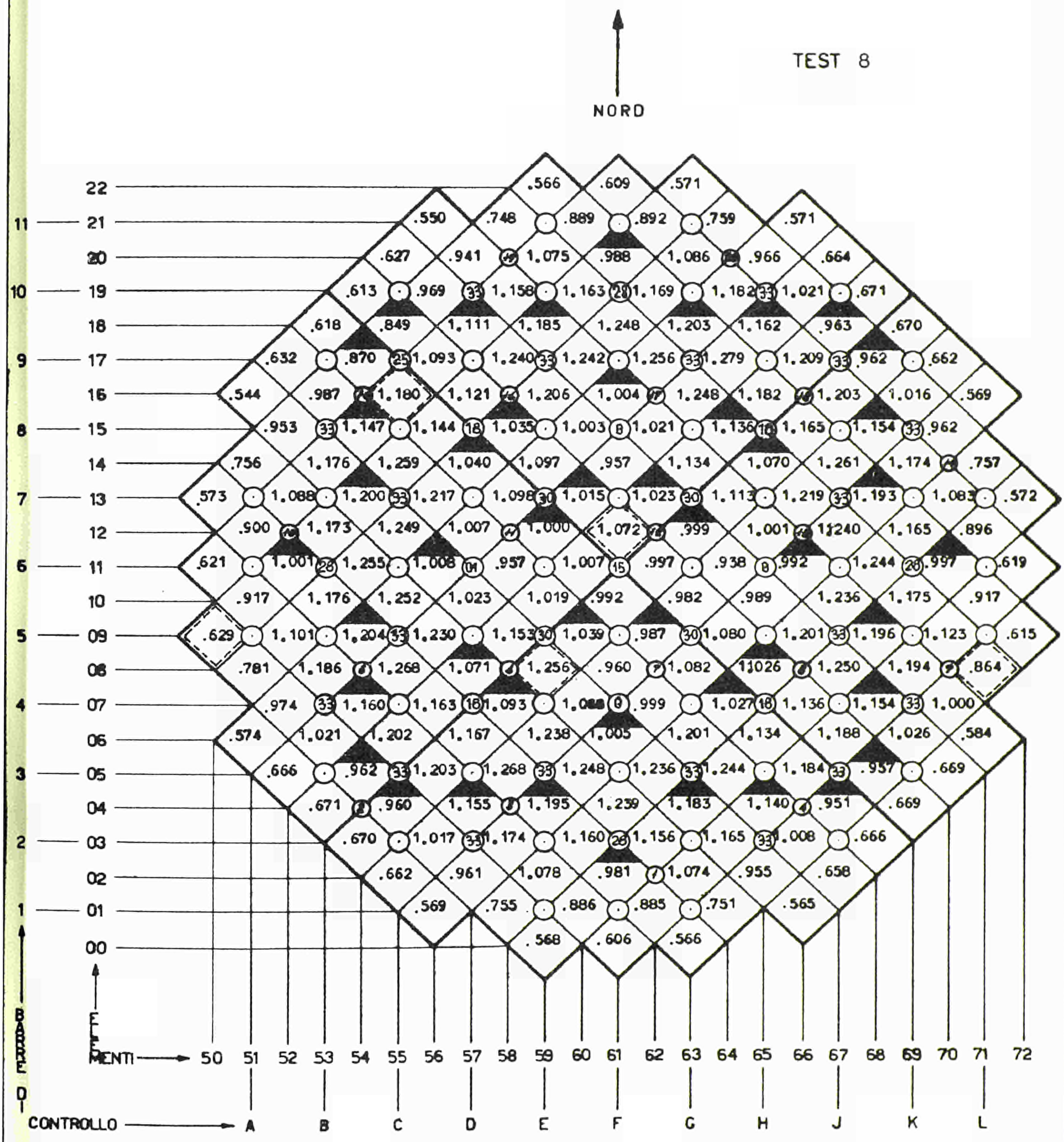

FIG. 23

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

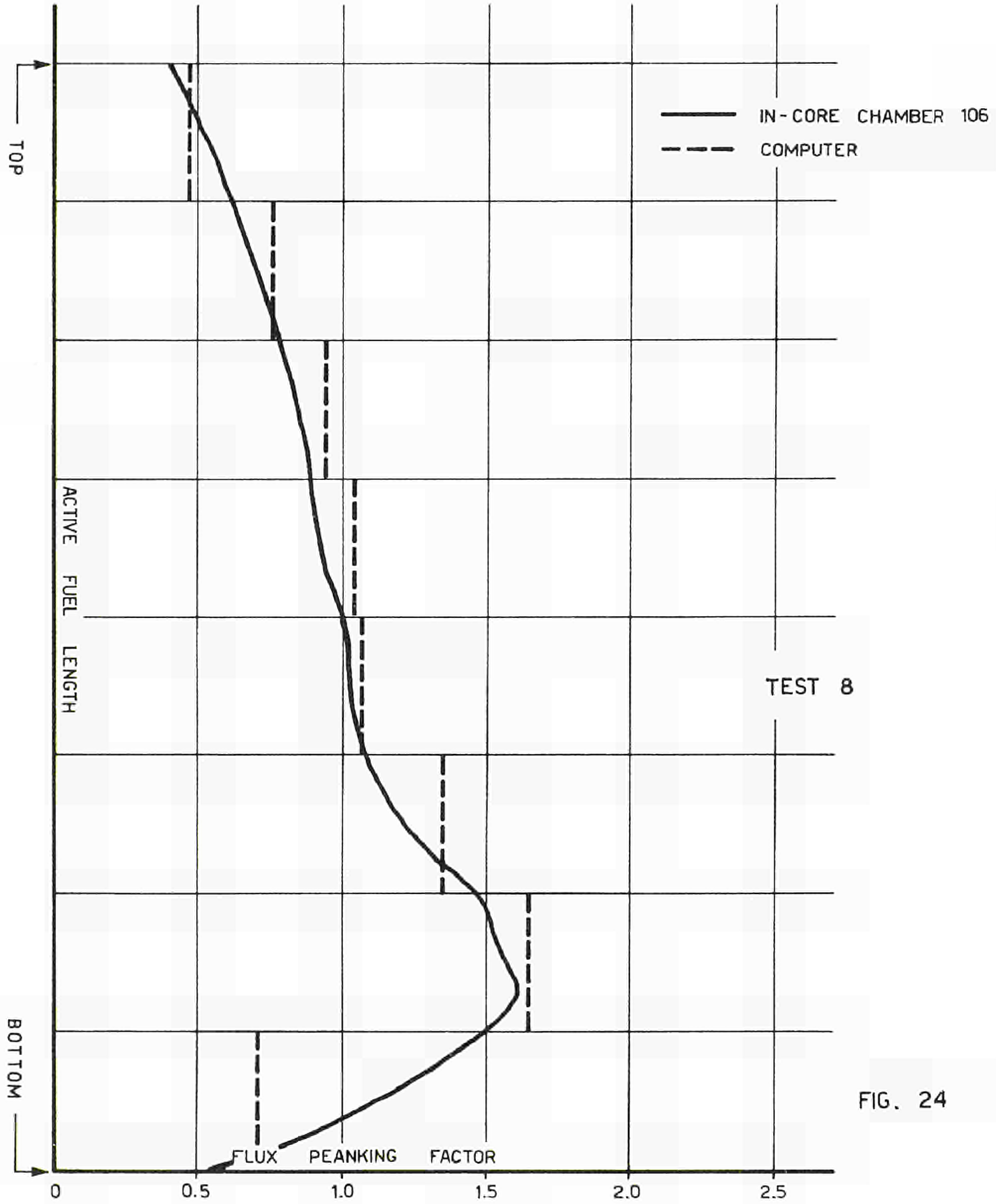

FIG. 24

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AXIAL FLUX DISTRIBUTION

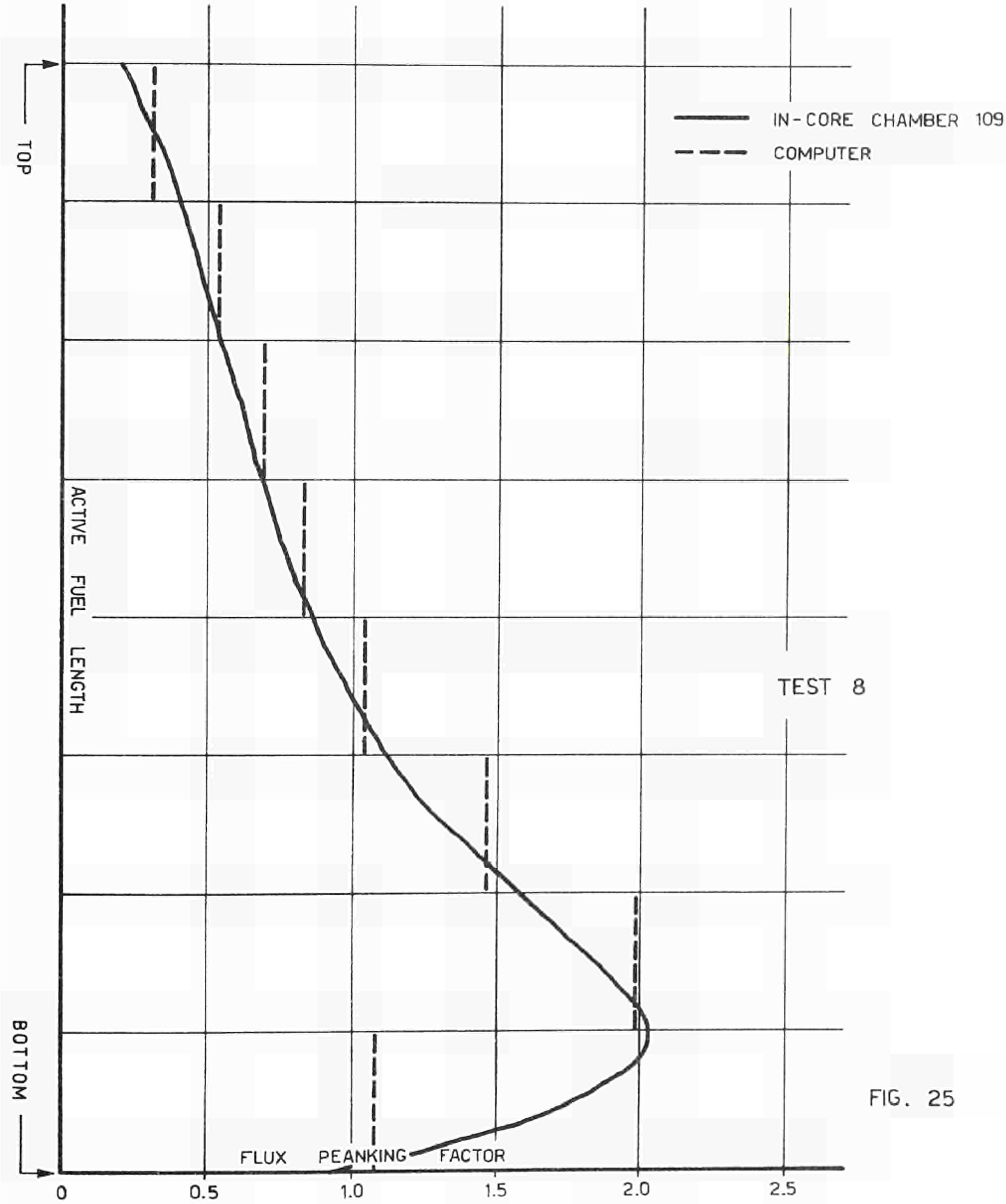

FIG. 25

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

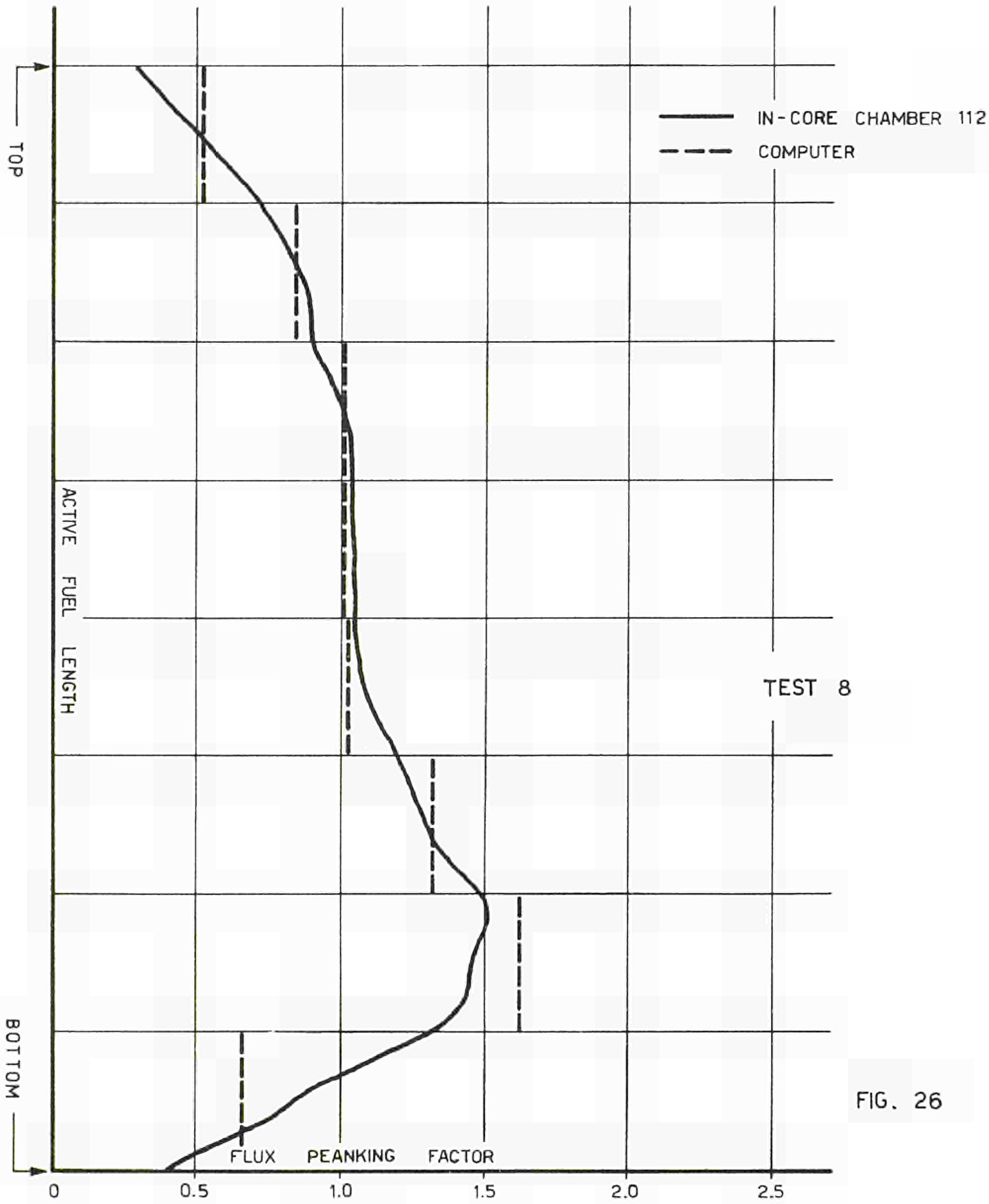

FIG. 26

4.7 Test 9 - Reactor power 65%, one loop in natural circulation, maximum subcooling

Test No. 9 can be compared directly with test No. 8 as it was performed at the same power level and recirculation flow, but with a different subcooling. In test No. 9 the power of the primary system was reduced by inserting some of the control rods (from 90% to 88.2% of withdrawn rods). The power level was then restored to 65% by generating 56 tons/hr of secondary steam. Therefore, subcooling changed from 25.5 kCal/kg to 31.1 kCal/kg.

The critical parameters, which normally improve with the higher subcooling, were instead worsened with respect to test No. 8. However, this is imputable to the unfavorable control rod configuration which had to be adopted during test No. 9. Indeed, rod D-4 (Fig. 27) stalled from mechanical reasons and was not possible to insert it beyond notch 20. Thus, an area of pronounced power was created which made it necessary to insert other rods around rod D-4 and to repeat a similar configuration in the symmetrical region to avoid undue radial dissymetries in the core. As a consequence, the axial distribution was not very good and the flux peaks were slightly higher than normal.

The strongest element was element 58-05 with a radial factor of 1.34, which is slightly higher than the values experienced previously just because that element was positioned near the stalled rod D-4.

The maximum local heat flux (71.9 W/cm^2) and the MCHFR (2.75) were observed in the west rod of element 68-11, at 2/8 from the bottom.

The normalized traces of the movable in-core chambers are not given for test 9 as it was not possible to obtain them during the test. However, Figures 28, 29 and 30 give the axial distribution determined by the computer.

GARIGLIANO RESEARCH PROGRAM

TABLE 4 - VII

Results of Test No. 9

Reactor Power (MW)	330.8
Gross Generator Power (MW)	100.7
Primary Steam Flow Rate (tons/hr)	510
Secondary Steam Flow Rate (tons/hr)	56
Feedwater Temperature (°C)	169.8
Steam Drum Pressure (ata)	58.3
Reactor Pressure (ata	69
Total Recirculation Flow Rate (tons/hr)	2835
Core Leakage Flow (tons/hr)	198
Core Inlet Subcooling (kCal/kg)	31.1
Core Average Heat Flux (W/cm^2)	17.2
Peak Heat Flux (W/cm^2)	71.9
Average Core Exit Quality (%)	21
Maximum Channel Exit Quality (%)	30
Average Channel Void Fraction (%)	47
Maximum Exit Void Fraction (%)	76
Min.Critical Heat Flux Ratio at Overpower (MCHFR)	2.75
Radial Power Factor for the MCHFR Fuel Element	1.253
Axial Power Factor at the MCHFR Point	2.16
Local Power Factor at the MCHFR Point	1.54
Overpower Factor	1.20
Steam Quality at the MCHFR Point (at Overpower) (%)	8.6
MCHFR Channel Flow Rate (tons/hr)	13.03
Average Core Exposure (MWD/MT)	5170
Percent of Withdrawn Control Rods	88.2

CENTRALE DEL GARIGLIANO

FUEL BUNDLE POWER FACTORS

TEST 9

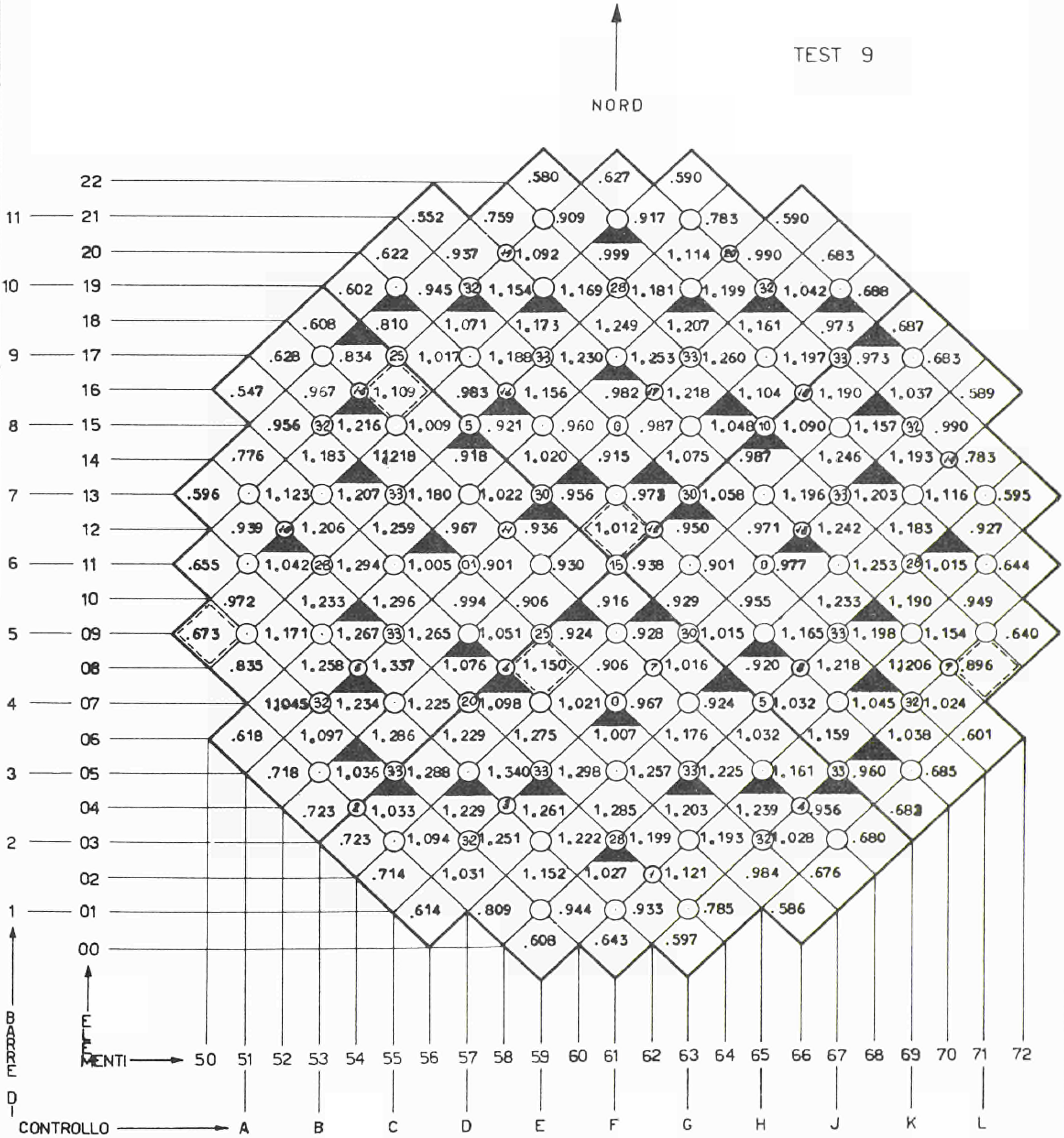

FIG. 27

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

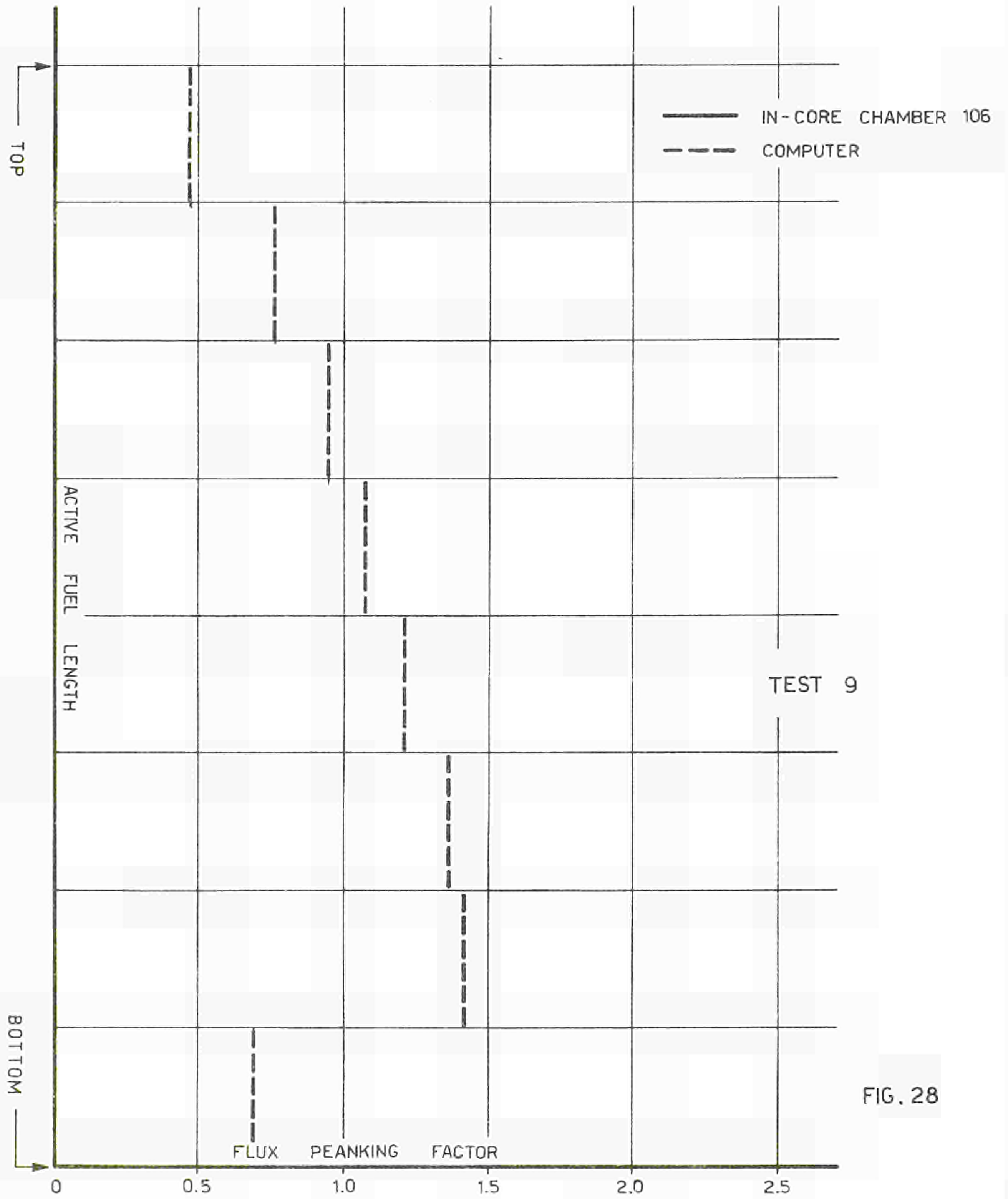

FIG. 28

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

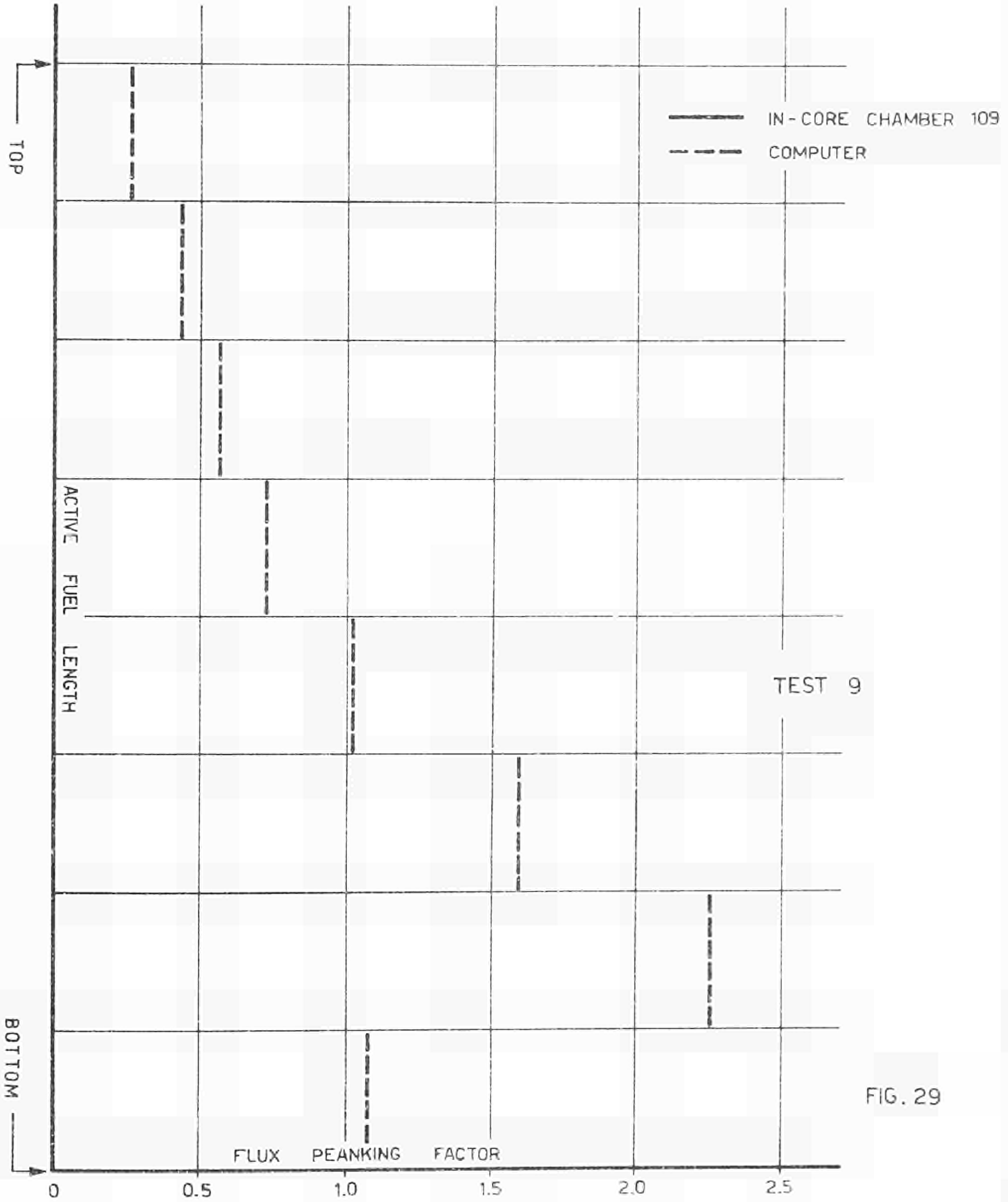

FIG. 29

CENTRALE DEL GARIGLIANO

AXIAL FLUX DISTRIBUTION

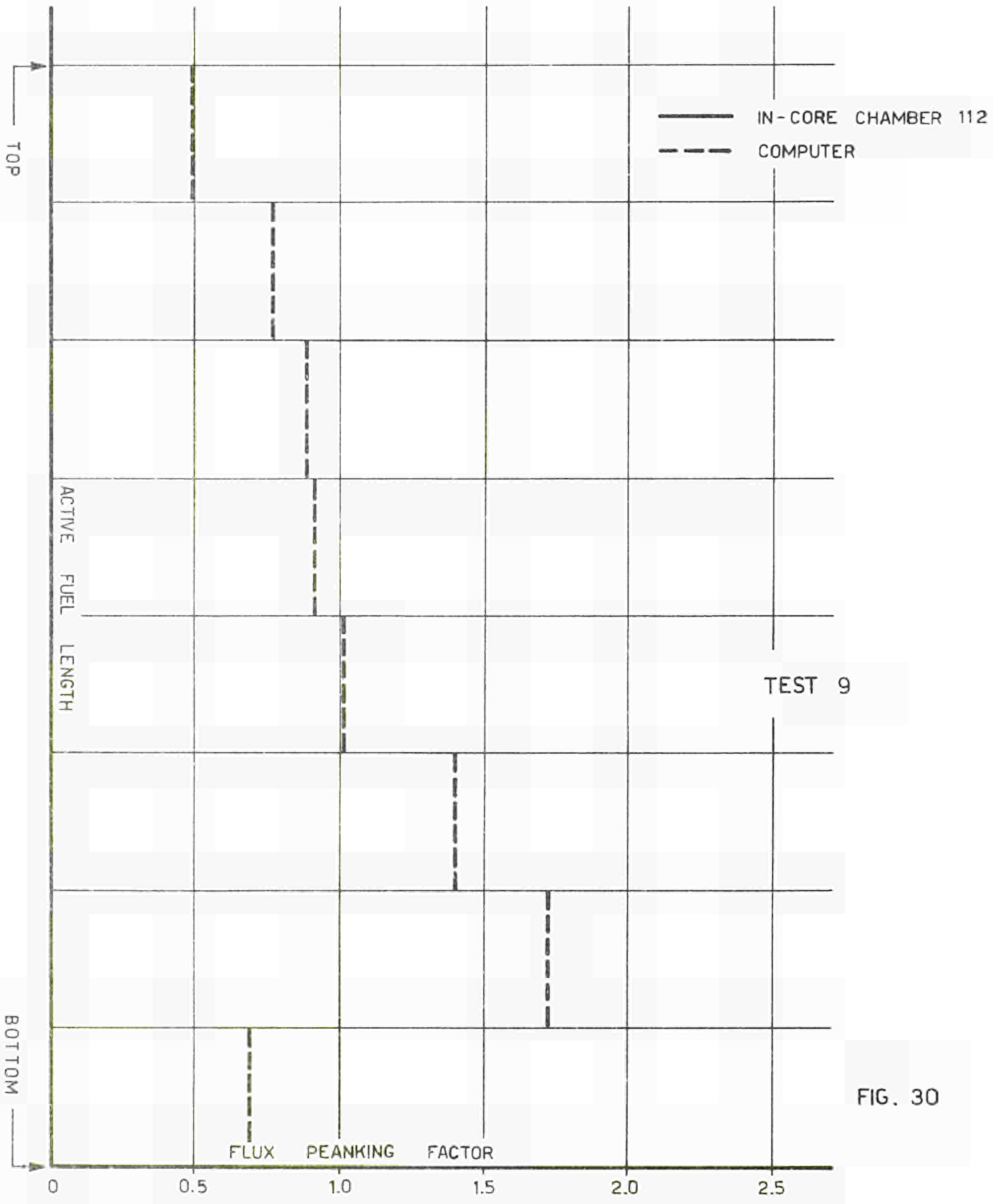

FIG. 30

5. Digital computer performance

The digital computer installed at the station proved to be a very useful tool in investigating and assessing plant performance during the tests. It demonstrated its great flexibility as it was able to follow all the special tests, even under conditions widely differing from normal operating conditions. The computer codes were prepared and the computer was designed specially for normal operating conditions. It was therefore felt that during the high void tests the computer would have had difficulties in establishing the performance of the plant. This was not so, which demonstrated the excellence and the outstanding usefulness of the computer.

The most important thing is that it was possible to program control rod configurations suited for each test by simulating the conditions in advance on the computer temporary off line.

Without the computer it would not have been possible to perform the careful investigations mentioned in Section 3 because we would not have had time to collect such an abundant amount of data and to perform all the required calculations so quickly.

During the processing of the data obtained from the tests, it was however necessary to make a certain number of modifications and corrections in the calculation programs, as is normal with new equipment when it is called upon for the first time to perform complex on-line operations. However, these are minor numerical corrections, slight modifications in the codes and a certain amount of refinement in the calculation methods. For instance, we realized that in the calculation of the maximum local heat flux the computer would not operate correctly because it averaged the corner rod factor relating to a given fuel element and a given level with the remaining three corner rod factors belonging to the three adjacent elements at the same level. In other words, the local heat flux peak of a corner rod was influenced by the adjacent corner rods of other fuel elements. Among the other, this was not prudential as a given heat flux peak could appear smaller than it actually was. This inaccuracy of the program was corrected and all the calculations relating to the special tests were performed again off line on the computer.

A first series of investigations was also performed with the physical computer code and specifically on the influence of fuel irradiation on the calculation of the neutron flux in power distributions. It is felt that the programs will have to be studied further to be perfected and to eliminate the inaccuracies that might result from lack of flexibility of some of the formulas especially after prolonged fuel irradiation.

However, we are confident that a well coordinated effort will be able to eliminate in future any inconvenience and render the computer even more accurate and more useful a means of investigation of plant behavior.

For instance, the computer already demonstrated its usefullness in following crud deposition on fuel elements. Indeed, as has been done during the tests, it is possible to make the calculated core pressure drop coincide continuously with the measured pressure drop by correcting the hydraulic coefficients to the element inlets. This will provide a pretty clear picture of the progress of the phenomenon and it will be possible to check the reactor safety margins more closely.

The comparison between calculated axial neutron flux distributions and the values determined by the movable in-core chambers substantially confirmed the observation of gamma scanning (*), that is that the computer generally tends to overrate the axial peak but above all it overrates the flux in the higher part of the core and underrates it in the lower part of the core. Therefore, for all the tests, the values of the maximum heat flux are higher and those of the MCHFR are lower than the actual ones. At any rate, the error involved is very small - on the order of a few units percent.

Finally, it should be stated that the computer does not adopt the exact values of the corner rod factor as a function of the void content; in fact, it assumes in all cases a void content equal to 20%. However, since these factors differ very little between zero and 50% voids and they tend to decrease (**) for voids above 45-50% it is felt that this approximation is quite acceptable.

(*) See ENEL, "Misura della distribuzione di potenza nel reattore del Garigliano con un nuovo metodo di scansione gamma", by M. Galliani, U. Cammarota et alt. - Rome, March 1966

(**) See GEAP, 4899 "Final Safeguards Report - Garigliano Development Program", June 1, 1965 - Pages 5-24

6. Movable in-core chambers

The movable in-core chambers installed near the fixed chambers in positions 106, 109 and 112 were found most valuable in determining the axial power distribution quickly as they gave an immediate picture of the trend of the flux during the severe tests, that is in reactor operating conditions which are not very well known. The results were so satisfactory that we almost regretted that the remaining 17 core positions had not been provided with movable chambers. For the purpose of demonstration, Fig. 31 reproduces the recording performed during test 1-A with the movable in-core chamber 112. As described in Para. 4.1 of this report the peak flux is definitely shifted upwards in the core. The spurious peak located at approximately midway is due to the presence of steel springs outside the in-core chamber tube to reduce vibrations induced by the hydraulic flow. These springs are held on the tube by two stainless steel collars (Fig. 32) which are responsible for the two peak depressions which precede and follow the spurious peak in Fig. 31. The inertia of the recorder pen also contributes to enhance the peak slightly so that in correcting the tracing (dotted line) the peak should be cut at about 2/3 of its height. This apparent inconvenience was later found very useful in the normalization of the curves as it represents a precise reference point to check the active height of the fuel.

The diagrams shown in Section 4 of this report give all the normalized and corrected tracings of the three movable in-core chambers. At the same time we have included the normalized axial distributions obtained by means of the digital computer. However, it should be pointed out that the latter were obtained by averaging the distributions relating to the four fuel elements which surround a movable chamber; specifically,

- for in-core chamber 106: elements 58-09, 57-08, 58-07, 59-08;
- for in-core chamber 109: elements 70-09, 69-08, 70-07, 71-08;
- for in-core chamber 112: elements 62-13, 61-12, 62-11, 63-12.

This procedure is correct up to a point because the neutron flux detected by the in-core chamber is mainly due to the four corner rods which surround it, whereas the flux distributions determined by the computer refer to an average value of the 81 rods forming a fuel element. Therefore, it would be advisable to multiply at each level the calculated flux by the corner rod factor, thus obtaining the axial distribution for each rod surrounding the in-core chamber. During the processing of the test data, however, the corner rod factors for the neutron

MOVABLE IN-CORE 112 AXIAL FLUX RECORDING

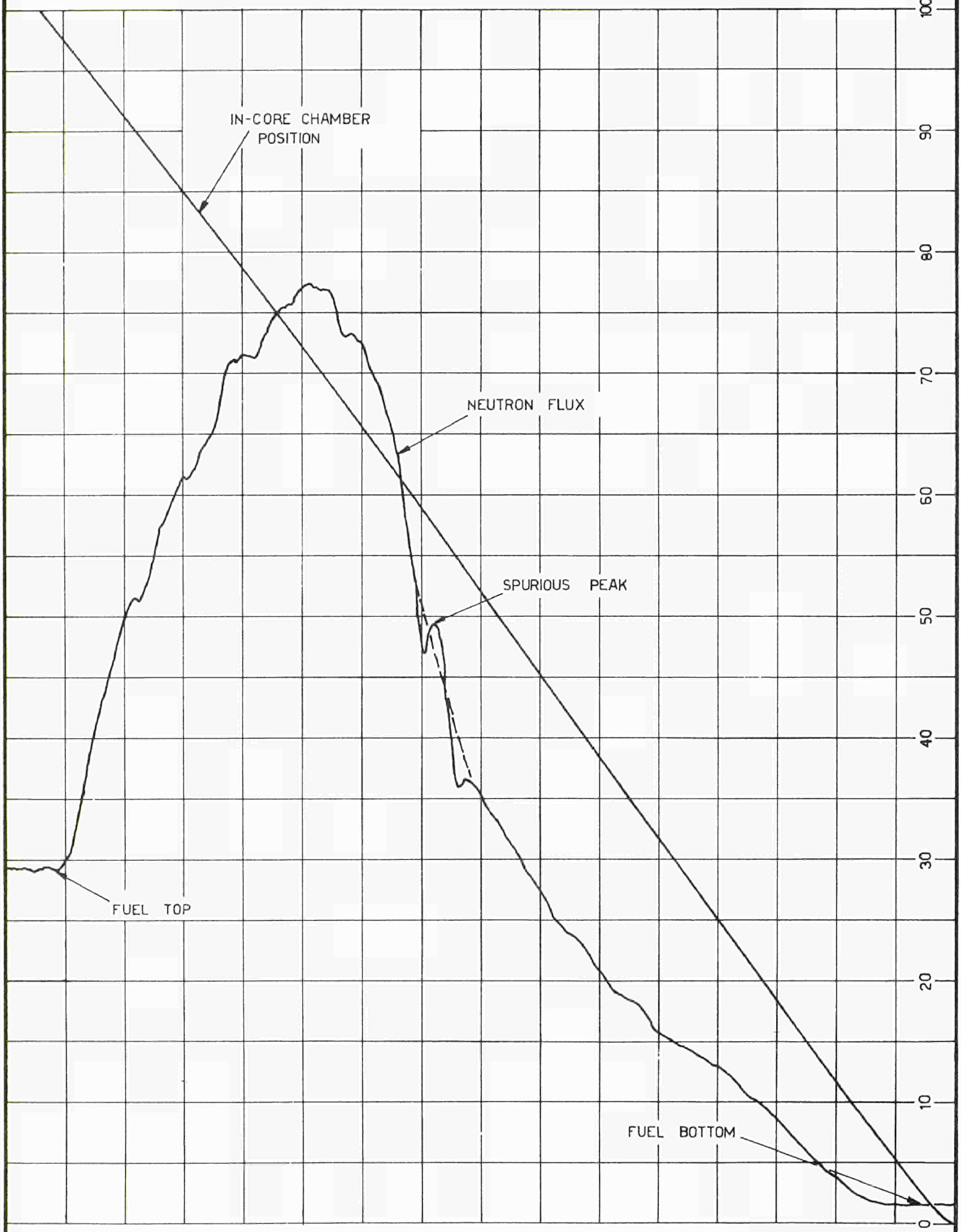

FIG. 31

CENTRALE DEL GARIGLIANO

SKETCH OF THE ANTI VIBRATION SPRINGS OF THE IN-CORE HOUSING TUBE

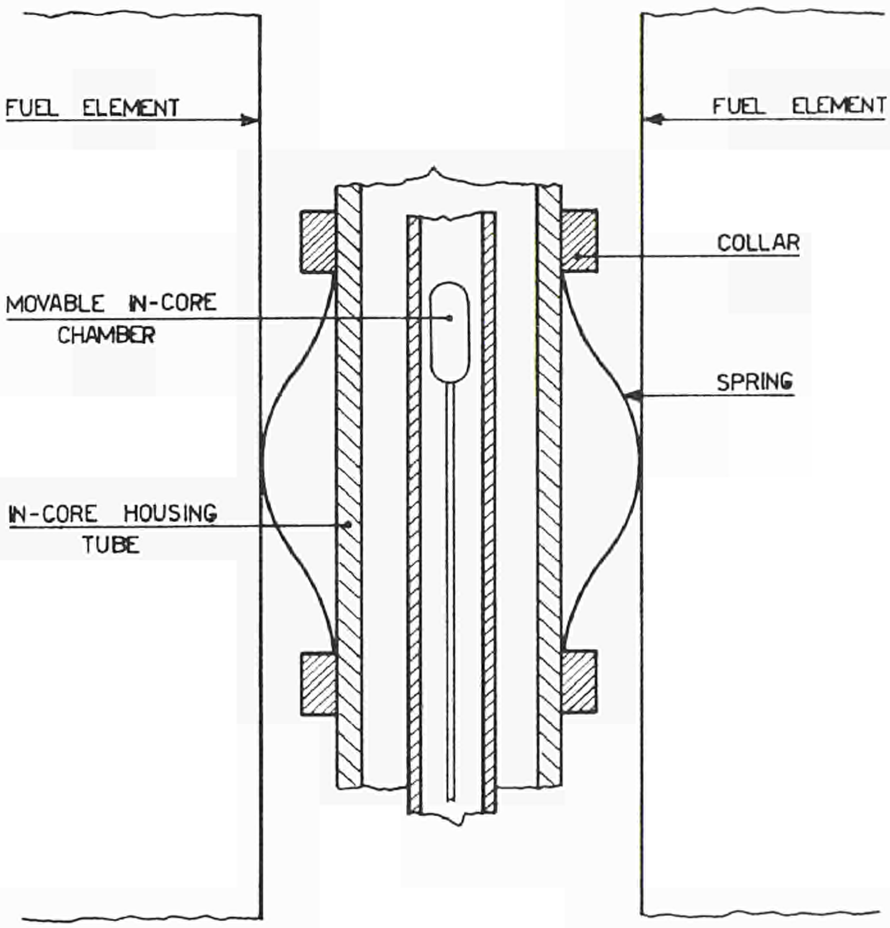

FIG. 32

flux were not available and it would not have been correct to adopt those for the heat flux which were available for the determination of the maximum local heat flux.

An interesting consideration was made for test No. 5 (Figs 12, 13, 14). It will be immediately noted that in this case the chamber tracings differ appreciably from the calculated distributions which are much more flattened than the former. Actually the movable in-core chamber recorded the axial distribution immediately after the power was increased from 75 to 94% of the rated value, in going from test 3 to test 5 for which, as previously said, it was necessary to withdraw the control rods considerably. Since the flux was depressed at the top by the voids, it was definitely shifted downwards (see Figs 8, 9 and 10) where the fuel was much less poisoned. Instead, the computer had been put on line after the rods had been withdrawn and since it was off line previously for the study of the subsequent rod configurations it did not see the axial xenon transients caused by the transfer of the flux peak downwards. Therefore, it assumed that the core was already at equilibrium with a flatter axial flux distribution. This is confirmed by the fact that for in-core chamber 109 (Figs 9 and 13) the effect is much less evident as the flux peak had already moved downwards in the condition called for by test No. 3.

7. Information from the instrumented assemblies

The four instrumented assemblies installed in the reactor before the beginning of the test were to measure the channel flows and inlet and outlet coolant temperatures directly. Unfortunately unlike all the other equipment, the instrumented assemblies supplied little reliable information. Just before the tests three flowmeters failed electrically so that the channel flow measurements were taken only in elements A (71-08) and D (55-16). All the flow measurements at the channel outlets (water-steam mixture) were found to be meaningless so that it was impossible to determine the outlet steam quality and the output of each element. Even the thermocouples, installed for direct measurement of the subcooling at the element inlets, supplied unreliable data.

The only valuable measurements where provided by the lower flowmeters in elements A and D. As we said in Section 3, they were used very advantageously in determining the channel flow distribution correctly, the calculated core pressure drop and in establishing the leakage flow.

Table 7-1 gives the channel flow values measured by the inlet turbine meters during the special tests as compared with the values determined by the digital computer. After consideration of the differences between the two sets of values it can be stated that the agreement between the measured flows and the calculated flows is fairly good save for the tests performed with very low flow rates.

In particular, the minor differences observed in test 1-A indicate the slight underrating of the leakage flow (13.5% of the total recirculation flow); this caused a smaller error in excess for the instrumented channels in the thermal hydraulic calculation of the flow distribution. The differences could also be imputed to a slight overrate of the total recirculation flow, but since the calculated core pressure drop coincides with the measured values this does not seem very likely.

For tests Nos 3 and 6 the differences are negligible so that it is reasonable to infer that all the parameters involved were evaluated correctly.

With regard to test No. 5, exactly the contrary of test 1-A happened.

For tests 7, 8 and 9 the differences are quite appreciable but hardly imputable to erroneous evaluation of the leakage flow or of the recirculation flow. As a matter of fact, the flometers (see test 8, element A; and test 9, element D) gave greater indication of malfunctioning which led to electrical trouble or complete locking. This whould explain for instance the 2.45% difference for element A in test No. 7 and, immediately after, -19.6% in test No. 9 for almost the same channel flow.

With regard to the behavior of the instrumented assemblies, it is necessary to point out that the turbine flometers worked for about one month before the special tests started.

In addition, the special instrumentation of the four assemblies had been inserted and removed from their enclosures several times because the instrumented assemblies themselves had to be removed from the reactor to allow various operations in the pressure vessel. It is therefore evident that the repeated handling of such delicate equipment has certainly not contributed positively to their performance.

TABLE 7-I

Test	Instrumented Assembly A Channel Flow			Instrumented Assembly D Channel Flow		
	Measured	Calculated	Difference	Measured	Calculated	Difference
1-A	70.23 tons/hr	73.94 tons/hr	+ 5.28%	67.56 tons/hr	70.58 tons/hr	+ 4.47%
3	31.92 "	31.61 "	- 0.97%	31.94 "	31.19 "	- 2.35%
5	32.20 "	30.82 "	- 4.29%	30.83 "	28.92 "	- 6.60%
6	32.41 "	31.87 "	- 1.66%	31.01 "	30.51 "	- 1.61%
7	16.32 "	15.92 "	- 2.45%	19.52 "	16.18 "	- 12.0%
8	no signal	16.90 "	---	12.61 "	16.73 "	+ 32.6%
9	20.79 tons/hr	16.71 "	- 19.6%	no signal	16.63 "	---

8. Conclusions

The special tests of the Research Program clearly demonstrated that:

1) The Garigliano boiling water plant is stable also under the severest operating conditions.

2) The plant generally responds promptly to any tipe of maneuver and remains well within the safety limits. Also during the most complicated operations no trouble was experienced to cause undesirable plant shutdowns. It will suffice to consider that in approximately ten days of testing (mostly at low power levels) the total generation loss was only 7.000.500 kWhrs against a full load generation of approximately 36.000.000 kWhrs.

3) The thermal performance of the reactor is excellent, even under high void conditions, because the axial flux distribution automatically locates itself so as to better exploit the subcooled part of the core.

4) It is possible to program the control rod configuration easily even with a high void content without causing severe radial power distributions.

5) In the extremely probable case of an outage of both recirculation pumps, the reactor can be operated at rated power in natural circulation.

6) The high void tests confirmed that the reactor can operate at higher power density in forced circulation and therefore generate a much greater output than rated. This statement does not take into account the effects of said increased output on the duration of the fuel cycle, and the limitations imposed by the other machinery in the plant.

7) The digital computer is a valuable means for proper conduct of nuclear plants. It allows a considerable amount of data to be collected and processed rapidly and these data, properly handled, are a precious guide for the operator in making the best use of the fuel and of the plant in general. In addition, the flexibility it proved to possess during the tests allows all the events that may occur in the plant lifetime to be followed.

With regard to the computer installed at the Garigliano, it is indispensable that the calculation codes be further perfected so as to obtain more and more accurate results and to further improve the plant operation criteria. The modifications could possibly be incorporated advantageously in programming future computers.

8) The movable in-core chambers are definitely a step forward in the design and manufacture of in-core instrumentation as they give the desired information immediately and not after about 15 hours as is the case with wire irradiation. This means that the core performance can be known at any moment and, if necessary, remedial action can be taken timely.

9) The instrumented assemblies, at least of the type used at the Garigliano, have not turned out to be an efficient investigation tool, probably because better technical solutions have not yet been found to make the special instrumentation work properly without modifying the fuel element considerably. They could be very useful if they were able to determine the neutron flux distribution inside the element, coolant temperatures, coolant flow rate and, above all, the element output from the outlet steam quality committing only small errors. In particular, the element output would serve the purpose of confirming the theoretical calculations performed by the computer.

10) A noteworthy result is the one obtained with the highest void content. The fact that the reactor could be operated with a ratio of total feedwater flow to recirculation flow equal to 1:5 makes it reasonable to expect that the jet pumps of the new boiling water reactors can be fed directly with the water from the feed pumps. Indeed, the ratio of 1:5 between the driving water flow in the jet pumps and the total recirculation flow through the core can be promising from the standpoint of pump efficiency. When the other problems associated with this solution are solved, recirculation loops and pumps could be entirely dispensed with.

ACKNOWLEDGEMENTS

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Alfred Nobel

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