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COMMISSION OF THE EUROPEAN COMMUNITIES

**PLUTONIUM RECYCLE IN ENEL'S
LIGHT WATER REACTORS**

by

A. ARIEMMA, U. BELELLI, M. PAOLETTI GUALANDI,
I. ROSA, L. SANI and B. ZAFFIRO

1972



Report prepared by ENEL
Ente Nazionale per l'Energia Elettrica - Rome (Italy)

Euratom Contract No. 092-66-6 TEEI

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ABSTRACT

This document is the final report on the work performed under the ENEL-EURATOM Research Contract No. 092-66-6 TEEI for utilization of plutonium in thermal reactors, which became effective in June 1966 and was completed in 1970.

The studies carried out for the selection of the reactor for the irradiation program with plutonium prototype assemblies are briefly summarized. A detailed description is given for the calculation methods and codes used for the design of the plutonium prototype assemblies that have been in the Garigliano reactor since summer 1968.

The paper contains also the main results of the experimental activities carried out under the program and, in particular, the results of the criticality measurements in the Garigliano reactor, of the gamma-scanning on the core containing twelve irradiated prototype assemblies, and of the post-irradiation measurements on an enriched-uranium assembly irradiated to about 10 000 MWd/MTU.

The operating experience gained up to mid-1970 with sixteen prototype plutonium assemblies is also summarized. At that date, the prototype assemblies had reached an average irradiation level of 7 000 MWd/MTM with a lead assembly value of 7 500 MWd/MTM.

The paper reports on the results of the optimization studies to determine the plutonium value, particularly where the plutonium is blended with depleted uranium recovered from the reprocessing of Magnox reactor fuel.

Finally, areas that require further studies for a specific reactor are indicated.

KEYWORDS

BOILING WATER REACTORS
RECYCLING
PLUTONIUM
DESIGN
ECONOMICS
OPTIMIZATION
IRRADIATION
REACTOR LATTICES

MOCKUP
MEASURED VALUES
CRITICALITY
GAMMA SCANNING
BURNUP
COMPUTER CALCULATIONS
PROGRAMMING

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1. Introduction *)

The main reason for the interest in recycling plutonium in thermal reactors pending the industrial development of fast reactors stems from the large quantities of plutonium that are being produced as a by-product in the reactor operating today and which will significantly grow in the future as the installed nuclear capacity increases. The feasibility of utilizing plutonium as a recycle fissile material and its economic advantages have been carefully investigated by ENEL. In mid-1966 ENEL, in cooperation with EURATOM, launched a research program, the purpose of which was to assess the economic potential of such a recycle, to prove the technical feasibility and to find out the best ways of utilizing plutonium in ENEL's water reactors. Therefore, the studies were devoted on one hand to evaluate the technical-economical advantages of using plutonium in a given reactor type and, on the other, to develop nuclear design criteria to optimize the fundamental characteristics of this fuel. This goal was approached through different experiments which increased the amount of information available to adjust the design calculation methods to be used, and through the demonstration of the technical feasibility of plutonium recycle with a relatively large irradiation program using prototype plutonium assemblies.

Therefore, studies have been conducted as realistically as possible on uranium and plutonium fuel cycles to establish on which reactor to carry out the demonstration irradiation program with prototype assemblies. The models used in these studies were the fuel cycles of the two water reactors operating on ENEL's network, namely, the 150-MWe Garigliano boiling water reactor and the 257-MWe Trino Vercellese pressurized water reactor.

Based on the results of the preliminary studies, the choice for the implementation of the irradiation program fell on the Garigliano reactor in which the prototype plutonium assemblies were loaded.

Irradiation started in the summer of 1968 and is still satisfactorily under way. By the end of June 1970, the first twelve assemblies had reached an average burnup of over 7000 MWd/MTM, without any significant difference in behavior from the enriched-uranium assemblies.

*) Manuscript received on January 19, 1972

The prototype assemblies were fabricated with the plutonium recovered from reprocessing of the fuel irradiated in the Latina gas-graphite reactor. Over 70 kgs of this plutonium was required to fabricate the sixteen assemblies, thus giving a thermal output due to the UO_2 - PuO_2 assemblies in the core of about 10% of the total. Thus the Garigliano station is the first commercial station in the world to carry out an experiment on plutonium recycle of such extent.

Since one of the main objectives of the program was the development of adequate design criteria for plutonium assemblies, attainment of this objective required a considerable amount of effort to produce suitable calculation methods and techniques, which were also verified against experiments conducted under the Research Program. For this part of the work ENEL also made available the results of the neutronics experiments performed by UKAEA for ENEL in the DIMPLE critical facility at Winfrith on a uranium-plutonium lattice like that of the Garigliano reactor.

A detailed discussion of all the phases of the program was given in several topical and progress reports prepared in compliance with the Contract, and a list of these reports may be found at the end of this document. However, all the main results are described again herein, grouped so as to provide a complete picture of all the activities and the conclusions, with no need to refer to other documents.

2. REPORT ON WORK PERFORMED

The Program for Plutonium Utilization in Thermal Reactors can be considered divided into two main phases.

In the first phase, an analysis was made of all the technical aspects associated with the introduction of plutonium fuel in the ENEL water reactors, to investigate the various potential problem areas and to assess the economic implications. In this phase, the criteria and techniques for nuclear design of prototype plutonium assemblies to be used for an irradiation program were also developed.

The second phase comprised experiments on the Garigliano fuel to obtain data which were then used to verify and trim the calculation techniques. In this second phase a total of sixteen plutonium assemblies were loaded into the reactor and are still being exposed.

Besides the main results obtained in the two phases, this Section provides a brief description of the characteristics of the prototype assemblies and the experience acquired at the Garigliano with the operation and handling of these assemblies.

2.1 Selection of the Reactor for the Irradiation Program with Plutonium Prototype Assemblies

Studies have been conducted on as realistic as possible uranium and plutonium fuel cycles for the two water reactors of ENEL to examine the main problem areas and to select the reactor in which to carry out the demonstration irradiation experiment with prototype plutonium assemblies. On the basis of the technical and economic evaluations performed, it was possible to draw a number of conclusions on each of the areas investigated⁽¹⁾.

(a) The analysis of the technical problems associated with the introduction of plutonium in the two reactors indicated that a slight perturbation on power distribution could be caused under certain circumstances.

Therefore, from the technical standpoint, the selection of the reactor to be used first for the experiment with prototype plutonium assemblies

was dictated by the greater power margin after the deduction associated with the perturbations due to plutonium loading. The investigations and high void tests previously performed on the Garigliano reactor under another research contract with EURATOM had indicated that this reactor does possess ample margins. On the other hand, the Trino reactor was being carefully examined with an aim at exploiting the available power margin to increase the electrical output. It was therefore reasonable to expect, from the standpoint of using prototype plutonium assemblies, that this reactor would no longer have the degree of flexibility desired for prudential reasons. However, it was possible that a better fuel assembly design or other improvements would give rise to a further power margin, but this would have to be considered in further detail.

- (b) The economic analysis relating to the assessment of the industrial value of plutonium revealed that it was more dependent on the assumption on plutonium fabrication overprice than on the reactor type. Thus, from the standpoint of economics, there were no prevalent reasons to prefer either reactor for the demonstration irradiation experiment.

Therefore, ENEL and EURATOM agreed to perform the experimental part of the research program on the Garigliano reactor, in which a number of prototype plutonium assemblies were loaded to demonstrate their behaviour under irradiation. However, the usefulness of the information derived also for application to PWR fuel has never been overlooked.

2.2 Description of the Calculation Methods

The calculation methods for the study of the various aspects of a core loaded with UO_2 - PuO_2 fuel was developed by adapting methods and techniques that had been developed for enriched-uranium systems. The adaptation was based mainly on the experimental results obtained with critical assemblies of plutonium fuel and on measurements purposely carried out on the Garigliano reactor (cfr. Sub-Section 2.4).

The use of plutonium fuel revealed intrinsic difficulties in respect of neutron calculation, essentially because of resonance at low energies. Generally speaking, it may be stated that for plutonium-bearing systems the calculation of the spectrum--and thus of the nuclear constants averaged over the spectrum--must be carried out with greater care than for all-uranium systems. Therefore, the calculation methods were developed with an aim at ascertaining what degree of precision in spectrum representation is required for the different kind of problems to be solved. To this purpose, use was made of digital-computer codes already prepared for uranium systems; it was first necessary to modify the capacity of these codes to allow a proper study of the characteristics of the two ENEL water reactors and subsequently to adapt them to take into account the presence of plutonium.

The computer codes were selected in relation to the different lattices of BWR's and PWR's; even though the work was devoted to adapting the program to the Garigliano boiling water reactor, some effort was made to develop similar methods for the Trino Vercellese pressurized water reactor. In the following description, the calculation methods for the two reactor types will be dealt with separately. Depending on the nature of the problems to be solved, the calculation methods have been divided in:

- (a) methods for power distribution and reactivity determination;
- (b) methods for fuel cycle studies;
- (c) methods for the verification of the shutdown margin;
- (d) methods for the study of transients.

2.2.1 Methods for Power Distribution and Reactivity Determination

The calculation method developed is based on the use of a diffusion code which is applied in a two- or three-dimension geometry according to the nature of the problem to be solved. In this connection, it should be pointed out that most of the operating situations of a BWR core are characterized by heterogeneities of a tridimensional type, especially because of the distribution voids

along a channel, so that recourse must be had to the technique of superimposition of effects. Instead, for PWRs it is possible to consider the axial and radial flux components separable and thus resort to a bidimensional representation of the problems.

Lattice constants and their variation with exposure were generally calculated by means of RIBOT code; the K_{eff} calculation normally follows the power distribution calculation. The codes used for these calculations can be conceptually grouped in two main categories:

- (a) Codes that permit the determination of the macroscopic power distribution of the whole core as a function of irradiation.
- (b) Codes that permit the evaluation, as a function of irradiation, of the extent to which the power distribution is affected by local conditions owing to non-uniformities in the lattice, such as water gaps between assemblies, different isotopic compositions of adjacent assemblies, and presence of control rods.

After careful examination of all the codes available for BWRs, for the category (a) a FLARE-type code, namely ERFLARE, was selected because of its flexibility and economic advantages, while a TURBO-type code, CONDOR, appeared more suited to PWR problems.

The category (b) code normally used was the two-group BURNY, whilst a new code, BURSQUID, resulting from linking the five-group RIBOT with the SQUID diffusion code, was prepared for more detailed calculations.

Whenever it was considered necessary for the lattice constants calculations to adopt a more sophisticated representation of the neutron energy distribution, use was made of the GAM or FORM code to calculate the constants relating to the higher than thermal energies, and of the THERMOS code for thermal energies. To solve problems characterized by the presence of control rods and to calculate the nuclear parameters in particular regions such as the neutron sources, the fuel assembly sheaths, etc., use was made of the GGC-II and DTK codes, as appropriate. In all these cases the diffusion code used was initially PDQ-2, and later the SQUID

of greater capacity. For a number of special problems for which the assumption of axial and radial neutron flux component separability is not justified, such as the evaluation of local perturbations induced by the control rods on the radial power distribution, use was made of the tridimensional diffusion code, TRITON.

The following is a description of the main characteristics of all the above-mentioned codes.

The ERFLARE Code

This code (ENEL Revised FLARE) is the ENEL version of the FLARE code⁽²⁾, adapted for the requirements of a large boiling water reactor.

The three-dimension FLARE code permits a fairly approximate calculation of reactivity, power distribution and burn-up, and also the representation of the control rod configuration during the periods in which the life of a BWR is subdivided for the purpose of the calculations.

The original version of the FLARE code was characterized by a relatively limited number of points (14 x 14 x 12 in the x, y and z directions) and could not be usefully applied to the Garigliano core. In fact, to represent each assembly on an horizontal plane with at least one point requires 16 points in both the x and y directions. The code was therefore modified to accommodate a 16x16x16 point arrangement. Incidentally, it should be noted that the larger number of axial points contributed to an improved representation of the axial power distributions.

The original version of the code was also modified on a semi-empirical basis in relation to the representation of the regions containing high control rod density and to the Haling technique⁽³⁾ to determine a power distribution capable of minimizing the core peaking factor throughout the cycle. The introduction of this technique considerably simplified the matter of obtaining the control rod withdrawal sequence, since it was reduced merely to the determination of a control rod pattern that would give that particular power distribution. The ERFLARE code was trimmed and verified against the data collected from the operation of the Garigliano reactor during the first two operating cycles and, in particular, from the first three gamma scanning measurements.

The CONDOR code

In PWR's the macroscopic power distribution can be calculated with a fair degree of approximation by assuming as valid the separability between the axial and radial flux components. The CONDOR⁽⁴⁾ code was used to determine the macroscopic radial power distribution in Trino Vercellese PWR as a function of burnup. This code presents the following main characteristics:

- (a) Two-dimension calculation technique.
- (b) Capability of representing fuel elements, fuel followers and structural material separately.
- (c) Capability of evaluating the effects of burn-up separately for each region.

The operation of this code can be summarized as follows:

- Storage in the library of the two-group microscopic cross-sections calculated a priori by means of an auxiliary code.
- Calculation of the macroscopic cross-sections of each region performed on the basis of the isotopic concentrations and microscopic cross-sections.
- Automatic search of the boron concentration required to make the reactor critical carried out by simplified diffusion calculations (i. e. harmonics method).
- Two-dimension diffusion calculation and determination of the macroscopic power distribution that is assumed as constant throughout the preset burn-up step.
- Determination of the isotopic concentrations of each region as a function of the attained burn-up.
- Repetition of the calculation cycle and use, when desired, of another library containing data calculated for the new mean burn-up level.

The validity of the calculation method was verified by using the operational data obtained with the Aeroball system on the first core of the Trino Vercellese reactor at different burn-up levels⁽⁵⁾. In these calculations the cross-sections introduced in the library were calculated by means of the RIBOT code for all the burn-up levels of interest.

The RIBOT Code

The RIBOT code⁽⁶⁾, developed by CNEN, performs zero-dimension calculations of K_{eff} or reactivity lifetime for water lattices; it utilizes a four-group scheme (1 thermal and 3 fast groups) lately modified to a five-group scheme (2 thermal and 3 fast groups) for the RIBOT-5 version, particularly suited for studying plutonium lattices. The nuclear constants are calculated from correlations resulting from a fitting procedure. Naturally, such a calculation model is not complete if it is not supported by experimental results or computations effected with more accurate models. The method proved to be fairly accurate in itself, and very useful especially for its flexibility in application to the nuclear design of fuel assemblies. The code collapses the three fast groups and/or the two thermal groups to yield four- (3 fast, 1 thermal), three- (1 fast, 2 thermal) and two-group (1 fast, 1 thermal) constants. Since this code constitutes the main sub-routine of the BURNY code, the main characteristics of the nuclear calculations are described below, together with the BURNY code.

The BURNY Code

The BURNY code^{(7), (8)} performs calculations of diffusion and lifetime in the x, y and r, z dimensions, utilizing a two-group scheme with energy cut-off at 0.625 eV. The neutron constants are calculated by means of the RIBOT code and are introduced into the EQUIPOISE code⁽⁹⁾ which performs the diffusion calculations.

The thermal constants are calculated by means of a correlation of the cross-sections based on the Wigner-Wilkins spectrum as a function of the following characteristic parameters: (a) absorption $1/v$ per atom of H; (b) U-235 concentration per atom of H; (c) Pu-239 concentration per atom of H; (d) absolute moderator temperature.

The correlation was carried out with recourse to the TEMPEST⁽¹⁰⁾ code; the cell disadvantage factors were calculated by means of the Amouyal-Benoist theory⁽¹¹⁾. With regard to the determination of the constants of the fast group, the method used is still quicker than the one used for the thermal constants, even though as accurate. To define Ombrellarotype⁽¹²⁾ microscopic cross-sections, the fast group is in turn subdivided into three sub-

groups, the lower limits of which are 183 KeV and 0.625 eV respectively. The cross-sections of the three sub-groups are then condensed in one fast group. The values of these microscopic cross-section are obtained by correlating the results of a series of calculations performed with the MUFT-IV code⁽¹³⁾ for a variety of water lattices. Instead, the resonance integrals relating to Sub-Group 3 were calculated case by case as a function of the characteristics of the lattice being considered.

The scheme for computing Pu-240 resonance integral has been deduced from heterogeneous Monte Carlo calculations. For small concentrations of Pu-240 a smooth transition curve was used from the infinite dilution resonance integral (8400 barns) to the actual heterogeneous value for high Pu-240 concentration.

Making use of the RIBOT technique which permits the neutron constants to be calculated in a relatively short time (about 0.1 sec), the BURNY code has the specific characteristic of calculating the constants of the individual regions after each irradiation interval.

The BURSQUID Code

This code is a link of the five-group RIBOT and SQUID codes⁽¹⁴⁾ and was prepared by ENEL in the framework of this Contract.

The five-group RIBOT code retains the main features of the calculation model of the two-group calculation method described earlier; the modifications involve only the subdivision of the thermal spectrum into two groups. In addition, the condensation of the fast groups is no longer carried out, and therefore the division in groups is the following:

Group 1	over 183 keV
Group 2	183 to 5.5 keV
Group 3	5.5 keV to 0.625 eV
Group 4	0.625 eV to 0.2 eV
Group 5	less than 0.2 eV

The macroscopic transfer cross-sections of Group 3 ($\Sigma_{R3,4}$ and $\Sigma_{R3,5}$) are obtained from Σ_{R3} on the assumption that the scattering is elastic and isotropic and due only to atoms of H.

With regard to downscattering phenomena of Group 4 and those of up-scattering from Group 5 to Group 4, it is assumed that these phenomena are due only to hydrogen according to the Wigner-Wilkins theory. The values of $\sum_{R4,5}$ and $\sum_{R5,4}$ were obtained by the TEMPEST code for a wide variety of water lattices.

For the diffusion calculation, use was made of the 10,000-mesh-point SQUID code which accepts a complete matrix of transfer cross-sections.

The GAM and FORM Codes

These codes represent a well-known tool for studying the slowing-down spectrum up to the thermal energies. Both codes are different versions of the early MUFT-IV code. GAM⁽¹⁵⁾ is the General Atomic version with a 100-group scheme for flux spectrum and is more flexible than the corresponding 54-group FORM⁽¹⁶⁾, especially for a more complete library.

The THERMOS Code

Also this code is a well-known tool for studying the neutron thermalization; it is based on the one-dimension calculations of the integral transport equation with isotropic scattering⁽¹⁷⁾. The version used was that with 50 groups up to 1.8 eV for flux spectrum analyses.

The GGC-II Code

This code is a combination of the GAM and GATHER codes of the General Atomic set⁽¹⁸⁾. The code is divided into three main parts: a fast section covered by GAM code, a thermal section covered by GATHER code⁽¹⁹⁾ and a section covered by COMBO code⁽¹⁸⁾ which combined fast and thermal cross-sections into single sets.

As compared with the earlier ones, this code is more flexible as it readily makes available lattice constants for each material, condensed in multi-group schemes and in some cases directly on punched cards which can be used as input for other codes such as the DTK.

The DTK Code

The DTK code⁽²⁰⁾ solves the transport equation in one dimension and in a multi-group scheme over the whole spectrum. It is used to calculate lattice constants in strong absorbing media and is particularly suited for the determination of control rod characteristics.

The TRITON Code

The TRITON code⁽²¹⁾ solves the diffusion equation in three dimensions with a maximum capacity of 20,000 mesh points. Calculations with this code are very time-consuming and therefore its use was limited to a few indispensable cases in which a high degree of precision in the local flux distribution prediction was required for particular core regions. This was the case, for instance, of the control-rod-induced perturbation of the radial power distribution in the 3x3 configuration of fuel assemblies in the open-vessel experiments carried out in the Garigliano reactor in 1968. The result thus obtained was utilized to optimize the level at which gamma scanning was to be performed after irradiation of the fuel assemblies.

2.2.2 Methods for Fuel Cycle Studies

Fuel cycles are studied both from the standpoint of reactivity life-time to determine the fuel loading strategy and from the strictly economic standpoint of evaluating the influence of the various parameters on the kWh cost and determining the industrial worth of plutonium. In addition to the tools previously described the MOVEL code was developed for the fuel assembly shuffling, and the AGENA and TITAN codes for the economic assessments. All these codes have been developed in the framework of this Contract.

Description of the MOVEL Code

The MOVEL code was prepared to study the Garigliano reactor fuel cycles and the details of the operating cycles. The inputs of this code are K_{eff} and isotope composition of the fuel as a function of irradiation for

various types of fuel assemblies up to a maximum of fifteen. The code calculates the variation in reactivity with time until the completion of each phase of the cycle, taking into account fuel shuffling and a pre-set flux distribution.

For each phase of the fuel cycle, the code calculates:

- (a) the criticality of the core by averaging the K_{eff} of the fuel assemblies by means of "statistical weights";
- (b) the burn-up of each assembly and of the average assembly in the core;
- (c) the isotope composition of all the discharged assemblies.

The K_{eff} variation versus burn-up, the power distribution and the statistical weights of the flux are calculated separately.

The validity of the code for the assessment of the reactivity lifetime during the single phases of the fuel cycle was confirmed by more refined calculations performed with other codes. The approximation resulting from the use of statistical weights is quite adequate for the evaluation of the average core K_{eff} (within +0.5%).

Description of the AGENA and TITAN codes

The AGENA code calculates the levelized average cost of kWh for both the uranium and the uranium-plutonium cycles as the ratio of the total net costs, inclusive of interest charges during the economic lifetime of the station, to the output of electricity in that period.

The total net costs are the algebraic sum of the following items:

- (a) Net fuel consumption cost, given by the difference between the value of the initial amount of fuel and the value of the final amount.
- (b) Plutonium credit.
- (c) Fuel cycle cost, representing the resultant of all expenses relating to fabrication, transport and chemical processing for recovery of fissile material.

The interest charges are the sum of interest on the capital invested during the accounting period for the fuel cycle considered. The present-worth method referred to a given date is used for the evaluation.

The TITAN code was developed to determine the industrial worth of plutonium in thermal reactors. It utilizes the AGENA code as a sub-routine and is based on the so-called indifference method. The indifference method consists in calculating the cost variations of a reference enriched-uranium cycle and a plutonium cycle as a function of the plutonium worth, taken as a variable, for a given reactor; the intersection of the two corresponding curves determines the industrial worth of plutonium for which use of either cycle is indifferent.

2.2.3 Method for the Verification of the Shutdown Margin

Reactivity determinations, such as for the shutdown margin and for the maximum control rod worth, are generally required for a core characterized by either fully withdrawn or fully inserted rod patterns. Since in this case it is fairly legitimate to assume that the flux can be separated in the two directions, axial and radial, it is possible to use the so-called "stick" technique for these calculations.

This technique entails a number of axial, one-dimension calculations for the various types of fuel assemblies in the core referred to as "sticks", including in this meaning also those assemblies that have the same characteristics but differ from one another either in the intensity of the control or in the void distribution or in irradiation. These calculations lead to the creation of files of assembly-wise lattice constants, averaged axially over the volumes and fluxes, for the various types of fuel assemblies and for different conditions of temperature, irradiation, power and control rod density. Once the files are complete, it is possible to examine any core situation with a two-dimension diffusion code in the x,y geometry (SQUID code), the input data of which are constituted by the lattice constants in the files.

The underlying criterion of the shutdown margin control system is that, in the cold condition and with all the xenon decayed, the reactor is to be subcritical by at least 1% with all the rods in and with the highest-worth rod all out.

For the preparation of the lattice-constant files it is necessary to calculate the different isotope concentrations in the fuel under the various core operating conditions, that is, for various values of void content and irradiation. Therefore, a number of assembly burn-up calculations are performed with the three-group (two thermal, one fast) BURSQUID code under operating conditions. On the basis of the concentrations so computed, the average assembly constants are determined, for the cold condition, as a function of irradiation, void content, presence of control rods, by means of the four-group (one thermal, three fast) BURSQUID code. It has, in fact, been found necessary to use a wider representation of the fast groups for the reactivity calculations than is necessary for the burn-up calculations. These constants are again averaged axially for several average assembly burnups, both with and without the presence of control rods by means of the FOG code⁽²²⁾; this code is capable of handling various types of reactor calculations based upon the solution of the one-dimensional diffusion equation. Consequently, for each real situation in which the reactor shutdown margin is to be verified there is a file of lattice constants to prepare the input data for the SQUID diffusion code, which is used to represent the core in cold condition with all the control rods in. On the basis of the resulting neutron flux distribution it is possible to identify the highest-worth control rod. The SQUID program is therefore run again for this rod all out, to check the core K_{eff} .

The validity of this approach was verified for several real conditions of the Garigliano reactor, and the calculations were always in excellent agreement with the experimental situations.

2.2.4 Methods for the Study of Transients

After the diffusion of the new fast digital computers equipped with plotters which record the variations of the significant quantities, it has become more and more frequent to use digital computers rather than analog computers for the solution of problems of dynamics. The possibility of representing a large number of physical phenomena very accurately--and thus

of improving the precision of the calculations--is the first advantage offered by numerical techniques. For instance, non-linearities, the representation of which by analog techniques is very expensive, can be represented very easily by digital computer codes. But it is mainly problems dealing with space-dependent phenomena, which were previously tackled with models of few nodes, that can now be processed in great detail.

In the case of nuclear reactor dynamics, the correct representation of the spatial distribution of neutron and heat fluxes in the core is very important, and the one-point model used with the analog technique is inadequate in most cases. Nevertheless, the analog-computer models are still valid for studies on the whole power plant.

The studies on transients and incidents described in Sub-Section 2.7 were conducted with an analog computer; the results can be used as the input data for a digital computer program to study core behavior in detail.

In order to have a suitable dynamics code to represent the behavior of the Garigliano core, ENEL developed the GARDIN code.

Description of the GARDIN Code

The code uses a model which represents a channel of average characteristics on which are imposed, as limiting conditions, the time-dependent variations of pressure (assumed uniform along the whole length), inlet temperature and flow rate. The fuel temperature variations and their influence on the Doppler effect are taken into account by means of the equation for heat transfer in the pellet, in the pellet-to-cladding gap, and in the cladding.

For the thermohydraulic analysis, the channel is divided into segments, represented by nodes, to each of which the equations of mass and energy conservation versus time are applied. The steam-to-water slip ratio is a function of the channel characteristics and of the local void fraction according to the Marchaterre and Hoglund⁽²³⁾ correlations. Based on the void and fuel temperature distributions, the code computes the nuclear

constants K_{∞} , M^2 , β_i and λ_i for each node from polynomial coefficients applied to irradiation, control rod density, void fraction and fuel temperature, for each type of fuel in the core.

The code computes the axial power distribution on an adiabatic model by means of one-neutron group equations. The power increase after a pre-established time interval is then calculated with kinetic equations using six groups of delayed neutrons. At this point, the fuel temperature, the axial void distribution and the new shape of axial power are calculated over again.

The input data for the code are:

- the geometric dimensions and densities of the channel materials;
- the percentages of different type of fuel in the core (≤ 10);
- the number of axial nodes (≤ 20);

and for each axial node of the channel:

- the exposure of each fuel type;
- the void fraction and control rod density averaged over the core lifetime.

To create the library of nuclear constants for each type of fuel assembly it is necessary to compute the constants for a set of values of the independent parameters by means of a cell code (for instance, RIBOT). From these data an auxiliary code derives the polynomial coefficients to be fed to the GARDIN code. This set of data allows many different cases to be set up quickly for different core conditions, without having to perform preliminary evaluations of the nuclear constants.

For each transient, the code also assesses the pre-existing steady-state conditions.

At each time interval and for each axial node the code prints out:

- the power shape factor;
- the temperature distribution in the fuel pellet, in the cladding and in the water;
- the heat flux on the cladding surface;
- the water and steam velocities;

- the enthalpy of the mixture;
- the void fraction.

The adequacy of the code in simulating transients of common interest in boiling water reactors was verified by comparing the theoretical neutron flux variations with those recorded during transients that occurred in the course of tests carried out on the Garigliano reactor (see Paragraph 2.6.5).

2.3 Prototype Plutonium Assemblies

The prototype plutonium assemblies fabricated for the Plutonium Irradiation Program are sixteen. Twelve assemblies, so-called "first set", were loaded during the summer 1968 shutdown, and by the end of their first operating cycle (June 1970) they had reached an average burn-up of 7000 MWd/MTM. Subsequently a second set of four assemblies fabricated entirely within the Community were loaded during the summer 1970 shutdown.

All the prototype assemblies have the same mechanical characteristics as the reload assemblies, as they were all fabricated and assembled with the same hardware supplied for the reload fuel. Each assembly consists of 64 Zircaloy-2-clad rods of the straight-through type arranged in an 8x8 square lattice.

2.3.1 Prototype Assemblies of the First Set

The prototype assemblies of the first set are of two types. One type contains only plutonium rods and is commonly called "standard type". The other contains plutonium rods at the center surrounded by uranium rods and is called "mixed type".

The eight standard-type assemblies were fabricated by the UKAEA to ENEL's nuclear design, whereas the four mixed-type assemblies were designed and fabricated by General Electric.

The fuel is in the form of cold-pressed and sintered pellets, except for the four mixed-type assemblies which also include rods fabri-

cated with hot-pressed pellets, vibrocompacted powder (VIPAC), and cold-pressed and sintered wafers. The pellets of the standard-type assemblies are all dished.

From the nuclear standpoint, these assemblies were designed to have the same performance as the reload uranium fuel. The nuclear design of the standard-type assemblies was based mainly on the following criteria:

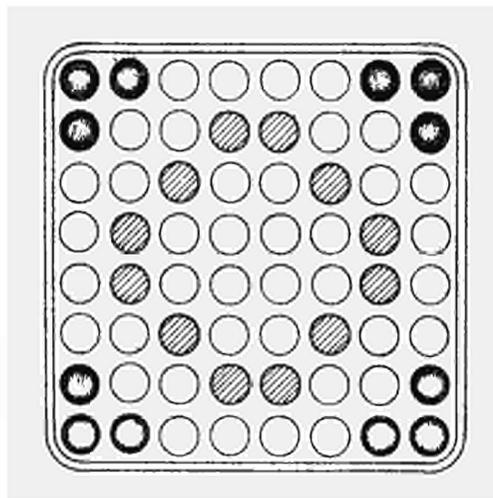
- (a) To have the same reactivity lifetime as the 2.3%-enriched-uranium reload assembly.
- (b) The local power peak due to the effect of the water gap and proximity of enriched-uranium assembly was not to exceed the value reached in the first-core assemblies.
- (c) The number of enrichments was to be the least compatible with the requirements (a) and (b) above.
- (d) Natural uranium oxide was to be the diluent for plutonium oxide.

The plutonium isotope composition in these prototype assemblies was as follows:

Pu-239	88.96%
Pu-240	9.77%
Pu-241	1.19%
Pu-242	0.08%

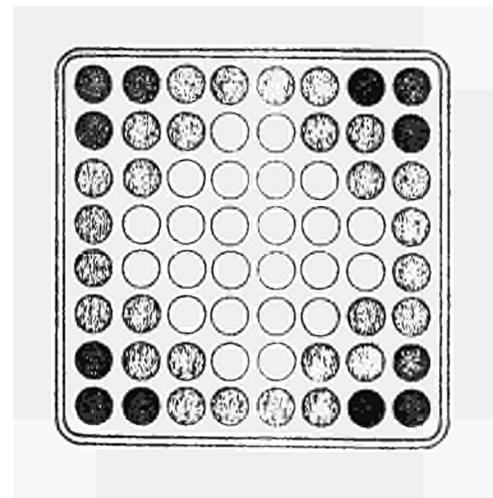
On the basis of the criteria described above and of the isotopic composition of the plutonium used, a three-concentration assembly was adopted. The distribution of the concentrations is shown in Fig. 2.1.

The mixed-type prototype assemblies, whose plutonium isotope composition is the same as for the standard-type assemblies, have four enrichments, two for the plutonium and two for the uranium. As shown in Fig. 2.2, twenty-four rods are plutonium and forty are uranium. The selected arrangement of rods (all uranium rods at the periphery) leads to less power perturbation at the interface between a normal uranium assembly and one containing plutonium, a higher average content plutonium per fuel rod and more flexible variation in the amount of plutonium deployed.



-  UO_2 - PuO_2 rods containing natural uranium and 2.89% of fissile plutonium
-  UO_2 - PuO_2 rods containing natural uranium and 1.80% of fissile plutonium
-  UO_2 rods enriched to 2.41% in U-235
-  UO_2 rods enriched to 1.83% in U-235

Fig. 2-2 - DISTRIBUTION OF ROD ENRICHMENTS IN THE "MIXED-TYPE" PLUTONIUM PROTOTYPE ASSEMBLY



-  Rods containing 2.85 w/o of fissile plutonium
 -  Rods containing 1.40 w/o of fissile plutonium
 -  Rods containing 0.74 w/o of fissile plutonium
 -  Rods containing 0.74 w/o of fissile plutonium
- Average fissile plutonium content : 1.82%

Fig. 2-1 - DISTRIBUTION OF PLUTONIUM CONTENT IN THE "STANDARD-TYPE" PLUTONIUM PROTOTYPE ASSEMBLY

2.3.2 Prototype Assemblies of the Second Set

Following the decisions made jointly by EURATOM and ENEL to proceed with the fabrication of a second set of prototype plutonium assemblies within the European Community, views were exchanged in 1967 with specialized Community manufacturers interested in fabricating plutonium assemblies. On the basis of the discussion, a technical specification for UO_2 - PUO_2 fuel assemblies was prepared⁽²⁴⁾, which took into account the main particular requirements of the Community manufacturers. In January 1968, ENEL issued an enquiry for the supply of four prototype plutonium assemblies to be loaded into the Garigliano reactor during the scheduled shutdown in Spring 1970.

From the analysis of the economic bids received, it emerged that the cost of the supply, based on the facilities at present available in the Community for the fabrication, was much higher than the cost of the reload uranium assemblies. In order to reach a solution that would be acceptable to ENEL and would at the same time permit the Community industries to acquire experience on plutonium fuel fabrication, ENEL preferred to have a few manufacturers in the Community separately fabricate the plutonium rods required for four 8x8 assemblies, for which ENEL would make available standard hardware procured with the reload uranium assemblies. In this connection, ENEL reached an agreement with the Community manufacturers ALKEM and Belgonucléaire, for the fabrication of the plutonium rods which were subsequently assembled into finished fuel assemblies by Fabbricazioni Nucleari at KRT works. Fabbricazioni Nucleari, which supplied the fuel for the second reload of the Garigliano reactor, also procured the hardware and fabricated the enriched-uranium spacer-capturing rods like those used for the normal reload fuel.

ENEL personnel followed all the phases of fabrication and testing at the manufacturers' shops, chiefly to ensure proper coordination of the activities of the three manufacturers.

The criteria adopted in the nuclear design of this second set of four assemblies were similar to those of the eight standard-type assemblies of the first set. The plutonium used for the fabrication of these assemblies had the following isotopic composition:

Pu-239	83.32%
Pu-240	14.31%
Pu-241	2.13%
Pu-242	0.24%

As will be noted, the content of Pu-240 is higher in the assemblies of the second set (14.31% versus 9.77%); this higher content should have a negligible influence on the local power peak factor. Therefore the fissile plutonium content in these assemblies was chosen equal to that already adopted for the standard-type assemblies of the first set (see Fig. 2-1).

The main difference from those assemblies lies in the use of an enriched-uranium capture rod as one of the four centrale rods.

This solution was adopted to simplify the fabrication process, and because recent on-site rod gamma scanning measurements have shown that plutonium causes power peaks higher than enriched uranium, in proximity of the connectors of the spacer rod itself.

The use of the enriched-uranium spacer capture rod in a plutonium assembly appears not to present inconveniences as to power distribution and its power density seems to be more favourable than that of the same rod in a reload fuel assembly (Fig. 2-3).

2.4 Experimental Activities

To gather useful information on the main technical aspects associated with the introduction of plutonium in power reactors, experiments were carried out in the Garigliano reactor with prototype plutonium assemblies. The purpose of these experiments was to measure the criticality of different arrays of fuel assemblies, including plutonium assemblies, and the power distribution inside uranium and plutonium fuel assemblies. The results of these measurements were used to verify the adequacy of the criteria followed in the design of the prototype assemblies and to check

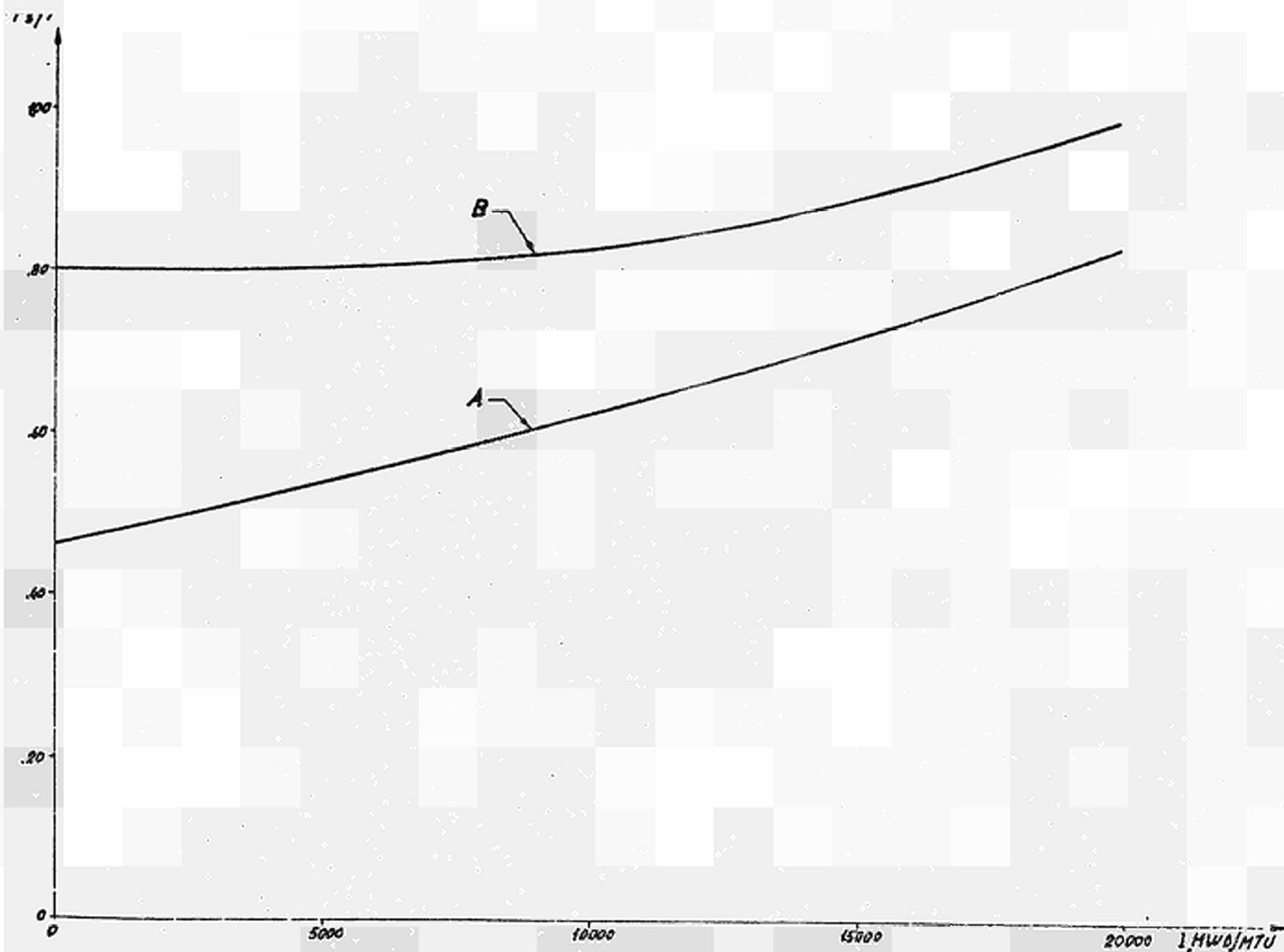


Fig. 2-3 - POWER DENSITY OF THE ENRICHED URANIUM SPACER CAPTURE ROD IN A PLUTONIUM ASSEMBLY (CURVE A) AND IN A RELOAD ASSEMBLY (CURVE B). P_s = SPACER CAPTURE ROD POWER DENSITY. \bar{P} = AVERAGE POWER DENSITY OF THE FUEL ASSEMBLY.

the magnitude of technical difficulties associated with the use of plutonium. The experiments brought to the fore certain aspects that must be taken into consideration for a better prediction of plutonium fuel behavior and increased the bulk of experimental data available for cross-checking and trimming the calculation methods for plutonium systems.

The measurements were performed during two station shutdowns for refueling, respectively summer 1968 and summer 1970.

Of the experimental data obtained under the subject Contract, also interesting are the results of the post-irradiation examination of an enriched-uranium fuel assembly, A-106, discharged from the Garigliano reactor at an average burn-up of about 10,000 MWd/MTU.

2.4.1 Open-vessel Experiments in the Garigliano Reactor During the Shutdown of 1968

During the shutdown of the Garigliano reactor for refueling in summer 1968, a series of experiments were performed on critical assemblies containing reload enriched-uranium fuel assemblies and prototype plutonium assemblies. The purpose of these experiments was to check the expected performance of plutonium fuel assemblies by assessing the accuracy of the calculation methods used in the nuclear design of plutonium fuel assemblies. Since they were performed on full-scale assemblies, they provide an integration of the experimental data previously obtained on critical facilities. In particular, these experiments allow an assessment of the accuracy of the calculations for the determination of criticality conditions and power distribution in mixed lattices, as well as the evaluation of the effects of water gaps and contiguity of plutonium assemblies to enriched-uranium assemblies (power sharing).

The experiments can be subdivided into two groups:

- (a) Criticality experiments
- (b) Measurement of the local power distribution through gamma scanning on slightly irradiated fuel rods removed from the assemblies.

The criticality experiments were performed in the reactor pressure vessel where a sufficiently large number of fuel assemblies was discharged to completely clear a core quadrant (see Fig. 2.4). The critical configurations were separated from the irradiated assemblies by a water belt of over 60 cm which was enough to segregate the measurement area so as to exclude any neutron interaction and to lower the gamma background to acceptable limits.

In order to acquire exhaustive information from the criticality experiments, it was deemed advisable to begin with a critical configuration with all fresh enriched-uranium fuel assemblies (2.3% U-235) and subsequently to replace an uranium assembly in different positions with a plutonium assembly so as to progressively obtain different configurations.

The criterion of retaining the same geometry in all the critical arrays would permit any error in the evaluation of neutron leakage to be approximately the same, thus permitting sufficiently precise estimate of the reactivity variations according to the type and location of the substituting fuel assembly.

The various configurations obtained with the successive replacements were (Fig. 2-5):

Configuration II	A standard plutonium assembly loaded into a corner position
Configuration III	The standard plutonium assembly shifted to the opposite corner position, to check reproducibility
Configuration IV	The standard plutonium assembly loaded into a more central position
Configuration V	A mixed-type plutonium assembly substituted for the plutonium assembly in the preceding configuration.

It was possible to pass from one configuration to the next with the control rods at the same levels, so that the ΔK involved in a replacement could be assessed on the basis of the difference between the related periods.

The K_{eff} value for the critical configuration formed by seven enriched-uranium fuel assemblies and with all the control rods out was estimated to be 1.0109 ± 0.0050 .

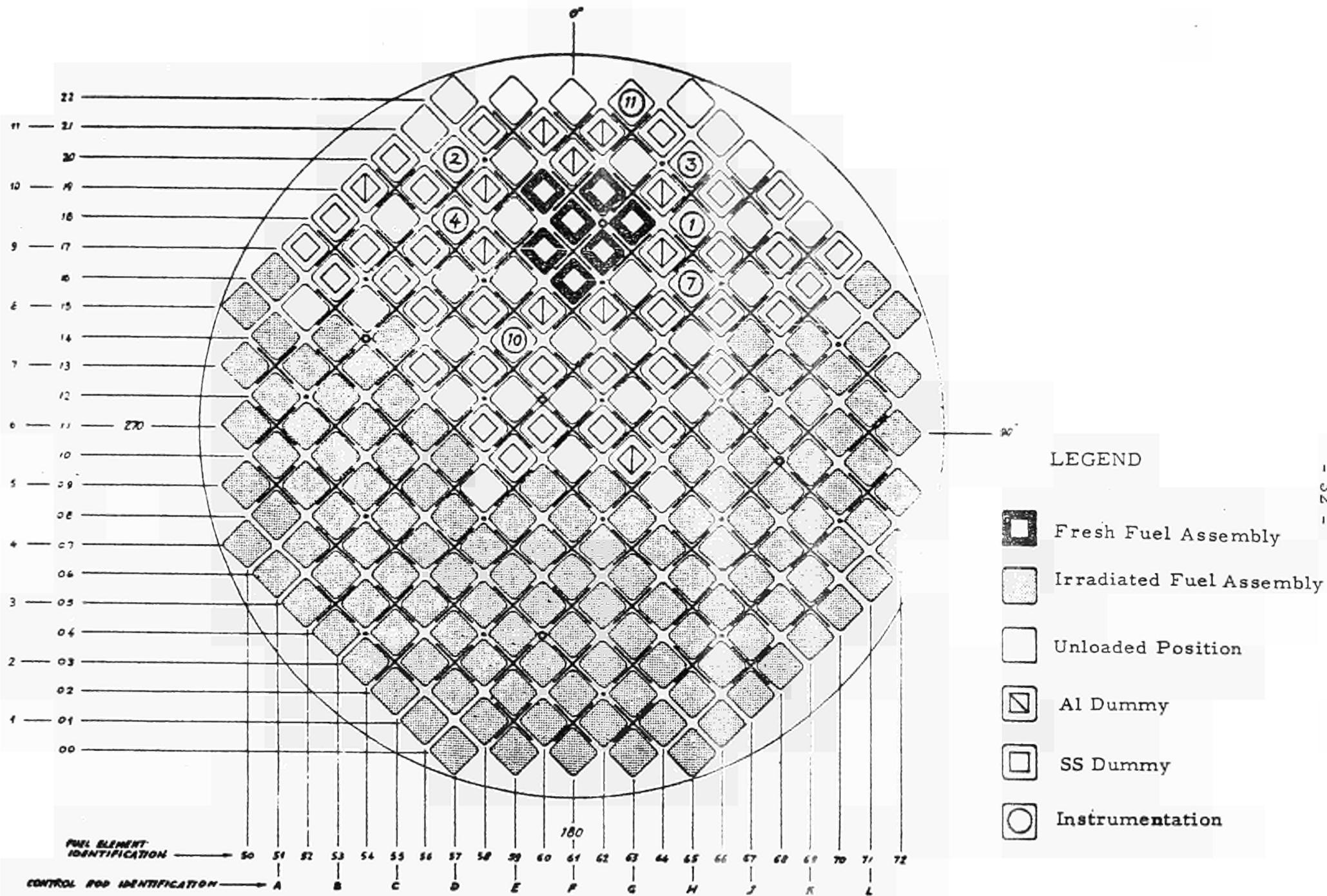
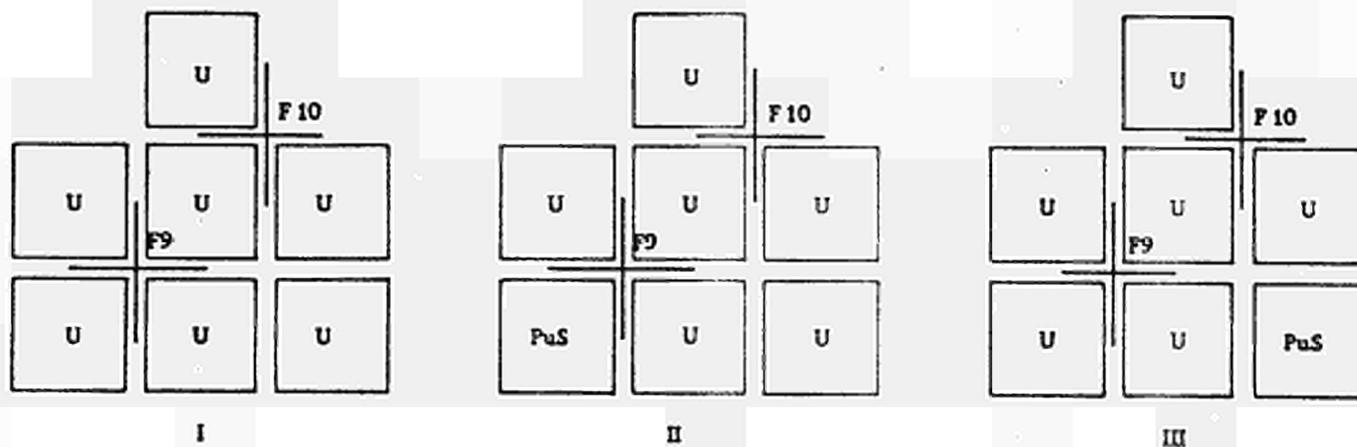


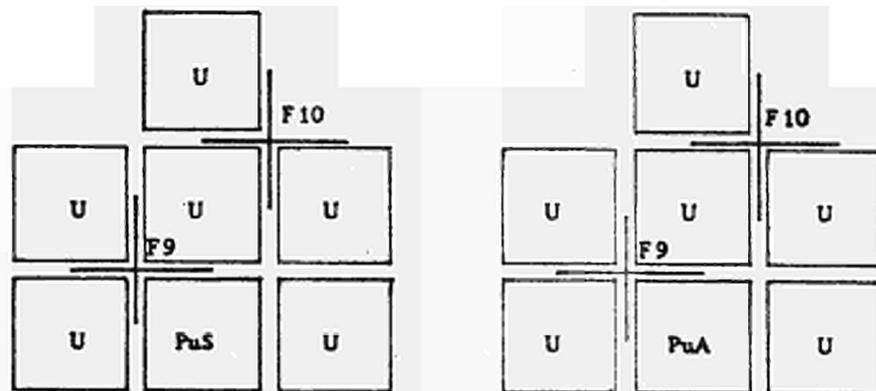
Fig. 2-4 - LOCATION OF THE CRITICAL ASSEMBLIES IN THE VESSEL



I

II

III



IV

V

Fig. 2-5 - CRITICAL CONFIGURATIONS

Table 2-I shows the differences in K_{eff} values obtained from the periods measured for the various configurations with the control rods at the same level.

TABLE 2-I

Configurations		$\Delta K = K_{effA} - K_{effB}$ pcm	$\pm \epsilon$ pcm	Control rod position (notch)		
A	B			F9	F10	Remainder
II	I	246.1	4.9	17	17	35
III	I	229.7	5.2	17	17	35
V	I	189.4	4.5	17	17	35
III	II	-16.5	4.1	17	17	35
IV	II	149.7	3.6	16	16	35
V	II	-41.2	4.4	16	16	35
V	II	-56.7	3.0	17	17	35
V	II	-46.3	4.1	17	16	35
V	III	-33.6	5.0	16	16	35
V	III	-40.2	3.4	17	16	35
V	IV	-194.2	24.4	16	15	35
V	IV	-190.9	5.6	16	16	35

The power distribution measurements consisted in slightly irradiating nine fuel assemblies in a small pile and in monitoring, by means of the NaI-detector technique, the 1.6-MeV gamma activity of the Ba-140/La-140 chain on the individual rods after disassembly of three selected fuel assemblies.

The La-140 gamma scan was preferred to the measurement of the total fission product gamma activity because the correlation factor between gamma activity and power can be calculated with fair approximation for different types of rods. Indeed, a few preliminary measurements of the total gamma activity, performed on short fuel segments previous to the formation

of the critical arrays, had indicated a substantial difference between the decay laws of the various types of rods.

Symmetry requirements led to the choice of a configuration of nine fuel assemblies in a 3x3 array: four were enriched-uranium assemblies, four were mixed-type plutonium assemblies and the one in the center was a standard-type plutonium assembly (Fig. 2-6).

The four uranium assemblies and the central plutonium assembly were housed in stainless steel rather than Zircaloy sheaths, to limit the excess reactivity of the assembly and thus the degree of control rod insertion. The use of the stainless steel sheaths lowered the K_{eff} to 1.006.

Under these conditions it should have been possible to obtain a sufficiently flat radial power distribution at a level of interest without any disturbance from the control rod bank, almost fully withdrawn. The configuration thus selected was characterized by a high degree of symmetry and by the presence of all three types of assemblies in one octant. This permitted the gamma scanning to be concentrated on the rods of an octant and to disassemble only three assemblies. With the control rod bank nearly all out (70 cm insertion, corresponding to $\Delta K=0.006$), the 3x3 configuration reached criticality, thus confirming the theoretical prediction.

The nine-assembly array was irradiated at a neutron flux of about 10^9 nv for about one hour. These conditions represent a satisfactory compromise between the requirement of sufficient La-140 gamma activity for the measurement, and the necessity of keeping the radiation level low enough to permit rod handling without undue exposure of the personnel. A set of specimens was placed near the boundary of the 3x3 configuration in order to check the gamma activity decay laws; the use of specimens was suggested by the requirement of reducing rod handling to a minimum.

After a decay period of 14 days, three assemblies were decontaminated and transferred one at a time to the fresh fuel vault where they were disassembled and gamma-scanned.

-  Enriched uranium fuel assembly
-  Standard plutonium fuel assembly
-  Mixed uranium-plutonium fuel assembly
-  Al dummy assembly
-  Rod sample position
-  Source
-  Guide tube

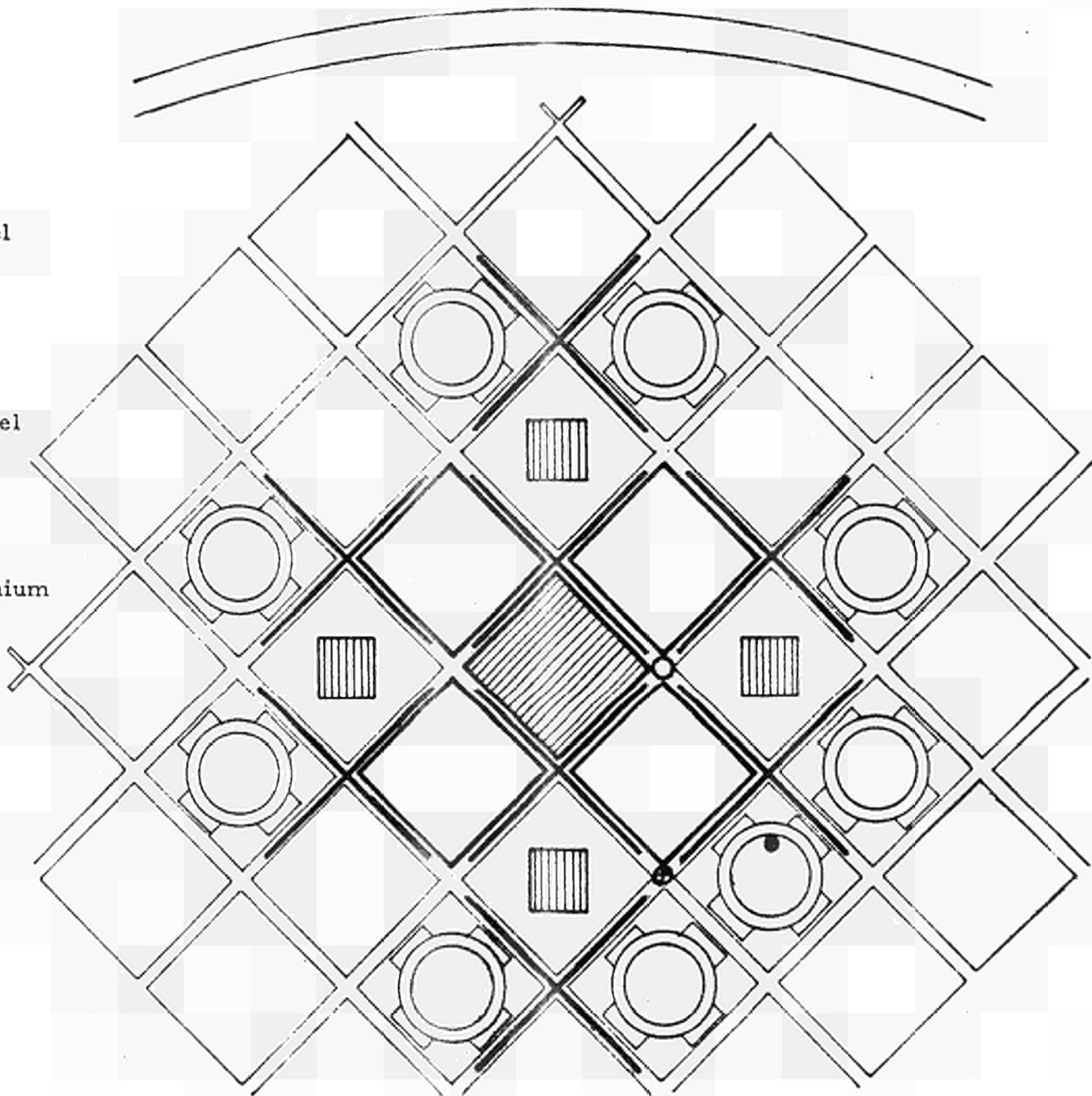


Fig. 2-6 - 3x3 FUEL ASSEMBLY CONFIGURATION

A total of 125 rods were scanned at two levels: at the top of the core in the uncontrolled region and at the bottom, in the fully controlled region.

The measured La-140 counting rates were corrected for the background and for the activity of the fuel rods before irradiation and were all brought back to the reference time (14 days after irradiation) by using the Ba-140/La-140 chain decay law obtained from periodical scanning of the three specimens over the whole duration of the experiment. The resulting decay law is an exponential with a half-life of 12.6 days versus the figure of 12.8 days given in the literature.

The La-140 gamma activity thus obtained was converted into power density by means of the conversion factors evaluated for each rod on the basis of the macroscopic fission cross-sections of the individual isotopes and the related fission yields. The cross-sections were obtained with the calculation method used for the programming of the experiment.

At the level not affected by the control rods, the 3x3 configuration is characterized by a diagonal symmetry; in giving the experimental distribution of the power density the data relating to each set of symmetrical positions were averaged. The values indicated in Fig. 2-7 are affected by a standard deviation of $\pm 0.7\%$, which includes the random error, the error associated with the correction for decay and the error due to engineering tolerances.

At the level influenced by the control rods the gamma activity of the fuel rods was very low because of the high depression in the neutron flux caused by the control rod bank; consequently there was an appreciable degree of uncertainty in these data.

The values of the La-140 gamma activity compared to those of the total gamma activity showed a systematic discrepancy which appears to be mainly due to the uncertainty in the total gamma activity data resulting from the strong component of the rod background.

As a result of carrying out the experiment on actual reactor fuel and hardware rather than on ideal critical facilities, a number of inter-

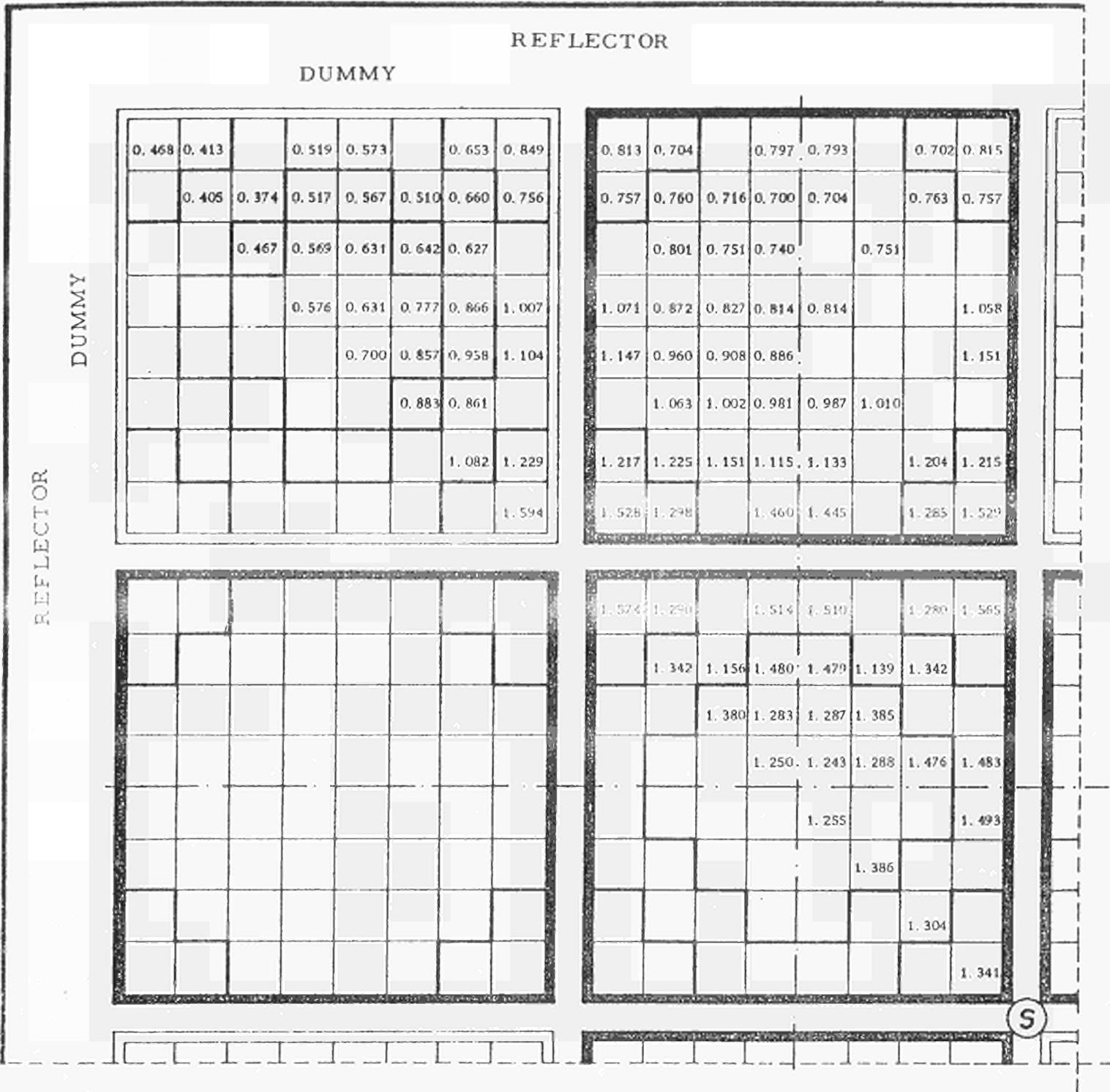


Fig. 2-7 - LEVEL NOT AFFECTED BY THE CONTROL RODS: EXPERIMENTAL DISTRIBUTION OF THE ROD POWER DENSITIES

esting, fine-structure effects were observed. With regard to axial power distribution, they included:

- (a) The depression ($\sim 4\%$) noticed in all the fuel rods not adjacent to the spacer-capturing rod, due to the steel spacer grids (Fig. 2-8)
- (b) Slightly pronounced peaks ($\sim 4\%$) (Fig. 2-8) present only in the rods adjacent to the plutonium spacer-capturing rod; these peaks are caused by a thermal flux rise in the Zircaloy end connectors. The effect is not visible in enriched-uranium spacer-capturing rods where the thermal flux increase in the end connectors is probably smaller and is compensated by the absorption in the grids.

In the radial power distribution, the following was observed:

- (c) The effect on rod power densities due to rod manufacturing tolerances. This effect was evaluated by comparison of a significant number of symmetrical rods. The value obtained (average: $\pm 0.5\%$) is net of the standard deviation of the measurement.
- (d) Strong effect of the neutron source on the power level of the corner rod adjacent to the source (about a 15% local reduction) (Fig. 2-7).
- (e) Power depressions in the peripheral fuel rods closest to the aluminum dummy assemblies (Fig. 2-7).

2.4.2 Gamma Scanning Measurements on the Garigliano Reactor Core Containing Twelve Plutonium Prototype Assemblies

During the shutdown of the Garigliano reactor for refueling of June 1970, a number of fuel assemblies sufficiently representative of the core was subjected to gamma scanning in order to determine, among the other, the actual operating conditions of the plutonium assemblies in the last month of operation before shutdown, and compare them with the design predictions (25). In particular, the power distribution through the core was determined by detecting the La-140 gamma activity from 52 fuel assemblies by means of the very-high-resolution solid-state technique (Ge-Li detector)

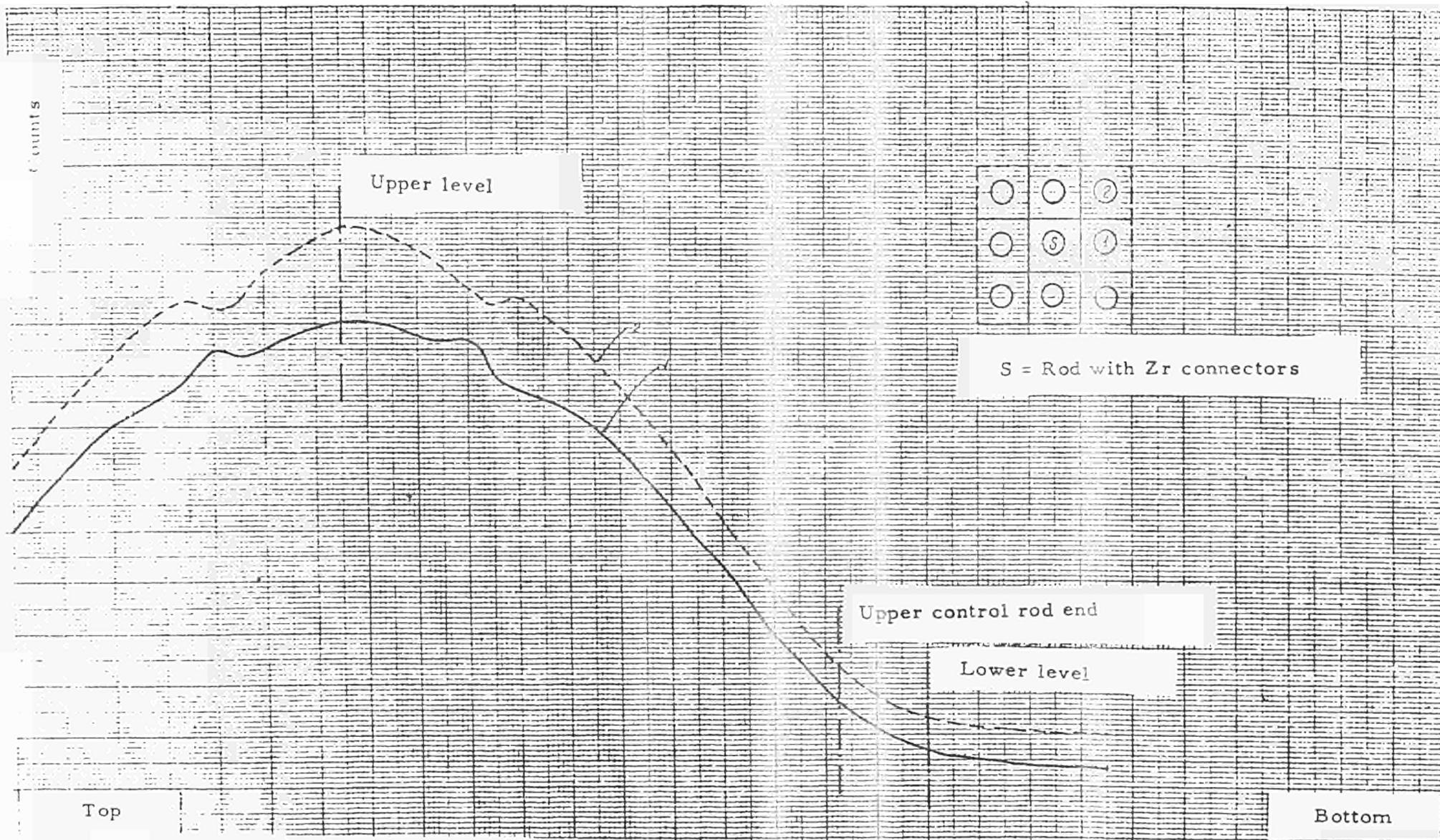


Fig. 2-8 - AXIAL POWER DISTRIBUTION IN THE 3x3 FUEL ASSEMBLY CONFIGURATION

which proved to have greater flexibility in respect to other techniques for measurements to be carried out during plant shutdowns.

This technique was used in the Garigliano plant by placing all the measuring instrumentation outside the fuel pool in order to avoid or minimize interferences with refueling operations. To this end, a circular hole was bored in the south wall of the fuel pool at about 2 m above the fuel racks in which irradiated assemblies are stored. This permits the measuring equipment to be located outside the pool on a platform, while only the equipment to move the fuel assembly in front of the collimator is located in the pool. This arrangement permitted also the elimination of the gamma background effect due to the irradiated assemblies in the pool. Fig. 2-9 shows the equipment lay-out.

In the hole, which extended to the external surface of the pool liner, a stainless steel blind pipe was inserted, anchored to the pool wall and sealed. A first collimating system was placed in the pipe, consisting of two cylindrical lead blocks that were rigidly held together by means of stainless steel clamps. During the measurements, a second collimator was placed between the pipe and the detector.

The detector was constituted of a Ge-Li crystal of an active volume of 30 cm³, equipped with a Dewar cryostat for temperature control; the instrumentation was calibrated by means of standard sources of about 10 microcuries each.

The fuel assembly handling and positioning equipment was constituted of a guide sliding on rails to move the assembly in the vertical direction, and a motor-operated chain drive. The guide supports the assembly also during its rotation around the main axis and during horizontal movement.

In preparing the power distribution measurement program, the following conditions were borne in mind:

- (i) the time available for the gamma scans was obviously limited;
- (ii) the core region to be scanned was to be sufficiently large to provide meaningful experimental data for comparison with the calculated values;

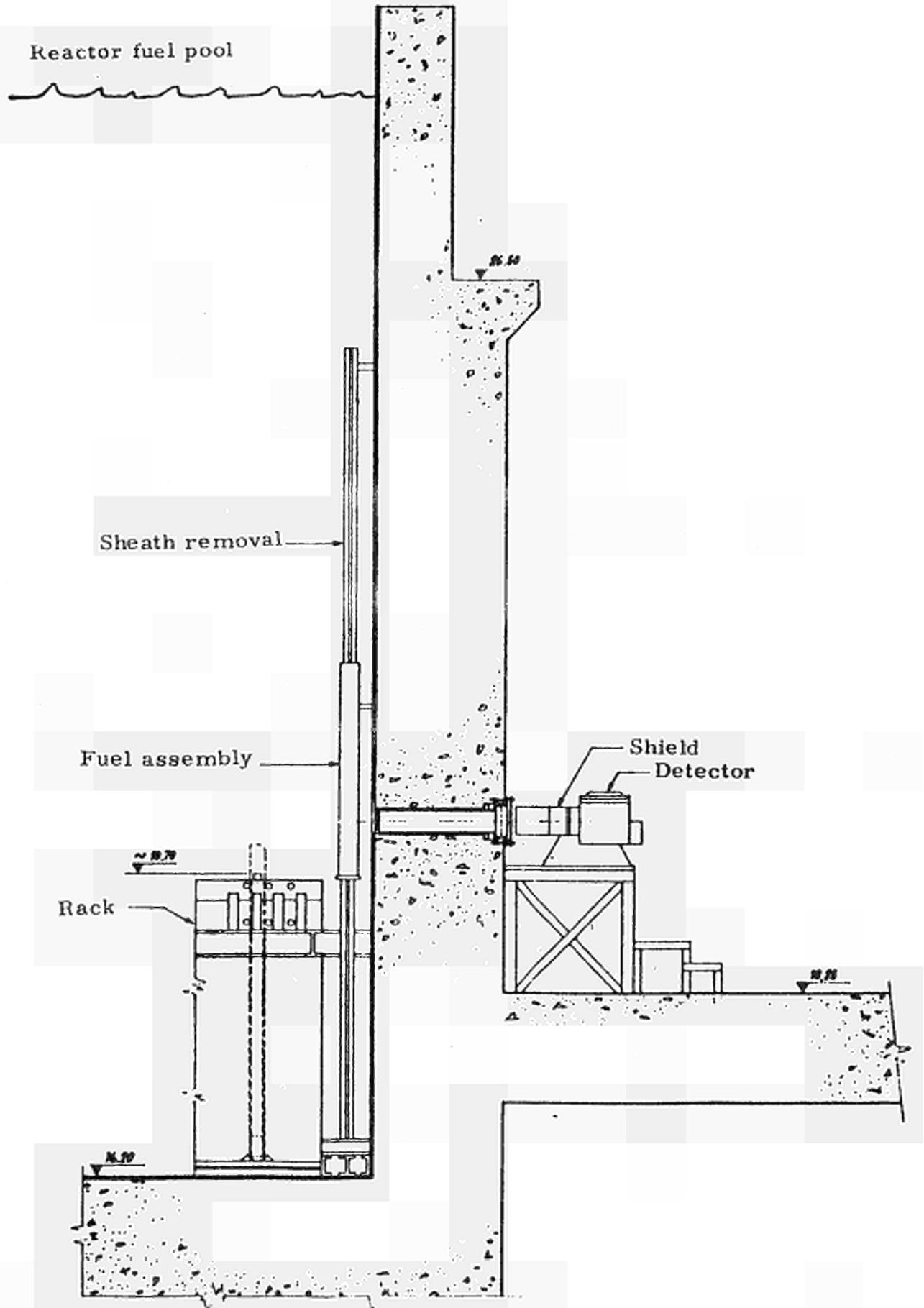


Fig. 2-9 - GAMMA SCANNING EQUIPMENT

- (iii) the region was to be located so as to include control-rod-affected fuel assemblies, standard and mixed-type plutonium assemblies and groups of four assemblies with and without plutonium assemblies centered in respect of the positions of the normal in-core flux monitors;
- (iv) the measurements were to be supplemented with a sufficient number of gamma scans on assemblies in symmetrical positions in the core.

The simultaneous determination of the power distribution by means of gamma scanning of the four assemblies referred to in (iii) above and by means of flux wire readings permits a comparison of the experimental values of the neutron flux at the position of the in-core monitor with the theoretical values.

The additional gamma scans mentioned in item (iv) are required to check the true symmetry of the core and to determine the difference between the power generated by the standard and mixed types of plutonium assemblies. On the basis of the conditions listed above, it was decided that gamma scanning should be performed on fifty-two assemblies placed in the positions shown in Fig. 2-10.

The actual measurements were preceded by the preliminary tests in conditions as close as possible to the actual working conditions with an aim at obtaining an advance indication of the difficulties that could be encountered in the measurements and to have a reliable check of the accuracy of the measurements to prevent an undetected error on all the experimental data, thus impairing the significance of the program.

The La-140 gamma activity was scanned at several elevations and from the four corners of the assembly at each elevation. This procedure was preferred to continuous scanning along the assembly length, as experience had indicated that scanning at an adequate number of elevations (eight for the Garigliano fuel assemblies) can give the assembly power with better accuracy.

For the sixteen assemblies adjacent to the in-core monitors for which a better knowledge of the axial distribution was desired, the number of eleva-

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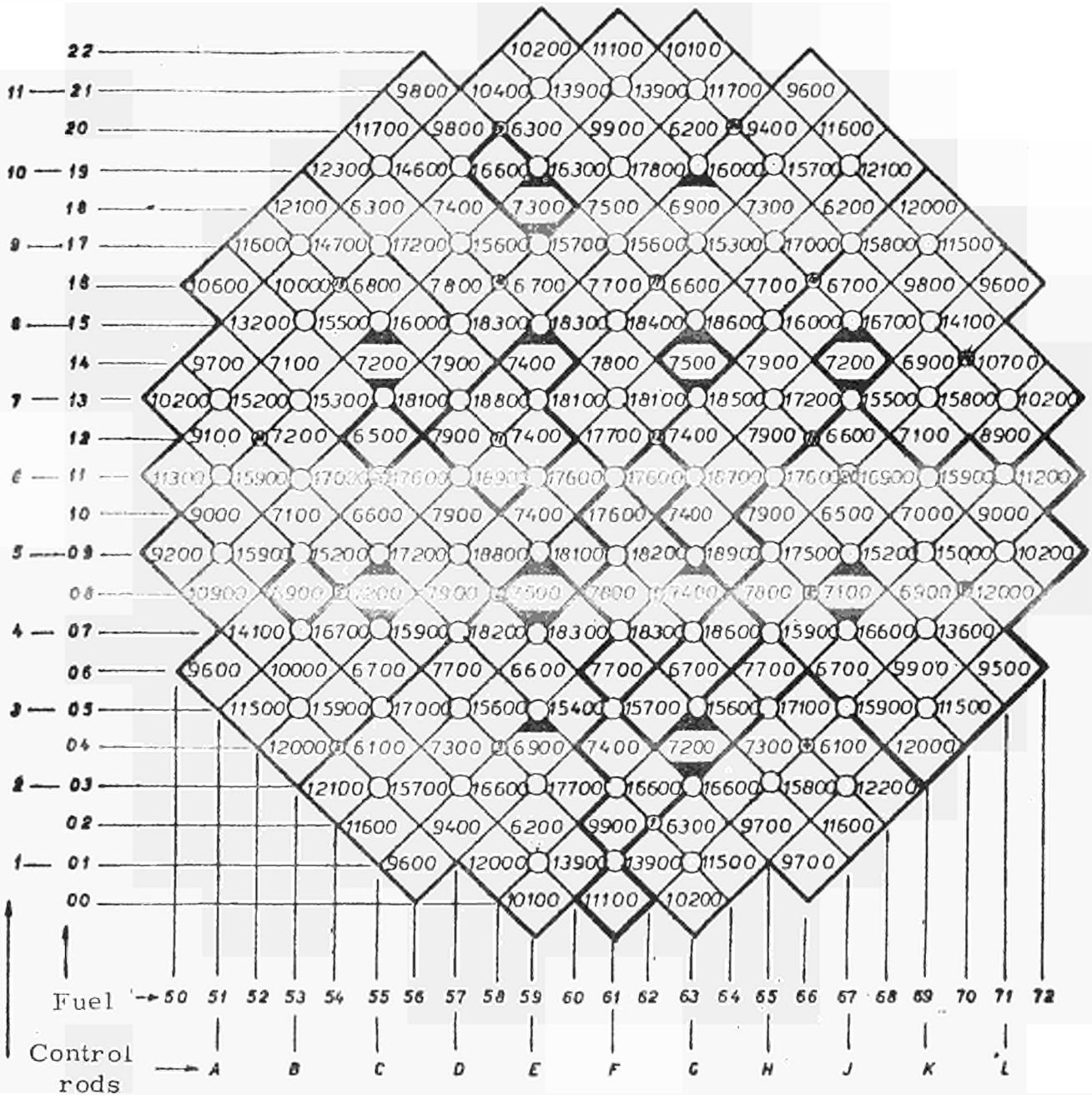


Fig. 2-10 - BURNUP DISTRIBUTION OF THE FUEL ASSEMBLIES ON JUNE 13, 1970, AS EVALUATED BY FLARE. THE HEAVY LINE DELIMITS THE ASSEMBLIES SUBJECTED TO GAMMA SCANNING.

-  Plutonium assembly
-  Mixed assembly

tions at which the measurements were to be performed was raised to twelve. These elevations were selected so that they would be sufficiently far from the spacer grids to avoid the depressions produced by the grids in the power distribution. The magnitude of these depressions referred to the total assembly power was subsequently assessed by means of very fine axial scans (every centimeter) on an assembly of the first-core load and one of the first reload.

The 1.6-Mev gamma activity of La-140 was obtained by integrating the area of the corresponding peak, as supplied by the multi-channel analyzer corrected for the background and for the Compton effect; other corrections were made later to allow for the instrumentation dead times and La-140 decay. The relevant values of each elevation of the assembly were summed and converted to power by means of a factor that took into account the different concentrations of fissile isotopes, the respective yields and the reciprocal shielding effect of the rods; the latter correction (shielding effect) was very important because the scanned assemblies differed in enrichment, number of rods, fissile material, and burnup. From the power values relating to the various elevations, the overall assembly power was derived. The macroscopic radial power distribution was obtained by normalizing the fifty-two experimental values to the corresponding calculated values. Fig. 2-11 shows the theoretical power distribution together with percentage deviation for the measured values.

2.4.3 Determination of Burnup and Heavy Isotope Content in a Uranium-Enriched Fuel Assembly Irradiated in the Garigliano Reactor

To complete and supplement the available experimental data to be used for the verification of the precision of the calculation technique on irradiated fuel, post-irradiation measurements of burnup and isotopic content were taken on an enriched-uranium assembly irradiated at about 10,000 MWd/MTU in the Garigliano reactor.

The fuel rods of this assembly (A-106) were sufficiently irradiated to contain an appreciable amount of plutonium and the assembly itself was exposed in a fairly central position of the core so that the power tilting effect could be expected to be rather limited.

Since the assembly was removed from the reactor during the May 1967 shutdown, in view of the long period of decay elapsed before the beginning of the hot cell measurements, the program was limited to the measurements of burn-up and fuel isotopic composition.

The program was logically broken into two parts: non-destructive measurements based on gamma spectrometry and gamma scanning, and destructive measurements of burn-up and isotopic composition based on gamma, alpha, and mass spectrometry of dissolved samples.

The non-destructive gamma-activity measurements were to determine the burn-up distribution at a pre-set level on the largest possible number of rods. These measurements are actually easier and more amenable than the destructive measurements, even though they can not provide the absolute value of burn-up, because in general the results are dependent on the geometry of the measurement itself⁽²⁶⁾. On the contrary, destructive measurements permit an assessment of the heavy atom content, specifically the depletion of U-235 and generation of plutonium, and offer the possibility of determining burn-up through the measurement of the concentration of a stable fission isotope, such as Nd-148⁽²⁷⁾.

Therefore, the absolute value of burn-up was determined on a limited number of rods and then correlated with the results of the non-destructive measurements of burn-up distribution. In addition, the availability of the dissolved fuel samples prepared for the destructive measurements offered the possibility of checking the burn-up by means of an additional destructive measurement, that is, through the determination of the specific activity of Cs-137.

For the purpose of adjusting the calculation method, it was of interest to determine the burn-up reached by the assembly at a given level, in order to apply the x-y geometry technique. In brief, the non-destructive measurements were taken on 34 rods at two levels corresponding to the positions of

the in-core instrumentation (levels C and D; see Fig. 2.12), whilst the destructive measurements were limited to 18 fuel sections taken all at the same level (level C). For a check of the axial distribution calculations, two rods were subjected to gamma scanning over their entire length.

The measurements permitted an integral and analytical verification of the calculation models and methods used in the nuclear design of the plutonium fuel assemblies. More specifically, the purpose of the measurements was to ascertain that the calculation technique was capable of:

- determining the correct burn-up distribution among the various rods;
- adequately assessing the concentrations of the heavy nuclides as a function of burn-up, with special reference to plutonium, while allowing for the effects of spectrum variations.

From the flow diagram of Fig. 2-13 the sequence of analyses performed can be seen.

The burn-up distribution in the fuel assembly was determined through the monitoring of the gamma activity from each fuel rod at the same axial position, by means of high-resolution gamma spectrometry (Ge-Li detector). This non-destructive technique is based on the possibility of correlating the gamma activity of a selected fission product to burn-up, so leading to a relative burn-up distribution.

The isotopes selected for the burn-up distribution determination were:

<u>Isotope</u>	<u>Energy, keV</u>	<u>Half-life, yrs</u>
Ru-106/Rh-106	512	1.008
Cs-137	662	30.600
Ce-144/Pr-144	2186	0.778

A typical spectrum obtained from a fuel rod is shown in Fig. 2-14 where the peaks due to these isotopes are clearly recognizable. The calculation procedure required for the interpretation of the spectra was con-

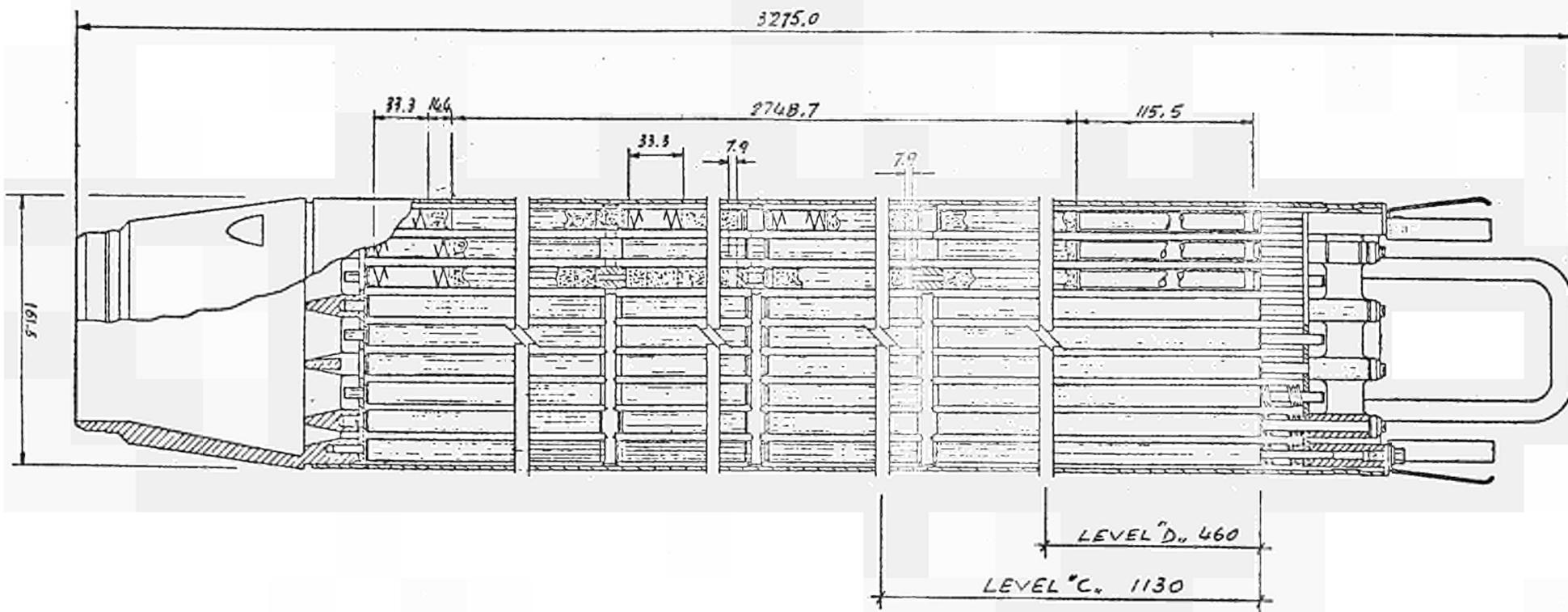


Fig. 2-12 - GARIGLIANO FIRST CORE FUEL ASSEMBLY
SHOWING IN-CORE POSITION

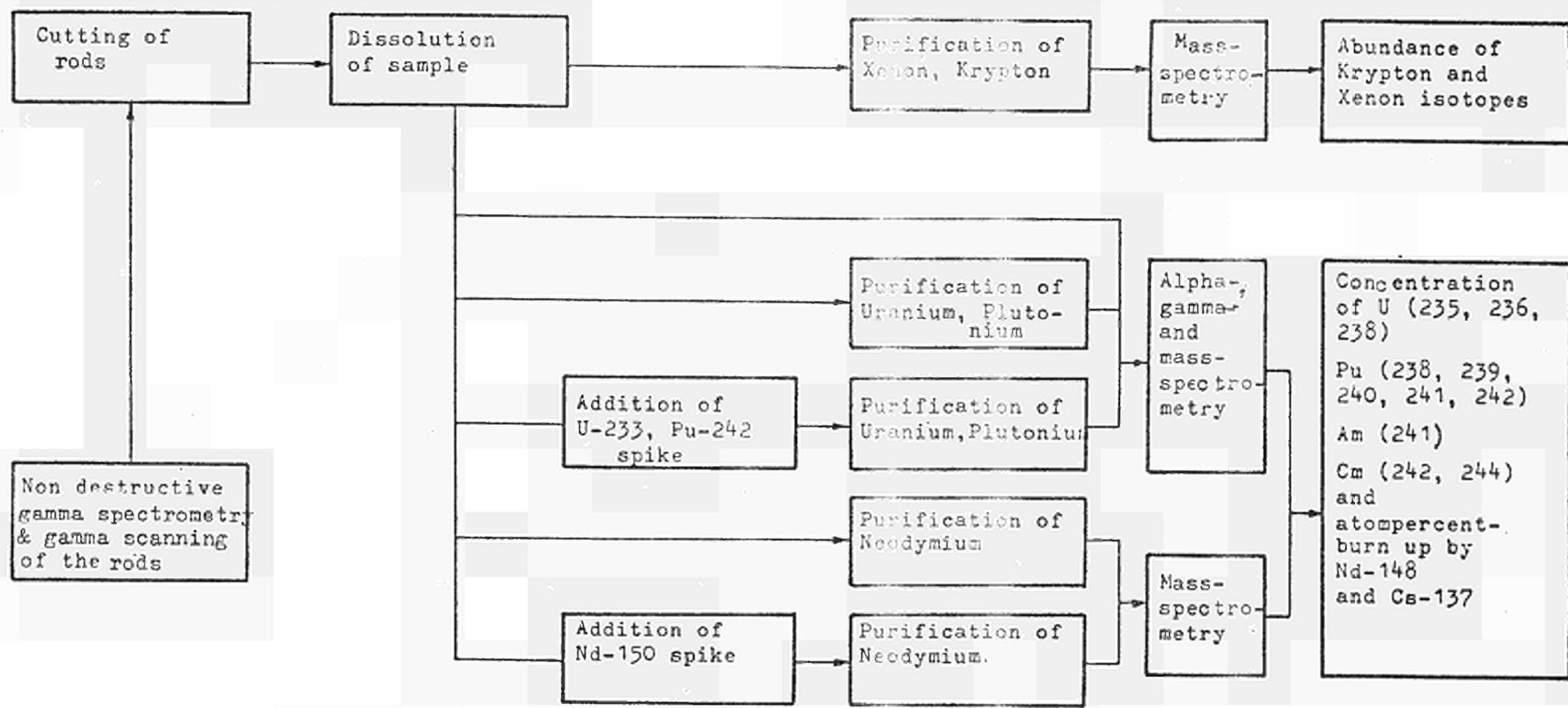


Fig. 2-13 - FLOW DIAGRAM OF POST-IRRADIATION EXAMINATIONS

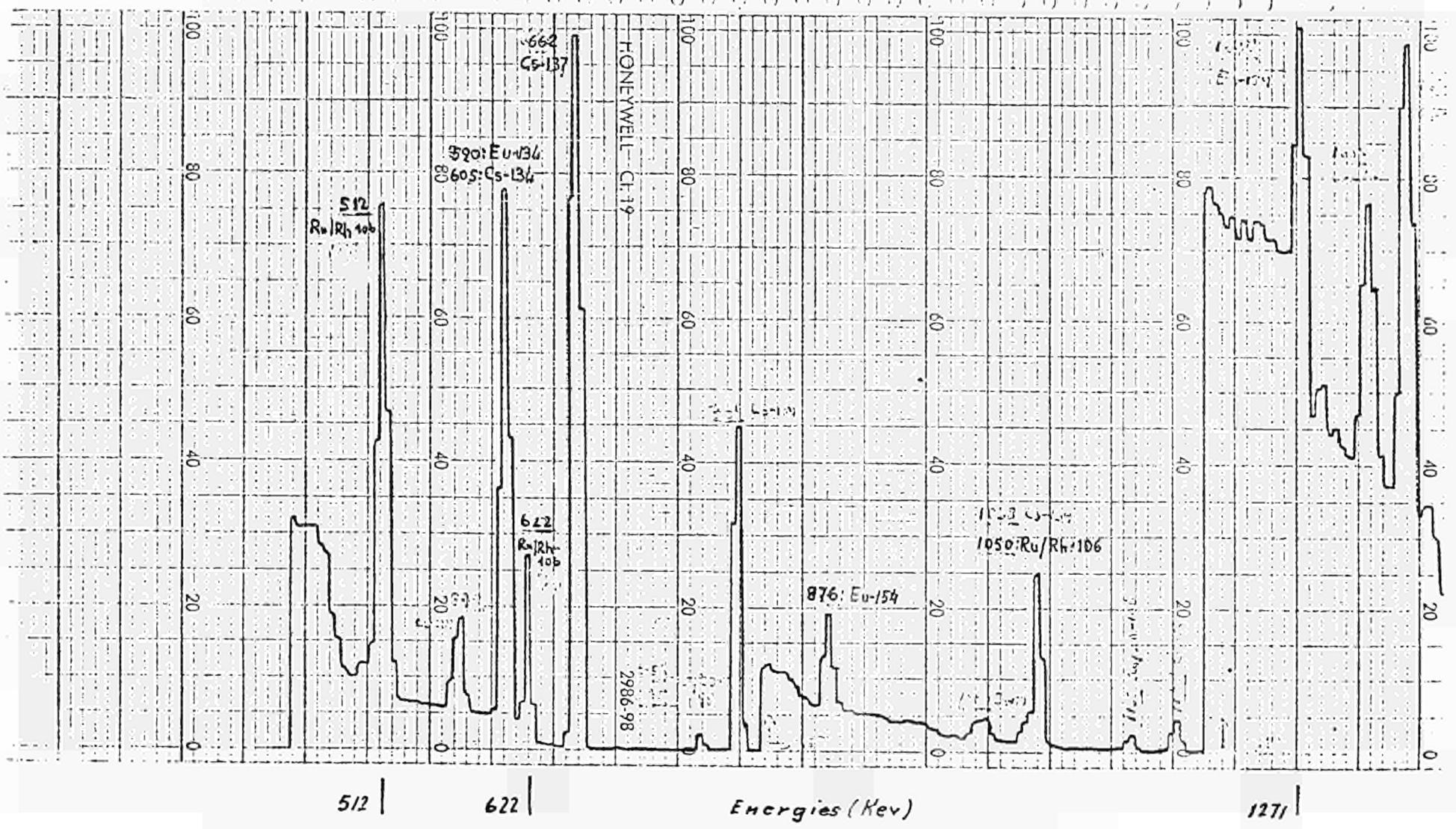


Fig. 2-14a - TYPICAL GAMMA SPECTRUM OBTAINED AT LEVEL "C"

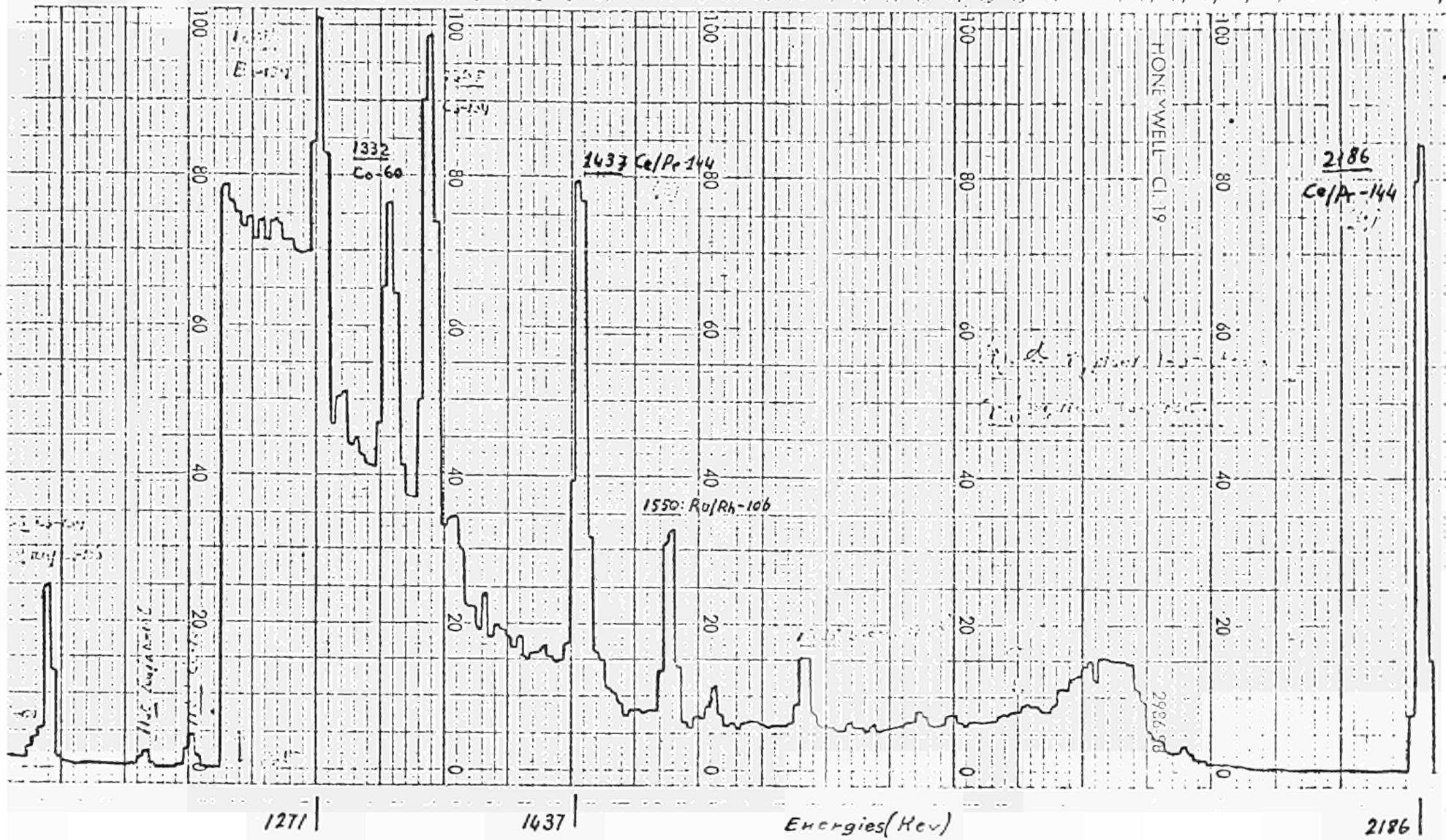


Fig. 2-14b - TYPICAL GAMMA SPECTRUM OBTAINED AT LEVEL "C"

siderably simplified because for these isotopes it was only necessary to establish the relative activity of the rods.

An axial gamma scanning was also carried out along the whole length of two rods. For this purpose a continuous advancing system was used and the individual activity of the selected isotopes was recorded.

Figures 2-15 and 2-16 give the normalized values of the measured activities; the two charts clearly show the diversity between the axial distribution of a corner rod (upwards tendency) and that of a central rod (tendency to shift toward the bottom of the core). This diversity is due to the combined effect of the voids and control rods.

At the beginning of the destructive measurement program it was decided to take advantage of the availability of dissolved fuel slices to check the burn-up level by determining the specific activity of the same fission products selected for the non-destructive analysis and in particular Cs-137. Therefore, a portion of each solution was subjected to gamma spectrometry with an absolute-calibrated system (Ge-Li monitor).

Once the specific activity of Cs-137 (in Ci/g) was known, it was possible to derive the burn-up (in MWd/MTU) by means of a conversion factor that takes into account in an appropriate way the characteristics of this fission product and the reactor history.

The concentration of U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, and Pu-242 were determined by mass spectrometry combined with isotopic dilution techniques. Each mass spectrometry measurement was generally performed three times. The experimental errors, expressed as deviation from the mean value for a single analysis, were on the average close to the method calibration error. Pu-238, together with americium and curium isotopes, was instead analyzed by means of the alpha spectrometry technique.

The isotopic composition results were related to the initial amount of fuel, that is, use was made of the ratio of each heavy isotope N_i to the total of heavy isotopes N_i^0 before irradiation (i. e. all initial uranium atoms);

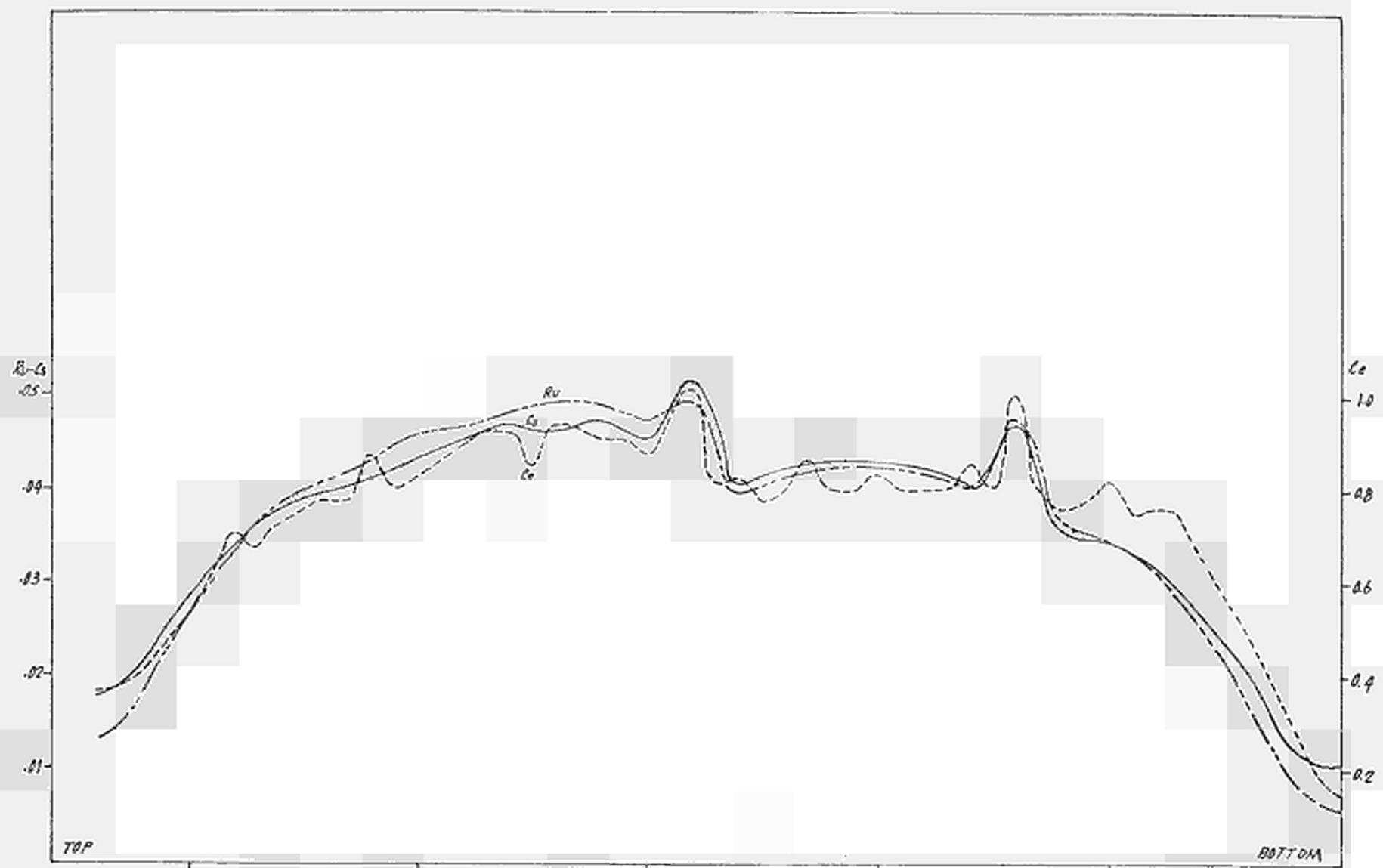


Fig. 2-15 - AXIAL ACTIVITY DISTRIBUTION OF Ru(512) . Cs(862) AND Ce(2186) PEAKS IN A CORNER ROD

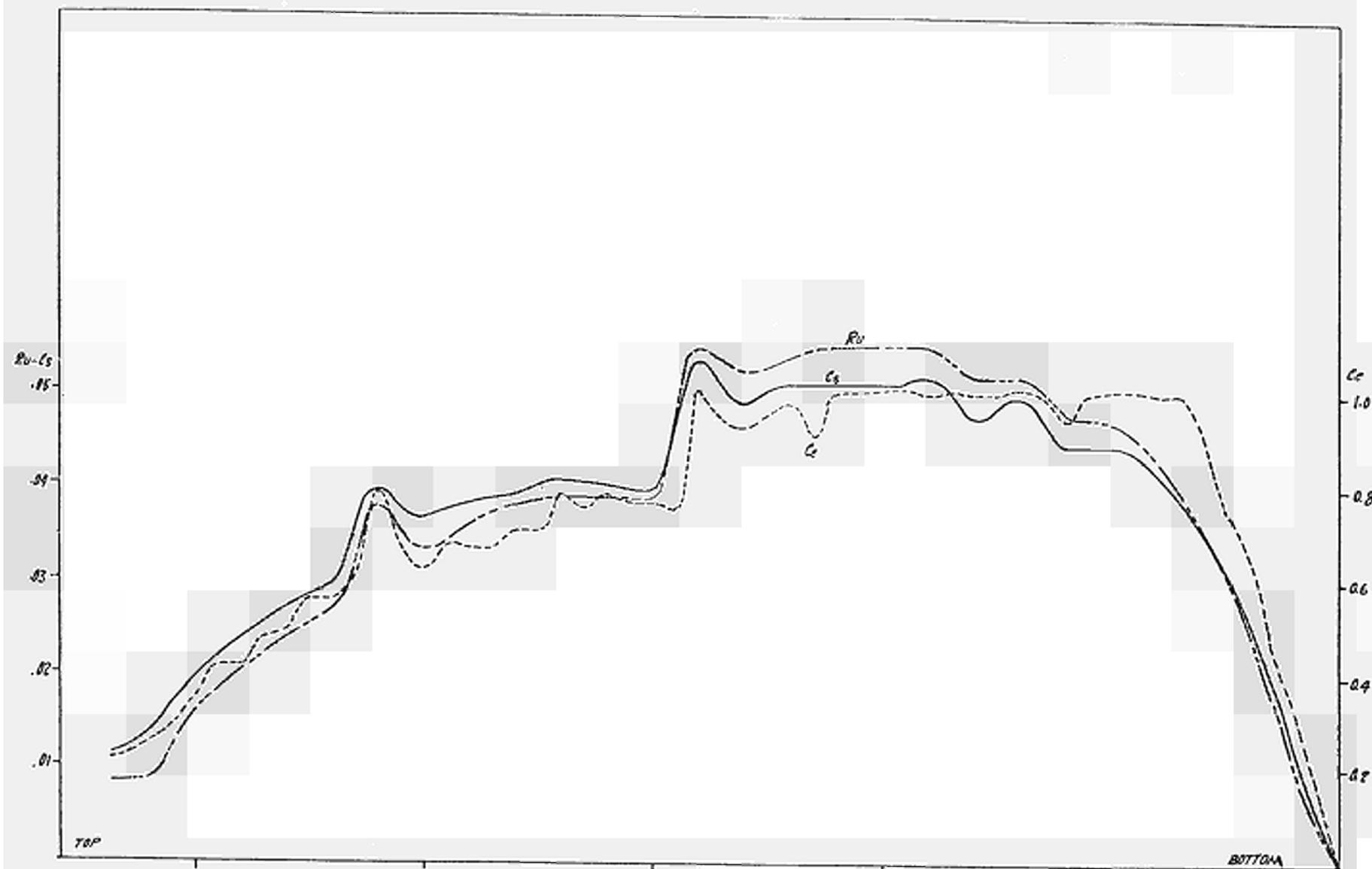


Fig. 2-16 - AXIAL ACTIVITY DISTRIBUTION OF Ru(512) . Cs(662) AND Ce(2186) PEAKS IN A CENTRAL ROD

in this way the results were easier to compare with the calculated values. Nd-148, as burn-up monitor, has been determined by mass spectrometry with isotopic dilution technique using a Nd-150 spike; the separation of neodymium was performed by chromatographic elution on resins. The obtained Nd-148 concentrations were converted into F_T , that is, the percentage of fissioned atoms referred to the initial heavy atoms, and subsequently into burn-up (in MWd/MTU) taking into account the average value of the Nd-148 fission yield, assumed as 1.965%⁽²⁸⁾, and an average value of 196 MeV for the total energy released per fission⁽²⁹⁾.

In the determination of burn-up as F_T , the error in fission yields prevails over all other errors and limits the accuracy of the burn-up analysis to about 1.5%

The measured contents of uranium and plutonium isotopes, together with the F_T values, are collected in Table 2-II.

To check the consistency of experimental results on isotopic contents, a systematic study of the relationships existing between different isotope ratios and certain reactor parameters was conducted⁽³⁰⁾. Several linear correlations were observed and some of these, especially those based on fission gas nuclides⁽³¹⁾, seem to be of general interest. In Fig. 2.17 and 2.18, two of these observed linear correlations are illustrated.

2.5 Operating Experience with Prototype Plutonium Assemblies

The presence of prototype plutonium assemblies in the Garigliano plant required a check of the activity due to these prototypes and to adopt adequate precautions in handling them in order to prevent or to limit the consequences of a spread of plutonium material, should an assembly be dropped accidentally.

This Sub-Section contains the results of the radiometric surveys, a description of the precautions taken, the main information on the opera-

TABLE 2-II

Results of F_T determination and of the contents of uranium and plutonium isotopes

Sample	F_T a/o	Uranium			Plutonium					$F_T \frac{\sum N_1}{\sum N_1^0}$
		$\frac{235}{\sum N_1^0}$	$\frac{236}{\sum N_1^0}$	$\frac{238}{\sum N_1^0}$	$\frac{^{238}}{\sum N_1^0} \times 10^{-3}$	$\frac{239}{\sum N_1^0} \times 10^{-3}$	$\frac{240}{\sum N_1^0} \times 10^{-3}$	$\frac{241}{\sum N_1^0} \times 10^{-3}$	$\frac{242}{\sum N_1^0} \times 10^{-3}$	
A. 1	1.126	0.00777	0.00161	0.9739	0.01860	3.725	1.117	0.4394	0.0861	0.99993
A. 3	1.118	0.01235	0.00187	0.9693	0.01751	3.884	0.919	0.3722	0.0544	0.99995
A. 5	1.128	0.01185	0.00173	0.9697	0.01586	3.977	0.935	0.3847	0.0555	0.99993
A. 9	1.499	0.00555	0.00184	0.9720	0.02669	3.439	1.420	0.5496	0.1649	0.99998
B. 1	1.046	0.00851	0.00142	0.9742	0.01864	3.859	1.010	0.4030	0.0676	0.99995
B. 2	1.094	0.01231	0.00189	0.9696	0.01701	3.857	0.879	0.3506	0.0494	0.99989
B. 8	1.293	0.01050	0.00199	0.9693	0.02004	3.685	1.064	0.4036	0.0735	0.99997
C. 1	1.138	0.01225	0.00188	0.9692	0.01774	3.929	0.919	0.3761	0.0561	1.00001
C. 3	0.972	0.01348	0.00168	0.9697	0.01332	4.148	0.807	0.3360	0.0384	0.99992
D. 2	1.008	0.01297	0.00173	0.9698	0.01427	4.028	0.809	0.3383	0.0411	0.99981
D. 4	0.941	0.01332	0.00172	0.9701	0.01691	4.181	0.764	0.3311	0.0364	0.99988
E. 1	1.153	0.01204	0.00190	0.9690	0.01739	4.058	0.934	0.3909	0.0559	0.99993
E. 5	0.950	0.01335	0.00164	0.9701	0.01780	4.221	0.770	0.3336	0.0359	0.99997
G. 7	1.121	0.01199	0.00183	0.9693	0.01890	4.167	0.941	0.3894	0.0542	0.99990
H. 2	1.273	0.01096	0.00195	0.9690	0.01892	3.820	1.028	0.4130	0.0693	0.99999
H. 8	1.351	0.01036	0.00200	0.9685	0.02047	3.855	1.133	0.4360	0.0827	0.99990
J. 1	1.370	0.00631	0.00180	0.9725	0.02900	3.663	1.324	0.5497	0.1407	1.00002
J. 9	1.542	0.00541	0.00191	0.9714	0.03246	3.518	1.474	0.5840	0.1820	0.99993

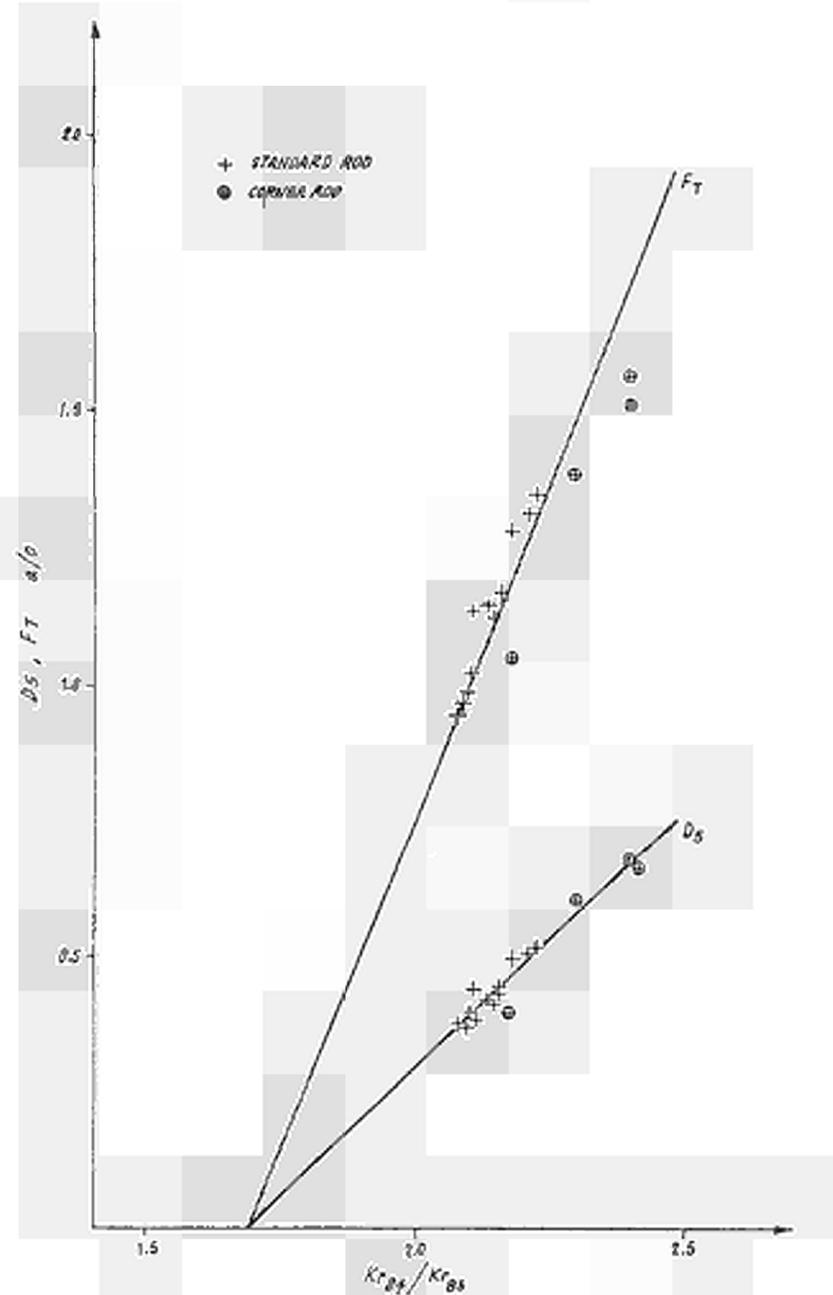


Fig. 2-18 - CORRELATION OF FRACTIONAL U-235 DEPLETION (D_5) AND HEAVY ATOMS BURNED (F_T) VERSUS KRYPTON ISOTOPE RATIO ($Kr-84/Kr-83$)

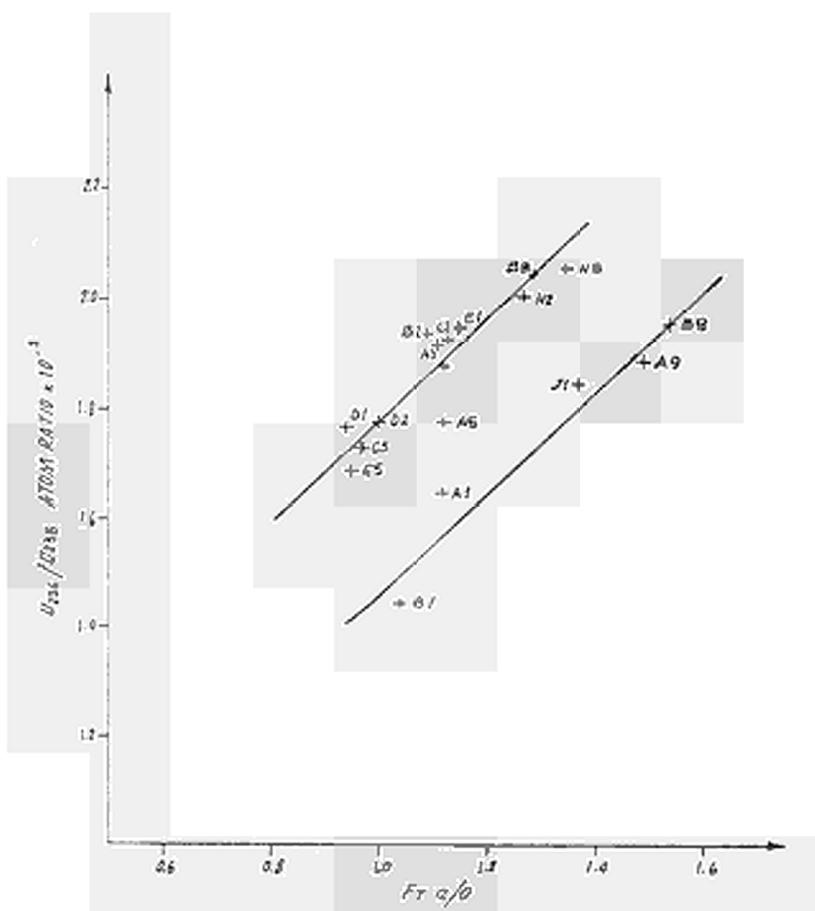


Fig. 2-17 - CORRELATION BETWEEN U-236/U-238 AND BURN-UP EXPRESSED AS F_T

tion of the Garigliano core, and the performance of the twelve plutonium assemblies loaded in the Garigliano core at the beginning of Cycle 2.

2.5.1 Health Physics Aspects Associated with Prototype Handling

The handling of the prototype plutonium assemblies did not give rise to any particular radiological problems. In fact, the average plutonium concentration was less than 2%, of which only about 1% was Pu-241. The radiological hazard of plutonium fuel comes mainly from the neutrons emitted in spontaneous fissions and (α, n) reactions with the oxygen in the pellets, and from gamma rays emitted by Am-241.

The neutron and gamma dose rates were measured on newly fabricated prototype plutonium assemblies in different positions and the results are given in Tables 2-III and 2-IV.

Table 2-III

Neutron Dose Rate, mrem/hr

Distance from assembly (cm)	Mid-height of assembly (150 cm)	50 cm from an assembly end
12.5 (instrument contacting assembly)	3	2
100	0.4	0.4

Table 2-IV

Gamma Dose Rate, mrem/hr

Distance from assembly (cm)	Mid-height of assembly (150 cm)	50 cm from an assembly end
5 (instrument contacting assembly)	6	6
100	< 0.5	< 0.5

The handling procedures called for checks for alpha, beta and gamma surface contamination, and measurements of airborne contamination. Surface contamination was monitored by alpha and gamma counting on swabs taken on the external rows of fuel rods, whilst the airborne activity was monitored by alpha and gamma counting of paper filters used for the continuous and intermittent sampling of the air in the fuel vault. During fuel handling, thanks to the strict fabrication specifications, no alpha contamination due to plutonium was detected.

The greatest radiological hazard in handling the prototype plutonium assemblies would be accidental dropping or violent impact of an assembly such as would cause escape of the oxide fuel. In addition to the strict instructions issued to the personnel handling these assemblies, several precautions were taken during the operations, consisting mainly in keeping the assemblies wrapped in plastic bags until they were loaded into the reactor. At any rate, the station staff was equipped with suitable portable instruments for emergency action.

2.5.2 Information on Core Operation

Since it was first placed on line in May 1964, the 160-MWe Garigliano nuclear power station has generated 6.0×10^9 kWh as of 30 June 1970. The core is composed of 208 fuel assemblies for a total of 50 tonnes of fuel; during Cycle 2, which lasted from October 1968 to June 1970, in the core there were 142 fuel assemblies of the initial load and 66 reload assemblies, twelve of which were plutonium assemblies. At the end of this period of irradiation, the average burnup of the initial fuel load was about 13,000 MWd/MTU (average design burnup 13,300 MWd/MTU) and the irradiation of the lead initial load fuel assembly was 18,900 MWd/MTU. Irradiation of the lead reload fuel assembly was 7,900 MWd/MTU, whilst the average and the peak irradiation of the prototype plutonium assemblies were 7,000 and 7,500 MWd/MTM respectively.

The assemblies of the first load, 9x9 lattice, have a design peak heat flux in overpower of 99 W/cm^2 ($315,000 \text{ BTU/hr-ft}^2$), that is, 12.8 kW/ft, whilst the reload fuel assemblies, 8x8 lattice, have a design peak heat flux in overpower of 133 W/cm^2 ($420,000 \text{ BTU/hr-ft}^2$), that is 18.9 kW/ft.

The maximum values actually experienced in Cycle 2 at rated power were around 7.2 kW/ft for the 9x9 initial load fuel assemblies and 10.4 kW/ft for the 8x8 reload fuel assemblies.

The maximum estimated linear power density for the plutonium prototypes was approximately the same as indicated for the normal reload 8x8 assemblies.

The operating cycle ending June 13, 1970, was one of the longest ever reached with boiling water reactors. The duration of this cycle was equivalent to about 17 months of operation with a 90% plant capacity factor.

The core operating conditions were analyzed periodically by means of follow-up calculations with the ERFLARE code; the value of K_{eff} obtained in all these calculations was systematically less than unity. The deviation, albeit relatively small ($< 0.006 \Delta K$), was practically constant throughout the irradiation cycle. These calculations thus confirm the validity of the method adopted to evaluate core reactivity depletion.

The radial power distribution is shown in Figures 2-19, 2-20 and 2-21, respectively at the beginning, middle, and end of Cycle 2. Also shown in these figures are the control rod patterns for each situation; each rod is represented by a circle at the center of the four assemblies controlled by that rod, and the number in the circle corresponds to the notches by which the control rod is withdrawn. These numbers are comprised between 0, rod all in, and 35, rod all out. (In the figures, the number 35 for the completely withdrawn control rods was omitted, so that a white circle means "rod all out".) Fig. 2-22 shows the burnup distribution at the end of Cycle 2.

GARIGLIANO STATION

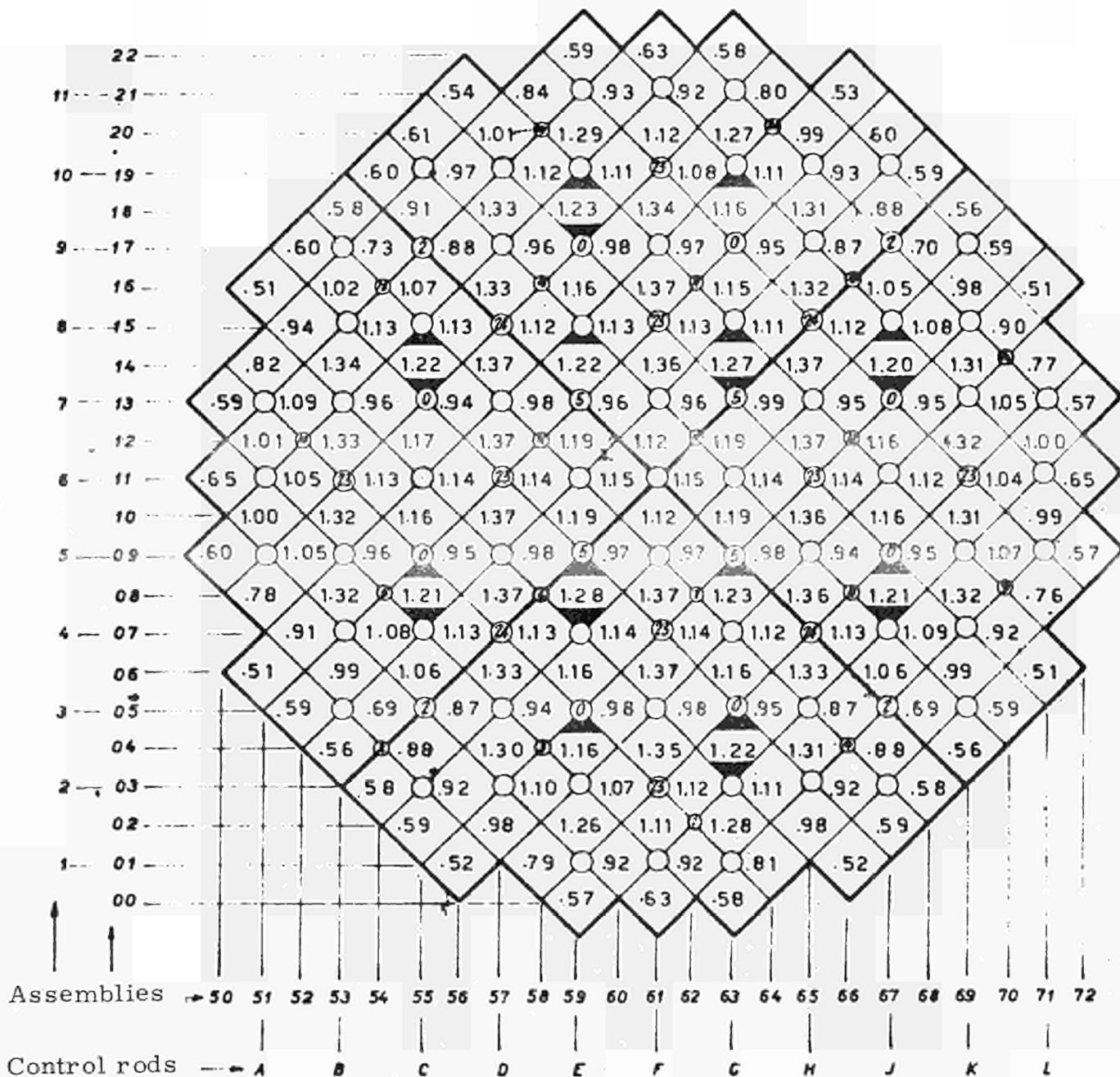


FIG. 2-19 - RADIAL POWER DISTRIBUTION AS OF DECEMBER 10, 1968.
AVERAGE CORE BURN-UP: 7,100 MWD/MTU

GARIGLIANO STATION

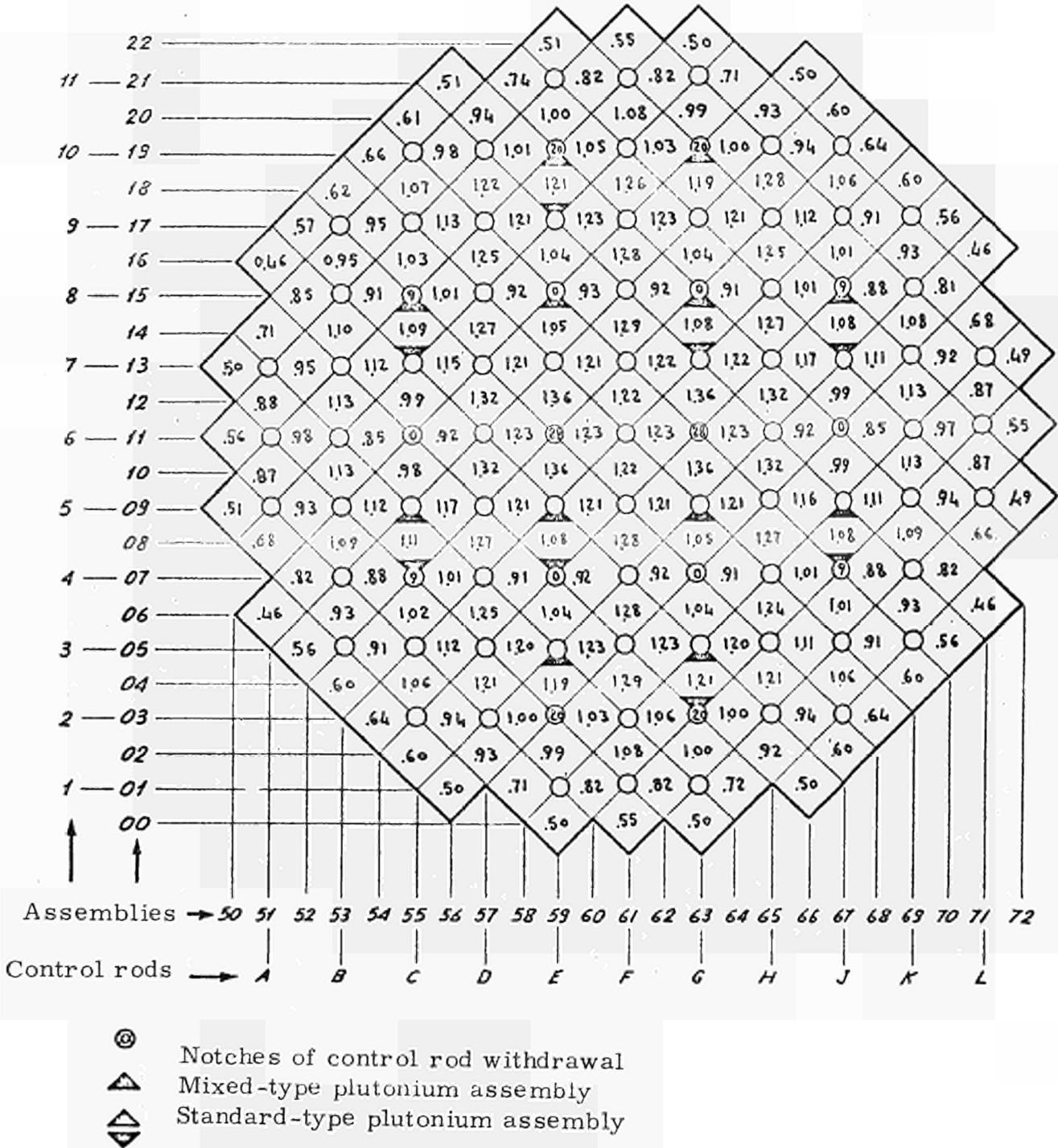


FIG. 2-20 - RADIAL POWER DISTRIBUTION AS OF JUNE 30, 1969

GARIGLIANO STATION

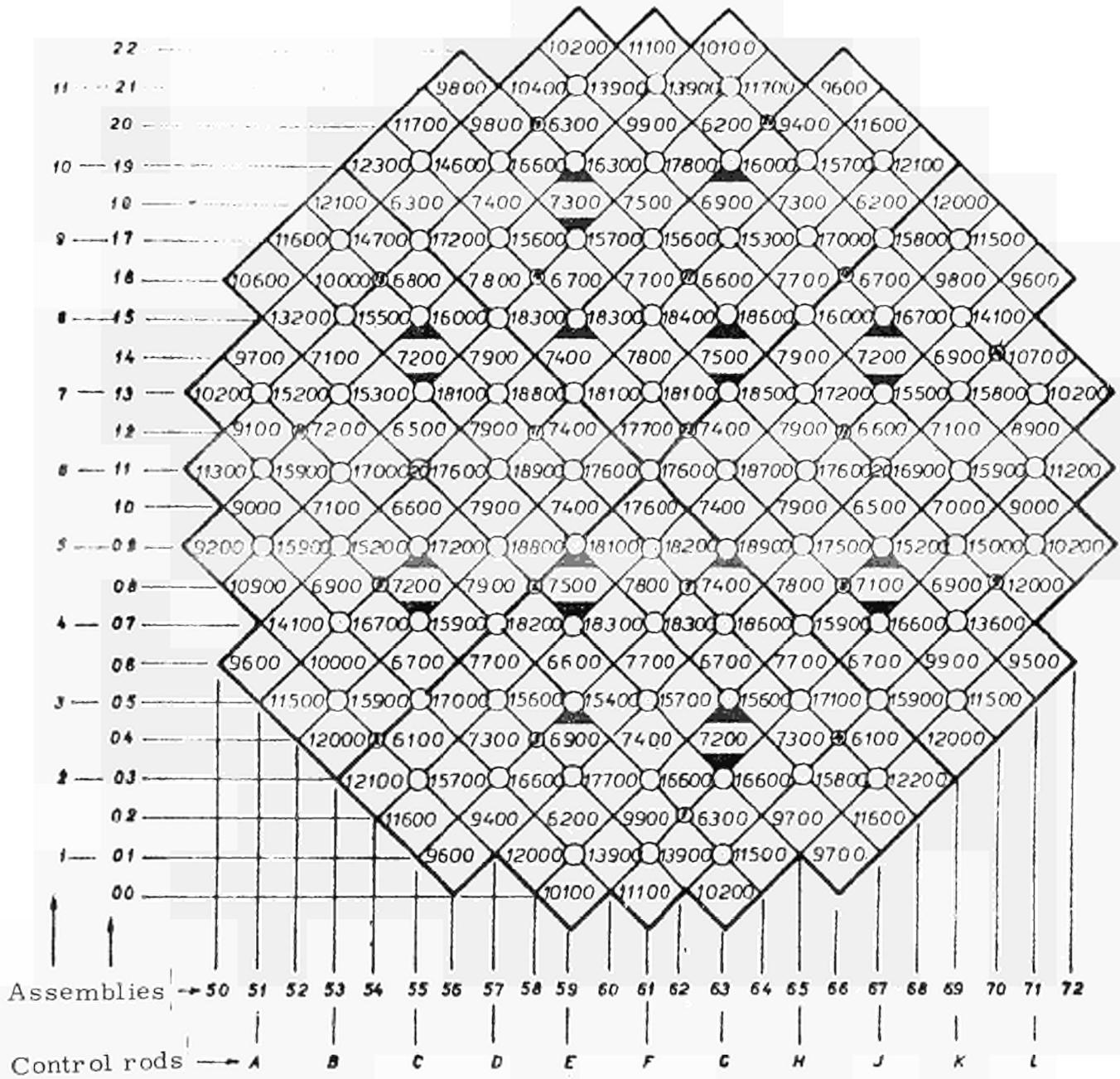


FIG. 2-22 FUEL ASSEMBLY BURN-UP AS OF JUNE 13, 1969.
AVERAGE CORE BURN-UP: 12,000 MWD/MTU



Standard-type plutonium assembly



Mixed-type plutonium assembly

GARIGLIANO STATION

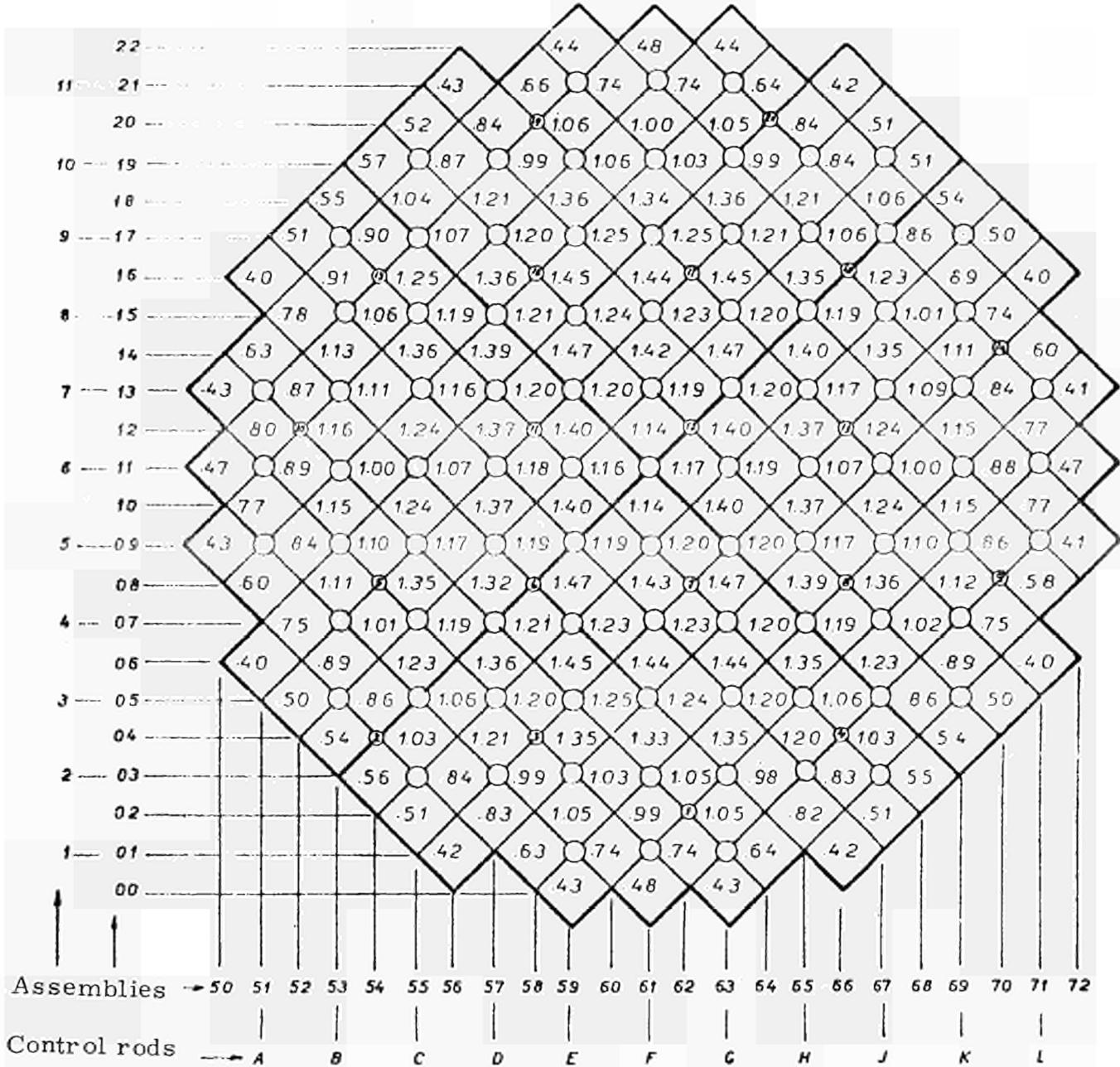


FIG. 2-21 - RADIAL POWER DISTRIBUTION AS OF JUNE 30, 1970
 AVERAGE CORE BURN-UP:12,000 MWD/MTU

From the operational standpoint it is useful to point out that during Cycle 2 generator tripping at full power generally scrambled the reactor, by contrast with the experience in Cycle 1 when this very rarely happened. Undoubtedly, the increased void coefficient associated with a higher plutonium content in the core contributed to this situation; the higher plutonium content is mainly due to higher fuel burnup and, to a smaller extent, to the presence of the twelve prototype plutonium assemblies. It should also be noted that the void coefficient decreases between the beginning and the end of the cycle as a result of the gradual withdrawal of the control rods, and consequently the phenomenon was felt more at the beginning of the cycle. At any rate, studies are under way to take provisions to eliminate the inconvenience.

Besides, during Cycle 2, the off-gas activity increased slowly and constantly. The rate of increase averaged about 4,000 microcuries/sec per month, and before shutdown the off-gas activity reached a ceiling of 35,000 + 40,000 microcuries/sec. From the examination of the type of activity of these gases, a slow change was noted from a "slow diffusion" mixture (denoting leakage of fission products from small cracks) to a "recoil" mixture (denoting the presence of exposed uranium). After plant shutdown, a sipping analysis of all the fuel assemblies was performed in the reactor pressure vessel. This analysis showed a certain number of suspect fuel assemblies in the initial load and reload. The sipping analysis was repeated in the pool on all the suspect assemblies and on a certain number of assemblies considered unfailed; this analysis confirmed the previous results.

The sipping signal for all the prototype plutonium assemblies was well within the average value for the whole core; some of the plutonium assemblies showed the lowest activities measured, thus confirming the high degree of integrity of these assemblies after one cycle of operation. This integrity was also confirmed by the TV examination performed on some of the plutonium assemblies and by the borescope examination of a few pins of an assembly that was disassembled for the gamma scans (see Sub-Section 2.4).

2.6 Verification and Trimming of the Calculation Methods

The calculation methods described in Sub-Section 2.2 were subjected to extensive investigation aimed at ascertaining their adequacy in dealing with the various aspects of UO_2 - PuO_2 cores. The data derived from the operation of the Garigliano and Trino Vercellese stations permitted a first verification of the methods on uranium systems.

The calculation techniques for plutonium systems were verified and trimmed by comparison with the experimental data gathered from experiments conducted abroad on critical facilities or from ad hoc experiments, such as the program implemented by the UKAEA for ENEL on the DIMPLE reactor or those carried out under the subject Program (cfr. Sub-Section 2.4).

The methods used for the calculation of reactivity and power distribution in these comparisons were applied with different energetic subdivisions of the neutron spectrum. The codes were the two-group BURNY (1 thermal and 1 fast); the five-group BURSQUID (2 thermal and 3 fast) and the four-group BURSQUID (1 thermal and 3 fast). In practice these are all versions of the same code, differing only in the number of energy groups.

The survey was also extended to the three-group GAM-THERMOS-SQUID system of codes (1 thermal and 2 fast) which is a conventional technique for a more detailed calculation of the lattice constants, but is much more time-consuming.

The use of the three fast groups permits a more detailed evaluation of the neutron diffusion that occurs in the fast region of the spectrum, the attribution of the U-238 absorption entirely to the resonance group (0.625 eV to 5.5 eV), and thus a more accurate assessment of Pu-239 generation.

With the adoption of two thermal groups (with energy cut-off at 0.2 eV) the variations in neutron reaction rates due to discontinuities (water gaps or heterogeneous enrichments) capable of modifying the thermal spectrum locally should be evaluated more accurately.

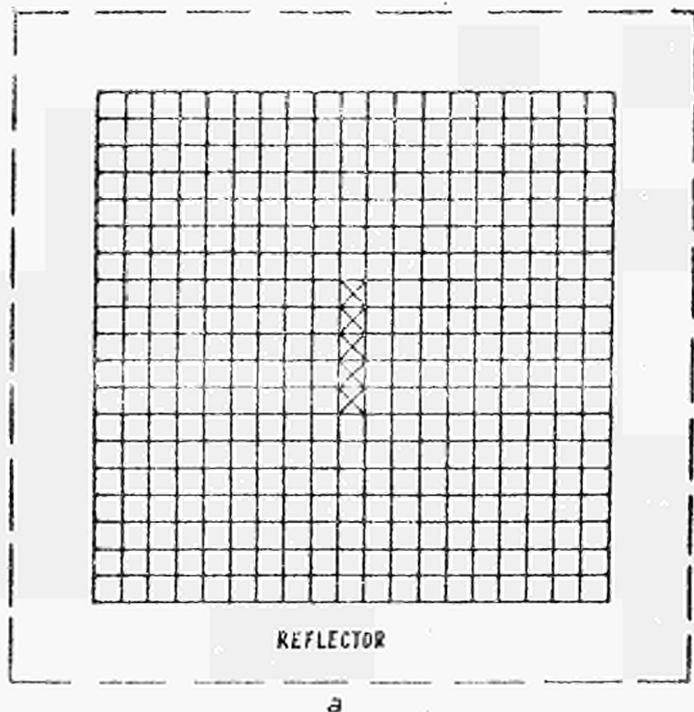
The following is a summary of the main results of the trimming of the calculation method in respect of the experiments considered.

2.6.1 Experiments on Critical Facilities

The information derived from a number of critical experiments were used mainly to check the method for the calculation of the local power distribution. The experiments used for this check were those of the PRCF and of the Saxton Program, as they were the most suited for this verification, and the configurations set up in the DIMPLE reactor for a lattice geometry nearer to that of the ENEL-designed plutonium assemblies.

PRCF Power Distribution Measurements - Of the PRCF measurements⁽³²⁾, two were of particular interest for the theoretical-experimental comparison, because they offered the possibility of studying the effect of different plutonium isotope compositions (from 7 to 23%) under the same geometrical and physical conditions. The results of these measurements were compared with those obtained with the two-group BURNY code⁽⁵⁾ and were generally found in good agreement, especially for the rods near the reflector; a discrepancy on the order of 4% was observed at the center of the critical assemblies and at the interface between the two regions of different isotope compositions.

Saxton Power Distribution Measurements⁽³³⁾ - Six configurations were selected from the experiments performed on a lattice of 19x19 rods (Fig. 2-23). These configurations were interesting because they provided information on the power trend in the plutonium rods near a water gap or near UO₂ rods. The experimental data were compared with the results obtained from the two-group BURNY and the five-group BURSQUID codes and found in fair agreement in both cases⁽⁵⁾. However, power sharing between UO₂-PuO₂ rods and UO₂ rods is reproduced better by the two-group code, because the five-group code gives a deviation of 4-5% for the rods at the boundary between the two core regions by underrating the power of the UO₂ rods in respect to those of

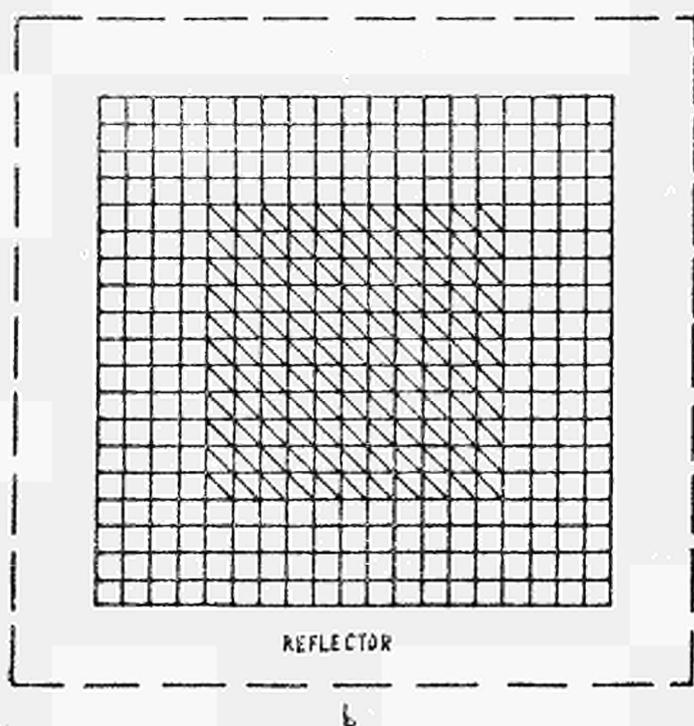


CONFIGURATION 1: Square core containing 19x19 rods of UO_2 enriched to 5.742 w/o in U-235.

CONFIGURATION 2: Ditto, without the five rods shown in Fig. (a).

CONFIGURATION 3: Square core containing 19x19 rods of $\text{PuO}_2\text{-UO}_2$ enriched to 6.6 w/o in PuO_2 .

CONFIGURATION 4: Ditto, without the five rods shown in Fig. (a).



CONFIGURATION 5: Two-region core (b). The internal region contains $\text{PuO}_2\text{-UO}_2$ rods enriched to 6.6 w/o in PuO_2 ; the external region contains UO_2 rods enriched to 5.742 w/o in U-235.

CONFIGURATION 6: Two-region core (b). The internal region contains UO_2 rods enriched to 5.742 w/o in U-235; the external region contains $\text{PuO}_2\text{-UO}_2$ rods enriched to 6.6 w/o in PuO_2 .

Fig. 2-23 - SAXTON FACILITY. Core pattern for different rod configurations.

$\text{UO}_2\text{-PuO}_2$. On the other hand, a systematic disagreement between the values of K_{eff} obtained from the two-group calculation methods and those from the five-group method is apparent (abt 1%). The five-group calculation is in better agreement (within 1%) with the experimental values, especially when high plutonium enrichment values are taken into account. The small dimensions of the critical assembly (19x19 rods) and the high degree of fuel enrichment used made this a very stringent test of the calculation techniques and therefore the agreement between theory and experiments were encouraging.

Power-Sharing Measurement in the DIMPLE Reactor at Winfrith - The UKAEA performed, on behalf of ENEL, a series of measurements on $\text{UO}_2\text{-PuO}_2$ assemblies in the DIMPLE reactor at Winfrith⁽⁵⁾. The assemblies consisted of 5x5 modular units, each containing 8x8 fuel rods; these units were arranged so as to give the closest reproduction possible of the lattice geometry of a BWR core, and particularly the Garigliano core.

The purpose of these measurements was to analyze the power sharing effects between UO_2 and $\text{UO}_2\text{-PuO}_2$ assemblies, and the local power distribution in the $\text{UO}_2\text{-PuO}_2$ module. The measurements consisted in determining the Pu-239 and U-235 fission rates by inserting foils of these isotopes into the rods that were most representative for these measurements. In this manner it was possible to deduce the power distribution and, at the same time, the distribution of the ratio between Pu-239 and U-235 fission rates along a horizontal section of the core. Once the ratio between Pu-239 and U-235 fission rates and how it varies with lattice heterogeneities is known, it is possible to ascertain the validity of the calculation method to foresee power distributions associated with irradiated cores.

Measurements were performed on four different core configurations (Fig. 2-24), that is:

(a) Core uniformly loaded with Al-clad UO_2 rods with 3% U-235 enrichment.

This configuration formed the basic core into which modular Pu units were subsequently loaded.

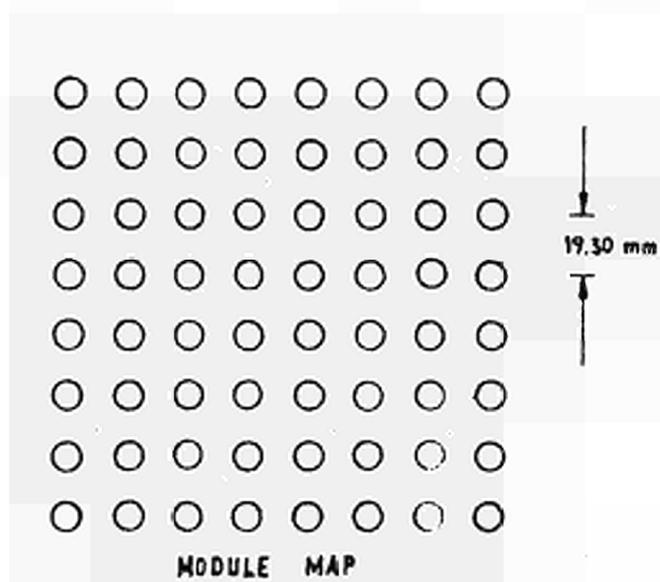
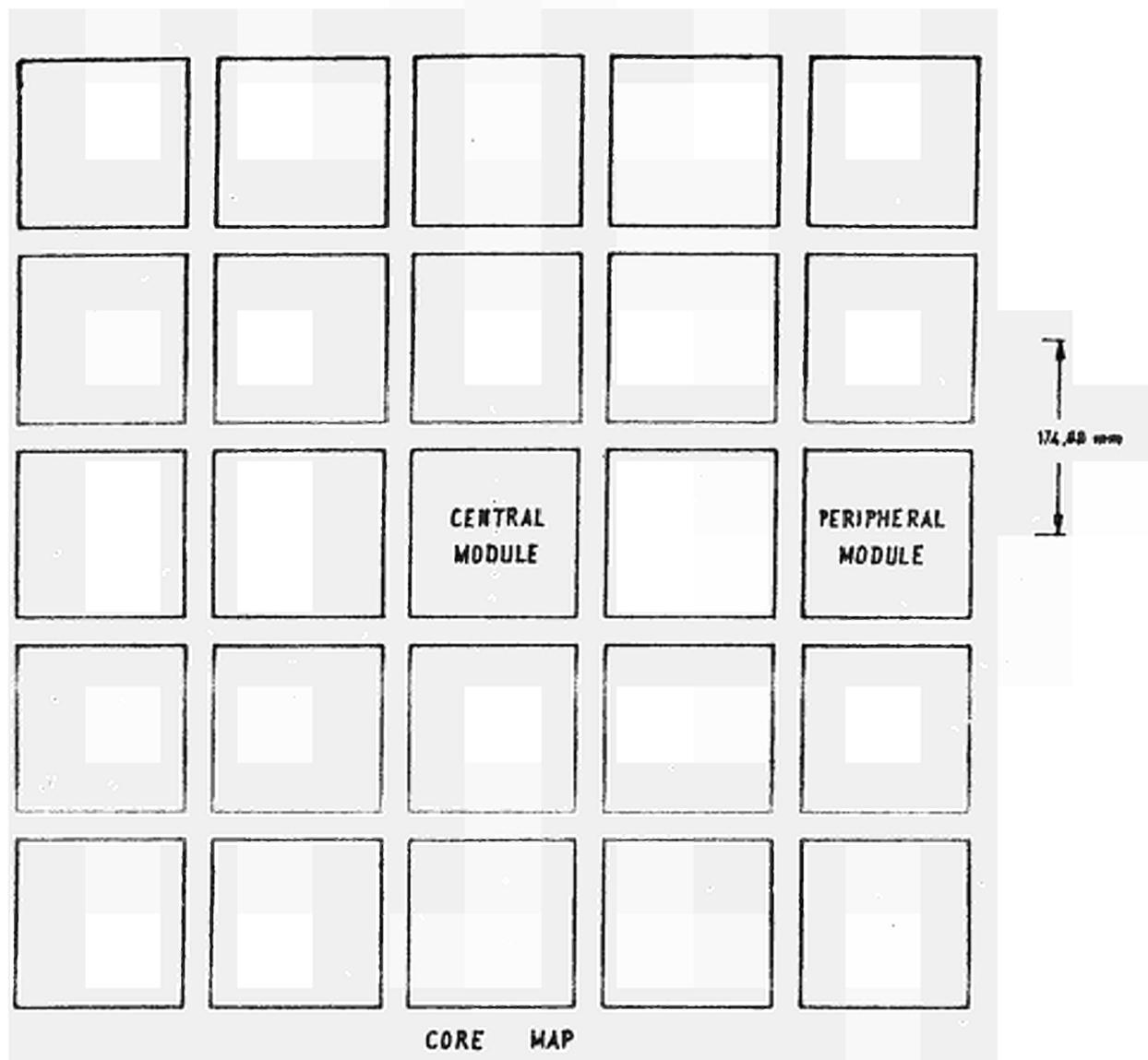


Fig. 2-24 DIMPLE ASSEMBLY. Core pattern for different rod configurations.

- (b) Core as in configuration (a), but with the central modular unit replaced by Al-clad UO_2 - PuO_2 rods consisting of natural uranium and 2.1% Pu.
- (c) Core as in configuration (b), but with the corner rods of the modular Pu unit containing 1.2% Pu in depleted uranium.
- (d) Core as in configuration (b), but with the modular Pu unit located at the core periphery so as to define the effect of the reflector on a Pu assembly.

The reactor was maintained critical by varying the moderator height, which was nearly the same for the four configurations.

The theoretical power distributions for the different configurations were calculated with the two-group BURNY code, the five-group BURSQUID code and the GAM-THERMOS-SQUID system to ascertain whether a larger number of groups for the representation of the neutron spectrum could substantially improve the estimates of power distribution and of the ratio between Pu-239 and U-235 fission rates.

Table 2-V lists the percentage standard deviations between theoretical and experimental values for power distribution, fission rate ratio and criticality checks. For completeness Table 2-V also contains the results achieved in the theoretical-experimental comparisons for the Saxton and the PRCF experiments.

On the basis of these results, the following main remarks may be made.

From the standpoint of the local power distribution, the two-group BURNY code appears to be in better agreement with the experimental data than the five-group BURSQUID code. The standard deviations from the experimental values were always lower than a few percent in all comparisons and with both codes. Thus, the two methods are satisfactory.

As to local effects, the five-group BURSQUID appears to overestimate the corner rod power of a fuel assembly by a few percent, whereas the two-group BURNY code gives the opposite effect.

TABLE 2-V

		Power distribution, σ_{\pm} (%)			Fission rate ratio, σ_{\pm} (%)			$(K_{\text{eff}})_{\text{theor.}}$ - $(K_{\text{eff}})_{\text{exp.}}$		
		Burny 2	Burny 5	G-TH-S	Burny 2	Burny 5	G-TH-S	Burny 2	Burny 5	G-TH-S
A	Array 1	1.4	2.5	-	-	-	-	+0.0117	-0.0012	-
	Array 2	1.4	2.8	4.2	7.6	2.3	2.3	+0.0075	-0.0015	-0.0101
	Array 3	1.1	1.7	-	7.7	1.9	-	+0.0065	-0.0026	-
	Array 4	3.2	2.2	-	9.3	2.8	-	+0.0164	+0.0073	-
B	6 Arrays	0.8-1.4	1.8-2.0	-	-	-	-	+0.0094 to +0.0180	-0.0053 to +0.0073	-
C	2 Arrays	1.6;3.4	-	-	-	-	-	+0.0089; +0.0170	-	-

A = DIMPLE REACTOR

B = SAXTON PROGRAM

C = P R C F

The ratios between Pu-239 and U-235 fission rates are evaluated better by the five-group BURSQUID, whereas the two-group BURNY reflects the limitation of having only one thermal group.

Fig. 2-25 shows the values of the percentage errors for each rod, for both the calculation methods. In this figure, it can be noted that the two-group BURNY gives a discrepancy in the fission rate ratio which increases from the center to the corner rod of the plutonium module where the error is in the range of 13 to 20%.

The clear improvement obtained with the five-group BURSQUID calculation method is due to the different representation of the thermal-neutron behavior. Indeed, the use of two thermal groups allows the local fission rate variations due to thermal-spectrum deformation to be determined for the two isotopes more accurately than one thermal group.

Such an accurate evaluation is impossible with the two-group BURNY method which uses only one-thermal-group constants computed on the assumption of an asymptotic cell spectrum, and thus it does not allow the thermal-spectrum variation due to the water-gap softening to be taken into account.

In this connection, it may be stated that the five-group BURSQUID code appears more promising for burnup calculations.

The results obtained with GAM-THERMOS-SQUID system are also in good agreement with the experimental data on local power distribution, whereas they are no better than the five-group BURSQUID code predictions when it comes to fission-rate ratios.

The K_{eff} values calculated for all the configurations are consistent with experimental data. Here again, a systematic difference is noted between the two-group and five-group calculation values. The K_{eff} values with the four-group model (1 thermal and 3 fast, with the same library as that of the two-group model) fall between the values obtained from the other calculation methods.

This fact would indicate that the discrepancy is due partly to the different representation of the fast-neutron events and partly to the different model used for the thermal neutrons.

2.6.2 Experiments in the Garigliano Reactor

The two experiments carried out in the Garigliano reactor with plutonium assemblies provided an interesting set of data for the trimming and verification of the theoretical methods, particularly for criticality and power distribution calculations. In connection with these experiments, the K_{eff} and power distribution were calculated with the BURSQUID code and the FORM-THERMOS-SQUID system; the macroscopic radial power distribution at full power was derived by means of the ERFLARE routine calculations.

The consistency of the criticality data from the first experiment (1968) and calculated values is quite satisfactory, the error being less than 0.5%. In Table 2-VI the K_{eff} values calculated for all the fuel assembly configurations, in the assumption of fully withdrawn control rods, are compared with the values derived from experimental results.

Table 2-VI

Configuration	Experimental K_{eff}	K_{eff} from BURSQUID	K_{eff} from FORM-THERMOS-SQUID
I	1.01090 ± 0.00050	1.00800	1.01560
II	1.01328 ± 0.00050	1.01070	-
IV	1.01486 ± 0.00050	1.01375	1.02140
V	1.01280 ± 0.00050	1.01130	-
3x3	1.00560 ± 0.00050	1.00600	1.00900

A more stringent check is provided by the comparison of the ΔK values arising from replacement of an enriched-uranium assembly with a plu-

onium assembly. This comparison is given in Table 2-VII.

Table 2-VII

	Experimental ΔK	ΔK from BURSQUID	ΔK from FORM- THERMOS- SQUID	ΔK from revised BURSQUID
ΔK_{II-I} , pcm	238 ± 9	270	-	210
ΔK_{IV-I} , pcm	388 ± 7	375	580	445
ΔK_{V-I} , pcm	191 ± 7	330	-	267

This comparison evidences a tendency of the calculation methods to overestimate the ΔK resulting from the replacement of one enriched-uranium assembly with a plutonium assembly. The overestimate is less appreciable when the replacement is made in a corner position of the configuration.

The fact that this tendency is common to both calculation methods, notwithstanding the different number of neutron groups, might indicate an overrating of the Pu-239 multiplication properties provided by the code libraries.

The calculations were repeated with a reduced epithermal fission integral of Pu-239 of the RIBOT-5 library from 338 to 293 barns according to a recent code revision⁽⁶⁾. The related results shown in the last column of Table 2-VII are in better agreement with the experimental values.

Figure 2-26 shows the percentage deviation between the calculated and experimental values of the power distributions for the level without control rods.

The power density was calculated with a standard deviation less than 2% more significant deviations being observed on the rods at the border of fuel areas within each assembly having different characteristics. The

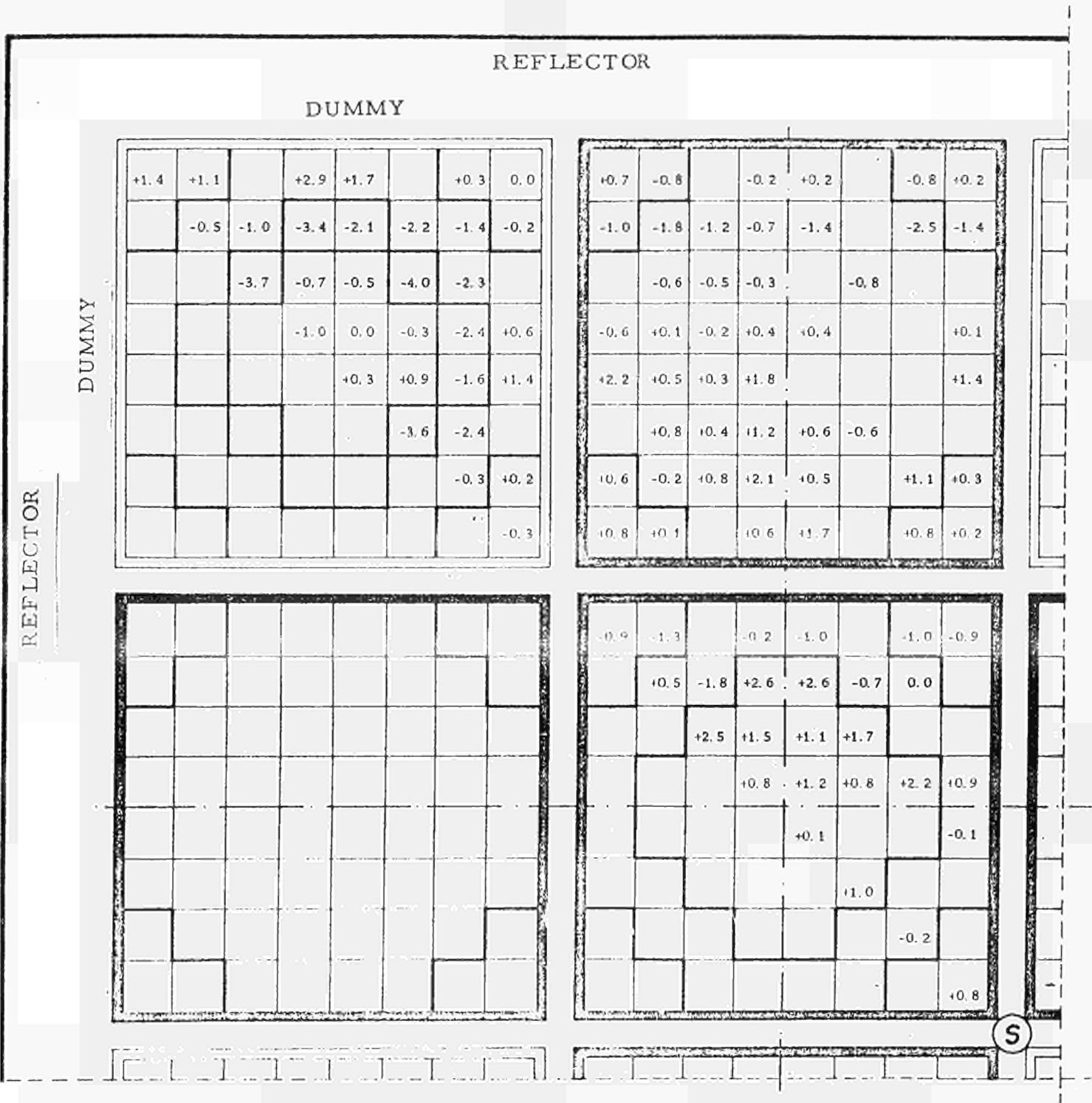


Fig. 2-26 - LEVEL WITHOUT CONTROL RODS: PERCENTAGE DEVIATIONS BETWEEN EXPERIMENTAL AND THEORETICAL ROD POWER DENSITIES, AS CALCULATED BY THE BURSQUID CODE.
 $\angle (T - E)/E \times 100.$ $\sigma = \pm 1.5\%$

values obtained for the corner rods with the FORM-THERMOS-SQUID three-group technique show a systematic deviation in respect of the experimental data. This deviation cannot be found in the results obtained with the first calculation method because of the adoption of the two thermal groups.

At the level characterized by the presence of control rods, the deviations between the theoretical and experimental values are greater, as evidenced in Fig. 2-27.

From this figure it can be noted that the deviations are distributed in a systematic pattern, the largest occurring in the fuel rods adjacent to the control rods, with opposite signs on the two sides. This trend may be due to off-center positioning of the control rods, which is actually possible also because of the 5-mm clearance between the control rod and the fuel channel.

To confirm this assumption a theoretical evaluation of this effect was performed with the transport code DTK for a control rod in the off-center position, that is, with one blade leaning completely on the side of one of the fuel assemblies.

On the fuel cells closest to the control rods, the effect determined a variation of about $\pm 14\%$, which is consistent with the value deduced from the deviation between theoretical and experimental values.

On the basis of the above considerations it seems that, at the time of the experiments, the control rods may have all been positioned more or less eccentrically.

The consistency of the K_{eff} values of the second experiment also appears to be very satisfactory. The macroscopic radial power distribution was obtained with the ERF LARE calculations that are regularly carried out to update the burnup reached by each fuel assembly for fuel accounting purposes. This consistency confirms the validity of the theoretical evaluation of the K_{∞} versus-burnup curve, and in particular the predictions of the fuel cycle length.

The comparison between the predicted radial power distribution at the end of Cycle 2 and the corresponding experimental values is shown in Fig. 2-11.

The mean square deviation is $\pm 2.5\%$, value that is on the same order of magnitude resulting from the evaluation of the other gamma scanning experiments carried out on the Garigliano core. Table 2-VIII summarizes the data relating to the different gamma scanning experiments.

Table 2-VIII
Comparison between theoretical and gamma-scanning data

Core irradiation (MWd/MTU)	Fuel cycle	Calculated K_{eff}	Power distribution σ (%)
70	1A	0.9956	± 2.41
4920	1A	0.9921	± 1.93
3315	1B	0.9970	± 5.02
12020	2	0.9945	± 2.50

In the gamma scanning at the end of Cycle 2 no particular bias was noticed in the evaluation of the power of the plutonium assemblies.

2.6.3 Post-Irradiation Measurements on an Enriched-Uranium Assembly

The calculation technique developed to assess power distribution was checked also at high irradiation levels by comparing the burnup and isotopic contents of individual rods, as derived from the post-irradiation measurements carried out at the Common Research Center, Karlsruhe, on an irradiated enriched-uranium assembly (A-106).

The calculations of the burnup and fuel isotopic composition were performed initially with the two-group BURNY code. In these calculations it was assumed that the neutron stream was nil around the assembly only during the exposure times when the assembly was not affected by the presence of control rods. On the contrary, when a control rod was inserted, use was made of

extrapolation lengths calculated by means of the transport code DTK.

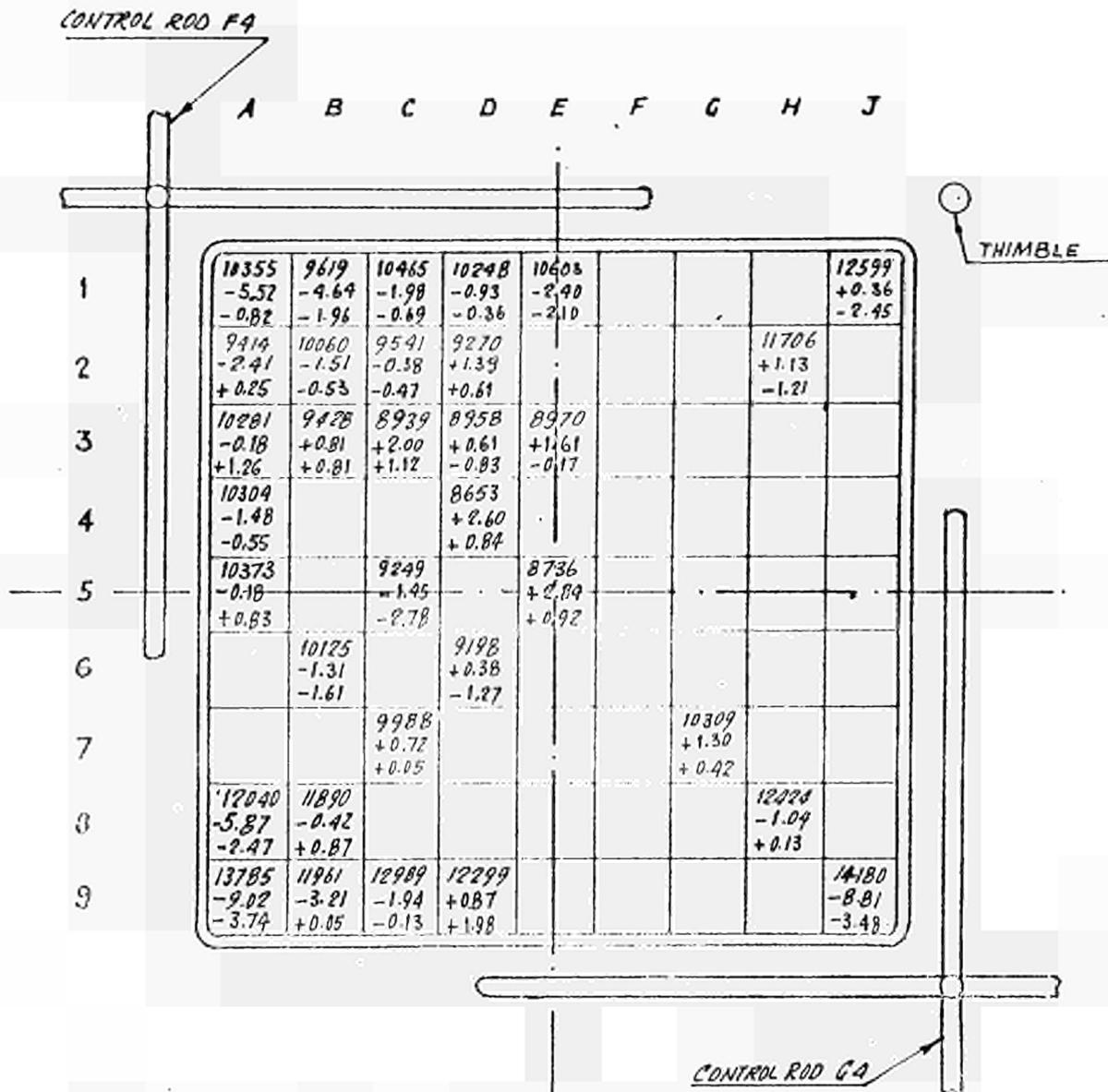
The simplified representation of the control rod blade intersection led to an overrating of the control effect. The same fact was observed for the first time in the experimental results of the measurements of the control-rod-affected power distribution carried out at the Garigliano in the 1968 summer shutdown (see Paragraph 2.6.2), and it has been confirmed by comparison with the burnup distribution measured at a fixed plane (Level C) in the post-irradiation examination of the A-106 assembly. The calculations were therefore repeated with the five-group BURSQUID code using a more detailed representation of the control rod blade intersection.

In Fig. 2-28 are indicated the experimental values of burnup (E) together with the percent deviations in comparison with the theoretical values (T) calculated with the two-group BURNY and five-group BURSQUID codes. The results of the two calculation techniques are well consistent with the experimental data, larger deviations being observed for the two-group BURNY calculation ($\sigma = \pm 3.1\%$ versus $\pm 1.5\%$).

By analyzing the deviations for the two-group BURNY calculation, it will be noted that the greater deviations occur in proximity of the areas where the enrichment differs and that the control rod effect on the corner rod is overrated.

As for the comparison with the five-group theoretical results, no systematic concentration of error is observed. It was therefore possible to conclude that at least up to 10,000 MWd/MTU the burnup of any single rod, including the corner and peripheral rods, i.e. rods in "difficult" positions, is well represented by both the two-group BURNY and the five-group BURSQUID calculations.

With regard to the isotopic abundance of heavy atoms, the experimental results, expressed as percent of initial uranium atoms still present after irradiation, were compared with the theoretical values calculated with the two-group BURNY and five-group BURSQUID codes. Table 2-IX compares the average deviations for the two calculations.



xxxx Experimental Values (E) MWd/MTU
xxxx $\frac{(T - E)}{T} \times 100$ - 2 Group BURNY $\sigma = \pm 3.088\%$
xxxx $\frac{(T - E)}{T} \times 100$ - 5 Group BURSQUID $\sigma = \pm 1.494\%$

Fig. 2-28 - THEORETICAL EXPERIMENTAL COMPARISON OF LEVEL C BURN-UP DATA

TABLE 2-IX

Comparison between the average deviations obtained with two-group BURNY and five-group BURSQUID calculations

Isotope	$\sigma\%$ (T - E/T)	
	Five groups	Two groups
U-235	-3.94	-5.66
U-236	-6.85	-7.38
U-238	+0.072	+0.19
Pu-239'	+3.69	+24.66
Pu-240	+3.26	+7.69
Pu-241	+10.12	+8.95
Pu-242	+25.50	+8.53

The two-group BURNY calculation appears unsatisfactory, particularly in respect of the Pu-239 concentration, which is systematically underestimated by 25% on the average. From the deviation distribution, it may be observed that the larger deviations occur in the corner and peripheral rods; this difference may be imputed to an inadequate representation of the thermal spectrum and to the library data used in the two-group BURNY. This assumption seems to be supported by the results obtained with BURSQUID, which more adequately represents the thermal spectrum and thus gives a better evaluation of the Pu-239 content. This effect had already been noticed in the measurements on the DIMPLE critical facility with plutonium-bearing fuel (see Para 2.6.1). More specifically, an evaluation with the five-group BURSQUID improves the Pu-239 and Pu-240 representation, but introduces higher systematic errors for Pu-241 and Pu-242 ($\sigma = +25\%$ for Pu-242). The U-235 content was slightly underestimated ($\sigma = \approx 5\%$) with both calculation techniques.

As a final comment, it should be pointed out that to obtain high precision with the BURSQUID calculation, it was necessary to use a more detailed calculational model and to take the irradiation history into account as closely as possible.

2.6.4 Verification of the Calculation Technique for the Determination of the Shutdown Margin

The validity of the calculation technique described under Paragraph 2.2.3 was checked with an a priori calculation of the cold criticality condition of the core at the beginning of Cycle 2.

The presence in the core of fresh and irradiated enriched-uranium assemblies with twelve plutonium prototypes provided an interesting opportunity for a verification of the accuracy of the calculation method in such a heterogeneous situation.

As a first trial, a calculation was performed with twenty-two control rods completely withdrawn according to the withdrawal sequence provided for Cycle 2. The resulting K_{eff} was 0.97667.

The calculation was repeated for the assumption that other eight control rods were withdrawn. In the calculation model this situation appeared to be practically critical ($K_{eff} = 1.00100$). Actually, the reactor went critical with one more control rod (C-8) withdrawn up to the 11th notch and at a core temperature of 50°C . The theoretical K_{eff} extrapolated to the real criticality situation was derived from the following balance:

K_{eff} with rod C-8 all in at $T = 23^{\circ}\text{C}$	1.00100
ΔK absorbed by rod C-8 at the 11th notch	<u>-0.00800</u>
	0.99300

The effect of the temperature variation from 23°C to 50°C was negligible.

The agreement between the calculated and experimental data was satisfactory and confirmed the validity of the calculation technique.

2.6.5 Verification of the Calculation Technique for Transients Studies

The validity of the GARDIN code was verified on the basis of experimental situations analyzed on the Garigliano plant. Use was made in particular of the data provided by the load rejection test and recirculation pump trip test carried out at the station on May 14 and July 31, 1969, respectively.

In carrying out the comparison for the load rejection test, it was noticed that the variation in water speed at the core inlet substantially affected the dynamics of the transient; this variation was not recorded during the test. Use was therefore made of an approximated theoretical equation which evaluates the variations in the circulating water flow rate on the basis of the pressure difference between the steam drum and pressure vessel. The variation so calculated permitted the reproduction of the neutron flux variation up to the time of scram (Fig. 2-29). The trend of the neutron flux in the subsequent time interval is similar to that recorded in another test carried out in May 1964 which had not caused reactor scram.

For the pump trip (Fig. 2-30) the agreement is also satisfactory. The time delay between the theoretical and experimental fluxes is due to the time constant (about 500 milliseconds) of the instrument recording of the core inlet water speed.

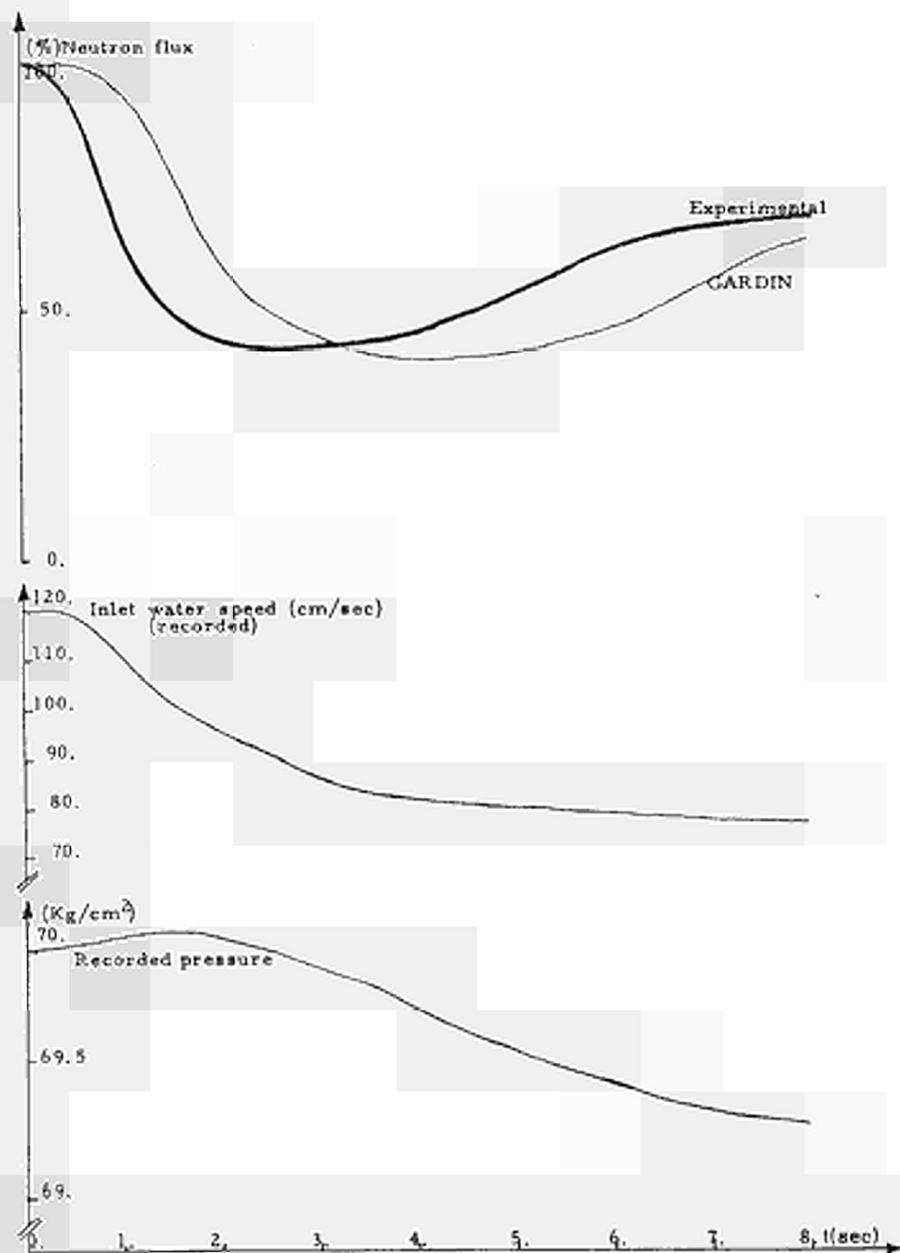


Fig. 2-30 - COMPARISON GARDIN-MEASUREMENT FOR THE TURBINE TRIP

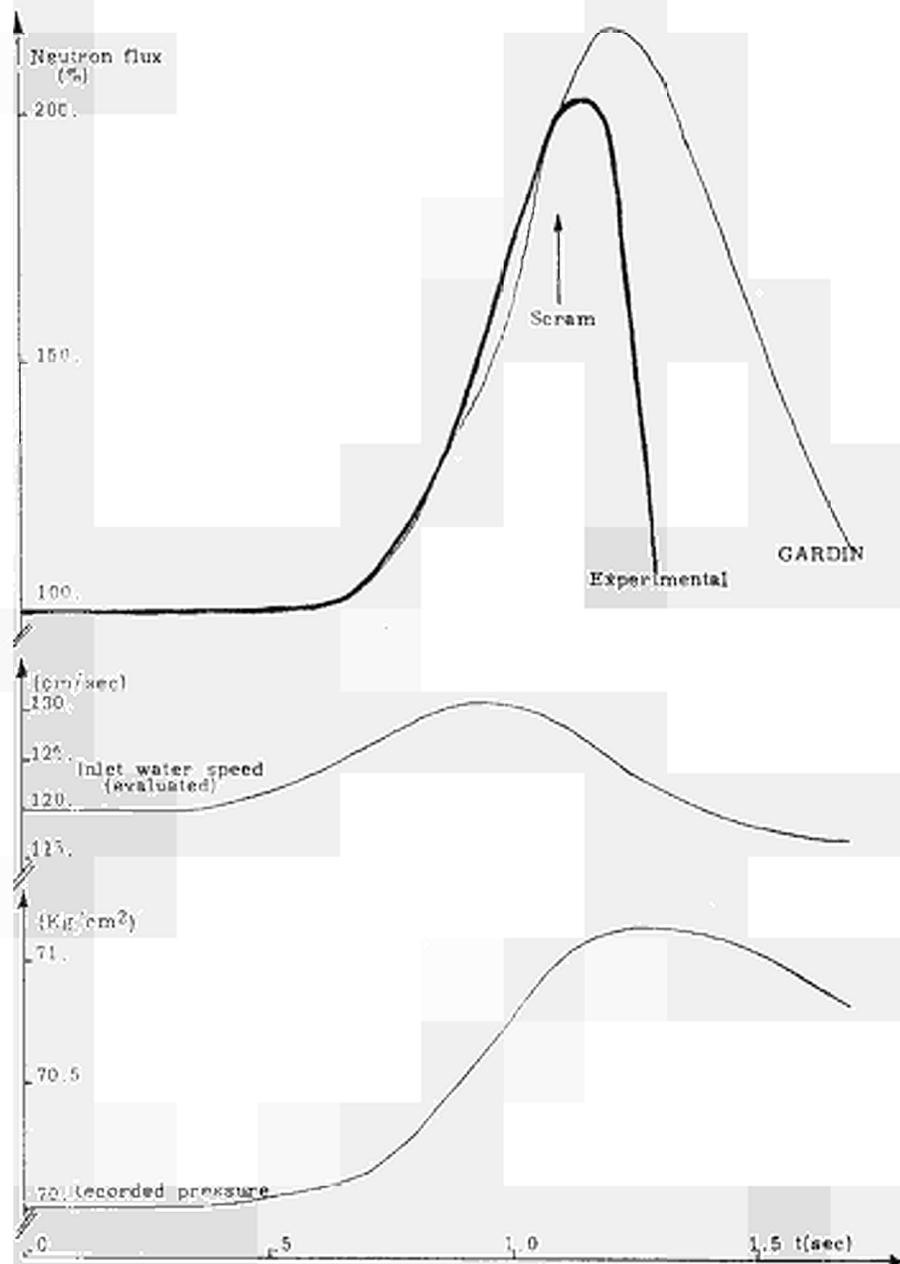


Fig. 2-29 - COMPARISON GARDIN-MEASUREMENT FOR RECIRCULATION PUMP TRIP

2.7 Transients Analyses

In respect of the implications to safety deriving from the use of plutonium assemblies, studies were carried out to determine the changes in core and plant behavior during the most important operational transients and incidents.

The dynamic behavior of the two stations of Trino Vercellese and Garigliano were examined for the two situations of all-uranium and all-plutonium cores and the results compared in order to find out the most important differences. The studies were carried out with the aid of an analog computer; the conditions examined for the main operational transients and incidents, both at startup and at full power, were those contemplated in the Safety Reports of these stations.

For Trino Vercellese, the following transients were considered:

- (a) uncontrolled control-rod withdrawal at full power;
- (b) boron removal;
- (c) steam manifold break;
- (d) load rejection;
- (e) loss of flow;
- (f) startup incident.

The results led to conclude that with a plutonium core the plant will suffer slightly worse consequences only in the case of a steam manifold break.

For Garigliano, the following transients were considered:

- (a) uncontrolled control-rod withdrawal at full power;
- (b) sudden reactivity insertion;
- (c) load rejection;
- (d) closing of turbine cut-off valves and failure of by-pass valves to open;
- (e) loss of flow;
- (f) startup incident.

The effects of the plutonium load on core behavior do not differ substantially from those of the uranium load. In the case of the closing of turbine

cut-off valves concurrent with failure of the by-pass valves to open, whatever the assumption on scram occurrence, the plutonium load gives a somewhat higher neutron flux peak than uranium, whereas the thermal power peaks differ by a few percents. In the case of load rejection from rated power concurrent with the closing of the primary and secondary steam valves, the neutron flux peak is slightly severer with plutonium fuel, but the other parameters of interest--above all, the thermal power--are practically the same.

In brief, from the analysis of startup and full power incidents it appears that the dynamic behavior of the station in respect of the most important parameters would not differ much with a plutonium core. With regard to the Garigliano reactor, the only incidents that cause a worse behavior are load rejection and closing of the turbine cut-off valves concurrent with failure of the by-pass valves to open; on the other hand, the worsening effects involve substantially only the neutron flux peak, since the other parameters vary only by a few percents.

3. TECHNICAL CONSIDERATIONS IN UTILIZING PLUTONIUM IN THERMAL REACTORS

The introduction of plutonium-bearing fuel into a core specifically designed for enriched-uranium fuel has several effects which have been widely studied during the implementation of the ENEL-EURATOM research contract. It must be recognized that several effects imputable to plutonium exist even without recycling plutonium. At high exposures uranium fuel contains plutonium in amounts on the same order of magnitude as the plutonium-bearing fuel. Thus the perturbation caused by plutonium recycle is not an entirely new problem. The quantitative values of these effects depends on the design concept of the fuel and can vary considerably (34). This Section covers the problem areas which have been brought to light by ENEL's studies and may require future efforts for a better understanding of the problems.

3.1 Accuracy of Nuclear Design Methods

Sub-Section 2.6 described the results of the analyses that confirmed that the calculation methods used for enriched-uranium cores can be fairly accurate when applied to mixed plutonium-bearing and enriched-uranium cores, provided that a limited number of modifications are made. The main modifications are the division of the thermal neutron spectrum into two energy groups for diffusion calculations, and a more accurate evaluation of the effective Pu-240 resonance integral for lattice constants calculations.

There remain a number of areas for which sufficient experimental information is lacking to establish the magnitude of error in the predictions. Specific reference is made to the assessment of plutonium assembly burn-up from the standpoints of fissile material depletion, isotopic composition variation, and reactivity loss rate with exposure. However, it should be recognized that, although not decisive, a first confirmation can be obtained from experimental data acquired from irradiated enriched-uranium cores, because their characteristics are to some extent affected by the plutonium produced.

3.2 Power Distributions

The plutonium assemblies or plutonium rods in an assembly generally tend to have an increased power density as compared with enriched-uranium fuel. Power peaking in plutonium rods can be particularly severe near large water gaps or at interfaces between plutonium and enriched-uranium zones. This trend is enhanced as the plutonium content of an assembly increases, both when highly exposed plutonium is to be used and when the burnup level is to be raised.

During the designing of the plutonium assemblies for the Garigliano reactor, it was found that it is generally possible to bring the local power density peak factor down to that of an enriched-uranium assembly by grading the plutonium content in the rods of an assembly much more than is necessary for enriched uranium. In this connection it should be remembered that plutonium permits the fabrication of a multiple-enrichment fuel assembly at a much lower penalty than enriched-uranium. In the case of the Garigliano reactor it was noticed that once an acceptable enrichment pattern is obtained, the local power density peak of the plutonium assembly increases but negligibly when the assembly is adjacent to a uranium assembly. In addition, the power mismatch between fresh plutonium assemblies and irradiated uranium assemblies is roughly the same as is observed between fresh and irradiated uranium assemblies.

Another way to contain possible power peaking, though difficult to realize, could be to improve the moderating characteristics of the lattice; the most promising of all the possible solutions appears to be the use of annular pellets.

The mixed assemblies, containing enriched-uranium rods at the periphery and plutonium rods at the center, may constitute a standby solution for BWR cores, should the all-plutonium assembly not allow adequate power flattening. This type of assembly avoids the increase in local power peaking due to the water gap, whereas sharing between enriched-uranium and plutonium zones can be contained by appropriate grading of the enrichments. It is

also a particularly suited solution for recycling self-generated plutonium in a reactor; in this case only a part of the total core load will be plutonium fuel.

On the contrary, the use of mixed assemblies to reduce power peaking is not justified in PWR cores. The latter have small water gaps around the assemblies (about 1/15 in thickness of the water gaps in the BWR cores), with the result that the power density increase in the nearby rods is very small. With a mixed core, flattening of the power distribution becomes only a question of minimizing power peaking at the interfaces between the enriched-uranium and plutonium zones. From this point of view the choice between the two assembly types (all-plutonium or mixed) would appear to be indifferent. However, it is necessary to distinguish between reactors of the first generation and those of the second.

In the first-generation reactors, the cruciform control rods are coupled to fuel followers, traveling adjacent to fuel assemblies; therefore, the necessity of avoiding a reduction in rod worth by the plutonium lattice would tend to favor the mixed assembly, because the control rod would contact an area entirely of enriched uranium.

The situation is totally different for the cores of the second generation. In fact, the cruciform control rods have been replaced by control rod clusters which penetrate the fuel assemblies. In normal operating conditions (rods out) the assemblies present numerous water holes (16-20) designed to accept the rods of the control cluster. Since these water holes are practically uniformly distributed, it is more difficult to obtain adequate power flattening inside an assembly of the mixed type, because there are two factors to be coped with to contain the power peaking increase in the plutonium rods, namely, the water hole and the Pu-U interface. In addition, the mixed assembly does not permit the avoidance of a significant penalty on the available control rod worth.

In view of the foregoing, the all-plutonium assembly appears to be more suitable in second-generation PWRs to meet power distribution constraints

without a lifetime penalty; this type of assembly also permits the loss in control rod worth in cores containing both plutonium and uranium fuel to be minimized by avoiding locating plutonium assemblies near control rods.

Another aspect of core physics that should be emphasized is the increase in total peak factor that may result from the axial distribution because of the more negative void and temperature coefficients of a plutonium core, which yields a higher power peaking near the core bottom. This increase is expected to be low in a PWR and much more pronounced in a BWR. At the Garigliano, it was estimated to be 10% on the basis of experimental measurements. In a BWR this phenomenon becomes manifest at the end of an operating cycle when the control rods are practically all out and it is no longer possible to shape the axial power distribution. This inconvenience can be remedied, however, by proper programming of the control rod withdrawal sequence so that at the end of an operating cycle the upper part of the core will have burned so as to compensate the void effect.

3.3 Reactor Control

The strength of the reactor control system is slightly reduced with plutonium fuel. Besides, an all-plutonium-fueled core has more negative moderator and void coefficients and a more negative Doppler coefficient than a uranium-fueled core, which results in more stringent control requirements. This might constitute the main problem area in converting an existing reactor to one fueled solely with plutonium. However, it should be emphasized that in each case reactor control may or may not be a problem according to the control system capability, single control rod worth, reactivity requirements, and fuel management. The analyses carried out for the Garigliano reactor indicated that an all-plutonium-fueled core, having the same burnup as the present enriched-uranium reloads, would meet the shutdown margin requirement.

It is interesting to recall the various implications that this problem entails for BWR and PWR cores.

The control rod system is generally designed to meet the following two reactivity requirements with the highest-worth rod all out:

- (a) provide the excess reactivity required to compensate the reactivity depletion over an operating cycle (when no shim poison is used);
- (b) absorb the hot-to-cold swing reactivity.

For a BWR there exists the possibility of compensating the increase in hot-to-cold swing by reducing the excess reactivity at the beginning of the operating cycle. This goal may be attained not only by changing the refueling strategy (shorter operating cycles), but also by using burnable poisons (e. g. Gd_2O_3). It is also interesting to note that, at parity of energy output between two refuelings, the initial excess reactivity at operating conditions is slightly lower for a plutonium core than for a uranium core; the reduction is more pronounced in the case of high-exposure plutonium because of the high content of Pu-240 acting as a fertile poison. This situation works in the direction of facilitating the fulfillment of the control system requirements of a BWR plutonium core.

On the contrary, for a PWR it is not possible to act on the excess reactivity. As a result, if the control system does not have sufficient margin to compensate the control requirements of a plutonium core, it is necessary to increase the number of control rods. Generally, in a chemically shimmed PWR, both head penetrations and core lattice positions are available to accommodate these additional control rods.

The problems associated with control are greatly reduced in the case of mixed uranium and plutonium cores, both because the increase in hot-to-cold swing reactivity is limited and because it is possible to minimize the reduction of the control system strength by adopting a mixed-type assembly (in a BWR or in a PWR with cruciform control rods) or an appropriate positioning of the plutonium assembly (in a second-generation PWR) (35).

In Trino Vercellese reactor it is possible to take advantage of the change in cladding material from the present SS to Zircaloy. The latter causes the moderator coefficient of the plutonium fuel to be almost the same as that of enriched-uranium fuel clad in SS.

3.4 Transients Behavior

The moderator and temperature coefficients, as already mentioned, are more negative in the case of a plutonium-fueled core; this fact is principally imputable to increased Pu-240 captures and is enhanced by fuel with greater contents of Pu-240. Furthermore, the delayed neutron fraction in Pu-239 is less than that in U-235 with a consequent decrease in total delayed neutron fraction; the latter effect is present also in exposed uranium fuel. Therefore, it is reasonable to infer that the plant behavior will be less favorable in the event of transients that cause temperature reductions or void collapse (cold water incident). This is supported by the results of the transients analysis referred to in Sub-Section 2.7.

It was found that for the Trino Vercellese reactor the only incident that worsens is the steam manifold break which causes inleakage of cold water into the core. In this incident, the levels at which power stabilizes after the scram are closely dependent on the overall control rod worth. Therefore, the reduction in control rod worth in a plutonium-fueled PWR is to be analyzed further, as mentioned previously.

For the Garigliano reactor, the transients that become worse are those initiated by a void collapse, and specifically the loss of load, and the closure of the turbine stop valves and concurrent failure of the bypass valves to open. This worsening is reflected almost exclusively on the neutron flux peak, which becomes higher, whereas it has practically no effect on the thermal power transferred to the coolant.

However, it should be pointed out that it has not been possible to make a detailed analysis of very rapid transients, such as the drop-out of a control

rod, nor studies on local criticality. These transients can best be analyzed by means of digital computers, with suitable codes. On the other hand, they are important only if the core is entirely loaded with plutonium so that control rods are completely surrounded by Pu assemblies. With the prototype assemblies loaded so far, the total amount of plutonium in the reactor has not increased appreciably, nor have the assemblies been loaded in adjacent positions; as a result, practically there should be no change as compared to the behavior of enriched-uranium cores.

4. ECONOMIC EVALUATIONS

The economic criteria at the basis of these studies were aimed at ascertaining the main factors affecting the utilization of plutonium for recycling in thermal reactors. The dominant factor in the plutonium value is the fabrication overprice relative to the price of enriched uranium fuel fabrication. In the next few years, the fabrication overprice will probably be high because of the limited quantity of plutonium utilized. Therefore, studies have been carried out in order to find the range of plutonium values which justify the economic incentive to recycle plutonium in thermal reactors.

In this section are summarized the main results of an examination of alternative solutions for the design of a plutonium-bearing fuel assembly and the results of the studies on the influence of certain economic parameters on the plutonium value.

4.1 Alternative Solutions for Plutonium Utilization

Various solutions for the utilization of plutonium in thermal reactors have been considered. The validity of each solution was examined by applying it in the design of a reload fuel assembly for the Garigliano reactor; this procedure appears to be the most reliable because the requirement of meeting the thermohydraulic limitations and nuclear performance of the project might entail, for some solutions, a substantial reduction in their incentive. On the other hand, this incentive is dependent on the values of some of the economic parameters, such as the overprice for plutonium rod fabrication, the cost of U_3O_8 , the cost of money, the fabrication cost for pellets having special characteristics, etc., the evolution of which in the short or medium term is difficult to forecast.

The uncertainty in the economic parameters makes it advisable to pursue the development of the various design solutions for utilizing plutonium in thermal reactors; the more or less positive evolution of a particular technological process or of an economic factor might condition the choice of one solution over another.

In addition, even if plutonium is generally evaluated on the basis of the fissile content only, this evaluation cannot but be affected by the presence of other isotopes, for instance Pu-240 which is a fertile isotope like U-238 but with a much larger cross-section, and Pu-242 which behaves as a poison. On the other hand, the quantities of plutonium available are characterized by percentage contents of Pu-240 and Pu-242 that differ substantially depending on the type of reactor from which the plutonium originates and on the fuel burnup at discharge. Therefore, the fissile plutonium to be recycled in a given reactor may assume a different worth depending on its isotopic content and may call for a different design to allow its best exploitation.

The alternatives subjected to technical and economic assessment by ENEL are:

- (a) plutonium blended with natural uranium for all the rods of the assembly (standard-type assembly);
- (b) enriched-uranium fuel assembly with plutonium in the central region (mixed-type assembly);

with the two subalternatives:

- (c) assembly with pellets of reduced density;
- (d) plutonium blended with the depleted uranium obtained from a Magnox reactor for all the rods of the assembly.

The analysis was completed by evaluating the influence on the value of fissile plutonium, of variations of some of the parameters that directly affect fuel cycle costs, such as burnup, ore cost and interest rate.

The economic analysis was performed by determining the industrial value of plutonium with the "indifference method", which consists in determining the costs of a reference enriched-uranium cycle and of the corresponding plutonium cycle as a function of the value of fissile plutonium taken as a variable. The intersection of the two curves determines the industrial value of plutonium at which the use of either cycle is indifferent.

For the cost of both the uranium and the plutonium fuel cycles, use was made of the ratio between the sum of total net costs and the corresponding gross electric output. The total costs and the electric output were both referred to the same date with the same rate of interest. In establishing the present-worth values, account was taken of the times at which money is paid out or cashed.

The fuel cycle now being carried out at the Garigliano was taken as a reference cycle; this is based on the use of assemblies enriched to 2.3% in U-235 for an average exposure of 20,000 MWd/MTU and on one-fourth reloads in a checkerboard pattern.

The reference economic parameters used for this study are given in the following Table 4-I.

Table 4-I
Reference economic parameters

<u>Unitary costs</u>	
Natural uranium	8 \$/lb of U ₃ O ₈
Depleted uranium	0
Conversion $\left\{ \begin{array}{l} \text{U}_3\text{O}_8 \text{ to UF}_6 \\ \text{U}_3\text{O}_8 \text{ to UO}_2 \end{array} \right.$	2.5 \$/kg of U
Separation cost	26.0 \$/kg separative work unit
Overprice for fabrication of plutonium assemblies in respect of uranium assemblies	0-100 \$/kg of fuel
Fresh fuel transport	4 \$/kg of fuel
Irradiated fuel transport and reprocessing	30 \$/kg of fuel
Interest rate	8.5 %/year
<u>Losses</u>	
Fabrication losses	2.00%
Uranium conversion	1.00%
Reprocessing	1.5%

4.1.1 Standard-Type Plutonium Assemblies

The studies performed and the experience gathered on the performance of the plutonium prototype assembly in the Garigliano reactor indicate the validity of the principle that the mechanical and thermal design of the enriched-uranium assemblies should be used also for the plutonium assemblies. This means maintaining the same coolant flow rate and considering the same heat transfer conditions as for the enriched-uranium assemblies, when the lattice pitch, the rod diameter and the spacer grids are the same. However, to limit power peaking to the same value as for the enriched-uranium assemblies, plutonium calls for a greater differentiation in the fissile contents inside the fuel assembly. In the case of the reload assemblies for the Garigliano reactor, the number of enrichments had to be raised from 2 to 3.

The characteristics of the plutonium assembly, considered for the present evaluation, are summarized in Table 4-II under the heading Pu-3N. In the design it has been assumed that the natural uranium is enriched with the plutonium discharged from a Magnox core at equilibrium, specifically as is produced in the Latina reactor (LE-type Pu in Table 4-III). However, the power distribution in an assembly is relatively insensitive to the plutonium isotopic composition, as demonstrated by the design for the two sets of plutonium assemblies that were loaded into the Garigliano reactor during the 1968 and 1970 refuelings.

Fig. 4-1 shows the trend of plutonium value related to the standard-type plutonium assembly, as a function of the fabrication overprice.

The economic analysis for this solution was completed with an assessment to establish how the values of plutonium vary depending on the composition. The characteristics of the various isotopic mixtures considered are tabulated in Table 4-III with an indication of their origin. The variations in plutonium value from the value of the LE-type Pu are shown in Fig. 4-2. Two interesting facts emerge from this figure, namely:

- (a) that the variations are small and become negligible at high fabrication overprices;

Table 4-II

Burnup and fissile content of the various types of assemblies

Type of assembly	Number of rods	Initial enrichment, gr/kg		Initial Pu content, kg	Irradiation at discharge, MWd/MTU	Final enrichment, gr/kg	
		U-235	(Pu-239+Pu-241)			U-235	(Pu-239+Pu-241)
U-2.3%	64	23.08	--	--	20,000	7.06	4.55
Pu-3N	64	7.00	20.68	5.26	20,300	3.18	11.97
MIXED-2D	28	20.60	--	--	19,700	5.07	4.41
	36	4.85	24.00	3.55		2.49	13.75
Pu-2D	64	4.85	23.67	6.02	20,300	2.28	13.54

Table 4-III
Plutonium types considered in the analysis

Type	Discharged from	Irradiation at discharge, MWd/MTU	Isotopic composition, %				
			Pu-239	Pu-240	Pu-241	Pu-242	Pu _{fissile}
L1	Latina Magnox reactor	1,000	89.0	9.7	1.2	0.1	90.2
LE	Latina Magnox reactor	3,500	74.5	20.5	4.5	0.5	79.0
GR	Garigliano BWR	20,000	60.0	26.0	11.0	3.0	71.0
T3	Trino Vercellese PWR	25,500	66.5	17.5	13.0	3.0	79.5
CR	Caorso BWR	27,500	50.0	31.0	14.0	5.0	64.0

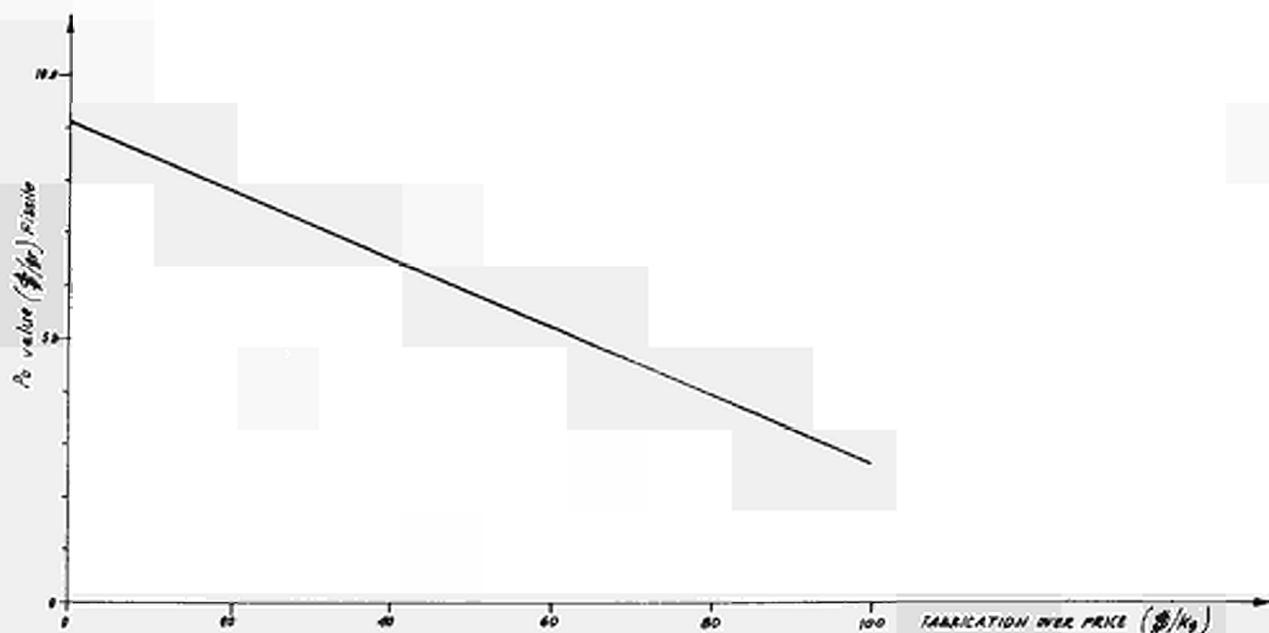


Fig. 4-1 - PLUTONIUM VALUE AS A FUNCTION OF THE FABRICATION OVERPRICE FOR THE STANDARD-TYPE ASSEMBLY CASE (PLUTONIUM BLENDED WITH NATURAL URANIUM FOR ALL THE RODS OF THE ASSEMBLY).

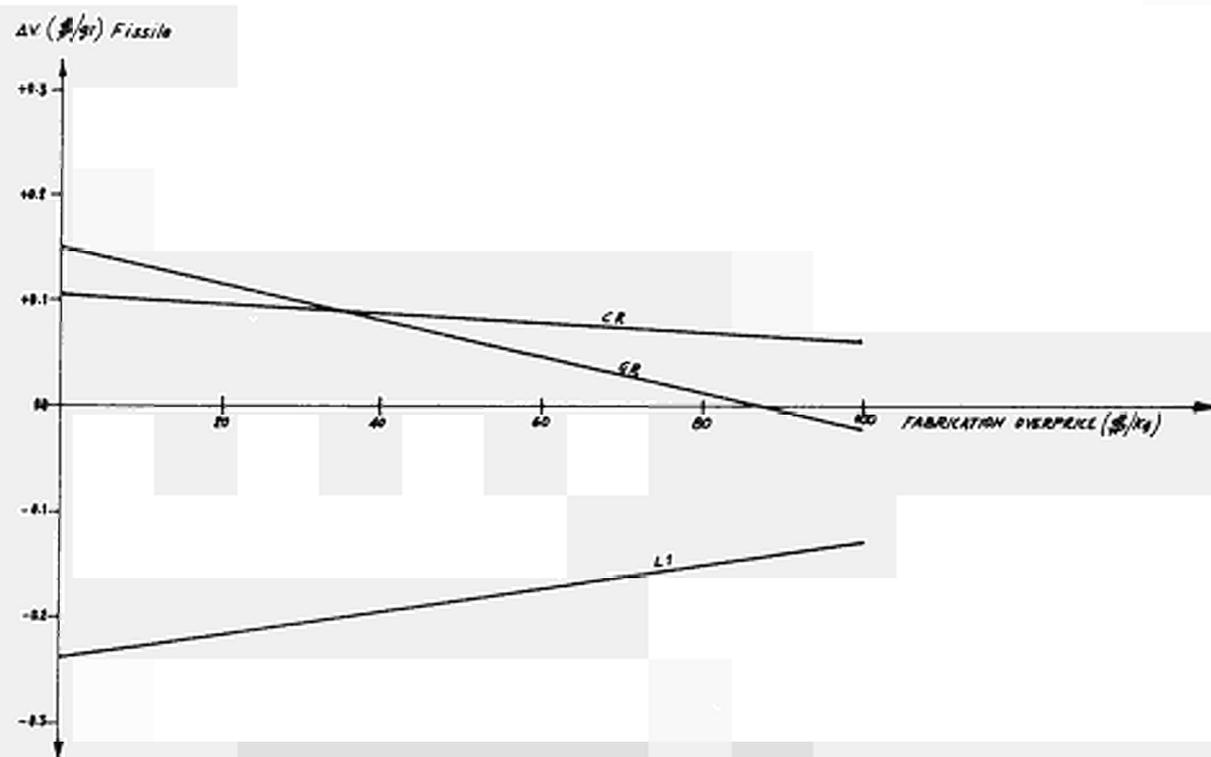


Fig. 4-2 - VARIATIONS IN PLUTONIUM VALUE, ΔV , RESULTING FROM THE USE OF DIFFERENT TYPES OF PLUTONIUM COMPARED WITH TYPE LE

(b) that plutonium containing large amounts of Pu-240 and negligible amounts of Pu-242 can be utilized for recycling to better advantage than "clean" plutonium (about 90% of fissile). This is true provided that the fabrication costs are not significantly affected by the Pu-241 content of the material.

The plutonium-natural uranium standard type assembly is taken as reference solution to assess the merit of the other alternatives. In fact, since the core was originally designed for enriched-uranium fuel, this solution has the advantage of introducing the smallest number of modifications in the original core design, even though it exploits the plutonium potential less (for instance, the moderator-to-fuel ratio is not optimized for a plutonium lattice).

4.1.2 Mixed-Type Plutonium Assembly

From a technical standpoint this solution presents both advantages and disadvantages in respect of the preceding solution. The disadvantages may turn into advantages and vice versa when referred to a BWR rather than a PWR and in passing from one assembly to another. As an example it will suffice to mention that one of the main problem areas of plutonium recycle in the existing reactors is the reduction of the control rod worth which determines an increased control rod requirement by the core. In a BWR this solution permits the reduction in control rod worth to be minimized, whereas in a PWR--which uses a cluster control rod system--it greatly penalizes the available control rod worth.

It is, however, a fact that the mixed-type plutonium assembly is an arrangement that allows great flexibility in the amount of plutonium to be recycled, which is a feature of particular interest in the case of self-generated plutonium recycles. The number of plutonium rods may, in fact, vary from a few units to more than half the number contained in a fuel assembly. Instead, with plutonium standard-type assemblies, to obtain this flexibility it is necessary to resort to mixed reloads of plutonium and enriched-uranium assemblies. However, the core loading do not always afford wide degrees of freedom.

Studies carried out confirmed the flexibility of the mixed-type assembly alternative mainly for the possibility of reducing the number of enrichments and, at the same time, of varying the number of plutonium rods. For instance, for this type of assembly a new design was developed by ENEL which considers the use of:

- three enrichments: two for the plutonium rods and one for the enriched-uranium rods;
- thirty-six plutonium rods out of sixty-four for the other characteristics (see Table 4-IV).

Of the two possible alternatives--reduction of the number of enrichments in the enriched-uranium rods or in the plutonium rods--we obviously preferred the former, as the cost penalty, associated with multiple enrichment is certainly higher for U-235.

In the design mentioned above, the reduction of the number of enrichments was obtained at the expense of power flattening in the fuel assembly.

The economic analysis demonstrated that in practice there is no difference between the two alternatives represented by the reference solution and the mixed-type assembly as concerns the value of plutonium. Specifically, with the mixed-type assembly solution, plutonium assumes a value by 0.2 \$/gr of fissile lower than the reference solution.

This small difference is conditioned on the assumption that the supply unit price for plutonium rods of a mixed-type assembly is the same as for standard plutonium assemblies. Actually, the fabrication price for a supply of both plutonium and uranium rods may be higher than the price for a supply of a single type of rod because of the fragmentation of the supply.

4.1.3 Recycling Plutonium with Depleted Uranium

Studies performed by other researchers indicated that the optimum U-235 content of the uranium to be blended with the plutonium is about the same as that of natural uranium or slightly higher (36). These studies are generally based on

a relatively high plutonium value and adopt the price of depleted uranium published in the USAEC Standard Table of Enriching Services.

However, the selection of the working assumptions is particularly important because the economic incentive to use depleted uranium is closely linked to the saving represented by the difference between the costs of natural and depleted uranium. Obviously the greater consumption of fissile plutonium when used with depleted uranium reduces this saving and the reduction is greater as the plutonium value is higher.

Actually, ENEL has considerable quantities of depleted uranium with a relatively high residual U-235 content (about 0.5%) that have been made available from the reprocessing of fuel of Magnox reactors and for which no use has been found so far. Therefore, its value is practically zero or, at the best, distinctly lower than the figures indicated in the USAEC Tables. On the other hand the value of fissile plutonium for recycle in thermal reactors is, at least initially, less than 10 \$/gr, since the fabrication price for plutonium fuel is appreciably higher than for enriched-uranium fuel. Under these circumstances, the use of depleted uranium as a diluent for plutonium in the fabrication of plutonium elements could be particularly promising.

To evaluate the merits of this combination, the standard-type plutonium assembly referred to in Sub-Section 4.1.1 was redesigned to allow the use of uranium depleted to 0.5% in U-235 (U-236 = 0.035%) instead of natural uranium, at par of burnup.

A mean content of 2.37% of fissile plutonium is necessary to obtain a burnup of 20,000 MWd/MTU instead of the 2.07% required for the reference element. (Compare the characteristics of the assemblies Pu-2D and Pu-3N in Table 4-II.)

In the economic assessment it was assumed that the cost of depleted uranium was zero and that its conversion to sinter-grade UO_2 could be obtained at the same price as natural uranium. Fig. 4-3 shows the results of this economic analysis; it can be noted that plutonium is considerably uprated by the

ΔV (\$/gr) Fissile

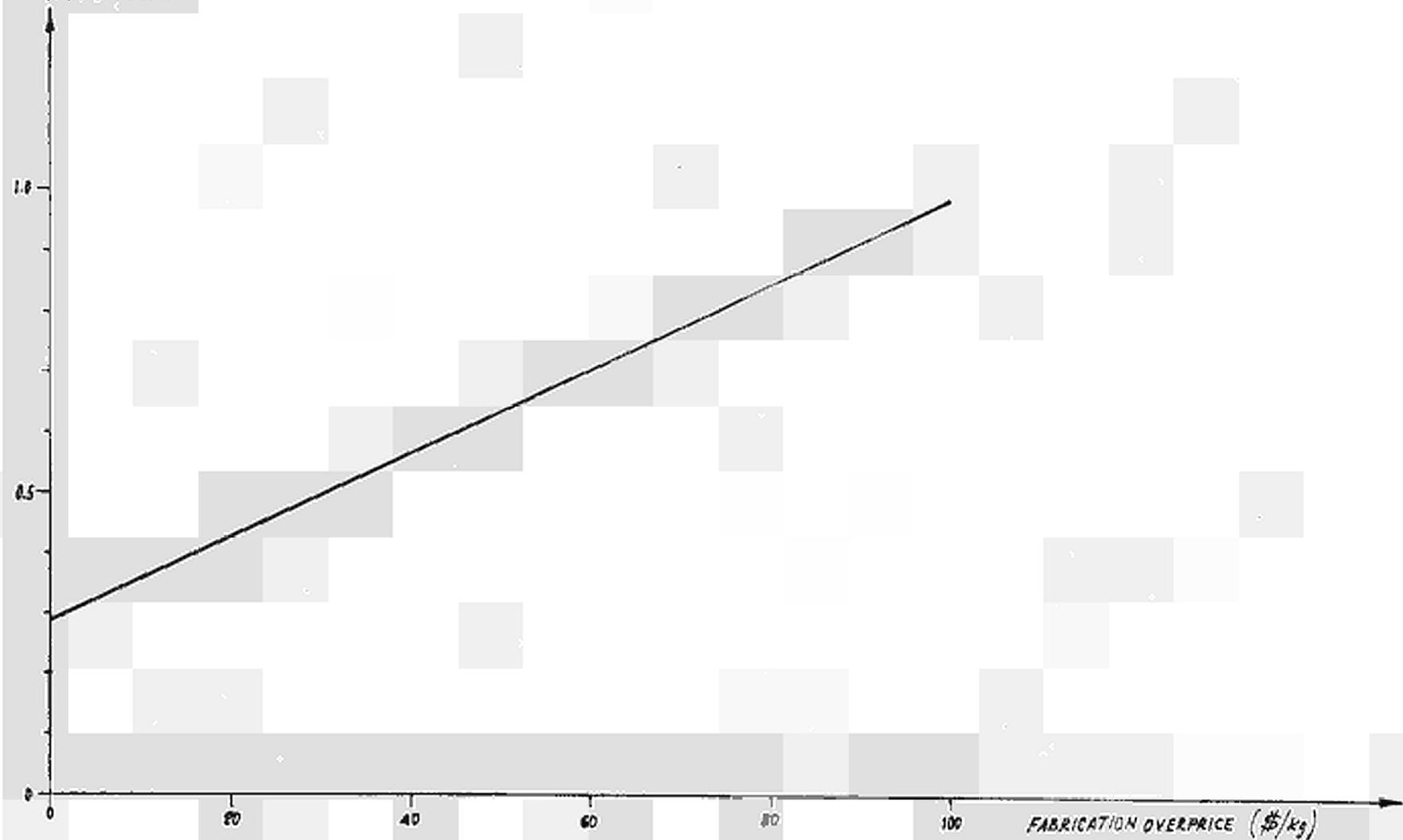


Fig. 4-3 - INCREASE IN PLUTONIUM VALUE, ΔV , RESULTING FROM USE OF DEPLETED URANIUM

use of depleted uranium, the increase in its value being between 0.4 and 1 \$/gr of fissile for fabrication overprices varying from 0 to 100 \$/kg.

The advantages of the use of depleted uranium were assessed also for the case of U_3O_8 prices less than 8 \$/lb. The incentive of using depleted uranium is smaller, but it is still appreciable when the fabrication overprice is high. For a U_3O_8 cost of 5 \$/lb, the range of increase is between 0.1 and 0.7 \$/gr of fissile

4.1.4 Plutonium Fuel Bundle with an Increase Moderator-to-Fuel Atom Ratio

All the alternatives considered above are conditioned by the criterion of adopting for the plutonium assemblies the same mechanical design as the enriched-uranium assemblies. As a result, the lattice of the uranium assembly does not give the plutonium fuel the best moderator-to-fuel, M/F, atom ratio. The nuclear characteristics of plutonium are fitter for higher moderator-to-fuel ratios. From a technical standpoint, a higher M/F ratio leads to less negative temperature and void coefficients and to improved behaviour during transients and improved thermohydraulic stability of the plutonium core.

The economic merits of using the optimum lattice for plutonium were considered on the condition that the plutonium and uranium reload assemblies are compatible and can be used interchangeably. This means that it is possible to change the lattice and rod dimensions, while keeping the channel dimensions unaltered.

The solutions available for consideration to increase the M/F atom ratio are:

- (a) increased number of rods per bundle and corresponding reduction of their diameter;
- (b) reduced fuel density.

The first solution would allow a higher M/F ratio without altering the heat transfer area of the core. However, the economic penalty associated with the larger number of fuel rods per bundle offsets the advantages of the better utilization of plutonium.

If the core presents a margin in respect of the limiting operating conditions and this margin is to be used up for better cycle economy, there is another alternative to be considered. At parity of number of rods used, it is possible to reduce their diameter without introducing a further penalty on the fabrication cost.

Since the Garigliano reactor is in the conditions described above, a preliminary assessment was made of the possible advantages to be derived from the use of smaller-diameter rods. The assessment was made for pellet diameter reductions of the plutonium assembly referred to in Sub-Section 4.1.1 (normal O.D. = 1.29 cm) corresponding to 10% and 20% decrease of fuel weight per assembly, as shown in Table 4-IV.

Table 4-IV

Percentage variations associated with rod diameter reductions

Item	Variation, %	Variation, %
Fuel weight per assembly	-10	-20
Fissile material weight per assembly	-2	-1.5
MWd/MTU	+11	+25
MWd/assembly	0	0
Average heat flux	+4.5	+10
Linear power density	0	0

At parity of MWd per assembly, the smaller-diameter rods allow the natural uranium requirements to be reduced appreciably without decreasing the plutonium inventory substantially. This is obtained at the expense of a higher heat flux and specific burnup. Therefore, the economic incentive is of little consequence, once it is borne in mind that this improvement means operating the fuel in severer conditions.

The alternative (b) above is a priori less attractive than the other because it allows a reduction of the amount of fuel in the lattice, but not an increase of the volume moderator, but it is a way of improving the lattice for a reactor which has no power margin. In addition, the low-density fuel is characterized by lower thermal conductivity and this may necessitate operation at a lower linear power, kW/ft, in order to retain an acceptable thermal performance.

This limitation can be overcome by fabricating the pellets in a hollow form. This type of fuel allows operation at higher linear power densities than normal pellets without causing center melting and at the same time it allows the apparent fuel density to be changed over a wider range than would be possible with low-density fuel. However, just as for the smaller-diameter rods, the economic incentive is still extremely limited. As a matter of fact, the plutonium value could be adversely affected if the fabrication of hollow pellets entails a penalty. However, this form of fuel could find its best application in the development of fuel having higher power densities.

4.2 Influence of Some Fuel Cycle Parameters on Plutonium Value

The assessment of the plutonium recycle presented in the preceding paragraphs is based on the economic parameters shown in Table 4-I. Of these, a few cost components fluctuate or may be expected to change in future, such as the price of U_3O_8 or the separative cost of enriched uranium. Moreover, both the uranium reload and the plutonium assemblies considered in the assessment are designed for a burn-up of 20,000 MWd/MTM. The present technological knowledge of uranium oxide rods and the experience acquired in uranium-plutonium rod fabrication justify the expectation that the plutonium assemblies will also be able to reach burn-ups of about 25,000 MWd/MTM or more. It should be noted that the incentive to increase the burn-up potential is certainly greater for plutonium assemblies than for enriched-uranium assemblies because of the higher fabrication cost of the former.

The influence of these parameters was assessed by assuming the intervals of variations that can reasonably be expected for the early period of plutonium recycle. It will be noted in Table 4-V summarizing the results of this analysis that, except only for the interest rate, the parameters all determine relatively large variations in plutonium value.

The increase in plutonium value obtained through higher burnup varies linearly with the fabrication overprice, and goes up to 1.4 \$/gr Pu_f of fissile for a fabrication overprice of 100 \$/kg of fuel.

It should be borne in mind that, in applying the indifference method, the reference uranium cycle was redesigned to give the same increase in burnup as the plutonium cycle. This explains why the plutonium value appears to be practically unaffected by a zero fabrication overprice and also indicates--as demonstrated by special studies--that the present Garigliano cycle is still liable to further optimization.

Table 4-V

Change in fissile plutonium value due to a variation of some economic parameters

Parameter	Reference value	Change	Change in Pu value, (\$/gr)fissile	
			(a)	(b)
Ore price, \$/lb U ₃ O ₈	8	± 2	± 1.3	± 1.3
Separative work cost, \$/kg _{swv}	26	± 4	± 0.7	± 0.7
Energy yield, MWd/MTM	20,000	± 5000	± 0.1	± 1.4
Carrying charge rate, %	7.5	1.0	negligible	negligible

5. CONCLUSIONS

The main interest in utilizing plutonium in thermal power reactors stems from the fact that very large quantities of this material are being produced as a by-product in the reactors operating today, and these quantities will continue to increase with the installed capacity. Although plutonium can best be utilized in fast reactors, today it appears unlikely that these reactors will develop soon enough to represent an attractive alternative for the use of plutonium. The predicted build-up of this material involves the immobilization of vast financial resources and requires costly storage facilities. It is therefore obviously important to find a form of plutonium utilization in the short term, and this is recognized by all the utilities concerned.

The accrued amount of fissile plutonium available to ENEL in 1973 will be about 900 kg, to which about 200 kg will be added each year. Part of this amount will be used to promote the implementation of any research or prototype fast reactor projects that ENEL may be interested in. At any rate, it is expected that there will remain a sufficient amount for recycle in Garigliano and Trino reactors.

So far, the analyses carried out under the subject Program and the experimental results acquired from irradiation of the plutonium prototypes in the Garigliano reactor prove that there are no technical difficulties opposing the use of plutonium in ENEL's water reactors, and that any problems encountered can be solved by tactical expedients. It is therefore merely a matter of economics. Indeed, the advantage of recycling plutonium is greatly dependent on the fuel fabrication costs that the operators will succeed in obtaining in the forthcoming years. ENEL's calculations indicate that with fabrication overprices in the range of 30 to 70 \$/kg--which are considered probable in the near future--the industrial value of plutonium in water reactors is on the order of 7 \$/gr of fissile, assuming that the depleted uranium discharged from the Latina Magnox reactor is used as a diluent.

Once the industrial value of plutonium in thermal reactors is established, it is necessary to compare this figure with the value of plutonium obtained in a mixed economy of thermal and fast reactors--after the advent of the latter--and compute the present worth referred to the date of interest. Studies are under way on ENEL's nuclear program for the 1980's, 1990's, 2000's and beyond, based on systems analysis, to verify the plutonium value indicated above. Of course, the results are greatly conditioned by the long-term assumptions of plant cost and fuel cycle cost for the different reactors; however, it is possible to establish a range of values for plutonium used in fast reactors.

The preliminary results of these studies, based on the foreseeable rate of increase of installed nuclear capacity on ENEL network, indicate that there will still be an incentive to recycle plutonium in thermal reactors for many years to come, provided that the fabrication overprices for thermal reactor fuel will remain within reasonable limits.

It is premature to set the date on which plutonium recycle will have to cease, because more accurate calculations will be possible only when more information is available on the costs respectively involved by fast and advanced thermal reactors, and on the trend of uranium ore cost, the latter depending on the results of prospections in the coming years.

In the meanwhile, ENEL contacted plutonium fuel manufacturers interested in fabricating assemblies for thermal reactors. More precise information on the matter will become available in the near future since ENEL has issued an enquiry for the supply of reload fuel for several years for Trino Vercellese station with an option to supply plutonium fuel commencing in 1973-74, and a similar policy will probably be adopted also for the Garigliano station.

6. LIST OF REPORTS ISSUED DURING THE IMPLEMENTATION OF THE CONTRACT

- Risultati dei lavori effettuati dal 1^o giugno al 30 settembre 1966. Relazione trimestrale n. 1. Doc. 4.811/3.
- Risultati dello studio relativo alle prospettive di utilizzazione del plutonio nei reattori ad acqua dell'ENEL. Doc. 4.811/4.
- Risultati dei lavori effettuati dal 1^o ottobre al 31 dicembre 1966. Relazione trimestrale n. 2. Doc. 4.811/5.
- Risultati dei lavori effettuati dal 1^o gennaio al 31 marzo 1967. Relazione trimestrale n. 3. Doc. 4.811/6.
- Risultati dei lavori effettuati dal 1^o giugno 1966 al 31 maggio 1967. Relazione annuale n. 1. Doc. 4.811/7.
- Risultati dei lavori effettuati dal 1^o aprile al 30 giugno 1967. Relazione trimestrale n. 4. Doc. 4.811/8.
- Risultati dei lavori effettuati dal 1^o luglio al 30 settembre 1967. Relazione trimestrale n. 5. Doc. 4.811/9.
- Risultati dei lavori effettuati dal 1^o ottobre al 31 dicembre 1967. Relazione trimestrale n. 6. Doc. 4.811/10.
- Risultati dei lavori effettuati dal 1^o gennaio al 31 marzo 1968. Relazione trimestrale n. 7. Doc. 4.811/11.
- Accuracy of Power Distribution Calculation Methods for Uranium and Plutonium Lattices based on Recent Experiments and ENEL Reactor Operation Data. Doc. 4.811/12.
- Risultati dei lavori effettuati dal 1^o aprile al 30 giugno 1968. Relazione trimestrale n. 8. Doc. 4.811/13.
- Second Annual Report for Year Ending 31 March 1968. Doc. 4.811/14.
- Risultati dei lavori effettuati dal 1^o luglio al 30 settembre 1968. Relazione trimestrale n. 9. Doc. 4.811/15.
- Risultati dei lavori effettuati dal 1^o ottobre al 31 dicembre 1968. Relazione trimestrale n. 10. Doc. 4.811/16.

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- Risultati dei lavori effettuati dal 1^o gennaio al 31 marzo 1969. Relazione trimestrale n. 11. Doc. 4.811/18.
- Third Annual Report for Year Ending 31 May 1969. Doc. 4.811/19.
- Risultati dei lavori effettuati dal 1^o aprile al 30 giugno 1969. Relazione trimestrale n. 12. Doc. 4.811/20.
- Risultati dei lavori effettuati dal 1^o luglio al 30 settembre 1969. Relazione trimestrale n. 13. Doc. 4.811/21.
- Risultati dei lavori effettuati dal 1^o settembre al 31 dicembre 1969. Relazione trimestrale n. 14. Doc. 4.811/22.
- Results of work performed from 1 January - 31 March 1970. Quarterly report No. 15. Doc. 4.811/23.
- Experimental and theoretical determination of burnup and heavy isotope content in a fuel assembly irradiated in the Garigliano reactor. Doc. 4.811/24.

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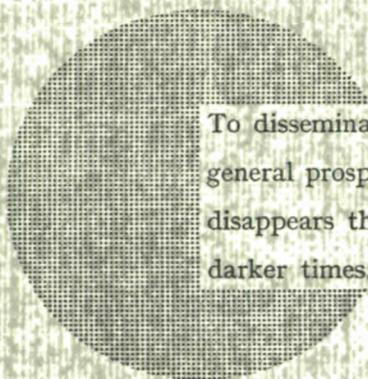
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To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

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