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Reactor Physics Activities Relevant to FBR and ATR Programmes in PNC, Japan November 1978 ∿ September 1979

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1. Introduction

Since last meeting of NEACRP, reactor physics activities have been performed in PNC, Japan, to support FBR and ATR development programmes.

The experimental fast reactor JOYO attained the second-stage thermal power of 75 MW in July 1979. Environmental impact of the prototype fast breeder reactor, MONJU, is now being assessed by both Japanese government and prefectural government and its construction will be started in 1981.

A prototype reactor of heavy water-moderated, boiling light water-cooled, pressure tube type, FUGEN, went into its commercial operation in March 1979.

2. Experimental Fast Reactor, JOYO[1]

After successful operation at the first-stage thermal power of 50MW during three cycles, the second-stage thermal power of 75MW was attained in July 1979 which was ceiling capacity of the present core configuration (MK-1 core) without any mechanical modification to the reactor.

This ceiling was being confirmed by evaluating the results of various tests and detailed analyses of core performance after the first construction permit relative to the first-stage thermal power of 50MW was issued in 1970.

As the performance tests at the thermal power up to 75MW, various transient tests such as the loss of external power supply, the trip of the main sodium circulating pumps or the small reactivity insertion etc. were carried out during about one month after attaining the first thermal power of 75MW.

At the end of last August, JOYO has reached the accumulated thermal power of 2.1×10^5 MWH and the maximum fuel burnup of about 1.4×10^4 MWD/T as the pin average. Five irradiated fuel assemblies were transfered to hot laboratories in order to conduct PIE.

3. Prototype Fast Breeder Reactor, MONJU

In the design refinement carried out in fiscal 1977 and 1978, some modifications and detailed specifications were made to get ready for safety evaluation and subsequent start of construction. In fiscal years 1978 and 1979, the preparatory manufacturing design is under way.

The Shiraki Region in Tsuruga City, Fukui Prefecture, the candidate construction site, was surveyed from the points of geological, meteorological and seismic conditions. After the findings of these surveys were submitted to the prefectural government where it was accepted in July 1978, environmental assessment was commenced, and following safety review by the regulatory authority, the construction of MONJU is to be started in 1981 with schedule of attaining criticality in 1987.

4. Design Study of the Demonstration Fast Breeder Reactor

Preliminary design studies have been continued for a 1000MWe LMFBR Demonstration Plant, whose target of construction will around the latter half of 1980s. In fiscal year 1978, the principal objective is to review the appropriateness of extending the MONJU concept and design techniques to a Demonstration Plant. Some results of the activities related to physics area were presented at Aix-en-Provence Symposium.(2)

5. Heavy Water Moderated, Boiling Light Water Cooled Reactor, FUGEN(3)

FUGEN (165 MWe), a prototype reactor, of heavy water-moderated, boiling light water-cooled, pressure tube type, reached the full power operation on 13th November 1978 and its commercial operation was licenced by the government on 20th March 1979 on the completion of 100 hours continuous full load operation.(4)

The produced power was 880,000 MWh(e), which corresponds to 220 full power day, by the end of August 1979.

Favourable reactor characteristics and performance have been shown by the operating experience, and steady operation is expected to continue, which will enable operation and maintenance data to be fully obtained to support the reliability of the plant, as well as supplying useful data for the design of the demonstration plant.

The 600 MWe demonstration reactor will be the second step of the FUGEN project, and its conceptual design is already completed.

6. Fast Reactor Physics Activities

1) Mockup Experiment and Analysis

As a part of the mockup experiment for MONJU, experiments of the B_4C absorber rod introduced in the radial blanket region have been carried out on FCA VII-2S assembly which was composed of the simulating blanket of depleted uranium oxide and sodium plates in a quarter of the radial blanket region, as shown in Fig. 1. Distortion of the radial power distribution was investigated by traversing a micro fission chamber. It was found that the power density was lowered by about 4% at the inner core edge and more than 10% in the outer core region by introducing the B_4C rod containing about 1 kg of ^{10}B, as shown in Fig. 2.

Analysis on reactivity worth of B_4C control rods which had been measured in the radial blanket of FCA VII-2S was made basing on two-dimensional X-Y diffusion theory with anisotropic diffusion coefficients and 25-group of JAERI-FAST set version 11. The effective cell averaged cross-sections are produced using a collision probability calculation code to take account of heterogeneity in the core, blanket and control rod. The calculated B_4C control rod worth underestimates the corresponding measured one by 10 $\sim$ 20%. The C/E of control rod worth is lower by 10 $\sim$ 20% in the radial blanket than in the center of the core. The difference of control rod worths obtained by diffusion and transport theories is fairly large in the radial blanket.

On FCA VIII assembly, experiments mocked-up fuel slumping and ones in order to measure the distribution of $^{10}B(n,\alpha)^7Li$ reaction rate in the control rod are being carried out for MONJU this year.

The large core critical experiments called "JUPITER" for the DOE-PNC joint program have been conducted at ZPPR. Preliminaly analyses for criticality, central reaction rate ratios and reaction rate distributions have been carried out on ZPPR-9. The evaluated nuclear data file "JENDL-1" was used to calculate. The agreement between the calculated and the measured values are satisfactory at present. Detailed analyses with "JENDL-2" have been undertaking.

2) Assessment of Nuclear Reactor Constants Sets

Interpolation method of resonance shielding factor, group collapsing method with bilinear weight, and group structure were assessed for the purpose of generation of new nuclear reactor constants set using the JENDL-2 nuclear data file, and an adjustment plan was proposed for the methodology used.

Concerned with the interpolation method of shielding factor, various method used so far and, additionally, spline fitting method and Akima's method were intercompared in connection with computing accuracy and time. The spline method was found to give the most excellent and stable accuracy. A recommendation is made for the nodal points of the spline interpolation ; four points for temperature, T=300, 900, 2100 and 3500°K, and eight points for background cross section, σ_0=0, 1, 10, 10^2, 10^3, 10^4, 10^5, 10^6b., respectively.

Concerning to the bilinear-weight collapsing method, the difference between the flux- and bilinear-weight collapsings was studied by considering a group collapse from seventy to twentyfive groups and by analyzing reactivity worth measurements for sodium void at ZPPR-2 and steam entry at ZPR-9 Phase 1. The flux-weight collapsing method used so far was shown to be sufficient for the analysis of the measurement of sodium void reactivity worth.

A group structure with equal lethargy width of $\Delta\mu$=0.25 and total group number of seventy is recommended for the generation of new nuclear reactor constants set.

3) Development of Core Analytical Method

Heterogeneous structure of the fuel pin assembly may exert influence on the neutronic properties of a fast reactor such as criticality factor, sodium-void reactivity, and Doppler coefficient. A study is performed to examine this effect quantitatively in a proto-type and a demonstration fast reactor. The effect comes both from the pin structure of the loaded fuel, and from the gross heterogeneity formed by two distinctive regions in a fuel subassembly, namely, the lumped fuel-pins in the central part and the peripheral wrapper-tube region.

It is shown that the effect on k_{eff} is as large as +0.6% $\Delta k/k$, more specifically, +0.3% from the fuel pin heterogeneity and +0.3% from the gross heterogeneity of the subassembly. The former is caused predominantly by the enhanced resonance shielding of the ^{238}U capture, and the latter comes also from the flux advantage in addition to the resonance effect.

The pin heterogeneity effect on sodium-void reactivity worth and Doppler coefficient turned out to be sizable. The maximum sodium - void worth is reduced by 20%, and the whole-core isothermal Doppler coefficient is enhanced by 10%.

Further, the plate heterogeneity effect in mockup critical assemblies is examined, and is compared with the pin fuel heterogeneity effect described above. In the case of the plate type fuel, both the spatial variation of the inner-cell flux and the enhancement of the ^{238}U resonance shielding are the main contributors to the heterogeneity effect on k_{eff}. On the other hand, the latter dominates the total effect as far as the pin-type fuel is concerned. In addition, it is shown that the component-wise contributions to sodium-void reactivity in pin environment are considerably different from those in plate environment. From these aspect, necessity and usefulness of the data from pin mockup experiment in a critical assembly are stressed.

4) Research on Shielding

For the shielding design and analysis of FBR, two numerical methods, discrete ordinates and Monte Carlo, are most important, and the applicabilities of these methods have been studied for years.

The two dimensional discrete ordinate transport codes have been widely used in the shielding analysis of fast reactors, and the method of these codes was proved to be very effective. Based upon these experiences, the development of a new technique which utilizes the spatial channel theory has been attempted to make the optimum spatial arrangement of the reactor shields. This technique, when applied to the system with various streaming pathes, finds the region which most contributes to the dose rates at the point of interest. The reactor vessel support structure as shown in Fig. 3 in Experimental Fast Reactor, JOYO, was analysed by this technique, and it was confirmed that the gap between the pedestal and the reactor vessel is the principal streaming path which contributes to the neutron dose rate at the driver mechanism of the rotating plug. One example of total response flux is shown in Fig. 4.

The analysis of the neutron streaming through the complex geometry requires the Monte Carlo method. We developed the albedo Monte Carlo code system for the estimation of neutron streaming along the ducts through the shield of the FBR. The code system was applied to the analysis of the mock-up experiment for the SNR at the EURACOS of the EURATOM, and the availability was confirmed. The calculation was performed with 11 group constants and albedo data, and the results were agreed to the experimental values within factor 3.

Reliability of shielding design calculation, by discrete ordinates or Monte Carlo method, greatly depends on the accuracy of cross-section data. Sensitivities of detector responces to the cross-section data for the radiation penetration through shields of JOYO were studied with linear perturbation theory. The analysis was based on the one dimensional calculation by ANISN, and the accuracies of the following cross-section data were shown to be important: neutron cross-sections of sodium and SUS304, gamma ray cross-sections of concrete and graphite. Moreover, it became clear that the responses (dose or counting rate) are highly sensitive for down scattering cross-sections of fast neutron in regions near the core, while sensitivities for neutron capture and in-group scattering cross-section and those for gamma ray cross-sections gradually increase as the location comes nearer to the detector region.

5) Energy Release Rate of Fission Products[5]

To estimate the decay heat of the fuel in FBR, the gamma ray energy release rates of the fission products have been measured for the fast neutron fission of U-235, U-238, natural U, and Th-232. (Experiments of Pu-239 are scheduled for the next year). The samples were irradiated at the fast neutron source reactor "YAYOI" of University of Tokyo. The gamma ray energy spectra were obtained by unfolding the measured pulse height distributions by NaI scintillation spectrometer, and the spectra were integrated to give the gamma ray energy release rates.

These experimental results of the gamma ray energy release rates were compared with the results of the calculations, and the decay data used in the analysis were discussed.

6) Study of Large Heterogeneous Core

Large heterogeneous core configuration (1000MWe) with the radial internal blanket has been optimized to improve breeding performance, and optimum fuel pin diameter has been studied. Details were presented at Aix-en-Provence Symposium.[6]

The safety as well as the neutronic and thermal-hydraulic characteristics of heterogeneous core have been also studied in contrast with the characteristics of the conventional homogeneous core. Although these comparisons have led to results somewhat in favor of the heterogeneous type, a definite conclusion could not be drawn because of rather rough models employed in the analysis, especially, concerning the fuel slumping and sodium void propagation. More detail studies will be continued.

7. Reactor Physics Activity in DCA[7]

For the investigation of the nuclear characteristics of the commercial-scale FUGEN type reactor, studies of reactor physics parameters have been continued by using uranium and plutonium fuels in DCA (Deuterium Critical Assembly)

Recent measurements have been made, to obtain basic information for nuclear design of the commercial-scale reactor, on 25.0 cm pitch square lattices loaded with 0.79 w/o enriched PuO_2-UO_2 clusters which have 54/60 fuel pins of 12.5 mm diameter. The measurement items in the DCA's core are: (1) lattice parameters such as δ^{25}, ρ^{28}, δ^{28}, δ^{49}_{25}, δ^{49}, and thermal disadvantage factors, (2) microscopic and macroscopic power distributions in the core, (3) material bucklings, and (4) loss-of-coolant reactivities.

Properties of Gd burnable poisoned fuel which aims higher burn-up are experimentally and calculationally studied.

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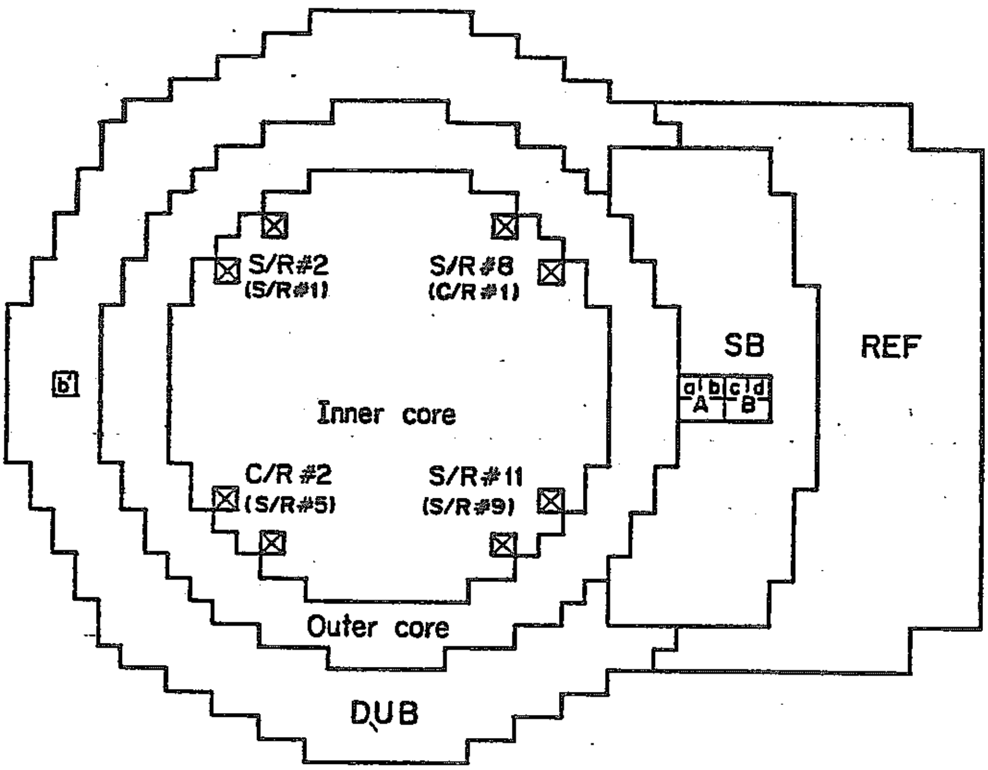


Fig. 1. Drawer Position for B_4 C Simulating Absorber Rod Worth Experiment

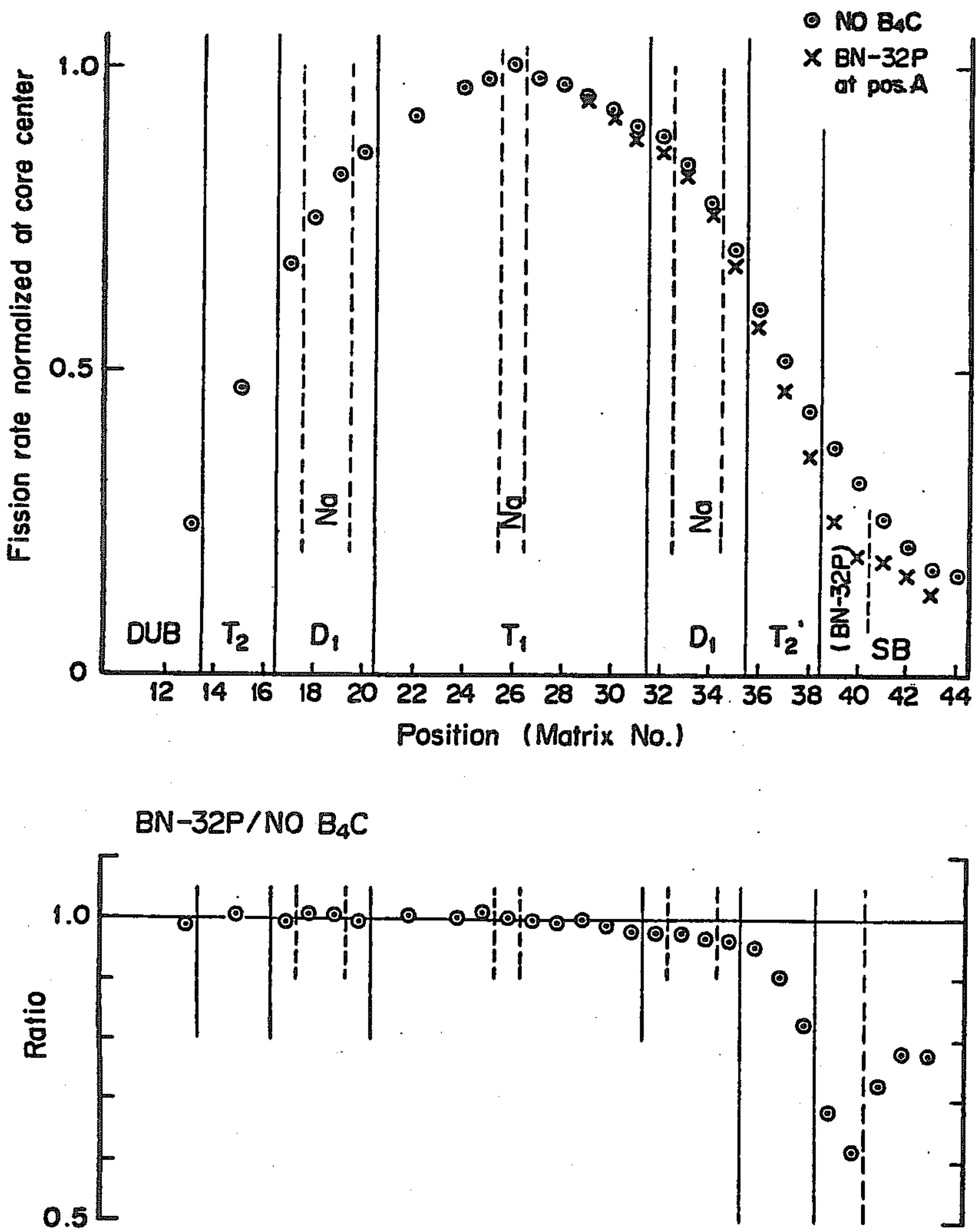


Fig. 2. Radial Fission rate Distribution of Pu-239 mfc in Assembly VII-2S

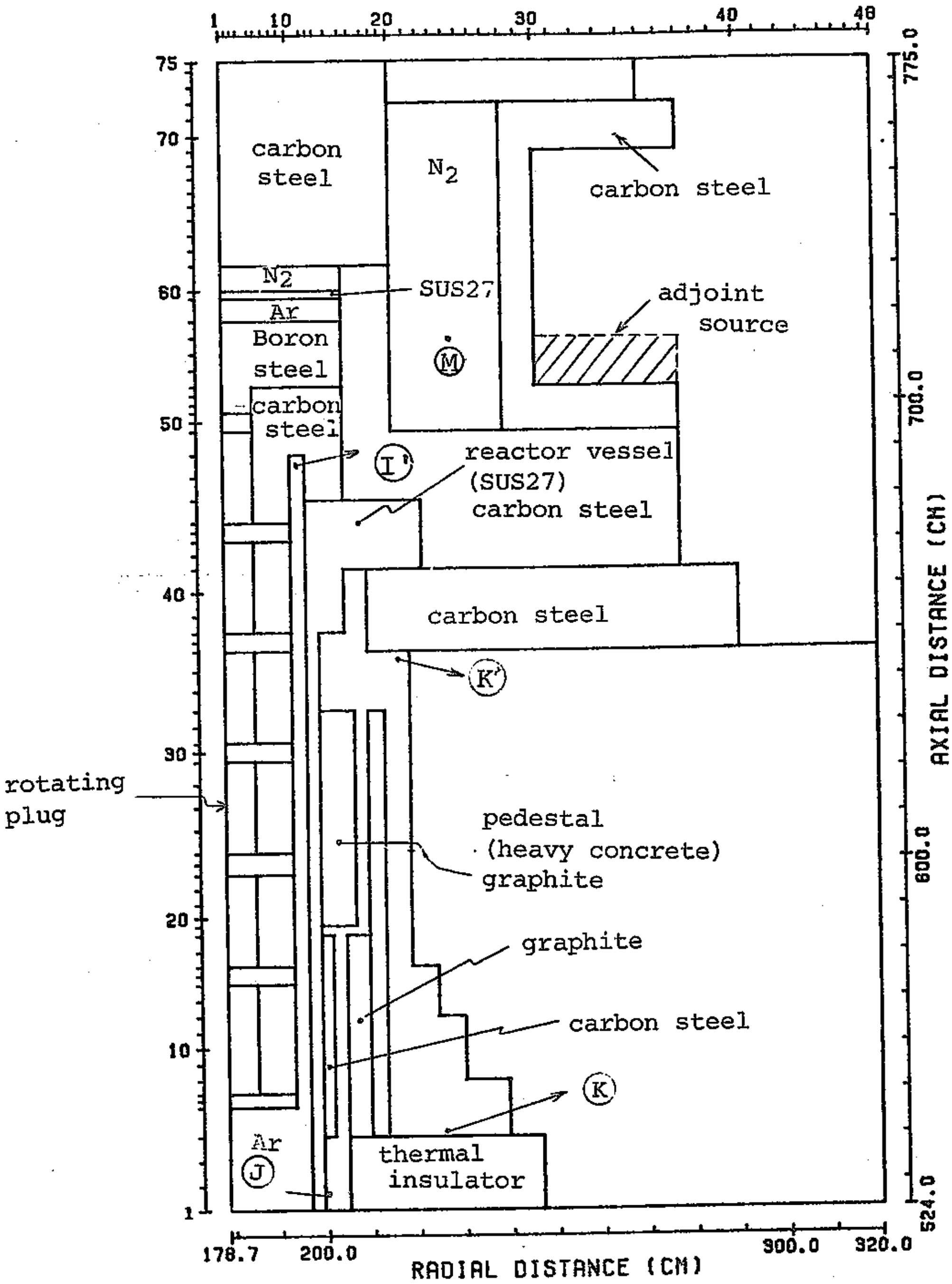
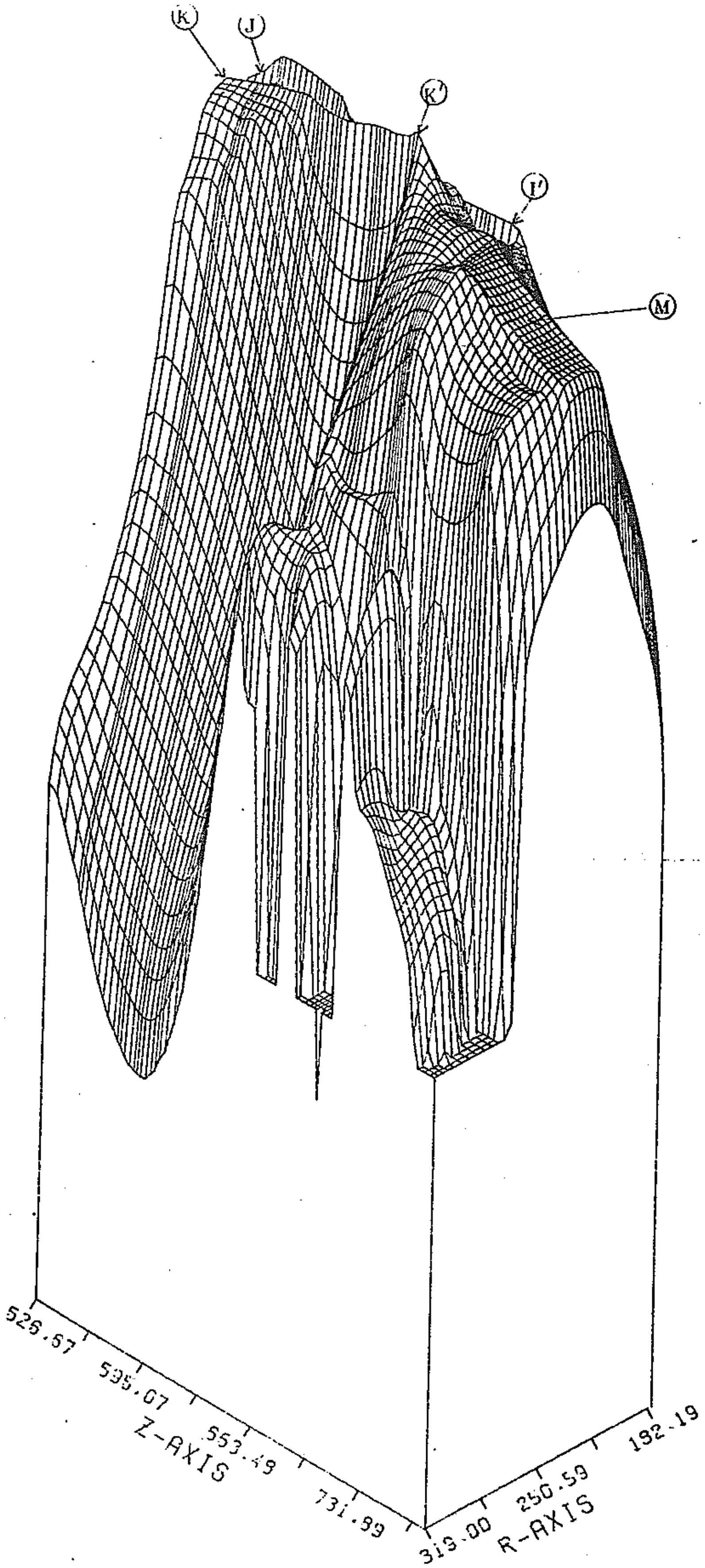


Fig. 3. Calculated System of Total Response Flux



JOYO MK1 (TOTAL)

Fig. 4. Total Response Flux