NEUTRONICS CHARACTERISTICS OF JRR-4 LOW ENRICHED URANIUM CORE

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ABSTRACT

JRR-4 (Japan Research Reactor No.4) had been operated using 93% enriched uranium (HEU) fuel since 1965. All of fresh HEU fuels had been consumed and stopped the reactor operation on January 1996. The reactor modification works started not only for fuel conversion but also for installation of a medical irradiation facility, repairing of the reactor building and so on. It has finished June 1998 and performance tests have begun. JRR-4 achieved the first criticality using 20% enriched uranium (LEU) fuel on July 14th, 1998. The minimum critical core consisted of 12 fuel elements, reflector elements and other reactor elements. During the tests, several reactor physics parameters have been measured. This paper presents the results of the tests. Methodology and results of neutronics design calculations are presented as well. The results are compared with the measured data.

INTRODUCTION

JRR-4 is a light water cooled and moderated swimming pool type research reactor with maximum thermal output of 3.5 MW. The core consists of fuel, reflector and neutron source elements, irradiation pipes and reactor control systems. JRR-4 has three kinds of control system. The fist one is shim control rod system (C1~C4) and the second one is fine control rod system (C5) and the last one is back up rod system (B1, B2). Fig.-1 shows the core arrangement of the JRR-4 after the modification work. The fuel elements arranged in a 4 by 5 array and they are surrounded by graphite reflector and other core elements. According to the Reduced Enrichment for Research and Test Reactors (RERTR) program, Japan Atomic Energy Research Institute (JAERI) has completed the JRR-4 modification work[1)] on June 1998. In the work, uranium enrichment of the fuel was reduced from 93% to 20%. To compensate the decrease of enrichment, uranium density of the fuel meat was increased by adopting uranium-silicon dispersion type fuel (silicide fuel). An irradiation pipe was replaced with a new one which can irradiate bigger size of samples. A medical irradiation facility was newly installed. To increase fast neutron flux goes to the facility, three graphite reflector elements closest to it were changed to aluminum reflector elements. In the modification work, not only the core modification but also repairing, renewal or installation of many reactor equipments including the reactor building have been done.

After the completion of the work, criticality approach experiment followed it immediately. JRR-4 achieved its first criticality with LEU fuel on July 14th, 1998. Loading of fuel elements was continued. JRR-4 has become the full core configuration which consists of 20 fuel elements on July 16th, 1998. Many characteristic tests have been performed on the core. The results were confirmed to meet the limitation permitted in the safety review by the regulatory body. The results of the design calculations were compared with the measured values to validate the calculation methodology.

EXPERIMENT

TEMPORARY NEUTRON DETECTION SYSTEMS

For the core characteristic tests, 4 temporary neutron detection systems were installed in the core. One was a FC detector and the others were BF3 detectors. 3 detectors were placed in the irradiation pipes and one detector was placed outside the core tank. 5 detectors including a permanent FC detector were used for the criticality approach test. Positions of the detectors are shown in Fig.-2. Signals from the detectors could be collected and processed in personal computers connected to the system.

CRITICALITY APPROACH

Number of fuel elements of minimum critical core was estimated as 12 by calculations. Then, the criticality approach experiment started from 6 fuel element core which was a half of minimum critical. At the fuel element positions without fuel, graphite reflector elements were installed. Fuel element loading was done by replacing the reflector element to the fuel element. Fig.-2 shows the order of fuel element loading. Inverse multiplication factor was measured and plotted on a paper at each fuel loading step. After adding 6 fuel elements, the core became critical as expected. The neutron source was removed from the core and it was confirmed that neutron flux kept a constant level for enough time with a shim rod and the fine rod were partially inserted. Another critical rod position was searched that all shim rods had the same position. Critical rod positions are listed in Table-1. Rod position of 650 mm is fully withdrawn and of 0 mm is fully inserted condition.

Fuel element loading was continued one by one until to be the full core configuration that consisted of 20 fuel elements. On the core, many characteristic tests have been done.

CONTROL ROD WORTH

Control rod worth was measured by the positive period method and the compensation method. The results are listed in Table-2. Control rod worth of the shim rod is about 4.1~4.2 %Δk/k. The value of the JRR-4 with HEU fuel was about 3.9 ~ 4.2 %Δk/k. Control rod worth of the shim rod does not change from that of the HEU core.

EXCESS REACTIVITY

Excess reactivity was evaluated from the critical rod position and the control rod worth. Summation of all negative reactivities inserted by partial insertion of control rods becomes the excess reactivity of the core. The measured value of the excess reactivity is 10.10 %Δk/k (corrected value for temperature of 300K). The excess reactivity of 12 %Δk/k is the maximum value permitted in the safety review by the regulatory body. It has been confirmed that the reactor meets the limitation.

ONE ROD STUCK MARGIN

One rod stuck margin is one of the most important parameter on the reactor safety. It was evaluated from the measured value of the excess reactivity and the control rod worth using the following equation.

$$r_s = r_T - r_{ex} - r_{\max} \tag{1}$$

where,

r_s : One rod stuck margin

r_T : Total control rod worth (= 16.98 %Δk/k)

r_{ex} : Control rod worth of the rod which has the largest worth (= 4.20 %Δk/k , C2)

$r_{\max}$: Excess reactivity (= 10.10 %Δk/k)

From the equation (1), the one rod stuck margin becomes 2.68 %Δk/k. The permitted minimum value of it is 1% Δk/k. The limitation is satisfied.

REACTIVITY COEFFICIENTS

The core temperature reactivity coefficient was measured. It is a synthesis of moderator temperature and fuel temperature reactivity coefficients. The moderator temperature was raised by the secondary coolant which was heated by the steam. Enough time was spent for raising it to make the fuel have the same temperature as that of the moderator. Change of reactivity is -0.302 %Δk/k according to the change of the temperature from 297.8 K to 313.9 K. The temperature reactivity coefficient becomes -1.875E-2 %Δk/k/K.

CALCULATION

COMPUTER CODES AND LIBRARY

The SRAC[2)] and the MVP[3)] codes were used for the design calculation of JRR-4 LEU core. The SRAC is an integrated code system which consists of neutron cross section libraries, a cell calculation module with burnup capability based on collision probability method, core calculation modules based on Sn transport or diffusion theory and an auxiliary code for core burnup calculation. The MVP is a vectorized continuous energy Monte-Carlo code. The cross section library for these codes are based on JENDL-3.2[4)] (Japanese Evaluated Nuclear Data Library ver. 3.2). Both of the codes have been developed by JAERI.

CALCULATION METHODOLOGY

SRAC CALCULATION

The SRAC calculation starts from a cell calculation of fuel plates. Fig.-3 shows the calculation model of the fuel plates. All of the fuel plates in a element are included in the model. Using the model, 107 energy group and 1 dimensional fixed source problem is solved with a cell calculation module named PIJ which is based on collision probability method. Homogenized cross sections of the fuel plate cell are obtained from the calculation. Next step is a fuel element cell calculation. Fig.-4 shows the fuel element cell model. The homogenized cross sections obtained in the previous step are used for the fuel plate regions of the model. A fixed source problem is solved in 2 dimensional and 107 group with PIJ. From this calculation, a homogenized fuel element cross section is obtained. Using the cross section, 107 energy group neutron spectrum is calculated by solving a B1 equation with a critical buckling. The homogenized cross section of 107 group is collapsed into 6 energy group using the spectrum as a weighting function.

The other core elements are divided into several vertical zones. Structure in a zone is smeared by atom number density and 107 group homogenized cross section of the zone is obtained. Two dimensional and 107 group core diffusion calculation is performed with the CITATION module stored in the SRAC. Fig.-5 shows the calculation model. From the calculation, space dependent neutron spectra of the core elements are calculated and 107 group cross sections of them are collapsed into 6 group using the spectra of them.

Few energy group (= 6) cross sections of every core elements have been obtained from the previous calculations. Three dimensional and few group whole core diffusion calculations are performed with the CITATION. From the calculations, neutronics parameters of the JRR-4 LEU core are obtained.

MVP CALCULATION

As the MVP is a continuous energy Monte-Carlo code, it can handle complicated structures with minimum geometrical approximations. In the JRR-4 calculation with the MVP, structure of all of the fuel plates in the core is treated with no approximation. The other parts of the core elements are divided into many zones adequately and homogenized within the zones by atom number density. History of 600,000 is selected to have one standard deviation error of about 0.1 %Δk/k.

COMPARISON OF MEASUREMENTS AND CALCULATIONS

MINIMUM CRITICAL CORE

Reactivity of minimum critical core was calculated with the MVP. The result is listed in Table-3. Calculation overestimates the measurement by about 1%. One of the reasons of the discrepancy is estimated as the neutron cross section library. It is known that calculations of uranium fueled cores with JENDL-3.2 overestimates a few point %Δk/k from many benchmark data analyses[5]. If the effect of JENDL-3.2 is removed, the calculated result may show good agreement with the measurement.

EXCESS REACTIVITY OF THE FULL CORE

Excess reactivity of the full core was calculated with the SRAC and the MVP. The results are listed in Table-4. Results of the both codes agree with the measurement very much but considering the tendency of the library, the SRAC calculations may underestimate the measurement slightly. Still, it can be said that the calculations agrees with the measurement well.

CONTROL ROD WORTH AND ONE ROD STUCK MARGIN

Control rod worth was calculated with the SRAC. The CITATION module which is based on diffusion theory was used for the calculation. Control rod was treated as a black absorber and a boundary condition was applied to the surface of it to have the same calculated worth with that of the MVP. In the calculations, all control rod were set at the same position that gives the effective multiplication factor (k_{eff}) of 1.0. From the position, a control rod was moved to the top or the bottom positions with the other control rods keep their original positions. At the two positions of the control rod, k_{eff}s were calculated and the worth of it was obtained. These calculations were performed for all control rods. Results are listed in Table-5. Results of C1~C5 overestimates the measurements. A control rod worth depends on positions of control rods in the core. It is estimated that one of the reason of the discrepancies between the calculations and the measurements is the difference of control rod positions of them. Though two control rods move in the measurement, only one control rod moves in the calculation. In the other way, calculated results of the back up rods agree with the measurements very much in all of three cases. Because the conditions of the control rods were very much similar between the measurements and the calculations. The moved distances of the control rods were short in the measurements and no control rod was moved in the calculations. Table-6 shows the critical position of the control rods of the calculation and of the measurement. Both positions agree very much. From the table, it is shown that the control rod worth is calculated correctly.

The one rod stuck margin is also listed in the Table-5. The calculation agrees with the measurement well.

REACTIVITY COEFFICIENTS

Moderator void fraction, moderator temperature and fuel temperature reactivity coefficients were calculated with the SRAC. Homogenized cross sections of the fuel element with several points of moderator void fraction, moderator temperature and fuel meat temperature were prepared and used for whole core calculations. From the change of effective multiplication factors and the change of parameters, reactivity coefficients were obtained. Number of few energy group was 8 for the coefficients calculations to keep a enough accuracy. The results are listed in Tables-7, 8 and 9. All calculated coefficients are negative within the calculated parameter ranges.

To compare with the measurement, the core temperature reactivity coefficient was calculated from the moderator and fuel temperature reactivity coefficients. They were fitted to first order functions of temperature by the least mean square method. The two functions were summed and averaged over the temperature range of the measurement. The result is listed in the Table-10. Agreement of the measurement and the SRAC is not bad.

CONCLUSIONS

Neutronics characteristics tests of the JRR-4 LEU core have been performed. All of the results meet the limitations permitted by the regulatory body. The results of the design calculations are compared with the measured values. Design value of excess reactivity shows good agreement with the measurement even if the tendency of JENDL-3.2, to give higher reactivity, is considered. Good agreements are also shown between the design and measured values of the criticality and of the one rod stuck margin. The core temperature coefficient of the calculation overestimates the measurements but it can be said the measurement is estimated well.

From the comparison, it can be said the design calculation estimates the core neutronics characteristics well.

Some tests have not been performed or not been processed yet. They will be completed soon and results will be compared with design values.

ACKNOWLEDGMENTS

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Table-1 Critical Rod Position of the Minimum Critical Core (mm)

C-Rod Pattern	C1	C2	C3	C4	C5
Critical-1	650	650	650	369	292
Critical-2	525	525	525	525	302

Rod position: 650 mm = Full Out, 0 mm = Full In

Table-2 Control Rod Worth of the Full Core (%Δk/k)

C1	C2	C3	C4	C5
4.09	4.14	4.10	4.20	0.46

Table-3 Criticality of the Minimum Critical Core* (k_{eff})

C-Rod Pattern	Measurement**	MVP	MVP / Measurement
Critical-1	0.99814	1.01071e+00 (0.1342%)	1.013

* Neutron source element is removed.

** After temperature correction

Table-4 Excess Reactivity of the Full Core (%Δk/k)

Measurement*	SRAC	MVP
10.10	10.16	10.58

* After temperature correction

Table-5 Control Rod Worth of the Full Core (%Δk/k)

	Measurement	SRAC
C1	4.09	4.80
C2	4.14	4.77
C3	4.10	4.71
C4	4.20	4.81
C5	0.46	0.55
B1	0.63	0.63
B2	0.94	0.86

B1+B2	1.33	1.32
One Rod Stuck Margin	2.68	2.58

Table-6 Critical Control Rod Position of the Full Core (mm)

	C1	C2	C3	C4	C5
Measurement	255	255	255	255	292
SRAC	258	258	258	258	258

Table-7 Moderator Void Reactivity Coefficients (%Δk/k/%void)

Void Fraction (%)	Reactivity Coefficient
0 => 5	-3.260E-1
5 => 10	-3.696E-1
10 => 25	-4.793E-1
25 => 50	-8.424E-1

Table-8 Moderator Temperature Reactivity Coefficients (%Δk/k/K)

Moderator Temperature (K)	Reactivity Coefficient
300 => 323	-2.696E-2
323 => 348	-3.236E-2
348 => 373	-3.756E-2
373 => 423	-4.570E-2

Table-9 Fuel Temperature Reactivity Coefficients (%Δk/k/K)

Fuel Temperature (K)	Reactivity Coefficient
300 => 373	-1.852E-3
373 => 473	-1.617E-3
473 => 573	-1.440E-3
573 => 673	-1.300E-3

Table-10 Core Temperature Reactivity Coefficients

	Measurement	SRAC
Temperature Change (K)	297.8 => 313.9	297.8 => 313.9
Reactivity Change (%Δk/k)	-0.302	-0.446
Temperature Reactivity Coefficient (%Δk/k/K)	-1.875E-2	-2.770E-2

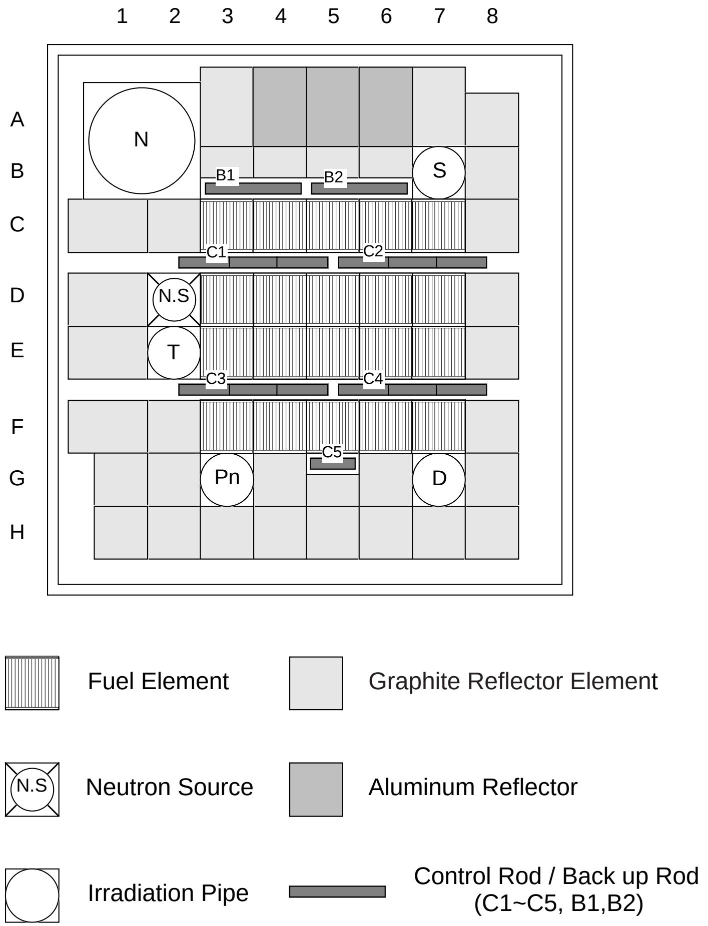


Figure-1 Core Configuration after the Modefication Work

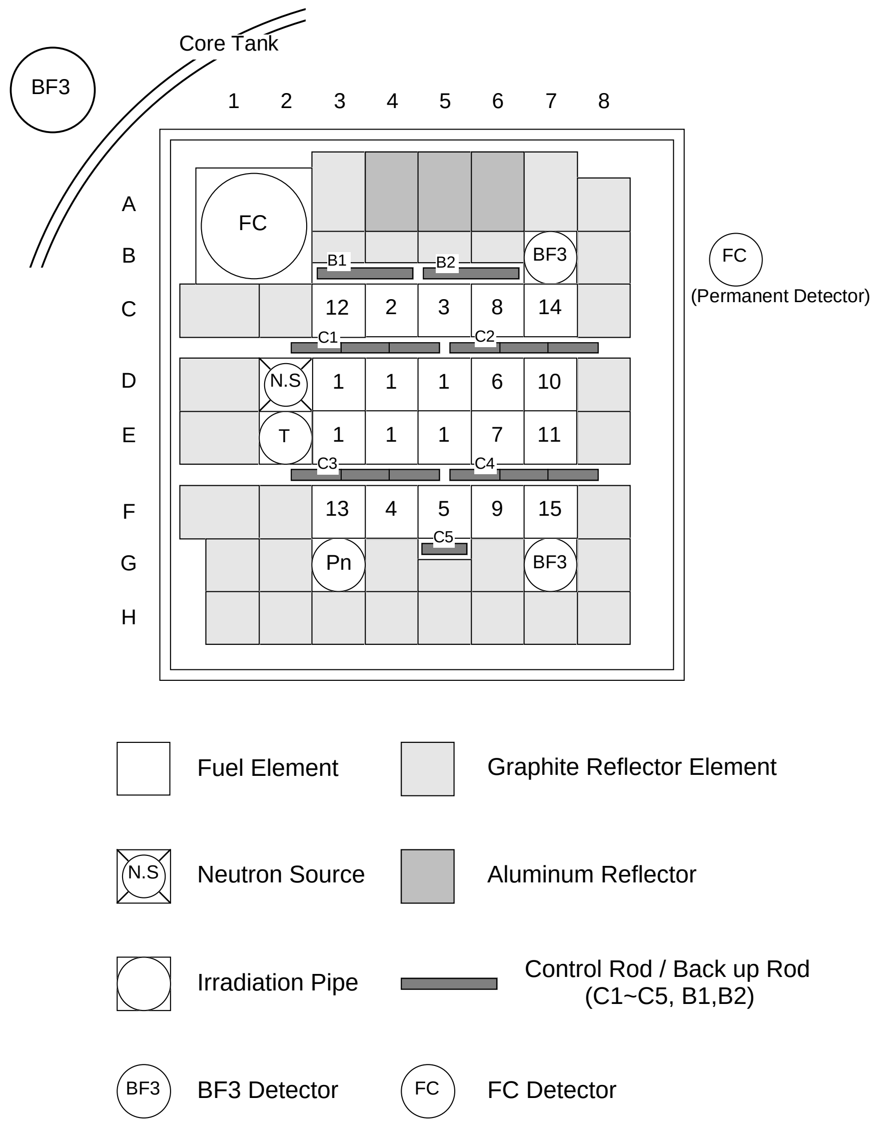


Figure-2 Order of the Fuel Element Loading

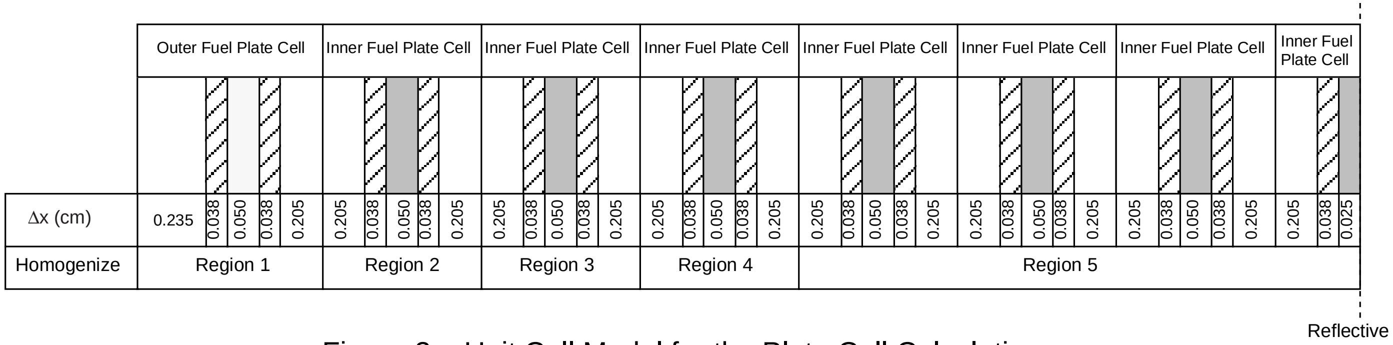
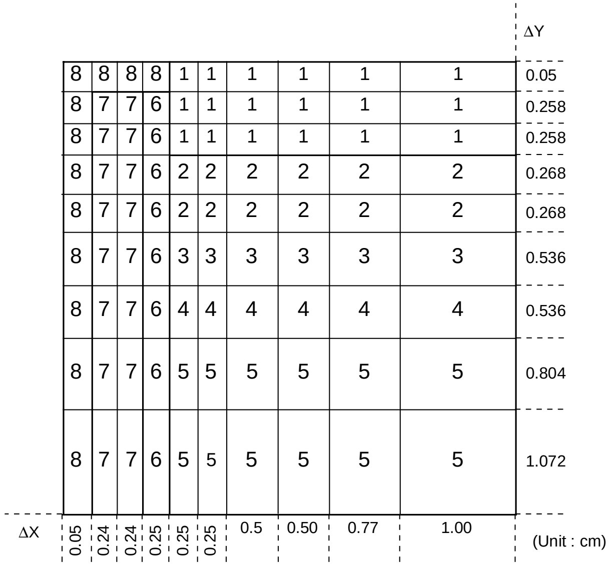


Figure-3 Unit Cell Model for the Plate Cell Calculation



Region	Mixture
5	Inner Fuel Plate Cell - 1
4	Inner Fuel Plate Cell - 2
3	Inner Fuel Plate Cell - 3
2	Inner Fuel Plate Cell - 4
1	Outer Fuel Plate Cell
6	Al+H_2O
7	Side Plate
8	H_2O

Figure-4 Unit Cell Model for the Element Cell Calculation

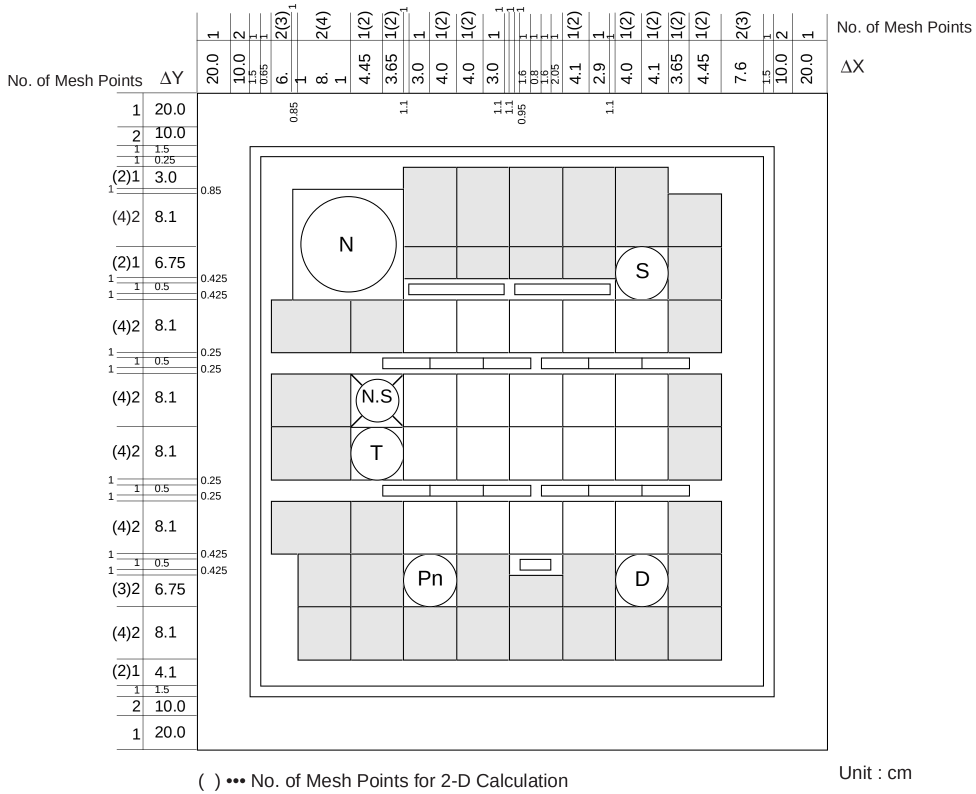


Figure-5 Horizontal Model for the Whole Core Calculation