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Neutronics Design of Upgraded JRR-3 Research Reactor

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ABSTRACT

The research reactor JRR-3 is under upgrading to enhance the usefulness of the reactor by increasing the power from 10 to 20 MW. The new reactor is a pool type fueled with a low-enriched uranium fuel.

The neutronics calculations were carried out on the reactor using 20% enriched U*Alx-Al plate-type fuel with 2.2 g U/cm^3. The results show that the core performances, such as reactivity, neutron flux, and burnup, are sufficient for beam experiments, material testing, and isotope production.

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1. INTRODUCTION

The Japan Research Reactor-3 (JRR-3) is a 10 MW tank-type facility located at the Japan Atomic Energy Research Institute, Tokai Research Establishment. After 21-year operation, the reactor was closed in 1983 and was subjected to upgrading in order to satisfy the changing needs of experimental programs. This upgrading involves a replacement of the core with a completely new one, together with an increase in power level from 10 to 20 MW. The upgraded reactor is scheduled to resume regular operations in 1989.

The design concept of the new reactor is that it has wide adaptability for utilization and that it provides sufficient neutron flux for beam experiments, material testing, and isotope production. To fulfill the requirements, a pool-type reactor with a heavy-water reflector surrounding the core is adopted. The design target of thermal neutron flux is 2.0E+14 n/(cm^2*s) in the heavy-water reflector.

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As an important result of the international reduced enrichment program, low-enriched and high-density uranium fuel (LEU fuel) became available recently for research reactors. This paper deals with the essential results of the neutronics design of the upgraded JRR-3 which uses the LEU fuel.[1]

2. CORE CONFIGURATION

A description of the reactor design is given briefly in Table 1. The reactor core is placed 5.5 m below the surface of the reactor pool. The core consists of 26 standard fuel elements, 6 control rods, and 5 irradiation elements. As illustrated in Fig. 1, they are arranged cylindrically in a square lattice of 7.72 cm pitch and surrounded with a heavy-water reflector.

The fuel elements are MTR plate type using 20% enriched U*Alx-Al dispersion fuel with 2.2 g U/cm^3, as shown in Fig. 2. The fuel meat thickness and coolant channel thickness in the fuel elements, as well as the enrichment and density of uranium, were determined after carrying out a feasibility study on the neutronics and thermal-hydraulics performances of the reactor.[2] The standard fuel element has a 7.62 cm square cross section and its active length is 75.0 cm. A control rod assembly consists of two major sections; the lower section of a control fuel element and the upper section of a hafnium box neutron absorber. The control rod is driven through the inside of a 0.5 cm-thick square aluminum guide tube. The control fuel element has the same active length as the standard fuel element, while the cross section is 6.36 cm square. The irradiation element is an aluminum block with a hollow where irradiation samples are located.

Heavy water is contained in a concentric cylindrical aluminum reflector tank. The thickness of the heavy-water reflector is 67 cm and the height is 160 cm. The space between the core components and the reflector tank is filled with beryllium blocks. Vertical irradiation holes are provided for irradiation services at 4 locations in the beryllium reflector and 8 in the heavy-water reflector. Nine horizontal beam tubes for physics experiments are extended into the heavy-water reflector. For calculational purposes, however, those irradiation holes and beam tubes in the heavy-water reflector were assumed to be filled with heavy water.

3. CALCULATION METHODS

Neutronics performances of the reactor, such as neutron multiplication factors, neutron flux distributions, and power density distributions, were calculated with a neutronics design code system SRAC.[3] The calculation methods used in this design are essentially the same as those which have been described in

Reference 4. A brief description of the methods is given below.

Neutron spectra and macroscopic cross sections were calculated for the fuel elements and the control rod absorber by using a collision probability method and for the reflectors by using a Sn transport method. Separate unit cells were employed for the fuel elements and the control rod absorber, respectively. Those calculations were carried out with a 37-fast group and a 35-thermal group microscopic cross-section libraries which were compiled mostly from ENDF/B-4 data. Burnup-dependent macroscopic cross sections were calculated only for nuclides in the fuel plate.

Three-group macroscopic cross sections were generated as a function of burnup to perform burnup and criticality calculations with a 2- or 3-dimensional diffusion model. The upper energy boundaries in the 3-group scheme are 10 MeV for the fast-energy group, 5.53 keV for the epi-thermal group, and 0.683 eV for the thermal group.

The macroscopic cross sections of the fuel elements were calculated by following the two successive calculation steps to take into account the heterogeneity of the fuel element cells. In the first step, a 1-dimensional plate cell which was consisted of fuel meat, clad, and moderator regions, was employed to calculate the homogenized cross section of the plate cell. Then, the homogenized cross section was used in the next step of a 2-dimensional calculation on a square fuel element cell.

The changes of nuclide densities due to fuel burnup were calculated after a neutron spectrum calculation of a fuel plate cell. This process is called a cell burnup calculation. Analytical solutions of the transmutation equations for nuclides were used to obtain the changes of the nuclide concentrations by employing a chain scheme of a Garrison model. In a process of a core burnup calculation, the cross sections of the fuel elements were obtained by looking up the tables of burnup dependent macroscopic cross sections. They were prepared for the possible ranges of burnup of the standard and control fuel elements, respectively, at the cell burnup calculation.

As the hafnium control rods have the very strong absorbing quality, their worths were evaluated using a diffusion theory model with an appropriate internal boundary condition for the thermal group and with macroscopic cross sections for fast and epi-thermal groups. To obtain the internal boundary condition, both a Sn transport calculation and a diffusion calculation were carried out on the same 2-dimensional reactor geometry in which a control rod was located at the central position. The internal boundary condition was applied only to the thermal group of the diffusion calculation and the value which gave the same reactivity worth as the transport calculation was used for the subsequent control-rod calculation.

4. NEUTRONICS PERFORMANCES

The neutronics parameters of the core were calculated as a function of burnup. As a calculation model of core loading, 28-days cycle length of operation and 5-batch scatter loading were employed. That is, the positions of the fuel elements are not changed during 140 days of operation (5 cycles of reactor operation). At the end of an operation cycle, the reactor is shut down and 5 or 6 fuel elements which irradiated for 5 cycles, are then removed. New fuel elements are loaded in the positions left vacant. The reactor is then taken to power again to run through another 4-week operation after one-week shutdown interval.

The essential parameters for the fresh fuel core, the beginning of equilibrium cycle (BOC) core, and the end of equilibrium cycle (EOC) core are listed in Table 2. The reactivity worths of the core and the control rods were calculated with a 2- or 3-dimensional diffusion model on the fresh fuel core and the typical equilibrium cores of 18% and 26% uniform burnups. The neutron flux, power, and burnup were evaluated by a 3-dimensional diffusion model at every cycle ranging from the fresh fuel core to the 10th EOC core in a 5-batch scatter loading pattern.

4.1 Reactivity requirements for operation

The core size and the initial atom concentrations have been chosen to allow for reactivity losses caused by fuel burnup, stable fission product buildup, equilibrium xenon and samarium poisoning, the cold-to-hot reactor reactivity change, and some irradiation sample margins in the experimental facilities. No allowance is made separately for transient xenon override.

The typical EOC core with equilibrium xenon and samarium concentrations at room temperature has an excess reactivity of 4.9%dk/k when all rods are fully withdrawn. This excess reactivity supplies the temperature defect, permits the addition of some absorbers in the experimental holes, and allows a minimum of about 0.5% for control purposes.

The reactivity requirement at each burnup step is listed in Table 3. In order that the reactivity requirement should be 8.8%dk/k at the point just before reloading is needed (at EOC), the initially loaded core (all fresh fuel elements) must have excess reactivity of 16.3%dk/k including the irradiation sample allowance.

4.2 Control rod worths

The total reactivity worth of all 6 control rods was calculated to be 31%dk/k. The least shutdown margin exists with the fresh fuel core. At the time, the shutdown margin of the cold xenon-free core with all rods inserted is 14%dk/k. As the maximum worth of a single rod is 10%dk/k, the shutdown margin with all rods inserted except only one rod withdrawn (one stuck rod situation) is 4%dk/k. Thus, the reactor sufficiently meets the one stuck rod criterion, even for the fresh fuel core. For the equilibrium cores, the shutdown margin is greater than 11%dk/k.

4.3 Flux and power distributions

Figures 3 through 5 show the calculated neutron fluxes in a BOC core. The thermal neutron flux has a peak in the heavy-water reflector with the maximum value of 2.7E+14 n/(cm^2*s) at an EOC core occuring about 7 cm from the core-reflector interface. Although not modeled in the calculation, the vertical irradiation holes and the beam tubes are to be located in the heavy-water reflector. Fluxes shown in those figures can be expected to be smaller when leakage through the beam tubes and the irradiation holes is taken into account.

The calculated profiles of power densities at a BOC core are shown in Figs. 6 and 7. The power distribution changes as the core burns up. The largest power peak occurs in a fuel element adjacent to the boundary between the core and the beryllium reflector, rather than in the core interior. The peaking factors were evaluated for the fresh fuel, BOC, and EOC cores, where the peaking factor is defined as the ratio of the power density at the hot spot to the average power density in the core. The maximum peaking factor is 2.63 at a BOC core, where radial, axial, and local components are 1.23, 1.42, and 1.51, respectively.

4.4 Burnup distributions

The burnup is defined as the fraction of original ^{235}U atoms that are lost by fission or capture. At the design power of 20 MW and 140 full power day loading, the average burnup of the standard fuel elements just prior to their removal is 40%. A horizontal distribution of average burnups of the fuel elements and vertical burnup distributions of typical several fuel elements in an EOC core are shown in Figs. 8 and 9, respectively. In those burnup distributions, the maximum burnup is 54% in the fuel element, 6-2, of which average is 40%.

4.5 Reactivity coefficients and kinetic parameters

The temperature coefficient of the moderator was evaluated for the combined effects of temperature and density changes in the light water as it heats. This reactivity change is due to the hardening of the thermal neutron spectrum resulting from an increase in the water temperature and a reduction in the water density. The void coefficient of the moderator was also calculated for the density changes as a function of void fraction in water. The temperature coefficient of the fuel was determined by a resonance calculation as a function of temperature. The absorption of the ^{238}U epi-thermal resonances increases as the temperature increases in the fuel meat. Thus, the temperature and void coefficients are always negative over the possible ranges of temperature and void fraction.

The prompt-neutron lifetime and the effective delayed-neutron fraction were evaluated by a perturbation theory, where required normal and adjoint neutron fluxes were calculated with a 3-dimensional diffusion model. All calculations were carried out using 4-group cross sections of which upper energy boundaries were 10 MeV, 0.82 MeV, 5.5 keV, and 0.68 eV. Those kinetics parameters are not so sensitive to the burnup of the equilibrium range.

5. CONCLUSION

Neutronics calculations have been carried out on the upgraded JRR-3 research reactor. The new reactor performs reasonably well using low-enriched and high-density U*Alx-Al fuel. The reactor performances, such as reactivity, neutron flux, and burn-up, are sufficient for beam experiments, material testing, and isotope production.

REFERENCES

1. H. Tsuruta, H. Ichikawa, J. Iwasaki, "Neutronics Design of Upgraded JRR-3 Research Reactor," (In Japanese), JAERI-M 84-099(1984).

2. H. Ichikawa, et al., "Neutronic and Thermo-Hydraulic Design of JRR-3(M) Reactor," in Proceedings of the International Meeting on Reduced Enrichment for Research and Test Reactors, Tokai-mura, Japan, October 24-27, 1983, JAERI-M 84-073(1984).

3. K. Tsuchihashi, et al., "SRAC: JAERI Thermal Reactor Standard Code System for Reactor Design and Analysis," JAERI-1285(1983).

4. K. Arigane, K. Tsuchihashi, "Analysis of Critical Experiments of FNR LEU Cores", ibid.

Table 1. Reactor Design Description

Reactor type	Pool type
Power level (MW)	20
Fuel	
Type	MTR type, flat plate
Meat material	U*Alx-Al
Uranium enrichment	19.75%
^{235}U content by element	300 g (Standard fuel element) 190 g (Control fuel element)
Horizontal cross section (cm^2)	7.62x7.62 (Standard fuel element) 6.36x6.36 (Control fuel element)
Active length (cm)	75.0
Number of fuel elements	26 (Standard fuel element) 6 (Control fuel element)
Lattice pitch (cm^2)	7.72x7.72
Moderator, Coolant	H_2O
Reflectors	D_2O, Be
Control rod absorber	Hf
Irradiation positions	5 (incore) 4 (Be reflector) 8 (D_2O reflector)
Number of beam tubes	9 (D_2O reflector)
Coolant temperature (^oC)	35 (Inlet) 44 (Outlet)

Table 2. Neutronics Parameters

(At 20 ^{o}C, 0% void)

	Fresh	BOC	EOC
Excess reactivity, All rods withdrawn (%dk/k)	16	11	9
Shutdown margin, all rods (%dk/k)	14	22	26
Shutdown margin, one rod stuck (%dk/k)	4	11	15
Total reactivity effect, all rods (%dk/k)	31	33	35
Total reactivity effect, one rod stuck (%dk/k)	21	22	24
Moderator temperature coefficient (%dk/k/^{o}C)	-1.8E-2	-1.9E-2	-1.8E-2
Fuel temperature coefficient (%dk/k/^{o}C)	-2.5E-3	-2.5E-3	-2.5E-3
Moderator void coefficient (%dk/k/% void)	-0.42	-0.31	-0.27
Prompt-neutron lifetime (s)	-1.7E-4	-1.8E-4	-1.8E-4
Effective delayed neutron fraction	7.6E-3	7.3E-3	7.2E-3

Table 3. Reactivity Requirements

(%dk/k at 20 ^{o}C, 0% void)

	Fresh	BOC	EOC
Fuel burnup	7.3	2.4	0.0
Xenon, samarium, and stable fission products	3.9	3.9	3.9
Irradiation sample allowance	3.7	3.7	3.7
Temperature defect and control margin	1.4	1.3	1.2
Total	16.3	11.3	8.8

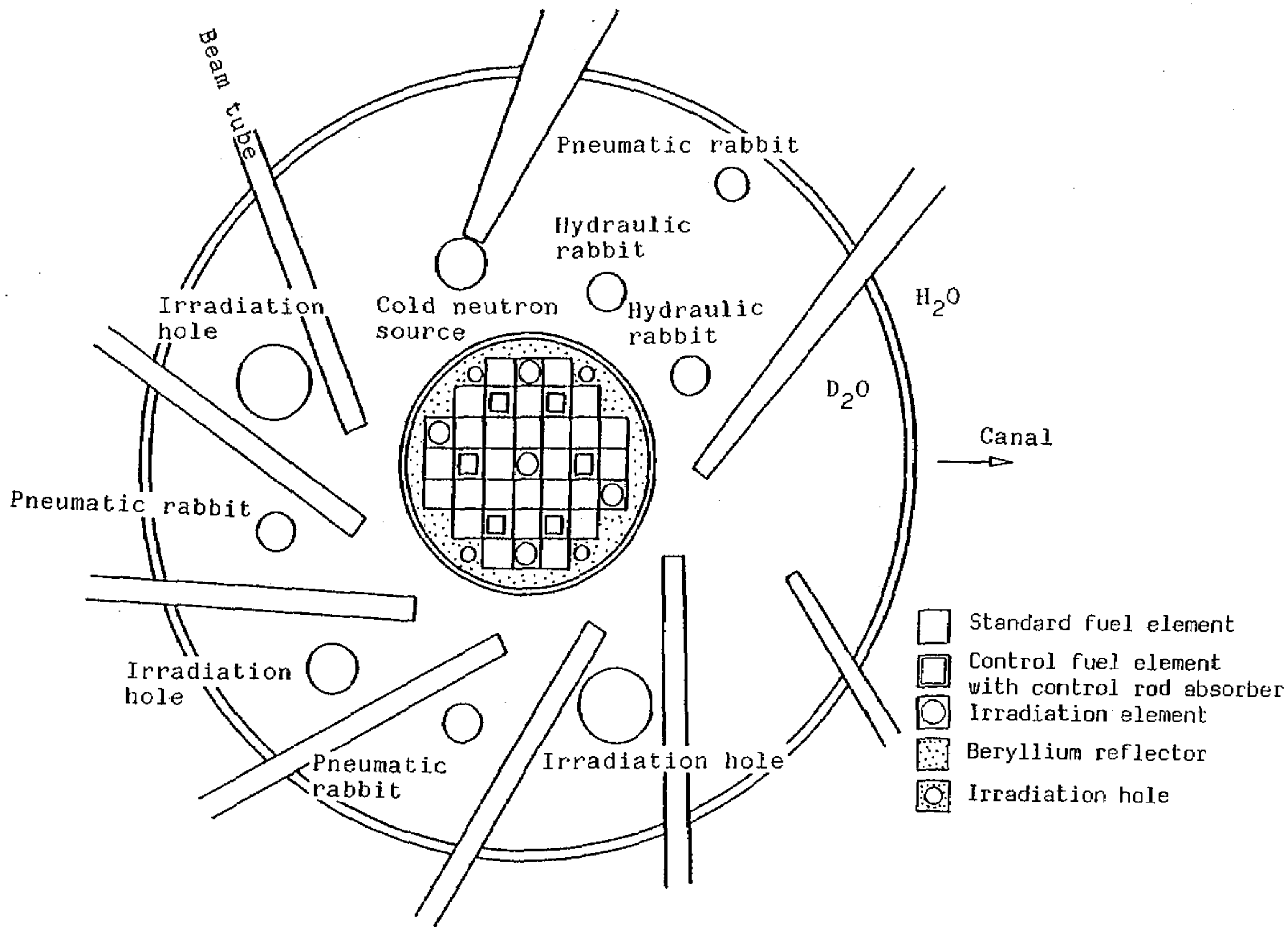
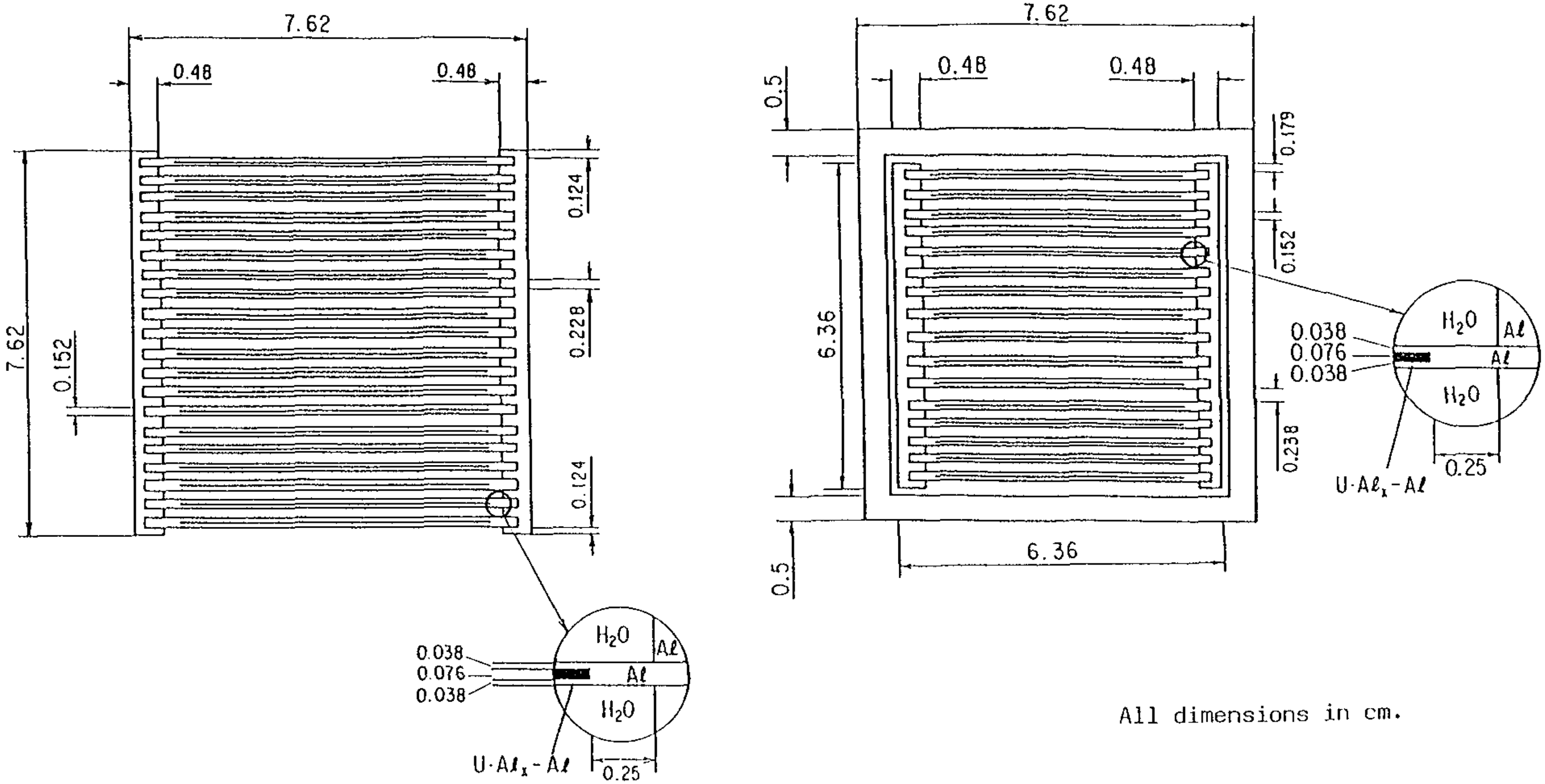


Fig. 1. Horizontal Cross-Sectional View of the Reactor.



(a) Standard fuel element

(b) Control fuel element

Fig. 2. Horizontal Cross-Section of Fuel Elements.

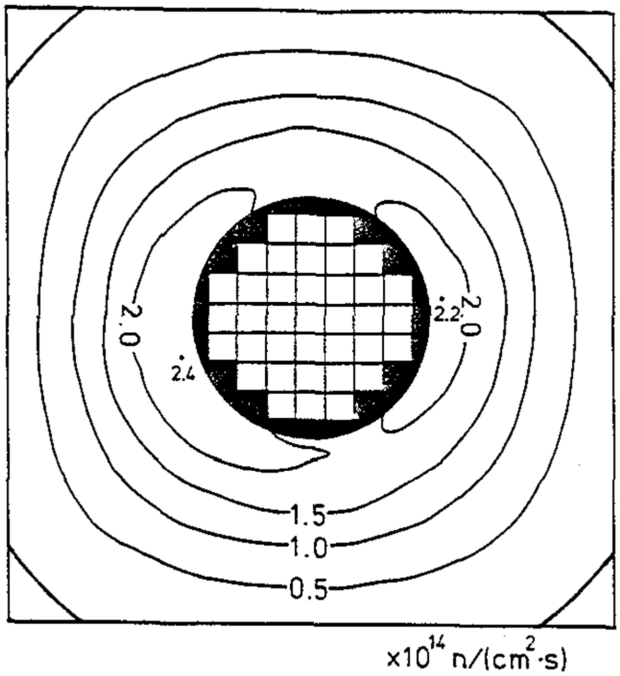


Fig. 3. Contour Map of Thermal Neutron Flux on an Axial Midplane in the D_2O Reflector of a BOC Core.

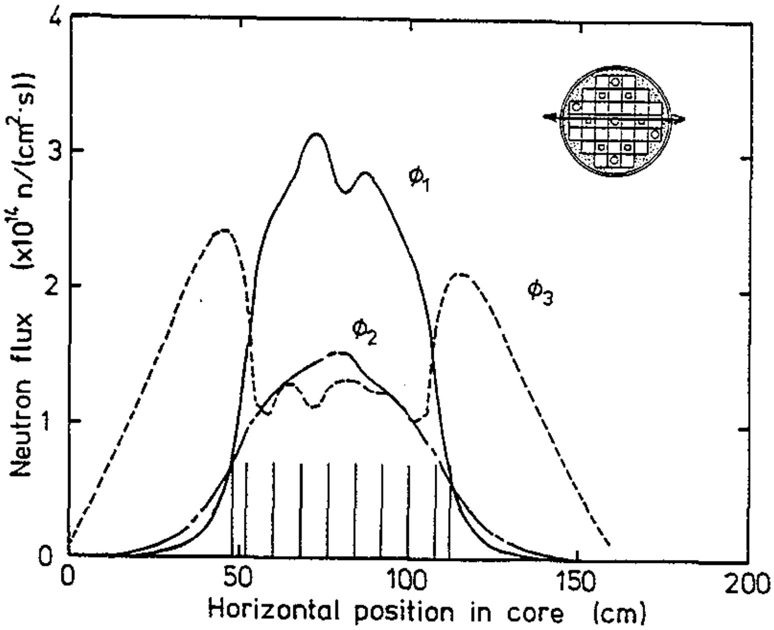


Fig. 4. Horizontal Distributions of Fast ($\emptyset_1$), Epi-Thermal ($\emptyset_2$), and Thermal ($\emptyset_3$) Neutron Fluxes in a BOC Core.

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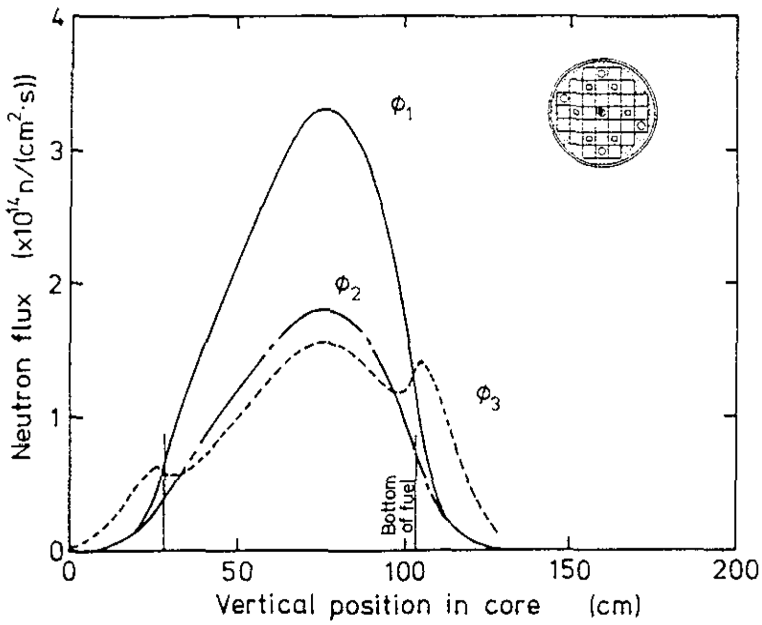


Fig. 5. Vertical Distributions of Fast (ϕ_1), Epi-thermal (ϕ_2), and Thermal (ϕ_3) Neutron Fluxes in a BOC Core.

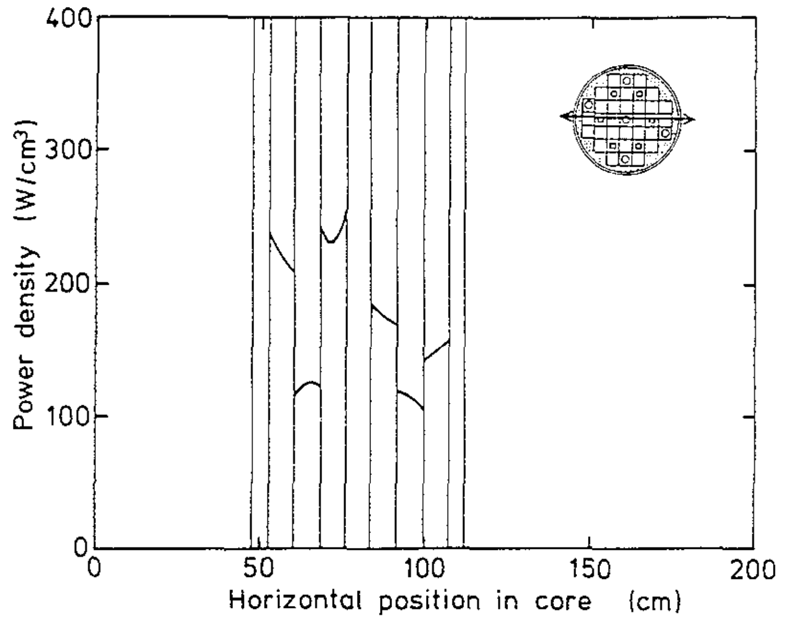


Fig. 6. Horizontal Distribution of Power Density in a BOC Core.

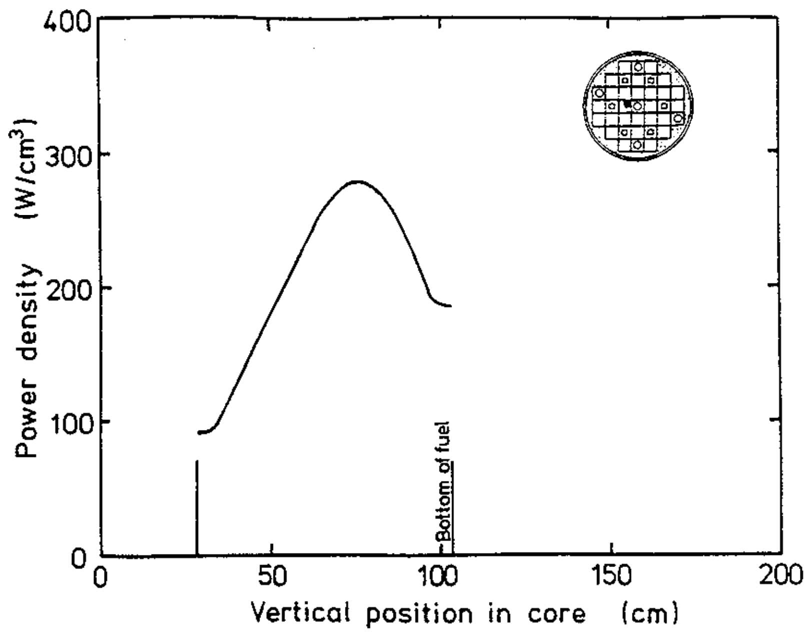
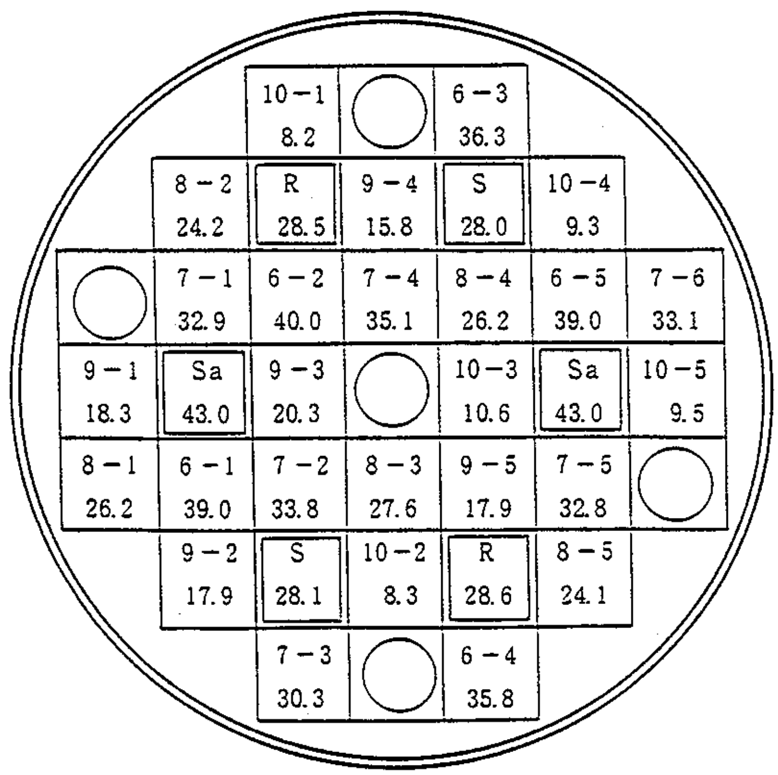


Fig. 7. Vertical Distribution of Power Density in a BOC Core.



Upper line: The number of fuel element
Lower line: Average burnup of fuel element (%)
Sa, S, and R: Control fuel elements

Fig. 8. Typical Distribution of Average Burnups of Fuel Elements in a EOC Core.

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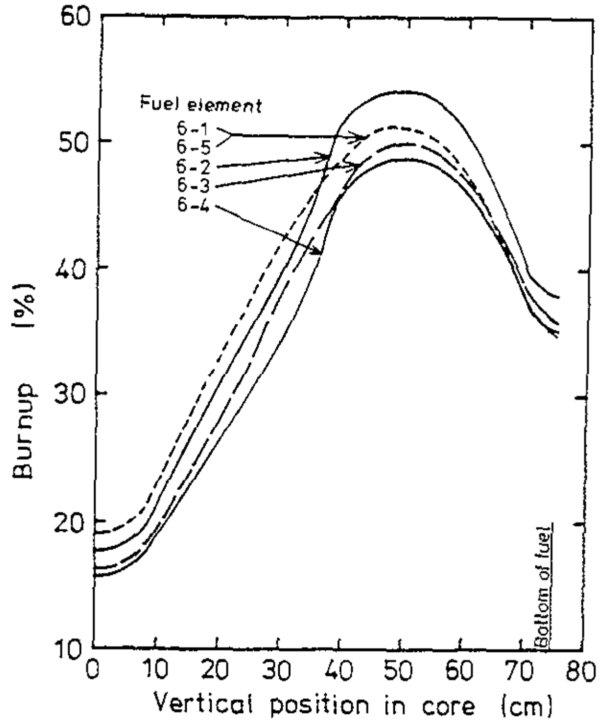

Fig. 9. Vertical Burnup Distributions of the Fuel Elements Loaded in the Core during 5 Cycles.
The locations of the fuel elements in the core are shown in Fig. 8.