JAERI - M
88-078

NEUTRONIC AND THERMO-HYDRAULIC DESIGN OF LEU CORE FOR JAPAN RESEARCH REACTOR 4

April 1988

Kenji ARIGANE, Shukichi WATANABE and Harumichi TSURUTA

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

編集兼発行　日本原子力研究所
印　　刷　㈱原子力資料サービス

Neutronic and Thermo-Hydraulic Design of LEU Core for Japan Research Reactor 4

Kenji ARIGANE, Shukichi WATANABE and Harumichi TSURUTA

Department of Research Reactor Operation
Tokai Research Establishment
Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken

(Received March 18, 1988)

As a part of the Reduced Enrichment Research and Test Reactor (RERTR) program in JAERI, the enrichment reduction for Japan Research Reactor 4 (JRR-4) is in progress. A fuel element using a 19.75% enriched UAlx-Al dispersion type with a uranium density of 2.2 g/cm^3 was designed as the LEU fuel and the neutronic and thermo-hydraulic performances of the LEU core were compared with those of the current HEU core. The results of the neutronic design are as follows: (1) the excess reactivity of the LEU core becomes about 1% $\Delta k/k$ less, (2) the thermal neutron flux in the fuel region decreases about 25% on the average, (3) the thermal neutron fluxes in the irradiation pipes are almost the same and (4) the core burnup lifetime becomes about 20% longer. The thermo-hydraulic design also shows that: (1) the fuel plate surface temperature decreases about 10 °C due to the increase of the number of fuel plates and (2) the temperature margin with respect to the ONB temperature increases.

Therefore, it is confirmed that the same utilization performance as the HEU core is attainable with the LEU core.

Keywords: JRR-4, RERTR, LEU Core, Neutronic, Thermo-Hydraulic, Design, SRAC, COOLOD, Reactivity, Control Rod Worth, Neutron Flux, Reactivity Coefficient, Kinetic Parameters, ONB Temperature, Fuel Temperature

ＪＲＲ－４低濃縮ウラン炉心の核・熱水力設計

日本原子力研究所東海研究所研究炉管理部
有金賢次・渡辺終吉・鶴田晴通

（1988年３月18日受理）

ＪＲＲ－４燃料の低濃縮化が原研の研究炉・試験炉燃料濃縮度低減化（ＲＥＲＴＲ）計画に基づいて進められている。燃料として，濃縮度19.75％，ウラン密度2.2 g／cm^3のＵＡℓx－Ａℓ分散型燃料を用いた低濃縮ウラン（ＬＥＵ）燃料要素が設計され，高濃縮ウラン（ＨＥＵ）炉心の核・熱水力特性と比較された。その結果，ＬＥＵ炉心の過剰反応度は約１％Δk／k，燃料領域の熱中性子束は約25％低くなるものの，照射筒における熱中性子束の低下はほとんどなく，炉心燃焼寿命も約20％増加することが明らかになった。また，燃料板が増加されたため，燃料板表面温度が約10℃低下し，熱的安全余裕が増加した。

以上により，ＬＥＵ炉心においても，ＨＥＵ炉心と同様の照射利用が可能であることが確認された。

東海研究所：〒319－11　茨城県那珂郡東海村白方字白根２－４

Contents

目　　次

1. INTRODUCTION

Japan Research Reactor 4 (JRR-4) is a 3.5 MW swimming pool type research reactor and uses 93% enriched UAl-alloy MTR-type fuel elements. The reactor has been utilized generally for shielding experiments, RI production, activation analyses, reactor engineering experiments, and training for reactor engineers.

The reduced enrichment program for JRR-4 was started in 1979 to convert the fuel from High-Enrichment Uranium (HEU) to Low-Enrichment Uranium (LEU). Some neutronic analyses and thermo-hydraulic experiments were carried out[1] and a UAlx-Al MTR-type fuel element of uranium density of 2.2 g/cm^3 was chosen as a candidate of the LEU fuel.

In order to confirm the neutronic performance and irradiation behavior of the LEU fuel element, two LEU fuel elements were fabricated. One fuel element was served for the neutronic performance measurement in JRR-4 and the other for the accelerated irradiation test in Japan Research Reactor 2 (JRR-2). The results indicate that the LEU fuel element has almost the same reactivity worth as the HEU fuel element as shown in Table 1[2] and irradiation behavior is satisfiable.[3]

Extensive design calculations have been further carried out on the neutronic and thermo-hydraulic performances of the LEU core. This report describes the results of the design calculations in comparison with those of the current HEU core.

2. DESCRIPTIONS OF FUEL ELEMENTS AND REACTOR CORE

2.1 LEU AND HEU FUEL ELEMENTS

The description of the LEU fuel element is shown in Table 2 together with that of the HEU fuel element. The drawing of the LEU fuel element is shown in Fig. 1.

The LEU fuel element is a plate type fuel using 19.75% enriched UAlx-Al dispersion fuel with uranium density of 2.2 g/cm^3. To compensate for the reactivity loss due to the greatly increased absorption by U-238, the LEU fuel element contains 221.4 g U-235 which is about 1.3 times larger than the HEU fuel element. In order to accommodate the increased uranium loading, fuel meat thickness and a number of fuel plates were modified from 0.05 to 0.089 cm and from 15 to 16 plates, respectively. Consequently, the water gap width was decreased from 0.41 to 0.335 cm.

1. INTRODUCTION

Japan Research Reactor 4 (JRR-4) is a 3.5 MW swimming pool type research reactor and uses 93% enriched UAl-alloy MTR-type fuel elements. The reactor has been utilized generally for shielding experiments, RI production, activation analyses, reactor engineering experiments, and training for reactor engineers.

The reduced enrichment program for JRR-4 was started in 1979 to convert the fuel from High-Enrichment Uranium (HEU) to Low-Enrichment Uranium (LEU). Some neutronic analyses and thermo-hydraulic experiments were carried out[1] and a UAlx-Al MTR-type fuel element of uranium density of 2.2 g/cm^3 was chosen as a candidate of the LEU fuel.

In order to confirm the neutronic performance and irradiation behavior of the LEU fuel element, two LEU fuel elements were fabricated. One fuel element was served for the neutronic performance measurement in JRR-4 and the other for the accelerated irradiation test in Japan Research Reactor 2 (JRR-2). The results indicate that the LEU fuel element has almost the same reactivity worth as the HEU fuel element as shown in Table 1[2] and irradiation behavior is satisfiable.[3]

Extensive design calculations have been further carried out on the neutronic and thermo-hydraulic performances of the LEU core. This report describes the results of the design calculations in comparison with those of the current HEU core.

2. DESCRIPTIONS OF FUEL ELEMENTS AND REACTOR CORE

2.1 LEU AND HEU FUEL ELEMENTS

The description of the LEU fuel element is shown in Table 2 together with that of the HEU fuel element. The drawing of the LEU fuel element is shown in Fig. 1.

The LEU fuel element is a plate type fuel using 19.75% enriched UAlx-Al dispersion fuel with uranium density of 2.2 g/cm^3. To compensate for the reactivity loss due to the greatly increased absorption by U-238, the LEU fuel element contains 221.4 g U-235 which is about 1.3 times larger than the HEU fuel element. In order to accommodate the increased uranium loading, fuel meat thickness and a number of fuel plates were modified from 0.05 to 0.089 cm and from 15 to 16 plates, respectively. Consequently, the water gap width was decreased from 0.41 to 0.335 cm.

2.2 CORE CONFIGURATION

The core consists of 20 fuel elements with a 4 by 5 array and 7 control rods as shown in Fig. 2. The core is surrounded by graphite reflector elements, aluminum irradiation pipes, and a neutron source. They are arranged on a grid plate with an 8 by 8 array of 8.1 cm square lattice pitch. The size of the core is 40.4 x 34.4 x 60.0 cm.

There are three kinds of control rods, such as shim rods (C1-C4), regulating rod (C5), and backup rods (B1, B2). They are made of 0.5 cm thick boron stainless steel containing 1.6% natural boron. The graphite of the reflector element is enclosed with an aluminum can. There are five irradiation pipes which are named as T-, S-, Pn-, L-, and D-pipe, respectively. The neutron source is Am-Be of 5 curies and always loaded at the lattice location D-2.

3. NEUTRONICS

3.1 CALCULATION METHODS

3.1.1 PROCEDURES

The neutronic design of the LEU core was performed using a neutronic design code system SRAC.[4)] The design method employed was confirmed by a series of benchmark calculations on the current HEU core.[5)]

A microscopic cross section library with 107 groups based on ENDF/B-4 was chosen from among the libraries provided for the SRAC and the cross section data collapsed into 61-group structure were used for the neutronic design. Three group macroscopic cross sections were generated for the core calculation. The upper energy boundaries in the 3-group scheme are 10 MeV for the fast group, 183 keV for the epi-thermal group, and 0.683 eV for the thermal group.

The 10-group macroscopic cross sections were also generated for the confirmation of the axial buckling used in the X-Y 2-dimensional calculation and for the adjustment of the internal boundary condition on the control rod surface.

The macroscopic cross sections of the fuel element were calculated using the collision probability method. Two kinds of heterogeneities were taken into account in this calculation step. The first heterogeneity is that of the fuel plate which consists of fuel meat, cladding and moderator. The second one is that of the fuel element which consists of several fuel plates, side plates and moderator.

2.2 CORE CONFIGURATION

The core consists of 20 fuel elements with a 4 by 5 array and 7 control rods as shown in Fig. 2. The core is surrounded by graphite reflector elements, aluminum irradiation pipes, and a neutron source. They are arranged on a grid plate with an 8 by 8 array of 8.1 cm square lattice pitch. The size of the core is 40.4 x 34.4 x 60.0 cm.

There are three kinds of control rods, such as shim rods (C1-C4), regulating rod (C5), and backup rods (B1, B2). They are made of 0.5 cm thick boron stainless steel containing 1.6% natural boron. The graphite of the reflector element is enclosed with an aluminum can. There are five irradiation pipes which are named as T-, S-, Pn-, L-, and D-pipe, respectively. The neutron source is Am-Be of 5 curies and always loaded at the lattice location D-2.

3. NEUTRONICS

3.1 CALCULATION METHODS

3.1.1 PROCEDURES

The neutronic design of the LEU core was performed using a neutronic design code system SRAC.[4] The design method employed was confirmed by a series of benchmark calculations on the current HEU core.[5]

A microscopic cross section library with 107 groups based on ENDF/B-4 was chosen from among the libraries provided for the SRAC and the cross section data collapsed into 61-group structure were used for the neutronic design. Three group macroscopic cross sections were generated for the core calculation. The upper energy boundaries in the 3-group scheme are 10 MeV for the fast group, 183 keV for the epi-thermal group, and 0.683 eV for the thermal group.

The 10-group macroscopic cross sections were also generated for the confirmation of the axial buckling used in the X-Y 2-dimensional calculation and for the adjustment of the internal boundary condition on the control rod surface.

The macroscopic cross sections of the fuel element were calculated using the collision probability method. Two kinds of heterogeneities were taken into account in this calculation step. The first heterogeneity is that of the fuel plate which consists of fuel meat, cladding and moderator. The second one is that of the fuel element which consists of several fuel plates, side plates and moderator.

Burnup dependent macroscopic cross sections of the fuel element were calculated under xenon-free condition using the chain scheme of Garrison model. The macroscopic cross sections of structure materials with and without moderator were collapsed using the neutron spectrum of the fresh fuel element and the asymptotic spectrum provided in SRAC, respectively. These macroscopic cross sections were generated at 300 K except for the calculation of the reactivity coefficients.

The control rod worth is underestimated by using the macroscopic cross section which is obtained by the collision probability method.[6] Therefore, the internal boundary condition for the control rod regions was employed for the thermal group instead of the thermal group macroscopic cross section in 3-group energy structure. To obtain the internal boundary condition, 10-group transport calculation and 3-group diffusion calculation were carried out by using the same 2-dimensional 1/4 core model. The internal boundary condition of the diffusion calculation was determined to obtain the same control rod worth as that of the transport calculation.

An axial buckling value for the 2-dimensional core calculation was adjusted so as to obtain the same k_{eff}'s between the X-Y 2-dimensional 3-group full core calculation and 3-dimensional 10-group full core calculation.

To evaluate the burnup dependence of the neutronic parameters, core burnup calculations were performed using the core burnup code COREBURN at three kinds of burnup steps, such as cold clean, beginning of equilibrium core (BOC, ρ_{ex}=6.4% $\Delta k/k$) and end of equilibrium core (EOC, ρ_{ex}=5.0% $\Delta k/k$). The xenon poisoning effect on the reactions was also evaluated by the one point approximation code XEBUILDUP.[7]

3.1.2 CORE MODELS

The core calculation models are shown in Figs. 3 and 4. The standard mesh dividing on the X-Y plane was 4 x 4 mesh in a fuel element for the 2-dimensional calculation and 2 x 2 mesh in a fuel element for the 3-dimensional calculation. Standard mesh interval of the Z-axis for the 3-dimensional calculation is 5 cm. The thickness of the light water reflector is 30 cm from the outside of the core tank in X- and Y- directions and 30 cm from the both ends of the fuel element in an axial direction.

3.2 NEUTRONIC PERFORMANCES

The neutronic performances, such as excess reactivity, xenon poisoning, control rod worth, neutron flux, power distribution, reactivity coefficient, and kinetic parameters are summarized in this section.

3.2.1 EXCESS REACTIVITIES AND BURNUPS

The changes of the excess reactivity and core average burnup from cold clean core to the equilibrium burnup core are shown in Figs. 5 and 6. The reactivities were calculated with the 2-dimensional diffusion model and the burnups were with 3-dimensional diffusion model, respectively.

The replacement of the fuel elements were carried out by a zone loading method. The first replacement was made when the excess reactivity of the core reached to the EOC (5.0% $\Delta k/k$). After that, it was made every 52.5 MWd operation. The number of the fuel elements to be replaced was decided by taking into account of the excess reactivity of the core and the burnup of each fuel element.

The excess reactivity of the LEU cold clean core is about 8.45% $\Delta k/k$ which is about 1% less than that of the HEU cold clean core. However, the decreasing rate of the excess reactivity of the LEU core is smaller than that of the HEU core and the burnup lifetime is about 20% longer than that of the HEU core. The burnups at the BOC and EOC of the LEU cores are 6.4 and 9.3% U-235, respectively. They are about 1 - 2% less than those of the HEU core.

3.2.2 XENON POISONING

JRR-4 is operated 4 days a week from Tuesday through Friday with about 6 hours daily operation. This weekly operation mode is called a cycle.

The negative reactivities caused by xenon poisoning and reactivity changes during a cycle operation at the EOC are shown in Table 3 and Fig. 7. The duration of operation is supposed at the calculation to be 5.5 hours on Tuesday and 6.0 hours from Wednesday through Friday. The operation time on Tuesday is shortened to spare time for checking the excess reactivity of the core.

The poisoning effect changes according to the reactor operation and becomes maximum on Friday during an operation cycle. The maximum negative reactivities during the operation and after shutdown are 1.8 and 2.5% $\Delta k/k$, respectively. The xenon poisoning effect of the LEU core is about 20% less than that of the HEU core due to the decrease of the thermal neutron flux in the fuel region.

3.2.3 CONTROL ROD WORTHS

Control rod worths were calculated with 3-dimensional model. The results are shown in Table 4.

Both control rod worths of the LEU and HEU cores are almost the same. The control rod worth becomes minimum at the cold clean core. The total shim rod worth and reactivity shutdown margin are 18.6 and 11.0% $\Delta k/k$, respectively. The one rod stuck margin is 4.7% $\Delta k/k$, so that the LEU core has enough shutdown margin to stop the reactor even under the one rod stuck condition.

3.2.4 NEUTRON FLUXES AND POWER DISTRIBUTIONS

The neutron fluxes and power densities were calculated by 3-dimensional diffusion model. The average neutron fluxes and typical thermal neutron flux distributions in the BOC are shown in Table 5 and Fig. 8. A power peaking factor is defined as the ratio of the maximum power density in each fuel element to the core average power density. The power peaking factors of the cold clean core which has the largest value throughout burnup and a typical power density distribution in the BOC are also shown in Figs. 9 and 10.

The thermal neutron flux in the fuel region of the LEU core decreases about 25% on the average because of the hardening of the neutron spectrum. The hardened neutron spectrum in the fuel region, however, increases the thermal component in the spectrum because of the thermalization by the moderator in the irradiation pipes. So that, almost the same thermal neutron fluxes are obtained as the HEU core in the irradiation pipes. This is why the power distribution in the LEU core becomes flatter than in the HEU core and the peaking factor of the hot channel (lattice location D-5) decreases by about 6%.

3.2.5 REACTIVITY COEFFICIENTS

The moderator temperature coefficient was calculated for the combined effects of temperature and density change of the moderator in the fuel region. The fuel temperature coefficient was calculated for the temperature change of the fuel meat. The void coefficient of the moderator was calculated for the density change as a function of void fraction in water in the fuel region. These calculations were performed by the 2-dimensional diffusion model using the axial buckling which makes k_{eff} of the reference core (at 300 K, with no void) unity.

Reactivity changes as a function of moderator and fuel meat temperature are shown in Fig. 11, and a reactivity change as a function of void fraction in Fig. 12. The reactivity coefficients at 300 K are shown in Table 6. There is no significant change at 300K in the moderator temperature coefficient between the LEU and HEU cores although the LEU core has slightly larger reactivity coefficient than that of the HEU core over 300 K. The fuel temperature coefficient of the HEU core is negligibly small because of the small amount of U-238, while that of the LEU core increases about 12 times due to the large amount of U-238. The void coefficient increases about 30% due to both effects of the hardening of the neutron spectrum and the increase of the neutron leakage caused by the decrease in the volume of moderator in the fuel element. So that, the LEU core has larger feedback reactivity than that of the HEU core.

3.2.6 KINETIC PARAMETERS

The prompt neutron lifetimes (l) and the effective delayed neutron fractions (β_{eff}) were calculated by the perturbation theory with 6 delayed neutron family using real and adjoint neutron fluxes calculated by 3-group 3-dimensional diffusion model with critical control rod positions.

The kinetic parameters are shown in Table 7. The prompt neutron life time of the LEU core is about 15% shorter than that of the HEU core because of hardening of the neutron spectrum and the increase of macroscopic absorption cross section. Since there is no significant changes in the effective delayed neutron fractions, however, kinetic performance of the reactor near the critical state would be almost the same as the HEU core.

4. THERMO-HYDRAULICS

The basic conditions on the thermo-hydraulic design to accommodate the use of the LEU fuel element are: (a) the geometries of primary components of the reactor should not be changed except for the fuel element and (b) coolant flow rate and cooling system should not be changed.

4.1 CALCULATION PROCEDURES

As the steady state thermo-hydraulic design of the LEU core, onset of nucleate boiling (ONB) temperature, fuel plate surface temperature, saturation temperature and coolant temperature were calculated by a computer code COOLOD.[8] In the calculation, the Dittus-Boelter's equation was chozen for the calculations of heat

Reactivity changes as a function of moderator and fuel meat temperature are shown in Fig. 11, and a reactivity change as a function of void fraction in Fig. 12. The reactivity coefficients at 300 K are shown in Table 6. There is no significant change at 300K in the moderator temperature coefficient between the LEU and HEU cores although the LEU core has slightly larger reactivity coefficient than that of the HEU core over 300 K. The fuel temperature coefficient of the HEU core is negligibly small because of the small amount of U-238, while that of the LEU core increases about 12 times due to the large amount of U-238. The void coefficient increases about 30% due to both effects of the hardening of the neutron spectrum and the increase of the neutron leakage caused by the decrease in the volume of moderator in the fuel element. So that, the LEU core has larger feedback reactivity than that of the HEU core.

3.2.6 KINETIC PARAMETERS

The prompt neutron lifetimes (l) and the effective delayed neutron fractions (β_{eff}) were calculated by the perturbation theory with 6 delayed neutron family using real and adjoint neutron fluxes calculated by 3-group 3-dimensional diffusion model with critical control rod positions.

The kinetic parameters are shown in Table 7. The prompt neutron life time of the LEU core is about 15% shorter than that of the HEU core because of hardening of the neutron spectrum and the increase of macroscopic absorption cross section. Since there is no significant changes in the effective delayed neutron fractions, however, kinetic performance of the reactor near the critical state would be almost the same as the HEU core.

4. THERMO-HYDRAULICS

The basic conditions on the thermo-hydraulic design to accommodate the use of the LEU fuel element are: (a) the geometries of primary components of the reactor should not be changed except for the fuel element and (b) coolant flow rate and cooling system should not be changed.

4.1 CALCULATION PROCEDURES

As the steady state thermo-hydraulic design of the LEU core, onset of nucleate boiling (ONB) temperature, fuel plate surface temperature, saturation temperature and coolant temperature were calculated by a computer code COOLOD.[8] In the calculation, the Dittus-Boelter's equation was chozen for the calculations of heat transfer coefficient and Bergles-Rohsenow correlation was used to evaluate the ONB temperature.

The differences in the thermo-hydraulic design between the LEU and HEU fuel element are the number of plates in the fuel element, the plate thickness, and the water channel thickness. Consequently, the element flow area decreased from 40 to 35 cm^2. In order to confirm the differences of the thermo-hydraulic parameters due to these modifications, coolant flow tests for the fuel element were performed by using a full scale mockup facility, and the core flow distributions and coolant velocities were experimentally obtained.

The results of the coolant flow test are summarized in Table 8 together with the input values for the thermo-hydraulic calculation. The fraction of total flow rate in the fuel region to the core at 7.0 m^3/min decreased from 84% of the HEU core to 82.5% of the LEU core. The average flow rate per fuel element decreases slightly 17.6 m^3/h for the HEU core to 17.3 m^3/h for the LEU core, but coolant velocity increases from 1.21 to 1.36 m/s.[1)]

4.2 THERMO-HYDRAULIC PERFORMANCES

The temperature distribution based on the power distribution along the fuel channel in the HEU and the LEU fuel elements and the maximum values are shown in Fig. 13 and Table 9. The inlet coolant temperature and pressure are 35 ℃ and 1.84 kg/cm^2 abs., respectively.

The flow resistance across the fuel element increases slightly in the LEU fuel element. However, fuel surface temperature has enough margin with respect to the ONB temperature. This is mainly because of low power density due to increase of the fuel plates and higher flow velocity in the LEU fuel element.

5. CONCLUSION

The neutronic and thermo-hydraulic design were carried out for the LEU core and its performances were compared with those of the HEU core. As the result, it was confirmed that the LEU core has almost the same neutronic and thermo-hydraulic characteristics and safety margins as the those of the HEU core. Therefore, there is no significant problem to convert the fuel from HEU to LEU.

transfer coefficient and Bergles-Rohsenow correlation was used to evaluate the ONB temperature.

The differences in the thermo-hydraulic design between the LEU and HEU fuel element are the number of plates in the fuel element, the plate thickness, and the water channel thickness. Consequently, the element flow area decreased from 40 to 35 cm^2. In order to confirm the differences of the thermo-hydraulic parameters due to these modifications, coolant flow tests for the fuel element were performed by using a full scale mockup facility, and the core flow distributions and coolant velocities were experimentally obtained.

The results of the coolant flow test are summarized in Table 8 together with the input values for the thermo-hydraulic calculation. The fraction of total flow rate in the fuel region to the core at 7.0 m^3/min decreased from 84% of the HEU core to 82.5% of the LEU core. The average flow rate per fuel element decreases slightly 17.6 m^3/h for the HEU core to 17.3 m^3/h for the LEU core, but coolant velocity increases from 1.21 to 1.36 m/s.[1)]

4.2 THERMO-HYDRAULIC PERFORMANCES

The temperature distribution based on the power distribution along the fuel channel in the HEU and the LEU fuel elements and the maximum values are shown in Fig. 13 and Table 9. The inlet coolant temperature and pressure are 35 ℃ and 1.84 kg/cm^2 abs., respectively.

The flow resistance across the fuel element increases slightly in the LEU fuel element. However, fuel surface temperature has enough margin with respect to the ONB temperature. This is mainly because of low power density due to increase of the fuel plates and higher flow velocity in the LEU fuel element.

5. CONCLUSION

The neutronic and thermo-hydraulic design were carried out for the LEU core and its performances were compared with those of the HEU core. As the result, it was confirmed that the LEU core has almost the same neutronic and thermo-hydraulic characteristics and safety margins as the those of the HEU core. Therefore, there is no significant problem to convert the fuel from HEU to LEU.

REFERENCES

1. S. Watanabe, T. Otsuka, K. Arigane, H. Shitomi, K. Kaieda, Y. Suzuki, "Neutronic and Flow Analysis of LEU Core in JRR-4", Proceedings of the International Meeting on RERTR, October 24-27, 1983, Tokai, Japan, JAERI-M 84-073(1984).

2. M. Morozumi, K. Kaieda, K. Arigane, H. Shitomi, "Irradiation and Nuclear Characteristics Measurements of JRR-4 LEU Fuel Elements", Proceedings of the International Meeting on RERTR, November 3-6, 1986, Gatlinburg, USA.

3. Y. Futamura, E. Shirai, H. Tsuruta, M. Adachi, T. Kikuchi, "Post-Irradiation Examination of Low Enriched UAlx-Al Fuel Element in JAERI", Proceedings of the International Meeting on RERTR, September 28 - October 2, 1987, Buenos Aires, Argentina.

4. K. Tsuchihashi, Y. Ishiguro, K. Kaneko, M. Ido, "Revised SRAC Code System", JAERI 1302(1986).

5. K. Arigane, "Benchmark Calculation on Nuclear Characteristics of JRR-4 HEU Core by SRAC Code System", (in Japanese), JAERI-M 87-063(1987).

6. K. Arigane, et al., to be published in JAERI-M report.

7. JRR-4 Operation Section, "Outline of Critical Experiment and Characteristic Test of JRR-4", (in Japanese), JAERI 1139(1967).

8. S. Watanabe, "COOLOD: Thermal and Hydraulic Analysis Code for Research Reactors with Plate Type Fuel Element.", (In Japanese), JAERI-M 84-162(1984).

Table 1 Comparison of reactivity worths between LEU and HEU fuel elements

(unit: %dk/k)

	Fuel Element		LEU/HEU
	LEU	HEU	
Experiment	5.85	5.94	0.98
Calculation	5.26	5.33	0.98

Table 2 Description of JRR-4 LEU and HEU fuel elements

Item		LEU		HEU	
		Inner Plate	Outer Plate	Inner Plate	Outer Plate
Meat	Enrichment (wt %)	19.75		93.15	
	Material	UAl_x-Al		UAl Alloy	
	U-Content (wt %)	52	31	20	11
	U-Density (g/cm^3)	2.20	1.10	0.66	0.33
	Void Content (vol %)	8	2	—	—
	Thickness (cm)	0.089	0.089	0.05	0.05
	U-Weight (g)	74.73	37.37	12.75	6.35
	U-235 Weight (g)	14.76	7.38	11.87	5.91
Clad	Thickness (cm)	0.038	0.038	0.038	0.038
Element	No. of Plates	14	2	13	2
	U-235 Weight (g)	221.4		166	
	Water Gap (cm)	0.335		0.410	

Table 3 Xenon poisoning reactivities in LEU and HEU cores

(unit:-%dk/k)

Core	Day of the Week							
	TUE		WED		THU		FRI	
	LEU	HEU	LEU	HEU	LEU	HEU	LEU	HEU
Clean	0.8	-	1.5	-	1.7	-	1.7	-
BOC	0.9	1.0	1.5	1.9	1.7	2.2	1.8	2.2
EOC	0.9	1.1	1.5	1.9	1.8	2.2	1.8	2.3

Table 4 Control rod worths in LEU and HEU cores

(unit:%dk/k)

Item	Clean		BOC		EOC	
	LEU	HEU	LEU	HEU	LEU	HEU
Total Shim Rod Worth(C1-C4)	18.6	-	19.2	19.3	19.8	19.9
Regulating Rod Worth(C5)	0.5	-	0.5	0.5	0.5	0.5
Total Backup Rod Worth(B1-B2)	1.8	-	1.9	1.6	1.9	1.7
Shim Rod Shutdown Margin	11.0	-	13.6	13.7	15.8	15.9
One Rod Stuck Margin	4.7	-	7.1	7.1	9.1	9.1

Table 5 Average neutron fluxes in LEU and HEU cores

(unit:10^{13}n/cm^2/s)

Zone	Energy Group	Clean		BOC		EOC	
		LEU	HEU	LEU	HEU	LEU	HEU
Fuel Region	1	3.9	-	3.9	3.6	3.9	3.6
	2	4.0	-	4.0	3.8	4.1	3.8
	3	2.5	-	2.6	3.5	2.7	3.6
Irradiation Pipe (T-Pipe)	1	1.1	-	1.2	1.0	1.2	1.0
	2	1.6	-	1.7	1.5	1.7	1.5
	3	4.0	-	4.1	3.9	4.2	4.0

Table 6 Reactivity coefficients in LEU and HEU cores

(at 300K)

Reactivity Coefficient	Clean		BOC		EOC	
	LEU	HEU	LEU	HEU	LEU	HEU
Moderator Temperature (-10^{-2}%dk/k/°C)	2.1	-	2.1	2.1	2.1	2.1
Fuel Temperature (-10^{-3}%dk/k/°C)	2.4	-	2.4	0.2	2.4	0.2
Moderator Void (-10^{-1}%dk/k/%void)	4.0	-	4.0	3.1	3.9	3.0

Table 7 Kinetic parameters in LEU and HEU core

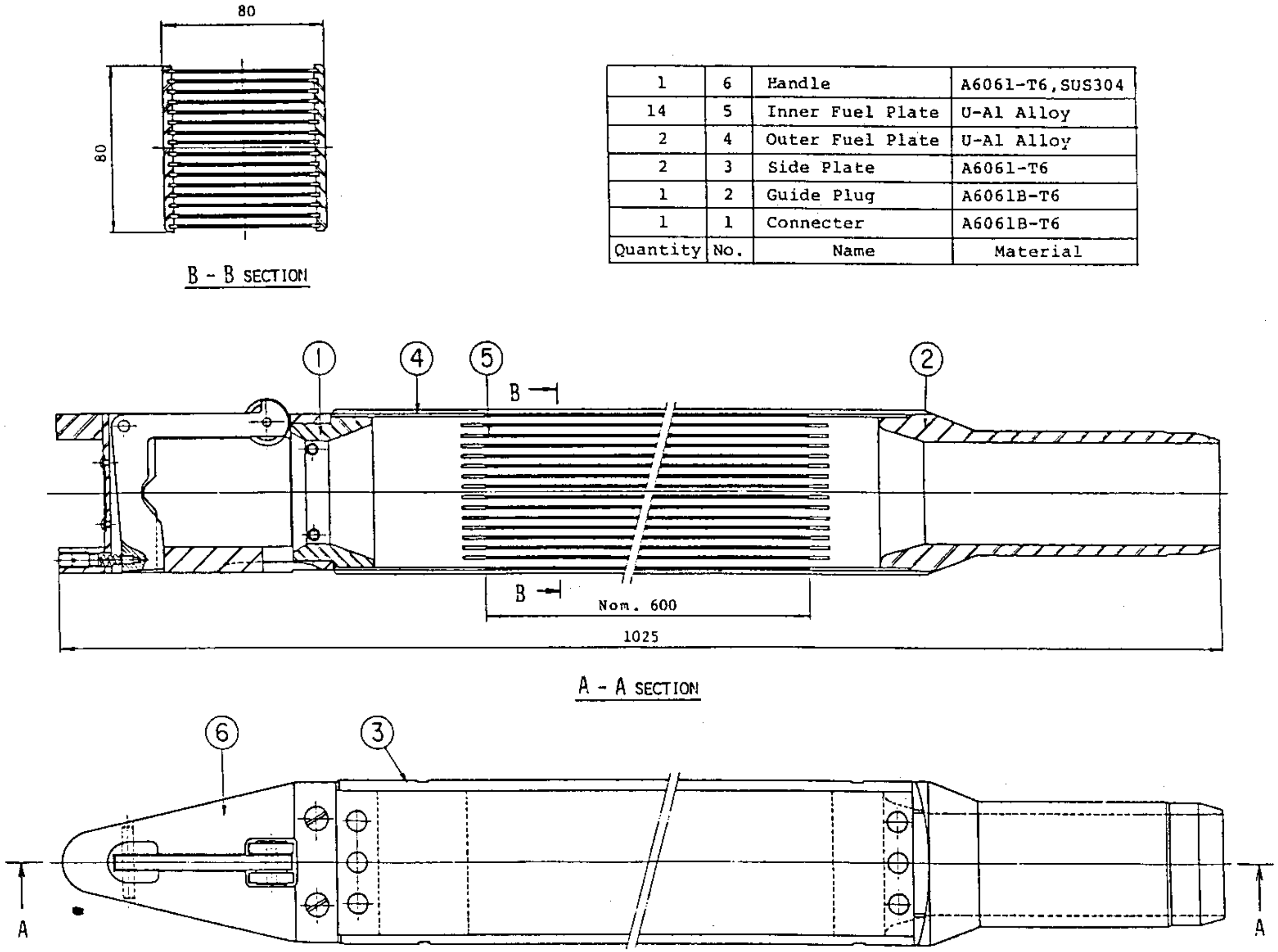
Kinetic Parameter	Clean		BOC		EOC	
	LEU	HEU	LEU	HEU	LEU	HEU
Prompt Neutron Lifetime(10^{-5}s)	5.9	-	6.0	7.0	6.1	7.2
Effective Delayed Neutron Fraction(10^{-3})	7.9	-	7.9	7.8	7.8	7.8

Table 8 Results of the coolant flow test and input data for the Thermo-Hydraulic calculations

	Item	LEU Core	HEU Core
Measured Thermo-Hydraulic Parameters (at 7m³/min)	Fraction of Total Flow Rate in the Fuel Region to the Core (%)	82.5	84.0
	Average Flow Rate per Fuel Element(m³/h)	17.3	17.6
	Coolant Channel Velocity (m/s)	1.36	1.21
Input Data for the Calculation	Coolant Inlet Pressure (kg/cm² abs)	1.84	1.84
	Coolant Inlet Temperature (℃)	35.0	35.0
	Hot Channel Factor		
	Nuclear Peaking Factor		
	Radial	1.47	1.47
	Axial	1.56	1.56
	Local	1.12	1.09
	Engineering Factor		
	for Coolant Temperature Rise	1.21	1.18
	for Film Temperature Rise	1.32	1.30

Table 9 Thermo-Hydraulic performances

Item	LEU Core	HEU Core
Average Heat Flux (w/cm²)	13.9	14.9
Coolant Outlet Temperature at Hot Channel (℃)	52.3	51.4
Maximum Surface Temprature of Fuel Plate (℃)	105.2	115.1
Saturation Temperature at Hot Spot (℃)	117.4	117.5
ONB Temperature at Hot Spot (℃)	125.3	124.9



Quantity	No.	Name	Material
1	6	Handle	A6061-T6,SUS304
14	5	Inner Fuel Plate	U-Al Alloy
2	4	Outer Fuel Plate	U-Al Alloy
2	3	Side Plate	A6061-T6
1	2	Guide Plug	A6061B-T6
1	1	Connecter	A6061B-T6

Fig. 1 JRR-4 LEU fuel element

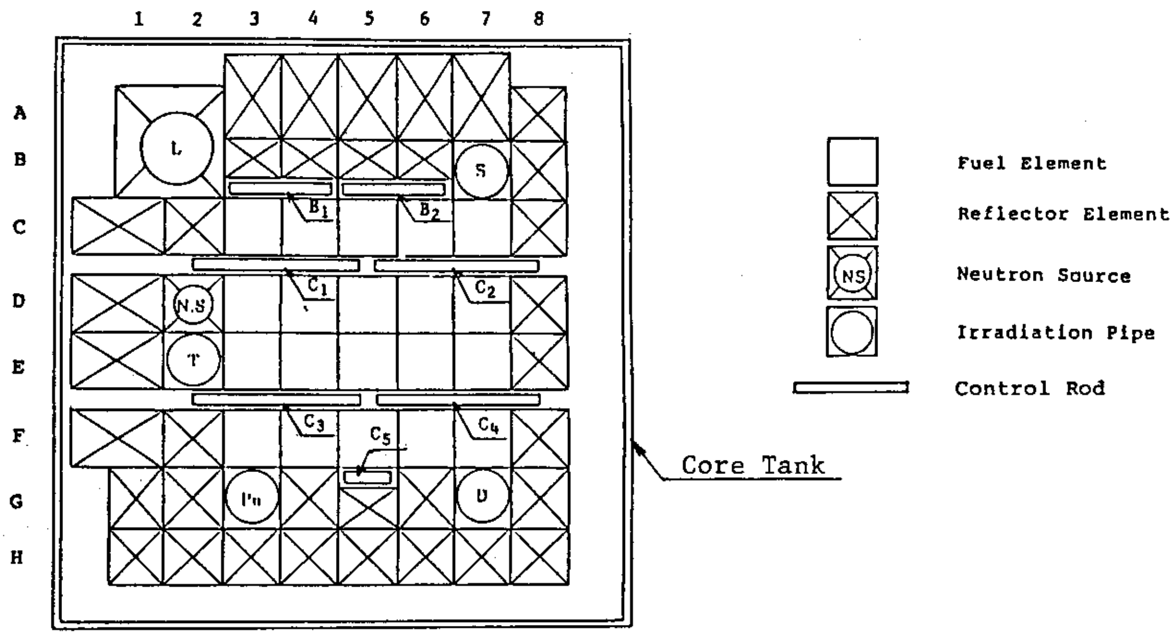


Fig. 2 JRR-4 core configulation

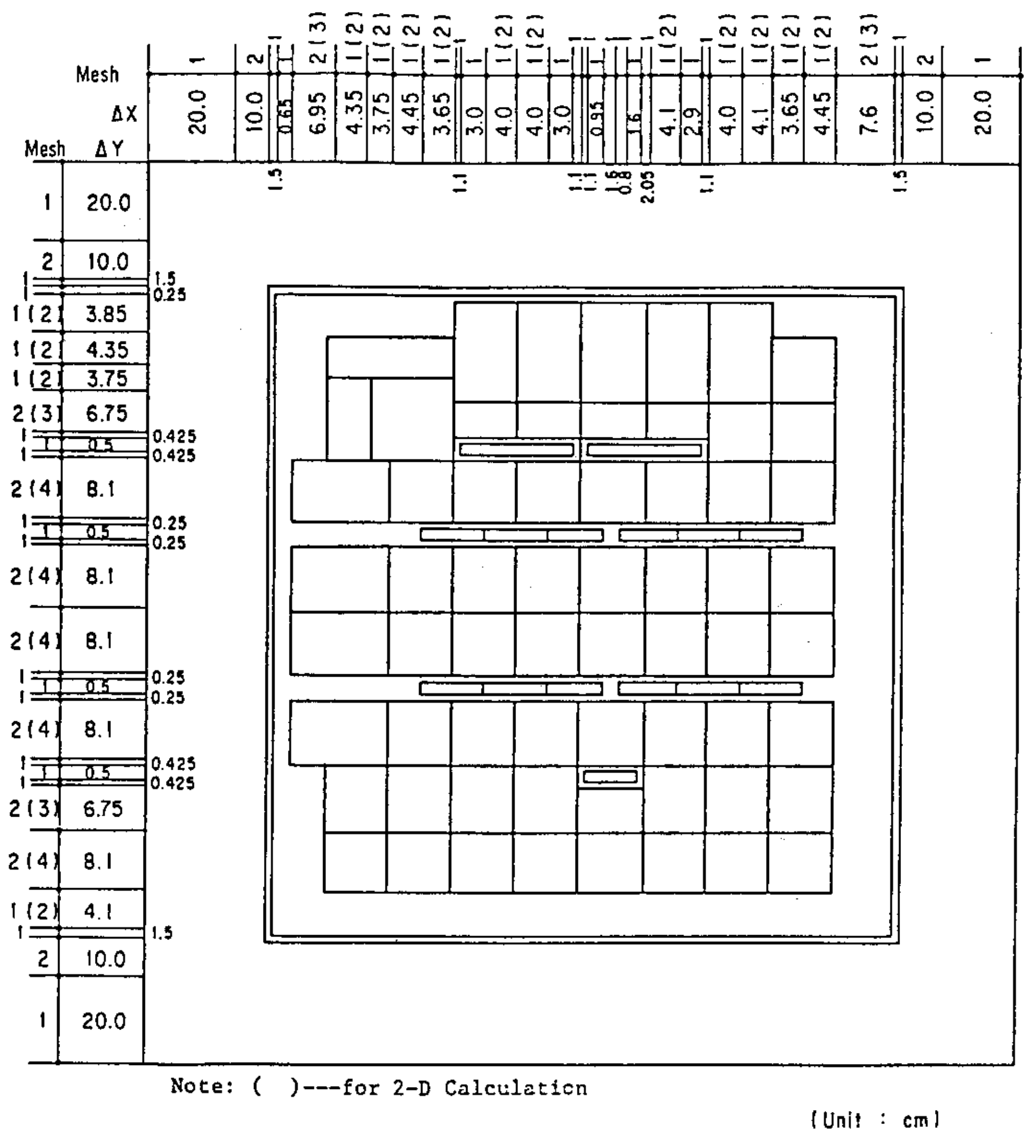


Fig. 3 Core model of X-Y plane for 3-Dimensional critical calculation

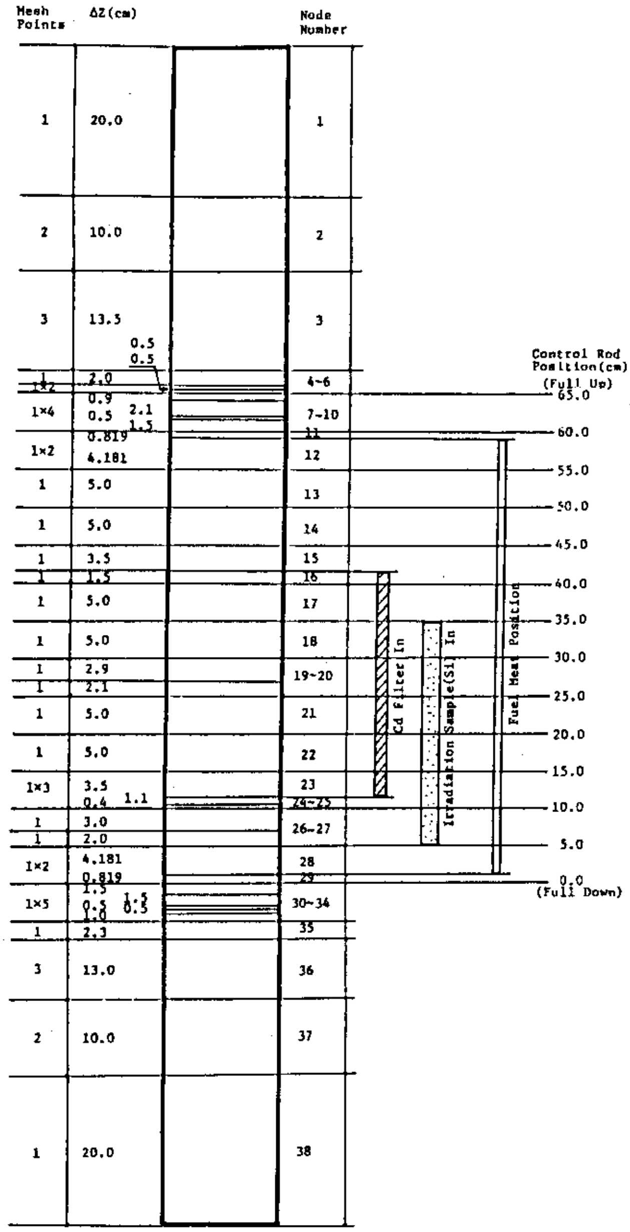


Fig. 4 Core model of Z-Axis for 3-Dimensional critical calculation (LEU core)

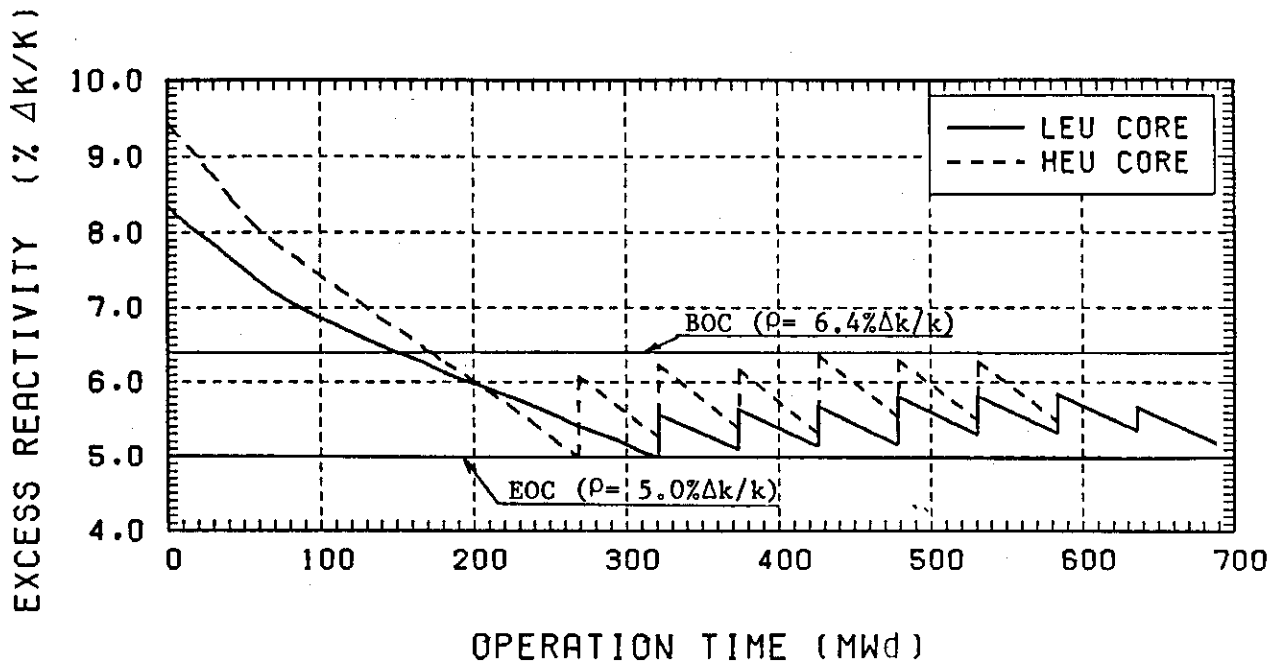


Fig. 5 Excess reactivities in LEU and HEU cores

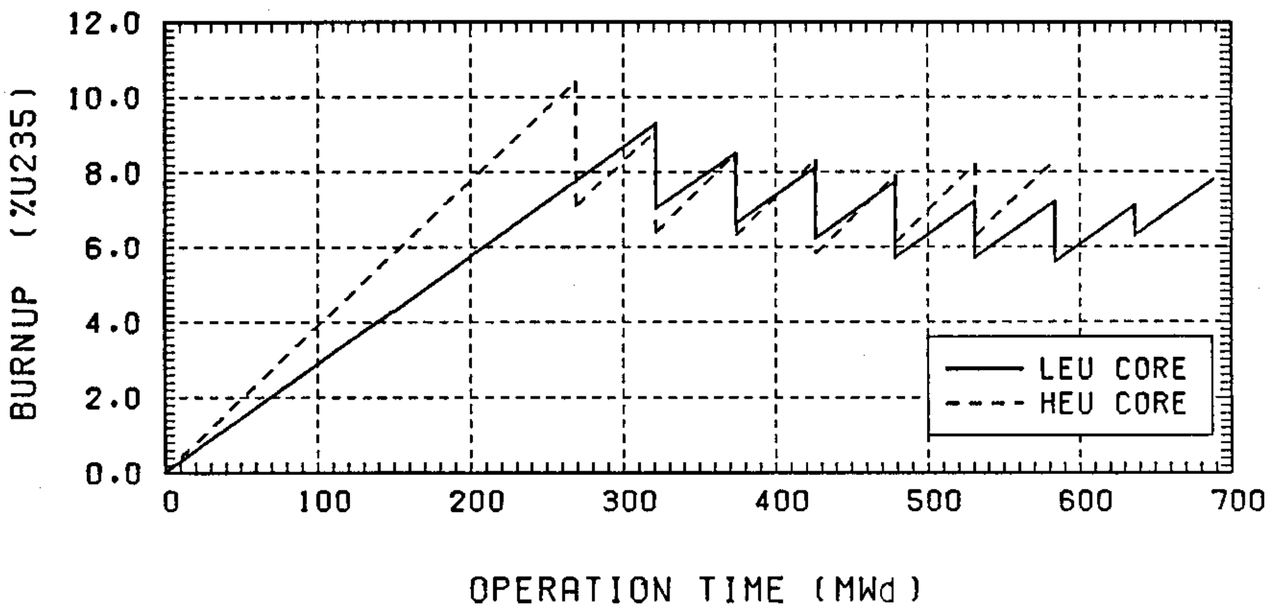


Fig. 6 Core average burnups in LEU and HEU cores

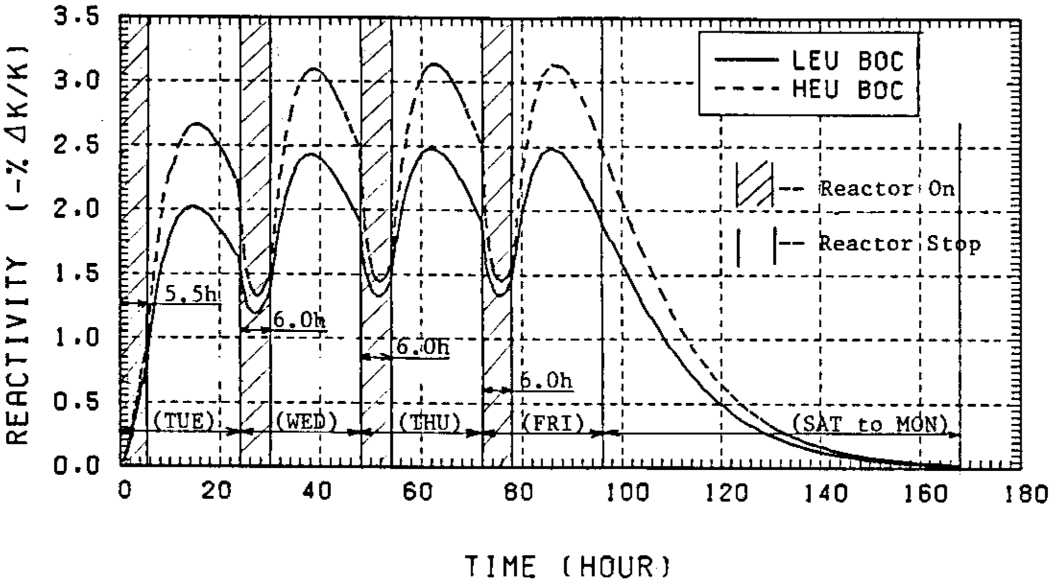


Fig. 7 Xenon poisoning reactivities in LEU and HEU cores

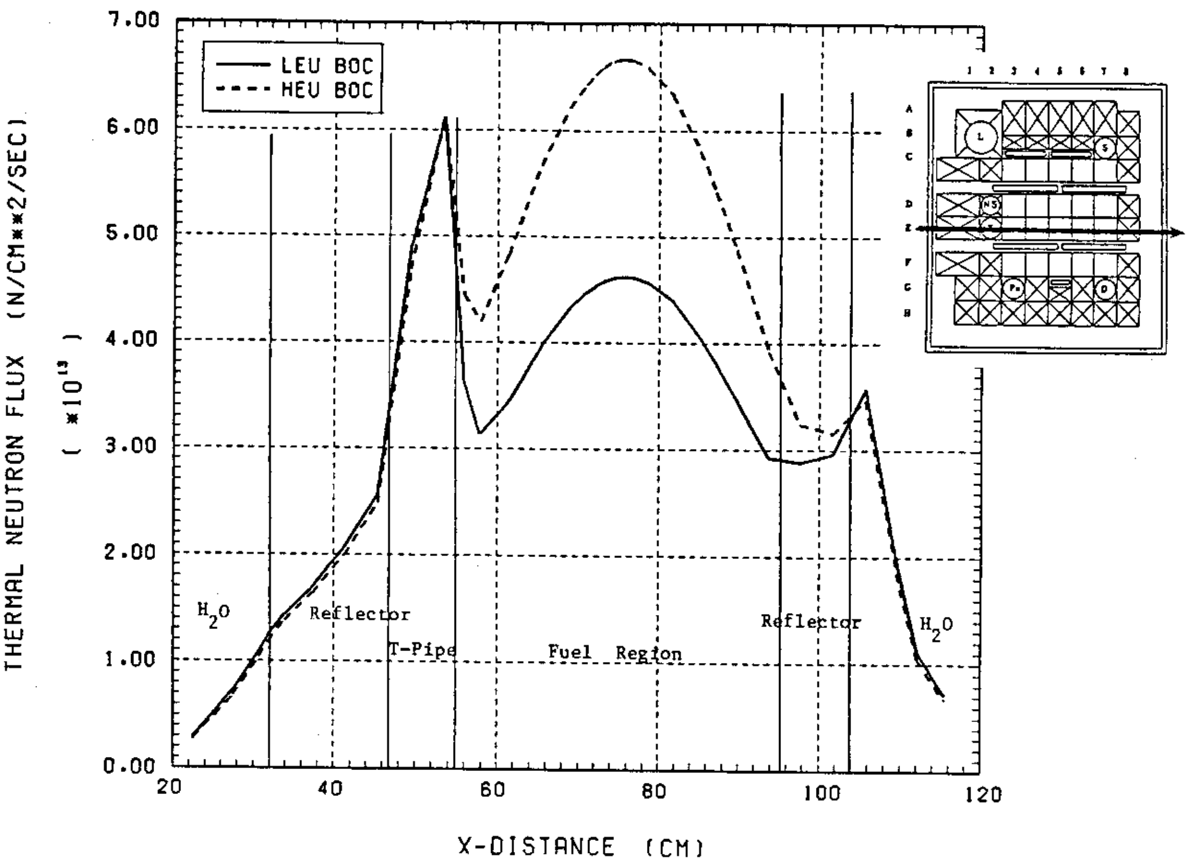


Fig. 8 Thermal neutron fluxes in LEU and HEU cores

	3	4	5	6	7	
C	1.19	1.52	1.68	1.51	1.18	---LEU Core
	1.13	1.52	1.69	1.51	1.13	---HEU Core
	1.05	1.00	0.99	1.00	1.04	---LEU/HEU
D	1.40	1.82	2.03	1.82	1.38	
	1.41	1.93	2.16	1.94	1.43	
	0.99	0.94	0.94	0.94	0.97	
E	1.45	1.81	2.02	1.81	1.38	
	1.46	1.92	2.16	1.93	1.41	
	0.99	0.94	0.94	0.94	0.98	
F	1.15	1.42	1.58	1.43	1.16	
	1.10	1.44	1.61	1.45	1.12	
	1.05	0.99	0.98	0.99	1.04	

□ Fuel Element

Fig. 9 Comparison of peaking factors between LEU and HEU cores

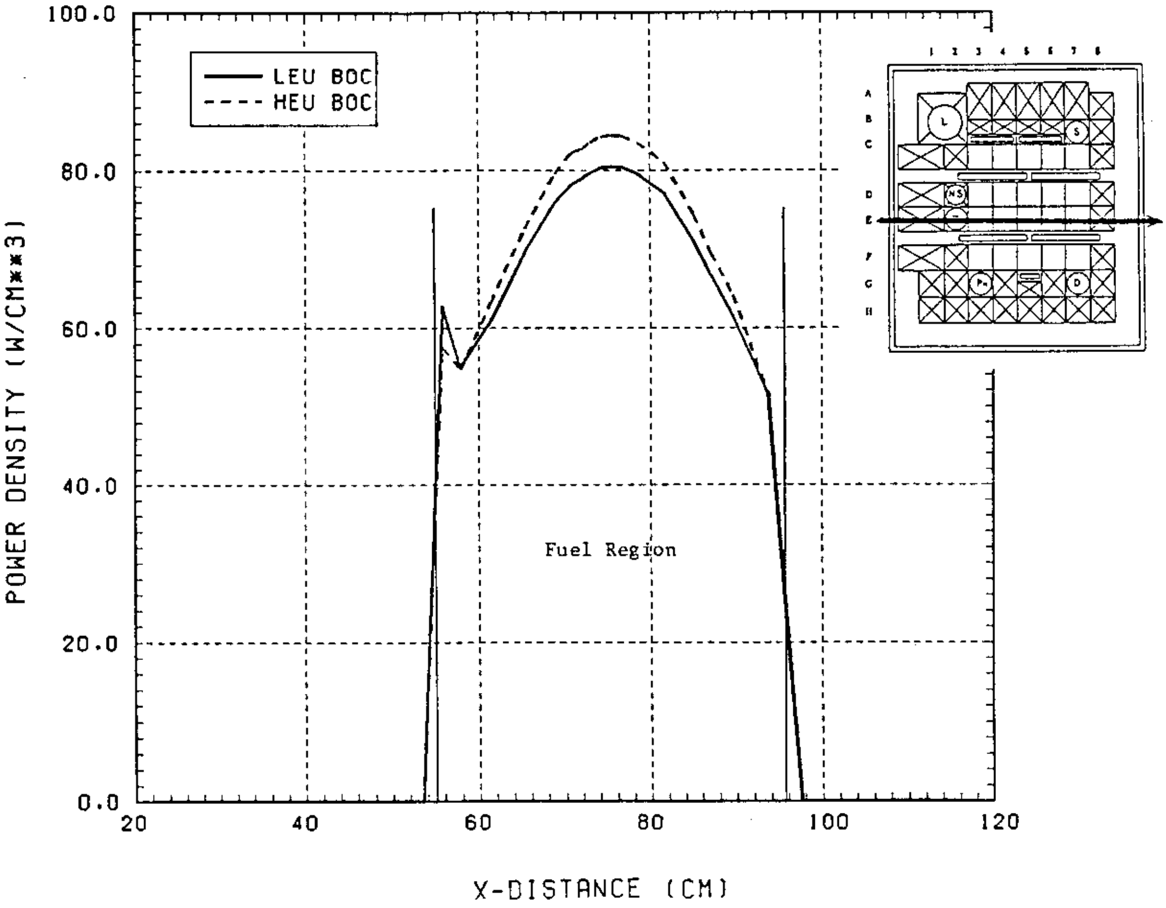


Fig. 10 Power distributions in LEU and HEU cores

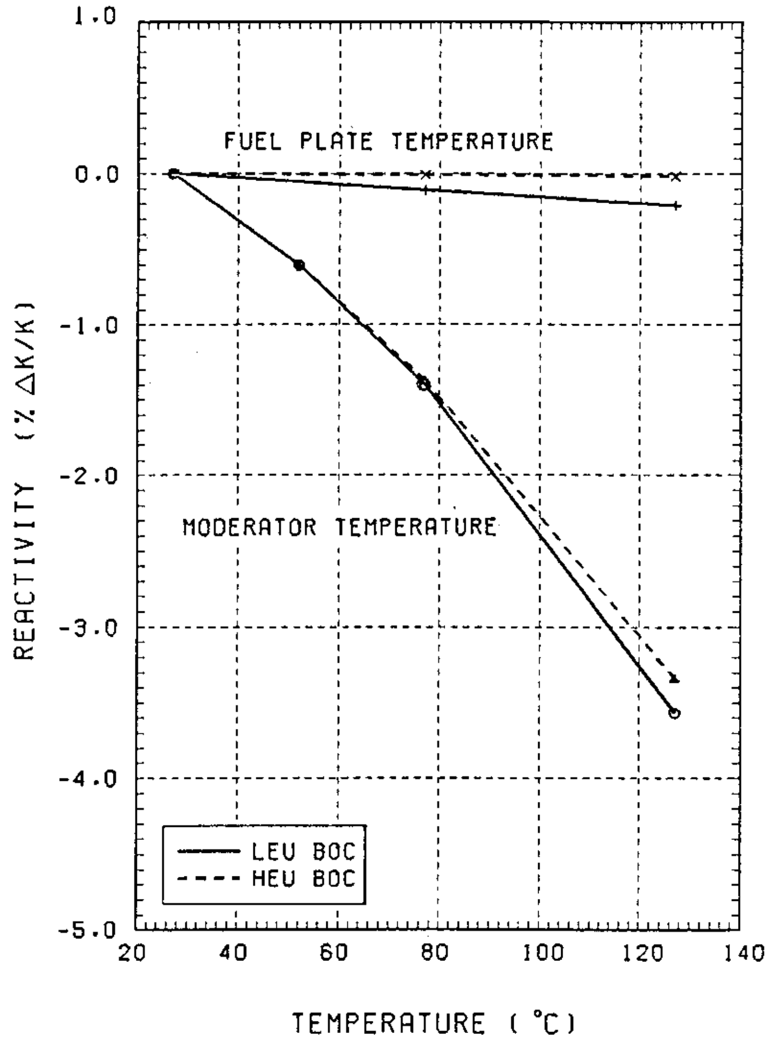


Fig. 11 Reactivity changes as a function of moderator and fuel plate temperature in LEU and HEU cores

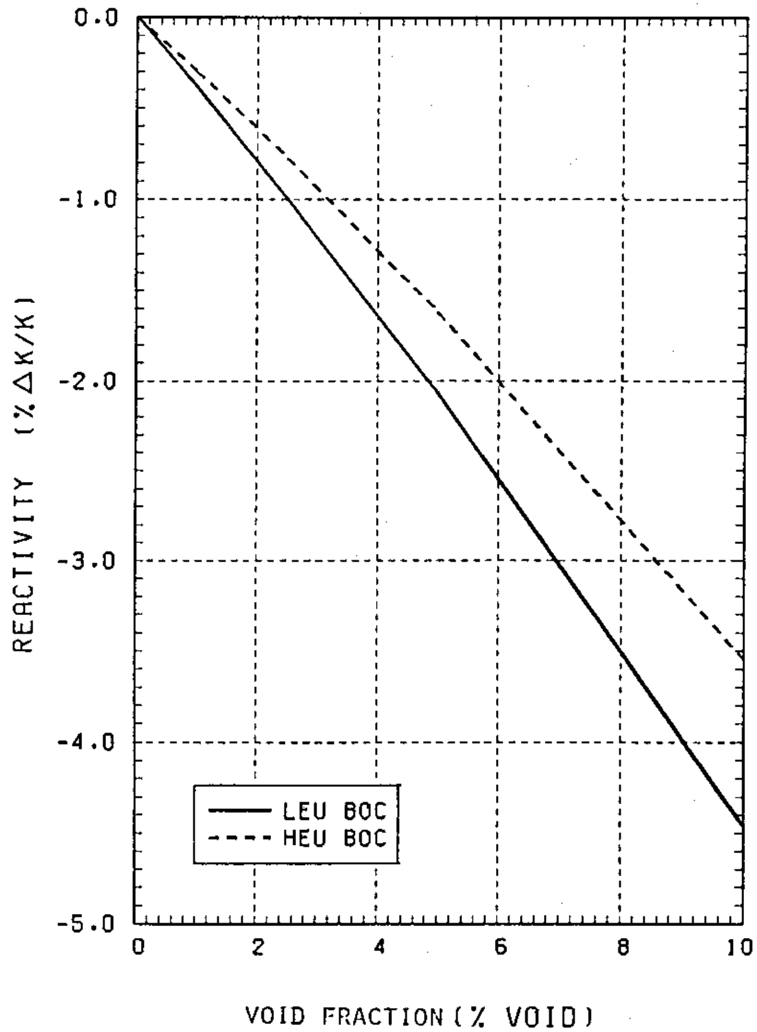


Fig. 12 Reactivity changes as a function of void fraction in LEU and HEU cores

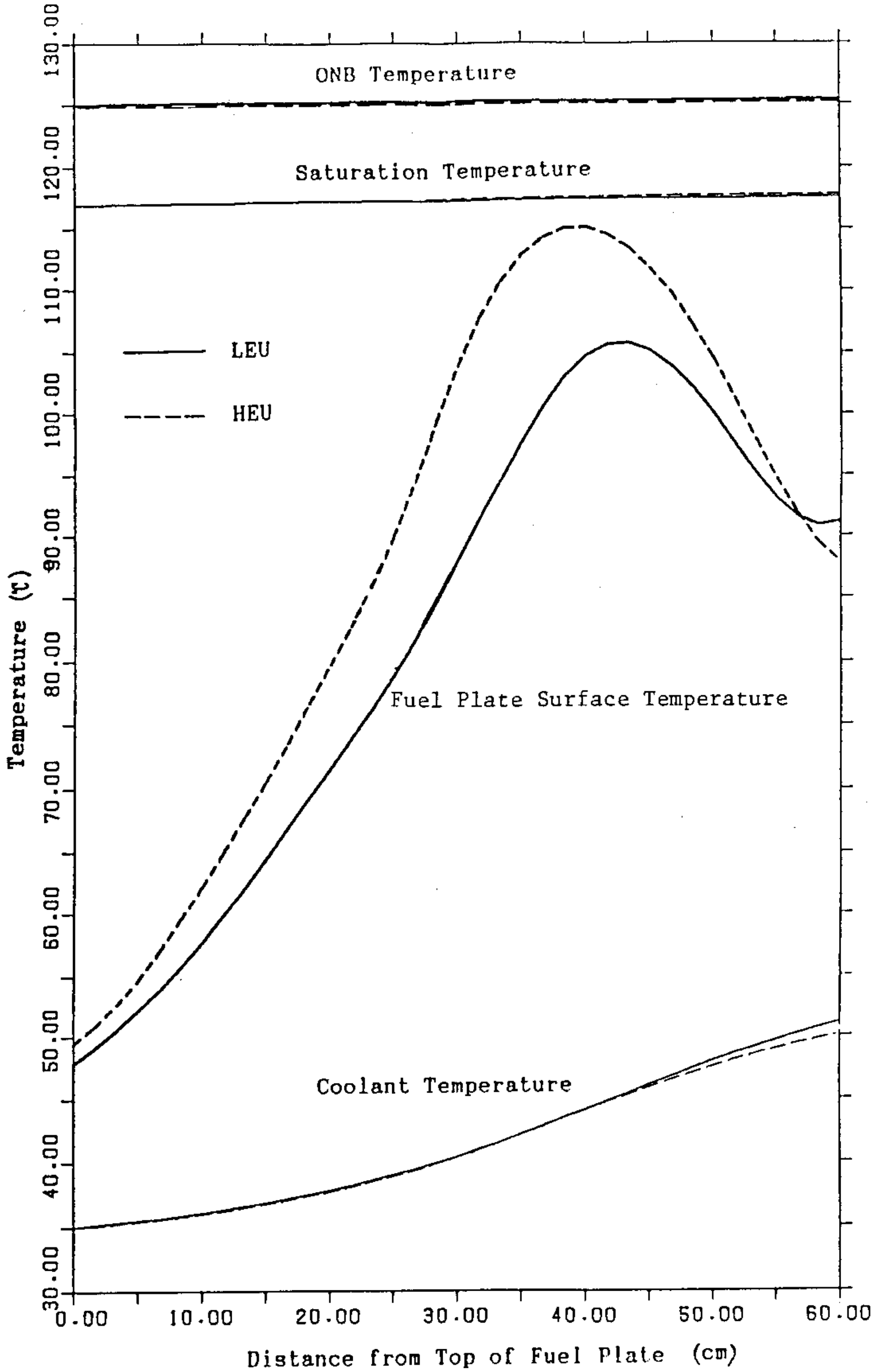


Fig. 13 Axial temperature distribution of LEU and HEU fuel elements