FULL CORE CONVERSION OF THE KYOTO UNIVERSITY RESEARCH REACTOR (KUR) FROM HEU TO LEU

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ABSTRACT

This paper describes the results of conversion of the Kyoto University research reactor (KUR) from HEU to LEU. The full core conversion of KUR to LEU has been achieved on March 2010. After the successful achievement of first criticality on April 15, 2010, the reactor is in full operation since May 28, 2010. Fuel design and fabrication, transport experiences, major results of initial core performance experiments and operational core characteristics are summarized and presented in this paper.

1. Introduction; the Kyoto University Research Reactor (KUR)

Kyoto University Research Reactor (KUR; Fig.1) is a light-water moderated tank-type reactor operated at the rated thermal power of 5MW, sited at the Kyoto University Research Reactor Institute (KURRI) as the KURRI's main neutron source. The core consists of plate-type MTR fuel elements and graphite reflector elements. KUR is operated by using four shim rods and a regulating rod; those are made of the stainless steel containing boron. The core is constructed at the bottom of the aluminum core tank with the size of 2m-diameter and 8m depth, which is filled with light-water. KUR is widely used for the experimental studies in physics, chemistry, biology, engineering, agriculture, medicine etc. Since its first criticality in 1964, it has been successfully operated for over than 40 years, and has served as one of the most useful inter-university research reactors in Japan.

The RERTR program in Japan was initiated in 1978. Four research reactors in Japan, i.e. JRR-3, JRR-4, JMTR of JAEA and KUR of Kyoto University were decided to be converted from HEU to LEU. The three JAEA reactors have been successfully converted by 1998, and the KUR has been the remaining reactor to be converted since then[1].

Conversion to LEU fuel started from 1992 at KUR as a partial core conversion. Two LEU silicide (U_3Si_2-Al) fuel elements have been loaded to the core in May, 1992, and has been successfully irradiated to its maximum burnup (35%-U235). These were the first LEU silicide fuel elements loaded and used in Japanese research reactor.   In 1994, the U. S. Government gave an approval to utilize the HEU fuel material prepared for Kyoto University High Flux Reactor (KUHFR) project[1] to be used in the KUR.   Since then, the KUR continued its operation with HEU until February 23, 2006.   After the Record of Decision (ROD) concerning the ten-year extension of Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program issued by the U.S. DOE in November 2004, a new contract concerning the FRR SNF Acceptance Program has been concluded between the U.S. DOE and Kyoto University in February 2006. This opened the LEU path for KUR beyond 2006, and the safety analysis for the conversion was officially started.

The safety review for the full core conversion to LEU silicide fuel has completed in February 2008. Following the fuel fabrication and transportation, the full core conversion of KUR to LEU was successfully achieved on March 2010, and the reactor is fully operated for joint use studies since May 2010.

The detail of the related activities on LEU core conversion will be described in the following sections.
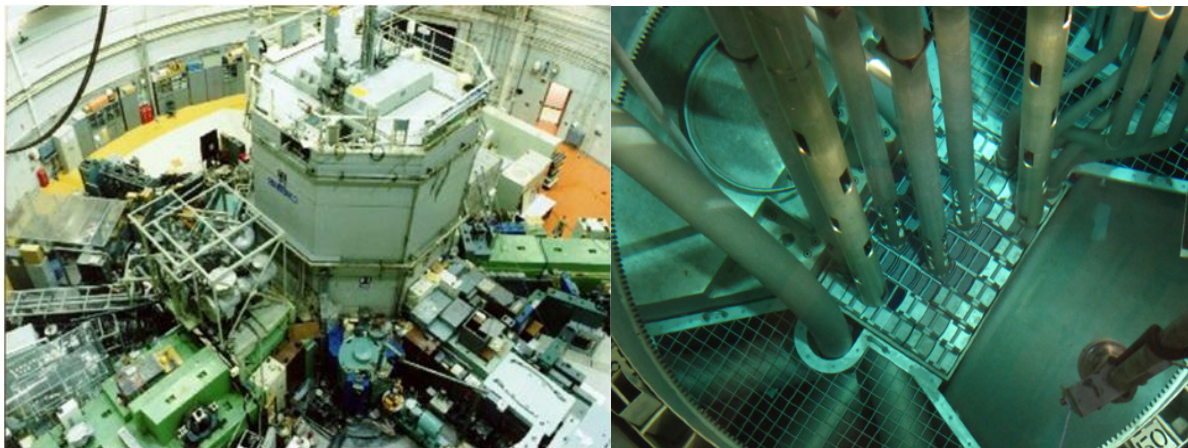

Fig.1   KUR (left: Reactor Biological Shield and Beam Lines, right: Reactor Core)

3. Full Core Conversion Process

3.1 Safety Analysis, Fuel Design, Fabrication and Transportation

Safety analysis for the LEU conversion[2-4] were performed using SRAC[5] code system and MVP[6] Monte Carlo simulation code for the reactor physics characteristics, COOLOD[7], THYDE-W[8-9] and EUREKA-2[10] codes for the thermo-hydraulic characteristics in steady state and transient conditions.   The detail of the reactor physics design and analysis will be presented elsewhere in this conference[11].

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[1] The KUHFR project has been officially cancelled in 1991.

The LEU fuel element was designed so as to have the same geometrical configuration with the HEU fuel element (Fig. 2) . This is in order to eliminate the modification of control rod system, reflector element design and thermo-hydraulic capabilities. Major specifications of fuel elements are summarized in Table 1.
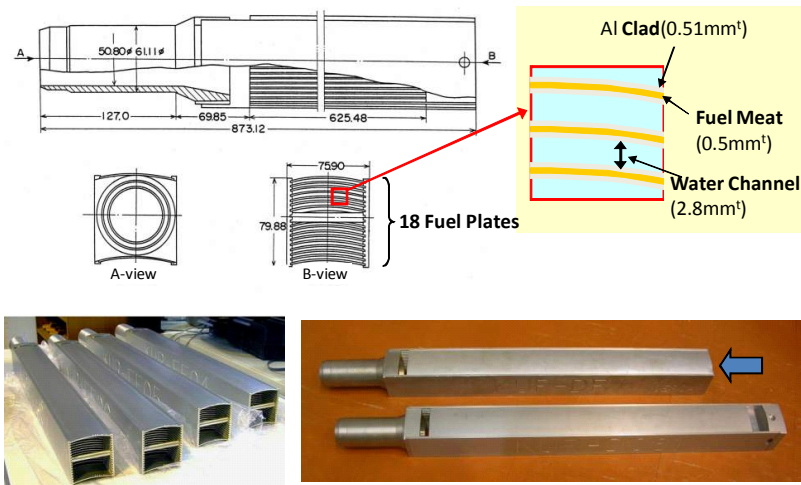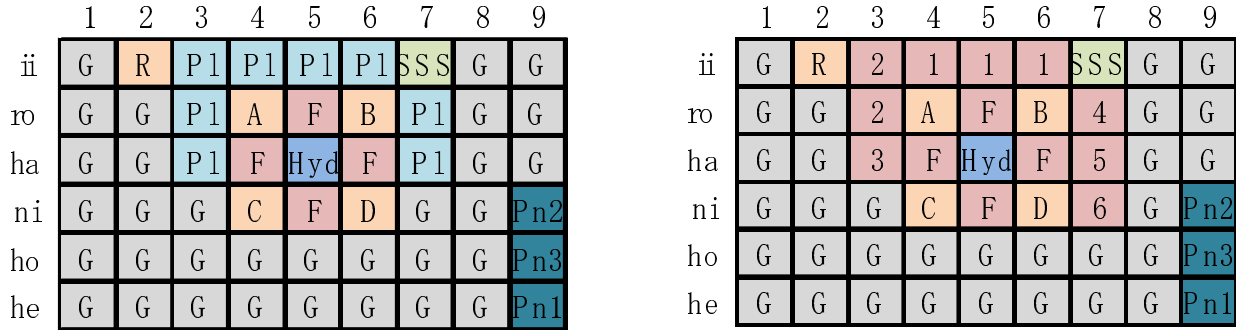

Fig.2   KUR Fuel Element

Table 1   Major Fuel Specifications

Design Data	LEU	HEU	Comments
Fuel Meat	U_3Si_2-Al	U-Al	
Enrichment (wt%U-235)	19.75%	93%	
Fuel Density	3.2gU/cc	0.58gU/cc	
Meat Thickness	0.5mm	0.5mm	Unchanged
Clad Material	Al Alloy	Al Alloy	Unchanged
Clad Thickness	0.51mm	0.51mm	Unchanged
#Fuel Plate / Element	18 (standard) 9 (special)	18 (standard) 9 (special)	Unchanged
U-235 mass / Element	213g (standard) 107g (special)	180g (standard) 90g (special)	
Max. burnup (%U-235)	35%	25%	

The fuel elements were fabricated at CERCA, France in two campaigns. The JRF-90Y-950K type transport casks were used for transportation. The unexpected delay in the fuel transportation of approximately 11 months occurred due to the regulatory requirement from French authority, which was initiated by the increased risk of nuclear fuel material transportation. This resulted to an overall alteration of the transport route and carriers, which also required extensive efforts in administrative procedures including additional validation process for the transport cask licenses. The fuel elements were received on early March, 2010, and were authorized by the competent authority of Japan for initial loading and subsequent performance tests on March 8, 2010.

3.2 Fuel Loading and Approach to Criticality Experiment

The first fuel was loaded in the core on March 23, 2010, which met the milestone of conversion at FY2009. The initial core configuration consists of four standard fuel elements (F) and five special fuel elements for control rods (A,B,C,D and R), surrounded by graphite reflector and water plugs as shown in Fig. 3.

	1	2	3	4	5	6	7	8	9
ii	G	R	P1	P1	P1	P1	SSS	G	G
ro	G	G	P1	A	F	B	P1	G	G
ha	G	G	P1	F	Hyd	F	P1	G	G
ni	G	G	G	C	F	D	G	G	Pn2
ho	G	G	G	G	G	G	G	G	Pn3
he	G	G	G	G	G	G	G	G	Pn1

	1	2	3	4	5	6	7	8	9
ii	G	R	2	1	1	1	SSS	G	G
ro	G	G	2	A	F	B	4	G	G
ha	G	G	3	F	Hyd	F	5	G	G
ni	G	G	G	C	F	D	6	G	Pn2
ho	G	G	G	G	G	G	G	G	Pn3
he	G	G	G	G	G	G	G	G	Pn1

Legend;
F: Fuel element for initial core loading, A-D: Special fuel element for control rods, R: Special fuel element for regulating rod, G: Graphite reflector, P: Water plug, 1-6 : Fuel elements loaded at approach to criticality steps 1-6
Hyd; Hydraulic Conveyor Irradiation Tube, SSS: Controlled irradiation port, Pn1-Pn3: Pneumatic tube

Fig.3   Initial Core Configuration (left) and Fuel Loading Sequence at Approach to Criticality Experiment(right)

A standard approach to criticality was performed by substituting the water plug and graphite elements with fuel elements in six loading steps as also shown in Fig. 3 and predicting the criticality at each loading step using inverse count rate plots. The fuel addition scheme was determined based on the full core MVP calculation, which indicated that the core would be critical after adding 9 standard fuel elements. During the approach to criticality , inverse count rate data obtained from fission chamber and BF$_3$ detector responses were plotted against the number of fuel elements loaded as shown in Fig. 4. The initial criticality was achieved after adding 9 standard fuel elements, as predicted by calculation, on April 15, 2010. Excess reactivity of the core was measured using the positive period method to be 0.38%Δk/k, which also showed an excellent agreement with the prediction by MVP full core calculation.
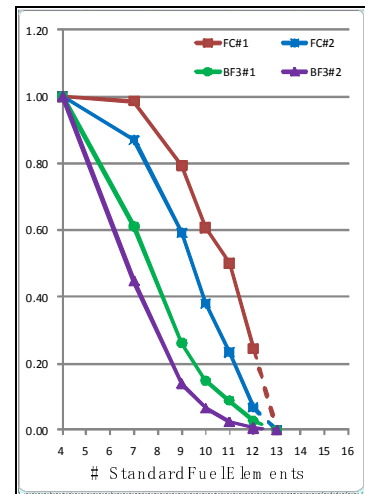
Fig. 4   Inverse Count Rate Curves

3.3 Operational Core and Reactor Physics Experiments

The operational core (Fig. 5) was constructed by substituting four graphite reflector elements with standard fuel elements at positions "Ni-3", "Ho-4", "Ho-5" and "Ho-6" so as to gain sufficient

excess reactivity for the full power operation.   A calorimetric calibration of the nuclear instrumentation was performed in order to verify the accuracy of the reactor power readings. After the completion of the calibration, control rod worth, temperature coefficient, neutron flux distribution and kinetic parameter measurement was performed as described in the following subsections.

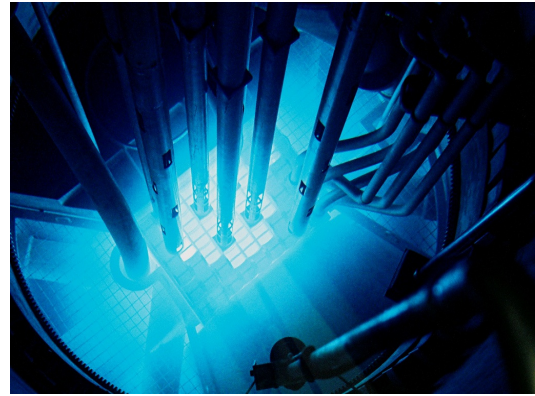|	1	2	3	4	5	6	7	8	9
ii	G	R	F	F	F	F	S S S	G	G
ro	G	P1	F	A	F	B	F	G	P1*
ha	G	P1	F	F	Hyd	F	F	G	G
ni	G	G	F	C	F	D	F	G	Pn2
ho	G	G	G	F	F	F	G	G	Pn3
he	G	G	G	G	G	G	G	G	Pn1

Fig.5   Operational Core Configuration and Core at Full Power Operation
(see Fig. 3 for legend)

3.3.1 Control Rod Worth Measurement

Control rod reactivity worth of the four shim rods (A,B,C and D rods) and regulating rod (R rod) was measured using positive period method and compensation method. The differential and integral rod worth is shown in Fig. 6.   The measured total rod worth of the control rods, as well as the shutdown margin of the core were confirmed to satisfy the safety limitations.
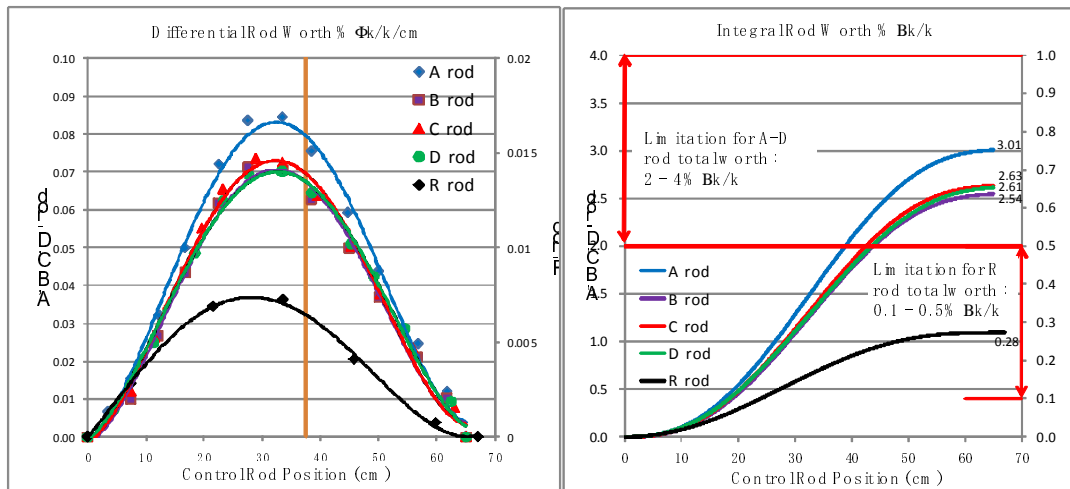

Fig.6 Results of Control Rod Calibration Experiment

3.3.2 Neutron Flux Distribution Measurement

Thermal neutron flux was measured by activation method using gold (Au) foils.   Six pairs of bare and Cd-covered Au foils (3mm diameter, 0.05mm thickness) were attached on aluminum foil holder blades, which were inserted in the central water channel of the fuel elements.   An example of the measured thermal neutron flux distribution along the vertical axis is shown in Fig. 7.   Compared to the HEU core, the thermal neutron flux at core fuel region was found to be reduced by about 20% and 10% at core fuel and central irradiation port.
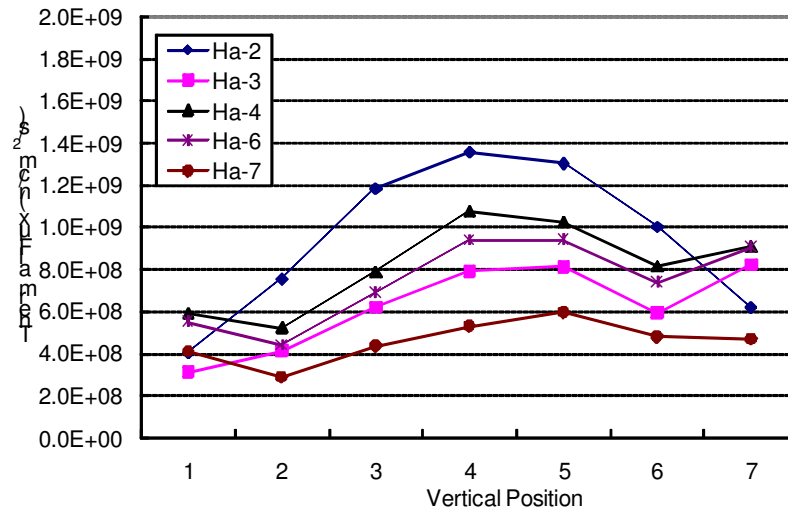

Fig. 7 Thermal Neutron Flux Distribution

3.3.3 Temperature Coefficient Measurement

The temperature reactivity coefficient of the core was measured at the reactor power of 20W by Joule heating of the moderator through primary pump operation. The reactivity change due to moderator temperature change was monitored by the critical control rod position.   The temperature reactivity coefficient was measured to be -6.0×10^{-3} +/- 0.3×10^{-3} %Δk/k/degree at moderator temperature of 30 degrees Celsius, which satisfied the safety limitation.

3.3.4 Kinetic Parameter Measurement

Reactor noise measurement was performed by collecting the time series data from the neutron detectors. The accumulated data was then analyzed using Feynman- α method to obtain the prompt neutron decay constant α.   The α–value at the critical state was determined to be 95.4 s^{-1}, which showed good agreement with the calculated value used in the safety analysis.

3.4 Operating Experience

The operational core was authorized by the competent authority for full-power operation on May 26, 2010.   The reactor is fully operating since May 28, 2010.   The current operating cycle is about 51

hours (Tuesday to Thursday) nominal, with 45 hours of 1MW operation for general use , followed by 6 hours of 5MW operation for neutron capture therapy irradiation.

For the fuel management of the converted LEU core, a new fuel depletion management system was constructed based on the core burnup simulation capability (COREBN) of SRAC code system. The performance of the system is currently under verification using the actual operation data.

A typical reactivity change within one cycle is shown in Fig. 8. The total reactivity was obtained from the control rod position change. Xenon reactivity was deduced by subtracting the reactivity caused by temperature change from the total reactivity, and was found to be in good agreement with the calculated reactivity by the new management system.

The burnup reactivity swing for the first 11 cycles are shown in Fig. 9, together with the prediction by the new management system. The predicted excess reactivity showed good agreement with the actual data for the very first stages of fuel depletion.
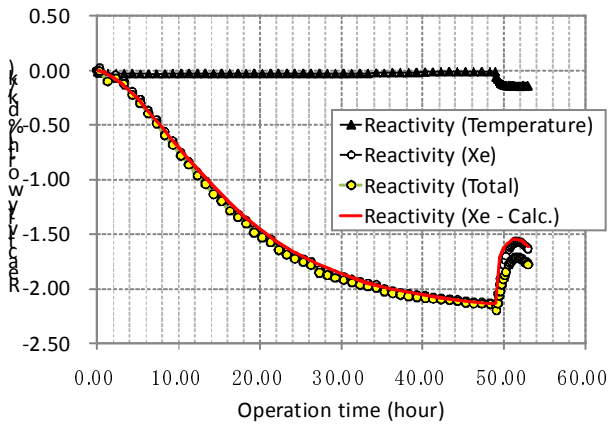

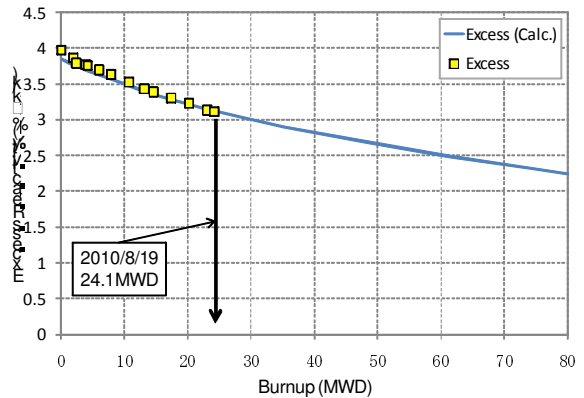Fig. 8 Example of Reactivity Change within one Cycle

Fig. 9 Burnup Reactivity (first 11 Cycles)

4. Concluding Remarks

The full conversion of KUR to LEU was finally and succesfully completed.   All measured safety parameters are demonstrated to be in compliance with the regulation limitations, which ensures the overall safety of the converted core.   The initial core and operational core test results demonstrated the validity of the neutronics characteristics design method used in the safety analysis and fuel management system.

5. Acknowledgements

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