Performance Evaluation of Converted and Upgraded PARR-1

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ABSTRACT

Pakistan Research Reactor-1 (PARR-1), a swimming pool MTR type research reactor which attained full power of 5 MW in June, 1966, with 93% high enriched uranium (HEU) fuel was converted to <20% Low Enriched Uranium (LEU) fuel in October, 1991. The reactor power was also upgraded from 5 MW to 9 MW and then to 10 MW. Different critical and full power operational core configurations were assembled with the new fuel. The final equilibrium core was assembled with 27 standard fuel elements (SFE) and five control fuel elements (CFE) having a central flux trap facility for high neutron flux. Detailed neutronics and thermal-hydraulic design calculations were made for the core conversion programme. After achieving the initial criticality several critical and power experiments were performed on the new core for the verification of design data and to determine the nuclear performance of the reactor. A comparison of the measured and the calculated results was also made. The results of the characteristics tests indicate that the performance of the new reactor is within the design limits. In flux trap thermal neutron flux is about 2×10^{14} n.cm^{-2}. s^{-1} which is five times higher than the average neutron flux of the core. seven standard and two control fuel elements have achieved designed burnup of 35%. Their physical inspection predicts excellent condition.

INTRODUCTION

PARR-1, a swimming pool, MTR type reactor attained full power of 5 MW on June 22, 1966 with 93% Highly Enriched Uranium (HEU) fuel. The reactor is cooled and moderated by Light water. Light water and graphite act as the reflector. Since its commissioning, PARR-1 has been mainly utilized for studies in solid state physics and neutron diffraction, nuclear structures, fission physics, Neutron Activation Analysis (NAA) radioisotope production and training of scientists, engineers and technicians. The reactor was operated with HEU fuel for about 30,000 hours and produced about 93,000 MWh energy.

The reactor was shutdown in 1990 for core conversion to commercially available LEU fuel. During the process of core conversion the reactor power was also upgraded to 10MW to meet the demand of higher neutron flux and to compensate the penalty in neutron flux due to conversion from HEU to LEU fuel. Most of the reactor systems including primary and secondary heat transport system, were renovated and several additional systems were installed. IAEA also provided technical assistance for the completion of this project. PARR-1 went critical with 20% LEU fuel on October, 31, 1991 and attained the upgraded power level of 9 MW on May 7, 1992. The reactor power was raised to 10 MW in 1998 after enhancing the primary flow rate. Following is a list of major achievements in the core conversion and power upgradation programme of PARR-1.

i)   Detailed reactor design calculations for core neutronics, thermal hydraulics and accident analysis were completed.

ii)   Final safety analysis report was prepared.

iii)   A storage bay was constructed for the irradiated fuel/active components. The reactor was partially decommissioned and the active core components were transferred to the storage bay in February 1991.

iv)   A fuel transfer cask was designed and fabricated for the transfer of HEU fuel.

v)   Stainless steel lining of the reactor pool and holdup tank was completed in September 1991.

vi)   Extensive building repairs were done

vii)   Heat transport system was modified and upgraded in September 1991. New HVAC system was also installed.

viii)   A laboratory for reactor startup studies and critical, low power and full power tests, was established in September 1991.

ix)   The new LEU core was made critical in October 1991. Several critical core configurations were made for experimental measurements.

ix)   The first operational core was assembled in January, 1992 and low power tests were started.

xi)   Based on the results of critical and low power experiments the reactor power was raised in steps to 9 MW in May 1992.

xii)   Emergency core cooling system (ECCS) was   installed in March, 1992.

xiii)   The reactor power was raised to 10 MW in 1998.

xiv)   The reactor is operating routinely satisfactorily since July, 1992.

REACTOR PHYSICS CALCULATIONS

Detailed neutronic calculations were made using different fuel types, uranium loadings and fuel burnups. Initially, U_3O_8-Al fuel was considered with the uranium density of 3.1 g/cm^3 corresponding to 270 g of U-235 per fuel element. However, to achieve a design burnup of the fuel of 35% it was necessary to increase the uranium loading to about 3.3 g/cm^3 which could be obtained in silicide fuel. Based on these considerations the LEU fuel selected was 20% enriched U_3Si_2-Al. The design data of the LEU fuel is given below.

Fuel material	U_3Si_2-Al
Fuel Enrichment (% by weight)	19.99
U-235 Contents (gm):	
· Standard Fuel Element	290
· Control Fuel Element	164
· Fuel Plate	12.81
Density of Uranium in Fuel Meat (gm/cc)	3.32
Cladding Material	Aluminum
Fuel Element Dimensions:	
· Total Length (cm)	87.33
· Cross-Section(cm^2)	7.96 x 7.59
No. of Fuel Plates:	
· Standard Fuel Element	23

For the selected fuel, number densities of all the materials used in different core regions i.e. fuel, moderator, reflector, etc. were calculated. These number densities were utilized to generate group constants using reactor lattice code, WIMS-D [1]. Five energy groups were considered as shown in Table-I and WIMS-D was employed for the generation of cross-sections for different core regions. For group constant calculations the standard fuel element was divided into two regions, active and nonactive. For the case of control fuel element a super cell was considered which included the control rod, the guide plates and two fuel plates.

Table I: Energy groups for cross-section generation

Group Number	Energy Boundaries (MeV)
1	10 MeV - 0.821 MeV
2	821 keV - 5.530 keV
3	5.53 keV - 0.625 eV
4	0.625 eV - 0.140 eV
5	< 0.14 eV

CORE CALCULATIONS

To determine the effective multiplication factor, reactivity and power distribution in the core, 3-dimensional diffusion calculations were performed using the neutronic computer code CITATION [2]. Different LEU core configurations were analyzed, which included the compact critical core, the first full power operational core and the equilibrium core. The calculations predicted that the most compact (water reflected) critical core could be assembled with 12 standard fuel elements and 4 control fuel elements arranged in a 4x4 array. The first operational core was designed with 17 standard and 5 control fuel elements. When all control rods were fully withdrawn, the excess reactivity was calculated to be 6400 pcm. The calculated results of reactivity values and neutron fluxes for different core configurations will be discussed later while comparing with the experimental measurements.

EXPERIMENTAL MEASUREMENTS

Experimental measurements include criticality experiment, reactivity measurement of the system and neutron flux inside and outside the core.

CRITICALITY EXPERIMENT

The initial startup of the first LEU core was performed in such a way so as to achieve criticality in a controlled and safe manner. All the prerequisites of reactor commissioning were completed prior to actual fuel loading [3,4]. The first critical core was designed to be critical with 13 standard fuel elements and four control fuel elements. Criticality was achieved with all safety rods in 85% withdrawn position. The mass of U-235 in the first critical core was 4427 grams. The first

critical core configuration along with the detector locations is shown in Fig. 1. The sequence of initial fuel loading steps are indicated by numbers 1 to 13. The criticality experiment provided a good test for the validation of fuel design data, since criticality was obtained with the same critical mass as predicted by analytical calculations[5].

THERMAL COLUMN

	A	B	C	D	E	F
9		12	SR#2	8	9	BF$_3$
8		7	4	1	SR#3	
7		SR#1	3	2	5	BF$_3$
6		11	6	SR#4	10	
5		13	FC-B		FC-A	
4						

SR: Shim rod, FC: Fission Chamber, Standard Fuel Elements : 13,   Control Fuel Elements : 4

Fig. 1: First critical core configuration with ex-core detector locations

REACTIVITY MEASUREMENTS

After achieving first criticality, various other critical cores and full power configurations were assembled both in the open and stall ends. Detailed reactivity measurements on these cores were made. Full power and equilibrium core configurations are shown in Figs. 2 &3.

THERMAL COLUMN

	A	B	C	D	E	F
9	S-66	C-22	S-71	S-75	S-76	S-68
8	S-74	S-70	C-19	S-69	C-21	S-72
7	S-78	C-18	S-73	C-20	S-77	S-67
6	S-82	S-79	WB	S-80	WB	S-81
5			FC-B		FC-A	
4						

S-XX: Standard Fuel Element,   C-XX: Control Fuel Element,   WB: Water Box,   FC: Fission Chamber

Standard Fuel Elements : 17,   Control Fuel Elements : 5

Fig. 2: First full power core configuration
THERMAL COLUMN

	A	B	C	D	E	F
9	S-67	S-66	S-84	C-22	S-81	S-68
8	S-72	C-19	S-71	S-70	S-83	S-86
7	S-78	S-75	WB-3	S-80	C-21	S-74
6	S-88	C-18	S-76	S-82	S-79	S-85
5	S-90	S-87	S-92	C-20	S-89	S-91
4	GR	S-95	WB-2	S-94	S-93	GR
3	GR	GR	WB	GR	GR	GR
2		FC-B	GR	WB	WB	FC-A

S-XX: Standard Fuel Element,   C-XX: Control Fuel Element,   WB: Water Box,   FC: Fission Chamber

Standard Fuel Elements : 27,   Control Fuel Elements : 5

Fig. 3: Final Equilibrium core configuration 93-A

Experimental measurements were made for the determination of excess reactivity, control system worth and reactor shutdown margin (sdm). The final equilibrium core configuration No. 93-A was assembled with 27 SFE's and 5 CFE's having an excess reactivity of 5.17% Δk/k and shutdown margin 5.05% Δk/k .

VOID COEFFICIENT OF REACTIVITY

Teflon strips were used to create void in the central flux trap and in the reactor core. Because of its near equivalence to air in small volumes teflon is being used to measure void coefficient of reactivity in many reactors [6]. In the equilibrium core configuration no 91, void was simulated by inserting aluminum plates into the standard fuel elements. It has been confirmed by neutronic calculations that discrepancy between aluminum and void effect is small enough to be ignored [7]. Aluminum stringers of 0.5 mm thickness, 65 cm long and 5 cm width were lowered in the water gap of the fuel plates. The effect of reactivity was observed at different locations in the fuel elements.

At PARR-1 the measured average reactivity effect in the central flux trap due to void is positive which is about +0.01 %Δk/k/%void.. The hollow aluminum pipe has the same effect as that of teflon with a variation of < 10%. The void coefficient of reactivity in all fuel elements and central flux trap was added and the overall void coefficient of reactivity in the core became negative of value 0.247%Δk/k/%void.

NEUTRON FLUX MEASUREMENTS

Neutron flux measurements for each full power core were made by foil irradiation. To determine the axial and radial flux distributions in SFE's, perspex stringers, each carrying seven gold foils, were installed in one coolant channel of each fuel element. Thin gold foils of 50 microns thickness and 99.99% purity were utilized to minimize the errors of self-shielding and flux distortion. For some of the fuel elements, cadmium covered gold foils were also used to measure the thermal-to-fast flux ratio. The foils were irradiated at about 40 W power level for 30 minutes. After irradiation and proper cooling, relative and absolute foil activities were determined by gamma-ray spectroscopic analysis . The experimental counting system consisted of a high efficiency HPGe gamma-ray detector with proper shielding, and PC-based 8k channels multichannel analyzer. The energy and efficiency calibration of the detection system was done with the help of multi-energy calibrated sources. The measured average thermal neutron fluxes in each fuel element, extrapolated to 10 MW full power level, are presented in Fig. 4. The measured neutron fluxes in the core agreed with the calculated fluxes within a maximum difference of about 30% [8]. The three dimensional measured flux profiles for the full power core are shown in Fig. 5.

THERMAL COLUMN

9	3.305 $\times 10^{13}$	3.446 $\times 10^{13}$	4.046 $\times 10^{13}$	C-R	3.582 $\times 10^{13}$	2.217 $\times 10^{13}$
8	2.518 $\times 10^{13}$	C-R	6.154 $\times 10^{13}$	6.109 $\times 10^{13}$	5.148 $\times 10^{13}$	3.654 $\times 10^{13}$
7	3.289 $\times 10^{13}$	2.815 $\times 10^{13}$	2.164 $\times 10^{14}$	6.883 $\times 10^{13}$	C-R	4.492 $\times 10^{13}$
6	3.368 $\times 10^{13}$	C-R	7.193 $\times 10^{13}$	7.115 $\times 10^{13}$	6.585 $\times 10^{13}$	3.365 $\times 10^{13}$
5	3.403 $\times 10^{13}$	4.104 $\times 10^{13}$	5.649 $\times 10^{13}$	C-R	5.433 $\times 10^{13}$	3.504 $\times 10^{13}$
4	GR	3.237 $\times 10^{13}$	1.06×10^{14}	4.641 $\times 10^{13}$	3.655 $\times 10^{13}$	GR
3	GR	GR	5.48×10^{13}	GR	GR	GR
2		FC-B	GR	2.79×10^{13}	2.53×10^{13}	FC-A

A	B	C	D	E	F

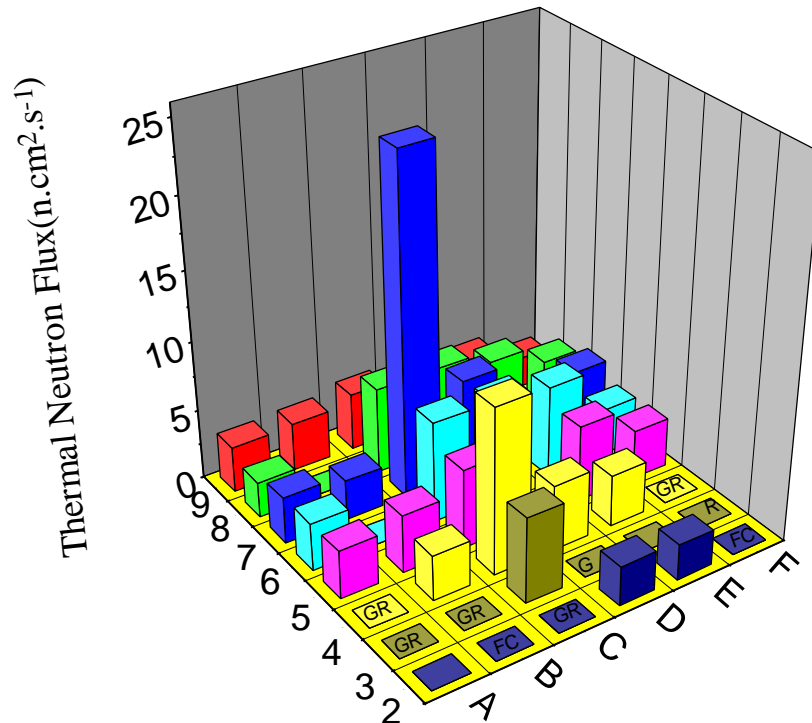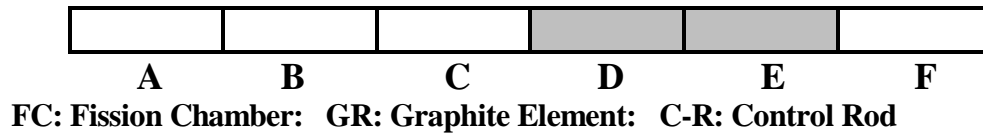**FC: Fission Chamber: GR: Graphite Element: C-R: Control Rod**

Fig. 5 Thermal Neutron Flux ($\times 10^{13}$) Inside The Core (Power = 10 MW)

Fig. 4: Thermal neutron flux inside the equilibrium core configuration 93-A

NEUTRON FLUX MEASUREMENTS IN EXPERIMENTAL FACILITIES

The layout of experimental facilities located around the core in the reactor pool is shown in Fig. 6. The measured values of thermal neutron fluxes in the experimental facilities are given in Table III. Neutron flux and cadmium ratio measurements in the thermal column were made in two different radial locations, 35 cm and 125 cm away from the core side (locations B and C respectively in Table II). It was observed that the cadmium ratio increased from 38 at location B to 273 at location C due to increased neutron thermalization.
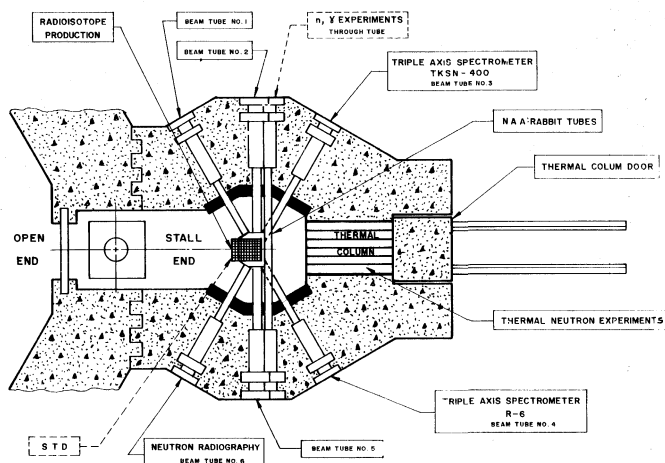
Fig. 6. Experimental facilities around PARR-1

Table II - Neutron Flux Measurements at Experimental Positions

Name	Thermal Neutron Flux ($n.cm^2 s^{-1}$)
RS # 1	3.27×10^{13}
RS # 2	3.50×10^{13}
RS # 3	2.07×10^{13}
Beam Tube No. 4	3.55×10^{13}
Beam Tube No.5	4.16×10^{13}
T. Column A	1.38×10^{8}
T. Column B	1.28×10^{10}
T. Column C	4.78×10^{11}

RADIATION DOSES AT UPGRADED POWER

As an essential requirement of reactor power upgradation, extensive radiation dose surveys were made on a routine basis, in and around the reactor building. The permanent area radiation monitoring channels in the new reactor were increased from six to ten. The maximum gamma radiation dose rate at the pool surface at full power was measured as 240 µSv/h, after about 30 hours of full power operation. However, the gamma radiation dose rate at the reactor bridge, which is about two meters above the water surface, was always within the permissible levels. A comparison of gamma radiation dose rates obtained in the old 5 MW reactor and new values at 9 MW, measured by the fission product monitor and the exhaust stack area monitor is presented in Fig. 7.
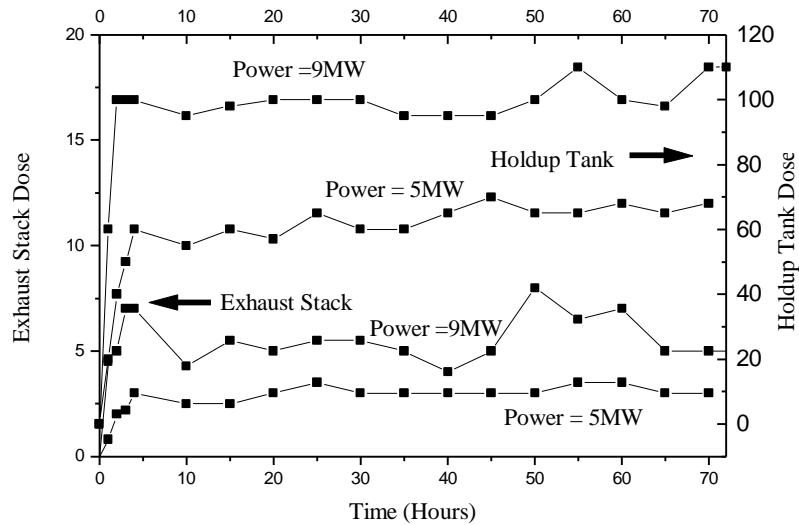
Fig. 7 Comparision of dose rates at 5 and 9 MW reactor power (µs/h)

BURNUP OF FUEL ELEMENTS

At the beginning of each new core the relative flux distribution has been measured regularly. This distribution has been measured after dividing the each fuel element in five regions axially. The U-235 contents in each fresh fuel elements are known, therefore burnup of U-235 in each region can be calculated after a certain period taking energy produced. Seven standard and two control fuel elements have achieved designed burnup of 35% satisfactorily without any problem and the fuel elements were found in excellent condition during physical inspection.

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