# BURN-UP DEPENDENT STEADY-STATE THERMAL HYDRAULIC ANALYSIS OF PAKISTAN RESEARCH REACTOR-1

**by**

***Atta MUHAMMAD*** [1], ***Masood IQBAL*** [2*], ***and Tayyab MAHMOOD*** [2]

[1] Department of Nuclear Engineering, PIEAS, Nilore, Islamabad, Pakistan
[2] Nuclear Engineering Division, PINSTECH, Nilore, Islamabad, Pakistan

Scientific paper
UDC: 621.039.517:66.011
DOI: 10.2298/NTRP1101045M

**The burn-up dependent steady-state thermal hydraulic analysis of Pakistan research reactor-1, reference operating core, has been carried out utilizing standard computer codes WIMS/D4, CITATION, and RELAP5/MOD3.4. Reactor codes WIMS/D4 and CITATION have been used for the calculations of neutronic parameters including peaking factors and power profiles at different burn-up considering a xenon free core and also the equilibrium xenon values. RELAP5/MOD3.4 code was utilized for the determination of peak fuel centerline, clad and coolant temperatures to ensure the safety of the reactor throughout the cycle. The calculations reveal that the reactor is safe and no nucleate boiling will commence at any part of the core throughout the cycle and that the safety margin increases with burnup as peaking factors decrease.**

*Key words: research reactor, fuel burn-up dependent cross-sections, nuclear peaking factors, power profile, thermal hydraulics, WIMSD/4, CITATION, RELAP5/MOD3.4*

## INTRODUCTION

Pakistan research reactor-1 (PARR-1) is basically a swimming pool type material testing research reactor (MTR), which is being utilized for the training of manpower, production of radioisotopes, and as a source of neutrons for basic and applied research. PARR-1 was originally a 5 MW research reactor using highly enriched uranium (>93% enriched) fuel. In the view of the proliferation concerns caused by the use of highly enriched uranium (HEU) and restricted supply of the HEU fuel to research reactors, PARR-1 was converted to operate on low enriched uranium (LEU) silicide fuel, containing 19.99% $^{235}$U. At the same time, in order to meet the high thermal flux requirements for isotopes production and other experimentation, the reactor output thermal power was initially upgraded to 9 MW; the further enhancement in the operating power level *i. e.* up to 10 MW was made. The MTR fuel element is a stack of straight fuel plates supported by side plates at both ends throughout its height. Fuel elements, control rods, graphite reflector elements, water boxes for irradiation of samples, and fission chambers with their guide tubes are assembled on a grid plate, having 54 holes arranged in 9 6 array, with a lattice pitch of 81.0 mm 77.1 mm [1]. Demineralized light water is used as a coolant and moderator. On one side the core is reflected by a thermal column, while the opposite side is reflected by graphite reflector elements and light water. The bottom side is reflected by a combination of aluminum and water. The rest of the three sides, *i. e.* the top and two lateral sides, are reflected by light water only. For reactivity control at PARR-1 five control rods (Ag-In-Cd alloy) are used. These control rods are employed for the startup, shutdown, and power leveling control. These control rods have enough strength that can scram the reactor in any unpredictable situation. The core configuration of PARR-1 reactor consists of 29 standard fuel elements (SFE) and 5 control fuel elements (CFE) as shown in fig. 1.

PARR-1 has two cooling modes:

– cooling by natural convection when the reactor power is <100 kW, and
– cooling by gravity driven downward flowing coolant when the reactor power is 100 kW.

The PARR-1 core provides numerous irradiation set-ups, which include water boxes, a graphite thermal column, pneumatic rabbit tubes, beam port tubes, a dry gamma cell, a bulk irradiation area, and a hot cell. The salient details of the PARR-1 core and its fuel elements are given in tab. 1 [1].

At PARR-1 the integrity of the fuel cladding is ensured by avoiding the nucleate boiling, *i. e.* cladding burnout throughout the cycle. The power profile al-

* Corresponding author; e-mail: masiqbal@hotmail.com

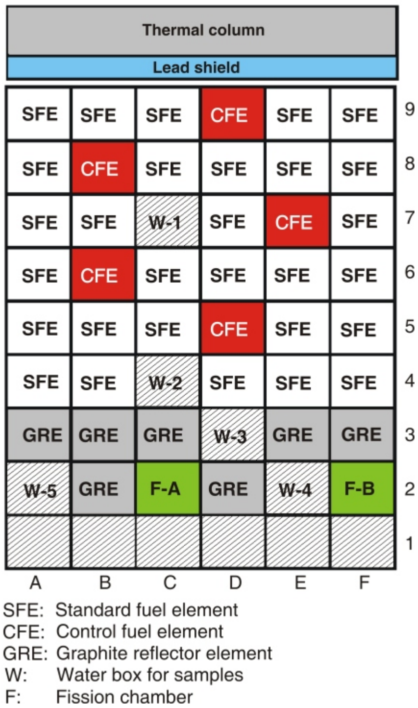


**Figure 1. Core configuration**

**Table 1. Main specifications of PARR-1**

| Reactor type | Swimming pool |
|---|---|
| Nominal core power [MW] | 10 |
| Lattice pitch [mm] | 81.0 77.11 |
| Fuel material and enrichment | $U_3Si_2$-Al (19.99% by wt) |
| Cladding material | Aluminum |
| Coolant/Moderator | Light water ($H_2O$) |
| Coolant flow rate [m$^3$h$^{-1}$] | 950 |
| Reflector | Light water and graphite |
| Fuel element description | Straight plate MTR type fuel element |
| $U^{235}$ contents per fuel plate [g] | 12.61 |
| Control rods | Oval shaped 5 rods |
| Composition of control rods | 80% Ag, 15% In, 5% Cd |
| Operational modes | Manual and automatic |
| Irradiation sites: | Neutron flux: |
| – Beam tubes | |
| Beam-tube #6 | 3.8 $^{13}$ cm$^{-2}$s$^{-1}$ |
| Beam tubes #2 and #5 | 1.1 10$^{14}$ cm$^{-2}$s$^{-1}$ |
| – Thermal column | Depends on the depth in thermal column |
| – Pneumatic rabbit | ~4.5 10$^{13}$ cm$^{-2}$s$^{-1}$ |

ways changes with the burn-up of the fuel as peaking factors are changed with the changes in the position of the control rods. Kinetic parameters also change with burn-up [2, 3]. The steady-state thermal hydraulic analysis of the PARR-1 equilibrium core at the beginning of the cycle with cold conditions has already been studied [4], as well as the simulations of reactivity transients [5-7]

In the present study the typical core of PARR-1 has been analyzed and power profile and peaking factors have been calculated at different burn-up steps during the life cycle of the core at critical position utilizing the standard computer codes WIMS/D4 [8] along with Borges [9] and CITATION [10]. The steady-state thermal hydraulic analysis of PARR-1 core was carried out at different burn-up utilizing RELAP5. Woodruff *et al.* [11] have used the code to compare the results for IAEA benchmark reactor and have shown good agreement with the PARET results. The analysis has been carried out at the beginning of the cycle (BOC) for 14 full power day and 35.26 full power day burn-up of PARR-1 core.

## METHODOLOGY

### Neutronic analysis

MTR_PC26 package was used for the generation of fuel burn-up dependent microscopic cross-sections for different regions of the core. This package uses WIMS/D4 code for this purpose along with BORGES. The WIMS/D4 code generates the cross-sections while BORGES reads the output of WIMS/D4 code and writes it in the format required for multidimensional diffusion theory based code, CITATION input. WIMS/D4 code uses its own 69 group library and solves the neutron transport equation in one dimension with reflective boundary conditions. At the beginning of the cycle, the burn-up of each fuel element was taken from the reactor operation group (ROG), incorporated by taking the burn-up option of WIMS/D4 to calculate the number density of fuel and fission products in the fuel region. Microscopic cross-sections were generated for the fuel region, structure region, water reflector, thermal column graphite, thermal column lead, graphite elements, control absorber, control follower, irradiation sites and fission chambers, and end caps of the fuel elements. All calculations were performed in ten energy groups [12] shown in tab. 2. For finding the power profiles and peaking factors at the corresponding burn-up, the reactor core (fig. 1) was modeled in XYZ geometry of CITATION. The total core active length of 60 cm was divided into five axial planes. The burn-up of every fuel element at each of the five axial planes was taken and accommodated in the code input. The control rods were fully modeled and at each burn-up step the criticality positions were found. Power profiles and peaking factors at the beginning of the cycle (BOC) for 14 full power days and 35.26 full power days burn-up were calculated. Burn-up through CITATION gives number densities at its output with xenon in the equilibrium level. Sometimes the reactor remains shut down for many days and xenon decays to about zero level. For the xenon free core, the number densities were taken from CITATION output at the corresponding burn-up and the

**Table 2. Ten energy group structure for generation of cross-sections**

| Group number | Energy intervals [ev] |
|---|---|
| 1 | 10 $10^6$ to 0.821 $10^6$ |
| 2 | 0.821 $10^6$ to 0.3025 10 |
| 3 | 0.3025 $10^6$ to 0.183 $10^6$ |
| 4 | 0.183 $10^6$ to 367.262 |
| 5 | 367.262 to 1.15 |
| 6 | 1.15 to 0.972 |
| 7 | 0.972 to 0.625 |
| 8 | 0.625 to 0.14 |
| 9 | 0.14 to 0.05 |
| 10 | 0.05 |

xenon concentration was set to zero. Again CITATION was executed without using the burn-up option employing the above mentioned number densities with zero xenon inventory. In this way, peaking factors were calculated for the critical core with zero xenon inventory. These peaking factors were used for the steady-state thermal hydraulic analysis of the core.

## Thermal hydraulic analysis

Two channel model approach was adopted in the RELAP5/MOD3.4 code, *i. e.* the hottest plate with the associated flow channel and other being an average plate with the associated flow channel. The standard fuel element consists of 23 fuel plates while the control element consists of 13 fuel plates as shown in figs. 2 and 3. Thus overall 732 coolant channels are formed within the fuel region of the core. One of these channels with the maximum power density is considered as the hot channel and the rest of the core channels are taken as the average channel. Axial profiles, represented by 21 equidistance mesh points at each burn-up step calculated through CITATION were incorporated in the RELAP5/MOD3 [13] code for the steady-state thermal hydraulic analysis of the core. To account for the uncertainties, the engineering hot channel factor 1.584 was incorporated using the conservative multiplicative method [14] which accounts for the factor 1.2 for the coolant temperature rise due to manufacturing tolerances in the coolant channel spacing, the factor of 1.2 for the film temperature rise due to uncertainties in the heat transfer coefficient and inhomogeneities in $^{235}$U distribution and the factor of 1.1 for uncertainties in the calculated power distributions. Assumptions were made that 90% of the total fission energy was deposited in the section fuel, about 4% was produced in the moderator, about 1% was produced in the other reactor materials and the remaining 5% was carried away by neutronos [15]. All calculation have been done with the coolant inlet temperature of 38 °C and the inlet pressure of 1.712 $10^5$ Pa, which corresponds to the static height of water from the core top to a point of 15 cm below the normal level of the pool (low level set point) [14].

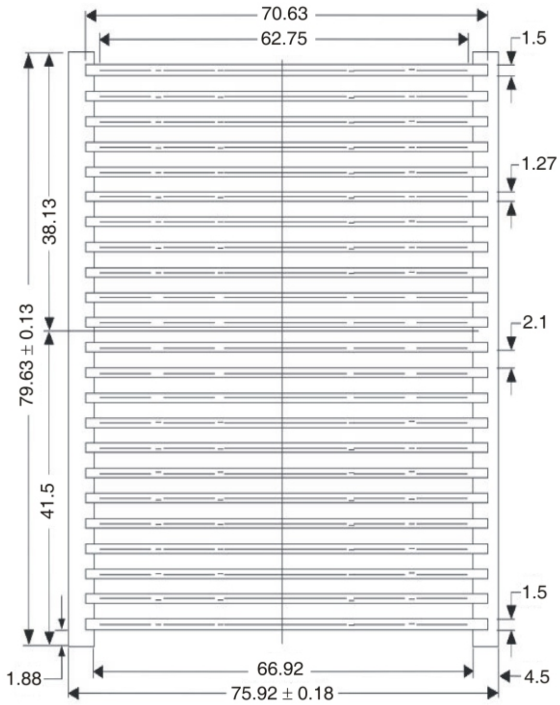


**Figure 2. LEU standard fuel element cross-section**

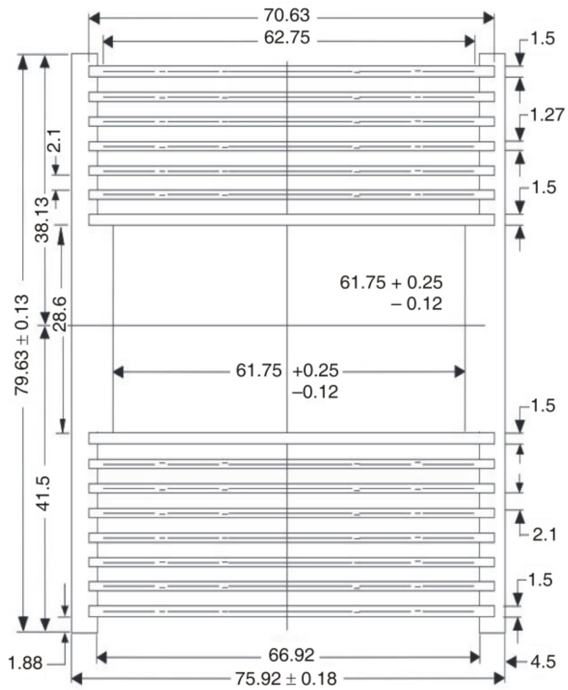


**Figure 3. LEU control fuel element cross-section**

**Table 3. Results of neutronic and thermal hydraulic analysis of a typical core of PARR-1**

| | Xenon free core | | | Equilibrium xenon | |
|---|---|---|---|---|---|
| | BOC | 14 full power days burn-up | 35.26 full power days burn-up | 14 full power days burn-up | 35.26 full power days burn-up |
| Radial peaking factor | 1.729 | 1.74 | 1.78 | 1.954 | 2.126 |
| Axial peaking factor | 1.798 | 1.75 | 1.67 | 1.457 | 1.237 |
| Engineering peaking factor | 1.584 | 1.584 | 1.584 | 1.584 | 1.584 |
| Total peaking factor | 4.92 | 4.82 | 4.71 | 4.51 | 4.17 |
| Steady-state coolant temperature rise across hot channel [°C] | 22.915 | 23.045 | 23.583 | 25.583 | 27.605 |
| Peak clad surface temperature [°C] | 102.32 | 100.54 | 99.69 | 98.27 | 95.95 |
| Peak fuel centerline temperature [°C] | 104.49 | 102.67 | 101.76 | 100.23 | 97.71 |

## RESULTS AND DISCUSSION

The calculated radial and axial peaking factors with zero/equilibrium xenon level are shown in tab. 3. At the beginning of the cycle, the axial peaking factor is larger than at the end of the cycle while the radial peaking factor increases towards the end of the cycle. The overall peaking factor decreases with the life cycle of the core for xenon free and also for equilibrium xenon. The steady-state thermal hydraulic analysis shows that the peak temperatures of fuel centerline and clad decrease with burn-up, because the axial peaking factor decreases with the life cycle of the core. The maximum coolant temperature of the core increases with burn-up due to the increase in the radial peaking factor. The total nuclear peaking factor at different states of the core is plotted against the active height of the fuel meat in fig. 4. The maximum value of nuclear peaking factor occurs at BOC due to the maximum insertion of control rods at this state. Axial flux depression at BOC places the peak value of nuclear peaking factor at 42.9 cm from the top of the core. The hot spot lies in the control fuel element at B-6 position of the core throughout the cycle; however, its axial position is changing with burn-up. The hottest plate is adjacent to the control follower. Thermal neutron flux at this location is higher due to the following reasons:

– the control fuel element at B-6 is the fresh fuel element,
– the control fuel element at B-6 is located close to the central water box set-up, and
– water is being used as the control follower in control fuel elements

Due to larger thermal neutron flux at this location, the peak power density lies in this fuel element. At the equilibrium xenon value, the peaking factor profile is nearly sinusoidal at the end of the cycle and the maximum value of the total peaking factor is decreased.

## CONCLUSION

The burn-up dependent steady-state thermal hydraulic analysis of PARR-1 typical core shows that the total nuclear peaking factor is higher at the beginning of the cycle; therefore, the maximum clad surface temperature also occurs at BOC which is 102.32 °C, while the saturation temperature at this point is 113.45 °C and the coolant bulk temperature is 53.33 °C, which shows that there is no chance of nucleate boiling throughout the burn-up cycle.

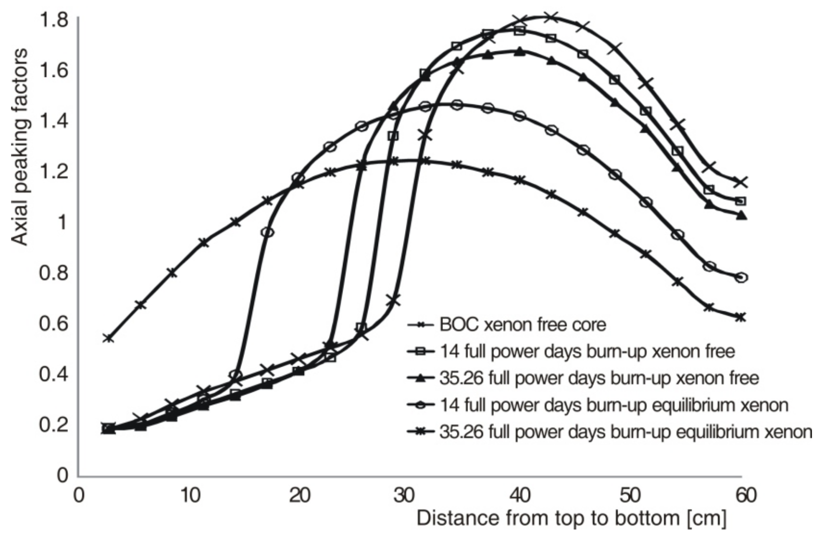


**Figure 4. Nuclear peaking factor (radial axial) along hot channel**

**ACKNOWLEDGEMENT**

Atta Muhammad gratefully acknowledges the financial support from the Higher Education Commission (HEC), Pakistan, for the Ph. D. (Indigenous) fellowship grant under PIN #074-1465-Eg4-022.

**REFERENCES**

[1] ***, FSAR, Final Safety Analysis Report of PARR-1, 2001
[2] Iqbal, M., Mahmood, T., Pervez, S., Flow of Kinetic Parameters in a Typical Swimming Pool Type Research Reactor, *Annals of Nuclear Energy, 35* (2008), 3, pp. 518-524
[3] Muhammad, A., Iqbal, M., Mahmood, T., Qadir, J., Calculation and Measurement of Kinetic Parameters of Pakistan Research Reactor-1 (PARR-1), *Annals of Nuclear Energy, 38* (2011), 1, pp. 44-48
[4] Bokhari, I, H., Mahmood, T., Chaudri, K. S., Steady-State Thermal Hydraulic Analysis of the Equilibrium Core of Pakistan Research Reactor-1, *Annals of Nuclear Energy, 34* (2007), 10, pp. 836-838
[5] Iqbal, M., Mirza, N, M., Mirza, S, M., Simulation of LEU-Transients under Reactivity Insertion Loss-of-Flow Conditions, *Nucl. Sci. Journal, 35* (1998), 2, pp. 81-90
[6] Mirza, A, M., Khanum, S., Mirza, N, M., Simulation of Reactivity Transients in Current MTRs, *Annals of Nuclear Energy, 25* (1998), 18, pp. 1465-1484
[7] Bokhari, I. H., Israr, M., Pervez, S., Analysis of Reactivity Induced Accidents at Pakistan Research Reactor-1, *Annals of Nuclear Energy, 29* (2002), 18, pp. 2225-2234
[8] Halsall, M. J., A Summary of WIMS/D4 Input Options, AEEW-M 1327, 1980
[9] Rubio, R. O., INVAP SE "BORGES v3.0, 1993"
[10] Fowler, T. B., Vondy, D. R., Cuningham, G. W., Nuclear Reactor Core Analyses Code CITATION, ORNL – TM – 2496 Rev. 2, Oak Ridge National Laboratory, Oak Ridge, Tenn., USA, 1971
[11] Woodruff, W. L., Hanan, N. A., Smith, R. S., Matos, J. E., A Comparison of the PARET/ANL and RELAP5/MOD3 Codes for the Analysis of IAEA Benchmark Transients, ANL, *Proceedings*, International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR), Republic of Korea, 1996
[12] Hamid, T., Reactor Kinetics Parameters as a Function of Fuel Burn-up, PIEAS-445, 1999
[13] ***, RELAP5/MOD3, Code Manual, Volume II: 'User's Guide and Input Requirements, NUREG/CR-5335, INEL-95/0174, Idaho National Engineering Laboratory, 1995
[14] Bokhari, I. H., Israr, M., Pervez, S., Thermal Hydraulic and Safety Analyses for Pakistan Research Reactor-1, *Proceedings*, 22nd International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR), Budapest, 1999
[15] El-Wakil, M. M., Nuclear Heat Transport, International Textbook Company, Scranton, Cal., USA, 1971

Received on December 8, 2010
Accepted on February 18, 2011

**Ата МУХАМАД, Масуд ИКБАЛ, Тајаб МАХМУД**

**ТЕРМО-ХИДРОДИНАМИЧНА АНАЛИЗА СТАБИЛНОГ СТАЊА ПАКИСТАНСКОГ ИСТРАЖИВАЧКОГ РЕАКТОРА-1 СА УРАЧУНАТИМ ИЗГАРАЊЕМ**

Коришћењем стандардних рачунарских програма WIMS/D4, CITATION и RELAP5/MOD3.4, извршена је анализа термо-хидродинамике стабилног стања стандардног језгра Пакистанског истраживачког реактора-1 у функцији од изгарања нуклеарног горива. Програми WIMS/D4 и CITATION употребљени су за прорачуне неутронских параметара, укључујући пикинг факторе и расподеле снаге језгра без ксенона и језгра са равнотежном концентрацијом ксенона за различита изгарања нуклеарног горива. Програм RELAP5/MOD3.4 коришћен је да се одреде температуре у центру горива, у кошуљици и хладиоцу, ради обезбеђења сигурности реактора током горивног циклуса. Прорачуни потврђују да је реактор сигуран и да током горивног циклуса нема започињања кључања у језгру, да се са изгарањем сигурност повећава, а пикинг фактори опадају.

*Кључне речи: истраживачки реактор, нуклеарни пресек, изгарање горива, нуклеарни пикинг фактор, расподела снаге, термохидраулика, WIMSD/4, CITATION, RELAP5/MOD3.4*