Estimation of Control Rods Worth for WWR-S Research Reactor Using WIMS-D4 and CITATION Codes

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ABSTRACT

An accurate estimation of the integral and differential worth of the control rods utilized in WWR-S research reactor is the main objective of the present study. The control rods were installed in nuclear reactors to provide precise and adjustable control of reactivity. WWR-S research reactor is controlled by nine control rods serving as shim, regulation and safety rods. Calculation of the reactivity worth of control rods in a reactor core constitutes an important factor of nuclear reactor design. In this study, the reactivity worth of the control rods is theoretically assessed using the neutronic codes; WIMS-D4 and CITATION for lattice cell and core simulation, in addition to the detailed Monte-Carlo simulation code MCNPX. The obtained results from WIMS-D4 and CITATION codes are compared with the corresponding measurements. A comparison with MCNPX evaluation of control rod worth for more validation is also presented.

Keywords: WWR-S/EK-10/WIMS-D4/CITATION/Worth

INTRODUCTION

Since the multiplication factor of the nuclear reactor core will change during operation due to fuel burn-up, because of fission product poisons and negative reactivity feedback effects which represented by temperature and power defects on reactivity. Therefore, at the beginning of each operational period, the reactors must be loaded with more fuel than that required to achieve criticality, to provide sufficient excess reactivity and to allow full power operation for a predetermined period of time. For WWR-S research reactors the critical mass when there are no experimental thimbles or absorbing rods in the reactor core is 3.2 kg U-235 (25 assemblies, 400 fuel elements). The maximum charge for which the excess reactivity could be compensated by inserting all the absorbing shim rods is 4.9 kg U-235 (38 assemblies, 608 fuel elements). The reserve reactivity under these conditions is ρ≈70 mk. The initial working charge chosen to allow for the loss of reactivity caused by poisoning, temperature coefficient of reactivity, and the introduction of materials for radio-isotope production to 32 assemblies (512 fuel elements, 4.1 kg U-235). The reserve reactivity of the unpoisoned reactor at the beginning of core life was ρ≈ 60 mk.

To compensate for this excess reactivity, it is necessary to introduce an amount of negative reactivity into the core to adjust and/or control it. This controlled reactivity can be used both to compensate for the excess reactivity necessary for long term core operation and also to adjust the reactor power level to bring reactor power, follow load demands, and shut the reactor down. This is achieved through the control rods and strong neutron absorbers that can be inserted into or withdrawn from the core. The assessment of the efficiency of a control rod to absorb excess reactivity in reactor core is very important, especially when major core changes are planned. It is also necessary to reassess the worth of a rod when the local burn-up around the rod is changing noticeably, as may be the case after refueling. The control rod worth is expressed by any of two terms; they are integral rod worth and differential rod worth. In integral rod worth, calculation is done based on reactivity change due to the insertion of control rod from top to bottom of the core. The effect of inserting each step part of control rod is calculated by measuring the reactivity change between the rod step withdrawal and its step insertion from top to that step part. In other method, differential rod worth, worth of each part of control rod is estimated based on reactivity change by inserting that part of control rod.
The theoretical calibration of the control rods as a function of the distance inserted in the reactor core is performed by the help of lattice cell code WIMS-D4 coupled with the core calculation code CITATION. A three-dimensional model of WWR-S research reactor core configuration is done using the Monte Carlo code MCNPX-2.7. It includes the details of all fuel elements, control rods, all irradiation channels and reactor pool. The neutron flux distributions within the core, control rods worth, core reactivity, and some kinetic parameters are estimated.

**REACTOR DESCRIPTION**

The WWR-S (water-coolant, water-moderator) research reactor \(^4\) was designed to produce average and maximum thermal neutron fluxes in the core are: \(\sim 1 \times 10^{13}\) and \(\sim 2 \times 10^{13}\) neutrons/cm\(^2\)/sec, respectively. The reactor load allows for an excess reactivity \(\rho = 60\) mk. Compensation of reactivity change and reactor control are carried out by nine control and safety rods. This control system that includes four shim control rods that serve to compensate excess reactivity, three rods act as safety rods, one fine rod and a regulatory rod used to automatically sustain the nominal power level. Each of three safety rods moves up sequently and all move down together. While each couple of manual rods (no. 4, 5 as manual-1 and no. 6, 7 manual-2) are moved up or down together. The automatic-control rod is made of steel, while the remaining rods are made of boron carbide. The rods are arranged crosswise in the core as shown in Fig.(1-a). Each rod consists of an aluminum tube (2 cm in outer diameter and 1.8 cm inner diameter) filled with an absorbing material as shown in Fig.(1-b). Each rod is placed with its guide tube (2.4 cm outer diameter and 2.1 cm inner diameter) between the fuel-element assemblies. The guide tubes for safety rods are filled with air while the others are filled with water. The effective length of control rods is 600 mm. The control rod positions are detected by position sensors.

In the WWR-S reactor \(^4\) the core is loaded with fuel elements of EK-10 type (UO\(_2\) in Mg enriched with 10% U-235). The fuel rods are encased in aluminum tubes having an external diameter of 10 mm and a wall thickness of 1.5 mm. The active length of the fuel elements is 500 mm. The fuel matrix is UO\(_2\)-Mg containing \(\sim 80\) g uranium per fuel rod. Each fuel assembly contains 16 fuel rods. There are four types of fuel assemblies, these include square, one cut corner, two cut corners and three cut corner as shown in Fig.(2).

![Fig. (1): (a) WWR-S core and control rods positions. (b) Vertical Cross section for a control rod](image-url)
The main objective of this study is using WIMS-D4 and CITATION codes to calculate the worth of control rods. The core configuration (Fig. 1a) is verified first through core simulation by using MCNPX code and WIMS-D4 /CITATION codes. In case WIMS-D4 /CITATION codes, this is done in two steps. Preparation of the cross sections data for different WWR-S regions, fuel assembly, control rod down, control rod up and reflector by WIMS-D4 code is done in the first step. In the second step, the reactor core is modeled by the CITATION-code.

The WIMS-D4 is a general lattice cell program which uses a transport theory to calculate flux as a function of energy and position in the cell (6). The fewgroup fluxes \( \Phi_{G,z}(G=\text{fewgroup index} , z= \text{zone index}, m=\text{material index}) \) are calculated in the main transport routine and averaged over the cell material volumes by formula (1).

\[
\Phi_{G,z}^T = \frac{\sum_m \Phi_{G,m}^T V_z(m)}{\sum_m V_z(m)} \tag{1}
\]

Where \( \Phi^T \) is the flux computed in the main transport routine and \( V \) is the volume (7).

For the sake of economy and saving machine time, WIMS-D4 is used first to calculate spectra for few spatial regions in the full number of energy groups of its library, and then uses these spectra to condense the basic cross-sections into few groups. Consequently, few groups’ calculation is carried out using much more detailed spatial representation. The resulting fluxes are then expanded using the spectra of the previous calculation, so that the reaction rates at each spatial point can be calculated in the library group structure. The code calculates cell-averaged diffusion coefficients, absorption, fission cross sections, the infinite multiplication factor (\( k_{\text{in}} \)) and effective multiplication factor \( k_{\text{eff}} \) (if
buckling is provided\(^5\)). For the present analysis, 69 energy groups neutron cross section library is used. The 69 energy groups consist of 14 fast, 13 resonance and 42 thermal groups. In the present work, the 69 energy groups in the data set library are collapsed to 12 energy groups’ cross section data set. The WIMS-D4 code solves the fuel burn up equations and fission products for any given power and then calculates the isotopic compositions and concentrations (g/cm\(^3\)) for important isotopes present in the reactor core. The burn-up equation (1) expressed by formula (2).

\[
\frac{dN_i(t)}{dt} = -\lambda_i N_i(t) - A_i N_i(t) + \sum_{k} \delta(i,j_1(k)) \alpha_{ki} C_k N_k(t) + \sum_{k} \delta(i,j_2(k)) \beta_{ki} \lambda_k N_k(t) + \sum_{k} \gamma_{ki} F_k N_k(t)
\]  

(2)

Where \(N_i\) = the number density of nuclide \(i\), 
\(N_k\) = the number density of nuclide \(k\), 
\(\lambda_i\) = decay constant for nuclide \(i\), 
\(A_i\) = absorption reaction rate for nuclide \(i\), 
\(C_k\) = capture reaction rates, 
\(\lambda_k\) = decay constant for nuclide \(k\), 
\(F_k\) = fission reaction rates 
\(\alpha_{ki}, \beta_{ki}\) = product fractions of isotope \(k\) 
\(\gamma_{ki}\) = yield of fission product \(i\) computed from the fission of nuclide \(k\) 
\(j_1(k), j_2(k)\) = identifiers of all products from isotope \(k\) and the delta functions \(\delta(i,j)\) indicate that the contribution occurs when \(i=j\).

Cross sections data library for different core regions is generated for fuel assemblies, control rod down, control rod up and reflector by cell models. In case of fuel square type assembly all pitches of fuel rods are equal to 1.7 cm as shown in Fig. (3-a). Therefore, one can represent it by no-cut corner model as shown in Fig. (4). This model consists of three regions the first region represents fuel with radius 0.35 cm, the second region for clad with radius 0.5 cm and the third region for coolant with equivalent radius equal to \(\sqrt{\frac{(pitch)^2}{\pi}} = \sqrt{\frac{1.7^2}{\pi}} = 0.959122 \text{ cm}\),

In case fuel assemblies of type one, two or three cut corners, the fuel rods pitches are different due to cut corner as shown in Fig. (3-b,c,d). Therefore, core fuel assemblies can be represented by two models one for no-cut corner section, where the pitches of fuel rods are equal to 1.7 cm and the other for cut corner section as shown in Fig. (4). In case cut corner section, the equivalent radius of coolant can be calculated as follows:

Cut corner section area = \(3.4^2 - 0.5 * 1.8^2 = 9.94 \text{ cm}^2\)

Area of cell for one fuel rod= \(\frac{9.94}{4} = 2.485 \text{ cm}^2\)

Equivalent radius = \(\sqrt{\frac{2.485}{\pi}} = 0.889382 \text{ cm}\)
There are two cell models to represent the control rod regions. In case of control rod inserted, this model consists of six regions as follows: absorber material region with radius 0.9 cm, clad with radius 1 cm, inner radius of tube guide 1.05 cm, outer radius of tube guide 1.2 cm, coolant and homogenized fuel–clad- coolant mixture region. The dimensions of coolant region and homogenized fuel–clad- coolant mixture region can be calculated as shown in Fig.(5-a). In triangle shaded (a b c)

\[
\frac{1}{2}ab = 1.8 \text{ cm}
\]

So \( \frac{1}{2}bc = \frac{1}{2} \times \sqrt{2} \times \overline{ab} = 1.272 \text{ cm} \)

This value represents half the length of the side of a square as shown in Fig.(5-a), its equivalent radius equal to \( \frac{2 \times 1.272}{\sqrt{2}} = 1.436192 \text{ cm} \). The homogenized fuel–clad- coolant mixture region can be represented by a square, the length of its side, equal to 6.8 cm, the equivalent radius equal to 3.836 cm.

The other case of control rod withdrawn, this model consists of four regions inner radius of tube guide 1.05 cm, outer radius of tube guide 1.2 cm, coolant and homogenized fuel–clad- coolant mixture region as shown Fig.(5-b).
To generate cross-section for water reflector, we homogenized fuel, clad and coolant as one region as shown Fig.(6).

CITATION code has been used for simulating the reactor core. Figure (7-a,b) shows the WWR-S core modeled by CITATION, where each number in Fig.(7-a) represents a cell region modeled by WIMS code.
CITATION code has been designed to solve problems that involve the finite-difference representation of the diffusion theory \(^8\). Explicitly, finite-difference approximation in space and time has implemented. The neutron flux eigenvalue problems are solved by direct iteration to determine the multiplication factor. The neutron flux, power distribution and burn-up can be determined for each fuel assembly \(^8\). The neutron diffusion equation expressed by formula (3).

\[
D \nabla^2 \phi - \sum_a \phi + S = \frac{1}{v} \frac{\partial \phi}{\partial t}
\]  

(3)

Where \(\phi\) is the neutron flux.

\(D\) is the diffusion coefficient \(D = \lambda_{tr} \frac{v}{3}\) where \(\lambda_{tr}\) the transport mean free path and \(v\) is the appropriate average neutron velocity

\(S\) is the production of thermal neutrons

The input file of CITATION code is created. This file defines three-dimensional geometry for the reactor core and using a previously generated macroscopic cross section library \(^8\).

The MCNPX-2.7 is a general-purpose Monte Carlo N–Particle code \(^9\) that can be used for neutron, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The reactor geometry and characteristics are carried out in MCNPX-2.7 code. Figure (8) shows the core configuration modeled by MCNPX-2.7.

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**Fig. (8):** Core configuration by MCNPX

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**RESULTS AND DISCUSSION**

In this paper, the \(k_{eff}\) for core loaded by 32 fuel assemblies (Fig. 9) is calculated assuming all fuel assemblies to be fresh. The calculations are done using WIMS-CITATION codes and MCNPX code. Table (1) gives the calculated \(k_{eff}\), which in a good agreement with the available values supplied in original design \(^{10}\).
Fig. (9): Core loading with 32 fuel assemblies

Table (1): \( k_{\text{eff}} \) for first core loading with 32 fresh fuel assembly

<table>
<thead>
<tr>
<th>Code</th>
<th>( k_{\text{eff}} )</th>
<th>Ref. (10)</th>
<th>Codes</th>
<th>WIMS/CITATION</th>
<th>MCNPX</th>
<th>Error (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>All control rods Up</td>
<td>1.06</td>
<td>1.05819</td>
<td>1.06144</td>
<td>0.31</td>
<td></td>
<td></td>
</tr>
<tr>
<td>All control rods down</td>
<td>-</td>
<td>0.90148</td>
<td>0.91635</td>
<td>1.64</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The estimation of the differential and integral control rod worth depends on the distance inserted in the reactor core. The change in reactivity that caused by control rod motion is referred as control rod worth. Integral control rod worth is the total reactivity worth of the rod at that particular degree of withdrawal. Differential control rod worth is the reactivity change per unit movement of the rod and is normally expressed in pcm/cm. The integral control rod worth is first calculated when the control rod is set at the upper level of the fuel, then control rod is gradually inserted down in the core to the final end position when the control rod is at the lower level of the fuel. Figure (10-a, b) illustrates the calculated differential and integral worth of all control rods from 1 to 9 and groups of them using WIMS-D4/CITATION codes. Also the worth of each control rod from 1 to 9 and groups of them is computed and compared with experimental results of LAZUKOV et al.\(^{(10)}\), as shown in Table (2).

Fig. (10): (a) Differential control rod worth  (b) Integral control rod worth
Table (2): The control rods worth calculated by WIMS-D4/CITATION codes and MCNP code compared with experimental results of LAZUKOV et al. (10).

<table>
<thead>
<tr>
<th>Rod</th>
<th>Material</th>
<th>Channel</th>
<th>Reactivity effectiveness of rod (Δk (pcm))</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>32 assemblies</td>
</tr>
<tr>
<td>Ref. [10]</td>
<td>WIMS-CITATION</td>
<td>MCNPX-2.7</td>
<td>Error(%)</td>
</tr>
<tr>
<td></td>
<td>Results</td>
<td>Results</td>
<td>Error(%)</td>
</tr>
<tr>
<td>ES</td>
<td>BC₄</td>
<td>1</td>
<td>2172.811</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>2184.029</td>
</tr>
<tr>
<td>ES</td>
<td>BC₄</td>
<td>2</td>
<td>2173.746</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>3.82</td>
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<tr>
<td>ES</td>
<td>BC₄</td>
<td>3</td>
<td>6858.287</td>
</tr>
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<td></td>
<td></td>
<td></td>
<td>2184.029</td>
</tr>
<tr>
<td>MC-1</td>
<td>BC₄</td>
<td>4</td>
<td>2470</td>
</tr>
<tr>
<td>MC-1</td>
<td>BC₄</td>
<td>5</td>
<td>1310</td>
</tr>
<tr>
<td>MC-1</td>
<td>BC₄</td>
<td>4,5</td>
<td>3370</td>
</tr>
<tr>
<td>MC-2</td>
<td>BC₄</td>
<td>6</td>
<td>1390</td>
</tr>
<tr>
<td>MC-2</td>
<td>BC₄</td>
<td>7</td>
<td>1390</td>
</tr>
<tr>
<td>MC-2</td>
<td>BC₄</td>
<td>6,7</td>
<td>2800</td>
</tr>
<tr>
<td>FC</td>
<td>BC₄</td>
<td>8</td>
<td>1390</td>
</tr>
<tr>
<td>AC</td>
<td>Steel</td>
<td>9</td>
<td>520</td>
</tr>
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</table>

The worth of control rods numbers 1, 2 and 3 (safety rods) are nearly the same because these rods are arranged in the reactor core symmetrically. The worth of control rod No. 4 has the largest value because it is located at the center point of the core where the flux has its peak value. Other shim control rods no. 5, 6 and 7 almost have the same worth to absorb neutrons in the nearby area of flux peak such that to reach the flatness radial flux pattern. The total worth of the safety rods is sufficient to bring the reactor to subcritical state in either the emergency situation or regular shutdown case.
The power level of the reactor depends on the macroscopic fission cross-section and the neutron flux. The effectiveness of a control rod largely depends upon the ratio of neutron flux at the location of the rod to the average neutron flux in the reactor. The effect of central control rod on neutron fluxes $\phi$ (thermal, epithermal, fast) is shown in Figure (11) declaring the flux distribution in case when all control rods are out of the core while central control rod is fully inserted. The peak of thermal flux at center disappeared when the central control rod inserted. The flux peak disappearance increased in fast flux more than that of epithermal more than thermal flux. i.e. the flux peak value has the least value in fast flux than epithermal than thermal.

CONCLUSIONS

In this study, WIMS-D4/CITATION codes have been used to estimate some reactor parameters such as $\phi$, $k_{\text{eff}}$ and control rods worth for WWR-S research reactor. The calculated $k_{\text{eff}}$ results using WIMS/CITATION codes are in good agreement with that done by using MCNPX code. The differential and integral worth are calculated for one control rod or group of control rods. The worth of one control rod or group of control rods is estimated by calculating the difference between reactivity of core when all control rods are withdrawn and when one control rod or a group of control rods are fully inserted. The calculated values are in a good agreement with the data provided in a study by LAZUKOV et al. The calculated results are in a good agreement with the referenced results. The 3-dimensional neutron flux distribution at different energy groups is obtained using WIMS/CITATION codes. The obtained results clearly show the effect of inserting central control rod in the core. The worth of central control rod is the largest because its position at the central core. The worth of scram rods is nearly the same, also manual control rods are nearly equal. The worth reactivity of the safety rods is enough to shut-down the reactor in an emergency situation.

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