Conceptual design of a high neutron flux research reactor core with low enriched uranium fuel and low plutonium production

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1. Introduction

For peaceful applications of nuclear energy, research reactors with capabilities of radioisotope production, material testing, neutron radiography, neutron activation analysis and more recently, Boron Neutron Capture Therapy (BNCT) are essential. Design of high neutron flux research reactors with low plutonium production while uses low enriched uranium as a fuel, with a small and compact core design for radioisotope production, fuel and material testing would be a good resource for research centers and universities. The first study for conceptual design of high neutron flux research reactor was done by Link from the Argonne National Laboratory. In this study, a research reactor with approximately \(5 \times 10^{15}\) n/cm\(^2\)-s thermal neutron flux was designed \[1\]. High Flux Isotope Reactor (HFIR) was designed to produce Trans plutonium isotope by using \(\text{Pu}_{242}\) which used HEU fuel (\(\text{U}_{3}\text{O}_8\)-Al with aluminum clad). Although the designed HFIR had high neutron flux, modifications needed to be performed on fuel to achieve a longer fuel cycle \[2\]. Olson from Argonne National Laboratory studied on very-high neutron flux research reactor design limitations such as power density, heat transfer and heat transport \[3\]. Sekimoto and Liem designed a high neutron flux research reactor with helium as coolant and a graphite moderator. They introduced a reactor with \(10^{15}\) n/cm\(^2\)-s neutron flux, maximum power density with the limitations of maximum system pressure and negative reactivity coefficient for fuel temperature and criticality \[4\]. Raina designed a multi-purpose high neutron flux research reactor with large irradiation volume in order to meet basic and applied research material irradiation \[5\]. Seo presented conceptual core design of a 20 MW research reactor using the HANARO Fuel Assembly (FA) which is high thermal neutron flux and multipurpose reactor with medium power and has sufficient space and expandability of the facility, also this designed research reactor uses low enriched uranium fuel \[6\]. Progresses in changing fuel from HEU (Uranium oxide) to LEU (U–10Mo) with increasing reactor power which could attain the same neutron flux performance for US high performance research reactors have been achieved in recent years \[7,11-13,15,17\]. Teruel and Rizwan-uddin presented a new and innovative LEU fuel research reactors design with neutron flux maximization for material tests, radiography, cancer therapy and radioisotope production purposes. They presented an
asymmetric cylindrical core in order to achieve high neutron flux [8]. Reactor engineer group from the Bhabha nuclear research center designed a high neutron flux research reactor for their needs. Reactor core characteristics are: 30 MW power, LEU as fuel, water as coolant and heavy water as moderator, high flux is provided in most places inside the core [9]. Seo and Cho from the

Fig. 1. Core design steps.

Fig. 2. a) Fuel pin, b) Circular fuel pin arrangement, c) FA with spacer grids.
Korean Advanced Institute of Science and Technology (KAIST) designed a high thermal neutron flux research reactor core which is a multi-purpose reactor. This designed reactor has a large core; therefore, many irradiation channels are allocated [10]. A low-power compact research reactor was designed by Vignolo. This reactor is a pool-type reactor with beryllium as moderator and cooled by natural circulation. Fuel plate can be added or removed from the core. This advantage makes the reactor core flexible and capable to compensate the reactivity decrease due to fuel consumption during the cycle by adding a block of beryllium as a moderator [14]. Wu from the University of Maryland designed a high neutron flux research reactor with $5 \times 10^{14}$ n/cm$^2$-s neutron flux for material irradiation and material science research [16]. Advanced test reactor (ATR) uses HEU as fuel. A study has been performed to replace HEU with LEU fuel and variations to the base core which is necessary for this conversion, has been discussed [18]. Based on international experience and homegrown technology in Iran, and considering the need for a high neutron flux research reactor for radioisotopes production, it is necessary to design a native reactor that satisfies all the design requirements as well as

<table>
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<th>Parameters</th>
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<tr>
<td>Fuel enrichment (%U$_{235}$ in UO$_2$)</td>
<td>3.7%</td>
<td>Inner/Outer diameter of fuel pin cladding (mm)</td>
<td>7.73/9.1</td>
</tr>
<tr>
<td>Fuel pellet height (mm)</td>
<td>11</td>
<td>Cladding</td>
<td>Zircoly + 1% Nb</td>
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<tr>
<td>Fuel pellet density (g/cm$^3$)</td>
<td>10.24</td>
<td>Thickness of the fuel pin clad (mm)</td>
<td>0.685</td>
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<tr>
<td>Fuel pellet outside diameter/center hole diameter (mm)</td>
<td>7.57/1.5</td>
<td>Gap between fuel pellet and fuel pin clad (mm)</td>
<td>0.08</td>
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**Fig. 3.** a) $K_{inf}$ versus core lattice pitch, b) $K_{inf}$ versus coolant void percent for core lattice pitch = 105 mm.

![Fig. 3.](image)

**Fig. 4.** Different cells of the designed research reactor core for generation of macroscopic cross sections in WIMS.

![Fig. 4.](image)
the international constraints as mentioned in the JCPOA. High neutron flux and low plutonium production (less than 1 Kg) at the end of cycle with plutonium grade less than 90% while using LEU fuel, small and compact core design with fuel temperature as low as possible are the main design objectives of this reactor. In this study, based on our experiences and incorporating the mentioned constructions, a high neutron flux research reactor with low plutonium production is designed.

2. Materials and methods

For designing our high neutron flux research reactor with low plutonium production, the following steps have been taken (Fig. 1):

According to step 1 of Fig. 1, we selected our fuel pins from WWER reactor fuel pin type (Fig. 2a) which has been experienced in Bushehr nuclear power plant. 18 fuel pins are arranged to form a FA (Fig. 2b). Fuel pellet and fuel pin characteristics are described in Table 1.

Fuel pin and circular fuel pin arrangement with spacer grids are shown in Fig. 2:

Although light or heavy water could be used for both coolant and moderator, considering the FA configuration in present design (Fig. 2b), light water is selected as coolant and heavy water as moderator. For choosing core lattice pitch, a FA with its own moderator (Fig. 5, cell 1) is considered as a cell. By varying the

![Fig. 5. Core cell with 96 meshes.](image)

![Fig. 6. a) The designed core with CBC at the center, b) Core schematic diagram.](image)

![Fig. 7. Applied calculation flow chart for neutronic and thermal hydraulic calculations in the FORTRAN program.](image)
moderator volume around the FA, core lattice pitch could be evaluated by: \( R \) (radius of moderator) = \( L \) (Lattice pitch) \( \times 0.525037 \), therefore \( K_{\text{inf}} \) for different core lattice pitch is calculated.

\[
R = L \times 0.525037
\]

Fig. 3 a shows the trend of \( K_{\text{inf}} \) versus different core lattice pitch. As shown, core lattice pitch can be selected less than/equal to 240 mm which is inside the under-moderated zone.

In order to select the core lattice pitch, coolant void reactivity effect is considered. Coolant void reactivity should be negative for selected core lattice pitch. Negative coolant void reactivity is tested for different core lattice pitches (less than or equal to 240 mm). Calculation is repeated with different core lattice pitches from 240 mm to 105 mm and void percentages from 5% void to 95%. For core lattice pitches greater than 105 mm, coolant void reactivity was positive. 105 mm was the first data of core lattice pitch in which coolant void reactivity is negative. Fig. 3 b shows the trend of \( K_{\text{inf}} \) versus coolant void percentage for the core lattice pitch of 105 mm. Furthermore, moderator void reactivity effect is negative because of being in the under-moderated zone. By selecting fuel pin, FA, core lattice pitch and coolant/moderator type, rough estimates for number of FAs and their height as well as top/bottom and side reflector volume, a preliminary core configuration is obtained which by numerous iterations the final core design is achieved. For cell calculations using WIMS code [19], different cells inside the preliminary reactor core have been considered. By considering a Central Beam Channel (CBC) for irradiation at the core center and 4 spacer grids for each FA along the core axial, 5 main cell types have been considered as follows (Fig. 4):

Cell 1 is a FA surrounded by heavy water as its moderator and cell 2 is similar to cell 1 with spacer grids to generate their macroscopic cross sections in WIMS code. To generate the macroscopic cross sections of CBC, 6 surrounding FAs (cell 3), which affect the macroscopic cross sections of CBC, are considered. In order to obtain the macroscopic cross sections of the side reflector, calandria, downcomer and vessel, the core is scaled down to a FA cell and the contribution of top/bottom reflectors are evaluated. Macroscopic cross sections for all different cells have been obtained by WIMS-D5 code. These macroscopic cross sections are used in the CITATION code [20] in order to evaluate neutronic core parameters. For performing core calculation by CITATION code, each cell is divided into 96 meshes (Fig. 5).

Also, we need to determine the number of FAs which are arranged with the 105 mm lattice pitch to form the reactor core. At first, a rough number of FAs is estimated taking into consideration the reactor power and plutonium production limitation. In the following, by detailed calculation, we reach the exact number of FAs which form the reactor core. By considering the selected fuel pin (Fig. 2a) and FA (Fig. 2b) with 70 cm height and UO2 fuel (10.24 gr/cm3) with 3.3% enrichment, and performing a rough estimates with regard to plutonium production limitation (less than 1 Kg per a reactor cycle), 72 FAs are specified inside the core for the first run. Core calculations are performed for different core configurations such as different number of FAs (72 FA, 66 FA and 60 FA), different enrichment (less than 5% enrichment), different dimension of reflectors (side and top/bottom reflectors) which results in different calandria and reactor vessel radius. In order to perform fast and accurate calculations, neutronic codes (WIMS-D5 and CITATION) and thermal hydraulic code (COBRA-EN) [21,22] are coupled by a FORTRAN program. CITATION output gives power and flux distribution, which are used as COBRA-EN input. The new temperatures extracted from COBRA-EN output are used to execute WIMS-D5 code. We repeated these calculations according to the calculation flow chart (Fig. 7). Calculations were repeated until power
differences between two successive iterations became less than $5 \times 10^{-5}$. Performing calculations is continued for the next time steps until the end of reactor cycle ($K_{eff} < 1$). Finally, after performing all calculations, the designed reactor core has 60 FAs with active height of 70 cm, 3.7% enrichment and 20 cm top and bottom heavy water as reflector. Radius of calandria is 75 cm with 1.5 cm thickness. The vessel has inside diameter of 159 cm and outside diameter of 162 cm (Fig. 6a) and core schematic diagram is shown in Fig. 6b.

The calculation process is automated using a FORTARN program whose flow chart is depicted in Fig. 7.

Reactivity effects for coolant, moderator and fuel temperatures are investigated and presented in Fig. 8a, b. These reactivity coefficients have been reported for steady state condition at nominal temperature. These graphs have been drawn by least squares method. As it is shown in Fig. 8a, coolant and moderator temperature reactivity coefficients is $-3.169$ pcm/{°C} and $-4.6339$ pcm/{°C} respectively and fuel temperature reactivity coefficient is $-3.441$ pcm/{°C}.

The outcome of coupled neutronic-thermohydraulic calculations are the neutronic and thermohydraulic parameters. These parameters include:

i) $K_{eff}$ and excess reactivity,
ii) Power and flux distribution at any steps (5 or 10 or 15 days) during reactor cycle,
iii) Reactor life time cycle length,
iv) Minimum DNBR,
v) Radial and axial PPF
vi) Pressure and temperatures for all channels,
vii) Material concentration variations during reactor cycle,
viii) Plutonium production at the end of cycle with its grade as well as uranium consumption.

Radial and axial PPF are derived by Radial $PPF = (\text{Maximum FA power})/(\text{Average FA power})$ and Axial $PPF = (\text{Maximum axial power in hot fuel assembly (HFA)}/(\text{Average power in HFA})$. Variations on fuel enrichment, power, core height and number of FAs and fuel pin arrangement in a FA have been examined to decrease plutonium production and increase neutron flux as well as keeping the maximum fuel temperature under 1883 °C (maximum fuel temperature criterion [23]) and achieving the design objectives. Having high neutron flux and compact core and low plutonium production (less than 1 kg at the end of cycle) with plutonium grade less than 90% while LEU fuel (according to JCPOA) was our challenge in this research. Therefore, in order to achieve these goals, parameters such as number of FA, fuel pin arrangement in a FA, core height, reactor power, fuel enrichment (from natural uranium to 5% enrichment) and coolant and moderator materials have been optimized by carrying out numerous calculations. Ultimately, our designed core characteristics are shown in Table 2.

In order to verify these calculations and data, MCNP Monte Carlo code [24] is used to calculate power, flux distribution, $K_{eff}$, reactor cycle length and the amount of plutonium production at the end of cycle.

### 3. Results

In this section, the results of the conceptual design are presented. Calculations with deterministic and Monte Carlo codes result in 158 and 155.5 days of cycle lengths and 14260 and 13760 pcm of excess reactivities correspondingly. $K_{eff}$ versus cycle length by two methods are shown in Fig. 9.

Table 3 shows comparison between two calculation methods and their relative error:

### Table 3

<table>
<thead>
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<th>Parameters</th>
<th>Deterministic</th>
<th>MCNP Monte Carlo</th>
<th>Relative Error, %</th>
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<tr>
<td>Reactor power (MW)</td>
<td>22</td>
<td>22</td>
<td>0</td>
</tr>
<tr>
<td>$K_{eff}$ (at the BOC)</td>
<td>1.16631</td>
<td>1.15955 ± 0.00055</td>
<td>0.58</td>
</tr>
<tr>
<td>Reactor cycle length (days)</td>
<td>158</td>
<td>155.5</td>
<td>1.61</td>
</tr>
<tr>
<td>Maximum power of a FA at the BOC (MW)</td>
<td>0.466</td>
<td>0.481 ± 0.58%</td>
<td>-3.12</td>
</tr>
<tr>
<td>Maximum power of a FA at the EOC (MW)</td>
<td>0.445</td>
<td>0.483 ± 0.58%</td>
<td>-7.87</td>
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</tbody>
</table>

Fig. 10. a) Power distribution at the BOC, b) Power distribution at the EOC.
Relative error, \( \% = \left( \frac{\text{Deterministic data} - \text{Stochastic data}}{\text{Stochastic data}} \right) \times 100\% \) \( \text{(2)} \)

Fig. 10a, b show color map of power distribution at the BOC and EOC. Power of each cells by two methods (deterministic and stochastic) and their relative errors which is derived by \( \left( \frac{\text{power by deterministic} - \text{power by stochastic}}{\text{power by stochastic}} \right) \times 100\% \), is shown on the color map for the BOC and EOC.

Fig. 11 is a color map thermal flux ratio of EOC/BOC which the thermal flux data at the BOC and EOC is shown on it.

This designed reactor has a Central Beam Channel (CBC) located at the center of the core which can be used for irradiation purposes and isotopes production. The average thermal and fast neutron fluxes in CBC are evaluated and plotted in Fig. 12. Based on the WIMS code, Neutron energies less than 4 eV is considered as thermal, between 4 eV and 9118 eV as epi-thermal and between 9118 eV and 10^7 eV as fast neutron.

Axial and radial PPF for all steps are calculated which are shown in Fig. 13.

Material variations (consumption and production) during the reactor cycle, especially plutonium, have been calculated. Plutonium-239 production weight is less than 1 Kg (0.9487 Kg) and its grade is 83.64% which is acceptable according to international constrains. Fig. 14a, b show plutonium (239, 240, 241, and 242) production during the reactor cycle.

Minimum DNBR at reactor channels during cycle is always over 5 which is presented in Fig. 15.

Inlet coolant temperature to the core is 50 °C and average outlet coolant temperature is 75.7 °C. Hot channel during reactor cycle is recognized by core analysis. As it is obvious from the thermal hydraulic core analysis, the worst case of hot channel has occurred at the BOC. Because of core symmetry, channels No. 22, 23, 30, 32, 39, 40 (according to Fig. 6a) are the hot channels. Therefore, sub-channel analysis for channel 23 using WIMS-D5 and COBRA-EN codes is performed. WIMS-D5 evaluates the fuel pin power in order to input COBRA-EN code and performing subchannel analysis to calculate axial temperatures profile. We repeated these calculations until axial temperatures were converged. Hot channel has 30 subchannels (Fig. 16) with 19 fuel pins, including central rod as a fuel pin with zero power.

Upon calculations termination, results show that subchannels 1, 2, 3, 4, 5, and 6 (shown in Fig. 16) which are the same, have the maximum temperature among subchannels. Coolant inlet/outlet temperatures for hot subchannel is 50/89.15 °C and maximum temperature of clad outer surface for hot fuel rod (HFR) is 129.15 °C. Maximum fuel temperature is compared with the WWER reactor
fuel pin type criteria [23,25–27] and maximum outer surface of clad temperatures is compared with the International Steam Tables Properties of Water and Steam based on the Industrial Formulation IAPWS-IF97 [28]. Minimum DNBR of HFR is compared with “guidelines for preparing and reviewing applications for the licensing of non-power reactors, standard review plan and acceptance criteria” (NUREG-1537, part 2) [29] which are shown in Table 4, and are acceptable.

### Table 4

<table>
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<tr>
<th>Parameter</th>
<th>Value</th>
<th>Criteria</th>
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<tr>
<td>Maximum fuel temperature for HFR (°C)</td>
<td>1612.75</td>
<td>1883 [23]</td>
</tr>
<tr>
<td>Maximum temperature of clad outer surface for HFR (°C)</td>
<td>129.15</td>
<td>145.5 [28]</td>
</tr>
<tr>
<td>Minimum DNBR for HFR</td>
<td>3.51</td>
<td>&gt;2 [29]</td>
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### 4. Conclusion

In the present study, an attempt has been made to design a high neutron flux with low enriched uranium fuel (3.7% enriched UO₂) and low plutonium production which can be used for peaceful nuclear activities based on nuclear non-proliferation treaty concepts. Regarding the presented results, a summary of the designed reactor is as follows:

- P = 22 MW
- Core geometry: hexagonal lattice with pitch of 105 mm
- Total fuel loading = 334.74 Kg
- Maximum thermal flux at the BOC = 1.572 × 10¹⁴ n/cm²·s⁻¹
- Average thermal flux at the BOC = 1.0811 × 10¹⁴ n/cm²·s⁻¹
- Maximum thermal flux at the EOC = 1.754 × 10¹⁴ n/cm²·s⁻¹
- Average thermal flux at the EOC = 1.229 × 10¹⁴ n/cm²·s⁻¹
- Average thermal flux of CBC at the BOC = 1.754 × 10¹⁴ n/cm²·s⁻¹
- Average thermal flux of the CBC at the EOC = 1.229 × 10¹⁴ n/cm²·s⁻¹
- Radial PPF at the BOC = 1.272
- Radial PPF at the EOC = 1.213
- Axial PPF at the BOC = 1.315
- Axial PPF at the EOC = 1.224
- Minimum DNBRs at the BOC are 5.319 and 3.51 for the HFA channel and HFA subchannel analysis respectively.
- Minimum DNBR at the EOC = 5.876 for a HFA channel
- Plutonium production (Pu₂³⁹) at the EOC is 0.9487 Kg with 83.64% grade

Some advantages of this reactor are:

i) Using one type of fuel enrichment (which facilitates refueling),
ii) Uses LEU (3.7% enriched UO₂) for fuel,
iii) Low plutonium production at the end of cycle,
iv) Small and compact core design (which allows this reactor to be installed and operated in educational and research centers),
v) Works at a low temperature and low pressure which make it safe to operate.

In order to complete the design, control system and shielding need to be designed in the next step.

### Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2019.09.002.
References


