

Physics and Thermal Hydraulics Design of a Small Water Cooled Reactor Fuelled with Plutonium in Rock-Like Oxide (ROX) Form

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Abstract

This paper describes the Physics and Thermal Hydraulics areas of a design study for a small water-cooled reactor. The aim was to design a Pressurised Water Reactor (PWR) of maximum power 80 MWt, using a dispersed layout, capable of maximising primary natural circulation flow. The reactor fuel consists of plutonium contained in granular form within a Rock-like Oxide (ROX) pellet structure.

KEYWORDS: *Small PWR, ROX fuel, Physics design, Thermal hydraulics design.*

1. Introduction

The present paper describes a study of the use of plutonium fuel in ROX form in a dispersed plant with significant natural circulation capability. ROX fuel has been investigated by JAEI [1] and seems to offer promise as a disposal route for plutonium stocks. The physics design is described first, then the thermal hydraulics followed by conclusions on the effects of plutonium ROX fuel on the design. Other aspects of the design, including dynamics were described in a paper given at ICAPP06 [2].

2. Physics design

The scope and objectives of this project are primarily focused on investigating the physics associated with using PuO₂ based ROX fuel, within a pin type arrangement including the following design parameters: optimal core geometry, achievable core lifetime, acceptable core kinetics, adequate safety margin for safe shutdown, control rod worths, nature of material produced during burnup.

Plutonium has rather different properties as a reactor fuel compared to low enrichment uranium. Particular attention was placed on the absence of significant resonance absorption, and hence Doppler broadening, in non-fissile plutonium. This leads to a smaller temperature coefficient of reactivity for the fuel ($\alpha_{T \text{ fuel}}$). Plutonium also has a smaller delayed neutron fraction than uranium leading to a shorter reactor period with consequences for reactor control.

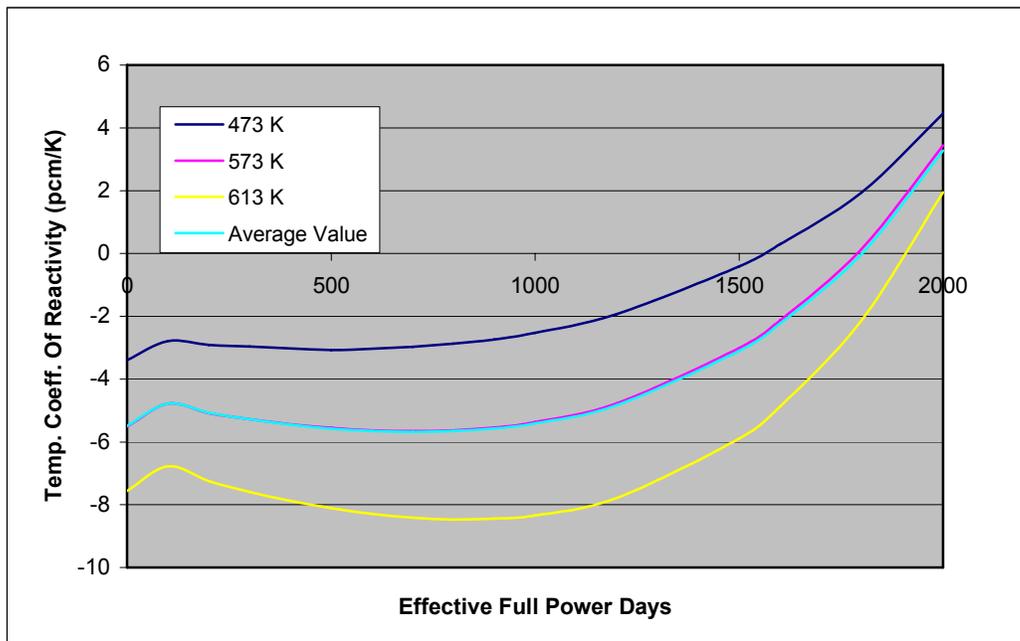
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2.1 Single fuel module calculations

The Serco Assurance ANSWERS codes, WIMS 9A [3] and MONK 8B [4] have been used for the reactor physics design aspects. The WIMS library based on JEF 2.2 was used. Calculations for an infinite lattice of fuel modules were made using the deterministic code WIMS. The design was refined to a 16×16 pin, closed square lattice with a cruciform control rod structure, detailed in table 1. Pin pitch was optimized to 1.4 cm and two Erbium poison pins were located in each quarter module to reduce the module PPF to 1.21. Erbium was selected as the burnable poison because it has resonances that closely match the fission resonances in ²³⁹Pu and ²⁴¹Pu.

Burn up calculations established that core life would be limited by reduction and eventual positive value of the temperature coefficient of reactivity for the moderator ($\alpha_{T \text{ mod}}$) rather than Xenon override considerations, as shown in Figure 1.

Figure 1: Temperature coefficient of reactivity for the moderator against core life in effective full power days at three different temperatures.



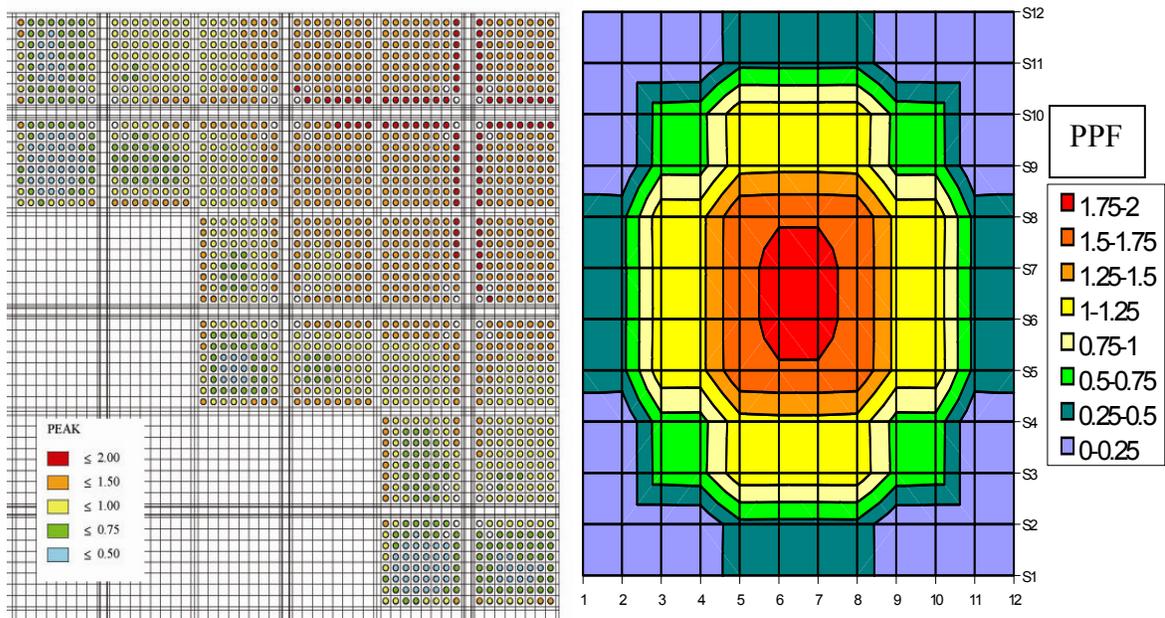
2.2 Two dimensional core calculations

Whole core calculations initially used a 2D model in the WIMS code. The core model consisted of 24 fuel modules arranged in a 6 x 6 grid surrounded by a water reflector (see Figure 2 showing a quarter core). The 2D model was used to investigate the effects of

distributed Erbium poison in the fuel. The final scheme had 3% Erbium in the four centre modules, 2% in the eight intermediate modules and 1% in the twelve outer modules. This design lowers the radial power peaking factor (PPF) from 2.04 to 1.76 at start of life (SOL).

These deterministic results were benchmarked against fully detailed 3D Monte-Carlo simulations made with the code MONK at a temperature of 300 K. Reasonable agreement was found between MONK and the 2D WIMS calculation for the multiplication factor ($k(\text{WIMS}) = 1.396$ and $k(\text{MONK}) = 1.392$) and exact agreement was found for the power peaking.

Figure 2: PPF from 2D WIMS calculation and 3D MONK calculation, distributed Erbium poisoning of 3% in 4 centre modules, 2% in 8 intermediate modules and 1% in 12 outer modules



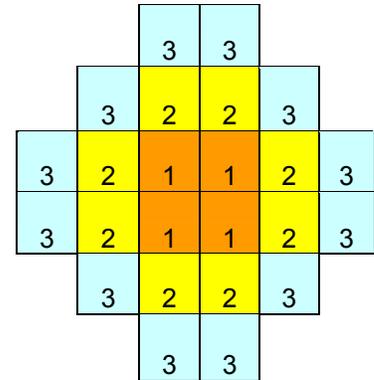
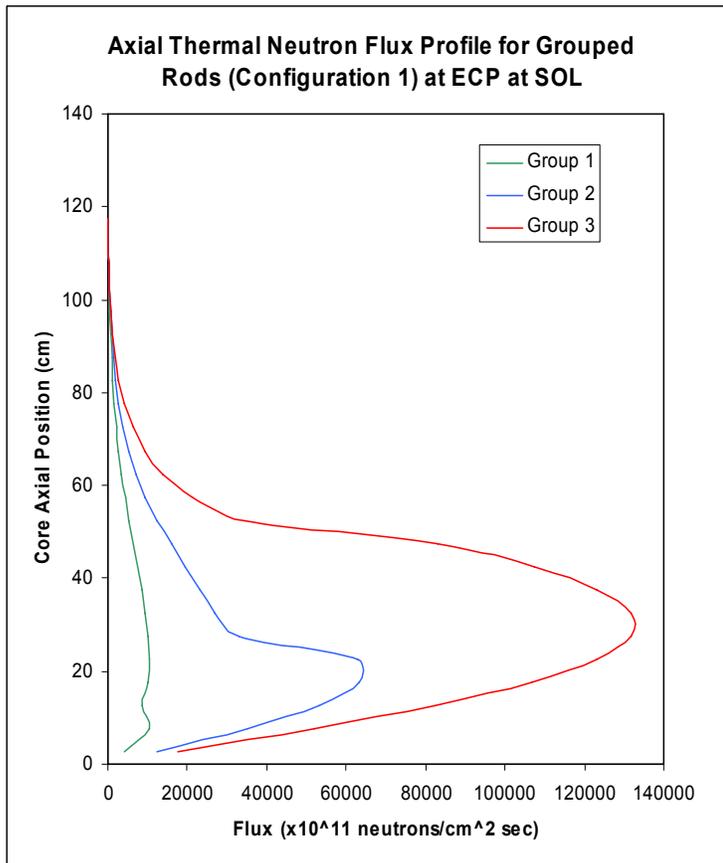
2.3 Three D whole core calculations

The effect of various control rod strategies was examined using the WIMS/SNAP diffusion code. SNAP requires the use of smeared geometry and condensation of energy groups. A two energy group scheme was chosen for these calculations.

One banked and two grouped schemes were calculated before the grouped scheme shown in Figure 3 was adopted for further study. This gave the lowest PPF and hence lowest maximum fuel temperature which materials studies indicated was a limiting constraint for ROX fuel.

Reactivity feed back coefficients were found from the whole core analysis at SOL and were more negative than the single module results but these were retained in the dynamics studies and to determine end of core life.

Figure 3: Axial flux profiles at SOL with grouped rods - SNAP calculation group 3 at 55 cm, group 2 at 30 cm and group 1 at 15 cm.



ECP = Estimated Critical Position

Table 1: Design details of core

Core Parameter	Value
Total Number of Fuel Pins in Core	5952
Total ROX Fuel Loading	1268 kg
Pu enrichment (²³⁹ Pu and ²⁴¹ Pu)	66.2%
Start of Life Pu loading	426 kg

Start of Life ²³⁹ Pu and ²⁴¹ Pu loading	282 kg
Number of Fuel Assemblies	24
Fuel Assembly Dimensions (lxbxh)	25.4 x 25.4cm x 1.2m
Number of Fuel Pins per Module	248
Fuel Cladding/Module Material	Zircaloy
Fuel Cladding Thickness	0.6 mm
Module Can Thickness	0.5 mm
Control Rod Guide Thickness	0.3 mm
Pin Pitch	1.4 cm
Fuel Pin Radius (excluding cladding)	0.35 cm
Fuel Pin Gas	He
Fuel Pin Gas Gap	0.5 mm
Fuel Pin Gas Pressure	40 bar
Burnable Poison Material (distributed and lumped)	Erbium
Poison Concentration (distributed)	1, 2 and 3% wt.
Core Width	1.6 m
Core Height	1.2 m
Number of Control Rods	24
Number of Control Rods in Bank 1	4
Number of Control Rods in Bank 2	8
Number of Control Rods in Bank 3	12
Control Rod Material	B ₄ C
Core Lifetime (Restricted by α_T)	1500 EFPD
Fuel Temperature Coefficient of Reactivity (α_{Tfuel})	-0.27 pcm/K (SOL) -0.019 pcm/K (EOL)
Coolant Temperature Coefficient of Reactivity ($\alpha_{Tcoolant}$)	-7.6 pcm/K (SOL) -5.6 pcm/K (EOL)
Average Fuel Temperature	1023 K
Average Clad Temperature	623 K
Average Coolant Temperature	573 K
Un-rodDED whole core k-snap (hot)	1.22639
RodDED whole core k-snap (hot)	0.75643

3. Thermal hydraulic design

The Thermal Hydraulic design aimed to remove the required maximum power using pumped flow and also to provide significant power removal using natural circulation. Codes employed were the sub-channel analysis code COBRA-EN [6] and the system code TRACPFQ [7].

3.1 Pumped Flow

For the design with pumped flow a radial PPF of 1.8 was taken (1.76 from the 2D physics studies) and an axial PPF of 1.4 with a chopped cosine shape assumed. As the final 3D physics studies were not available the radial power shape was assumed to be radial PPF of 1.8 in the 4 central modules, 1.2 in the 8 intermediate modules and 0.6 in the 12 outer modules. The thermal limits were a DNBR of 1.3 at 115% of nominal full power, maximum fuel temperature less than 1500 °C and max clad temperature less than 600 °C.

DNBR was evaluated by the COBRA-EN code with the EPRI CHF correlation selected. This correlation has recently been shown to be appropriate at the lower flow rates typical of the present design [5]. A core mass flow-rate of 750 kg/s was found to be sufficient to satisfy all the thermal limits with key parameters are given in Table 2.

Table 2 Key Thermal hydraulic parameters

Nominal Power	80.0 MW
Nominal Pressure	15.0 MPa
Nominal Core T _{in}	290 °C
Nominal Coolant Mass Flux	860 kgm ⁻² s ⁻¹
Height difference core to SG	5.0 m
No. of SG tubes	800
Length of SG tubes	4.0 m
Diameter of SG tube	15 mm
Tube Thickness	1.5 mm
Primary to sec. Temp. diff.	40.0 K

Steam generators were designed to transfer 40 MW at the nominal primary and secondary conditions with a limited tube bank height to allow large vertical separation of the SG and core within available space. The tube size and numbers were also adjusted to reduce flow losses and promote natural circulation.

3.2 Natural circulation flow and Loss of Coolant Accident calculation

The flow rate in the primary circuit at various powers was calculated using standard single phase equations for the head losses in pipes and area changes and the expression for driving head using thermal centres and temperature differences as in [8]. This was programmed into a spreadsheet. For a power of 19 MW the core flowrate was 178 kg/s.

This was also calculated by a TRACPFQ model with an RPV, 2 Steam Generators and connecting hot and cold legs shown in Figure 4. The heat transfer coefficient on the SG secondary side cells is calculated using Chen’s correlation [9]. There are 12 cells in the RPV component and 6 primary and 2 secondary cells in each SG. Table 3 shows natural circulation flowrates calculated by TRACPFQ, which indicates good agreement with the spreadsheet calculation between 15 and 21 MW power.

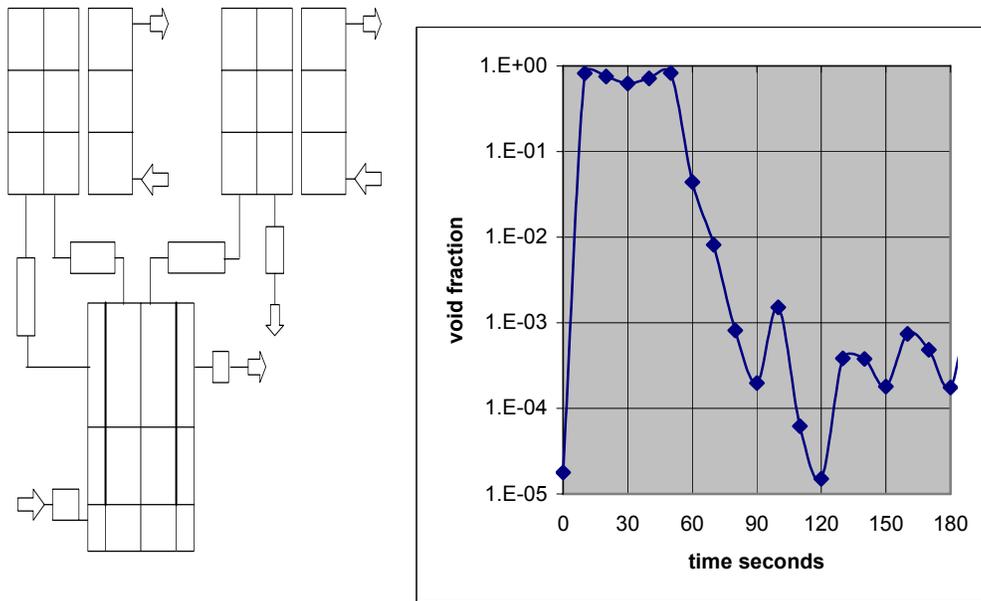
Table 3 Flow rates in natural circulation – TRACPFQ calculation

Power MW	8	9	12	15	18	21
Flow rate kg/s	133	138	152	163	175	185

The power available in natural circulation was assessed against thermal limits of min DNBR > 1.3 and no bulk boiling using a COBRA-EN model which included core regions of three different powers in which core flow was allowed to redistribute itself. Minimum DNBR is >2 for a power of 19 MW and this was taken as the allowable power in natural circulation.

Loss of Coolant Accident (LOCA) transients for a large break and a medium sized break were calculated using the TRACPFQ model with pressure of 0.15 MPa at the break and an additional pipe providing a constant injection flow rate to the vessel. Figure 4 shows the void fraction at the top of the vessel for a Large Break LOCA, showing that it empties in about 10s, starts to refill at 40s and has refilled at 120 s. The volume of water injected is 2.50 m³.

Figure 4 Cells in the TRACPFQ model and void fraction predicted at top of core



4.0 Conclusions

This investigation into the use of a plutonium based ROX fuel in a small reactor plant presents the following conclusions:

The larger fission energy yield, number of neutrons emitted post fission and Doppler broadening of Pu fuel compared to uranium fuel result in high PPF for this type of core. High PPF were found within the fuel pin, caused by resonance self shielding, across a module and across the whole core (radial and axial).

Temperature coefficients of reactivity ($\alpha_{T_{fuel}}$ and $\alpha_{T_{coolant}}$) are significantly smaller than those for a typical uranium fuelled civil core. This gives a smaller thermal feedback effect, presenting a problem for reactor control systems and for prevention of fuel melt from a Reactivity Insertion Accident.

The overall design characteristics of this Pu ROX reactor, as described here and elsewhere [2] suggest it is feasible provided a responsive control system to limit reactivity addition is included.

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