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Experimental study of neutronic parameters in Tehran research reactor mixed-core

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In this work, general characteristics of a typical mixed core, including HEU & LEU fuel is studied. The study is performed in the Tehran research reactor (TRR). In this study the neutronic parameters, reactivity feedback coefficients and kinetic parameters are investigated. The reference core designated for such study is the equilibrium core (No. 61) with an average bun-up of 27% & 36% for SFE's & CFE's, respectively. The MTR_PC package is used for neutronic analysis. In this research, experimental and computational results for the reference and mixed core are compared. Meantime, the obtained values for neutronic parameters are mostly below the adopted safety criteria and they are in good agreement with the experimental results. However β_{eff} and ℓ_p are a little bit higher in the mixed core with respect to the reference core, but in practice, these small changes will not cause substantial impacts on the dynamic behaviour of the reactor core. The absolute values of the fuel temperature, moderator density and void coefficients of reactivity, are less in the mixed core and only the moderator temperature coefficient is higher. The calculated values of power defect, based on the reactivity coefficients; in both core configurations are in good agreement with the experimental values.

1. Introduction

Tehran research reactor (TRR) achieved its first criticality using 90% enriched uranium, in 1967. The core conversion from HEU to LEU was carried out in 1992. Unlike other research reactors the TRR has not experienced the step by step mixed-core period, in which the previous HEU fuels were gradually replaced by the new LEU fuels. In fact all HEU were substituted by the LEUs at once. Since the most HEU burn up is 20%, then designing a mixed-core for TRR is a perfect solution for optimal utilization of this kind of fuels. Also considering that the LEU–SFE and CFE of TRR are approaching the range of permissible burn-up, using HEU–SFE and CFE can be a proper alternative to survive the TRR and overcoming the shortage of fuel. Therefore, the use of HEU–Fes in a mixed core becomes significant either economically or from the research point of view.

Feasibility study of using HEU–CFE in the reference, namely the core No. 51 has also been performed in our previous papers in which, neutronic and kinetic parameters of the reference and mixed cores of TRR has been investigated (Lashkari et al., 2012, 2013). This study showed that increasing the number of HEU–CFEs, will cause reduction of the shutdown margin as well as regulating rod (RR) worth. Meantime, the radial peaking factor increases, but all the neutronic parameters are kept far below the safety limits. The kinetic parameters, namely the β_{eff} decreases but the ℓ_p increases as the fuel burn-up proceed. Moreover, increasing the number of HEU–CFE in the reference core No. 51, the ℓ_p increases, but with little change in β_{eff}.

In this research, a new mixed core of TRR, in which 5 LEU_SFEs replaced with 5 HEU_SFEs in the reference core No. 61, has been studied both experimentally and computationally. MTR_PC package was used to calculate the neutronic parameters, kinetic parameters and reactivity feedback coefficients of the reference core. The methodology used to calculate the kinetic parameters is different from the one applied for neutronic and reactivity feedback coefficients (Lashkari et al., 2012, 2013). In this study the IAEAs' documents (IAEA, 1980; IAEA, 1992) and TRR amendment to the SAR[1] were used as a guide in order to perform the analysis.

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[1] AEOI, 1989. Tehran Research Reactor Amendment to the Safety Report, Tehran-Iran.

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Nomenclature

TRR	Tehran Research Reactor
LEU	Low Enriched Uranium
HEU	Highly Enriched Uranium
SFE	Standard Fuel Element
CFE	Control Fuel Element
LEU–CFE	Low Enriched Uranium–Control Fuel Element
HEU–CFE	Highly Enriched Uranium–Control Fuel Element
HEU–SFE	Highly Enriched Uranium–Standard Fuel Element
BOC	Begin Of Cycle
SAR	Safety Analysis Report
GR.B	Graphite Box
E.B	Empty Box
SR	Shim Safety Rod
RR	Regulating Rod
ρ	Reactivity
$\alpha_{T,f}$	Fuel temperature coefficient
α_m	Moderator temperature coefficient
$\alpha_{T,m}$	Moderator temperature only
$\alpha_{D,m}$	Moderator density
α_V	Void reactivity coefficient
β_{eff}	Effective delayed neutron fraction
ι	Prompt neutron life time
$\Delta\rho_{power}$	Power defect of reactivity
T_w (in)	Core inlet water temperature
T_w (out)	Core outlet water temperature
ΔT_{wm}	Mean water temperature difference
T_{fm}	Mean fuel temperature
ΔT_{fm}	Mean fuel temperature difference

Table 1
Specifications of TRR fuel assemblies.

Parameter	Fuel assembly type		
	HEU–SFE	LEU–CFE	LEU–SFE
Meat material	U_3O_8–Al	U_3O_8–Al	U–Al Alloy
Enrichment	20%	20%	93.15%
Number of fuel plates	19	14	16
No of outer dummy plates	0	0	2
Meat thickness	0.07 cm	0.07 cm	0.05
Cladding thickness	0.04 cm	0.04 cm	0.038
Water channel thickness	0.27 cm	0.27 cm	0.31
Meat width	6 cm	6 cm	6.1
Meat length	61.5 cm	61.5 cm	59.9
Side wall thickness	0.45 cm	0.45 cm	0.48
Total plate width (wall to wall dist.)	6.7 cm	6.7 cm	6.6
FE dimensions	$8.01 \times 7.71 \times 61.5$ cm^3		$8 \times 7.61 \times 59.9$ cm^3
Uranium per fuel plate		15.26 g	12.2 g
Weight of U-235 per fuel assembly		213.7 g	195.2 g
Density of total uranium in meat		3.0 g/cc	0.69951 g/cc
Total density of meat		4.8 g/cc	3.16367 g/cc
Density of U-235 in meat		0.591 g/cc	0.6516 g/cc

Description of TRR

The TRR is a pool type research reactor, in which light water serves as coolant, radiological shielding as well as neutron moderating medium and reflector. The reactor is designed and licensed to operate at a maximum thermal power level of 5 MW. The reactor core assembly is located in a two-section pool and may be operated in either pool. One of the sections contains experimental facilities, like beam tubes, rabbit system, and thermal column. The other section is an open area for bulk irradiation studies. The major components of TRR are the pool (including embedment and accessories), bridge and support structure, core, cooling system, control and instrumentation, ventilation system, and the experimental facilities. Details of reactor description and core parameters are given in TRR-Safety Analysis Reports (SAR[2]).

Elements of the reactor core are arranged in a 9 by 6 grid plate structure. Specifications of TRR fuel assemblies are given in Table 1.[3] HEU–SFE is different in composition and dimension with LEU–SFE. Main differences are enrichment and the number of fuel plates. To study the HEU–SFE replacement in a mixed core, equilibrium core 61-B was selected as a reference core. The core configuration of the reference core and burn-up of the fuel elements (in percent of the initial value of ^{235}U) at the BOC is given in Fig. 1.

2. Methodology

2.1. Simulation methodology

The MTR_PC package has been developed by INVAP (Argentina) in order to perform neutronic, thermal hydraulic and shielding calculations of MTR-type reactors. In this research, WIMSD-5B (NEA, 2003), POS_WIMS, HXS, BORGES (Rubio, 1993), and CITVAP v.3.1 (Villarino and Carlos, 1993) neutronic part of MTR_PC package are used to calculate kinetic and neutronic parameters of TRR reference and mixed-core. CITVAP is a new version of the CITATION-II code. It solves 1, 2 or 3-dimensional multi-group diffusion equation in both rectangular and cylindrical geometries. WIMSD with ENDF/B-IV library was employed for macroscopic cross-section generation which provides nuclear cross-sections in the form of 69-energy group structure. POS_WIMS is a post processor program of WIMS code used to condense and homogenizes macroscopic cross section for CITVAP from WIMS output for neutronic and reactivity feedback coefficient calculations. The HXS program (Handle Cross-Section) makes the connection between cell and core calculations. The BORGES code prepares microscopic

[2] AEOI, 2001. Safety Analysis Report for the Tehran Research Reactor (LEU), Tehran-Iran.
[3] INVAP, 1990. T.R.R: Technical Guaranties Report.