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Using PARET Code for Analyzing Research Reactor Cores with Two Fuel Geometries

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ABSTRACT

The PARET code is limited to one fuel dimension and channel spacing. The research reactor cores of mixed fuel or cores utilized for fuel irradiation are with two fuel geometries. This paper proposed a method of two calculations for analyzing these cores using PARET code. In this method, the first calculation is for total core power with reactivity input. The calculated core power and its distribution are used as input to the second calculation for temperature calculations of the fuel element (plates or rod) with different geometry. The power input to the second calculation is the power produced in this fuel element with time. The method was tested compared with IAEA benchmark core calculations and RELAP code calculation of a typical research reactor core utilized for uranium plates irradiation. It is shown that the proposed method gives accurate results for analyzing research reactor cores with two fuel geometries using PARET code.

1. Introduction

PARET is a coupled neutronic- thermal hydraulic code designed for use in prediction of the course and consequence of nondestructive reactivity accidents in research reactor (RR) cores. It is not applicable to either destructive excursions or situations in which there are space-time variations in the neutron flux [1]. PARET has been shown to give accurate prediction of RR core power and the maximum fuel, clad and coolant temperatures [2, 3, 4]. However, it is limited to one fuel dimensions and channel spacing [1, 5].

RR cores of mixed Low Enriched Uranium (LEU) and Highly Enriched Uranium (HEU) fuels and cores utilized for fuel irradiation are with two fuel geometries. The use of less burnt HEU spent fuel elements with the LEU fuel elements is being considered in RR to save money on the purchase of costly fuel elements or in transition cores from HEU to LEU core [5]. Irradiation of fuel plates (or rods) is for the purpose of ^{99}Mo production or new fuel tests [6, 7].

This paper proposed a method of two calculations for analyzing two fuel geometries cores using PARET code. With this method, analysis of a mixed fuel core and RR cores utilized for fuel irradiation can be done using PARET code. In this method, the core power versus time is calculated first and the fuel, clad and coolant temperatures are calculated in the second calculation for a specific fuel element in the core. The proposed method was used in analyzing reactivity insertion transients for IAEA benchmark core and typical RR core utilized for uranium plates irradiation. The results are compared with benchmark calculations and RELAP5/MODE3 calculation. The proposed method has been shown to give accurate prediction of fuel, clad, and coolant temperatures.

2. Reactor core description

The reactor core is the main component concerned with analysis. A description of the IAEA benchmark core is shown in Fig. 1. The in-core positions can be used for fixing a fuel element, a water box, or one (or more) HEU fuel. Typical RR core consists of fuel elements and uranium plates irradiation box is shown in Fig. 2. The main design parameters and core specifications for IAEA and typical RR cores are summarized in Table 1[3, 8, 9, 10].

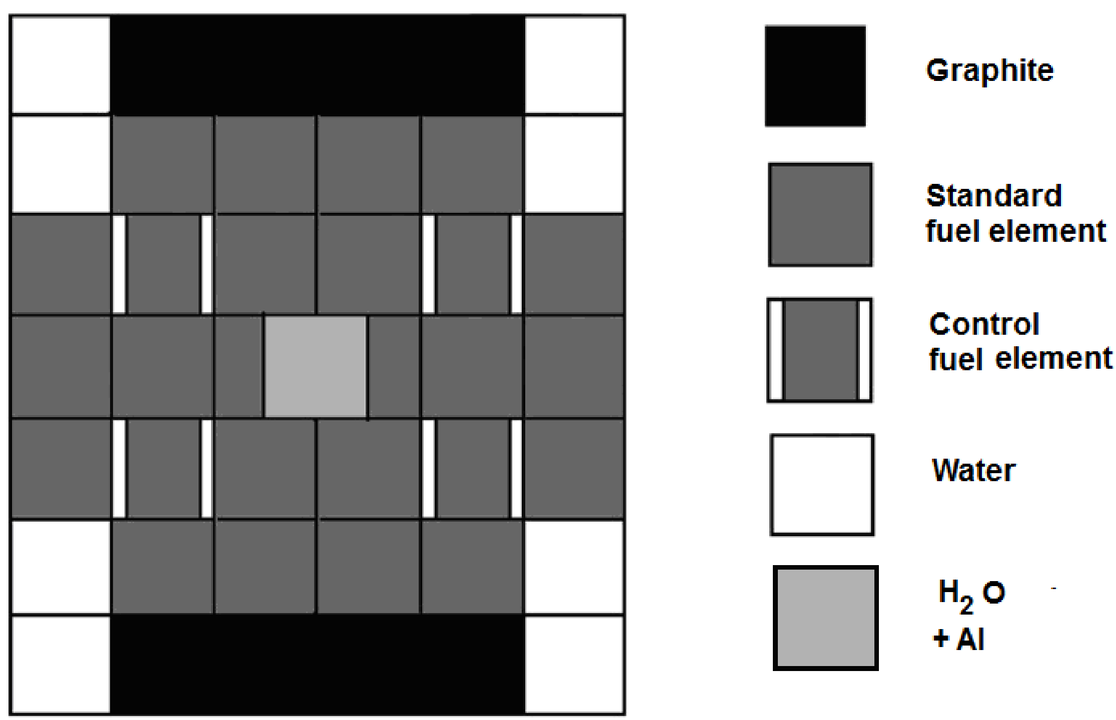

Figure 1: IAEA benchmark core

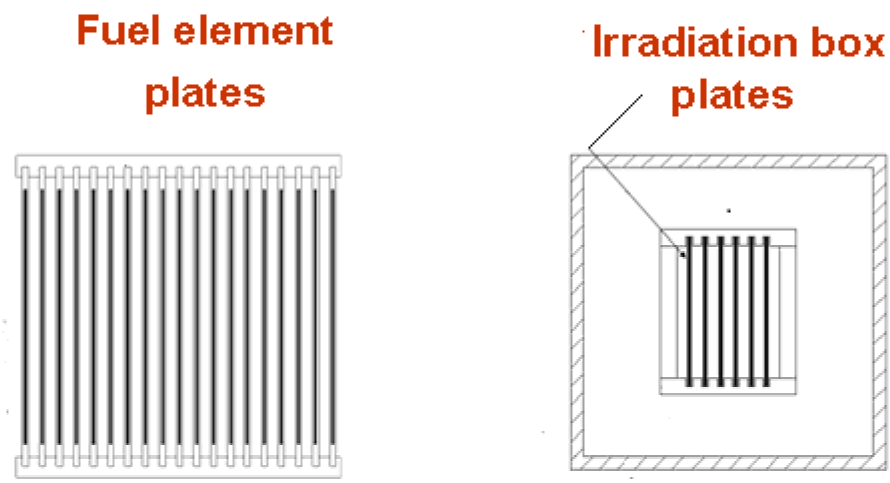

Figure 2: RR core with fuel elements and uranium plates in irradiation box

Table 1: Main design parameters and core specifications

Parameters	IAEA core	Typical core	
		Fuel element	Irradiation box
Total fuel volume (m^3)	0.01062	0.01838	0.0000145
Meat dimensions (mm × mm × mm)	0.51 × 63 × 600	0.7 × 64 × 800	0.7 × 30× 115
Cladding thickness (mm)	0.38	0.4	0.35
Plate width (mm)	65	70	32
Hot channel total peaking factor	2.5	3 (maximum)	0.286 % of core power
Axial peaking factor	1.5	1.31	1
Inlet core temperatures (^oC)	38	20	
Coolant channel mass flux (kg/s.m^2)	3215.8	5090	6000
Channel spacing (mm)	2.23	2.7	2.33
Total cross section flow area (m^2)	0.085792725	0.0980	0.0005220
Delayed neutron fraction	0.007275	0.007050	
Prompt neutrons generation time (s)	43.74×10^{-6}	75×10^{-6}	
Coolant temperature coefficient ($/$^oC$)	-0.01082	-0.01540	
Void coefficient ($/%)	-0.407	-0.241	
Fuel temperature coefficient ($/$^oC$)	-0.003310	-0.002482	

3. Method

A two channels model (see Fig. 3) is usually adopted in the PARET code calculations with hot channel has the hottest fuel plate and average channel has the remaining plates in the core. PARET input is limited to one fuel dimensions and channel spacing. Hence, it is not used for analyzing cores of mixed fuel or cores that have fuel irradiation boxes. This paper proposed two calculations method to calculate the HEU fuel maximum fuel, clad, coolant temperatures. With reactivity versus time input, the first calculation is for the core power including the total fuel volume of the two fuels and the dimensions of the common fuel. The output core power versus time (power history) is the input to the second calculation of temperatures for HEU fuel or irradiated fuel. It should be note that the input power to the second calculation is a percentage (%) of the total core power.

The core power space dependence is postulated to be the static neutronic calculation of the power distribution. When analyzing practical situations some calculations of power and flow distribution will be required first. Axial power distribution of HEU fuel or irradiated fuel is chopped cosine or uniform.

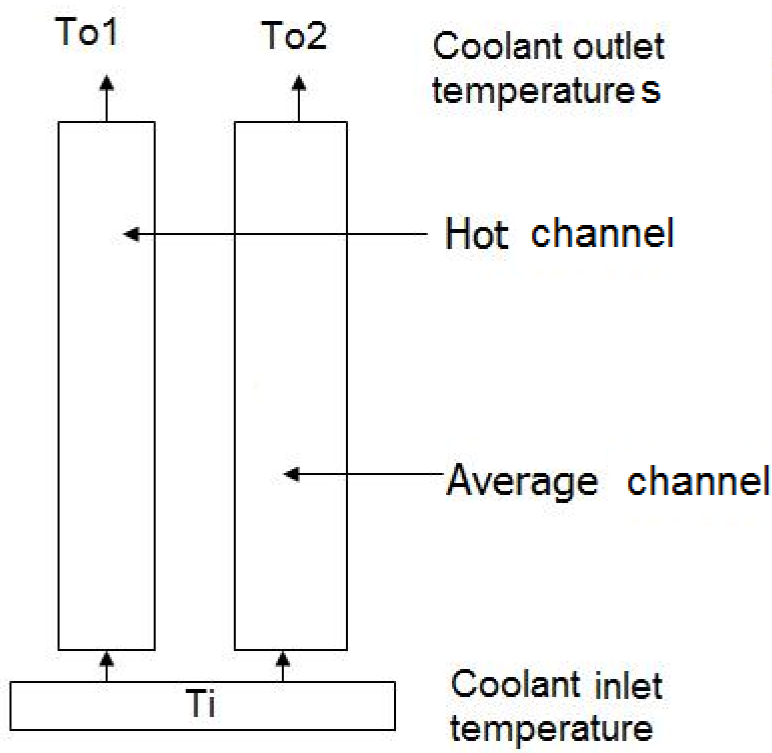

Figure 3: Two channels model

4. Results

4.1 IAEA benchmark core

The analysis of IAEA benchmark transients was done using two calculations method. The benchmark transients for the IAEA core with LEU fuel include insertion of 0.9 $/s and 1.50 $ /0.5 s in a critical core of initial power of 1 Watt. The scram in is a linear insertion of -10 $ in 0.5 seconds with set point of 12 MW and a time delay of 25 ms. The results are summarized in Table 2 showing good agreement with benchmark and RElAP5/MODE3 calculations.

Table 2: Comparison of Peak Temperature for Reactivity Insertion Transients

Reactivity Insertion Transients	**0.9 $/s Peak Temperatures, °C**			**1.5$/0.5s Peak Temperatures, °C**		
	Fuel	Clad	Coolant	Fuel	Clad	Coolant
PARET (IAEA-TECDOC-643)	80.6	77.7	53.9	183.4	156.7	82.0
PARET/ANL [11]	81.2	78.0	54.1	188.3	157.9	83.3
RELAP5/MOD3 [11]	80.8	77.8	53.2	196.2	169.8	82.0
PARET (2 calculations method)	82.6	79.4	55.7	183.1	156.4	88.1

4.2. Typical Core Utilized for Fuel Irradiation

The analysis of 1$/s reactivity insertion in a typical RR critical core of initial power of 1 KW was done. The scram is a linear insertion of -4$ in 0.5 seconds with set point of 24 MW and a time delay of 25 ms. The verification of two calculations method is shown in Table 3 compared to RELAP code calculations for fuel plates in irradiation box.

Table 3: The comparison between PARET and RELAP results for irradiation box

Reactivity Insertion Transients	**1 $/s Peak Temperatures, °C**		
	Fuel	Clad	Coolant
RELAP5/MOD3 [10]	114.0	103.0	27.1
PARET (2 calculations method)	107.0	98.2	26.3

5. Conclusions

The results show that the proposed method gives accurate prediction of fuel, clad, and coolant temperatures. As conclusion with this method, the PARET code can be used for analyzing the reactivity accidents of RR cores of mixed LEU and HEU fuels or cores utilized for fuel irradiation.

6. References

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