

# Investigation of the Capability of REALP5 to Solve Complex Fuel Geometry

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**Abstract**—This work is developed within IAEA Coordinated Research Program 1496, “Innovative methods in research reactor analysis: Benchmark against experimental data on neutronics and thermal-hydraulic computational methods and tools for operation and safety analysis of research reactors”.

The study investigates the capability of Code RELAP5/Mod3.4 to solve complex geometry complexity. Its results are compared to the results of PARET, a common code in thermal hydraulic analysis for research reactors, belonging to MTR-PC groups.

The WWR-SM reactor at the Institute of Nuclear Physics (INP) in the Republic of Uzbekistan is simulated using both PARET and RELAP5 at steady state. Results from the two codes are compared.

REALP5 code succeeded in solving the complex fuel geometry. The PARET code needed some calculations to obtain the final result. Although the final results from the PARET are more accurate, the small differences in both results makes using RELAP5 code recommended in case of complex fuel assemblies.

**Keywords**—Complex fuel geometry, PARET, RELAP5, WWR-SM reactor.

## I. INTRODUCTION

THE RELAP5 code was developed by Idaho National Engineering Laboratory (INEL) for thermal hydraulics studies in PWRs. However, current investigations and code adaptations have demonstrated that the RELAP5 code can be also applied for thermal hydraulic analysis of nuclear research reactors with good predictions. RELAP5/MOD3.4 is a coupled kinetics, one-dimensional heat transfer and two components hydrodynamics code and is capable of modeling all of the components of the systems in a very general manner [1].

However the PARET code was developed by Argonne National Laboratory (ANL). This code had been designed to calculate fuel and coolant behaviors during transients or accidents initiated by reactivity or power changes. It solves the energy balance and heat transfer equations for coolant and fuel. The coolant may be in the single or two-phase states. Power is calculated following the point kinetics model, while reactivity feedback terms are taken into account through expansion in each of the following variables: coolant temperature, moderator density and fuel temperature [2].

The WWR-SM reactor has been operating since 1979 on 90% highly enriched uranium (HEU) fuel. It was gradually converted to LEU. The reactor operates at 10 MW maximum

powers and is currently used for research and to produce isotopes for medical purposes.

WWR-SM reactor is a light water moderated and cooled reactor. The reactor is capable of producing 10 MW of thermal power and generating average thermal neutron flux of  $2.0 \times 10^{14}$  n/cm<sup>2</sup>.sec. The nominal flow is 1250m<sup>3</sup>/hr; the water flows downward under the condition of the forced convection induced by the primary cooling pumps.

## II. REACTOR CORE DESCRIPTION

The WWR-SM reactor is a pool type reactor. The core is located in a cylindrical reactor vessel, having inner diameter of 0.652m and height of approximately 2.8m. The reactor vessel is located off-center within a central tank, having diameter of 1.1m and height of approximately 6.3m. The central tank is located within a shielding tank having the same height as central tank and diameter of 2.3m. The core is water-cooled by forced convection; the water enters the bottom of the central tank through a single 0.35m diameter pipe, flows upward, turns toward the center and flows down through the core, exiting the center of the reactor vessel through a single 0.35m diameter pipe. There is no other piping penetration in the reactor vessel and central tank. An elevation schematic of the reactor is shown in Fig. 1 [3].

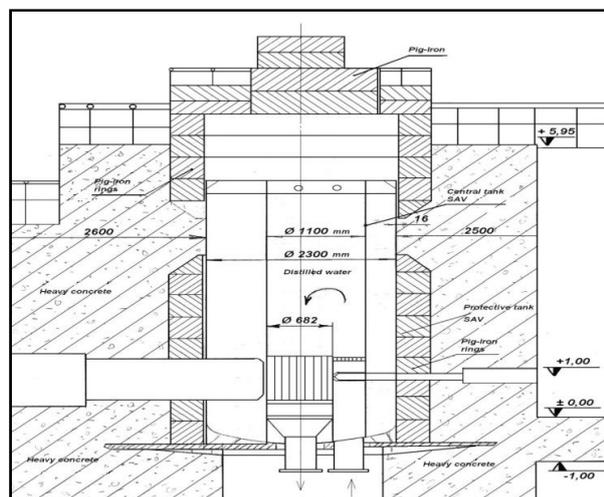


Fig. 1 An elevation schematic of the WWR-SM reactor

The reactor operates at maximum power of 10 MWth using 20 IRT-4M fuel assemblies having 6 fuel tubes. The core is made up of square shaped fuel elements of 7.15\*7.15cm, with an active length of 0.6m. Each fuel element is composed of

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6coaxial squares plates with a 19.7% enriched uranium load; the clad is aluminum [4].

Beryllium is used as reflector material to surround the active core by an optimum reflector [4].

The reactor core parameters and the Fuel Assembly Parameters are summarized in Tables I and II, respectively. The core configuration and a horizontal cross section of the IRT-4M Fuel Assemblies (FA) used in the reactor are presented in Figs. 2 and 3, respectively [4]. As seen in Fig. 3, the fuel assembly is made of six fuel round-cornered square tubes, which makes its simulation a challenge.

TABLE I  
REACTOR CORE PARAMETERS

Moderator type	Water
Coolant type	Water
Recirculation	1 circuit
Nominal moderator temperature	45 °C
Nominal fuel temperature	60 °C
Nominal moderator density	0,980 g/cm <sup>3</sup>
Nominal inlet temperature	40 °C
Nominal outlet temperature	45 °C
Nominal Reflector temperature	47 °C
Operating outlet pressure	150.0 kPa
Operating Inlet pressure	137.22 kPa

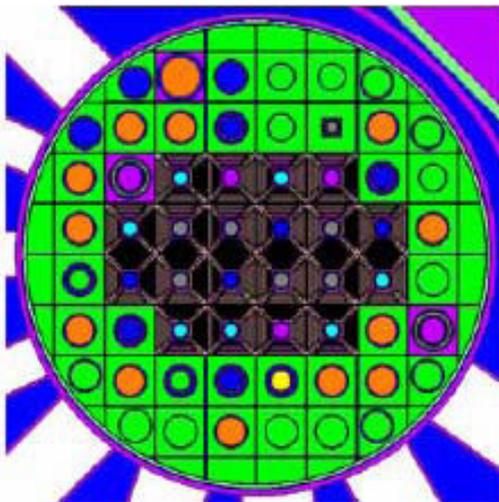


Fig. 2 WWR-SM reactor core

TABLE II  
FUEL ASSEMBLY PARAMETERS

Parameter	6-Tube IRT-4M
Dispersant	UO <sub>2</sub> -Al
Wt. % <sup>235</sup> U	19.7
(gU/cm <sup>3</sup> ) <sub>meat</sub>	2.8
Void fraction %	31.7
<sup>235</sup> U/FA, g	266
H <sub>meats</sub> , cm	60
t <sub>meats</sub> , mm	0.70
t <sub>clad</sub> , mm	0.45
t <sub>coolant</sub> , mm	1.85
Vol <sub>meat</sub> , cm <sup>3</sup>	481
Clad Temperature	98 °C

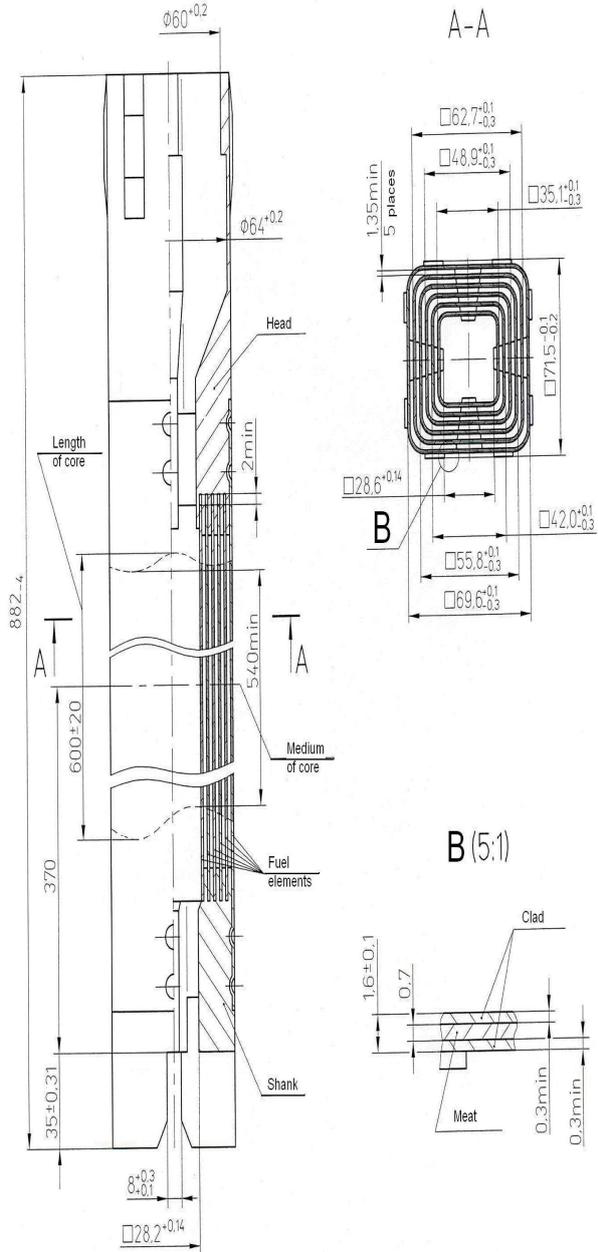


Fig. 3 Fuel assembly scheme

The hydraulic characteristics of IRT-4MFA are shown in Table III [4].

TABLE III  
HYDRAULIC CHARACTERISTICS OF IRT-4M FA

Gap in FA	Area, [mm <sup>2</sup> ]	Velocity [m/s]	Hydraulic diameter [mm]	Flow [m <sup>3</sup> /h]
1	342.3	3.04	5.199	3.75
2	488.8	2.82	4.001	4.96
3	436.6	3.09	4.004	4.86
4	384.5	3.21	4.008	4.44
5	332.3	3.08	4.013	3.68
6	280.2	3.6	4.019	3.63
7	228.0	3.52	4.029	2.88
Sum – 6 tubes	2492.7			28.20

The specifications of heat exchanger and pumps are summarized in Table IV and heat exchanger schemes presented in Fig. 4 [4].

TABLE IV  
HEAT EXCHANGER AND PUMPS SPECIFICATIONS

Heat Exchanger	
Type	Shell & tube : TNG-1200
Number	2
Pressure drop (primary side)	2.1 atm
Heat transfer area	358 m <sup>2</sup>
Heat transfer coefficient	1.391 W/cm <sup>2</sup>
Pump	
Number	5
Power	Pri. = 40 kW
Nominal Flow rate	Pri. = 220 - 340 m <sup>3</sup> /h
Rotation speed	1460 rpm
Elevation height	Pri. = 25-32 m

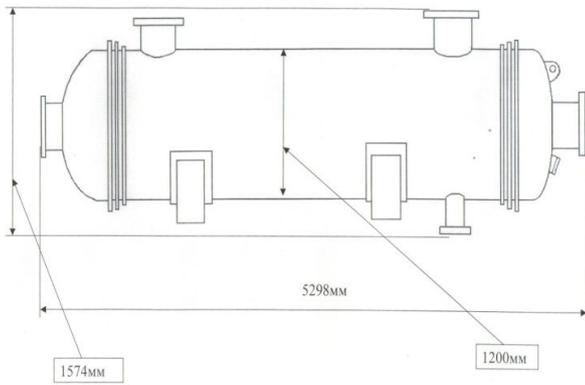


Fig. 4 Heat exchanger scheme

### III. REACTOR MODELS

#### A. PARET Model

Due to the complexity of the design of the fuel assemblies, the PARET Code uses an equivalent geometry to simplify the analysis. In addition, PARET simulates the core as 6 sub-core in 6 inputs as illustrating below.

The IRT-4M Fuel assembly may be treated as 6 fuel round-cornered square tubes, each having its associated coolant, flowing on the inner and outer surface as shown in Fig. 5. The figure shows the fuel assembly and the dimension of fuel element.

By using log mean method, equivalent widths are calculated (each fuel plate and channel have equivalent width). Using the same method, half coolant channel before and after fuel element are taken as show in Fig. 6, except in the last channel, half coolant channel before and total channel after.

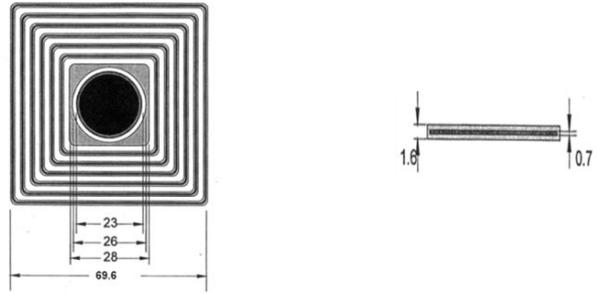


Fig. 5 Fuel Assembly (left) and fuel element (right)

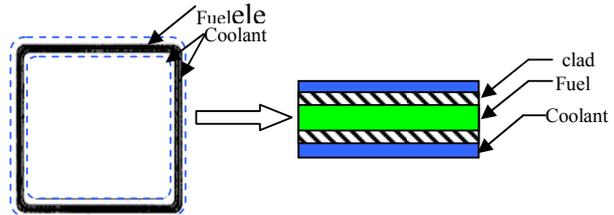


Fig. 6 The real fuel element (left) and its equivalence (right)

The methodology of log mean is as follows:

Equivalent Width:

$$W_e = \frac{W_1 - W_2}{\ln\left(\frac{W_1}{W_2}\right)} \quad (1)$$

where;

$W_e$ : equivalent width.

$W_1$ : outer width of coolant channel in real plate.

$W_2$ : inner width of coolant channel in real plate.

The equivalent plate has the thickness of real plate. Due to log mean method, the velocity of coolant in the channel, the coolant area, the mass flow rate, and the hydraulic diameter differ from real plate, so that, new values need to be calculated.

Equivalent Cross Section Area:

$$A_e = 0.5(A_1 + A_2) \quad (2)$$

where:

$A_e$ : equivalent area.

$A_1$ : outer area of coolant channel.

$A_2$ : inner area of coolant channel.

Equivalent Velocity:

$$v_e = \frac{m_1 + m_2}{A_e} \quad (3)$$

where:

$v_e$ : Equivalent velocity.

$m_1$ : Mass flow in the coolant channel before fuel plate ( $v_1 \times A_1$ ) (m<sup>3</sup>/sec).

$m_2$ : Mass flow in the coolant channel after fuel plate ( $v_2 \times A_2$ ) (m<sup>3</sup>/sec).

Equivalent Mass Flow:

$$m_e = v_e \rho \tag{4}$$

where:

$\rho$ : coolant density (kg/m<sup>3</sup>)

$m_e$ : Equivalent mass flow (kg/sec)

PARET needs the total power, total fuel volume and total cross section area. The Uzbekistan core is divided into 6 different cores (with 6 different channels). Each sub-core has a different fuel element and in the end the average value of 6 cores is calculated. The equivalent values of the 6 channels entered into PARET are shown in Tables V and VI.

TABLE V  
EQUIVALENT VALUES OF EQUIVALENT PLATE

Channel	Equivalent width (mm)	Equivalent velocity (m/sec)	Equivalent area (mm <sup>2</sup> )	Equivalent mass flow (kg/sec)	Equivalent hydraulic D(mm)
1	68	4.26	510.35	4.25×10 <sup>3</sup>	14
2	61.035	5.91	452.14	5.9×10 <sup>3</sup>	14
3	54.13	6.28	401.08	6.3×10 <sup>3</sup>	14
4	47.22	6.28	350.02	6.3×10 <sup>3</sup>	14
5	40.3	6.29	298.96	6.7×10 <sup>3</sup>	14
6	32.38	4.73	355.66	4.7×10 <sup>3</sup>	20

TABLE VI  
DIFFERENT DATA OF THE 6 CHANNELS USED IN PARET

Channel	Total power (MW)	Total volume(m <sup>3</sup> )	Total equivalent cross section area(m <sup>2</sup> )
1	2.24	2.284×10 <sup>-3</sup>	10.207×10 <sup>-3</sup>
2	2.006	2.050×10 <sup>-3</sup>	9.043 ×10 <sup>-3</sup>
3	1.78	1.820×10 <sup>-3</sup>	8.022 ×10 <sup>-3</sup>
4	1.56	1.590×10 <sup>-3</sup>	7.000 ×10 <sup>-3</sup>
5	1.32	1.357×10 <sup>-3</sup>	5.980 ×10 <sup>-3</sup>
6	1.1	1.126×10 <sup>-3</sup>	7.110 ×10 <sup>-3</sup>

After running PARET code, there are 6 output values shown in Table VII.

TABLE VII  
OUTPUT DATA OF EACH CHANNEL

Channel	Channel power	Output temperature (°C)
1	2.95E+05	77.5
2	2.45E+05	68.3
3	2.31E+05	66.9
4	2.02E+05	67.0
5	1.72E+05	65.5
6	2.07E+05	74.2

To obtain the average output temperature from the core the following equation used;

$$T_0 = \frac{\sum_i m_i A_i C_p (T_i - T_{in})}{m_i A_t C_p} + T_{in} \tag{5}$$

where;

$m_i$  and  $m_t$  mass flow (kg/.sec),  $T_{in}$  = 45°C.

B. RELAP5 Model

It should be noted that the PARET code is a core code designed to only simulate the fuel assembly and core whereas, RELAP5 simulates all the reactor components.

RELAP5 has one input file that contains the all the information it requires. The reactor model includes two loops. The first is the Core Cooling System (CCS) including the corresponding pipes, pumps and heat exchangers and the second is the Secondary Cooling System (SCS) specified in terms of an inlet and outlet flow rate to allow the power removal generated in the core.

The water flows downward under forced convection induced by the primary cooling pump. The coolant from different parts of the core components are mixed in the lower plenum before entering the heat exchanger [3].

The input file of RELAP5 includes hydrodynamic components (single volumes, junctions, pipes, branches, ....etc) with their data, heat structures of these hydrodynamics, and tables containing information about core material, heat exchanger material, initial reactor power, reactivity insertion, points kinetics,... etc.

The coolant system nodalization is given in Fig. 7. In the figure, each number corresponds to a component included in the model.

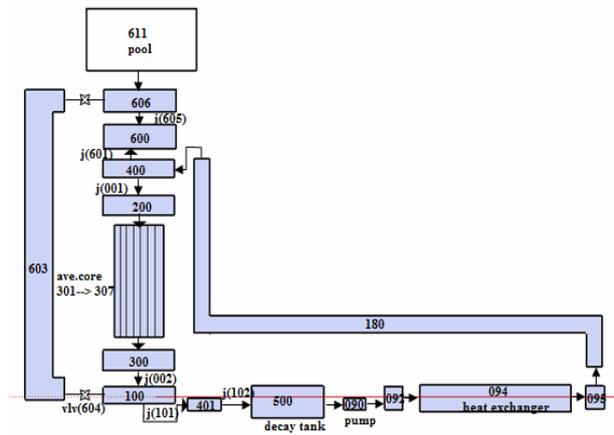


Fig. 7 The coolant system nodalization

The hydrodynamic components are:

1. Pump

Components 090 (SNGVOL) express the main pump which is presented as single volume for simplicity. Only the geometric characteristics of the pump are taken into account (length, width and flow direction).

2. Heat Exchangers

This reactor has a horizontal shell and tube heat exchanger and is represented by annulus and pipe. The pipe (094 in Fig. 7) is the primary side (tube part) and the annulus is the secondary side (shell part). The input flow is from one side and the output flow from the other side, as shown in Fig. 4. It is noted here that the details of the heat exchangers were

not sent by the reactor manager. Thus the heat exchanger is redesigned in this study depending on the pressure drop through shell and tube, its transfer area and removal of the heat generated in the CCS circuit. The redesign methodology depends on three equations solved together;

Heat exchanger transference area: [6]

$$A = n \pi D_o L A \tag{6}$$

where;

A: transference area (m<sup>2</sup>),

n: number of tubes in heat exchanger,

D<sub>o</sub>: outside diameter of tubes (m),

L: length of tubes (m).

Flow rate inside the tube or shell:

$$m' = n \frac{\pi}{4} D_i^2 v \tag{7}$$

where;

m': Flow rate (m<sup>3</sup>/hr)

$\frac{\pi}{4} D_i^2$ : Cross section area (m<sup>2</sup>)

v: Flow velocity (m/s)

Pressure drop through the tube or shell:

According to IAEA-TECDOC 233 [5], the pressure drop across any tube includes the pressure loss at the entrance to the fuel channel ( $\Delta P_{en}$ ), the friction loss in the fuel channel ( $\Delta P_f$ ), and the pressure loss at the exit of the fuel channel ( $\Delta P_{ex}$ ). That is:

$$\Delta P = \Delta P_{en} + \Delta P_f + \Delta P_{ex} \tag{8}$$

The entrance, friction, and exit pressure losses for single-phase flow can be calculated from the following standard formulae [5];

$$\Delta P_{en} = \frac{K \rho U^2}{2} \times 10^{-5} \tag{9}$$

$$\Delta P_f = \frac{4f L_c \rho U^2}{2D_e} \times 10^{-5} \tag{10}$$

$$\Delta P_{ex} = \frac{\rho(U - U_o)^2}{2} \times 10^{-5} \tag{11}$$

where;

K: A conservative value = 0.5,

ρ: density (kg/m<sup>3</sup>),

U: Coolant velocity (m/s),

f: The friction factor =  $0.0791/Re^{0.25}$  (for  $5000 < Re < 51094$ ),

Re: Reynolds number =  $\frac{\rho U D_e}{\mu}$

L<sub>c</sub>: tube length (m),

D<sub>e</sub>: hydraulic diameter (m).

The final formula is:

$$\Delta P = \frac{\rho U^2 \times 10^{-5}}{2} \left( 1.5 + \frac{0.3164 L_c \mu^{0.25}}{\rho^{0.25} U^{0.25} D^{1.25}} \right) \tag{12}$$

Equations (6)-(8) have 4 unknowns (n: number of tubes in heat exchanger, D<sub>o</sub>: outside diameter of tubes, D<sub>i</sub>: inside diameter of tubes, and v: flow velocity). Therefore, in order to solve them, value of D<sub>o</sub> was assumed. The choice of the initial values of D<sub>o</sub> (3/4" and 1") is according to the recommendations in [6]. Firstly it assumed as 1", and by using properties of steel pipe tables [7], the rest of equations are solved using iteration. However, a value of D<sub>o</sub> = 1" failed to solve these equations according to [7]. Therefore, the value of D<sub>o</sub> = 3/4" was used. This value succeeded in obtaining the values of n, D<sub>i</sub> and v from the equations and the details of heat exchanger are known and used as a part of RELAP5 input.

### 3. Core

The RELAP5 model avoids all the PARET model calculation, where it uses the core as is in one input with only one change. A rectangular ring is converted to circular ring as show in Fig. 8, where this square shape is converted into circular shape (square perimeter = circular perimeter:  $4a=2\pi R$ ). The dimensions of the two shapes are shown in Table VIII. The whole core is divided into 7 sub-cores. Each sub-cores has different dimensions, flow, and heat structure.

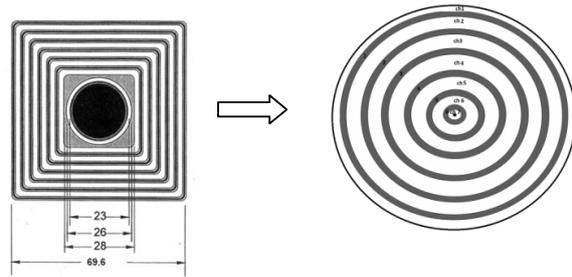


Fig. 8 The real assembly (left) and the equivalent assembly (right)

Component 400 (see Fig. 7) is a single volume that represents the upper plenum of the core. At this volume the flow divided into 7 parts, each part of flow goes to one of average sub-core.

Pipes 301 to 307 represent the average sub-cores.

### 4. Pool

The pool is simulated as time dependent volume (component 611 in Fig. 7). This part is worked only during LOFA or LOCA accidents, and after scram or shutdown to cool the core. The flap valve remains closed as long as the primary flow is driven by the core cooling pumps (during normal operation). It is opened by gravity after shutdown or after scram, then natural convection is established and the cooling flow in the channel turned from downward to upward.

The natural convection is simulated by using pipe 603 which it carries the coolant from pool to the bottom of the core and valve 604 (see Fig.7).

TABLE VIII  
6-TUBE UO<sub>2</sub>-AL FUEL GEOMETRY

Tubes number	Material	Rectangular		Circular	
		Flat OD	Flat ID	OD	ID
1	Water	71.5	69.6	45.518	44.309
	Clad	69.6	68.7	45.518	44.309
	Fuel-1	68.7	67.3	43.736	42.845
	Clad	67.3	66.4	42.845	42.272
2	Water	66.4	62.7	42.272	39.916
	Clad	62.7	61.8	39.916	39.343
	Fuel-2	61.8	60.4	39.343	38.452
	Clad	60.4	59.5	38.452	37.879
3	Water	59.5	55.8	37.879	35.523
	Clad	55.8	54.9	35.523	34.950
	Fuel-3	54.9	53.5	34.059	34.059
	Clad	53.5	52.6	34.059	33.486
4	Water	52.6	48.9	33.486	31.131
	Clad	48.9	48	31.131	30.558
	Fuel-4	48	46.6	30.558	29.666
	Clad	46.6	45.7	29.666	29.094
5	Water	45.7	42	29.094	26.738
	Clad	42	41.1	26.738	26.165
	Fuel-5	41.1	39.7	26.165	25.274
	Clad	39.7	38.8	25.274	24.701
6	Water	38.8	35.1	24.701	22.345
	Clad	35.1	34.2	22.345	21.772
	Fuel-6	34.2	32.8	21.772	20.881
	Clad	32.8	31.9	20.881	20.308
	Water	31.9	28.2	20.308	17.953

5. Other Components and Junctions

Additional components are used to complete the model loop and its geometry is adjusted depending on flow and pressure drop. Also the geometrical data of junctions (i.e. area and hydraulic diameter) are taken directly from the corresponding connected volumes. Where changes in size take place, data from the smaller components are taken.

It is worth noting that most of the components as connecting pipes, pump, and heat exchanger are not sent in the reactor report, but its dimensions are adjusted to obtain the correct core temperatures in steady state.

The heat structures are represented as follows:

1. Core

Heat structure of WWR-SM fuel is different, since the coolant channel is annulus. Therefore it has two heat sources; one from outside and the other from inside the fuel ring as show in Fig. 9.

The geometrical characteristics of the fuel are accurately described [4]. The fuel described has cylindrical geometry, with an aluminum cladding and an UO<sub>2</sub>-Al meat the material properties (thermal conductivity and volumetric heat capacity) were taken from [4].

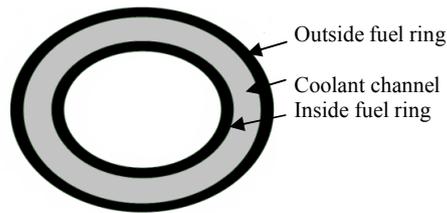


Fig. 9 WWR-SM coolant channel

2. Heat Exchangers

The components of the secondary loop are not included in the model except the secondary side of heat exchanger, where this part is simulated as an open loop and is adjusted to obtain the correct primary side of heat exchanger outlet temperature in steady state.

IV. RESULTS AND DISCUSSION

The output temperatures calculated by PARET, RELAP5 and the output temperatures measured in the reactor are given in Table IX.

TABLE IX  
EXPERIMENTAL AND CALCULATED OUTPUT TEMPERATURE

	Output temperature (°C)		Difference (%)
	Experimental results	PARET results	
	51	51.1	< 0.3
		52.2	< 2.5

The PARET code is a core code and it need symmetrical fuel assemblies because it was designed to simulate plate and cylindrical fuel assemblies. Its input needs a single value for each fuel assembly characteristic (active length, width, flow, flux profile,...etc). So it fails to simulate the core of WWR-SM reactor which has annulus fuel assembly simultaneously. It needs to divide an asymmetrical core into 6 symmetrical sub-cores and simulate each one separately. Each sub-core needs an equivalent data for power, flow, coolant velocity, fuel width, flux profile ...etc, as mentioned earlier. Thus extra time and effort are needed to calculate these new values. Finally the output results of the 6 sub-cores obtained from the code are used to calculate the final overall results manually.

On the other hand, the RELAP5 code succeeded in simulating this type of fuel assemblies (annulus fuel) in one input without spending extra time in manual calculations.

V. CONCLUSION

In the capability test of the RELAP5, WWR-SM reactor and PARET code are used. The WWR-SM reactor was simulated by using PARET code and RELAP5 code. It was found that the RELAP5 succeeded to solve the complexity of fuel geometry in one input file and the output temperature is obtained from the code directly. The PARET code use six input files then used a general equation (5) to determine the output temperature. Although the result from PARET is more accurate, RELAP5 is recommended to solve this type of problems.

## ACKNOWLEDGMENT

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## REFERENCES

- [1] Antonella L. Costa, Patricia Amelia L.Reis, et al. "Thermal Hydraulic Analysis of the IPR-R1 TRIGA Research Reactor Using a RELAP5 Model", Nuclear Engineering and design 240, pp 1487-1494, 2010.
- [2] INVAP," Replacement Research Reactor Project- SAR-Ch 16- Safety Analysis", Australian Nuclear Science and Technology Organization, November 2004.
- [3] P.L. Garner and N.A. Hanan, "Transient Analyses for the Uzbekistan WWR-SM Reactor with IRT-3M HEU Fuel and IRT-4M LEU Fuel", Nuclear Engineering Division, Argonne National Laboratory, August 2007.
- [4] IAEA, Proceeding Series on "Innovative Methods in Research Reactor Analysis: Benchmark against Experimental Data on Neutronics and Thermal Hydraulic Computational Methods and Tools for Operation and Safety Analysis of Research Reactors", Vienna (2010).
- [5] IAEA-TECDOC 233, "Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels", 1980.
- [6] John E. Edwards, "Design and Rating Shell and Tube Heat Exchangers", P & I Design Ltd, Teesside, UK, Issued 29 August 08.
- [7] Green, Don W.; Perry, Robert H., "Perry's Chemical Engineers' Handbook", 8th ed., McGraw-Hill, December 2008.