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NATIRT – Model of the Loss of Flow Transient for Tajoura Research Reactor with LEU Fuel

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Abstract:

Design parameters are presented for Tajoura reactor core utilizing the new fuel assemblies with low enriched uranium (LEU, using IRT-4M fuel assemblies) in the steady state safety operational parameters and Loss of Flow transient mathematical models (NATIRT - computer program. The calculated results of the model are presented in the cases of forced convection steady state, transient during emergency tank filling and natural convection after emergency tank filling modes at different reactor core thermal power level. The results of NATIRT for all cases of flow were in good agreement with the PARET and PLTEMP computer programs.
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Keywords — **Tajoura Reactor, NATIRT, IRT Reactor, LEU Fuel, Loss of Flow, TNRC.**

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I. INTRODUCTION

The Tajoura reactor is a pool type reactor, moderated and cooled by light water located at the Tajoura Nuclear Research Center (TNRC). The reactor is designated to carry out experiments in field of nuclear physics and nuclear engineering, neutron activation analysis, solid state physics and isotope production. The reactor was put into operation at a power level of 10 MW in September 1983 with using the high enriched fuel (HEU) [1].

The base of the Tajoura core is a square grid plate with 36 identically formed places with a lattice pitch of 71.5 mm as shown in Figure (1). The fuel assemblies (FA), the removable beryllium units, and guide tubes of the control rods (8 shim control rods, 2 safety rods and one automatic regulating rod) can be put into these places. The compact core loading of Tajoura consists of 16 FAs. The FAs are surrounded by 20 removable beryllium units. Stationary beryllium reflector surrounds the removable core units.. The active fuel length is 600 mm. The fuel is cooled by the pumped flow of water from top to bottom of the core.

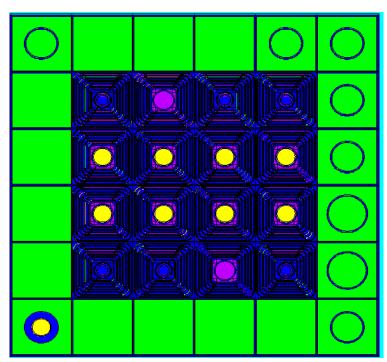

Fig. 1 Horizontal Cross Section of Tajoura Reactor Compact Core.

The reactor is completely converted to Low Enriched Uranium (LEU, 19.7% of ^{235}U) fuel of type IRT-4M at the end of 2006; the new fuel is an alloy (matrix) of aluminum and uranium-dioxide (UO_2–Al) with aluminum cladding [2, 11].

The LEU core is composed of IRT-4M Fuel Assemblies: 10×6-tube fuel assemblies (6TFA) and 6×8-tube fuel assemblies (8TFA). The IRT-4M FA geometries are shown in Figure (2). The 8TFA consists of 8 fuel elements and 9 coolant channels; the 6TFA is identical to the 8TFA except the 2 innermost fuel tubes that are replaced with control rod guide tube. The fuel material is UO_2–Al matrix with 19.7% of ^{235}U enrichment and its thickness is 0.7 mm covered by cladding material type SAV-1 (Russian Al alloy material) and its thickness is 0.45 mm. The 8TFA contains 300 g of ^{235}U and the 6TFA contains 263.8 g of ^{235}U [3].

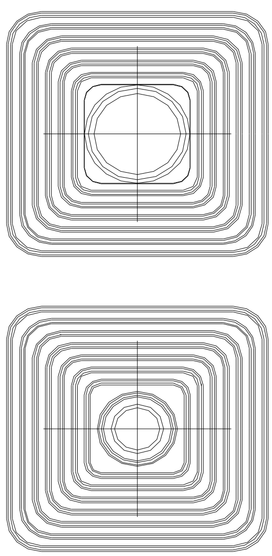

Fig. 2 IRT-4M Fuel Assembly Cross Section.

This report presents the analysis of the Tajoura core for power level (maximum) of 10 MW. The NATIRT computer program has been employed to calculate different parameters such as coolant, cladding and centerline temperatures at coolant forced convection modes before filling of emergency tank and natural convection mode after filling of emergency tank, and other thermal hydraulic critical parameters at steady state and unsteady state conditions.

The unsteady state results of NATIRT computer program are compared with the results of PARET [4] and RELAP [5] codes for response to flow-induced accidents [11]. Also the steady state results were compared with the IRTM code. The main reactor parameters are shown in Table (1).

II. NATIRT - MATHEMATICAL MODEL

The NATIRT is one dimensional computer code in axial direction, it is especially developed to simulate the hot channel for Tajoura reactor core,

Table (1):
TAJOURA REACTOR CORE THERMAL HYDRAULIC DESIGN PARAMETERS

Parameter	LEU
Reactor Core Power, MW	10
Inlet Pressure , MPa	0.16932
Pressure Drop, MPa	0.06570
Active Mass Flow Rate , kg/s	147
8TFA Coolant Volume Flow Rate, m^3/h	33.89
6TFA Coolant Volume Flow Rate, m^3/h	31.93
Active Coolant Volume Flow Rate, m^3/h	533
Primary Coolant Volume Flow Rate, m^3/h	1350
Primary Inlet Temperature, ^oC	45
Coolant Velocity, m/s	3.08

and it calculated the fuel element temperatures distribution (coolant, clad surface and fuel centerline) and other thermal hydraulic parameters at steady state and as a function of transient time in the cases of forced convection and natural convection modes during the power cut off the primary pumps.

The compact core loading of Tajoura reactor consists of 16 fuel assemblies. This reactor has one hot cell (8TFA), this cell also has the hottest channel in the reactor core. The purpose of NATIRT model is simulating the thermal analysis of the reactor core with the hot channel model. Therefore, Figure (4) shows the schematic flow chart of the NATIRT model and this model has the following objectives:

A. Forced Convection Cooling Mode

The forced convection mode of reactor cooling includes two sub-modes; the first is the steady state reactor operation mode, and the second during the emergency tank filling time (79.9 s).

1. Normal Operation Mode: During normal operation of the reactor core, the pumps work to circulate the coolant through the core. The actual operation of the pumps causes negative pressure under the emergency tank that leads to the discharge of water from the emergency tank and leaves it empty. The quantity of water that is discharged from the emergency tank is added into the reactor pool. The direction of the water through the core is downwards, the water flows through the core to the delay tank and then to the pumps. Each pump of the primary loop delivers the water into the heat exchanger to exchange the heat between the primary and the secondary loops then the water re-enters to the core.

The heat transfer coefficient ($h_s = \frac{k.Nu}{D_h}$) is related to the Nusselt number (Nu) There are several international correlations that can be used to obtain a Nu number for turbulent flow in the case of nuclear research reactors, we use a correlation that had been obtained experimentally for IRT reactor, which has the same characteristic as Tajoura reactor and is given as [5]:

$$Nu = 0.021\ Re^{0.8} Pr^{0.43} \left(\frac{Pr}{Pr_s}\right)^{0.25}$$

(1)

This equation gives the Nu number as a function of Re number and Pr number for forced convection and were evaluated at the bulk temperature of the coolant, but Pr_s is evaluated at the fuel cladding surface temperature T_s. Where, k is the thermal conductivity of the reactor coolant [W/m.K] and D_h the hydraulic diameter of the channel, [m].

Once the heat transfer coefficient is evaluated, then the fuel surface temperature can be determined using Newton's law of cooling [6, 7].

$$q'' = h_s(T_s - T)$$

(2)

Where: q'' is the surface heat flux, [W/m²], and h_s Heat transfer coefficient, [W/m².K].

The surface temperature of the fuel plate T_s is calculated from the Newton's law of cooling; the fuel-clad interface temperature T_c is obtained from T_s using the following equation:

$$T_c = T_s + \frac{x_c q_i''}{k_c}$$

(3)

The maximum fuel temperature T_s is evaluated as follows:

$$T_m = T_c + \frac{x_m q_i''}{k_f}$$

(4)

Where: k_c and k_f are clad and fuel thermal conductivity, respectively, [W/m -K]. The coolant temperature difference along each coolant channel can be obtained as:

$$T_{out} = T_{in} + \frac{Q}{\dot{m} \times c_p}$$

(5)

Where: T_{out} is the coolant temperature at the coolant channel outlet, [^oC], T_{in} coolant temperature at the coolant channel inlet, [^oC], Q heat removed from the channel, [kW], c_p specific heat of the coolant, [kJ/kg- K], and $\dot{m}$ coolant flow rate, [kg/s].

2. Emergency Tank Filling Mode: The reactor fuel is normally cooled by a forced downward circulation of coolant; the primary pumps also provide cooling of the reflector, the experiment tube structures, and the pool walls. Following loss of electric power to the primary pumps, the primary flow rate decreases, the pressure increases in the delay tank (DT) under the reactor pool, and the water level rises in the emergency tank (ET). The 7 m^3 ET is located in the fuel storage portion of the reactor pool and is connected to the DT (which is the exit path for coolant from the core) by a pipe; during normal operation, the ET has a low water level. During a loss of flow, the water level rise in the ET helps maintain the downward flow of coolant in the core for a longer time. The various relations among coolant flow and water level may be obtained by applying and solving the mass, momentum, and energy conservation equations. One such set of relations. The coolant velocity, V(t) [m/s], leaving the reactor pool as function of time, t [s], is assumed to decrease linearly in time during loss of primary pumps:

$$V(t)=a_o-b_o\times t$$
(6)

Where a_o and b_o are constants depending on the reactor configuration and fuel loading and t is the time, [s]. The values of the constants are set by knowing the velocity at t = 0 and requiring V = 0 at the time determined by experiment when the natural circulation values open (i.e., 79.90 s for LEU and water stops entering the ET in ~ 155 s.). [8]

This mode of cooling the reactor is not steady state mode, because the coolant velocity Equation.(6) decreases as time increases, therefore, coolant mass flow rate is calculated as follows:

$$\dot{m}(t)=\rho A_h V(t)$$
(7)

Where ρ is the reactor coolant density, [kg/m^3] and A_h hydraulic area of coolant channel, [m^2]

The unsteady state thermal hydraulic parameters of the reactor in this case were calculated using the equation (1) through equation (5).

3. Natural Convection Mode: Once the pressure in the DT increases to be the same as the pressure in the bottom of the reactor pool, the natural circulation valves (NCV) open, allowing a path for natural circulation of coolant from the DT up through the reactor core, down through the reactor pool, through the NCVs, to the DT. Two NCVs are provided for redundancy; the flow through either NCV is sufficient for reactor cooling. At some time during this transient, the pressure in the delay tank (DT, under the reactor pool) and the bottom of the reactor pool will become equal, and two natural circulation valves (NCV) will automatically open allowing natural circulation of coolant to remove the decay heat from the reactor fuel.

The level of water in the reactor pool is the same as the level of water in the emergency tank after the reactor was shut down.

Natural convection is observed as a result of the motion of the fluid due to density changes arising from the heating process (buoyancy force effect) a velocity field is set up with the fluid as a result of the buoyancy forces [9].

The heat transfer coefficient is related to the Nu number. There are many correlations that can be used to obtain a Nu number for natural convection heat transfer coefficient. The Nu number for the natural convection is a function of the Pr number and another dimensionless parameter called Grashof number (Gr), defined as:

$$Gr=\frac{g\beta(T_s-T)L^3}{v^2}$$
(8)

Where L is the core active length, [m], ν coolant kinematic viscosity= μ/ρ, [m²/s], μ coolant viscosity, [kg/m. s], g the acceleration gravity force, [m/s²], β coefficient of volumetric thermal expansion, $\beta=-\frac{1}{T}$, [K^{-1}].

Since there are two regions of flow in the coolant channel namely; developing and fully developed boundary layer, for developing boundary layer condition the local Nu number is given by:

$$Nu_z=0.508\ Pr^{0.5}(0.952+Pr)^{-0.25}Gr^{0.25}$$
(9)

And for fully developed boundary layer region, there are several of correlations for the Nu number with uniform heat flux condition at the fuel element surface. One of those correlations is widely used in like this channel as follow:

$$Nu_z=0.6(Gr\times Pr)^{1/5}$$
(10)

Equation (10), Is used for Nu determination in the laminer flow in the rang of $10^5<Gr^*Pr<10^{11}$

$$Nu_z=0.568\ (Gr\times Pr)^{0.22}$$
(11)

Equation (11) uses for Nu determination in the turbulent flow in the rang of $10^5< Gr^*Pr<10^{11}$, where, $Gr^*=Gr\times Nu$.

In natural convection, water flows from the delay tank through the core (upward flow condition), where it is heated and moves in the upward directions. The cold water flows in the downward direction through the natural circulation valves. Then, the two streams mix together and enter at the bottom of the core.

The core inlet coolant temperature is taken as the average temperature of the delay tank and reactor pool. The instantaneous average coolant temperature of the coolant water is calculated using the following equation:

$$\frac{\partial T}{\partial t}=-\frac{q'''}{\rho c_p}$$
(12)

$$q'''=\frac{Q(t)}{V_{WC}}$$
(13)

$$V_{WC}=V_{RP}+V_{DT}$$
(14)

Where Q is the core power, [W], V_{wc} the total coolant volume, V_{RP} the coolant volume in the reactor pool, [68 m^3], and V_{DT} the coolant volume in the delay tank, [36m^3]. Since, we assume that most of the hot water is contained in the top third volume of the delay tank then;

$$V_{WC}=V_{RP}+\frac{V_{DT}}{CF}$$
(15)

Rearrange equation (1) using equations (13), (14), and (15), we get:

$$\frac{\partial T}{\partial t}=-\frac{Q(t)}{\rho c_p\left(V_{RP}+\left(\frac{V_{DT}}{CF}\right)\right)}$$
(16)

Where: $\left(\frac{\partial T}{\partial t}\right)=T(t+\Delta t)-T(t)$, Δt is the Time difference, [s], CF the delay tank coolant part sharing in the cooling process of the reactor during natural cooling mode. Then:

$$T(t+\Delta t)=T_{in}(t)-\frac{Q(t+\Delta t)\Delta t}{\rho c_p\left(V_{RP}+\left(\frac{V_{DT}}{CF}\right)\right)}$$
(17)

The resultant of the fuel element heating due to the fission heat energy are the raising of the coolant temperature near the fuel element surface and forming of a buoyancy force due to the difference of the coolant densities, and the body forces resulting from the buoyancy give rise to free convection currents upward along the fuel element surface.

Using differential formulation method for natural convection method from the vertical fuel element of the Tajoura fuel element, the governing equation of continuity, momentum and energy equations for transient state and one dimensional flow (z-direction) we get the general solution for these equations the following explicit finite difference formulas. The general solution using finite difference technique for the coolant temperature and coolant velocity distribution of the Tajoura fuel coolant channel with $\Delta t \le \Delta z/u$ for stability of the method as follows:

$$T_{j+1}^{k+1}=\left(1-\frac{u_{j+1}^{k}\Delta t}{\Delta z}\right)T_{j+1}^{k}+\frac{u_{j+1}^{k}\Delta t}{\Delta z}T_{j}^{k}+\frac{\Delta t}{c_{p_{j+1}^{k}}\,\rho_{j+1}^{k}}q_j'''$$
(21)

$$u_{j+1}^{k+1}=u_{j+1}^{k}+g\beta\Delta t\left(T_s^{k+1}-T_\infty^{k+1}\right)-u_{j+1}^{k}\frac{\Delta t}{\Delta z}\left(u_{j+1}^{k}-u_j^{k}\right)$$
(22)

The unsteady states of the other thermal hydraulic parameters of the reactor in this case were calculated using equation (1) through equation (5). To perform the analysis and evaluate the temperature of the coolant, cladding and fuel using a NATIRT computer program is written in FORTRAN language. The program employs the discretization scheme shown in Figure (3) and the flow chart of the NATIRT program is shown in Figure (4).

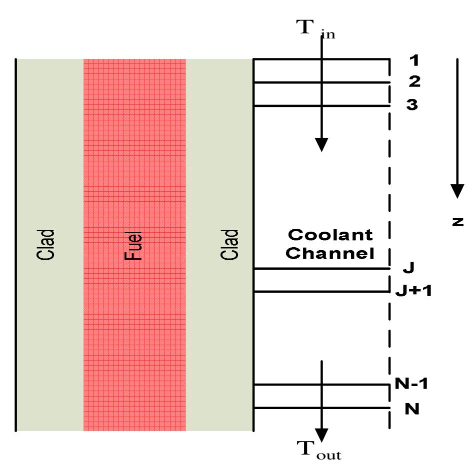

Fig. 3 Discretization of the hot channel of the 8TFA Modeling in Axial Direction

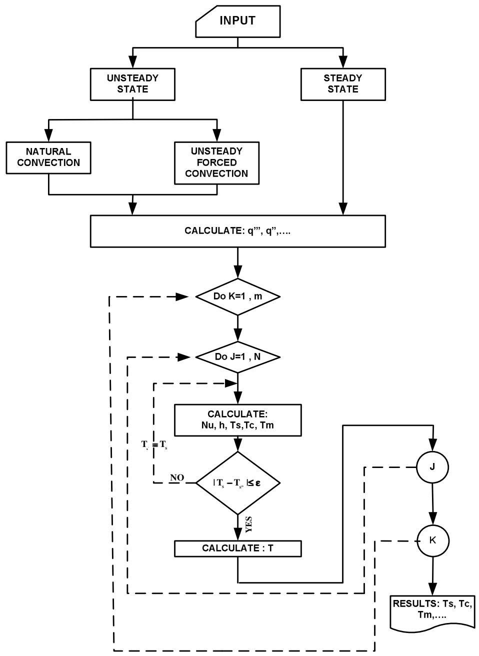

Fig. 4 NATIRT Program Flow Chart.

III. THEORETICAL RESULTS

The theoretical results for the hot channel in the hot cell (8TFA) is given in Table (1) the results for the hot cell in the hot channel (8TFA) only, which it is have the coolant velocity at the hot channel is equal to 3.08 m/sec, maximum heat flux 1.13 MW/m^2 at 10 MW and the other parameters are shown in the Table (1).

A. NATIRT Results

The results of NATIRT program in the case of steady state forced convection downward flow for the hot channel of the Tajoura reactor is shown in the Figure (4), where the maximum outlet temperature of channel coolant at the bottom of the channel is equal to 70.6 ^0C, cladding temperature is 113.1^0C, and fuel centerline temperature is 117.4^0C at the reactor power level of 10 MW (q$^"$ = 1.13 MW/m^2 on the lift side surface of fuel element No. 8 of the 8TFA).

The results of NATIRT program in the case of unsteady state forced convection downward flow for the hot channel of the Tajoura reactor at condition of primary pump cut power off is shown in the Figure (5), where the maximum outlet temperature of coolant at the bottom of the channel is equal to 70.6 ^0C, cladding temperature is 113.1^0C, and fuel centerline temperature is 117.4 ^0C at the reactor power level of 10 MW (q$^"$ = 1.13 MW/m^2 on the lift side surface of fuel element No. 8 of the 8TFA).

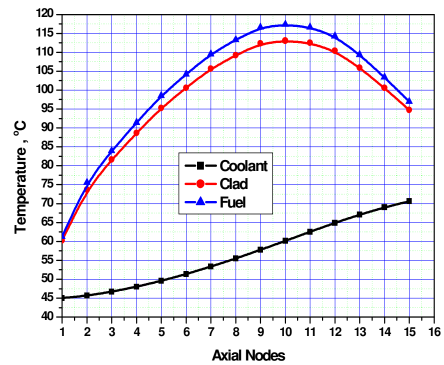

Fig. 4 NATIRT: The Axial Distribution of the Fuel Centerline, Clad Surface and Coolant Temperatures of the Hot Channel at steady state.

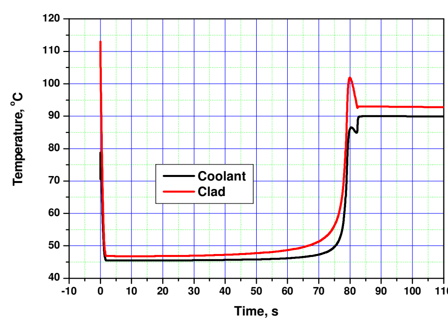

Fig. 5 NATIRT: The variation of the Clad Surface and Coolant Temperatures of the Hot Channel as a Function of Time during LOFA.

B. PARET Results

The results of PARET program in the case of steady state forced convection downward flow for the hot channel of the Tajoura reactor are shown in the Figure (6), where the maximum outlet temperature of coolant at the bottom of the channel is equal to 84.5 ^{o}C, cladding temperature is 114.9 ^{o}C, and fuel centerline temperature is 117.4 ^{o}C at the reactor power level of 10 MW (q'' = 1.13 MW/m^2 on the lift side surface of fuel element No. 8 of the 8TFA).

The results of PARET program in the case of unsteady state forced convection downward flow for the hot channel of the Tajoura reactor at condition of primary pump power cut off is shown in the Figure (7), where the maximum outlet temperature of coolant equal to 81.3 ^{o}C, cladding temperature is 111.3 ^{o}C, and fuel centerline temperature is 111.5 ^{o}C at the reactor power level of 10 MW (q'' = 1.13 MW/m^2 on the lift side surface of fuel element No. 8 of the 8TFA).

IV. COMPARISON OF RESULTS

The comparison of results for hot channel of Tajoura reactor at forced convection downward flow with different rated power levels at steady state condition are given in Table (2).These results of the these codes were in good agreements with some deviations due to the modeling methods were used in two codes (NATIRT, IRTCO [10]).

The unsteady state results of NATIRT and PARET codes for forced convection and natural convection during the pumps power is cut-off are given in Table (3) and Figure (8) and Figure (9), respectively. These results of the three codes were in good agreements with some deviations due to the modeling methods were used in two codes (NATIRT, and PARET).

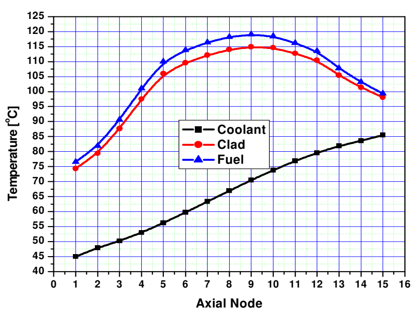

Fig. 6 PARET: The Axial Distribution of the Fuel Centerline Clad Surface and Coolant Temperatures of the Hot Channel.

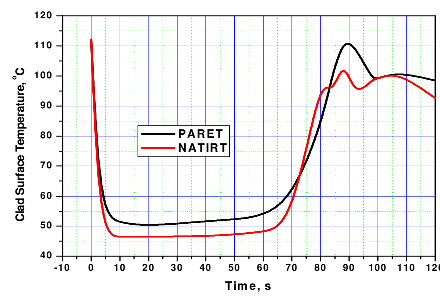

Fig. 8 Comparison of Clad Surface Temperature vs. Time after Primary Pumps Power Cut-off.

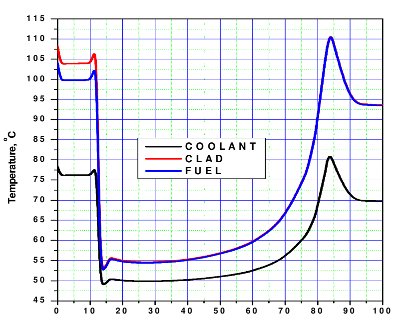

Fig. 7 PARET: The variation of the Fuel, Clad Surface and Coolant Temperatures of the Hot Channel as a Function of Time during LOFA.

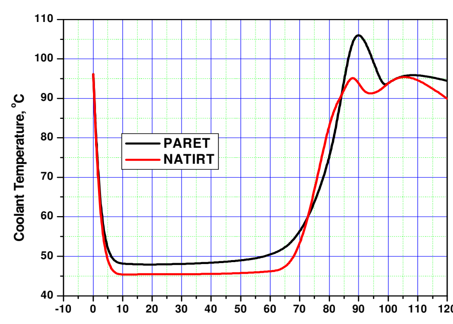

Fig. 9 Comparison of Coolant Temperature vs. Time after Primary Pumps Power Cut-off

Table 2.
FORCED CONVECTION DOWNWARD FLOW and STEADY STATE COMPARISON of RESULTS.

Power MW	Parameter	NATIRT	PARET	IRTCO	NATIRT/PARET	NATIRT/IRTCO
10	ONBF	1.25	1.18	1.22	1.10	1.02
	T_{fuel}, ^oC	117.40	117.43	116.06	0.99	1.01
	T_{clad}, ^oC	113.10	113.42	114.90	0.99	0.98
	$T_{coolant}$, ^oC	70.60	84.51	71.66	0.84	0.99
8	ONBF	1.45	1.42	1.45	1.02	1.00
	T_{fuel}, ^oC	106.30	104.30	103.70	1.02	1.03
	T_{clad}, ^oC	102.90	101.30	102.86	1.02	1.00
	$T_{coolant}$, ^oC	68.30	70.40	66.40	0.97	1.03
5	ONBF	2.18	2.14	2.11	1.02	1.03
	T_{fuel}, ^oC	84.70	83.43	83.74	1.02	1.01
	T_{clad}, ^oC	82.60	81.42	83.39	1.01	0.99
	$T_{coolant}$, ^oC	57.00	64.80	58.43	0.88	0.98

Table 3.
COMPARISON of RESULTS for UNSTEADY STATE FORCED CONVECTION DOWNWARD FLOW and NATURAL CONVECTION MODES. (NATURAL AFTER 180 s.)

Power MW	Parameter	NATIRT			PARET			NATIRT/PARET		
		10 s	79.9 s	Natural	10 s	79.9 s	Natural	10 s	79.9 s	Natural
10	T_{fuel}, ^oC	46.98	108.01	92.4	54.30	111.5	---	0.87	0.94	---
	T_{clad}, ^oC	46.79	107.80	92.3	54.00	111.3	98.44	0.87	0.97	0.94
	$T_{coolant}$, ^oC	45.47	89.98	89.8	49.20	81.3	94.44	0.92	1.11	0.95

V. CONCLUSIONS

From the previous results and analysis of the hot channel of the Tajoura reactor, we conclude that the results of the new computer program (NATIRT) were in good agreement with the results of the international computer program (PARET). At last, the calculated average error percentage between the two computer packages programs were in the range of 2% to 6%, which is due to the modeling procedures which were used in both codes.

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