THERMAL HYDRAULIC ANALYSIS OF TAJOURA REACTOR CORE DURING TRANSIENT STATE AFTER COST-DOWN OF COOLING PRIAMERY PUMPS
PART II: DURING THE FILLING OF THE EMERGENCY TANK

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الملخص

تمت في هذه الورقة محاكاة وتحليل للحالة العابرة المفترض حدوثها في قلب مفاعل تاجوراء مباشرة بعد إيقاف المفاعل نتيجة تعطل مضخات المبرد. هدف هذه الدراسة هو حساب العوامل الأساسية المتصلة بالسلامة وذلك بعد توقف المضخات والذي يؤدي مباشرة إلى انطفاء المفاعل وبدأ انسياب ماء التبريد خلال قلب المفاعل من حوض المفاعل إلى خزان الطوارئ والذى يوفر تبريد للمفاعل لمدة 89 ثانية بعد تعطل المضخات. قلب المفاعل بمركز البحوث النووية بتاجوراء هو من نوع IRT-4 ووقوده منخفض التخصيب (UO2-Al of 19.7 %) ومجموعات الوقود مصنعة على مجموعتين، احداهما تحتوي على 6 أنابيب والمجموعة الاخرى تحتوي على 8 أنابيب. الحد الأقصى للطاقة التشغيلية للمفاعل هو 10 ميجاوات. في هذه الدراسة للحالة العابرة تم تحليل وحدة الوقود الأكثر سخونة، حيث حددت الوحدة الأكثر سخونة بناء على نتائج دراسة سابقة للحالة المستقرة. أظهرت النتائج للحالة العابرة أن درجة حرارة مائع التبريد تزيد من 67.7 إلى حوالي 75 درجة مئوية في غضون 86.5 ثانية بعد إيقاف المفاعل كما أن درجة الحرارة للغلاف تنخفض من 105.5 إلى 70.6 درجة مئوية في غضون 12 ثانية من توقف المفاعل ثم تزيد إلى أقصى درجة حرارة لها مقدارها حوالي 89 درجة مئوية بعد 86.5 ثانية. تمت مقارنة نتائج الحالة العابرة أثناء ملء خزان الطوارئ مع النتائج التي تم الحصول عليها من مختبر أرجون الوطني (ANL). وأوضحت المقارنة أن هناك اختلافات معتبرة بين درجات حرارة غلاف الوقود التي تم حسابها في هذه الدراسة وتلك المتحصل عليها من مختبر أرجون الوطني (ANL) حيث أن نتائج هذه الدراسة أكثر منطقية لأن النموذج الرياضي لهذه الدراسة أستخدم فرضيات قليلة لمحاكاة الحالة العابرة في مفاعل تاجوراء بالمقارنة بتلك التي استخدمها ANL، كما أن باحثو مختبر أرجون الوطني لم يقوموا بمحاكاة فترة العشرة ثواني الأولى التي تلت إيقاف المفاعل.

ABSTRACT

Simulation and analysis has been performed for the transient state after core scram caused by postulated cost-down of all three operating primary pumps of the Tajoura nuclear reactor. The scope of this study is to calculate the core important safety related parameters during the filling of the emergency tank after scram. The reactor at Tajoura Nuclear Research Center (TNRC) is a pool type and the core is IRT-4M type. The core fuel is LEU UO2-Al of 19.7 % U^{235} . The fuel assemblies include 6-tubes and 8-tubes. The maximum operating power for the reactor is 10 MW. The hottest fuel-assembly is used in the analysis and the initial conditions for the transient state is taken from the results of previous study which was established for the steady state core operation. The results from the analysis of the transient state showed that the fluid temperature increases from 67.7 ^oC to about 75 ^oC within 86.5 seconds from the core scram, where the clad temperature drops from 105.5 ^oC to 70.6 ^oC within 12 seconds from the scram, then increases to its maximum temperature of about 89 ^oC at 86.5 seconds from the core scram. The results of the transient state are compared with the results obtained from Argonne National Laboratory (ANL). The comparison shows significant differences between the temperatures calculated in this study and those obtained from Argonne National Laboratory (ANL). The results from this model appear more logical than those of ANL. This is because of the fewer assumption made in this study compared with that of ANL, moreover ANL does not simulate the first 10 seconds follows the scram.

KEYWORDS: Research Reactor; TNRR; IRT-4M; Hot Assembly; Transient State.

INTRODUCTION

Tajoura reactor is a pool type with maximum power of 10 MW. The core consists of 16 fuel assemblies surrounded by a beryllium reflector and irradiation position formed by removing beryllium plugs. The previous core uses IRT-2M fuel assemblies containing high enriched uranium (HEU) with 80% enrichment. The assemblies of previous core have 3-tubes and 4-tubes, while the new core uses IRT-4M fuel assemblies containing low enriched uranium (LEU) with 19.7% enrichment. The current assemblies have 6-tubes and 8-tubes. The 6-tubes IRT-4M fuel assemblies are designed to use the existing control rods [1,2]. The maximum normal operating power for the reactor is 10 MW. Forced convection cooling of the core during normal operation is provided by three pumps operating in parallel. When the pumps stop working, the core is cooled by convection of water flows to the emergency tank for 89 second and latter on, the core is cooled by free convection of water in the reactor pool. Tajoura reactor has complicated shape fuel elements. Modifications are usually introduced to the codes used in the core calculations. WIMS code, CITATION code and the MCNP code are usually used for the neutronic calculation, while the PLTEMP code and PARET code are used for the thermal calculation. previous studies [3-4,7] calculated the important safety related parameters during steady-state operation of the Tajoura' Reactor. The scope of this study is to calculate the important safety related parameters during postulated transient case caused by cost-down of all three operating primary pumps. This transient is initiated by loss off-site power. In this study, the thermal hydraulic analysis for the transient state is considered and the important safety related parameters are calculated. Also a comparison is made between the results of this study and that of ANL.

REACTOR DESCRIPTION

Core description

The core consist of 36 cells, six cells contain 8-tube assemblies and 10 cells contain 6-tube assemblies while the rest of the cells are removable beryllium blocks [1,2], the assembly arrangement is presented in Figure (1).

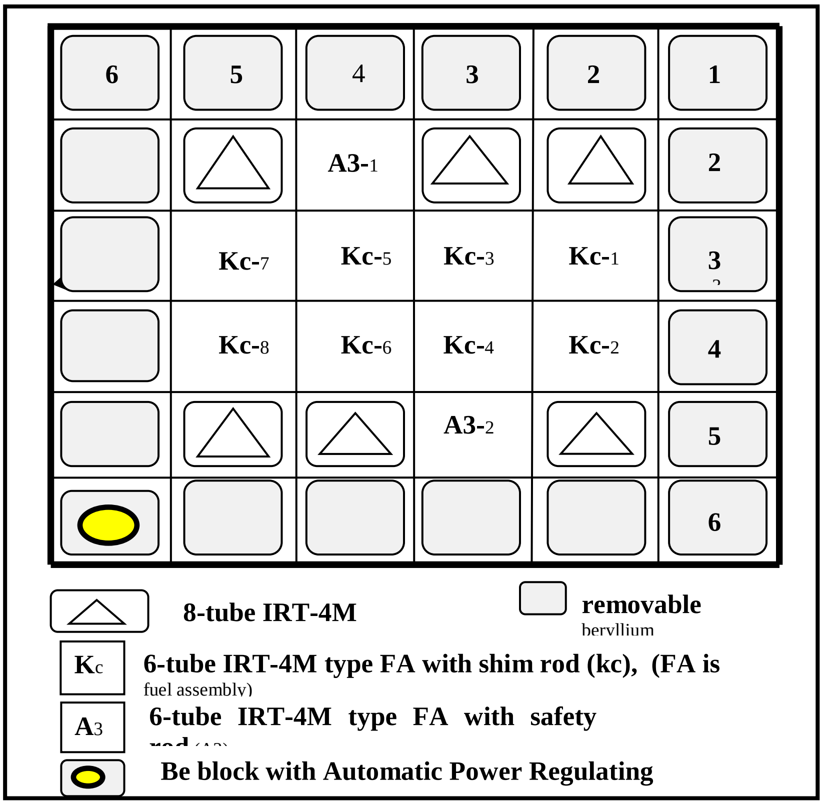

Figure 1: Core assemblies arrangement

The 8-tube-IRT-4M type FA consists of one central displacement tube, one fuel element is cylindrical shape and seven fuel elements are square shape as shown in Figure (2-a). The 6-tube IRT-4M type FA consists of central channel in which a control rod can be inserted and six fuel elements are square shape as shown in Figure (2-b).

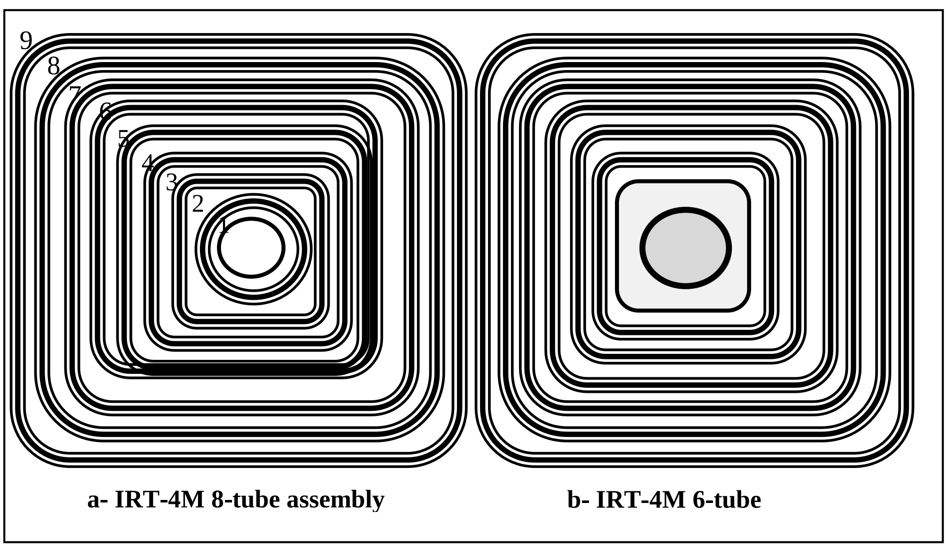

Figure 2: Cross section of IRT-4M fuel

The fuel material is UO_2- AL as a sandwich surrounded by aluminum clad type JAV-1 alloy. The assemble dimensions are presented in Table (1). The fuel elements are separated from each other by the coolant channel with thickness of 1.85 mm. The dimension of the fuel assemblies and other parameters of the core are listed in Table (2).

Table 1: Dimension of 8-tube fuel element[1,2]

Fuel no.	shape	Size (mm)		Corner radius (mm)	
		out	in	out	in
1	ring	21.3	18.1		
2	square	28.2	25	4.5	2.9
3	square	35.1	31.9	5.3	3.7
4	square	42	38.8	6.1	4.5
5	square	48.9	45.7	6.9	5.3
6	square	55.8	52.6	7.7	6.1
7	square	62.7	59.5	8.5	6.9
8	square	69.6	66.4	9.3	7.7

Table 2: Core Parameters Tajoura research reactor[1,2]

Parameter	(core - LEU)	Parameter	(core - LEU)
Fuel assembly design	IRT-4M	Uranium density in meat, gU/cc	2.77
U-235 Enrichment, wt%	19.7	Dispersant volume fraction , %	32.1
Fuel meat material	UO_2-AL	Uranium weight fraction, %	88.1
Fuel assemblies pith, mm	71.5	Meat/clad /element thickness,mm	0.7/0.45/1.6
No. of fuel tubes in assembly	6/8	Coolant channel thickness, mm	1.85
No. of assemblies in the core	10/6	Meat length, mm	600
U235 Loading /assembly, g	265/300	Assembly fuel meat volume, cc	483/551

Cooling system

The cooling system of the reactor consists of primary coolant loop which contain the pumps and heat exchanger. The heat exchangers exchange heat between the primary loop and secondary loop and then the heat is removed from the secondary loop to the third loop and finally to the cooling tower [1-2]. The emergency tank with a volume of 7 m^3 is connected to delay tank of 36.7 m^3 volume by a pipe of 0.18 m diameter and 5.10 m length as shown in Figure (3). When the pumps stop working the coolant will flow to the emergency tank from the reactor pool through the core. The flowing continues until the water levels in the emergency tank and the reactor pool are equal.

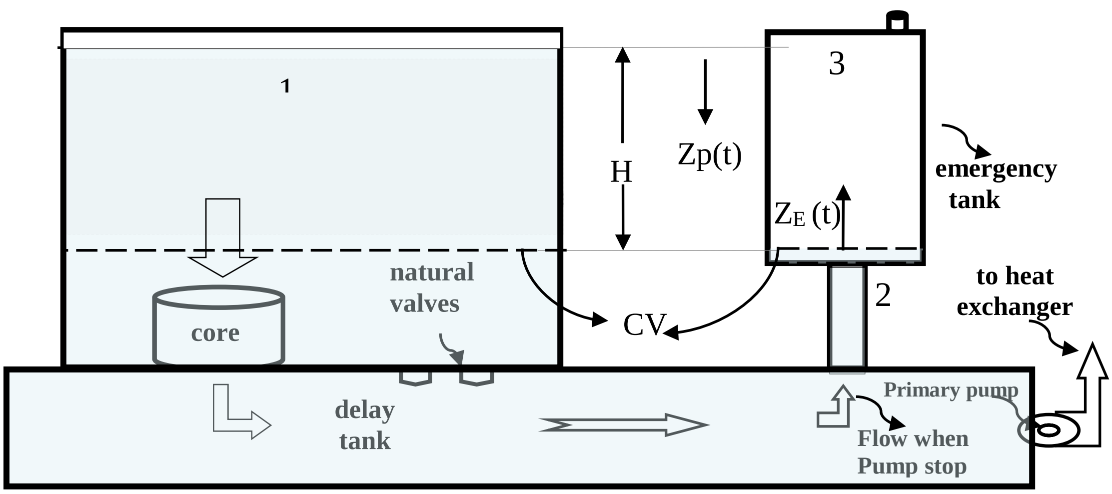

Figure 3: Cooling system during filling of the emergency tank

Technical data of the reactor

The important technical data for Tajoura research reactor are presented in Table (3).

Table 3: Technical data of the core of Tajoura reactor [1-2]

Parameter	value	Parameter	value
power	10 MW	Coolant core flow rate	533 m^3/h
Number of core Cells	36 (10 cells of 6 Tube-FA & 6cells of 8 Tube-FA)	8-tube flow	36 m^3/h
Fuel thermal conductivity	138 W/m ºC	6-tube flow	31.6 m^3/h
Clad thermal conductivity	187 W/m ºC	Core pressure drop	0.0657 MPa

THEORETICAL ANALYSIS

When the pumps stop operating due to a sudden electric cut-off, or due to any other reason, the reactor immediately shut down (scram) and the self sustaining fission chain reaction is terminated but a considerable amount heat continues to be generated by the decay of the fission products or by fission caused by delayed neutron, hence cooling of the reactor core must be provided for some time after shut down. The emergency tank in Tajoura reactor is designed to provide cooling to the core for certain time after scram, the emergency tank is connected to the a reactor pool as shown in Figure (3).

Decay heat after shutdown

After shutdown, the core continues to generate power (heat) as a function of time, the amount of such power generation (P_s) depends on the level of power before shutdown (P_o) and on the duration of time it operated at such level; since both of these factors determine the amount of fission product present. The equation governing the power after shutdown for uranium fuel is [5]:

$$P_s = P_o\,[0.1(t_s+10)^{-0.2} - 0.087(t_s+2\times10^7)^{-0.2}] + (-[0.1(t_s-t_o+10)^{-0.2} - 0.087(t_s+t_o+2\times10^7)^{-0.2}]) \quad (1)$$

Where t_o is the time of operation and, t_s is the time after shutdown in seconds. Providing coolant to the core is necessary after scram to maintain the clad surface temperature at a safe level and avoiding clad damage. The heat generation before shut down in the fuel cell is;

$$q_{cell} = PP_r \quad (2)$$

Where q_{cell} is the heat generation before shut down in the hottest cell, P is the core power before shutdown in watt and P_r is the relative power of the hottest cell. The heat generation rate in every fuel element (H_{GRi}) of the cell depends on the volume of the fuel element and calculated as,

$$H^{j}_{GR_i} = q_{cell}\,\frac{V_{Fi}}{V} \quad (3\text{-a})$$

the volumetric heat generation $q_i'''^{\,j}$ and the heat flux $q_i''^{\,j}$ are calculated as,

$$q_i'''^{\,j} = \frac{H_{GR_i}^{j}}{dz\, t_f\, d_i} \quad \text{and} \quad q_i''^{\,j} = \frac{H_{GR_i}^{j}}{A_i} \tag{3-b}$$

Where V_{Fi} is the volume of the i[th] fuel elements, *V* is the total volume of fuel elements, d_i is the fuel element width, t_f is the fuel thickness, A_i is the surface area of the i[th] fuel element and dz is the segment length.
A previous studies [3-4,7] found channel number 8 is the hottest channel, therefore the volumetric heat generation in channel 8 (see Figure (2-a) is given as:

$$q_{ch8}''' = q_{L7}''' + q_{R8}''' \text{ and the heat flux is } q_{ch8}'' = q_{L7}'' + q_{R8}'' \tag{4}$$

Where the subscripts L7 and R8 are the left hand side of fuel element 7 and right hand side of fuel element 8 respectively

Cooling system after scram

The emergency tank is subjected to the atmospheric pressure as shown in Figure (3). When the primary pumps suddenly stop working, and due to the elevation difference between the water levels in the reactor pool and the emergency tank, the coolant continues to flow through the core provides cooling to the reactor core for several seconds (89 s) after core scram. If a control volume as shown in Figure (3) is taken, the velocity entering the emergency tank may be found as a function of time by using the Bernoulli equation and continuity equation as;

$$\frac{P_1}{\rho} + \frac{u_1^2}{2} + gz_1 = \frac{P_2}{\rho} + \frac{u_2^2}{2} + gz_2 \text{ (neglect the friction)} \tag{5}$$

$$\frac{d}{dt}\int_{cV} \rho dV + \int_{cs} \rho \vec{u} d\vec{A} = 0 \tag{6}$$

Where P_1 is the reactor pool pressure (atmospheric pressure), P_2 is the emergency tank pressure (atmospheric pressure), u_1 is the velocity of the reactor pool, u_2 is the velocity entering the emergency tank.
since $V_p = A_p Z_p$, equation (6) becomes:

$$\frac{-A_p dz_p}{dt} = u_2 A_2 \tag{7}$$

The change in the volume of water in the reactor pool is equal to the change of volume of water in the emergency tank (see Figure 3), $\Delta V_P = \Delta V_E$, $\Delta V_E = \Delta Z_E A_E$ and $\Delta V_P = \Delta Z_P A_P$. Where V_E is the volume of water in the emergency tank, A_E is the cross-section area of the emergency tank and Z_E is the water level in the emergency tank and $\Delta z_P = H - Z_P$. Where H is the initial elevation difference between the water level in the reactor pool and that in the emergency tank, this gives:

$$z_P(t) - z_E(t) = z_p(t)\left(1+\frac{A_P}{A_E}\right) - \frac{HA_p}{A_E} \quad (8)$$

From the conservation of mass;

$u_1 A_1 = u_2 A_2$ and $u_1 = u_2 A_2/A_1$ (9)

Where $A_1=A_p$ is the cross-section area of the reactor pool, $u_1=u_p$ is the velocity of the water in the reactor pool, $A_2=A_t$ is the cross-section area of the tank, $u_2=u_t$ is the velocity entering the emergency tank. Substitute equation (9) into Bernoulli equation (5) where $P_1 = P_2 =$ atmospheric pressure, this gives:

$$u_2^2 = 2g(z_1 - z_2) + u_2^2 \frac{A_2^2}{A_1^2} \quad (10)$$

Where $Z_1=Z_P(t)$, $Z_2=Z_E(t)$, Substituting equation (8) into equation (10) gives:

$$z_P\left(1+\frac{A_p}{A_E}\right) - H\frac{A_p}{A_E} = \frac{u_2^2}{2g}\left(1-\frac{A_t^2}{A_P^2}\right) \quad (11)$$

Substituting equation (11) into equation (7), the velocity entering the emergency tank can be found as;

$$u_2 = \left(2g\frac{(CH-D)}{F}\right)^{\frac{1}{2}} - \frac{CA_t gt}{A_p F} \quad (12)$$

Where $C=1+A_P/A_E$, $D=H A_P/A_E$ and $F=1-A_t^2/A_P^2$

The mass flow rate during filling of the emergency tank is

$$\dot{m}_c = \dot{m}_E = \rho A_E u_E \quad (13)$$

Where $\dot{m}_c$ and $\dot{m}_E$ is the mass flow rate through the core and mass flow through emergency tank respectively.

The mass flow rate through the core is equal to the sum of the mass flow rate through the channels of the fuel assemblies and the mass flow rate through the Beryllium;

$\dot{m}_c = \dot{m}_{ass} + \dot{m}_{Be}$ and $\dot{m}_{ass} = 10\dot{m}_{6T} + 6\dot{m}_{8T}$

Where $\dot{m}_{ass}$ is the mass flow rate through channels of the fuel assemblies, $\dot{m}_{Be}$ is the mass flow rate through the berylliums, $\dot{m}_{8T}$ is the mass flow rate through the 8-tube channels and $\dot{m}_{6T}$ is the mass flow rate through the 6-tube channels. According to the hydraulic tests for Tajoura reactor core with LEU (16 FA) [1,2], the flow rate versus the pressure drop across the core is shown in Table (4).

Table 4: Flow rate through the core vs. pressure drop [1,2]

Flow rate (m^3/hr)	415	450	910	1190	1350
Pressure drop MPa	0.008	0.009	0.033	0.053	0.065

The flow rate at a pressure drop of 0.0657 MPa across the core is equal to 1350 m^3/h [1,2]. The mass flow rate through the core distributed between the 8-tube and 6-tube fuel assembly according to the area ratio and can be evaluated as:

$$\dot{m}_{8T}(t) = \dot{m}_{ass}(t)\frac{A_{8T}}{A_{ass}}$$

The mass flow rate in the fuel assemblies during filling of the emergency tank is:

$$\dot{m}_{ass}(t) = 0.395\ \dot{m}_c(t),\ \dot{m}_{8T}(t) = \frac{\dot{m}_{ass}(t)\ A_{8T}}{A_{ass}},\ \ \dot{m}_{Hcel}(t) = \frac{\dot{m}_{8T}(t)\ A_{Hcel}}{A_{8T}} \tag{14}$$

Where A_{ass}, A_{Hcel} and $\dot{m}_{Hcel}$ is the cross-section area of the assemblies channels, the cross-section area of the hottest cell channel within the 8-tube assembly and the mass flow rate of the hottest cell channel respectively.

The hottest channel is channel number 8 in the 8-tube fuel assembly where the maximum coolant temperature takes place and the clad surface temperature contact with this channel is the maximum, the mass flow rate and the velocity through channel 8 can be found as follows:

$$\dot{m}_{ch8}(t) = \frac{\dot{m}_{Hcel}(t)\ A_{ch8}}{A_{Hcel}} \quad \text{and} \quad u_{ch8}(t) = \frac{\dot{m}_{ch8}(t)}{\rho A_{ch8}} \tag{15}$$

Convection

The fundamental equation of convective heat transfer (q'') for both free and forced motion of the fluid is the so-called Newton´s law of cooling which may be written as follows [5,6]:

$$q'' = h\ A_s\ \Delta T,\ \text{where}\ h = Nu\ .\ k / D_e \tag{16}$$

Where h is the heat transfer coefficient, A_s is the area normal to the direction of heat transfer, ΔT is the temperature difference between the surface and the cooling fluid, *Nu* is the Nusselt number, D_e is the hydraulic diameter (m) and k is the conductivity of the fluid W/m.k

Forced convection correlation

For fully developed turbulent flow between parallel plate, the asymptotic value of the Nusselt number (Nu) is found by the Seider and Tate equation [5,6] is,

$$Nu = 0.027\ Re^{0.8}\ Pr^{\frac{1}{3}}\left(\frac{\mu}{\mu_w}\right)^{0.14},\ 0.7 \le Pr \le 16700\ ,\ Re \ge 10000 \tag{17}$$

And $P_r = c_p\ \mu / k$, $R_e = D_e\ \rho\ u / \mu$

Where *Re* is the Reynolds number, *Pr* is the Prandtl number, μ is the viscosity of the fluid (kg/s m), c_p is the specific heat of the fluid (J/kg. k), u is the velocity of the fluid (m/s) and ρ is the density of the fluid (kg/m^3). All the properties to be evaluated at bulk temperature except the μ_w which is evaluated at wall temperature.

Free convection in vertical channel

For a narrow channel the flow is fully developed and the velocity takes the parabolic shape Figure (4), so the parabolic velocity distribution (u) is given by[6]:

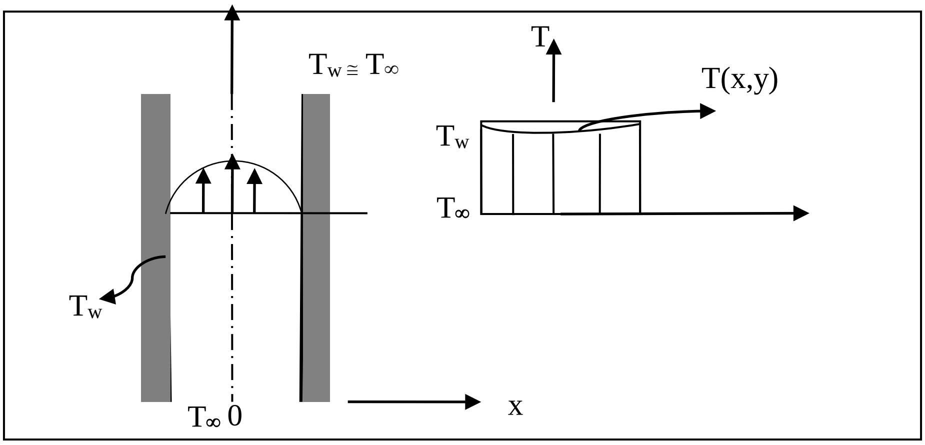

Figure 4: Vertical narrow channel fully developed flow

$$\mathrm{u} = \frac{g\beta\Delta T L^2}{8\nu}\left[1-\left(\frac{x}{(L/2)}\right)^2\right] \tag{18}$$

Where x is the distance from the center of the gap of the channel, L is the channel thickness and H is the channel height. The vertical mass flow rate per unit width is,

$$\dot{m}' = \frac{\rho g\beta\Delta T L^3}{12\,\nu} \tag{19}$$

Where $\beta = \frac{-1}{\rho_\infty}\left(\frac{d\rho}{dT}\right)_P$ is the volumetric thermal expansion coefficient, $\nu = \mu / \rho_\infty$ is the kinematic viscosity of the fluid, ρ_∞ is the density of reservoir fluid and L is the gap thickness of coolant channel. The total heat transfer per unit width is given by :

$$q' = m' c_p (T_w - T_\infty) = \frac{\rho g\beta c_p (\Delta T)^2 L^3}{12\nu} \tag{20}$$

the average heat flux is :

$$q'' = \frac{q'}{2H} \tag{21}$$

where

$q' = q'_L + q'_R$, the subscripts L and R are the Left side and Right side

from equation (20)and (21)we get

$$(\Delta T)^2 = \frac{24Hq''\nu}{\rho g\beta c_p L^3} \tag{22}$$

Substitute equation (22) into equation (18) the velocity of fully developed laminar flow between two vertical channels with unequal uniform heat fluxes is given by :

$$U = \left(\frac{g\beta LHq''}{6c_P\mu} \right)^{0.5} \tag{23}$$

Where L is the channel thickness and H is the channel height. The Nusselt number is $Nu = \frac{hH}{K}$ (24)

And $h = \frac{q''}{\Delta T}$ so $Nu_H = \frac{q''H}{\Delta T\ K}$ (25)

rearranging equation (22) and (25), the average Nusselt number can be given by:

$$Nu_H = \frac{\rho g\beta c_P L^3 \Delta T}{24K\nu} \tag{26}$$

Coolant and clad temperatures

The coolant temperatures distribution are found using the energy equation in one direction ;

$$\frac{dT}{dt} + u\frac{dT}{dz} = \frac{q'''}{\rho c_p} \tag{27}$$

The clad temperature T_s is calculated from equation (16) as,

$T_s = T_c + q'' / h\ A_s$ (28)

where q''' is the volumetric heat generation and T_c is the coolant temperature.

NUMERICAL SOLUTION OF THE TRANSIENT STATE

The hottest channel is divided into segments of size dz and appropriate interval of time Δt is considered during filling of the emergency tank. Using finite difference method, equation (27) can be solved and the coolant temperature $Tc^n_{(j+1)}$ during the filling of the emergency tank at node j+1 and time n calculated as follows;

$$T^n_{c\,(j+1)} = T^{n-1}_{c\,(j+1)}\left(1 - u^{n-1}_j \frac{\Delta t}{\Delta z}\right) + u^{n-1}_j \frac{\Delta t}{\Delta z} T^{n-1}_j + \frac{q'''^{\,n-1}_j \Delta t}{\rho^{n-1}_j c^{\text{n-1}}_{p\ \text{j}}} \tag{29}$$

The clad surface temperature Ts^n_{j+1} at node j+1 at time n can be calculated using equation (21) as: $Ts^n_{j+1} = T^n_{c\,(j+1)} + \frac{q''^{\,n}_{j+1}}{h^n_{j+1}}$ (30)

The velocity entering the emergency tank at time step n can be calculated using equation (12) as: $u^n_2 = \left(\frac{2g(CH - D)}{F}\right)^{\frac{1}{2}} - \frac{gCA_t t_{(at\ step\ n)}}{A_p F}$ (31)

The coolant velocity during the early stages of the filling of the emergency tank is high, the flow regime is turbulent and the heat is removed by forced convection. During the late stages, when the water level in the emergency tank approaches the level in reactor pool, the flow becomes mixed convection and then; at the last few seconds of the filling, it becomes purely natural convection. The natural convection velocity during filling of the emergency tank can be calculated using equation (23) as,

$$u_{Fd}^{n} = \left[\frac{q_{ch8}^{\prime\prime n}\, g\, \beta^{n}\, L\, H}{6\, \mu^{n}\, c_{p}^{n}} \right]^{0.5} \tag{32}$$

The dominant velocity may be determined by making a comparison between the forced convection velocities and the free convection. A computer program (FORTRAN) is constructed for transient state and used to calculate the important safety parameters after scram during the filling of emergency tank which include forced and free convection.

RESULTS AND DISCUSSION

The results include, decay power inside hottest cell, coolant velocity, mass flow rate, water level in the emergency tank, mass flow rate through all channels of the hottest cell, coolant velocity through the core, free and forced convection velocity, outlet coolant temperature, clad surface temperature, and heat transfer coefficient.

Decay Power

To calculate the decay heat within the core, the reactor was assumed to have operated continuously for one month at its maximum power of 10 MW. According to the equations (1 to 3), the decay power of the hottest cell after core scram is initially about 40.1 kW and decreases to 26.4 kW at the end of the filling of the emergency tank, as shown in Figure (5).

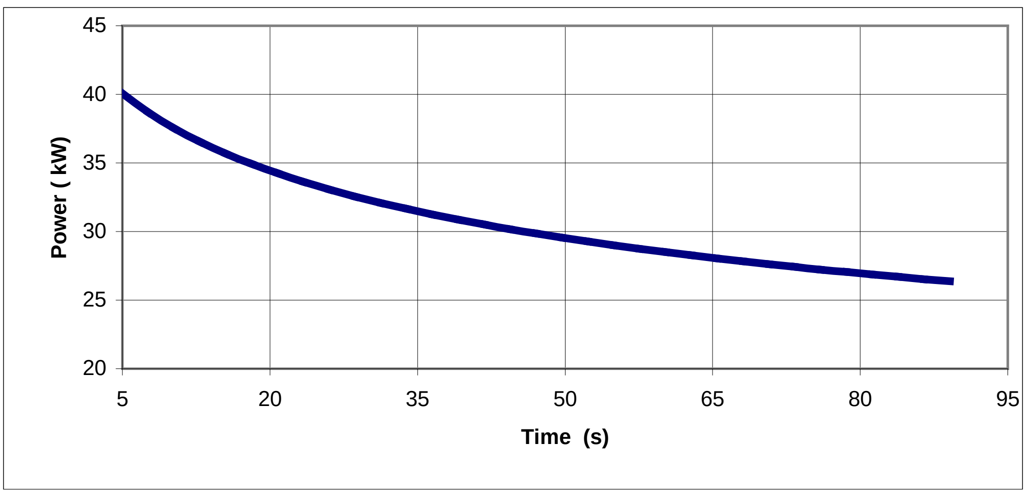

Figure 5: Hottest cell decay power during filling of the emergency tank

Mass flow rates through all channels of the hottest cell

The level of water in the emergency tank increases with time while the level in reactor pool decreases until the level of water in the emergency tank and that in reactor pool are equal. Figure (6) shows the levels variation with time after core scram.

The mass flow rates through the hottest assembly and cells are calculated according to equations (12 to 14) and presented in Figure (7). The flow rates decrease with time and become zero (no mass flow) when both water levels at reactor pool and emergency tank are equal. The figure shows that the flow in channel 8 is the highest while the lowest flow is in channel 1.

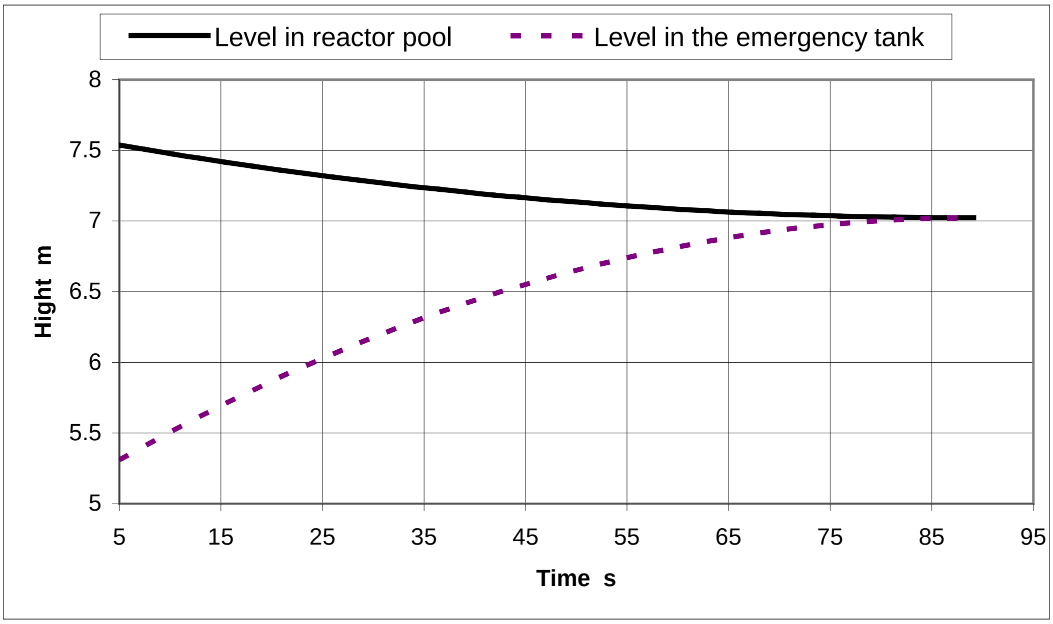

Figure 6: Water levels in the emergency tank and in the reactor pool.

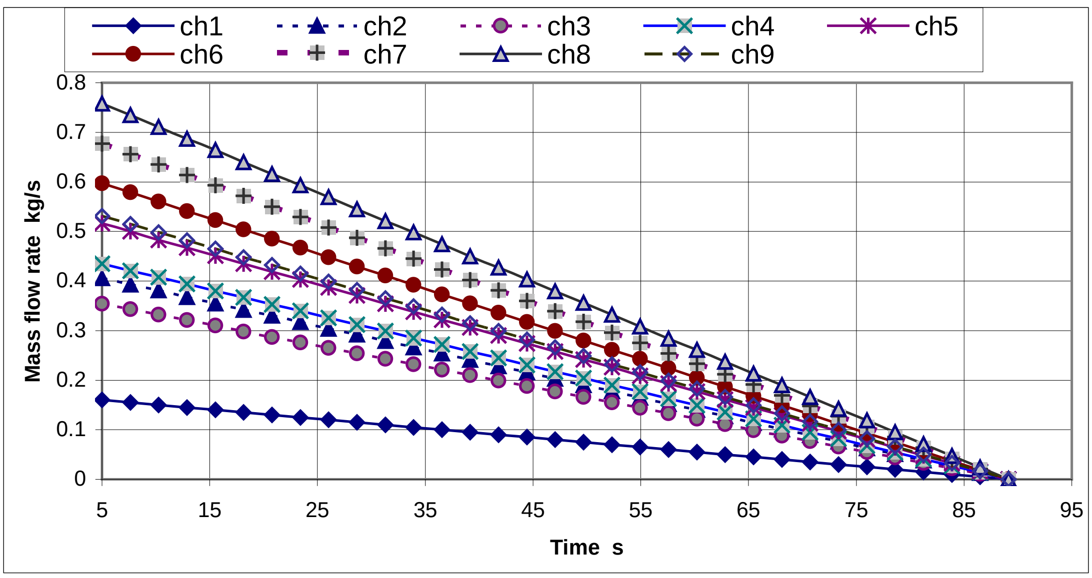

Figure 7: Mass flow rates through the channels of the hottest cell

Forced and free convection during filling of the emergency tank

The coolant velocity during the early stages of the filling of the emergency tank is high, the flow regime is turbulent and the heat is removed by forced convection. During the late stages, when the water level in the emergency tank approaches the level in reactor pool, the flow becomes mixed convection and then, at the last few seconds of the filling; (after 86.5 seconds from the core scram), it becomes purely natural convection, this behavior is presented via the values of upward and downward velocities of the coolant inside the channel as shown in Figure (8).

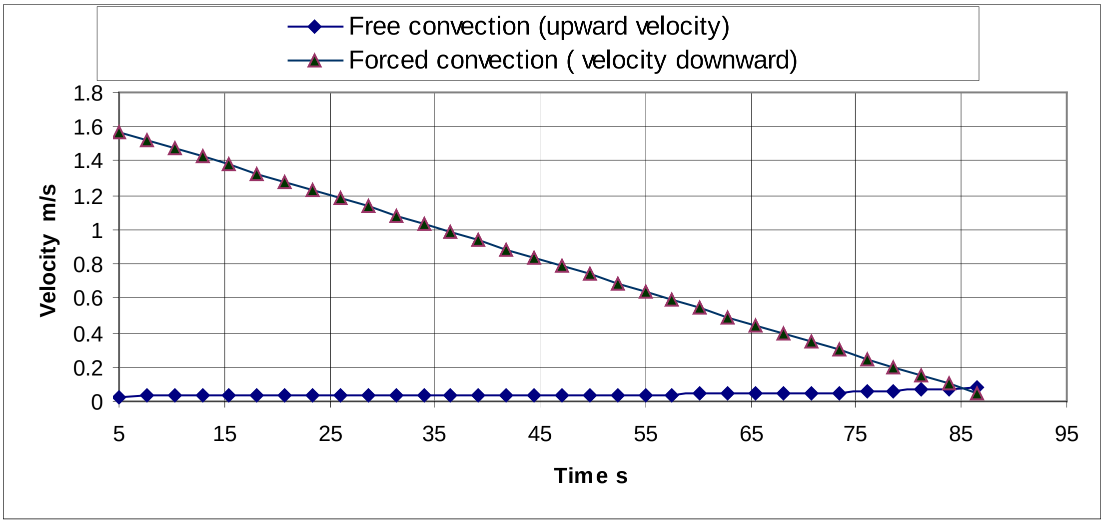

Figure 8: Forced and free convection velocities during filling of the emergency tank

Coolant and clad temperatures

A previous study [3,7] determined the hottest coolant channel in the core during the steady state operation. The channel number 8 (see Figure (2a)) is found the hottest channel. A summary of the important results are shown in Table (5).

Table 5: Summary of some results during steady state operation(hottest channel) [3,7].

Highest outlet coolant temperature	67.7 ^{o}C
Maximum clad surface temperature	105.5 ^{o}C
Maximum centerline fuel temperature	108.1 ^{o}C

The transient is initiated by loss of off-site power where all the three operating primary pumps are down and the coolant of the core is provided by the coolant flow to the emergency tank. According to the model of this study the flow through the emergency tank is assumed to be start immediately after scram. As shown in Figures (9, 10), the fluid temperature increases from 67.7 ^{o}C to about 75 ^{o}C at 86.5 s from scram, where the clad temperature drops from 105.5 ^{o}C to 70.6 ^{o}C within about 12 seconds from the scram, then increases to about 89 ^{o}C at about 86.5 seconds from the core scram, when the emergency tank is nearly full. The natural convection starts 3 seconds before the emergency tank becomes completely full. According to the analysis by Argonne National Laboratory[3], the coolant temperature drops to 52 ^{o}C within a time less than 4 second, and the clad temperature drops to 54 ^{o}C within the same time, (see Figures (9,10)) . ANL does not model the first 10 seconds after scram. The coolant and the clad gradually decreasing abut 2 ^{o}C within 40 seconds. As the flow rate approach zero after 80 seconds from the core scram, the temperatures of the fluid and the clad increases with high rate and within 8 seconds the temperatures peak to their maximum values of 108 ^{o}C for coolant and 114 ^{o}C for clad surface. Table (6) gives a summary of the maximum values of the outlet coolant temperature and the clad surface temperature during the filling of the emergency tank.

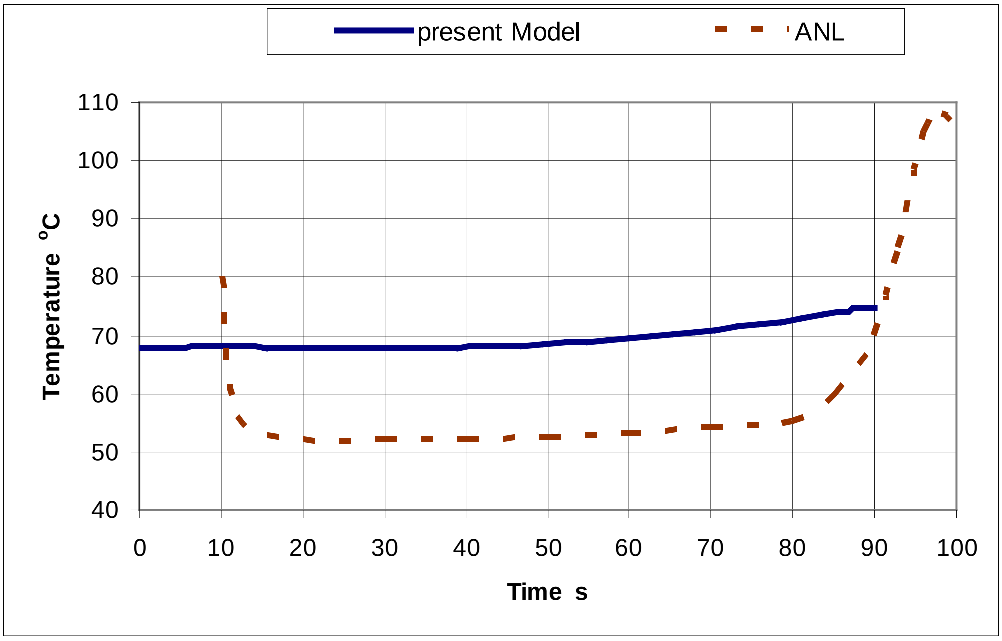

Figure 9: Coolant outlet temperature after scram

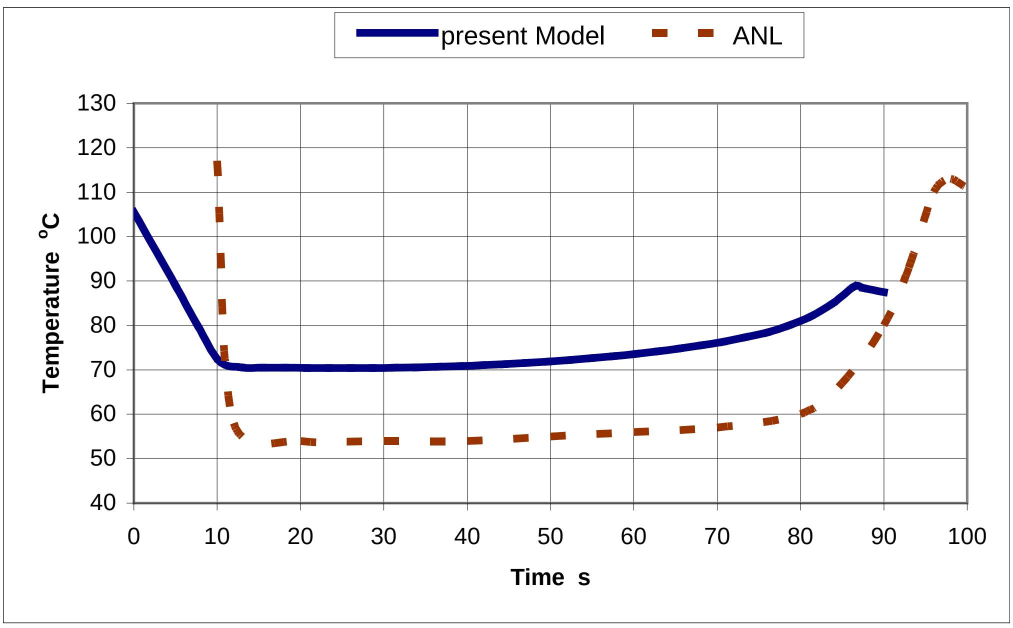

Figure 10: Clad surface temperature after scram

Table 6: Highest fluid and clad temperatures after core scram

Study	Model of this study	ANL
Maximum coolant temperature	≅ 75 °C	≅ 108 °C
Maximum clad surface temperature	≅ 89 °C	≅ 114 °C

The results from this model appear more logical than those of ANL. This is because of the few assumption made in this study compared with that of ANL, also in this study, a computer program is constructed especially for simulating the case of this study, where ANL use a PARET code which is need a number of assumptions must be adopted to simulate Tajoura reactor core.

CONCLUSION

The calculations were performed for the low- enriched uranium (19.7% UO2-Al) core with maximum power of 10 MW. The model established for this analysis simulates the transient state when the core is cooled by water flowing to the emergency tank immediately after scram. The model used for this analysis simulates the fuel tubes as parallel plates separated by their respective coolant channels. The hottest fuel assembly was used in the analysis. The following conclusion can be drawn:

- The emergency tank takes 89 s to be full.
- The free convection starts 86.5 s after scram.
- The highest outlet fluid temperature was found about 75˚C at 86.5 seconds from the core scram.
- The clad surface temperature reaches its maximum value about 89 ˚C at 86.5 second.

According to the fuel specification, the maximum temperature allowed for the clad surface is 102 ˚C. This value (89 ˚C) is still within the safety limit. Also this maximum value of clad temperature remains much lower than that temperature calculated by ANL which is equal 114 ˚C.

The results from this model appear more logical than those of ANL. This is because of the few assumption made in this study compared with that of ANL, also the ANL does not simulate the first 10 seconds after scram.

REFERENCE

[1] TNRC, internal report TNRC reactor. Report number R-cf-05-2006. TNRC, Tajoura (2006).

[2] Liaw J.R., Hanan,N.A. and Matos J.E. "Feasibility study for LEU conversion of the 10MW IRT-1 Research reactor at the Tajoura Nuclear Research Center in libya" RERTR program, Argonne national Laboratory, Argonne, IL 60439, USA (February 4, 2004)

[3] Garner P.L., Hanan N.A. "IRT-1 LEU core: Power densities for peak 8-tube FA" Argonne National Laboratory, Draft (September 2006).

[4] Abuhaffa H. M, “Thermal Hydraulic Analysis of the New Core of Tajoura Reactor Msc. Thesis, Nuclear Engineering Department, Faculty of Engineering, Tripoli university - Libya (2007).

[5] EL-wakil, M.M. "Nuclear heat transport", American Nuclear Society (1993)

[6] Incropera F. P. and Dewitt D. P. "fundamentals of heat and mass transfer", John Wiley and Sons, Inc.(1996).

[7] ابو حافة ح. م. "التحليل الهيدروحراري لقلب مفاعل الأبحاث في مركز البحوث النووية بتاجوراء" مجلة البحوث الهندسية، العدد السابع عشر (سبتمبر 2012 م)