TRANSIENT THERMAL PERFORMANCE OF ET-RR-II CORE

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ABSTRACT

During reactor scram or shutdown, the reactor power does not immediately drop to zero but falls off rapidly according to a negative period, eventually determined by the half-life of the longest-lived delayed neutron group. The objective of this work is to study the temperature distributions in fuel, clad, and coolant of the core of ETRR-2 during steady state (normal operation). The variations of this temperature distribution with time, in case of loss of flow accident with control plates insertion, are studied in order to observe the different limits of temperature values in each region. These limitations should be taken into consideration from the thermal hydraulic safety point of view. In case of the loss of flow accident due to a loss of power supply, pump failure, blockage or any other reason. It is very important for the thermal hydraulic designer to take into account that the coolant flow rate after LOFA (due to pump fly wheel) is enough to remove the residual heat (after scram) from the fuel to avoid fuel thermal damage. Through out this work, a mathematical model for representing the physical process that govern the heat transfer through the fuel, clad, and coolant in reactor core had been introduced. A computer code named TRANS was developed to solve the governing equations using Cranck-Nicklson finite difference scheme. The transients in fuel, clad and coolant had been simulated to study temperature variation within the core cell. The results show good agreement with the results of the code of the core designer company INVAP.

Keywords: Transient heat transfer, ET-RR-II, Fuel temperature transients, Safety, LOFA.

INTRODUCTION

For the successful implementation of nuclear research reactors, their design must be safe and their performance must be adequate under a variety of conditions. To achieve this goal, design and modeling of these systems must be performed. The previous efforts are mainly categorized into three classes. The first class deals with the lumped parameter model. R.J.Onega and K.E.Karcher[1]developed a nonlinear mathematical model for determining the dynamic response of a pressurized water reactor that incorporates both prompt and delayed temperature feedbacks. They assumed that in the PWR core (1)The mass flow rate of the coolant entering the core is not rapidly varying in time.(2)The feedback is determined by the fuel and coolant temperature coefficients of reactivity. The pressure coefficient of reactivity is assumed to be negligible.(3)A lumped parameter model of the core is an adequate representation. Only one node is used for the fuel and one for the coolant.(4) The effect of cladding is lumped with the fuel in the heat transfer coefficient calculations. The average time it takes for the heat to be transferred from the fuel to the coolant includes the cladding, but cladding temperature is not a state variable. They used a modified form of Hansen's method of largest eigen values to obtain the solution by transforming the governing equations into an integral equation then solving it iteratively. A comparison of the results of this model with those of Kerlin et al precise multi-nodes linearized model shows good agreement. T. W. Kerlin, E. M. Katz,J. G. Thakkar J.E. Strange[2]developed a mathematical model for dynamic response of the H. B. Robinson pressurized water reactor plant. The model was based on the basic conservation laws for neutrons, mass, and energy. The model included representation for point kinetics , core heat transfer, piping, pressurizer, and the steam generator. The system of nonlinear equations were solved using MATEXP code which uses a matrix-exponential-type of solution. Theoretical and experimental frequency responses were obtained from the model and the test data, the comparison showed that the model was capable of good predictions for reactivity perturbations and fair predictions for steam valve perturbations. A method was also demonstrated for using the test data for at-power determination of the differential control rod worth.

The second class deals with the one dimensional model. Kratwerk Union Erlangen (Federal Republic Of Germany)[3]developed a model for simulating the transient analysis of a 1000-MW gas cooled fast reactor, the emphasis being placed on the solution of one-dimensional unstationary helium flows. The fluid dynamics equations were solved one by one by a combination of implicit and explicit methods. In case of accident, the shutdown system is always activated. The results showed a necessity of backup pressures above 150KPa for depressurization accidents and a minimum circulator frequency of 5 Hz for the flow coast down accidents. M.Gaeta and F.Best [4] developed a transient thermal analysis model of a space reactor power to apply it to scenarios of interest. The scope of the simulation includes the thermal and neutronic behavior of a liquid metal cooled fast reactor. They assumed that: (1) The point kinetics equations with one delayed neutron group may be used assuming reactivity is constant during time step. (2) The core coolant model treats the coolant as one lumped node. It is assumed that the coolant is well stirred and has a uniform temperature. All thermophysical properties are assumed constant during time step. The treatment of conduction equation in the fuel is one dimensional in the radial direction. They also assumed that the point kinetic equations can be solved using the standard techniques for ordinary differential equations. The conduction equation is discretized by finite difference technique producing a matrix which is easily solved. A computer program that simulates the transient and steady state behavior of a space nuclear power system was named EETAP.

The third class deals with the two dimensional model. M. El-Genk and H. Xue [5] developed a two dimensional transient model to simulate steady state and transient operations of a single cell thermionic fuel elements (TFEs) for incore electric power generation in space nuclear reactor power systems. A major advantage of single cell TFEs is that they can be fully integrated in the reactor core and tested using electrical heaters. This nonnuclear testing provides confidence in the design and improves the system reliability, at much lower cost compared with nuclear testing. The general transient conduction equation in the solid region of the TFE was applied , the different solid regions of the TFE were divided into a number of finite volumes with axial and radial dimensions. A trapezoidal time integrator with a fully implicit finite difference scheme was used to calculate the transient temperature distribution. Model predictions were in good agreement with published data to within 4.5% and 5.5% for fission and electrically heated TFEs of the TOPAZ-II type, respectively.

THE REACTOR CORE

ETRR-II is a MTR (material test reactor) which has two main design characteristics : (1) It uses light water flowing inside a cylindrical housing tank. This water plays different roles as a moderator, as a coolant, and sometimes as a reflector.(2)Fuel elements are fuel plates, sometimes curved plates, conforming a 'sandwich' with an Aluminum alloy and enriched Uranium (meat) lined with an Aluminum cladding. These plates are placed inside an Aluminum frame, resulting in a fuel element (FE).The MTR cores are built with a variable number of FEs surrounded by one or more different types of reflectors . Some of the FEs are specially designed to house in the inside the control plates .The core is submerged in water. The ET-RR-II core consists of 29 FEs, each FE consists of 19 fuel plates. It contains 6 control plates.

TRANSIENT THERMAL MODEL

To form the heat balance governing equations, the following assumptions were used [6]: (1)The variation of flux in the axial direction is sinusoidal (i.e., a cosine function of y). (2)The maximum values of flux and volumetric source strength occur in a single fuel element at its center. (3) The thermal conductivities of fuel, clad, and coolant are constants.

The fuel plate of the MTR is shown in Figure 1. The governing transient heat conduction equations are given as:

For fuel region

$$(\rho c)_f \frac{\partial T_f}{\partial t} = k_f \frac{\partial^2 T_f}{\partial x^2} + k_f \frac{\partial^2 T_f}{\partial y^2} + q'''(x,y,t) \qquad (1)$$

For clad region

$$(\rho c)_c \frac{\partial T_c}{\partial t} = k_c \frac{\partial^2 T_c}{\partial x^2} + k_c \frac{\partial^2 T_c}{\partial y^2} \qquad (2)$$

The heat balance equation for the coolant region is given by

$$A_f (\rho c)_w \frac{\partial T_w}{\partial t} = hw(T_c - T_w) - mc_p \frac{\partial T_w}{\partial y} \qquad (3)$$

The initial conditions are steady state (normal operation) temperature distributions which are calculated by using the finite difference scheme and compared with the analytical solutions.

The boundary conditions are:

$$\frac{\partial T_f}{\partial x}(0,y,t) = 0 \qquad 0 \le y \le l_y \qquad (4)$$

$$\frac{\partial T_f}{\partial y}(x,0,t) = 0 \qquad 0 \le x \le l_f \qquad (5)$$

$$\frac{\partial T_f}{\partial y}(x,l_y,t) = 0 \qquad 0 \le x \le l_f \qquad (6)$$

$$k_f \cdot \frac{\partial T_f}{\partial x}(l_f,y,t) = k_c \cdot \frac{\partial T_c}{\partial x}(l_f,y,t) \quad 0 \le y \le l_y \qquad (7)$$

$$T_f(l_f,y,t) = T_c(l_f,y,t) \quad 0 \le y \le l_y \qquad (8)$$

$$\frac{\partial T_c}{\partial y}(x,0,t) = 0 \qquad l_f \le x \le (l_f + l_c) \qquad (9)$$

$$\frac{\partial T_c}{\partial y}(x,l_y,t) = 0 \qquad l_f \le x \le (l_f + l_c) \qquad (10)$$

$$k_c \cdot \frac{\partial T_c}{\partial x}(l_f + l_c, y, t) = h \cdot (T_c(l_f + l_c, y, t) - T_w(y,t)) \quad 0 \le y \le l_y \qquad (11)$$

$$T_w(0,t) = T_{in}(t) \qquad (12)$$

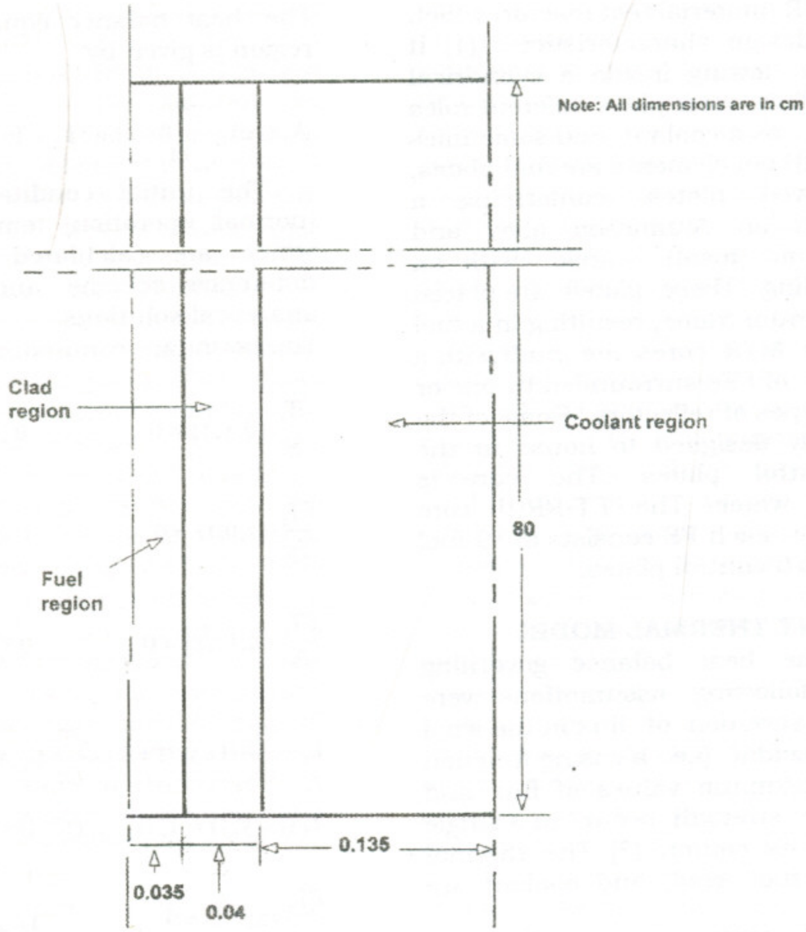

Figure 1 Fuel plate unit cell

THE VOLUMETRIC THERMAL SOURCE STRENGTH

The volumetric thermal source strength during the normal operation takes the following function:

$$q'''(x,y,0) = q_o''' \cdot \cos(\frac{\pi . y}{l_y})$$

and the volumetric thermal source strength after scramming is

$$q'''(x,y,t) = q_o''' \cdot \cos(\frac{\pi . y}{l_y}) \cdot FF(t) \tag{13}$$

where FF(t) is a fitting function derived from the deduced data of neutronic calculations of ETRR-II [7] and takes the following form:

$$FF(t) = EXP[-3.95247 + 0.41486 \cdot e^{\frac{-t}{12.29517}} + 0.83985 \cdot e^{\frac{-t}{264.952}}] \tag{14}$$

$$q_o''' = \frac{\pi \cdot POWER}{4 \cdot NFE \cdot NFP \cdot l_y \cdot w \cdot l_f}$$

THE NORMALIZED FLOW RATE FUNCTION

Flow coast down may result due to loss of electrical power, primary pump failure, primary coolant flow reduction(e.g. valve failure, blockage in piping or heat exchanger), fuel channel blockage, or coolant reduction in core by pass.

The normalized flow rate in case of loss of flow accident is shown in Figure 2. It is fitted [7]as a function of time as follows:

$$yy(t) = 1 - 0.036 \cdot t \qquad t \le 20s$$

$$yy(t) = 0.408 \cdot e^{-0.0188776 \cdot t} \qquad 20s < t \le 100s \qquad (15)$$

$$yy(t) = 0.0818 - 0.0002 \cdot t \qquad 100s < t \le 120s$$

$$yy(t) = 0.0578 \qquad t > 120s$$

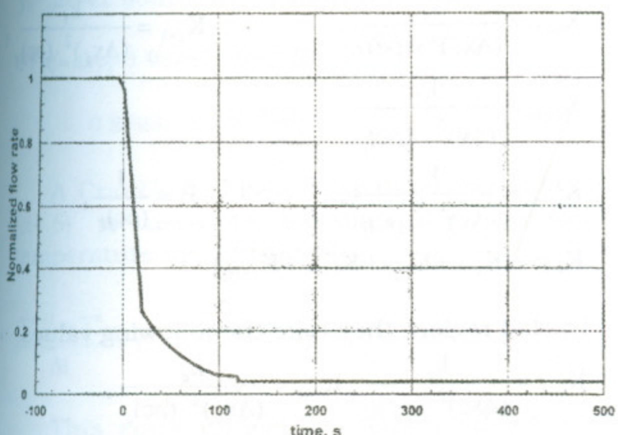

Figure 2 Flow coast down versus time in case of LOFA

HEAT TRANSFER COEFFICIENT CALCULATION

Heat is transferred by convection between clad surface and coolant. The regime of convective heat transfer depends on the dimensionless group Gr/Re^2 [8].

Convection may be forced, mixed or natural convection. The condition for specifying the regime is:

1. Forced convection $\frac{Gr}{Re^2} << 1$

2. Mixed convection $\frac{Gr}{Re^2} \approx 1$

3. Natural convection $\frac{Gr}{Re^2} >> 1$

The following correlations are going to be used to represent heat transfer coefficient in each regime where $h = \frac{k_w \cdot Nu}{D_e}$:-

Pure Forced Convection Regime

For laminar flow, the empirical correlation of Sieder and Tate [8] is used.

$$Nu = 1.86 \cdot (Re \cdot Pr \cdot \frac{D_e}{l_y})^{\frac{1}{3}} \cdot (\frac{\mu_w}{\mu_c})^{0.14} \qquad Re \le 2300$$

For turbulent flow, the used correlation known as the Dittus-Boelter [8] is:

$$Nu = 0.023 \cdot Re^{0.8} \cdot Pr^{0.4} \qquad Re \ge 10000.$$

For transition region between laminar and turbulent flows, linear Interpolation between turbulent and laminar flows heat transfer coefficients is employed.

Mixed Regime

For turbulent flow, the following correlation [8] is used

$$Nu = 4.69 \cdot Re^{0.27} \cdot Pr^{0.21} \cdot Gr^{0.07} \cdot (\frac{D_e}{l_y})^{0.36}$$

$$Re \ge 2000 \text{ and } Ra \cdot (\frac{D_e}{l_y}) \le 20000$$

$$\text{or } Re \ge 800 \text{ and } Ra \cdot (\frac{D_e}{l_y}) \ge 20000$$

For laminar flow , the following correlation [8] is used

$$Nu = 1.75 \cdot (\frac{\mu_m}{\mu_c})^{0.14} \cdot (Gz + 0.012 \cdot (Gz \cdot Gr^{\frac{1}{3}})^{\frac{4}{3}})^{\frac{1}{3}}$$

Pure Free Convection Regime

For turbulent flow, Churchill and Chu correlation is used [8]

$$Nu = (0.825 + \frac{0.387 . Ra^{\frac{1}{6}}}{(1 + (\frac{0.492}{Pr})^{\frac{9}{16}})^{\frac{8}{27}}})^2 \qquad Ra \ge 10^9$$

For laminar flow, Churchill and Chu correlation is used [8]

$$Nu = 0.68 + \frac{0.67 \cdot Ra^{\frac{1}{4}}}{(1+(\frac{0.492}{Pr})^{\frac{9}{16}})^{\frac{4}{9}}} \qquad Ra < 10^9$$

THE INLET COOLANT TEMPERATURE

According to INVAP data results [7], and using curve fitting data, the inlet coolant temperature after control plates insertion can be represented as follows :

$$T_{in}(t)=40.\,^{o}c \qquad t<20s$$
$$T_{in}(t)=50.\,^{o}c \qquad 20s \le t < 140s \qquad (16)$$
$$T_{in}(t)=(85.5-0.25.T)\,^{o}c \qquad 140s \le t < 180s$$
$$T_{in}(t)=40.\,^{o}c \qquad t>180s$$

Figure 3 shows a schematic diagram of this data.

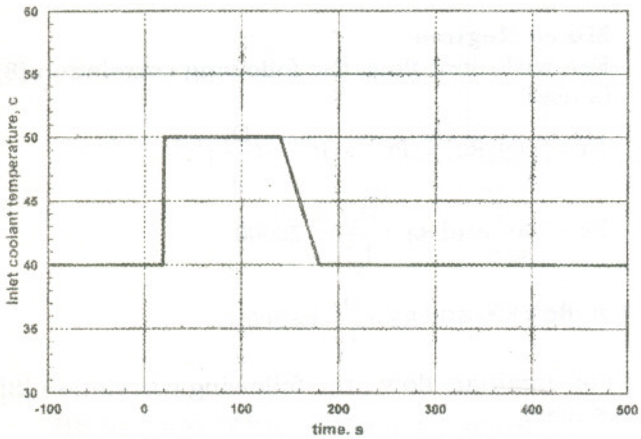

Figure 3 Inlet coolant temperature versus time after control plates insertion in case of LOFA

METHOD OF SOLUTION

The different solid regions are divided into a number of finite volumes with axial and radial dimensions. The dimensions of the control volumes are given by (Δx_f, Δy) for fuel region, (Δx_c, Δy) for clad region, and (ℓ_b, Δy) for coolant region. The control volume finite difference technique is used to represent the governing equations. The equations are transformed to a set of linear algebraic equations. For the radial direction, there are 11 nodes in fuel region, 11 nodes in clad region and one node in the coolant region. There are 11 nodes in the vertical direction. The equations take the form

$$\frac{dT_i}{dt} = K_{i-1} \cdot T_{i-1} + K_{i+1} \cdot T_{i+1} + K_{i-m} \cdot T_{i-m} + K_{i+m} \cdot T_{i+m} +$$
$$K_i \cdot T_i + P_i \cdot q'''(x_i, y_i, t) \qquad (17)$$

where K_{i-1}, K_{i+1}, K_{i-m}, K_{i+m}, K_i, P_i are the coefficients of each control volume and they take the following values in fuel region:

$$K_{i-1} = \frac{k_f}{(\Delta x_f)^2 \cdot (\rho c)_f}, \qquad K_{i+1} = \frac{k_f}{(\Delta x_f)^2 \cdot (\rho c)_f},$$

$$K_{i-m} = \frac{k_f}{(\Delta y)^2 \cdot (\rho c)_f}$$

$$K_{i+m} = \frac{k_f}{(\Delta y)^2 \cdot (\rho c)_f}, \qquad P_i = \frac{1}{(\rho c)_f},$$

$$K_i = -(K_{i-1} + K_{i+1} + K_{i-m} + K_{i+m})$$

In clad region, they take the following values:

$$K_{i-1} = \frac{k_c}{(\Delta x_c)^2 \cdot (\rho c)_c}, K_{i+1} = \frac{k_c}{(\Delta x_c)^2 \cdot (\rho c)_c},$$

$$K_{i-m} = \frac{k_c}{(\Delta y)^2 \cdot (\rho c)_c}$$

$$K_{i+m} = \frac{k_c}{(\Delta y)^2 \cdot (\rho c)_c}, P_i = 0.,$$

$$K_i = -(K_{i-1} + K_{i+1} + K_{i-m} + K_{i+m})$$

In coolant region, they take the following values:

$$K_{i-1} = \frac{h}{\ell_b \cdot (\rho c)_w},$$

$$K_{i-m} = \frac{(mc_p)_w}{\Delta y \cdot (\rho c)_w . w . \ell_b}$$

$$K_{i+m} = 0, P_i = 0., \; K_i = -(K_{i-1} + K_{i+1} + K_{i-m} + K_{i+m})$$

$K_{i+1} = 0\,, \qquad P_i = 0; \qquad K_{i+m} = 0$

$K_i = -(K_{i-1} + K_{i+1} + K_{i-m} + K_{i+m})$

and $q^{\backslash\backslash\backslash}(x_i, y_i, t)$ is the average volumetric thermal source strength after scramming within the control volume i. It takes the following form:

$$q^{\backslash\backslash\backslash}(x_i, y_i, t) = \frac{q_o^{\backslash\backslash\backslash} \cdot l_y}{\pi \cdot \Delta y} \cdot [\cos(\frac{\pi \cdot y_{ii}}{l_y}) - \cos(\frac{\pi \cdot y_{fi}}{l_y})] \cdot FF(t) \qquad (18)$$

where $\Delta y = y_{fi} - y_{ii}$

y_{fi}= upper boundary of control volume i
y_{ii}= lower boundary of control volume i

$$FF(t) = \exp[-3.95247 + 0.41486 \cdot e^{\frac{-t}{12.29517}} + 0.83985 \cdot e^{\frac{-t}{264.952}}]$$

A Cranck-Nicklson finite difference scheme (α=.5) is used to calculate the transient temperature distributions[8]

$$\frac{T^{n+1} - T^n}{\Delta t} = (1-\alpha) \cdot \dot{T}^n + \alpha \cdot \dot{T}^{n+1} \qquad (19)$$

This equation together with the discretised heat conduction equations are solved for the radial and axial temperature distributions. The boundary conditions are applied to these equations. The initial conditions are taken to be the steady state operational values which are determined using finite difference.
A computer code **TRANS** is developed to perform these calculations for each time step. It uses Gauss Siedel iteration technique to solve the set of linear equations at each time step. The heat transfer coefficient is iteratively determined each time step to account for temperature and flow changes.

INPUT DATA

Figure 3 shows the inlet coolant temperature as a function of time. The inlet temperature is constant during the first 20s. Then it rises up rapidly to 50°C. It stays constant for 120s then it decays back to 40°C. This data is taken from INVAP results. It is fitted and fed to the code.
Figure 4 shows the normalized fission power after scramming. It is fitted from INVAP neutronic results.

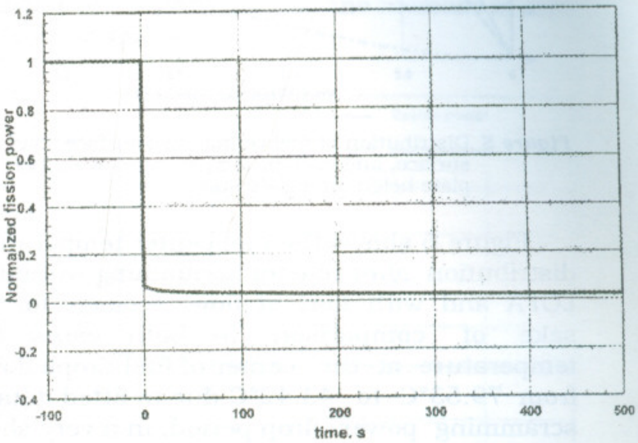

Figure 4 Normalized fission power versus time after control plates insertion

RESULTS AND CONCLUSION

Figure 5 shows the steady state temperature distributions in fuel center, fuel-clad interface, clad surface, and coolant. The data of the steady state solution is considered as the initial temperature distribution for the transient runs. To check the TRANS code, the steady state solution is compared to analytical solutions. The two solutions perfectly agree with each other.

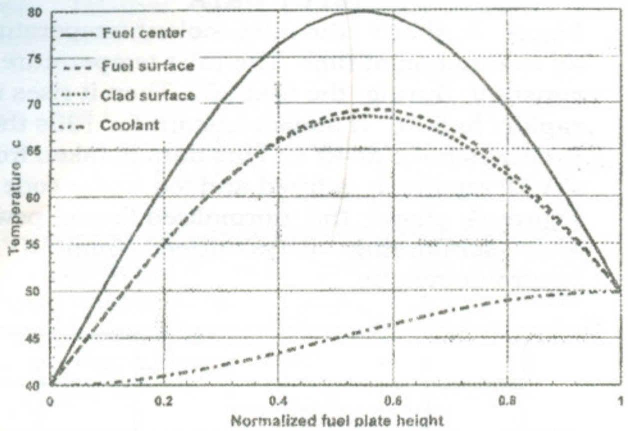

Figure 5 Distribution of fuel center, fuel surface, clad surface, and coolant temperatures with fuel plate height at steady state.

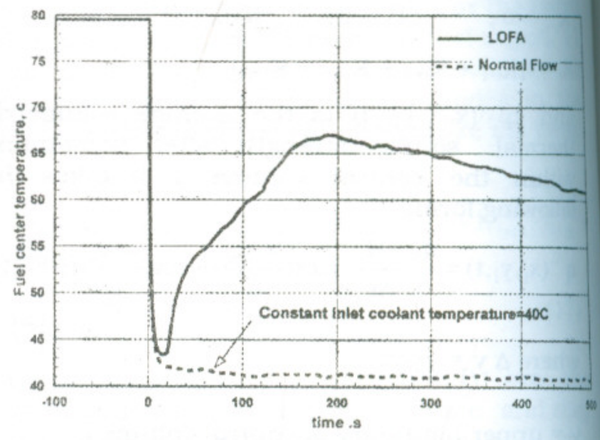

Figure 6 Fuel center temperature at position (y/l_y=0.5) versus time after control plates insertion

Figure 6 shows the fuel center temperature distribution after reactor scramming without a LOFA and with loss of flow accident for the sake of comparison. In both cases the temperature at the center of fuel drops down from 79.53°C to 43.42°C due to fitted data of scramming power drop period, in a very short period (20 seconds). In case of no LOFA the fuel center temperature continues to drop. While in case of LOFA, the temperature rises up again till it reaches a maximum value 67°C after 188 seconds from scramming then it drops down slowly. The rise up of fuel center temperature is due to the flow coast down as a function of time (shown in Figure 2). Using the fitted inlet coolant temperature data of INVAP the temperature reaches its maximum value after about 200 seconds. This period of time is the crucial time interval in safety calculations.

Figure 7 shows the clad surface temperature after scramming with and without flow coast down. The clad surface temperature shows the same behavior of fuel center temperature but the values of temperatures are lower than those of fuel center temperature due to thermal resistance of fuel. The clad surface temperature in case of flow coast down rises up to its maximum value of 66°C in about 200 seconds, then it goes down slowly.

Figure 8 shows the outlet coolant temperature versus time with and without flow coast down in case of control plates insertion. The figure shows that the outlet temperature follows the same behavior as fuel center and clad. The response to change in inlet coolant temperature is faster in coolant outlet temperature than the response in fuel center and clad. The maximum coolant temperature is at 140 seconds.

The oscillation in maximum outlet temperature occurs due to the mathematical representation of flow coast down. At 120 seconds the flow drops suddenly by a small amount due to opening of the flapper valve. This leads to oscillations at the outlet coolant temperature at this time but the oscillation die out due to the continuos mathematical representation of flow coast down after 120 seconds.

Figure 9 illustrates the normalized fission power generated after scramming compared to the normalized thermal power transferred from the core due to inlet /outlet temperature difference in case of flow coast down. The core thermal transferred power is calculated as $P_{thermal} = mc_p(T_{out}-T_{in})$.It can be noticed that fission power needs about 17 seconds to drop to 0.07 of the nominal power at steady state. The thermal transferred power then undershoots to negative values. The undershooting occurs because of the sudden rise up of the inlet coolant temperature in the data fed to the core. After 134 seconds, the two values agree for the rest of time.

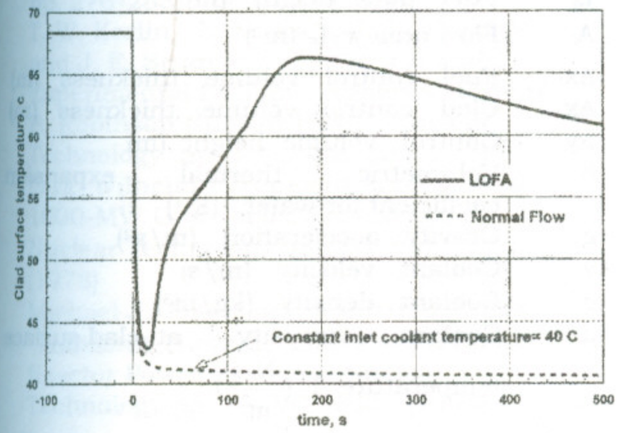

Figure 7 Clad surface temperature at position (y/l_y=0.6) versus time after control plates insertion

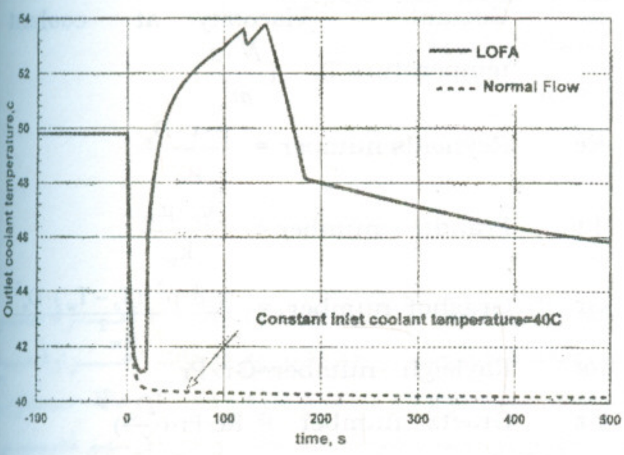

Figure 8 Outlet coolant temperature versus time after control plates insertion

Figure 10 shows the temperatures of outlet coolant, clad surface, and fuel center in case of control plates jamming i.e. nominal power continues to generate. The temperatures of fuel center, clad surface, and outlet coolant increase and after nearly 35s they reach to 142°C 130°C, 88°C respectively. These values are comparable to reported values of 135 °C, 134°C, 99°C reported by INVAP in the safety analysis report. In this report these values are sufficient to begin the onset of nucleate boiling (ONB). There is a difference in evaluating coolant temperature after 35 seconds because of the following two reasons (1) This temperature depends on fitted coast down data fed to the program.(2)The assumption that the inlet coolant temperature is constant is used to emphasis the effect of flow coast down only regardless of the increase in inlet coolant temperature.

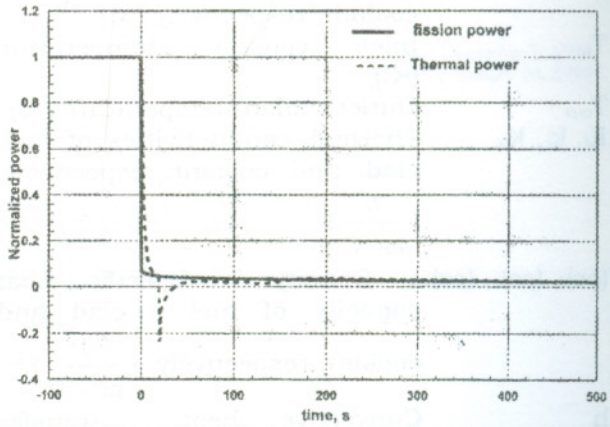

Figure 9 Normalized fission power, and core thermal transferred power versus time after control plates insertion in case of LOFA

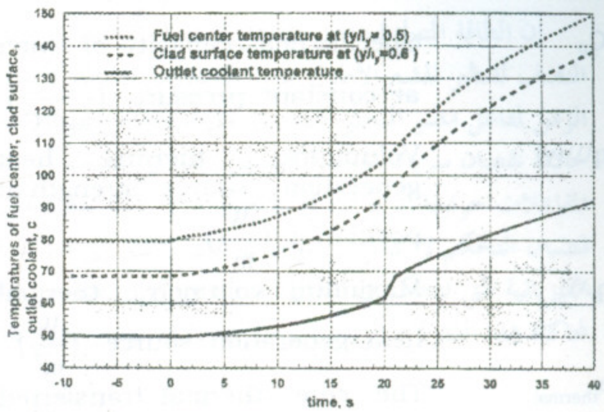

Figure 10 Fuel center, Clad surface, and outlet coolant temperatures versus time in case of control plates jamming with LOFA

The 130 °C of clad surface is enough to onset the nucleate boiling regime of the coolant. This, consequently, forms a vapor film which increase thermal resistance between clad surface and coolant. This time interval should be taken into consideration in safety analysis for the startup time and response time in emergency core cooling system design.

NOMENCLATURE

- t — Time (s)
- T_f, T_c, T_w — Temperatures of fuel, clad, and coolant respectively (°C)
- T_{in} — Inlet coolant temperature (°C)
- T_{out} — Outlet coolant temperature (°C)
- K_f, k_c, K_w — Thermal conductivities of fuel, clad, and coolant; respectively. $(\frac{W}{m \cdot {}^\circ C})$
- $(\rho c)_f$, $(\rho c)_c$, $(\rho c)_w$, — Density × Specific heat capacity of fuel , clad ,and coolant, respectively $(\frac{J}{m^3 \cdot {}^\circ C})$.
- h — Convective heat transfer coefficient of coolant $(\frac{W}{m^2 \cdot {}^\circ C})$
- m — Mass flow rate $(\frac{Kg}{s})$
- C_p — Specific heat capacity of water at constant pressure $(\frac{J}{kg \cdot {}^\circ C})$
- $q'''(y)$ — Volumetric thermal heat generation source strength at position y $(\frac{W}{m^3})$
- q_o''' — Maximum volumetric thermal heat generation source $(\frac{W}{m^3})$
- $P_{thermal}$ — The core thermal transferred power (w)
- w — Fuel plate width (m)
- L_f — Fuel half-thickness (m)
- L_c — Clad thickness (m)
- L_b — Half distance between two fuel plates (m)
- D_e — Hydraulic diameter (m)
- L_y — Fuel plate height (m)
- A_f — Flow area $w \times L_b$ (m^2)
- Δx_f — Fuel control volume thickness (m)
- Δx_c — Clad control volume thickness (m)
- Δy — Control volume height (m)
- β — Volumetric thermal expansion coefficient for water (K^{-1})
- g — Gravity acceleration (m/s^2)
- v — Coolant velocity (m/s)
- ρ — Coolant density (kg/m^3)
- μ_c — Coolant viscosity at clad surface temperature T_c $(\frac{N \cdot s}{m^2})$
- μ_m — Coolant viscosity at $(\frac{T_w + T_c}{2})$ temperature $(\frac{N \cdot s}{m^2})$
- μ_w — Coolant viscosity at coolant temperature T_w $(\frac{N \cdot s}{m^2})$
- Re — Reynolds number = $\frac{\rho \cdot v \cdot D_e}{\mu_w}$
- Pr — Prandtl number = $\frac{c_p \cdot \mu_w}{k_w}$
- Gr — Grashof number = $\frac{g \cdot \beta \cdot \rho^2 \cdot (T_c - T_w) \cdot D_e^3}{\mu_w^2}$
- Ra — Rayleigh number=Gr×Pr
- Gz — Graetz number = $Re \cdot Pr \cdot (\frac{D_e}{l_y})$
- Nu — Nusselt number
- NFE — Number of fuel elements in the core
- NFP — Number of fuel plates per fuel element
- POWER — Reactor total power (W)

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الأداء الحراري الانتقالي لمفاعل مصر البحثي الثاني

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ملخص البحث

إن الطاقة المتولدة داخل المفاعل لا تصل إلى الصفر عند إطفاء المفاعل ولكنها تنقص إلي كمية محددة وتستمر تبعا لفترة عمر النصف للنظير المشع الأطول عمرا الناتج من انشطار نوات اليورانيوم. يهدف هذا البحث إلى معرفة توزيع درجات الحرارة مع الزمن لمفاعل مصر البحثي الثاني في كل منطقة من مناطق عنصر الوقود سواء منطقة الوقود و غلافه و المبرد وذلك في حالة حدوث حادثة نقصان سريان المبرد الناتجة من عطل في المضخة أو انقطاع التيار الكهربي أو انسداد قنوات السريان . و من المهم لمصممي المفاعلات حراريا أن يؤخذ في الاعتبار - في حالة نقصان سريان - المبرد أن درجات الحرارة الناتجة عن الحرارة المتولدة بعد إنزال قضبان التحكم لا تصل إلى درجات عالية تؤدى إلي إتلاف الوقود أو تبخر المبرد. وقد تم عمل نموذج رياضي لوصف العمليات الفيزيائية التي تتحكم في انتقال الحرارة في كل منطقة من مناطق عنصر الوقود , كما تم عمل برنامج(TRANS) بلغة الفورتران لمحاكاة هذا النموذج الرياضي وحل معادلاته. وتبين نتائج هذا البرنامج إتفاق تام مع نتائج الشركة المصممة للمفاعل INVAP .