THERMAL-HYDRAULIC ANALYSIS OF LOSS OF FLOW ACCIDENT IN THE IAEA 10 MW MTR RESEARCH REACTOR

R. ALTAMIMI*, M. ALBATE

JORDAN RESEARCH AND TRAINING REACTOR, JORDAN ATMOMIC ENERGY COMMISSION

70 AMMAN, 11934 - JORDAN

OMAR S. AL-YAHIA

SCHOOL OF MECHANICAL ENGINEERING, KYUNGPOOK NATIONAL UNIVERSITY

80 DAEHAK-RO, BUK-GU, DAEGU 702-701, REPUBLIC OF KOREA

* CORRESPONDING AUTHOR: RAGAI.TAMIMI@JAEC.GOV.JO

ABSTRACT

A useful dynamic model for the analysis of flow inversion and establishment of natural circulation core transient of LEU fueled research reactors with scram enabled under natural circulation condition has been developed. The model includes point kinetics, single-phase thermal-hydraulics, and a one dimensional heat conduction based on lumped parameters method, with continuous feedback due to coolant and fuel temperature effect. The model predictions are qualified by comparing with the detailed calculations conducted in various institutions using different codes. The cladding peak temperature remains below the onset of nucleate boiling. The results are very encouraging and the model is useful for the analysis of most MTR research reactors encountered in practice.

Keywords— Flow Inversion, Lumped Model, Research Reactor, Runge-Kutta, Thermal-Hydraulics.

1. Introduction

Research reactors around the world are generally used as a source of neutrons; they are classified as non-power reactors. In contrast, power reactors are for electricity generation, co-generation, or heat generation. Research reactors are valuable tools which can supply high-intensity neutrons to be used for basic scientific studies as well as industrial applications, serving academia and research institutes. In addition, research reactors have served as important tools to train and educate personnel working in the nuclear field. There are many different types of research reactors, and plate type fuel research reactor is one of the most common ones. [1]

Most of the research reactors have a forced downward flow, designed to minimize the amount of nuclear radiation at the pool surface and to reduce the resistance during the shutdown rod insertion. At the same time, these reactors will experience a flow inversion in case of loss of flow accident due to a failure in the primary cooling pump that may leads to the most unfavorable thermal hydraulic condition. [2]

The flow inversion phenomena is a process in which the direction of the flow inside the core will invert, this happens for research reactors with forced downward flow cooling in the core. During this process the flow will go down to zero and then grow up again in the upward direction by natural convection. The aim of studying this phenomenon is its effect on the temperature of the cladding and the fuel that may lead to the Onset of Nucleate Boiling (ONB) during the inversion of the flow, where the degradation in the heat removal capability of the coolant occurs. And as the flow velocity goes down in the core channels, depending on the decay power of each channel, the coolant temperature will gradually increase until it exceeds a point at which steam voids appear on the surface of the clad and then a vapor slug along the channel appears.

The analysis of the transient behavior of research reactors has received a noticeable and recent attention because of its importance in determining the limits imposed by clad melting temperature. [3] Most of the existing computer codes for the commercial power reactors have a limited applicability on the research reactors with plate type fuel elements. It is often necessary to modify the codes for flow and heat transfer in narrow rectangular channels with coolant velocities, high heat flux, and high sub-cooled core conditions. [4]

The IAEA 10 MW MTR benchmark research reactor was analyzed by many institute and scientists to compare and verify different models used to study the reactor transients. [5]

Many engineering problems cannot be solved analytically, due to the complexity of the system equations. In this work, a modified fourth order Runge-Kutta method for better solution to the set of differential equations for the lumped model was used to investigate the transient thermal hydraulics during the process of flow inversion and establishment of natural circulation for the IAEA 10 MW research reactor.

2. Mathematical models

Dynamic model to analyze the transient behavior of MTR research reactors during LOFA have been done. The model based on; reactor point kinetics, single-phase thermal hydraulic, one dimensional heat conduction based on lumped parameters and the feedback due to the; void, doppler and moderator temperature effects.

The reactor power is calculated from a point reactor kinetics model with six groups of delayed neutrons. The neutron balance inside the reactor core is express as:

$$\frac{\partial Q}{\partial t} = \beta \frac{Q}{\Lambda}(\rho(t) - 1) + \sum_{i=1}^{6} \lambda_i C_i \quad (1)$$

$$\frac{\partial C_i}{\partial t} = \beta_i \frac{P}{\Lambda} - \lambda_i C_i \ , \quad \text{i=1,,, 6} \quad (2)$$

Where Q is the average power in the core, $\rho(t)$ is the reactivity, β is the delayed neutron fraction, Λ is the mean neutron generation time, C_i is the number of delayed neutron precursors in group i and λ is the precursor decay constant.

The initial conditions for these differential equations are;

Power at time zero is Q_0 and $c_i(0) = \frac{\beta_i}{\lambda_i \Lambda} Q_0$.

The continuous reactivity feedback can be expressed as:

$$\rho(t) = \rho_{in}(t) + \alpha_m\left(\rho_l(t) - \rho_{l,0}\right) - \alpha_f\left(T_f(t) - T_{f,0}\right) - \alpha_l\left(T_l(t) - T_{l,0}\right) \quad (3)$$

Where in the right hand side, the first term is the externally introduced reactivity and the other terms are the various feedback contributions. Those feedback contributions are introduced by changes of fuel temperature (Doppler effects), coolant temperature (spectrum effects only) and the density of the coolant (due to the changes in the temperatures) where they are multiplied by their reactivity feedback coefficients;α_f, α_l and α_m respectively. [6]

The heat transfer from the fuel element to the core coolant is expressed by a set of one-dimensional equations of heat conduction and convection. The temperatures were taken as an average value in the core components (lumped parameter) to simplify the governing equations. [7]

The heat transfer equation for the fuel element, cladding and for the channel coolant average temperatures with time can be written as:

$$\frac{\partial T_f}{\partial t} = -\frac{k_f}{ad\rho_f Cp_f}\left(T_f - T_C\right) + \frac{Q}{V_f\rho_f Cp_f} \tag{4}$$

$$\frac{\partial T_C}{\partial t} = \frac{k_c}{d(d-a)\rho_c Cp_c}\left(T_f - T_c\right) - \frac{h_s}{(d-a)\rho_c Cp_c}\ \left(T_c - T_l\right) \tag{5}$$

$$\frac{\partial T_l}{\partial t} = -\frac{u}{l}\left(T_l - T_p\right) + \frac{h_s}{b\rho_l Cp_l}\ \left(T_c - T_l\right) \tag{6}$$

Where the value of the surface heat transfer coefficient, h_s is written as:

$$h_s = Nu\left(\frac{k_l}{D_h}\right) \tag{7}$$

The value of the Nusselt number, Nu can be written as [8]:

$$\mathrm{Nu} = \begin{cases} 7.63 & Re < 2300 \\ 0.023Re^{0.8}Pr^{0.4} & Re > 3100 \\ Nu_{2300} + \left[e^{\left(\frac{(1766-Re)}{276}\right)} + \ {Nu_{3100}}^{-0.955}\right]^{-0.955} & 2000 < Re < 3100 \end{cases} \tag{8}$$

To analyze the pool temperature raise over the pool volume, a heat balance equation was used as:

$$\frac{\partial T_p}{\partial t} = \frac{Q}{V_p\rho_l Cp_l} \tag{9}$$

The flow inversion phenomena calculations start when the flow velocity decreases to 3 cm/s. At this velocity, the flow inversion model by a first order velocity differential equation written as follows start [9]:

$$\frac{\partial v}{\partial t} = -\frac{f}{2D_h}\left(\frac{\rho_m}{\rho_p}\right)\ |v|v + g\left(\frac{\rho_m}{\rho_p} - 1\right) \tag{10}$$

Where ρ_m is the density at mean coolant temperature and ρ_p is the density at pool temperature and v is the upward flow velocity.

The previous equations are all first order differential equations that can be solved numerically. For this purpose, fourth order Runge-Kutta method is used.

3. Numerical Method

In this work, fourth-order Runge-Kutta method has been used to solve the set of differential equations. Runge-Kutta methods are among the most popular ordinary differential equations solvers and based on using higher order terms of the Taylor series expansion. First studied by Carl Runge and Martin Kutta in 1900s, the modern developments are mostly due to john butcher in 1960s. One of the most widely used and efficient numerical integration methods is the fourth-order Runge-Kutta method. It is simple to implement and yields good numerical behavior in most applications. Also, it is generally recommended over Euler integration. [10]

The formula for the Runge-Kutta 4th order method is:

$$y_{i+1} = y_i + \frac{1}{6}\left(k_1 + 2k_2 + 2k_3 + k_4\right)h \tag{11}$$

Where:

$$k_1 = f(x_i, y_i)$$

$$k_2 = f\left(x_i + \frac{1}{2}h, y_i + \frac{1}{2}k_1 h\right)$$

$$k_3 = f\left(x_i + \frac{1}{2}h, y_i + \frac{1}{2}k_2 h\right)$$

$$k_4 = f(x_i + h, y_i + k_3 h)$$

Small time steps are required in the modeling for a good accuracy and maintaining stability. The advantage of this numerical method is allowing the controller of the time step to perform in a systematic manner and at each time step; the truncation error can be obtained. [4]

4. IAEA benchmark system description

The investigation of the flow inversion phenomena with the lumped model was carried out for the IAEA 10 MW light water, open-pool type MTR research reactor. It is a safety related benchmark problem for an idealized research reactor specified by the IAEA (1980, 1992) in order to constitute a verification test for the used codes in absence of validation against experimental measurements. It is performed for transients governed by the core kinetics and thermal-hydraulics. A forced downward circulation system was used for the light water coolant and moderator [1]. The LEU 10 MW IAEA research reactor core detailed specifications are given in table 1. ANSI/ANS-5.1-1973 was used to simulate the decay heat. [Ans-5.1 1973 4]

The core contain a total number of 25 fuel assembly, 21 are MTR standard fuel assemblies and the remaining four are control fuel assemblies, each standard fuel assembly contains 23 standard plates, whereas each control fuel assembly contains 17 standard plates and contains fork type absorber blades . The core is reflected by graphite on its two opposite sides and by light water by the two other sides.

Table 1 IAEA benchmark core specifications.

Core thermal–hydraulics		**Fuel assembly dimensions**	
Fuel thermal conductivity (w/cm k)	1.58	Length (cm)	8.0
Cladding thermal conductivity (w/cm k)	1.80	Width (cm)	7.6
Fuel specific heat (j/g k)	0.728	Height (cm)	60.0
Cladding specific heat (j/g k)	0.892	Number of fuel elements sfe/sce	21/4
Fuel density (g/cm3)	0.68	Number of plates sfe/sce	23/17
Cladding density (g/cm3)	2.7	Plate meat (mm)	0.51
Radial peaking factor	1.4	Width (cm) active/total	6.3/6.65

Axial peaking factor	1.5	Height (cm)	60.0
Engineering peaking factor	1.2	Water channel thickness (mm)	2.23
Inlet coolant temperature	38.0	Plate clad thickness (mm)	0.38
Operating pressure (bar)	1.7		

5. Results & Discussion

The relative flow during the loss of flow accident in the reactor core is exponentially related with time, and can be expressed as:

$$\frac{v(\mathrm{t})}{v_0} = \exp\left(-\mathrm{t}/\tau\right) \tag{12}$$

Where τ is the time constant for the pump coast down which equals 1 sec for fast LOFA and 25 sec for slow LOFA. At time zero, the reactor power is considered to be 120% of the nominal power.

For the FLOFA, the reactor trip signal is initiated when the flow rate reaches 85% of its initial value and the reactor scrams with a delay of 200 ms. Coolant temperature increases as the flow decreases before the reactor scram and it reaches its first peak of 59.23 ℃ after 0.47 seconds from the initiation of the accident. The cladding temperature increases with the increase of the coolant temperature and it reaches its first peak of 87.19 ℃ after 0.38 seconds from the initiation of the accident. The increase in the coolant, cladding, and fuel temperatures decreases the reactor power due to the temperatures and density feedback effects. With the scram of the reactor power, the coolant and cladding temperatures start to decrease as the decrease of power has more effect on the temperature than the loss of flow but for a certain time. With decrease of flow to a point where buoyance force becomes significant, the cladding and coolant temperatures starts to increase again as the effect of flow decrease is more prominent than the decrease in power. At 5.2 sec the flow inversion occurs and the coolant and cladding temperatures keep increasing until they reach their second peaks. The mass flow, bulk and Cladding temperatures, and relative power evolution during FLOFA is shown in figure 1 for the first 2 seconds and in figure 2 for the first 30 seconds from the initiation of the FLOFA accident.

For SLOFA, The mass flow, bulk and Cladding temperatures, and relative power evolution follows the same behavior as the FLOFA. Coolant temperature increases as the flow decreases before the reactor scram and it reaches its first peak of 62.49 ^{o}C after 4.27 seconds from the initiation of the accident. The cladding temperature increases with the increase of the coolant temperature and it reaches its first peak of 90.77 ^{o}C after 4.26 seconds from the initiation of the accident. In contrast to the FLOFA, the flow inversion happens in slower slope and takes longer time (54.9 sec) to reach the point of flow inversion. After the inversion, the coolant and cladding temperatures keep increasing until they reach their second peaks. The mass flow, bulk and Cladding temperatures, and relative power evolution during FLOFA is shown in figure 3.

Table 2 shows a comparison between the results obtained in this analysis and the main benchmark results for the Fast Loss of Flow Accident (FLOFA) and the Slow Loss of Flow Accident (SLOFA). The comparison shows a very good agreement between the results.

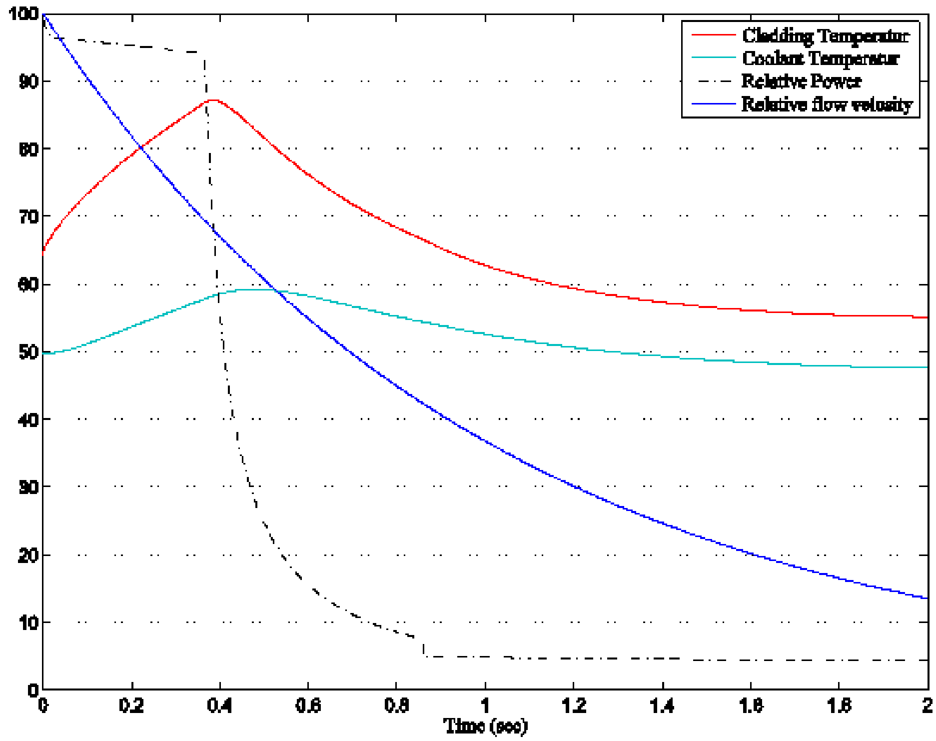

Figure 1 Cladding, bulk temperatures and relative power evolution during FLOFA

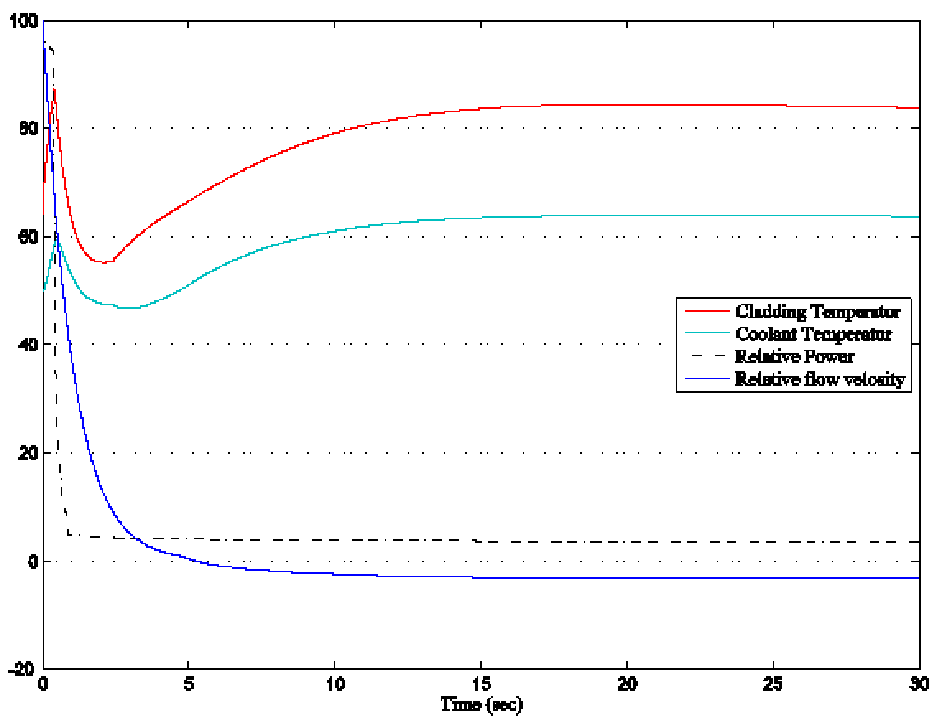

Figure 2 Cladding, bulk temperatures and relative power evolution during FLOFA

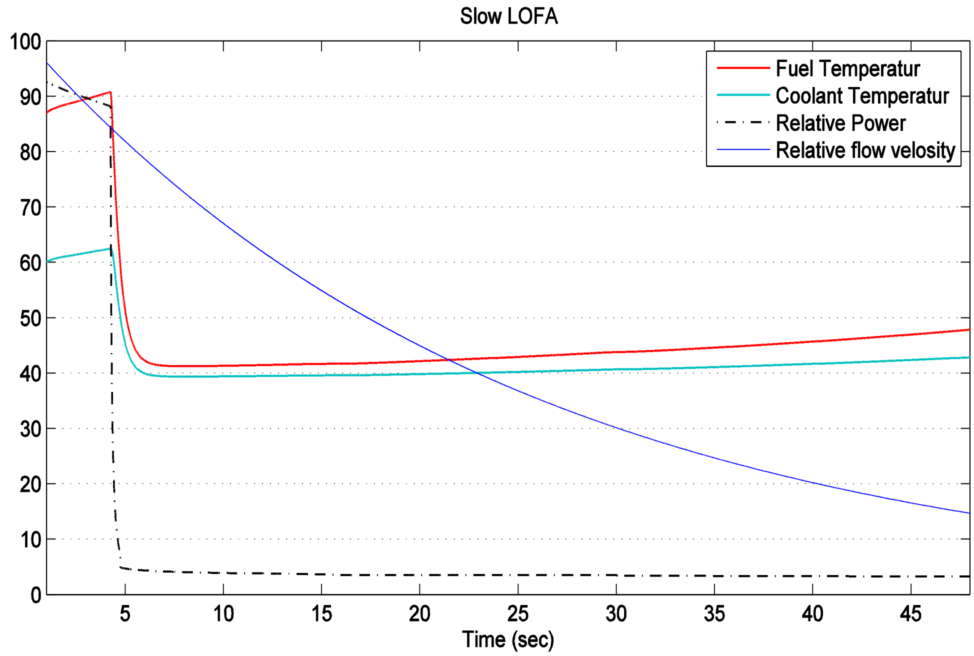

Figure 3 Cladding, bulk temperatures and relative power evolution during SLOFA

Table 2: **Comparison with main benchmark results for Loss of Flow Accident (LOFA).**

	Present study	ALYAHIA ET. AL	Kazeminejad NSRI	RELAP5/3.2 UPISA	PARET ANL	RETRAC-PC LAS	COSTAX-BOIL JEN	EUREKA-PT JAERI	COBRA III-C INTERATOM
FAST LOFA									
Power at scram	11.30 MW	11.54 MW	11.26 MW	11.83 MW	11.86 MW	11.72 MW	11.67 MW	NA	11.4 MW
	(0.35)	(0.36)	(0.37)	(0.19)	(0.30)	(0.19)	NA	NA	(0.36)
1st cladding temperature peak	87.19^{o}C	90.30^{o}C	90.20	92.58^{o}C	89.46^{o}C	87.92^{o}C	93.9^{o}C	97.1^{o}C	89.3^{o}C
	(0.38)	(0.35)	(0.38)	(0.40)	0.505)	(0.40)	(0.37)	(0.40)	(0.36)
1st coolant temperature peak	59.23^{o}C	59.15^{o}C	63.20	59.50^{o}C	60.84^{o}C	59.92^{o}C	59.3^{o}C	58.1^{o}C	56.4^{o}C
	(0.47)	(0.38)	(0.49)	(0.50)	(0.60)	(0.47)	(0.43)	(0.48)	(0.46)
Flow inversion	5.2 sec	4.28 sec	4.60	7.4 sec	4.415 sec	7.36 sec	NA	NA	NA
2nd cladding temperature peak	84.45^{o}C	91.95^{o}C	107.30	120.73^{o}C	105.9^{o}C	128.25^{o}C	NA	95.2^{o}C	NA
	(17.71)	(8.95)	(15.25)	(10.00)	(8.51)	(7.36)	NA	(10.00)	NA
2nd coolant temperature peak	63.9^{o}C	73.03^{o}C	101.60	105.3^{o}C	101.68^{o}C	69.76^{o}C	NA	49.3^{o}C	NA
	(16.32)	(10.18)	(17.97)	(11.90)	(9.14)	NA	NA	(10.00)	NA
SLOW LOFA									
Power at scram	10.58 MW	11.25 MW	11.1 MW	11.56 MW	11.64 MW	11.56 MW	11.7 MW	NA	11.46 MW
	(4.24)	(4.28)	(4.28)	(4.10)	(3.86)	(4.05)	(4.06)	NA	(4.26)
1st cladding temperature peak	90.77^{o}C	86.97^{o}C	84.90^{o}C	88.41^{o}C	84.56^{o}C	84.63^{o}C	90.3^{o}C	96.1^{o}C	85.5^{o}C
	(4.26)	(4.03)	(4.29)	(4.30)	(4.07)	(4.24)	(4.27)	(4.2)	(4.26)
1st coolant temperature peak	62.49^{o}C	57.52^{o}C	61.60^{o}C	57.97^{o}C	58.83^{o}C	58.82^{o}C	58.1^{o}C	57.5^{o}C	55.4^{o}C
	(4.27)	(4.25)	(4.31)	(4.30)	(4.08)	(4.27)	(4.3)	(4.3)	(4.26)
Flow inversion	54.94 sec		57.56 sec	57.40 sec	62.84 sec	57.26 sec	NA	NA	NA

(): time in sec. NA: not available.

6. Conclusion

The present study reports one dimensional dynamic model for the analysis of the flow inversion scenario associated with loss of flow accident and the establishment of the natural circulation in the IAEA 10 MW research reactor core.

The scenarios of fast and slow LOFA were considered. In which, the pump coast down is decayed as exponential function with time constant of 1 second and 25 seconds . It is assumed that the reactor is operated at a (120%) of its full power when the transient starts. Moreover, when the flow decreases to 85% of the initial flow rate, control plates scram with delay time of 200 ms.

The results of the study were compared with other results reported with different codes. Generally, good agreement was achieved, which indicated that the developed lumped model is proper to thermal hydraulic analysis and safety limits of plate type research reactors, as the IAEA 10 MW MTR reactors.

7. Acknowledgement

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Nomenclature

a	Fuel half thickness, m	t	Time, sec
b	Channel half thickness, m	v	Flow velocity, m/sec
C_i	i^{th} group precursor density	V	Volume, m^3
Cp	Heat capacity, j/kg.k	***Greek symbols***	
D_h	Hydraulic diameter, m	α	Reactivity feedback coefficient
d	Plate half thickness, m	β	Average delayed neutron fraction
g	Gravitational acceleration, m/sec^2	β_i	i^{th} group of delayed neutron fraction
h_s	Surface heat transfer coefficient	ρ	Reactivity or density
K	Thermal conductivity, w/mk	Λ	Prompt neutron life time
L	Channel height , m	***Subscripts***	
Nu	Nusselt number	*b*	Bulk
Pr	Prandtl number	*c*	Cladding
Q	Power, w	*f*	Fuel
Re	Reynolds number	in	Insertion
T	Temperature, C	*l*	Coolant