MATHEMATICAL MODELING OF CANDU-PHWR

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ABSTRACT

The paper deals with the transient studies of CANDU 600 pressurized Heavy Water Reactor (PHWR). This study involved mathematical modeling of CANDU-PHWR to study its thermodynamic performances. Modeling of CANDU-PHWR was based on lumped parameter technique. The reactor model includes the neutronic, reactivity, and fuel channel heat transfer. The nuclear reactor power was modelled using the point kinetics equations with six groups of delayed neutrons and the reactivity feed back due to the changes in the fuel temperature and coolant temperature. The CANDU-PHWR model was coded in FORTRAN language and solved by using a standard numerical technique. The adequacy of the model was tested by assessing the physical plausibility of the obtained results.

1-INTRODUCTION:

The CANDU 600 pressurized heavy water reactor (PHWR) is a large, horizontally oriented cylindrical tank, called the calandria, which contains the cool, low pressure heavy water moderator. This tank is penetrated by a number of horizontal tubes, called fuel channels which contain the natural Uranium fuel and the pressurized high temperature heavy water coolant. This coolant is pumped through the fuel channels, removing heat from the fuel, and then through a preheater type U-tube steam generator (PUTSG) where this heat is given up to produce steam which is fed to the turbine. The steam generators and coolant pumps are located at each end of the reactor so that flow is in one direction through one half of the fuel channels and in the opposite direction through the other half. A pressurizer maintains the coolant circuit pressure at a relatively high value. High circuit pressure permits high coolant temperatures which in turn permits the generation of steam at high enough pressure to achieve reasonable turbine cycle efficiencies. The CANDU reactor simplified flow diagram is shown in figure 1. The reactor consists of a cylindrical stainless steel calandria structure which contains the heavy water moderator, reactivity control mechanisms and 380 fuel channel assemblies. The fuel channel assemblies contain the fuel and heavy water coolant and pass through the calandria tubes. The gap between each fuel channel and calandria tube is gas-filled to provide thermal insulation. A cooling system is provided to dissipate heat transferred to the moderator from the high temperature fuel channels and heat generated in the heavy water itself through interaction with fission neutrons and gamma radiation. The reactor is fuelled with natural Uranium in the form of pellets, of Uranium dioxide (UO_2).

Approximately 29 pellets stacked end-to end, are sealed in zirconium alloy to form a fuel element. Thirty-seven of these elements are assembled into a fuel bundle (Figure 2). Each fuel channel accommodates twelve fuel bundles.

2-MATHEMATICAL MODELING:

2.1 Reactor Model:

The CANDU-PHWR reactor power was modeled using the point kinetics equations with six groups of delayed neutrons and reactivity feed back due to changes in fuel temperature and coolant temperature as follows:

$$\frac{dn}{dt}=(R-B_T)\frac{n}{L}+\sum_{i=1}^{6}\lambda_i C_i$$

$$\frac{dC_i}{dt}=\frac{B_i.n}{L}-\lambda_i C_i$$

$$R=R_c+R_{fb}$$

$$R_{fb}=\alpha_F.\Delta T_F+\alpha_c.\Delta T_c$$

Where:
n =Neutron density,

B_T =total delayed neutron fraction,

B_i =delayed neutron fraction for the ith group,

L =neutron generation time,

C_i, λ_i =precursor concentration and decay constant respectively,

R , R_{fb} , R_c = total, feedback and control rod reactivity respectively,

α_F, α_c = Doppler coefficients of reactivity and coolant temperature coefficient of reactivity respectively,

ΔT_F , ΔT_c = changes in fuel and coolant temperature respectively.

The reactor heat transfer model was based on lumped parameter technique, which assumes average system parameters over defined lumps. This approach gives moderate internal information of the system depending on the number of lumps used with reasonable accuracy at moderate computer time and cost. The fuel channel heat transfer model is simulated by an average channel composed of horizontal central fuel region surrounded by coolant. Two coolant lumps were used for the fuel lump to obtain good approximation of the heat transfer driving force between fuel and coolant which is taken

as the differences between the average fuel temperature and the average temperatures of each coolant lump respectively. The model equations are derived from the basic principles of thermohydroulics by applying the laws of conservation of the mass and energy on each model lump, and the equations of state for the heavy water coolant as follows:
For the fuel-lump:

$$M_F.C_{PF}\frac{d}{dt}(\overline{T}_F) = P_0.N - U_{FC}.A_{FC}.(\overline{T}_F - \overline{T}_C)$$

For coolant lumps:

$$M_{C1}.C_P.\frac{d}{dt}(T_{C1}) = \frac{1}{2}U_{FC}.A_{FC}(\overline{T}_F - \overline{T}_{C1}) + W_P.C_P(T_L - T_{C1})$$
$$M_{C2}.C_P.\frac{d}{dt}(T_{C2}) = \frac{1}{2}U_{FC}.A_{FC}(\overline{T}_F - \overline{T}_{C2}) + W_P.C_P(T_{C1} - T_{C2})$$

Where:
A_{FC} = Heat transfer area between fuel and coolant,

C_{PF} , C_P = specific heat of fuel and coolant respectively,

M_f =fuel mass,

M_{C1} , M_{C2} =coolant mass in the first and second coolant lumps respectively,

N =n/n$_0$ = normalized neutron density,

P_0 = reactor power at normal operation,

$\overline{T}_F$ = average fuel temperature,

T_L =temperature of coolant entering the fuel bundle in the first coolant lumps,

T_{C1} , T_{C2} = output coolant temperature of the first and second coolant lumps respectively,

U_{FC} = overall transfer coefficient between fuel and coolant,

W_P= primary heavy water coolant flow rate,

$$\overline{T}_{C1} = (T_L + T_{C1})/2$$

$$\overline{T}_{C2} = (T_{C1} + T_{C2})/2$$

$$\overline{T}_C = (\overline{T}_{C1} + \overline{T}_{C2})/2$$

For the application of the mathematical model, it is necessary to obtain the effective heat transfer coefficient between the fuel with an average temperature $\overline{T}_F$ and the heavy water coolant with an average temperature $\overline{T}_C$. this heat transfer coefficient can be derived by equating the heat generated in the fuel with the heat transferred by conduction through the cladding material and the heat transferred by convection through the coolant. It is assumed that no heat generated in the cladding and the resistance to heat at the fuel cladding interface is neglected.

3. CANDU-PHWR PROGRAM DESCRIPTION:

For the state variables chosen, CANDU-PHWR model equations are solved using standard numerical method called "Merson" for solving a set of coupled nonlinear differential equations.
The equations form is then given by:

$$\frac{d}{dt}(\overline{Y}) = \overline{F}$$

where:
$\overline{Y}$ = state variable vector,
$\overline{F}$ = Vector of non linear and forcing terms (including differential terms)

The elements of vector F will be listed in subroutine "DIFF", which is a part of the "Merson" program used. The Merson program concerned with the calculation of the transient behavior of the state variables using the forcing and steady values as an input. This program consists of three sections, the first is the main program that includes the forcing values, the basic model data, the geometric parameters and the steady state values as an input to the system. The second section is the subroutine "DIFF" which includes the state equations, the heat transfer correlations, algebraic variables and all physical properties of heavy water coolant as time dependant for the CANDU-PHWR model. The third section is the subroutine "Merson" which includes the algorithm for solving the system equations.

4.TRANSIENT RESPONSE OF THE CANDU-PHWR MODEL:

The transient responses of the CANDU-PHWR mathematical model for various step reactivity changes are shown in Figure 3.
The first effect of the reactivity decrease is shown as a rapid decrease in power generated due to the external reactivity step decrease. Decreasing the fuel temperature is followed by a decrease in the coolant temperatures. The rapid increase of the nuclear power generated following its initial rapid reduction is due to the negative feed back reactivity. This rapid increase is followed by relatively slower increase of power due to the decrease in the negative feed back reactivity till the power reaches a steady state value.
The effect of the reactivity increase is shown as a rapid increase in the nuclear power generated. Increasing the fuel temperature is followed by increasing coolant temperature

inside the core. The rapid decrease of the nuclear power following its initial increase is due to the negative feed back reactivity increase. This rapid decrease is followed by slower power increase (for small external reactivity increase) or rapid power increase (for large external reactivity increase).
Figure 4 illustrates the CANDU reactor transients for various step reduction in the primary coolant flow. The simulation of this perturbation is done through the change in the primary flow valve coefficient.
The decrease in the primary flow increases the fuel temperature. The rapid increase in the reactor power is followed by a rapid decrease due to the negative feed back effect till the power reaches a steady state value.

5. CONCLUSIONS:

Starting from the first principles and employing a nonlinear lumped parameter state variable formulation, a mathematical model for the CANDU-PHWR has been produced. This model has been coded in a general purpose language(FORTRAN) and solved using a standard numerical technique. The adequacy of the model was tested by assessing the physical plausibility of the obtained results.

The results showed that for both the external reactivity increase and primary flow decrease perturbations the CANDU-PHWR reactor is inherently stable due to its high negative temperature coefficient of reactivity.

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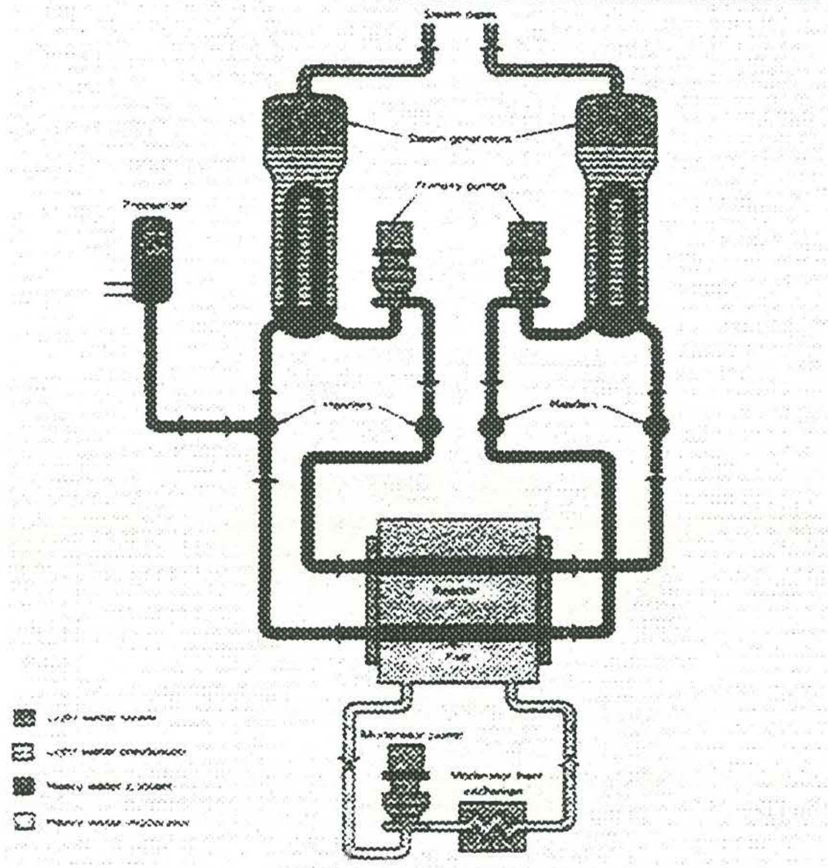
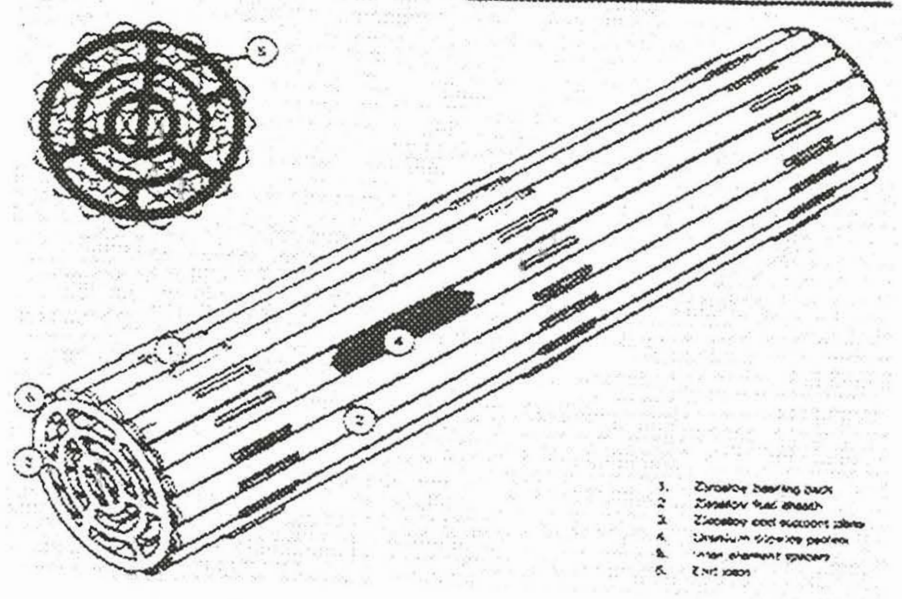
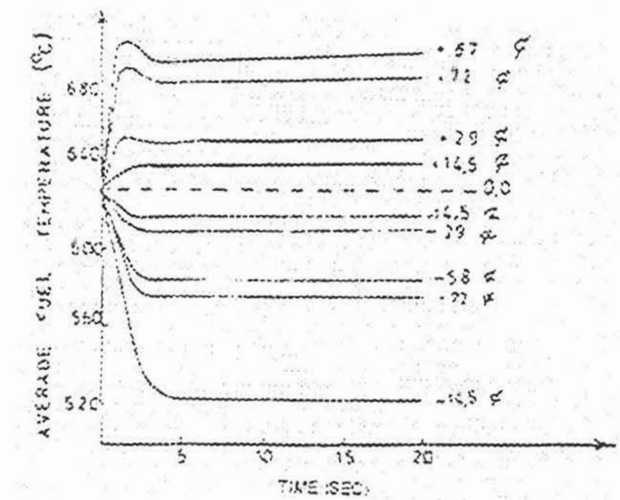
CANDU Reactor Simplified Flow Diagram , Figure 1



37 Element Fuel Bundle — Figure 2





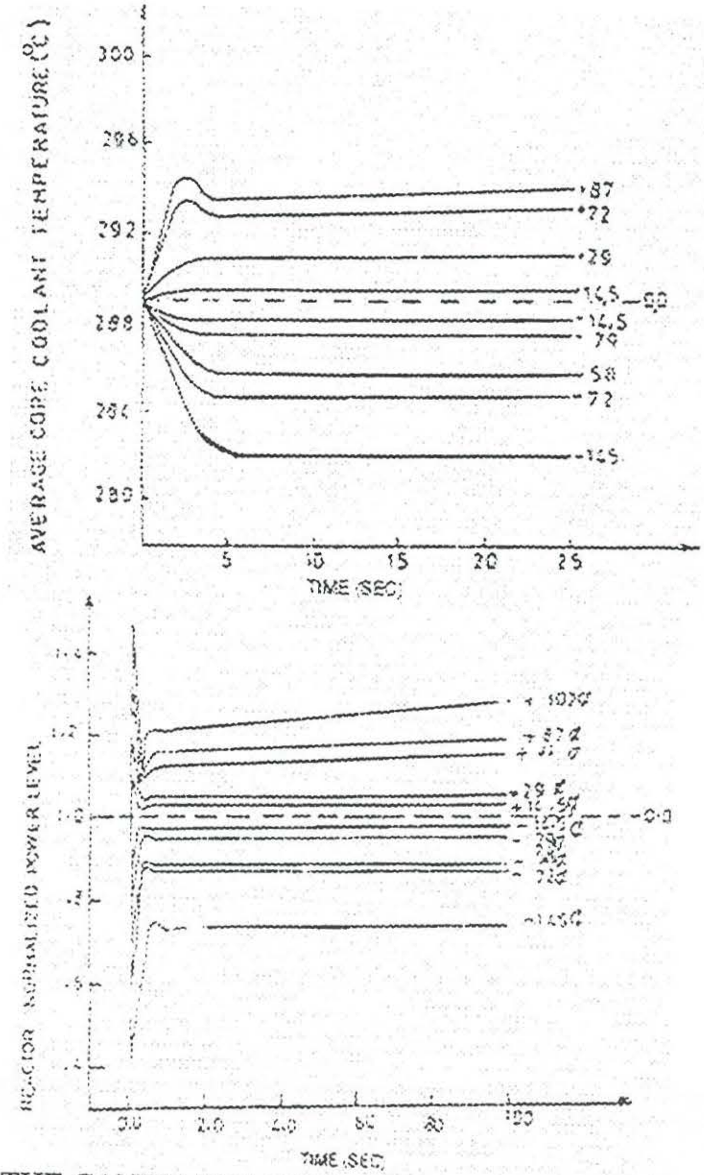


FIGURE (3) THE CANDU REACTOR TRANSIENT FOR VARIOUS STEP REACTIVITY CHANGES

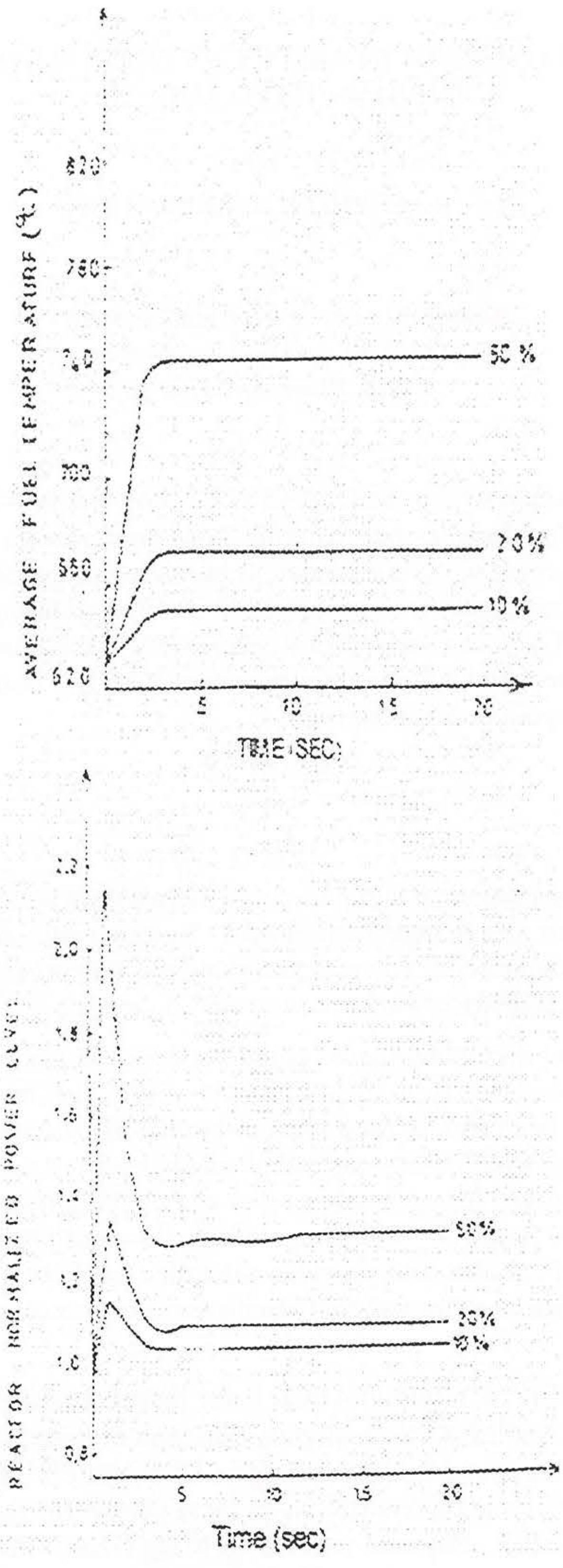


FIGURE 4 THE CANDU REACTOR TRANSIENT FOR VARIOUS STEP CHANGES IN THE COOLANT FLOW RATES