Chapter 1
INTRODUCTION

1.1. General

The Dalat Nuclear Research Reactor (DNRR) is a 500 kW pool-type reactor with natural convection cooling. Light water is used as both the coolant and the moderator. More detailed description of the DNRR is given in Appendix A. The main uses of the DNRR are: (i) irradiation of samples for radioactive isotope production and neutron activation analysis, (ii) nuclear and reactor physics research, and, (iii) education and training [1].

To date, several studies have been done in the areas of reactor physics and thermal-hydraulics, including theoretical research and experimental measurements, but most of these have looked at only steady-state conditions. Based on the results of these studies, some safety analyses have been performed to ensure that the reactor operation is safe under the design conditions.

Recently, the first attempts to study reactivity accidents and to simulate reactor operation, have raised requirements for solving transient problems. For the DNRR, point kinetics is sufficient to predict neutron power with time. Reactivity feedback must be considered during a transient. The most difficult reactivity feedback phenomena to model are the ones resulting from changes in the temperature of coolant and fuel, as these require solving time-dependent thermal-hydraulic equations for natural convection. Generally, mass, momentum and energy conservation equations together with state equations (i.e. fluid properties) have to be solved as a function of time to find such hydrodynamic parameters as flow, mass, enthalpy and pressure. The most important dependent parameter is mass flow as recognized by many authors. In the case of forced convection, mass flow is developed by circulation pumps and is determined by the pump capacity. In the case of natural convection under steady-state conditions, mass flow can be found by the balance of buoyancy and friction forces, and will therefore be a function of the given steady-state conditions.

For non-steady natural convection, mass flow takes a certain time to develop from the initial steady-state conditions to the new stabilized value state. In some simulation models, to avoid solving the complex equation of momentum, mass flow is given by a function of power with empirical time constant based on practical observations [2], which, of couse, will not necessary give the correct transient response for all transients.

In this thesis, we wish to establish a code that can solve the system of thermal-hydraulic equations numerically for the DNRR model. Coupled with the point kinetics, this code is expected to simulate normal operations of the DNRR. Normal operations should be understood as any reactor maneuvers under the following conditions [1]:

- Power is not more than 110% of full power,
- Period is not less than 30 sec., and
- No surface boiling is allowed.

It should be pointed out that the power and period limit settings are designated to prevent fuel surface boiling. Since no boiling is allowed in normal operations, only single phase subcooled flow is considered in the model. Theoretically, the code could also be applied to heat transfer with boiling, and having the coolant range from subcooled liquid, through the two-phase regime, and up to superheated steam, by choosing an appropriate set of correlations. In practice, we will not be able to validate the code under accident scenarios, so using empirical correlations would imply a significant level of uncertainties. Void formation will cause major distortions in the neutron flux distribution, with the result that our model would not be sufficiently realistic. Therefore, boiling is beyond the scope of this thesis.

The code is also intended to be used for thermal-hydraulic safety analysis of the DNRR. It is known that the surface temperature of the aluminium cladding of the fuel must not exceed a certain maximum value, particularly for the fuel staying in the core for the whole life of the reactor, i.e. tens of years. Aluminium cladding corrosion is influenced by various factors, such as coolant temperature, pH, chemical composition, velocity, etc. Each of these factors, either individually or in some combinations, induces or accelerates aluminium corrosion, resulting in the cladding weakening in the long term, and consequently, increasing the risk of fission product release. For design purposes, the maximum wall temperature of aluminium cladding is recommended to be below **105°C** [3]. That is, for long term operations, this maximum wall temperature value should not be exceeded. However, for short transients, the onset of nucleate boiling is accepted as the thermal-hydraulic limit.

As indicated by the title, the emphasis in this thesis is on the thermal-hydraulics of the reactor. We will assume that all neutronic parameters are known from previously published works, including experimental results. To simulate thermal-hydraulic parameters of the DNRR, we will first model the reactor components: the core, the chimney and the pool. Then, each component will be subdivided into an arbitrary number of regions. Conservation of mass, energy and momentum will be used to write the lumped parameter equations for the flow regions. Heat conduction equation will be written for the fuel and cladding regions. Equations of state and boundary conditions between fluid flow and heating surface will be added to make closure of the system of equations. Necessary assumptions and limitations will be included to simplify derivations of finite difference forms. By choosing an appropriate set of empirical correlations, computer programs will be written to solve the derived equations. Finally, the code will have to be validated by code-experiment comparison.

Summary:

The ***objectives*** of this thesis are *to simulate the thermal-hydraulic behaviour of the Dalat Nuclear Research Reactor under normal operations, including steady-state conditions and power transients.*

The ***scope*** of the research is:

1. *Neutronic parameters are determined experimentally and used as constants when applicable.*
2. *Fixed shape for the neutron flux and the heat distribution in the reactor core is assumed.*
3. *The onset of nucleate boiling is considered as the reactor safety limit.*

This work is expected to have the following ***benefits*** :

1. *the coupled point kinetics and thermal-hydraulics code can be used as a simulation code to investigate reactor operations .*
2. *the code can be used for safety analysis in order to ensure the safe and reliable operation of the reactor.*

1.2. Literature Survey

i) <u>NATCON</u> - (Source: MTR-PC 2.6, INVAP, Argentina, 1997).

The NATCON code is designated for <u>steady-state</u> heat transfer calculations for pool-type research reactors with natural convection cooling. One hot or average channel model is used. The coolant flow, W, is calculated based on the balance of buoyancy ($\Delta P_{buoyant}$) to friction pressure drops across the channel ($\Delta P_{friction}$).

$$\Delta P_{buoyant} = \Delta P_{friction}$$

$$\Delta P_{friction} = KW^2$$

$$\Delta P_{buoyant} = gH(\rho_o - \rho)$$

where ρ_0 is the density of pool water at ambient temperature, ρ is the average density of water in the hot column with height H. ρ is calculated at temperature $(T_1 + T_2)/2$, while

$$T_2 = T_1 + Q/c_pW$$

where T_1 and T_2 are the inlet and outlet coolant temperatures, Q is thermal power and c_p is the specific heat of water.

ii) <u>PARET</u> - (Source: NESC0555, NEA-Data Bank, France)

C.F. Obechain (1969) and W.L. Woodruff (1982) developed and improved the PARET code, a program for the analysis of reactor transients. The PARET code provides a coupled thermal, hydrodynamic and point kinetics capability with continuous reactivity feedback. It is designed to analyse reactivity transients in nuclear reactors with <u>forced convection.</u>

The hydrodynamics calculations are based on a modified Momentum Integral Model in which equations representing the conservation of mass, momentum, and energy are solved in each moderator channel at each time node:

Mass Conservation: $$\frac{\partial \rho}{\partial t} = -\frac{\partial W}{\partial z}$$

Momentum Conservation: $$\frac{\partial W}{\partial t} + \frac{\partial}{\partial z}\left(\frac{W^2}{\rho}\right) = -\frac{\partial P}{\partial z} - K|W|W - \rho g$$

Energy Conservation: $$\rho\frac{\partial H}{\partial t} + W\frac{\partial H}{\partial z} = q$$

where ρ is coolant density, W is coolant flow, P is pressure, H is enthalpy and q is the heat flux.

iii) <u>SLOWKIN</u> - (Source: Report IGE-219, Ecole Politechnique de Montreal, Canada, 1997)

D. Rozon and S. Kaveh (1997) developed the SLOWKIN, a simplified model for the simulation of transients in a SLOWPOKE-2 reactor. The SLOWKIN model uses coupled point kinetics and thermal-hydraulics to investigate transients in the SLOWPOKE-2 reactor, a pool-type reactor with natural circulation. A major simplification is introduced to <u>avoid solving the conservation of momentum</u>: coolant flow is calculated at steady-state conditions and changes with a time constant during transients. Thus,

$$W(p,t) = W_{ss}(p)[1 - \exp(-t/\tau)]$$

where $W_{ss}(p)$ is coolant flow calculated at steady-state conditions with power p, and τ is time constant determined experimentally.