CHAPTER II
LITERATURE SURVEY

2.1 Background

Nowadays, resources of fossil fuels are dwindling. The other energy sources are needed to substitute for fossil fuels as well as to provide for any additional demand. Nuclear energy has proved to be an economic source of power generation. This energy is a sort of between renewable and non-renewable resource. The important nuclei used as a source of fuel for nuclear reactors are the two isotopes of uranium, U-235 and U-238, and two isotopes of plutonium, Pu-239 and Pu-241. For these four fissile materials, only Uranium-235 is naturally occurring and the primary fuel of all commercial power producing nuclear reactors even if natural uranium contains only 0.7 percent of U-235 (Cameron, 1998). However, it is possible to produce this uranium in the breeder reactors, which are essentially renewable. In any case the supply of uranium is vast.

In addition to the main source of heat in the reactor, the moderator is the substantial element for reducing neutron velocity to increase fission probability. Water (H_2O), heavy water (D_2O) and graphite (C) all make good moderators for nuclear reactors. In order to maintain a thermodynamic cycle to produce work, the heat must be removed continuously while it is produced. A suitable coolant, such as light water, heavy water, helium and carbon dioxide, is thus required to flow over the fuel elements and to remove the heat. To achieve the required balance, maintaining and controlling the reactor at a steady load, movable neutron absorbing control rods are used within the reactor. The basic reactor components are shown in Figure 2.1 (Lister and MacNeil, 1994).

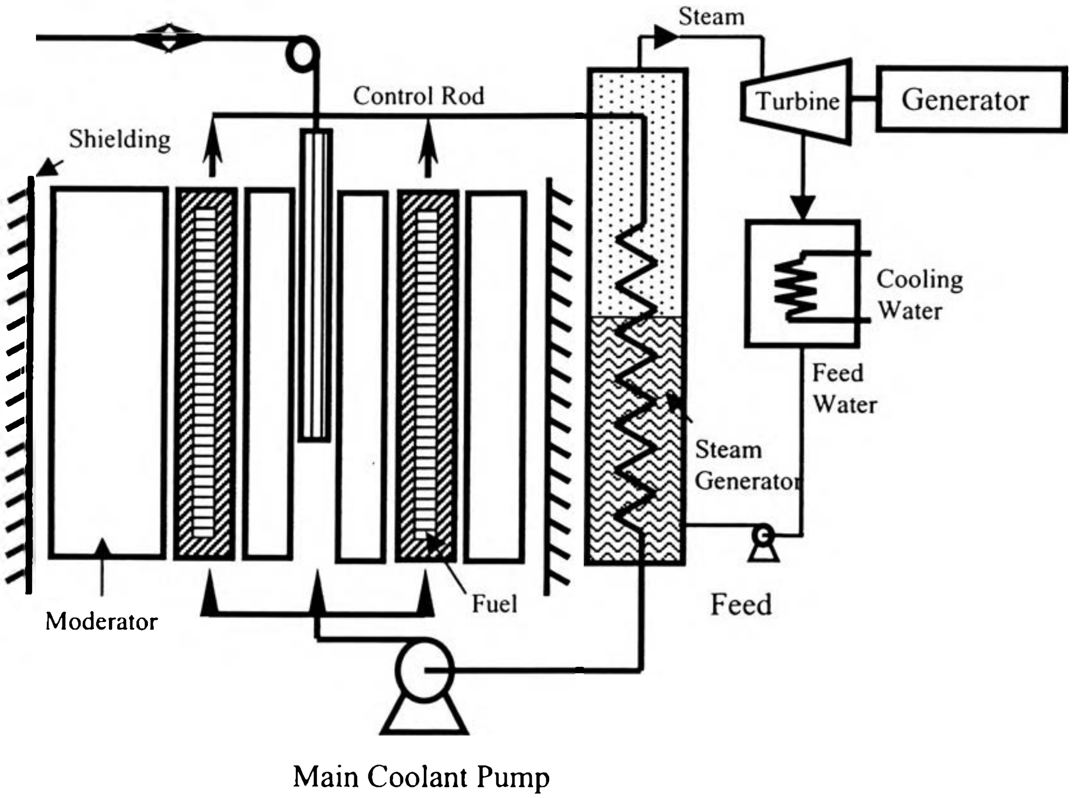

Figure 2.1 Basic reactor components

There are many types of nuclear reactors conveniently categorized by the moderator used. The following lists of abbreviations are representative of the different nuclear reactors.

AGR	Advanced Gas Cooled Reactor
BWR	Boiling (Light) Water Reactor
GCR	Gas Cooled Reactor
HWLWR	Heavy Water Light Water Reactor
LGR	Light Water Graphite Reactor
LMFBR	Liquid Metal Fast Breeders Reactor
PWR	Pressurized (Light) Water Reactor
PHWR	Pressurized Heavy Water Reactor

CANDU, Canada Deuterium Uranium, which is a commercial nuclear reactor of Canada, is one type of PHWR. CANDU reactors play an important role in the power generation for Canada. Canada has 22 nuclear generating stations, which supply approximately 20 percent of the nation electricity. CANDU reactors have also been supplied to other countries throughout the world. For CANDU structure, the reactor is enclosed in a reactor vessel called the calandria. The calandria is penetrated horizontally by a large number of horizontal channels as shown in Figure 1.2 and Figure 1.3. Each channel contains a pressure tube carrying twelve fuel bundles, each of which has 37 fuel pins arranged in a circular pattern. In each fuel pin, it contains the hard and insoluble uranium-oxide pellet, which is refined from natural uranium. This fuel can be shifted while the reactor operation by special refuelling machines, which can be attached to the ends of any pressure tube, are used to push new fuel elements in at one end of the pressure tube and remove spent fuel elements at the other end. Besides, high-pressure heavies water coolant that transports the heat to heat exchanger boilers where it generates stream to drive the turbines. Surrounding each pressure tube is a concentric calandria tubes. Outside the calandria tubes are the heavy water moderator that is separated from the pressure tube by the calandria tube, creating an annulus shape containing an insulating gas. A trickle flow of CO_2 occurs along the channel in the annulus between the two tubes. The purpose of the gas-filled annulus is to keep heat losses from the coolant to the moderator at acceptably low level during normal operation. The circuit of the moderator system is separated from the heat transport system. The calandria is located inside a concrete reactor vault that is filled with light water. This water provides shielding to protect the concrete walls from fast neutron irradiation and overheating, and also supports the calandria shell at a constant temperature. In the whole process of power generation, there are two circuits: the primary circuit within the nuclear reactor cycle and the secondary circuit steam cycle. Heat generated in the nuclear reactor is removed in order to used in the steam cycle. The first step in heat removal is conduction through the fuel to its surface and then transfer by convection to the coolant that circulates thoroughly the channels in the reactor. Certain channels around the periphery of the reactor do not produce as much heat as those in the center due to the lower neutron flux near the reactor boundary. Accordingly every fuel channel can produce heat at different rates. After the coolant leaves the fuel channels, heat exchanger between the two circuits transfers heat from the coolant to the stream cycle. The stream cycle drives the turbines, which generate power as shown in Figure 2.1 (Chaplin, 1999).

The reaction for generating energy is self-sustaining chain reaction, based on the fission in the fuel. In order to reduce the loss of neutrons by non-fission capture or leakage, the moderator, used heavy water, is the important component to slow the neutrons and thereby increase fission probability to produce the required thermal energy. Cameron (1998) showed that A good moderator has to have two basic properties; it must slow fast neutrons to decrease the uranium resonance captures compared to fission captures and it must have a low absorption cross-section to overcome neutron captures by itself. The main moderator system consists essentially of two main pumps, two heat exchangers, a head tank and the necessary instruments, valves and piping. The moderator system is operated at a low pressure with helium being used as a cover gas over the heavy water. The cover gas prevents accumulation of deuterium and oxygen gases, from radiolysis of the heavy water moderator, that could cause an explosion hazard, by catalytic recombining hydrogen and oxygen to form heavy water. Apart from the main circulation system, the moderator system is used to be (AECL, 1973):

2.1.1 Moderator Purification System

This system performs the following functions:

- Maintains the purity of the heavy water, thereby minimizing radiolysis, which may cause excessive production of deuterium in the cover gas.
- Minimizes corrosion of components and activation of the system by removing impurities present in the heavy water and bycontrolling the pH.
- Reduces the concentration of the soluble poisons, boron and gadolinium in response to reactivity demands.
- Removes the soluble poison, gadolinium, after initiation of the poison injection shutdown system.

2.1.2 Moderator Deuteration and De-Deuteration System

The system is used for upgrading ion exchange resins, from the purification system, used to remove corrosion products, ionic impurities and liquid poisons (boron and gadolinium) that is rejected into the moderator for reactivity control purposes.

2.1.3 Moderator Cover Gas System

The inert gas, helium, covers the free surface of the moderator heavy water, which recirculates to the reactor core, to maintain the concentration of deuterium gas (D_2), which results from the radiolysis of heavy water in the gas space, below 4 percent. This system also provides a vent to the shutdown rods and the poison injection shutdown system.

2.1.4 Moderator D_2O Collection System

The moderator D_2O collection system is designed to collect heavy water from various points in the moderator system in order to pump to either the moderator system or to the D_2O cleanup system.

2.1.5 D_2O Sampling System

The moderator D_2O sampling system is designed to provide samples from various systems, main moderator system, D_2O collection system, moderator purification system and D_2O cleanup system, for the purpose of performing the tests listed below:

- pHs test
- Conductivity test
- Test for chlorides
- Test for isotopic
- Test for tritium
- Gamma scan
- Test for fluorides
- Test for organic
- Test for boron concentration

- Test for gadolinium concentration

2.1.6 D_2O Supply System

The D_2O supply system receives D_2O from two sources:

- Fresh D_2O is received from tank trucks or drums
- Upgraded D_2O is received from the purification system

2.1.7 Liquid Poison System

Both boron and gadolinium are liquid poisons. Boron is injected into the moderator system to make the solutions of boron and heavy water, B_2O_3. they are used for long-term reactivity shim control. Gadolinium is used as the solutions of gadolinium and heavy water to compensate for the absence of xenon (Xe-135) during start-up of the reactor. Gadolinium is also injected to shut the reactor down via the liquid injection shutdown system.

The moderator is also required to act as a heat sink in certain accident situations, such as a loss of coolant accident coincident with a failure of cooling system.

Heavy water, which is used in the CANDU reactor, is naturally occurring compound in ordinary water and is formed from the combination of two deuterium atoms and one atom of oxygen. Heavy water has the high neutron moderating ratio, which is the nuclear property. The high moderating ratio is essential for achieving a high neutron economy. However, the heavy water separation is somewhat difficult due to the similar chemical and physical properties as those of light water, low quantity of heavy water in nature, and the necessity of large quantities of fluids in separation process. At the same time, this process requires a large amount of energy to maintain favorable bithermal conditions. The Girdler-Sulphide or GS process involving a dual temperature chemical exchange between hydrogen sulphide and water, which used carbon steel and the austenitic stainless steels, was one of the reliable and economical heavy water produced methods, which is employed in Canada. austenitic stainless steels (Twigg and Chang, 1983).

2.2 Development of the CANDU System

Abdelbaky *et al.* (1996) studied the simulation of photon-electron transport in CANDU reactor fuel channels using the Monte Carlo methods for calculating the energy deposition in the coolant. Two different codes, Electron Transport-based code (SANDYL, MCNP and the integrated tiger series (ITS) code system, includes the TIGER, CYLTRAN and ACCEPT) and Electron Gamma Shower (EGS4) code have been justified and compared by using the simplifying assumptions and simplified geometrical models. A simplified computational model based on decoupling photon-electron transport simulations is also introduced in order to speed up the computation without prohibitively expensive and long estimating time. Since, the coolant in a nuclear power reactor is subjected to a high radiation field comprised of fast neutron, protons and fast electrons. This irradiation results in chemical decomposition (radiolysis) of water-based coolants, leading to the creation of chemically reactive species, which can induce corrosion and cracking in the surrounding metallic surfaces. The damaging aspects of coolant radiolysis can be minimized by appropriate control of the coolant chemistry. The models, the EGS4 and SANDYL codes, using coupled photon-electron transport calculations give agreement with successfully validated results, although EGS4 codes are shown to be more sensitive than SANDYL to the value of the energy cut-off assigned to electron tracking. Besides, the model of decoupled photon-electron transport calculations shows the identical results. Thus, the computational efficiency of the decoupling model makes it suitable for using in the repetitive calculations needed to assess the extent of radiolysis occurring in fuel channels under many varying operating conditions.

In CANDU reactors, the heavy water moderator is physically separated from the coolant and fills the calandria. A large amount of heat is transferred to the moderator due to the heat loss from the fuel. The moderator heat load is considered varying about from 82 to 91 MW in the reactor (Thompson *et al.*, 1996). A moderator cooling system is established for the continuous removal of this heat load. The warm heavy water moderator is circulated through the moderator heat exchangers where its heat is transferred to the recirculating cooling water (RCW). Eventually this heat is lost to the environment through the raw service water (RSW). Usually, the moderator outlet temperature (from the calandria) varies from 60-65°C. Consequently, an amount of heat will be transferred to the moderator under normal operation, which is lost to the surroundings. To overcome this heat loss, Fath and Ahmed (1986) proposed a CANDU reactor moderator heat recovery (MHR) scheme, which is the circuit utilizing all of the moderator heat to the first stages of the plant feed water-heating system. The steam saved from the turbine extraction system was found to produce an additional electric power ranging from 5 to 11 MW. This additional power represents a 0.7-1.7% increase in the plant electric output power and a 0.2-0.7% increase in the plant thermal efficiency. The other advantage of this scheme, the feed water is clean reducing the problem of fouling or corrosion in the moderator heat exchanger. Moreover, the raw service water (RSW) heat exchangers used as a backup are relatively small, due to the high logarithmic mean temperature difference between the moderator and RSW.

2.3 Loss of Coolant Accidents (LOCA)

There are a large number of studies concerned with loss of coolant accidents (LOCA) in the CANDU power reactor. The cool moderator surrounding the calandria tubes provides a potential heat sink following LOCA should the emergency coolant injection (ECI) fail or be impaired. The fuel is prevented from melting in such an accident because of heat transfer to the moderator. Rogers (1984) demonstrated the failure of the moderator cooling system in a severe accident sequence. For a moderator cooling system failure, After the period, about 20 minutes, that moderator is provided an alternate heat sink, the calandria pressure increases until the rupture disks on the calandria relief ducts break, at about 140 kPa(g). Some moderator is lost through the ducts and begins to boil. Liquid moderator is rapidly expelled through the relief ducts by void formation in about 45 to 50 minutes. No gross fuel melting would occur even when fuel channels are uncovered. Fuel in the core debris in the bottom of the calandria would not begin to melt until about 135 minutes after the accident begins. Moreover, the calandria walls would not melt nor suffer significant damage from molten core debris provided that the shield-tank water cooling system remains operational. Core debris would be contained within the calandria and would begin to re-solidify in the period of 10 to 50 hours after the initiation of the accident. The calandria serves as in inherent core catcher in a CANDUreactor. The last result is that pressurization of containment during the accident sequence would not cause containment failure nor result in excessive releases of fission products outside containment. Thus, it is judged that the core debris will be contained in the calandria vessel, which acts as an effective core catcher. However rough estimates of the probability of this accident sequence indicated a value considerably less than 10^{-7} per reactor year.

For the analysis of postulated loss-of-accident (LOCA) scenarios and transient fault conditions in CANDU nuclear reactors, ATHENA, Algorithm for THErmalhydraulic Network Anlysis, the advanced thermalhydraulic computer code was presented. This code uses a six-equation (two-fuild) hydrodynamic model to describe two-phase flow. System models are necessary for the idealization of a CANDU reactor system under LOCA conditions. Heat transfer between the fluid and piping walls (or fuel) is modellied using applicable correlations for boiling, condensation and forced convective heat transfer. Radiation heat transfer between the individual fuel rods in a pin bundle is also modelled. ATHENA could be able to model both horizontal and vertical separated flows acculately (Richards *et al.*, 1985).

Sandersan *et al.* (1993) introduced one of the ways of enhancing the safety of a CANDU-PWR reactor by reducing the moderator subcooling requirements during a postulated LOCA. The increased moderator temperatures would elevate the heat transfer from the moderator, which acts as a heat sink, to the surrounding shield tank during an accident, with impaired emergency cooling. This reduction in subcooling requirements can be achieved by investigating experiments of the change in heat transfer characteristics between a pressure tube and a calandria tube, while incorporating a zirconium metal wire screen placed in the fuel-channel annulus, right next to the calandria tube. Since in some LOCA scenarios, the pressure tube overheats, looses its strength, and then deforms through the annulus gap into contact with its surrounding moderator-cooled calandria tube. Upon contact, heat stored in the pressure tube would be rapidly transferred across the interface to the calandria tube and then to the moderator. The sudden transfer of heat could exceed the critical heat flux and cause film boiling on the calandria-tube outside surface. The film boiling causes the severely reduced rate of heat removal from the fuel channel to the moderator. In this technique, the screen helps to reduce the PT/CT thermal conductance by slowing down the rapid transfer of heat upon ballooning to cantact and to reduce the probability of film boiling on the calandria tube without any important effect on the normal system operation of the fuel channel. The reduction was significant enough that the calandria tube was not forced into film boiling upon pressure-tube ballooning contact with 0°C subcooling (100°C water) outside the calandria tube. Hence, it may be possible to raise the moderator temperature, during normal reactor operation. Since, the requirement of the moderator subcooling was decreased. Moreover, the use of a wire would have a minimal economic impact on the neutrons produced.

The problem causes by impairment of both the heat transport system (HTS) and the emergency core cooling system (ECCS) can result in high fuel temperatures and low steam flow rates over extended periods of time. Under these conditions, high temperatures can increase the quantity of fission products released from the fuel, and lead to the production of large quantities of hydrogen gas. Eventually, they may cause failure of one or more pressure tubes. In order to study this severe accident conditions, the CHAN-IIA code has been designed to simulate the heat transfer behavior of the CANDU fuel channel and validated against the CS28-1 experiments, provided data on the temperatures, the rate of hydrogen production, and the status of the fuel channel. Besides, The CATHENA code is designed for the analysis of two-phase flow and heat transfer in piping networks and fuel channels. The CHAN-IIA code over-estimates the fuel temperature transient and hydrogen production. It is observed that the sub-channel model and simple radiation model in fuel rings would contribute to under-estimation of heat removal in fuel channel. However, the CATHENA code shows a reasonable agreement with the CS28-1 experiments data and can be applied to the fuel and fuel channel analysis for impaired cooling conditions such as loss of a coolant accident with emergency core cooling system unavailable (Lim *et al.*, 1995).

Choi *et al.* (1996) studied the theoretical method to predict the transient behavior of helium gas and heavy water in reactivity mechanism thimbles during an in-core break in the CANDU reactor. By the structure of the CANDU, this study utilized the pressure tube concept that consists of an array of pressure tubes, containing the reactor fuel, passing through a large cylindrical vessel (the calandria), which contains the heavy water moderator and reflector. Each pressure tube is isolated from the heavy water moderator by a calandria tube. Moreover, the calandria is designed to withstand an in-core break such as pressure tube and calandria tube rapture. There are two types of in-core break: one is the spontaneous pressure tube rapture and the other is the pressure tube rapture owing to channel flow blockage. The designed reactivity thimbles are installed vertically in the calandria while applying and comparing two approaches, static and dynamic model. The application of the static model at the peak moderator pressure is simple. However, this can not follow the real phenomena and gives unrealistically high helium gas pressure. The dynamic model can simulate the actual phenomena and provides reasonable values for maximum helium gas pressure.

The Canadian Algorithm for Thermal hydraulic Network Analysis (CATHENA) thermalhydraulic code was developed by Hanna (1998) primarily for the analysis of postulated accident conditions in CANDU reactors. As a result of the unique design of the CANDU reactor, the CATHENA thermalhydraulic code has been developed with a number of unique modelling capabilities. The development of CATHENA began by following from earlier thermalhydraulic codes, which used equilibrium thermalhydraulic models. Currently the code switched to a one-dimensional, nonequilibrium two-fluid model to more accurately represent the horizontal CANDU fuel channels for two-phase stratified flow conditions predicted during some postulated loss-of-coolant accidents (LOCA). Some of the unique features of the CATHENA code are the one-step semi-implicit numerical method used and the solid heat transfer modelling capability that allows horizontal fuel bundles to be represented in detail. The highlight characteristic of this code is its flexible heat transfer package (GENHTP). GENHTP represented the wall heat transfer model, which consists of three major modelling components: wall-to-fluid heat transfer, wall-to-wall heat transfer and conduction within solid models. It is a flexible with respect to model geometry and the connection of all heat transfer surfaces to the thermalhydraulic model. Since this code is a versatile tool, it can be been utilised for designing and analysing the multiple applied lattice experimental (MAPLE) class of research reactors, thermalhydraulic test facilities to extendedly the integrated analyses of the CANDU reactor.

2.4 Moderator Temperature

Generally, the capability of the moderator to act as a heat sink in certain accident scenarios is a function of the amount of available sub-cooling. The available sub-cooling is established primarily from a prediction of the moderator temperature.

Sion (1983) studied in-core moderator temperatures within CANDU reactors. The program required the measurement of the temperature profile of the moderator water (D_2O) inside the calandria vessel, by means of a specially instrumented probe introduced within the core. He verified a theoretical computer program code-MODCIR (a Company generated computer program for MODerator CIRculation) with the practical experimental data of the temperature profile. Measurements were made under steady and transient reactor conditions. Due to the intense radiation environment within the reactor, two different sensors, viz. Resistance temperature detectors (RTD), which is made from nickel to neglect the transmutation problem for the duration of the tests and type

K chromel-alumel thermocouples are used. Criteria of sensor selection, type and design, took into consideration the possible means of minimizing source of errors, which are caused by gamma ray heating, transmutation effects caused by neutron capture, radiation-induced currents and alloy inhomogeneity, etc. The results established the feasibility of in-core moderator temperature measurements (within the range of 30-80°C) and are close to the data predicted by a theoretical computer code-MODCIR. It is indicated that the thermocouples used are not significantly affected by the intense radiation fields, thus producing more accurate data.

The CANDU reactor vessel, it is filled with a matrix consisting of a large number of tubes (the calandria tube) that contain the fuel pressure tube. The heat generated within the moderator is assumed to drive natural convection currents that are upward in the reactor core (calandria centre) and downward in the reflector zone. On the other hand, the moderator inlet jet at the reflector zone will generate a momentum force that opposes the natural convection circulation. Then, the transient two-dimensional computer code, which is capable of predicting the moderator flow and temperature distribution inside CANDU calandria was developed. This code uses a new approach in simulating the calandria tube matrix by blocking the cells containing the tubes in the finite difference mesh. The developed code can also be used to study the effect of different parameters on the moderator circulation. The transient and steady state, isothermal and non-isothermal results are predicted for a CANDU reactor. The steady-state isothermal flow pattern predictions are in reasonable agreement with measurements made in a full-scale mock-up. A jet momentum-dominant flow pattern is predicted in the non-isothermal case, and the effect of the buoyancy force, resulting from nuclear heating, is found to enhance the speed of circulation. Hot spots are located in low-velocity areas at the top of the calandria and below the inlet jet level between the fuel channels. A parametric study was carried out to investigate the effect of moderator inlet velocity, moderator inlet nozzle location, and geometric scaling. The results indicate that decreasing the moderator inlet velocity has no significant influence on the general features of the flow pattern; i.e., the flow is still momentum dominant. The temperature field, however, has certain high-temperature hot spots within the fuel channels as a result of reducing the moderator flow rate. A moderator inlet nozzle located 174 cm. below the horizontal mid-plane shows no significant changes in the main features of the circulation. However the new location is found to give a more uniform temperature distribution with fewer and lower temperature hot spots than the present design arrangement. For the last parameter, the flow pattern, and temperature distribution are conserved under geometric scaling (Fath and Hussein, 1989).

Thompson *et al.* (1996) proposed work on three fronts to study the local moderator temperature derived by analytical modelling based on thermal-hydraulic analysis. Firstly, efforts have been made to resolve the uncertainties, associated with moderator temperature predictions, moderator heat load, and D_2O flow as input parameters, and have a significant bearing on the output of, the moderator temperature prediction codes. Secondly, the diversification of modelling tools is examined to provide a range of independent analysis results against measured data. 2DMOTH code is custom-written for analysing the flow and temperature distribution in a CANDU reactor. PHOENICS code is the other general-purpose model of one-, two- or three-dimensional fluid flow/heat transfer calculation, which can handle steady state or transient simulations. MOD_TURC_CLAS code has been validated against the limited number of single-point in-reactor measurements. Thirdly, the feasibility of developing a methodology to permit the three-dimensional moderator temperature distribution measurements is being assessed. The primary difficulty associated with the measurement of the moderator temperature distribution is that direct measurement is not possible, due to access limitations. Vertical Flux Detector (VFC) assemblies are used for measurement purposes. The gamma and neutron heating is determined by the sequence of codes: ORIGEN-S, the Integrated Tiger Series, and MORSE-SGC. This can provide direct indication of the moderator temperature distribution, and also provide data for validation of the analysis codes.

In order to study the heat transfer characteristics of VFD, a test cell was fabricated from a small section of VFD. This test cell was inserted in the Gamma cell, which generated heating by using cobalt 60 (C-60) as a fuel simulating the irradiative heating from the fission reaction of natural uranium in the CANDU nuclear reactor. This test cell provided the temperature data that were recorded by type K chromel-alumel thermocouples operated in two modes: unsteady state experiments on the test cell outside the gamma cell and steady state experiments on the test cell outside and inside the gamma cell. All experiments in the gamma cell were performed with helium flowing through the cavity containing the detector well at 2 ml/min. The water temperature and flow rate remained unchanged throughout each experiment. As long as the ultimate utilization of a device is to measure the temperature distribution within the reactor moderator, the steady state experiments are the more appropriate to imitate the manner in which it would be operated in the plant. The test cell successfully represents the flux tube assembly for the purpose of assessing irradiative heating. Nevertheless, there is a problem in all cases of the experiments since the irradiation field of the gamma cell is not sufficiently strong to give adequate temperature rise for the determination of the temperature difference under simulated reactor moderator conditions. This does not allow the accurate determination of the heat transfer characteristics of the vertical flux tube assembly. Thus, the test cell must be exposed to an irradiation field at least an order of magnitude greater than the gamma cell, which is used in these experiments to obtain a precise determination of the heat transfer characteristics of the vertical flux assembly (Steward, 2000).