Thermalhydraulic Design Methods and Computer Codes

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Abstract
Methods used for thermal and hydraulic design are typical of those available in the world, enhanced by major experimental programs. These programs are used to develop and confirm mathematical modeling. In particular, experiments are concentrated on modeling of CANDU-specific aspects of:

1. Thermalhydraulics, such as channel flows and pressure drop, critical heat flux, blowdown and refill behaviour in the heat transport system;
2. Flow and temperature distribution in the moderator system; and
3. Thermalhydraulics in the containment safety system.

Both steady-state and transient conditions are included.

This paper describes computer-modeling approaches used for design along with the experimental and theoretical verification programs which confirm the results.

1. HEAT TRANSPORT SYSTEM

1.1 Heat Transport System Description

Figure 1 shows a simplified schematic of the CANDU heat transport system (HTS). The HTS circulates pressurized heavy water coolant through the fuel channels to remove the heat produced by fission in the nuclear fuel. The coolant transports the heat to steam generators, where it is transferred to light water to produce steam to drive the turbine. Two parallel HTS coolant loops are provided in CANDU 6. The heat from half of the several hundred fuel channels in the reactor core (380 in CANDU 6) is removed by each loop. Each loop has one inlet and one outlet header at each end of the reactor core. Heavy water is fed to each of the fuel channels through individual feeder pipes from the inlet headers and is returned from each channel through individual feeder pipes to the outlet headers. Each HTS loop is arranged in a 'Figure of 8', with the coolant making two passes, in opposite directions, through the core during each complete circuit, and the pumps in each loop operating in series. The coolant flow in adjacent channels is in opposite directions. The pressure in the HTS is controlled by a pressurizer connected to the outlet headers at one end of the reactor.

1.2 Heat Transport System Modeling

CATHENA, developed by AECL, has evolved with the objective of providing a high degree of flexibility in modeling thermalhydraulic systems. Although developed primarily for the analysis of CANDU nuclear reactors, the code has been successfully applied in the analysis and design of experimental test programs. CATHENA is also being used in support of the design, safety and licensing of research reactors developed by. A comprehensive description of the code can be found in [1].

1.2.1   Thermalhydraulic Model

The CATHENA code uses a non-equilibrium, two-fluid thermalhydraulic model to describe fluid flow. Conservation equations for mass, momentum and energy are solved for each phase (liquid and vapour), resulting in a 6-equation model. Also, up to four noncondensible gases may be represented as part of the vapour phase, yielding a 7- to 10-equation model. Interphase mass, momentum and energy transfer are flow-regime-dependent, and are calculated using constitutive relationships obtained from the literature or are derived from single-effects experiments.

Noncondensable gas properties are available in CATHENA for H_2, He, N_2, Ar, CO_2 and air. The thermodynamic properties for the noncondensable gases are assumed to follow the ideal gas law.

The numerical solution technique used to solve the conservation equations is a one-step, staggered-mesh, semi-implicit, finite-difference method. The dependent variables defining the state of a node or cell are pressure, void fraction, and phase enthalpies. If noncondensable gas(es) are present, the noncondensable fractions are also dependent variables. For connections between nodes (called links), the dependent variables are the velocities of the gas and liquid phases. Conservation of mass is achieved using a truncation error correction technique.

A one-step finite-difference numerical solution scheme has been adopted that is not transit-time-limited. The resulting set of equations is not reduced to a pressure- or flow-field approach. A time-step controller implemented in CATHENA automatically selects the next time step at each finite-difference time step. This is accomplished by monitoring changes in the dependent variables, the selected derived variables, and the truncation error. If the maximum change is below a prescribed value, the time step is increased; if the change is above a maximum prescribed value, it is decreased. The user may alter the default selection criteria through input data and thus check the temporal convergence of a given simulation.

1.2.2   Heat Transfer

Heat transfer from metal surfaces is handled by an extensive wall-heat-transfer package, GENHTP (GENeralized Heat Transfer Package). A set of flow-regime-dependent constitutive relations specifies the energy transfer between the fluid and the pipe wall and/or the fuel element

surfaces. A variational finite-element method is used to model the heat transfer by conduction within the piping and fuel in the radial direction, and the heat transfer can also be modeled in the circumferential direction. The radiative heat transfer and the zirconium/steam reaction rates can also be calculated. The ability to calculate the heat transfer from individual groups of pins in a fuel bundle subjected to stratified flow is built into this package. Under these conditions, the top pins in a bundle are exposed to vapour, while the bottom pins are exposed to liquid.

The GENHTP model also allows for the calculation of pressure tube (and calandria tube) strain, caused by pressure tube heat-up in a pressurized channel. Under certain postulated conditions, the pressure tube can "balloon" into contact with the calandria tube. When this occurs, heat will be transferred from the channel, through the pressure tube and calandria tube to the D_2O moderator. CATHENA is able to model all these processes, including the contact conductance between the pressure tube and calandria tube.

CANDU fuel is usually modeled by a user-specified UO_2 region, gap, and fuel sheath region. Temperature-dependent properties, available within GENHTP, are generally used for the UO_2 and sheath. Although a constant gap conductance is usually used for LOCA analysis, a user-specified time dependent value may be applied.

1.2.3   Component Models

Component models that describe the behavior of pumps, valves, pressurizers, steam separators, and discharge through breaks are available to complete the idealizations of the reactor systems.

The Generalized Tank Model (GTM), a two-region, two-fluid, non-equilibrium thermalhydraulic model is provided to model tanks, allowing a variable cross-sectional area. The upper and lower tank regions are modeled as independent volumes which are allowed to exchange heat and mass through mechanisms such as condensate fall, bubble rise or inter-region condensation.

1.2.4   Modeling Control Systems

Control systems may be modeled using a simulation-like language within CATHENA. Included within this system is the ability to use FORTRAN-like statements. The majority of variables calculated by CATHENA can be accessed, and used in these calculations. The results of these calculations can then be used to "control" models such as valve (opening fraction), reactor kinetics (reactivity insertion), etc.

In some cases, complex reactor control systems are already available in FORTRAN code. For these cases it is more efficient to use the existing control program, and "couple" it with CATHENA. CATHENA provides the required information to the Controller Program, and the Controller Program provides the required valve opening fractions, etc., to CATHENA.

1.3 Experimental Verification

The validation of CATHENA is supported by experimental data from a variety of CANDU-typical experimental facilities. The experiments cover a wide range of operating and postulated accident conditions. The major experimental programs are described below.

1.3.1 RD-14M

Figure 2 shows a simplified schematic of RD-14M, a multiple-heated channel, full- elevation, scaled, integral test facility, possessing most of the key components of a CANDU Primary Heat Transport System (PHTS). The facility is arranged in the standard CANDU two-pass figure-of-eight configuration. The facility is designed to produce similar fluid mass flux, transit time and pressure and enthalpy distributions as those typical of CANDU reactors under both forced and natural circulation conditions [2].

The reactor core is simulated by ten, 6 m-long horizontal test sections. Each test section has simulated end-fittings and seven electrical heaters, or fuel element simulators (FES), designed to have many of the characteristics of a CANDU fuel bundle. Test sections are connected to headers via full-length insulated feeders. Feeders are equipped with trace heating tapes to minimise heat losses under natural circulation conditions. Pipework connecting outlet headers can also be valved in to study the effect of outlet header interconnect geometry on mitigating oscillatory behaviour at full and low power conditions.

Above header piping is also CANDU-typical including two full-height, U-tube steam generators or boilers (BO1 and B02) and two bottom-suction centrifugal pumps (P1 and P2). Steam generated in the secondary, or shell, side of the steam generators is condensed in a jet condenser (CD1) and returned as feedwater to the boilers. For natural circulation experiments conducted post 1990 a customised secondary system, designed to operate at reduced power levels typically encountered under natural circulation conditions, was utilised.

The primary-side pressure is controlled by a pressurizer/surge tank (TK1) using a 100-kW electric heater (HR1). The facility operates at typical CANDU primary system pressures and temperatures (typically 10 MPa(g) and 310°C at the outlet header).

For the natural circulation experiments, fluid removed from the primary circuit at header 7 (HDR7) is cooled and stored in an inventory tank (TK3). Level monitoring of the inventory tank provides a record of the quantity of primary fluid removed.

The RD-14M facility is extensively instrumented. FES sheath and centreline temperatures up to 1000 C can be measured axially in five of the seven heaters in each simulated fuel channel to provide a comprehensive picture of the FES temperature distribution. In addition flow, temperature, pressure and the void fraction of the fluid entering and leaving each test section is

measured. Gamma densitometers are used to measure the void fraction of fluid at the entrance and exit to both steam generators and at the discharge of both pumps. Fluid temperature, pressure and flow rates are measured at regular intervals throughout the facility. In addition, over 50 differential pressure measurements provide an accurate picture of the pressure distribution throughout the facility. Key secondary-side measurements such as pressure, steam flow rate and temperature, feed water temperature and flow rate and internal shell-side recirculation rate are also recorded. Overall, approximately 600 instruments are scanned and recorded using a dedicated data acquisition system.

1.3.2 Header facilities

The core of a CANDU reactor consists of many fuel channels connected by feeder pipes to common inlet and outlet headers. Two-phase flow may occur in the headers during various postulated loss-of-coolant accidents (LOCA). Analysis of these postulated reactor accidents requires an understanding of the flow behaviour of the various reactor components, such as the headers. CANDU Owners Group (COG) has two facilities to study the two-phase flow behaviour in CANDU-like headers:

LArge-Scale Header (LASH) facility:

An overview of the LASH facility set-up is shown in Figure 3. The facility consists of two horizontal headers (inlet and outlet) connected by six banks of five "U" shaped feeders, for a total of thirty feeders [3]. Two basic types of tests can be performed in the LASH facility: two-phase steady-state injection tests and blowdown/refills tests.

The test loop consists of a steam separator, a condenser, a fossil-fuel boiler, a preheater, an inventory tank, various boiler and liquid-water circulating pumps, a mixer, interconnecting piping and valves, and a blowdown line. The test loop is designed to operate at a maximum pressure of 6 MPa. For two-phase injection tests, boiler steam and liquid water are fed into the two-phase mixer, and the resulting two-phase mixture flows into the inlet header. The boiler is capable of supplying a superheated-steam flow rate of approximately 4 kg/s at 7 MPa and 350∞C to the mixer. Liquid water flow rates up to 80 kg/s can be supplied to the mixer. Discharge from the output header is returned to the separator tank, where the liquid is recirculated through the liquid pump(s). Steam from the separator tank is returned to the condenser. The steam and liquid flows supplied to the mixer are varied to provide a range of qualities and flow rates to the inlet header.

The inlet and outlet headers are CANDU typical in diameter, but are half-length intended to be representative of a half-length Pickering header. Each header is 4.1 m in length and 0.325 m in diameter (internal). Two vertical turrets, 0.308-m internal diameter, are attached to the top of each header. The turret locations are biased towards one end of the header. Each turret contains vertical and horizontal nozzle connections.

Thirty feeder lines, 49.2-mm inside diameter, extend vertically downwards from the inlet and outlet headers. The maximum vertical drop to the horizontal-feeder section is 10 m below the header centreline, and the minimum vertical drop is 8.6 m below the header centreline. Each feeder consists of a vertical inlet section and a vertical outlet section connected by a horizontal section.

The set-up for blowdown/refill experiments is similar to that for two-phase injection experiments, however, a blowdown line and a cold-water-injection line connects to both headers. The set-up shown in Figure 2 is for inlet and/or outlet header breaks. Other configurations of the blowdown piping were used (for example, tests with feeder breaks), but are not shown here. For blowdown/refill tests, the headers and feeders are preheated by injecting superheated steam into the two-phase mixer. After preheating, the headers are pressurised with superheated steam, and a blowdown is initiated by opening one or more of the blowdown line quick-acting valves. Cold water is then injected into the inlet header, or both headers, to simulate the injection of emergency coolant liquid.

The LASH facility is extensively instrumented with over 500 channels. The phase distributions in the headers are inferred using differential pressure and conductivity probes located axially along the headers. Various fluid and surface-mount thermocouples are installed in the headers and feeders. Pressure cells are used to measure the header pressures, the header-to-header differential pressure, and various differential and absolute pressures in the feeders. All horizontal feeder sections contain turbine flowmeters, and selected feeders are instrumented with gamma densitometers.

Transparent Header/Feeder facility: This facility has been running at AECL Whiteshell Laboratories since 1989. It consists of inlet and outlet headers, and thirty connecting feeders, with nearly identical internal dimensions to the LASH headers and feeders. However, the facility is constructed of clear acrylic, which allows for the visual observation of flow. Two-phase injection experiments are performed by injecting an air-water mixture into the inlet header. Experiments are performed at near atmospheric pressure, liquid injection flow rates between 6 and 60 kg/s, and air injection flow rates between 0 and approximately 24 litre/sec (standard conditions). This facility is mainly used to visual confirm behaviour inferred from experiments performed in the LASH facility. This facility is also used as a test bed for the development and testing of new instrumentation.

1.3.3 Cold-Water Injection Test Facility

Figure 4 shows a simplified schematic of the Cold-Water Injection Test (CWIT) facility. This facility is designed to study the thermalhydraulic behaviour of a CANDU fuel channel under various postulated accident conditions, such a emergency-coolant injection following a LOCA [4] . The facility contains all of the key components of a CANDU fuel-channel assembly, as discussed below.

The CWIT facility consists of a channel assembly, two headers, inlet and outlet feeders, two break-simulation devices, a blowdown tank, and a cold-water injection system. The full-scale channel assembly consists of a 6-m long, electrically heated 37-element fuel bundle (FES). The fuel bundle has an axially-cosine power distribution, and by applying different voltages to the individual FES rings, a radial power depression ratio, representative of that in a CANDU reactor is maintained. The FES is housed in an actual CANDU pressure tube.

The pressure tube attaches to full-scale CANDU 6 end-fittings. Sections of vertical and horizontal feeder pipe connect the end-fittings to the headers, and is representative of the arrangement in a CANDU reactor. Breaklines are attached to each header to facilitate the simulation of a LOCA. Blowdowns are initiated by opening pneumatic quick-acting values installed on both blowdown lines. A cold-water injection system simulates the injection of emergency coolant into the headers.

The test facility is extensively instrumented with nearly 250 data channels recorded. Thermocouples are used to measure fluid, pipe-wall, and FES heater surface temperatures. Differential and gauge pressure cells measure local pressure at various locations in the loop. Gamma densitometers are used to measure feeder void fractions.

1.3.4   End-Fitting Characterization Facility

The End-Fitting Characterization (EFC) facility was designed to investigate the thermalhydraulic and heat transfer behaviour of CANDU end-fittings when subjected to a wide range of two-phase flow conditions. Tests were performed to study the flow resistance, the depressurization and the heat transfer behaviour of the end-fitting under postulated accident conditions.

The facility consisted of a CANDU 6 end-fitting, as shown in Figure 5 for heat-transfer tests. The test loop consisted of a steam generator, a condenser, a fossil fuel boiler, a preheater, a storage tank, various liquid-water pumps, a steam Venturi, interconnecting piping, and pressure and temperature control systems. The test loop was designed to operate at a maximum pressure of 5.0 MPa. The fossil boiler was able to supply superheated steam at 7 MPa and 450°C and saturated water to the test assembly via the steam/water mixer.

The end-fitting consisted of the end-fitting body, a liner tube, and a shield plug. The test assembly consisted of a feeder pipe, a simulated fuel channel, a water line, a steam/water mixer, return line to the condenser, an end-fitting bypass line, interconnecting pipes, and various flanges that were used to direct flow to various components of the end-fitting.

The simulated fuel channel consisted of a 2.0-m long pressure tube containing four copper pelleted 37-Element bundles installed to simulate the entrance and exit effects of reactor fuel bundles. The

Zircaloy-4 pressure tube was rolled into the end-fitting and extended inside the end-fitting by 112.6 mm.

The instrumentations in the test assembly consists of single-beam gamma densitometers, water and steam flow orifices, various thermocouples for temperature measurements, and capacitance type transimitters for absolute pressure measurements. Inlet, outlet and return flows to the end-fitting and flow from the dead space are measured with turbine flowmeters. The instrumentations in the test assembly varied from test to test.

The instrumentations in the end-fitting are clustered around six major axial planes. A total of 79 thermocouples are installed along 5 radial plains at 6 axial locations. Multiple thermocouples at each axial plane and around the outside of the end-fitting components ensure that the temperature distribution that may arise due to flow stratification can be captured. Absolute and differential pressure measurements are made using pressure cells. A 35-mm flowmeter is installed in each of the four holes in the shield plug. Void-fraction measurements are made within the end-fitting assembly with 18 single-beam gamma densitometers. Pitot tubes are used to measure the single-phase velocities in the annulus and in the channel. Collapsed fluid level in the annulus, dead space, and in the channel are inferred from differential pressure measurements.

2.   MODERATOR SYSTEM

2.1   Moderator System Description

The prime operating purpose of the heavy water moderator in the calandria is to provide a medium to slow down high energy fission neutrons in the reactor to the appropriate thermal energy level to promote further nuclear fission. In addition the moderator removes the heat that is continuously generated in the moderator as a result of heat production associated with neutron moderation and gamma ray absorption processes, heat transfer from the pressure tubes across the annular gap to the calandria tubes, and heat transfer from reactor structures due to nuclear heat generation and temperature differences.

The heat is removed by circulating the moderator heavy water through an external circuit which is fully independent of the heat transport system. The moderator heavy water is extracted from the calandria vessel through two outlet ports located at the bottom of the vessel. After discharging through the outlet ports , the fluid mixes in a header and passes through one of two 100 percent pumps to be cooled via two parallel 100 percent tube and shell heat exchangers (where the heat is rejected to the Recirculated Cooling Water) and is returned to the moderator through inlet nozzles located at the side of the vessel. The MTC (Moderator Temperature Controller) maintains the outlet temperature at a prescribed setpoint. The location of the inlet and outlet nozzles on the sides of the calandria ensures a uniform moderator temperature distribution inside the calandria. The operating pressure at the moderator free surface, which is maintained within a specified range

above the top row of channels, is slightly above atmospheric. The moderator head tank maintains the moderator level in the calandria within the required range by accommodating moderator swell and shrink resulting from temperature fluctuations. Figure 6 shows a schematic of the Main Moderator System

2.2 Moderator function following a loss-of-coolant accident

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of coolant accident with failure of emergency core cooling. The capability of this heat sink is assured by controlling the heavy water temperature in the calandria within specified limits.

Following a loss-of-coolant accident with loss of emergency core cooling there may be early PT/CT contacts due to high pressure in the PHTS and further PT/CT contacts due to pressure tube sag. The primary purpose of the moderator during a loss-of-coolant accident coincident with loss of emergency cooling with or without class IV power is to act as a heat sink following PT/CT contact. In particular one of the factors that determines the nature of heat transfer at the surface of the calandria tube in the event of PT/CT contacts is the degree of subcooling available (margin to boiling). The degree of local subcooling is the difference between the local saturation temperature and the local moderator temperature. The degree of subcooling is an important factor in determining whether a critical heat flux "condition" exists at the surface of the calandria tube. It is also a major factor in determining post contact behavior, the latter being a function of the circulation pattern within the moderator and hence the temperature distribution in the moderator.

Thus it is important to be able to predict local moderator temperatures, as the temperature of the moderator fluid surrounding a calandria tube has an important effect on the type of boiling which occurs following pressure tube/calandria tube contact. If film boiling takes place, the local moderator temperature is also an important factor in determining the time required to rewet the calandria tube and the amount of any possible post-contact deformation. The circulation pattern and temperature distribution existing at the time of PT/CT contact will to a large degree depend on that existing during steady state operation.

The circulation pattern and temperature distribution within the calandria vessel during full power operating conditions is dictated by the interaction between momentum and buoyancy forces. Momentum forces are generated by the incoming jets and buoyancy forces are generated from heat addition within the core and reflector regions (causing density differences). Depending on the relative magnitude of these forces, different flow patterns can exist within the moderator varying from momentum-dominated to buoyancy dominated. A momentum dominated flow can be visualized as one in which the incoming jets penetrate to the top of the calandria, collide and are redirected downwards through the central core towards the outlet. A buoyancy-dominated flow is one in which the jets do not penetrate to the top of the calandria but instead are redirected in the reflector region towards the outlet. Figure 7 illustrates the two basic types of flow patterns

that could exist within the calandria. Various intermediate or mixed flow regimes are also possible.

2.3   Moderator Circulation Modelling

In order to predict local moderator temperatures during steady state operation or following a loss-of coolant accident with loss of emergency core cooling it is necessary to employ CFD computer codes due to complexity of the physical system/processes. The problem requires advanced numerical techniques as moderator circulation combines the effects of forced convection due to inlet jets and natural convection due to internal heat generation in the fluid, in a complex geometry.

As part of the IST (Industry Standard Toolset) initiative, the computer program MODTURC_CLAS (MODerator TURbulent Circulation using Co-Located Advanced Solution) [5,6,7] has been adopted as the moderator circulation analysis tool within the Canadian Nuclear Industry. MODTURC_CLAS was derived from the CFD code TASC_flow by adding particular CANDU specific models to the code such as the heat exchangers, temperature controller, etc.

MODTURC_CLAS makes use of the most recent advances in computational methods to provide greater flexibility and economy in arriving at moderator temperature and flow predictions. It is a state-of-the art three–dimensional computational fluid dynamics computer code used to predict the flow and temperature fields of a single–phase turbulent incompressible fluid, subject to general boundary conditions in a prescribed geometry. The numerical method used in MODTURC_CLAS is a fully implicit, co-located, finite volume method with a flux element–based domain discretization. This combines the well known geometric flexibility of the finite-element method with the desirable conservation properties of the finite-volume method.

The governing partial differential equations for conservation of mass, Cartesian components of momentum, and energy form the basis of the governing equations in MODTURC_CLAS. The principal solution variables are the Cartesian components of velocity, pressure, temperature, and turbulent kinetic energy and dissipation, with all other variables being functions of these. The fully Cartesian co-ordinate system and co–located nodal arrangement for the solution variables utilised by MODTURC_CLAS avoids the complexity of a general tensor or non–orthogonal staggered grid formulation and, coupled with the numerically conservative finite volume approach, yields the desirable strong conservation properties.

2.4   Experimental Support-Code Validation

MODTURC_CLAS has undergone extensive validation[8] against small scale tests performed at STERN Laboratories in Hamilton, Ontario, Canada. The small-scale test facility embodied all of the salient features of a typical CANDU 6 calandria, i.e., inlet jets, a tube–matrix (representative of the CANDU core), and internal heat generation. Several experimental tests were performed at

STERN Labs including both isothermal and heat addition cases. Several distinct experiments were conducted with the test apparatus. These include isothermal tests, a nominal-flow test(approximating real calandria operating conditions) and a low flow test(corresponding to a reduction in flow which generates a buoyancy dominated flow), and an enforced symmetry test(performed by placing a divider in the vessel). Transient experiments were also carried out to simulate postulated accident scenarios corresponding to those analyzed in CANDU safety reports. Pickering B moderator temperature measurements (over a vertical line within the moderator) indicate that MODTURC_CLAS reasonably well. MODTURC_CLAS has also been validated against in-core measurements taken in the Bruce A and Pickering B calandrias.

In addition to the current work, steady-state and LOCA/Loss of Class IV Power transient analyses have been performed using MODTURC_CLAS for the CANDU 9 moderator system [9]. Other accident scenario results for the CANDU 9 moderator system (e.g., LOCA/LOECC) using MODTURC_CLAS has recently been analysed. Other analysis applications can be found in [10,11,12]

More recently as part of the CANDU 9 validation effort a part scale three-dimensional moderator rig was constructed a Chalk River Laboratories. Validation exercises are now under way for MODTURC_CLAS simulation of the CANDU 9 configuration for both steady state and transient analyses. Plans are also in progress to perform experiments relating to the CANDU 6 configuration.

3.   CONTAINMENT SAFETY SYSTEM

3.1 Containment System description

Under the most simplistic usage, the word "containment" implies the reactor building, which is also referred to as the containment building. Under broader use the word signifies a combination of systems, phenomena and processes which, when acting together, ensures that the environmental risk posed to the public is acceptable if radionuclides (radioactive materials) escape from the cooling system of the reactor. In general usage containment means a special safety system that is designed to seal or envelope (contain) coolant and radionuclide release from the reactor assembly. Containment encompasses all of the phenomena involving containment in fuel, containment in the primary circuit and containment in water.

Containment in the reactor building considers the driving force for escape of radionuclides and containment processes. The driving force is a pressure difference across the containment envelope caused by steam/water discharges inside containment and/or by the continued operation of the ventilation system. This driving force is primarily responsible for the transport of radionuclides within containment. Containment processes include radionuclide chemistry,

gravity forces (buoyancy, settling etc.), deposition and binding of radionuclides on surfaces and retention of radionuclides by filters and concrete pores, cracks, pools etc.

Containment design considers both the natural processes (physical and chemical aspects) and the effect of the equipment inside containment, which is provided for either normal operation or to supplement the containment design. Designed and engineered subsystems improve containment performance and provide in depth defense.

Engineered containment subsystems include the primary enclosure, the secondary enclosure, the isolation system, pressure relief systems (dousing), air coolers, heat exchangers, filter systems, gross leakage detection systems, etc. Of all these possible groups, CANDU 6 containment design includes a leak tight reactor building which can withstand high pressures (with leakage much less than five percent by volume per day at a design pressure of 124kPa(g)), a very fast isolation system, a large dousing system for pressure suppression, local air coolers. Additional heat sinks may include emergency core cooling and moderator system heat exchangers, and as an option and not part of the design basis, an operator initiated post-accident containment depressurization facility. Figure 8 depicts the primary containment thermalhydraulic phenomena.

3.2   Safety Concerns

During normal operation, the containment system serves to control minor releases of radionuclides from the reactor assembly but its primary purpose is to contain any radionuclides that may be released into containment if a breach occurs in the heat transport system boundary. Containment provides the final barrier to limit the release of radionuclides to the environment.

In addition to radionulides being released from the break and hence having the potential to be released to the external environment there also exist the concern of a hydrogen burn in containment due to hydrogen release. This safety aspect is as important as possible radionuclide release since a hydrogen burn deflagration/detonation could possible damage safety systems.

During certain postulated accidents, (LOCA+LOECC), hydrogen gas would invariably be transported from the fuel channel, where it is produced, to the containment atmosphere. The hydrogen/steam mixture may be released either as a jet or a plume into the reactor vault or into the feeder cabinet through a break in the heat transport system. Once hydrogen is released, various mixing mechanisms are available to promote dispersion within the accident vault and the containment building in general. These include natural convection, due to density differences and thermal gradients for inter room mixing, and forced convection from local air cooler fans for mixing within the accident vault. Since the CANDU 6 containment is large, the uniform concentration of hydrogen predicted to occur during these postulated accidents is near or below the flammability limit for hydrogen/steam/air mixtures. During the hydrogen transport and mixing period, the hydrogen concentration in the accident vault may exceed the flammability limit and undergo ignition if a sufficiently strong ignition source present to overcome flammability

suppression due to the presence of water mist and steam. Furthermore, if complete mixing is not achieved, there may be local areas containing hydrogen/air/steam mixtures within the flammable range.

3.3 Containment Modelling

The objectives of performing containment thermalhydraulic analyses at AECL are to estimate the following safety related quantities following postulated accidents:

1. Global pressure history to assess containment integrity, and estimate the timing and initiation of pressure dependent signals;
2. Pressure differentials between adjacent compartments to assess internal structural integrity;
3. Release path flow/energy rates from containment to the external atmosphere via various release paths for the purpose of estimating radionuclide releases from containment (subsequently used for dose analysis);
4. Local temperature and humidity conditions for environmental qualification purposes;
5. Steam/non-condensable gas distribution and concentration inside containment (especially hydrogen) to evaluate the potential of a hydrogen burn.

Objectives 1,2 and occasionally 4 can be adequately modelled by using lumped parameter codes (which assume that all physical properties within a volume are constant, i.e., temperature, pressure, density). The AECL computer program PRESCON2 (a lumped parameter code) [13] has been used extensively over the past 20 years to perform containment safety analysis[14-17]. PRESCON2 solves the conservation equations for mass, momentum and energy based on the homogeneous equilibrium model. Flows between nodes are governed by a one dimensional momentum equation. The program also has submodels for wall heat transfer, dousing, fans, air coolers, etc. It's companion code SMART (also a lumped parameter code)[18] is used to address objective 3. SMART tracks aerosol transport and behaviour (with embodied radionuclides) as well as radioactive noble gases in containment based on the flow history predicted by PRESCON2 and subsequently estimates releases to the environment. SMART also tracks the iodine inventory in the sump. The program assumes a drop size aerosol distribution for the liquid portion of the break and models aerosol depletion processes such as agglomeration, gravitational settling, etc[19,20].

Lumped parameter codes are not particularly suitable for buoyancy driven flows that result from small temperature gradients and hence small pressure differences. These flows are usually not well mixed and as a result three-dimensional codes are more suited for this purpose. The general-purpose containment analysis code GOTHIC (Generation of Thermalhydraulic Information in Containment) has been adopted as the Canadian Nuclear Industry Standard tool for performing containment analysis. GOTHIC has lumped parameter capability but has the additional capability to model flows in 1,2 or 3 dimensions using cartesian co-ordinates and was specifically written for design, licensing safety and operating analysis of nuclear containments and

confinements, auxiliary buildings and related equipment, with applicability to a wide range of other non-containment specific problems. GOTHIC solves mass, momentum and energy balances for three separate phases: vapor, continuous liquid (pool, films, etc) and dispersed liquid (drops). The vapor phase can be a mixture of steam and non-condensible gases and a separate mass balance is maintained for each component of the vapor mixture.

The phase balance equations are coupled by mechanistic models and correlations for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow as well as single phase flows. The interface models allow for the possibility of thermal non-equilibrium among the phases and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms (in the Navier-Stokes equations) in multi-dimensional models together with he k-e turbulence model. Special models for engineered safety equipment such as pumps, fans, valves, heat exchangers, coolers etc are also included.

In the near future GOTHIC may replace PRESCON2 following validation of its lumped parameter capabilities as applied to CANDU 6 containment.

The primary use of the GOTHIC code at this time is however to determine hydrogen distribution in CANDU containment following a LOCA/LOECC event and the consequences of a hydrogen burn should it occur [21-25]. GOTHIC has the capability to model hydrogen deflagration in 3 dimensions but not as yet detonation. GOTHIC has also been used for various CANDU applications [26-27] including future passive design studies[28-29]

3.4 Experimental Support-Code Validation

PRESCON2 has an extensive validation base relating to pressure and pressure-differential prediction within containment. It has been validated against some of the HDR hydrogen mixing experiments with limited success[30,31]. A similar statement can be made concerning the GOTHIC code when used in lumped parameter mode. Many of the tests are for blow-down simulations with basic lumped volume modeling. Both SMART and GOTHIC have recently been adopted as the Canadian Nuclear Industry Tools for radionuclide behaviour and transport in containment and hydrogen distribution/burn in containment respectively. As part of the Industry Code Validation Plan both SMART and GOTHIC are undergoing validation at Whiteshell Laboratories for their intended application.

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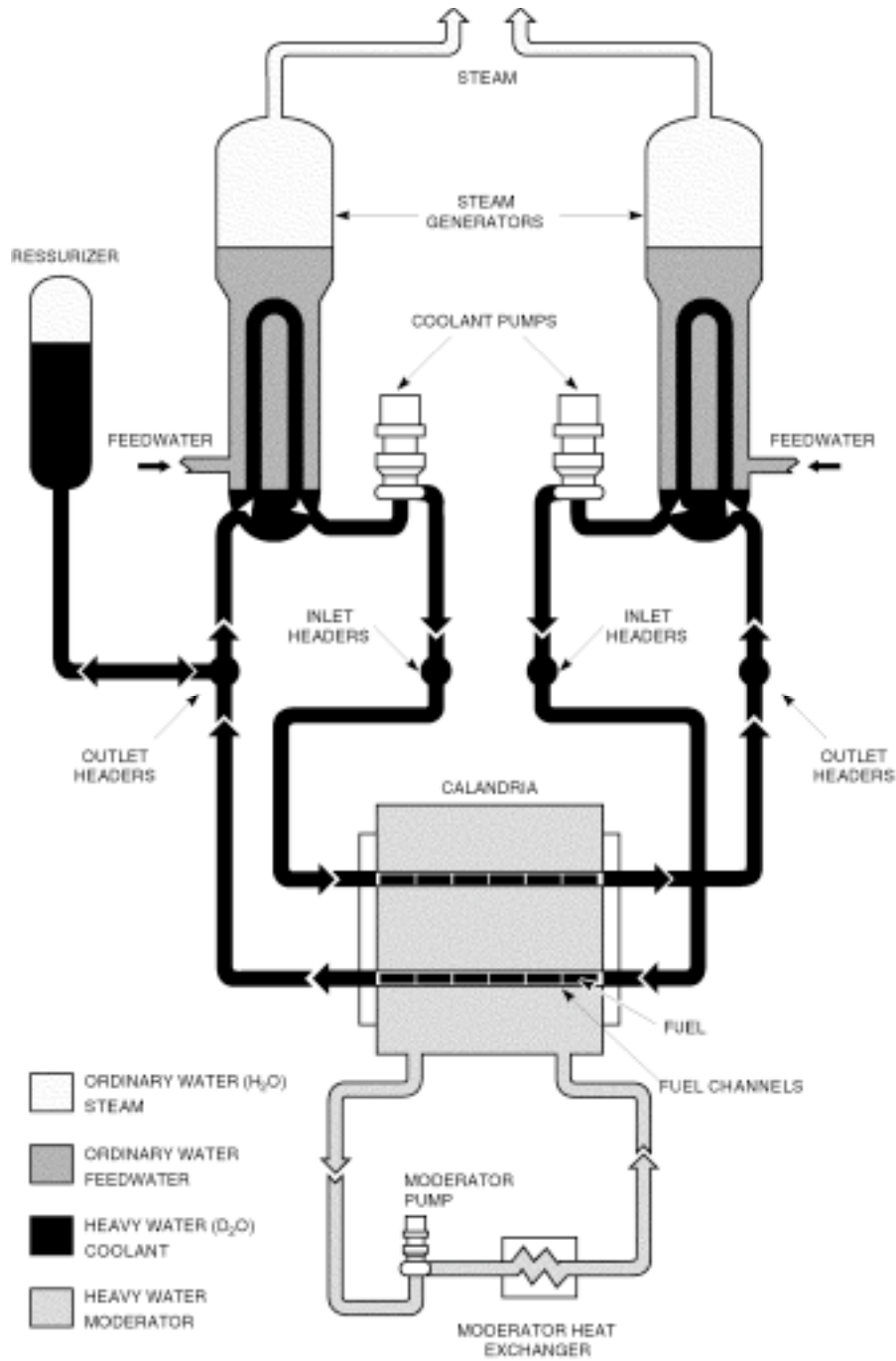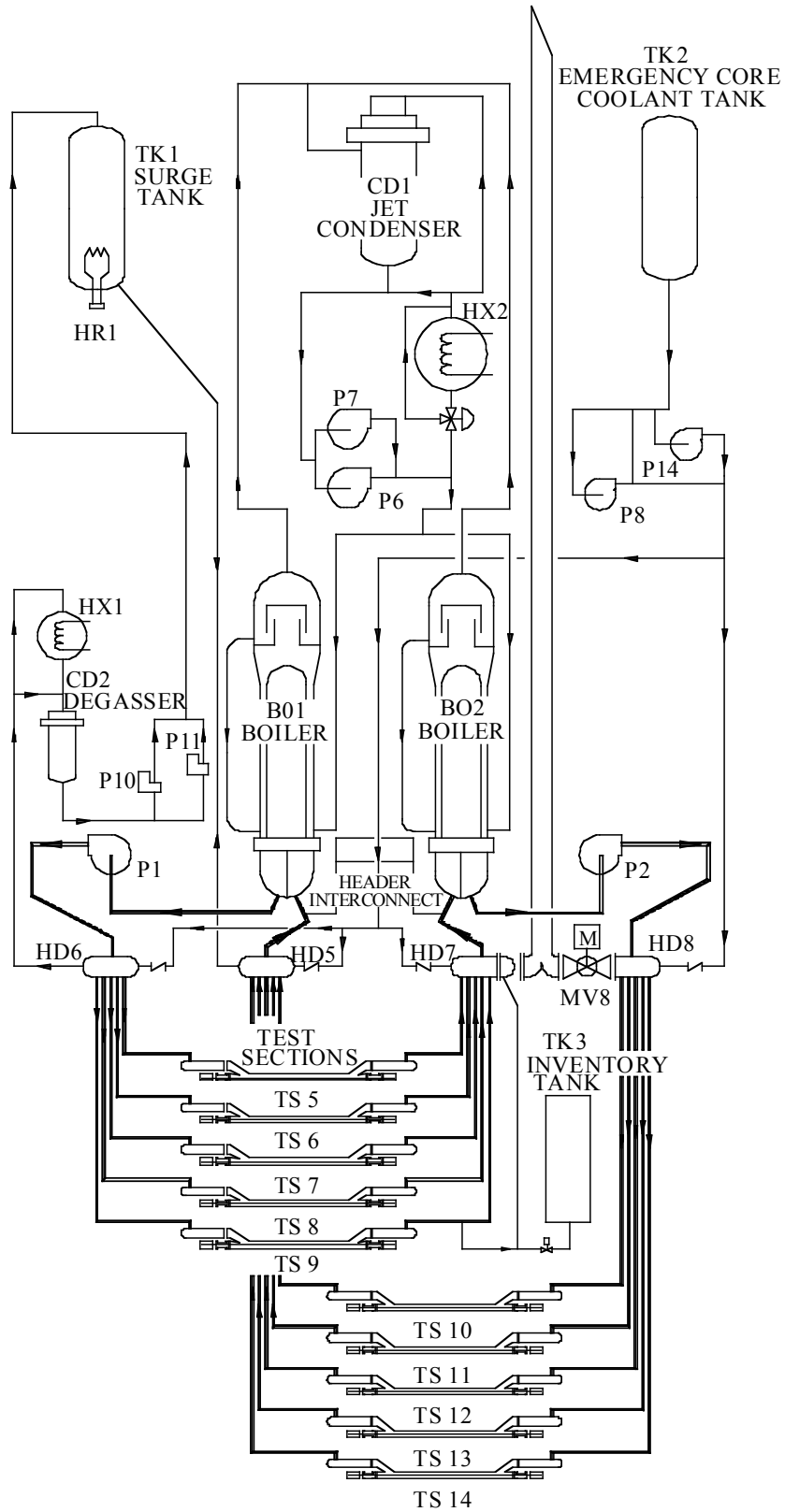
Figure 1: Schematic diagram of the CANDU heat transport system

Figure 2: Schematic Diagram of the RD-14M Integral CANDU Test Facility
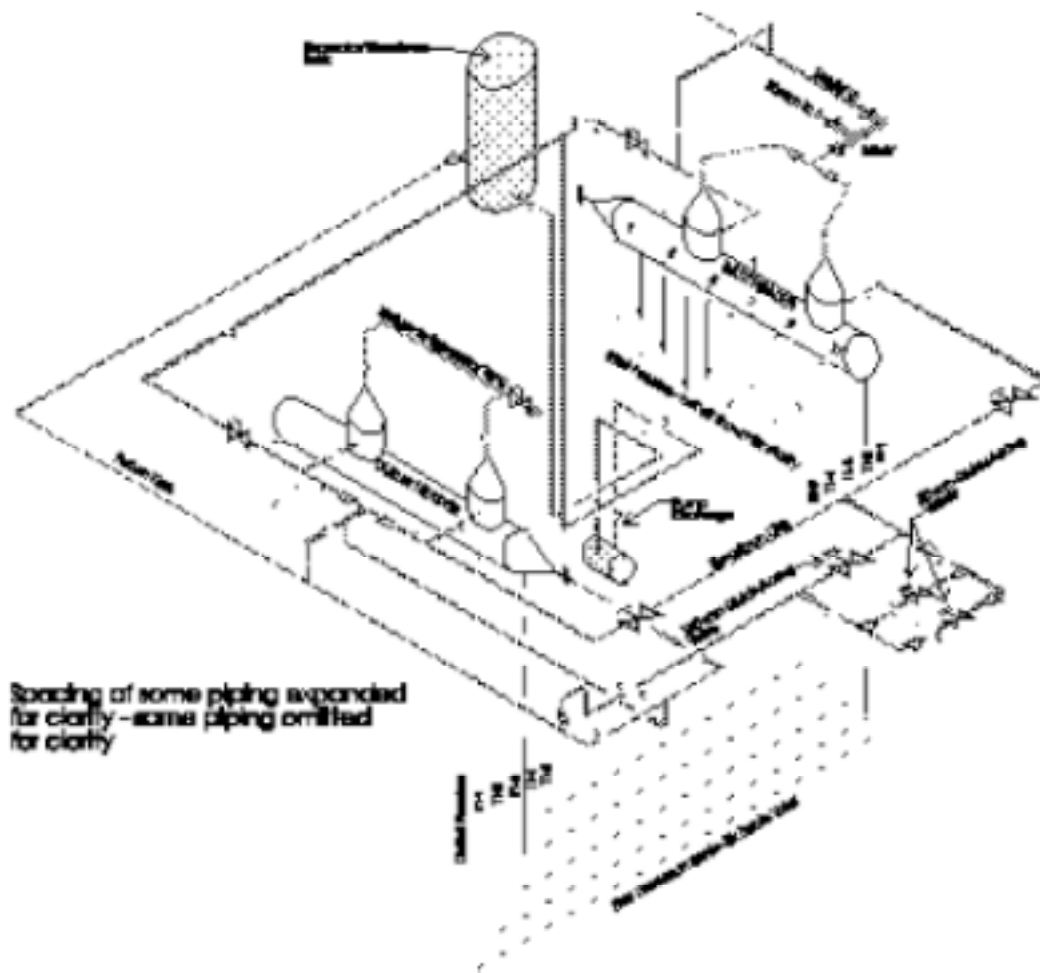
Figure 3: Schematic Diagram of the LASH Facility
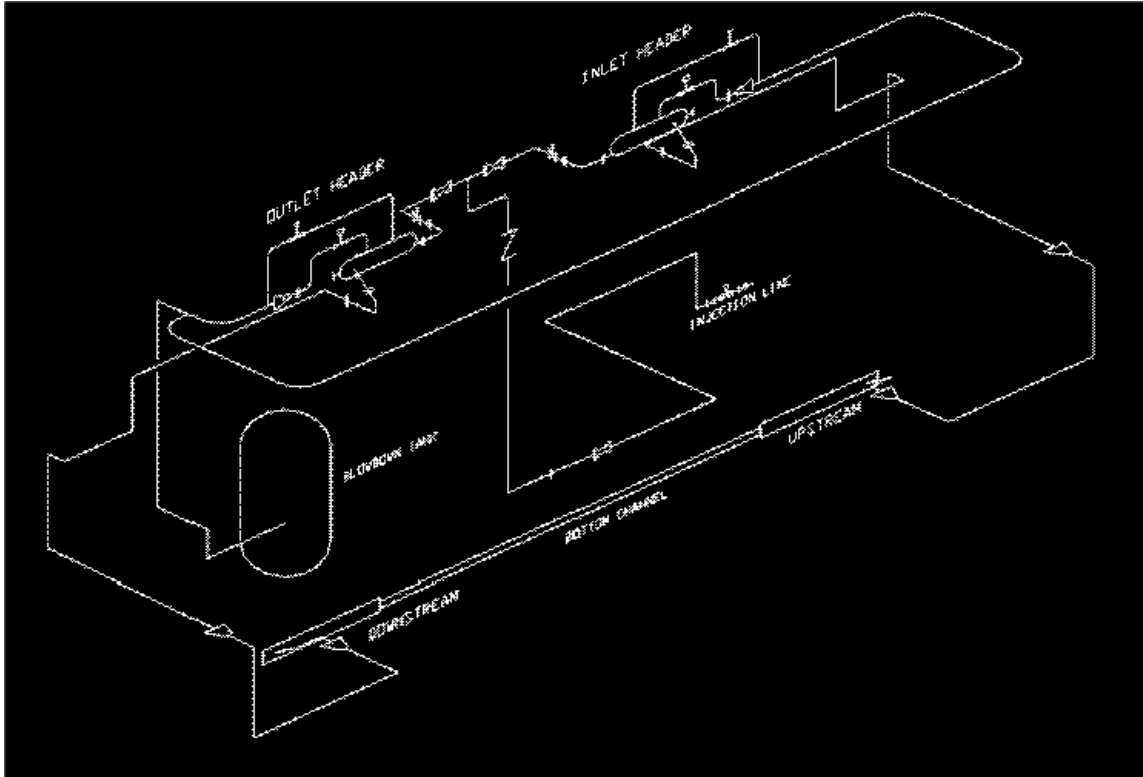
FIGURE 4: Schematic Diagram of the CWIT Facility
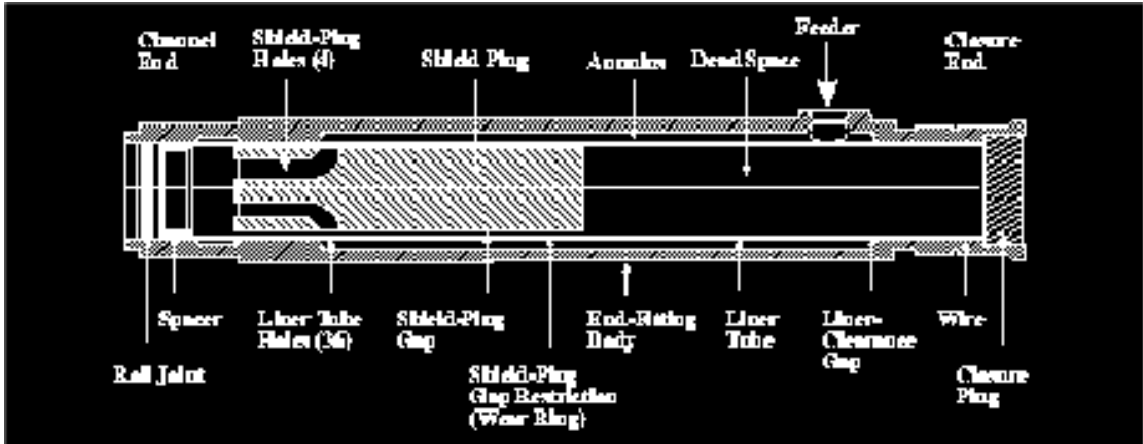
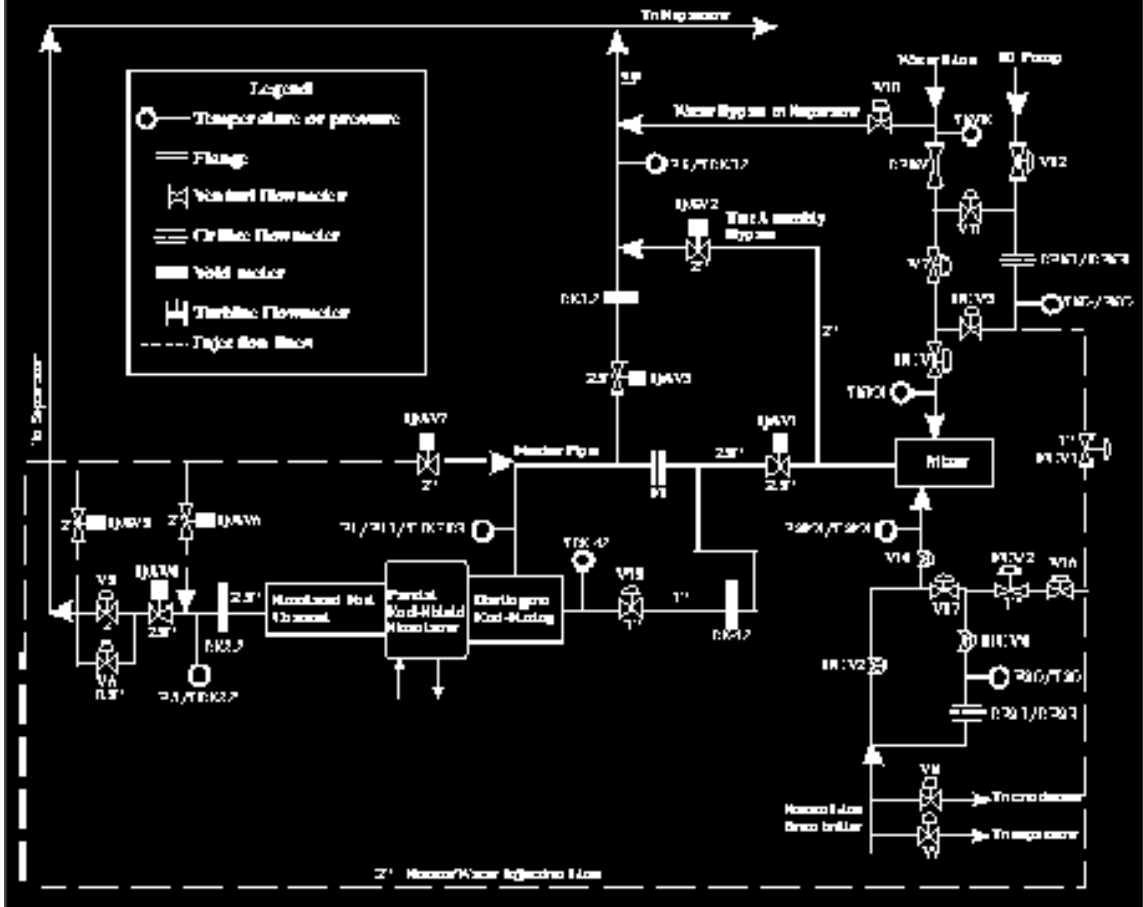FIGURE 1: Schematic Diagram of Darlington End Fitting

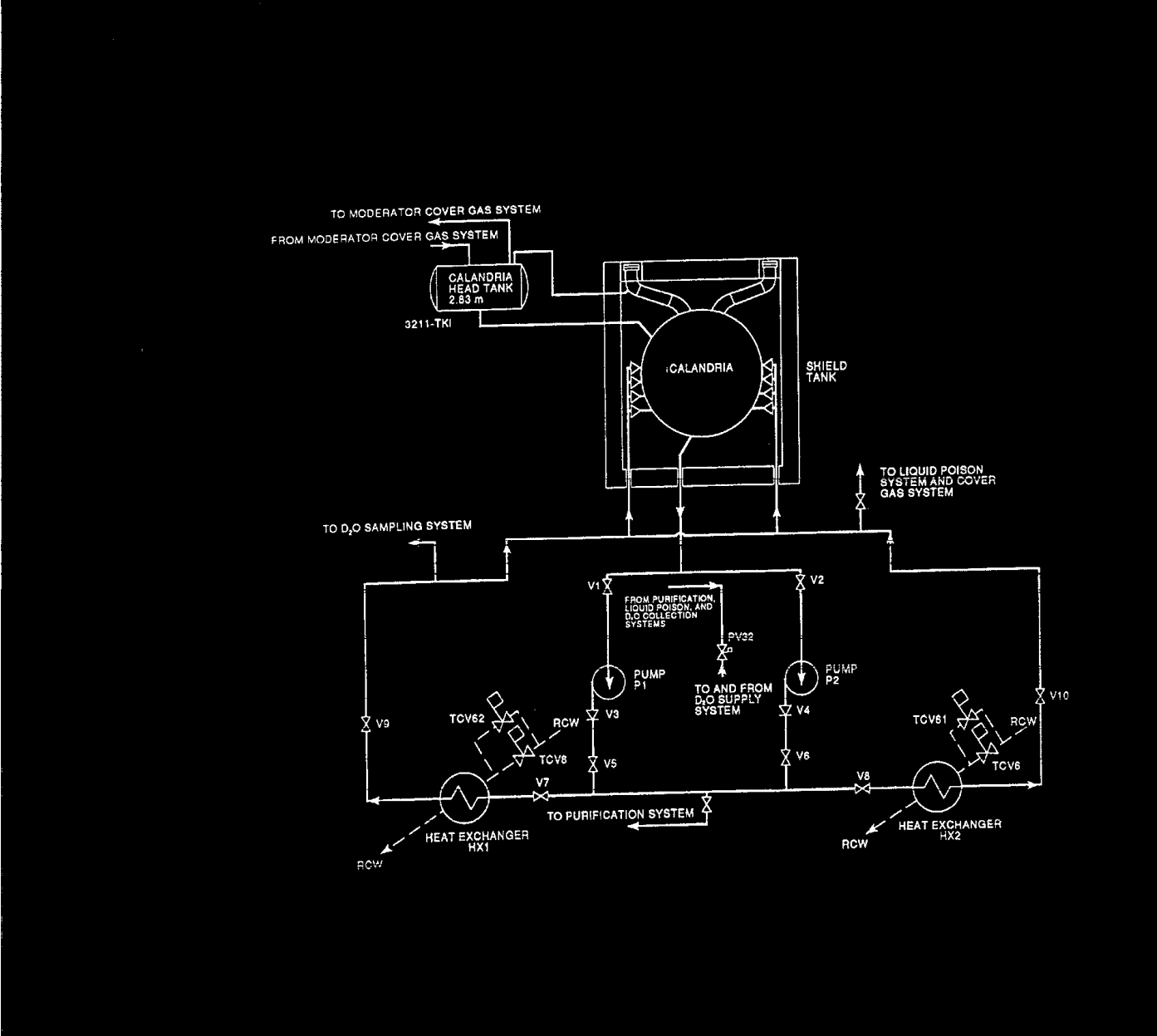Figure 5: Diagram of the End-Fitting Characterization Facility

Figure 6: Diagram of the main moderator system

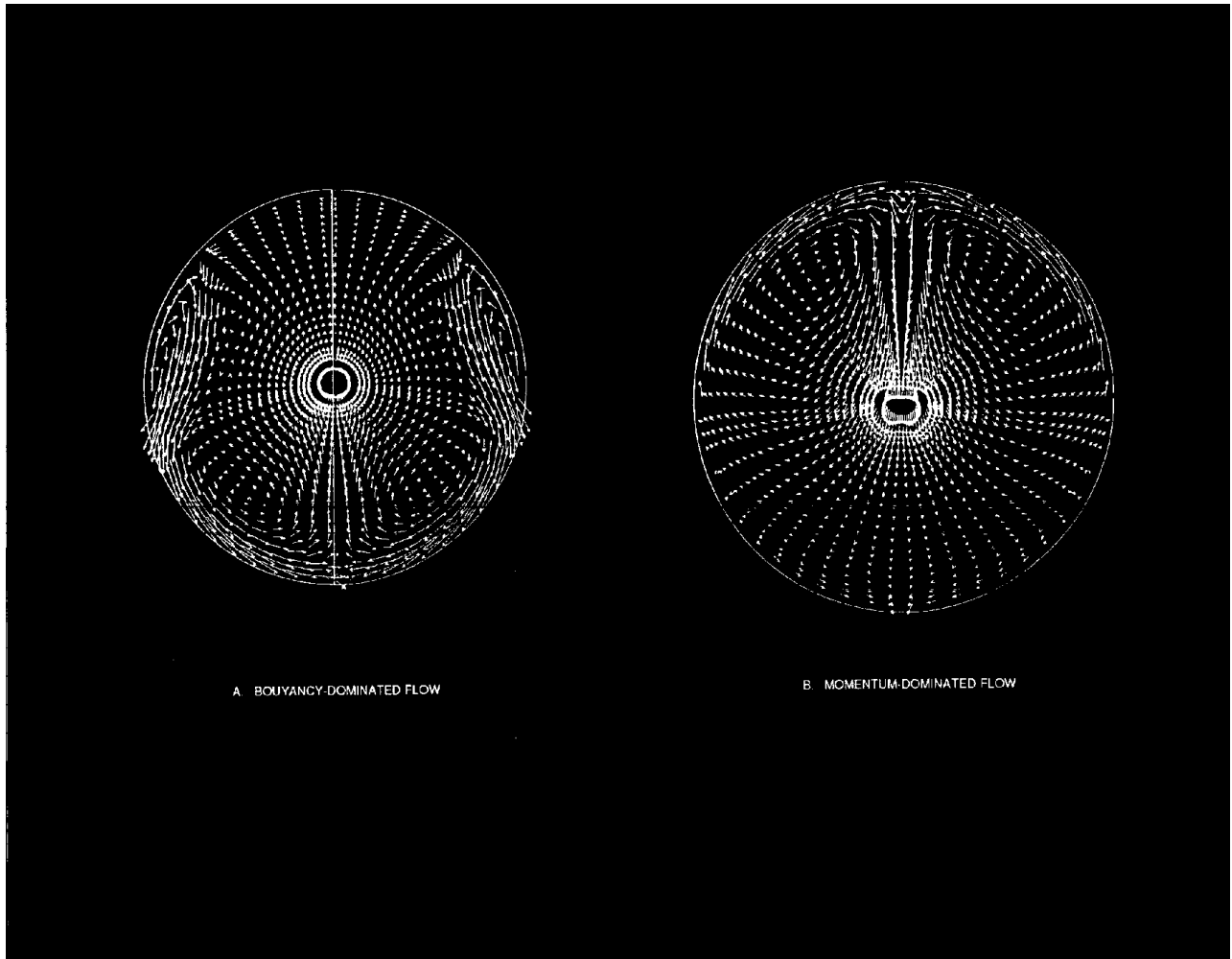A. BOUYANCY-DOMINATED FLOW

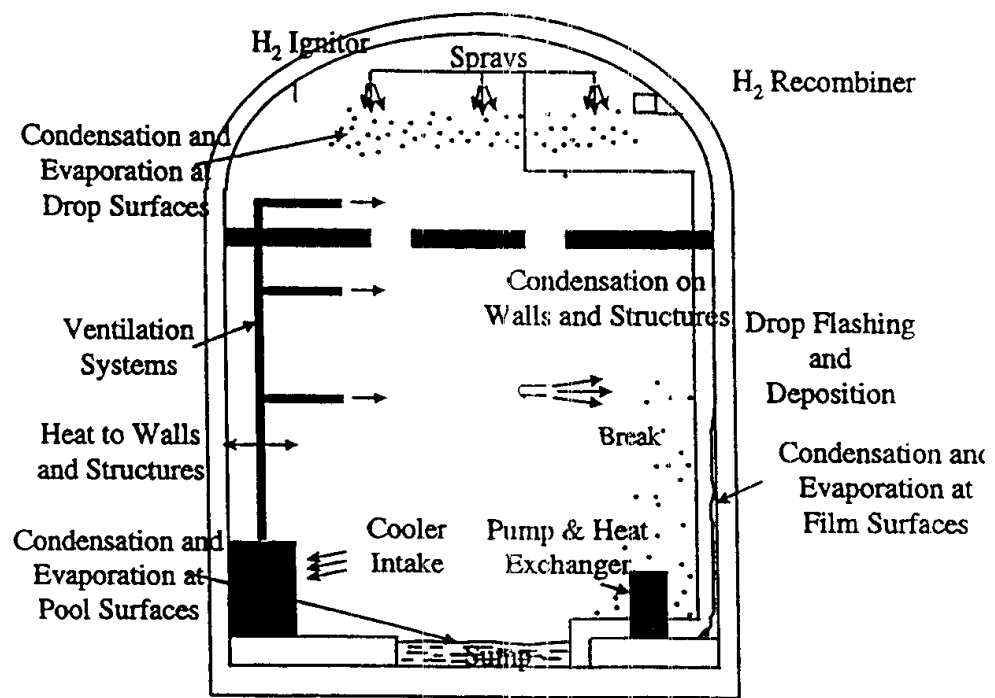B. MOMENTUM-DOMINATED FLOW

Figure 7: Types of flow patterns inside the calandria

Figure 8: Containment thermalhydraulic Phenomena