MODULAR ACCIDENT ANALYSIS PROGRAM FOR CANDU REACTORS

C. BLAHNIK: C.S. KIM: S. NIJHAWAN*, R. THURAISINGHAM
Ontario Hydro, 700 University Avenue, Toronto, Ontario, M5G 1X6
*MIR Consulting Inc. Halifax, N.S. B3H 3P2

Abstract

MAAP-CANDU is an integrated computer code for the best estimate analyses of severe accident scenarios in CANDU nuclear power stations, including those with significant core damage. It is based on the widely used MAAP-LWR code with a number of models specially developed for CANDU. Dynamic feedbacks between plant systems and all known natural severe accident phenomena are modelled. This paper describes the key features of the code with focus on CANDU-specific models.

1. INTRODUCTION

The Modular Accident Analysis Program (MAAP) is a family of integrated computer models for the analysis of severe accidents in nuclear power plants. The severe accidents are those not routinely analyzed as part of the design and licensing process for a plant, because the probability of such events is extremely low. They can involve extreme temperature excursions, a large release of radioactive fission products from the fuel and severe damage to the plant.

The CANDU version of the code (MAAP-CANDU) has been developed between mid 1988 and the end of 1990. The development facilitates a recommendation of the Ontario Nuclear Safety Review (ONSR) that severe accident analyses be analyzed for Ontario Hydro's plants[1]. MAAP-CANDU is based on the MAAP-LWR[2] used widely around the world for risk assessment studies. The source code has been reviewed by experts and extensively validated[2]. The CANDU-specific models were developed by a team from Ontario Hydro and international experts under strict quality assurance (QA) guidelines. The purpose of the QA is to maintain the integrity of the generic and phenomenological models while ensuring the validity of new, CANDU-specific models. The MAAP-CANDU code thus retains the benefit of extensive international research and development (R&D) in severe accident phenomenology, which is included in the MAAP code family, while representing the unique features of a CANDU plant.

MAAP-CANDU is fully documented. Detailed descriptions of models and input parameters are contained in the User's Manual, along with extensive references and various validation and verification documents The code is operational on micro, mini and main-frame computers. It is currently being employed in the analyses of severe accidents for the Darlington Nuclear Generating Station (NGS) which quantitatively explore the propagation of certain initiation sequences with the potential for a severe core damage. The sequences are extracted from the risk assessment study for this plant[3].

This paper discusses the MAAP-CANDU code with focus on the CANDU-specific features. Following an overview of the MAAP modelling approach, the main system models are discussed, and the experience with code application is described.

2. MAAP MODELLING APPROACH

The MAAP code has been developed according to the following principles: all reactor systems and structures (including the engineered safety systems and natural heat sinks) should be represented; all known severe accident phenomena shall be represented; the process and phenomenology models shall be fully integrated to dynamically simulate feedback effects; the code shall be flexible to allow a detailed representation if a certain process or phenomenon is found to significantly affect the accident progression or consequence; and the code should be efficient (fast running) to facilitate analyses of long duration accident sequences with alternate progression pathways. These principles have been implemented in a highly modular FORTRAN program. The modules, which can consist of a number of the models and sub-programs, are implemented in the code in accordance with the following five categories:

High level routines: Direct the computation sequence through the code and do not contain any physical models. They include the main program, the input-output, data storage and retrieval subroutines, and routines that perform integration, control time step and direct calls to system and region subroutines.

System status routines: Monitor and record the status of

the entire plant in each time step. The status is monitored by setting and removing event flags for the individual systems (i.e. pressurizer empty, moderator cooling systems off, non-accident unit vault coolers operating, hydrogen burn occurring in containment, etc.) and interventions (safety relief valves manually opened, lost power restored, etc.). The events flags are subsequently used by the region routines to direct the calls to the appropriate phenomenology subroutines.

Region routines: Each routine represents a physical region of the plant such as the primary heat transport system, the calandria vessel, the vacuum building, etc. The region routines assemble the rate equation for the integrator using results from phenomenology routines (e.g. break discharge, boil-off rate, hydrogen burns, fission product transients).

Phenomenology routines: These routines are the fundamental elements of the code. They describe the physical processes that occur in each system and region of the plant. They include conventional models (e.g. two phase break discharge, transient heat conduction) as well as models specific to severe accidents derived from the international R&D program over the past decade or so. Examples of the latter are the various chemical reactions at high temperatures, debris behaviour, flow mixing in multi-compartment volumes, fission product release and aerosol transport.

Property & Utility routines: These routines supply physical properties of materials and fluids.

MAAP-CANDU solves a set of coupled, first order ordinary differential equations. Conservation equations for mass and energy are set up for each physical region of the plant. The momentum balances of the regions are assumed to be quasi-steady. This assumption reduces them to algebraic expressions and eliminates the need for differential equations describing the conservation of momentum.

3. HEAT TRANSPORT SYSTEM MODELS

The heat transport system models are schematically shown in Figure 1. They consist of three region routines, namely the primary heat transport system, the pressurizer and the steam generator adapted from the MAAP-PWR code. Each region is represented by a single control volume from the standpoint of thermal-hydraulics. Since accident sequences of interest involve a loss of coolant without makeup, any further detail in the representation is not warranted for these systems. For the same reasons, the Emergency Coolant Injection System is modelled only to the extent that the user may specify an addition of water or steam to the primary heat transport system according to the accident scenario circumstances as a part of recovery actions The emphasis is placed on the representation of the heat transfer to the various heat sinks including the engineered and structural heat sinks, the chemical environment in the systems and the transport and deposition of fission products along with their decay heat. Appropriate control logic is available for the various relief valves, pressurizer heaters and steam generator feedwater supply. Structural heat sinks are modelled using a two-dimensional slab model.

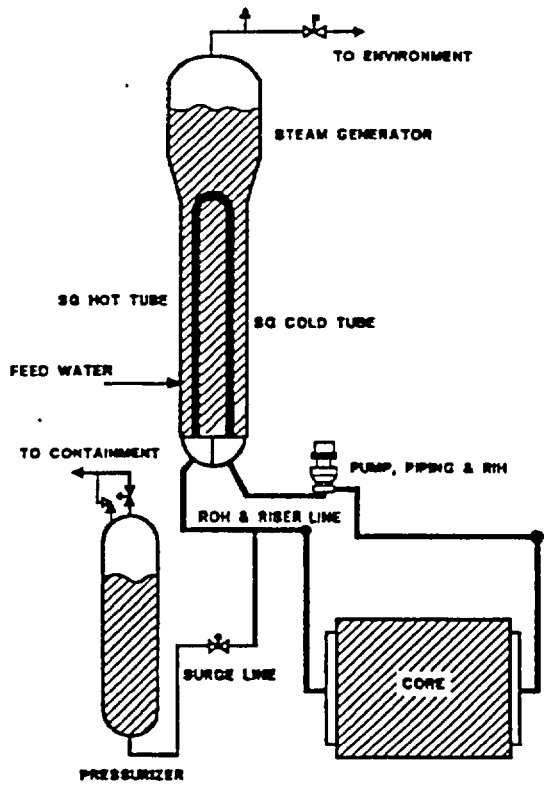

Figure 1 : Schematics of the Heat Transport System Model.

The heat transport system employs a two-phase, homogeneous thermal-hydraulic model prior to phase separation. Once the phases become separated, a lumped, multi-component, non-equilibrium model is activated. The components treated include not only water and steam, but also the full range of non-condensable gases which may be generated within the system (e.g. H_2) or enter it through the breaks (e.g. O_2, N_2 and CO_2).

The core is represented in the primary heat transport system model by a lumped parameter model until the onset of heatup. Subsequently, the primary heat transport

system model activates a separate core heatup model described in the following section. At all times, rates of interchange of mass and energy, including those of fission products, with the interfacing systems are calculated. The heat transport models interface with the calandria vessel via channel failures, the containment via piping failures or pressurizer relief, the confinement via pump seal failures and the environment via steam generator steam discharge valves. The breaks are triggered by separately derived failure criteria[4] and the relief valves operate according to the appropriate control logic. The models are also thermally coupled with all the above systems as well as the shield tank via the end shield lattice tubes.

The rates of change of the dynamic variables within a region are calculated from the net balance of mass and energy flows into or from the region (e.g. break flows, conduction to heat sinks), the rates of the internal processes (e.g. flashing, rainout, interfacial heat transfer) and the local heat source terms (e.g. heat generation in the fuel, debris and volatilized fission products). Concentrations and temperatures of all fluid components are evaluated. Inventories of twelve fission product groups are tracked in gaseous, suspended aerosol and deposited aerosol states. The changes in local heat source terms due to the release and transport of fission products are dynamically updated.

4. CORE HEATUP MODELS

The core heatup models are unique to CANDU reactors. They are activated if and when the liquid and steam phases separate in the primary heat transport system during the accident. First, a boil-off of any residual water in the fuel channels is simulated until the channels dry out. During the boil-off period, the channel is represented by a lumped heat source with heat losses to the calandria vessel and the end-shields accounted for. The amount of residual water depend upon accident scenario and is specified from a separate analysis. Typically, very small breaks in the heat transport system can result in the channels and feeders full of water at the time of phase separation, while large breaks can lead to essentially fully voided channels. The fuel conditions at the end of boil-off are specified in the input. In most circumstances, the channels experience only mild temperature excursions during this period.

When the channels dry out, a series of thermal-mechanical models are activated. The processes and phenomena represented during this period are highlighted in Figure 2 include:

Thermal-hydraulic, thermal-mechanical and thermal-chemical transients in intact and partially disassembled fuel channels with a steam/hydrogen flow on the inside and with their calandria tubes either submerged in moderator (intact channels) or uncovered and exposed to the steam and H_2 environment in the calandria vessel (intact or broken channels). The channel phenomena modelled are the deformation and relocation of channel components including effects on the flow patterns within the channel, the exothermic reaction between Zircaloy and steam including the resulting changes in the fluid properties due to chemical conversion of H_2O to H_2, the release of fission products and their associated decay heat and the disassembly of channel segments either due to an excessive strain or by melting of the channel walls.

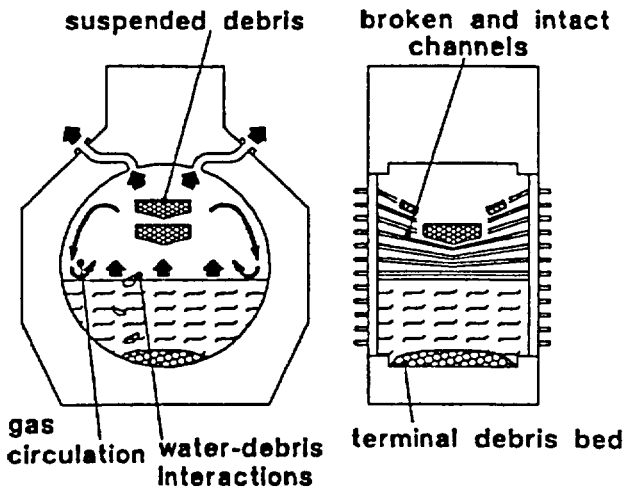

Figure 2 : Core Heatup Phenomena

The formation, heatup and motion of suspended channel debris beds (i.e. solid channel debris temporarily supported by the underlying channels), the metallurgical transformations within these beds (i.e. alloying of Zr, ZrO_2 and UO_2) and the release of volatilized fission products and associated decay heat from these beds.

The thermal and chemical interactions of the debris falling into a water pool at the bottom of calandria vessel (i.e. the quenching of debris and the reaction of molten Zr with liquid water). The behaviour of the terminal debris bed at the bottom of the calandria vessel is computed by the model described in the following section.

Local steam flow and temperature patterns on the outside of calandria tubes in the uncovered region of the core.

governed by forced flow due to steaming from the calandria vessel pool or by buoyancy when the steaming subsides The flow fields take into account any obstructions formed by the suspended debris beds

The core heatup models are fully integrated with the interfacing systems (i.e. with primary heat transport system, calandria vessel and end-shield models). The progression of core degradation influences the thermal-hydraulic response of these regions with a feedback on the core behaviour via altered boundary conditions.

To facilitate this level of integration, the fuel channels are grouped according to power and elevation into sets of entities with similar characteristics. Each set is then represented by a "characteristic" channel which is modelled in detail. The behaviour of all other channels (called the "associated" channels) within a set is assumed to be identical to that of the characteristic channel. It may be offset in time for the associated channels if the group represents several channel rows which may then dry out at different times due to different water volumes in the feeders.

The characteristic channel is represented by a generalized, multi-node annular ring model. Each ring preserves material properties, areas and volumes of the component it represents. The number of rings and that of the axial nodes can be user specified. In the reference model, 9 radial and 13 axial nodes are employed as shown in Figure 3. Steam and H_2 flows within an intact channel are determined dynamically from the chemical environment in the primary heat transport system, a user specified pressure differential across the reactor header, the channel resistance and the fuel heat output. Once the channel fails, the internal flows are determined by differences in the fluid conditions between the primary heat transport system and the calandria vessel.

The channel failure and disassembly are modelled by a separately derived failure criteria[4]. Any channel segments that meet the disassembly conditions move into a debris bed which may be suspended within calandria vessel (i.e. rest on still intact channels or located on the bottom of calandria vessel). A spillage of peripheral fuel can also be triggered when a user specified number of the channel central segments have disassembled. The pressure and calandria tube masses of these peripheral nodes are assumed to remain attached to the calandria vessel tube-sheet.

The boundary conditions on the outside of the calandria tubes are evaluated locally for different regions of the core. The core is divided into of core nodes within which the fluid conditions are assumed to be the same.

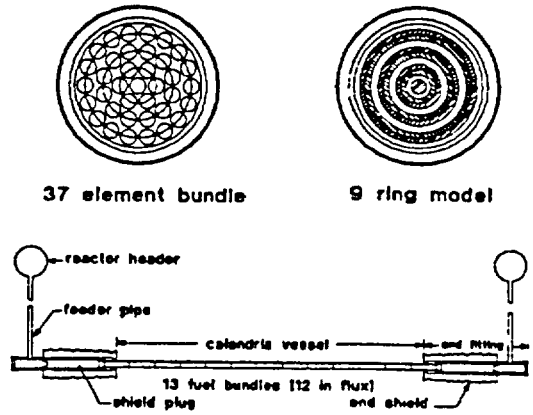

Figure 3 : Fuel Channel Model

The nodes are defined by up to 24 equal, horizontal slices and 5 vertical slices or axial segments. Each horizontal slice contains three characteristic channels. In the reference nodalization scheme, 30 core nodes are employed, formed by 5 axial slices and 6 vertical slices through the calandria vessel. Thus, calandria vessel fluid conditions are the same within 2 or three bundle lengths over 4 channel rows in the reference scheme. The flow and temperature fields through these nodes are then evaluated based on the steaming rate in the calandria vessel and the presence of suspended debris within the nodes.

The suspended debris behaviour is modelled on a core node basis. Since there is uncertainty with the motion of debris through a maze of underlying horizontal obstructions, the models have been designed to facilitate parametric analyses. Thus, the effects of different debris motion patterns (e.g. from an immediate relocation to the terminal debris bed upon disassembly, to the longest retention in the suspended form until the materials melt) can be studied.

The fission products are modelled on the fuel ring basis until the melting of channel segment walls. The residual inventories are then homogenized within the debris bed as are the fuel temperatures. The fuel is assumed to be failed at the onset of core heatup which is adequate for most severe accident analyses. A fuel failure model is also available which correlates the burnup-power dependant ring location with the failure temperature[4]. The release is modelled by user selected correlations of

experimental data [6]. The fuel decay power in the channels as well as the suspended debris beds is corrected at all times for the heat carried away by the fission products.

5. CALANDRIA VESSEL AND SHIELD TANK MODELS

The calandria vessel and the shield tank are systems unique to CANDU reactors. Region routines to represent these systems have been specifically developed utilizing the MAAP phenomenology and property routines. The systems are schematically shown in Figure 4. A lumped, multi-fluid, non-equilibrium thermal-hydraulic model is employed in these region routines, similar to that used in the heat transport system models.

The calandria vessel is represented by a single control volume which can contain intact or broken fuel channels, water, steam, non-condensable gases and debris. This model interfaces with the primary heat transport system via channel failures, the containment via rupture disc discharge, lattice tube leaks or seam failure and the shield tank via melt-through failure. The moderator cooling system, if available in the accident scenario, is also modelled. For scenarios with a loss of moderator cooling, a boiled-up water level is calculated, tracking the extent of core uncovering. The steam generation due to the heat transfer from submerged fuel channels, falling debris and the terminal debris bed at the bottom of the vessel is accounted for. Gas flow and temperature distributions in the uncovered region of the calandria vessel are calculated for forced and/or natural circulation, depending on the steaming rate from the liquid pool at the bottom. The steam condensation on the walls of the shield tank and end-shield is also modelled, as is the deposition and re-volatilization of fission product aerosols.

The quantity and the state of debris at the bottom of the calandria vessel is tracked (the suspended debris are tracked by the core heatup model). The quenching of debris, the chemical interactions of molten Zr with the water pool and the subsequent re-melting of debris are all modelled. The growth or shrinkage of crust thickness surrounding the molten pool is evaluated to calculate the heat transfer from the molten debris pool to the overlying water or gas, and to the calandria vessel wall.

The shield tank is represented by four control volumes, one for the main body of the tank including the shield tank extension, two for the end-shield and one for the head tank. The multi-volume representation is necessitated by the complex flow interconnections in this system. The shield tank cooling system network, the relief valves, the expansion tank vent and the expansion tank overflow to the active drain are modelled. The heat losses to the internal structures and to the containment

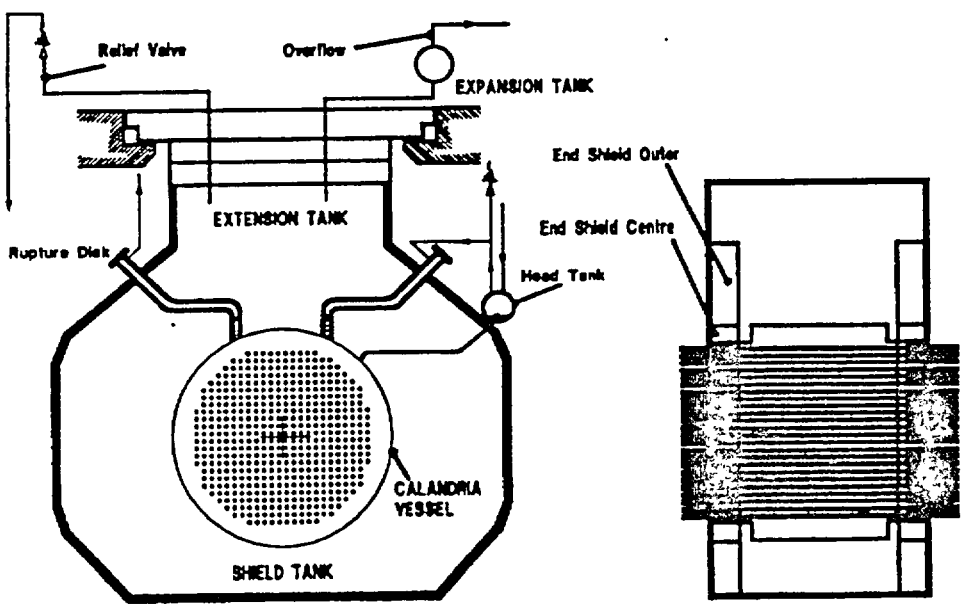

Figure 4 : Calandria Vessel and Shield Tank

through the walls are also modelled. The shield tank model interfaces with the calandria vessel via melt-through failure, the containment via pressurization and/or melt-through failure and the environment via reactivity deck seal failure. The breaks are activated by separately derived failure criteria[4]. The properties of water, steam, debris and fission products are dynamically processed and updated in each of the control volumes.

6. CONTAINMENT, CONFINEMENT AND VACUUM BUILDING MODELS

The nodalization scheme for the multi-unit Darlington NGS plant is schematically shown in Figure 5. The generalized model consists of up to 20 control volumes, each representing a region of the containment or confinement. The user specifies the properties of these regions (dimensions, material properties, heat sink characteristics, etc.) and how they are interconnected among each other as the plant-specific input. The input also defines the regions into which the molten core debris can flow and potentially interact with the concrete, the locations of sumps and the locations of engineered safety systems. The latter includes the vacuum building and its subsystems, the vault coolers, the post-accident water cooling system, the emergency filtered air discharge system, the hydrogen igniters. Each engineered safety system is represented by a separate model.

There are six possible sources of mass and energy discharge to the containment from the damaged reactor: The relief discharges from the pressurizer, shield tank and calandria vessel and the break flows from primary heat transport system, calandria vessel and shield tank. The discharge can consist of water, gases and molten debris. Other sources of heat modelled are the structural heat losses from the damaged reactor and the non-accident reactors, the heat transfer from the core debris on the containment floor, the decay heat carried by the fission products suspended in the atmosphere or deposited on the surfaces, and the chemical heat generated by the combustion of flammable gases. The heat is dissipated to the engineered safety systems, if available in the accident scenario and to the natural heat sinks (equipment metal, building walls and containment leakage).

In order to simultaneously evaluate the responses of all the containment and confinement volumes, various rates of changes are calculated. They are based on the rates of the interfacing regions and the phenomena occurring internally in each of the containment regions. The rate information is then fed back to the interfacing regions in the next time step with local implicit calculations performed when necessary.

The gas flow through vertical or horizontal junctions is modelled by considering the natural circulation (one directional Bernoulli flow as well as counter-current flow).

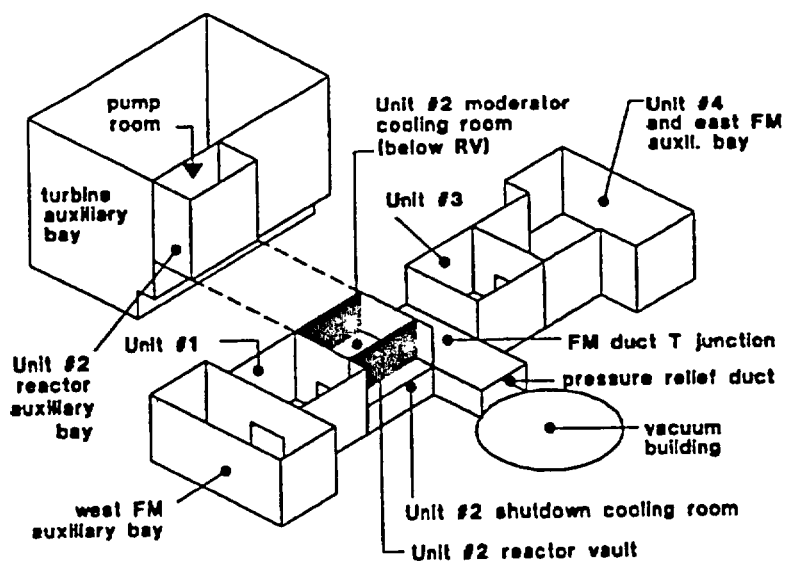

Figure 5 : Containment Nodalization

and the forced flow. The counter-current flow model for intercompartment flows is based on recent experimental data[7]. The forced flow is caused by a break discharge, the vacuum building pressure suppression processes and rapid steaming within the containment following an interaction of molten debris with water.

The transport of fission products is evaluated in all regions to which these material have access. With the exception of noble gases, the fission products and non-radioactive structural materials may be present in three states: vapour, suspended aerosols and deposited aerosols. The suspended aerosols are deposited by sedimentation, inertial impaction and thermophoresis using an experimentally determined correlation to the aerosol mass concentration in the atmosphere[8,9]. Re-suspension and re-volatilization of the deposited materials are also modelled[10,11].

If molten core debris comes into contact with concrete in the accident scenario, a model for core-concrete interaction is activated. This model simulates a one dimensional ablation of the concrete. The extent of attack predicted by this model differs slightly from the experimental observations which exhibit a two-dimensional behaviour (i.e. different attack rates at vertical and horizontal surfaces)[12]. If a more detailed representation of the core-concrete interaction process is required, a recent EPRI developed two-dimensional model can be employed. The products of concrete ablation (molten concrete constituents and gases) mix with the molten core debris, altering its properties and causing various chemical reactions with the molten metals. Some 21 reactions of 46 constituents are represented[13,14].

Flammable gases are produced by metal-steam interactions (H_2) and by chemical reduction of CO_2 generated by concrete decomposition (CO). These gases enter the containment atmosphere and are distributed among the regions by the inter-compartment flow. Models for a global compartment burn, an incomplete compartment burn (vented combustion) and standing flames (hydrogen-laden jets) are available [15]. The burns are initiated by igniters when the gas mixture in a compartment becomes flammable. If the ignition sources are unavailable in the accident scenario, the burns are initiated by auto-ignition or by a user defined criteria for spurious ignition.

7. EXPERIENCE WITH MAAP-CANDU

The MAAP-CANDU code has been successfully tested for several severe accident scenarios, including the total loss of heat sinks in the accident unit due to a loss of all electrical power. Figure 6 shows some results from a preliminary test run for the later scenario. Immediately apparent from this figure is the need for the computational speed, since the accident sequence can last for days. This run took 25 hours on a personal computer (PC) with a 386-20 processor. The same run on a 486-25 PC required 8 hours, while 4 hours were needed on a UNIX based engineering work station. In all instances, the running time is shorter than the accident time. Thus, the computing efficiency does not represent a constraint to the analysis. The results can typically be produced faster than can be absorbed by the analyst, since more than 2000 system variables are available for examination. Some system variables may consists of many values. For example, 2106 channel temperatures are available for 18 characteristic channels. Since the processes and phenomena evolved during the accident transients are complex and highly interrelated, it is essential to examine many of these variables simultaneously. Specialized plotting and process visualization programs are essential and have been developed to effectively analyze the vast amount of information generated.

The accident progression sequence was predicted to be identical on all machines, but some minor differences in the timing of events were noted between the DOS and UNIX based machines. These were well within the range of uncertainty of severe accident phenomena and the differences caused by the nodalization choices routinely covered by parametric analyses for each scenario. Nevertheless, efforts are under way to trace and eliminate these machine differences.

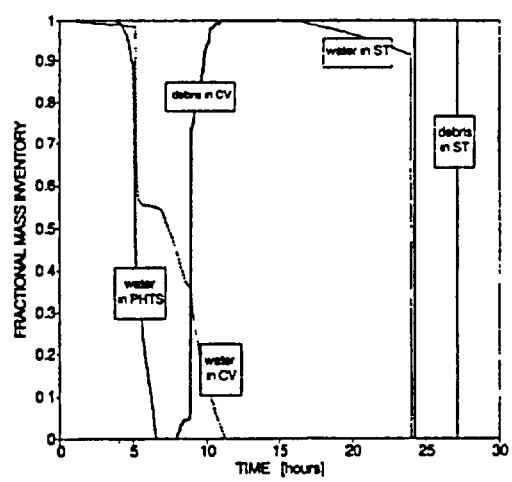

Figure 6 : Inventory Transients for Unit Blackout Scenario

The program requires a large amount of plant and equipment information (in excess of 1500 input parameters are needed to describe the plant). Furthermore, a number of model parameters such as equipment failure criteria had to be derived by separate analyses. The process of assembling and verifying this information has turned out to be as manpower consuming as the development of the models, particularly since high QA standards are set and maintained. To ensure the integrity of the results, strict configuration management controls are implemented to cover both the models and plant data.

8. CLOSING REMARKS

A state-of-the-art, fully integrated computer code is now available in Ontario Hydro for the analysis of severe accidents in the CANDU plants. The code combines the fundamental thermal-hydraulics, physics and chemistry from the literature with the up to date results of the international R&D programs on severe accidents. All known accident phenomena are represented, as are all the engineered systems and structures in the plant. Many phenomenology models have been validated and benchmarked for the source code and the integrity of these models has been strictly maintained. The remaining models are employed parametrically to explore the uncertainties.

A complete set of MAAP-CANDU input parameters and models for the Darlington NGS has been assembled and documented. Severe accident analyses are now in progress for this plant and the results will be published following a comprehensive expert review.

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