# OVERVIEW OF TUF CODE FOR CANDU REACTORS

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## ABSTRACT

An overview of the TUF code is presented. The important physical parameters used in the following code modules are briefly discussed: reactor control and safety systems, thermal-hydraulics, heat conduction, reactor physics and system components. The approach for code qualification as an analytical tool for CANDU reactors is described.

## 1. INTRODUCTION

The role of system analysis codes in reactor simulation has long been recognized in the nuclear industry (Reference 1). Reactor system codes are usually designed for reactor operating support and licensing analyses. Also, they can be used in the analysis of reactor process design. In the operational support analysis (for example trip parameter assessment and update of operation manuals), the precise interactions among the various control systems or system components are the most important function in the simulation. In the safety analysis, conservative assumptions are usually imposed in the models in order to insert the safety margins in the simulation. In the utility industry, certain requirements for a system analysis code have to be satisfied before it can be accepted as an analytical tool for the operational support analysis: (1) a consistent steady state initialization technique, (2) a control system model that can be implemented for the reactor plants, (3) ability to incorporate the data generated from the commissioning tests into the input data set, and (4) availability of special auxiliary or component models that are important in plant simulation. These requirements for the utility industry were generally ignored in most system analysis codes.

The main differences between reactor system analysis codes and reactor simulator codes are in their requirements. The reactor simulator code must provide a real time dynamic simulation of the plant systems which include plant logic simulation, control system, plant modelling, electrical system, protection system and data process system. To achieve this requirement, the depth of mathematical modelling has to be relatively homogeneous and coarse through the whole plant. Usually, a mixture of low order engineering models and table look-up techniques is applied. As a result, the simulator code is not constructed to accurately make predictions for disaster transients such as large loss of coolant accident (LOCA) transient. On the contrary, the system analysis code has no real time restriction. System analysis codes are expected to yield thermal-hydraulic parameters for system transients with engineering accuracy and at reasonable cost. Also, the plant electrical systems and the other auxiliary systems not related to thermal-hydraulics are not required in the system analysis code. Therefore, the plant evolutions simulated by a system analysis code are smaller than those provided by a simulator code. Nevertheless, both types of codes must be capable of simulating abnormal operating conditions resulting from malfunctions to demonstrate inherent plant response and functioning of automatic plant controls.

In the early years of safety analysis, conservative bounding analyses were used for plant licensing evaluations without impacting the economic operation of the plants. Recently, further conservation was added to the licensing models that significantly reduced the safety margins inherent in previous analyses. For example, the uncertainty in the reactivity increment due to coolant void in a large LOCA has been included in the safety analysis. As a result, the economic operation of the plant was impacted. To improve the realism of safety prediction methods and to qualify the actual safety margins, the advanced thermal-hydraulics codes were developed.

Due to the unique design features of the CANDU reactors and the intrinsic safety related features distinguishing it from other types of reactors (References 2 and 3), several system codes have been developed in Canada to provide analytical tools for the thermal-hydraulic analysis of CANDU reactors. AECL developed the FIREBIRD (Reference 4), RAMA (Reference 5) and CATHENA (Reference 6) codes for reactor safety analysis. In parallel with the developmental activities at AECL, Ontario Hydro Nuclear (OHN) developed the SOPHT (Reference 7) and TUF (Reference 8) codes for both operational support and safety analyses. A one-fluid model was used in the FIREBIRD, RAMA and SOPHT codes, while a two-fluid model was implemented in the RAMA, CATHENA and TUF codes. The separated flow approach for two-fluid model was used in the RAMA and CATHENA codes, which is different from the mixture flow approach used in the TUF code. Currently, only CATHENA (at AECL) and TUF (at OHN) are actively used in the safety analysis of CANDU reactors. The main features of the SOPHT and TUF codes that distinguish them from the system analysis codes at AECL are: (1) capability to simulate the normal operating conditions coupled with the reactor control systems of CANDU reactors, and (2) integrated reactor control systems specific for each station. These features enable analysts to perform both operational support and safety analyses for CANDU reactors in OHN.

The TUF code is made up of two separated programs: steady state and transient. In the steady state program, the equations dealing with thermal-hydraulic variables, nodal heat flux, heat exchanger film resistance and valve position (or special link resistance) for a control system are solved. The set of simultaneous non-linear equations is solved by the Newton-Raphson iteration method. To match the steady state solutions with normal operating conditions, different control flags are used in the input data. These flags are used to define the degrees of freedom for the steady state simulation, particularly when the control systems are involved. This program is used to calculate the normal operating conditions of a reactor at a specific operating power level. TUF contains modules dealing with thermal-hydraulics (one-fluid, drift-flux and two-fluid), reactor physics (point kinetics or external coupling with other reactor physics code), heat conduction (pipe wall, heat exchangers, pressure/calandria tubes and fuel pins), system components (pumps, valves, boilers, pressurizer, bleed condenser, turbine and accumulator), special models (discharge model, level swell, bundle movement, pipe strain model and metal-water reaction), and station controllers. The reactor control systems used in the code are station dependent. The reactor controllers simulate the following control systems: overall unit control, reactor regulating system, steam generator pressure and level controls, heat transport (HT) system pressure and inventory controls, bleed condenser pressure and level controls and safety systems.

Both the SOPHT and TUF codes are maintained and supported by the Reactor Safety and Operational Analysis Department (RSOAD). Table 1 compares the various models available in SOPHT and TUF. In the TUF code, the one-fluid model implemented in the SOPHT code is retained. This feature enables the TUF code to identify the differences between the one-fluid and the two-fluid models (Reference 9). All the developmental activities related to thermal-hydraulics at RSOAD have been concentrated on the TUF code. These developments are mainly generated from station requests for operational support and the code improvements. The current RSOAD activities associated with the TUF code are: (1) providing training and support to users, (2) assuring to comply with quality assurance (QA) procedure for code and input data changes, (3) conducting code validation, and (4) continuing code development and improvement.

The main purposes of this paper are to provide an overview of the modules in the TUF code and to describe the important physical parameters in the TUF code for the qualification of CANDU operational support and safety analyses.

## 2. FUNCTIONS OF REACTOR CONTROL AND SAFETY SYSTEMS

The reactor control and safety systems implemented in the SOPHT and TUF codes were specifically designed for CANDU reactors. Except some auxiliary systems which are not important in thermal-hydraulic simulations (for example the moderator and containment systems), most control and safety systems of CANDU reactors have been modelled. It should be noted that each nuclear power plant has its unique control characteristics: the controllers for Pickering [540MW(e)x8], Bruce [750 MW(e)x7] and Darlington [850 MW(e)x4] NGS in OHN are different from each other. For the Pickering and Bruce stations, minor differences also exist between the Stations A and B. Since the heat transport circuit and the control systems for Darlington NGS are similar to those for CANDU-6 [600 MW(e)] reactors, the functions of the control and safety systems for Darlington NGS are briefly described in this section for illustration. The details of the control and safety systems for each CANDU station can be found in the plant design manuals.

Similar to other pressurized water reactors, the CANDU reactor control system must satisfy two basic requirements: deliver steam at acceptable pressure and maintain a stable reactor power. All of the control systems, with the exception of HT pressure control, are implemented in a system of two dedicated, redundant digital computers to perform the following major control functions: reactor power control, plant load control, steam generator pressure control, steam generator level control, deaerator level control, moderator temperature control, control of miscellaneous systems, alarm annunciation and data logs. The three major areas which require constant monitoring and control under all operating conditions are: reactor, steam generator and turbine-generator. These areas are controlled in accordance with a pre-established overall plant control scheme. Therefore, the control program may be categorized into two groups: overall plant control and independent of overall plant control.

**Simplifications**

In the code, several simplifications have been made in the control systems: (1) In the overall unit control, the turbine run-up and monitor programs are not modelled. The turbine run-up program is used only to take the turbine-generator up to speed and to synchronize the generator to the grid. (2) The control of electrical output in the unit power regulator is not modelled since it is not important in the system analysis code. (3) The turbine load is calculated as a function of steam flow through the governor valve. (4) The deaerator is simulated as a boundary control volume; the level control of deaerator is not simulated. (5) The calibration of the in-core flux detector and ion chamber signals due to thermal power measurement is not modelled. (6) The light water zone controllers (14 in Darlington NGS) are simulated as one group with an average level. The flux tilt control is not simulated since the point kinetics model can not simulate the spatial flux distribution. (7) In the stepback routine, signals from the two central pairs of flux detectors are used to determine whether the reactor power reaches the suitable low level or not in the station controller. In the code, the average neutron power calculated from the point kinetics model is used instead. (8) Some general purpose control programs, for example the HT pump speed control and moderator temperature control, are not simulated. The HT pumps are assumed to be operating at constant speed at normal conditions.

None of the above simplifications significantly affect the accuracy of the control systems implemented in the code.

**Reactor Control Systems**

**Overall Unit Control:** The function of this system is to match the reactor power and the turbine load while maintaining steam generator drum pressure at its setpoint value. It is executed by three control programs: the unit power regulator, the steam generator pressure control and the demand power routine of the reactor regulating system. There are two distinct modes of plant control: normal (auto) and alternative (manual) modes.

The unit power regulator either changes the turbine load as demanded or controls the turbine governor valve to follow the load. The demand power routine computes the reactor flux power setpoint and the flux rate setpoint based on the reactor power setpoints obtained from (1) steam generator pressure control program when the reactor control is in normal mode, (2) operator keyboard when the reactor is operated in alternative mode, and (3) the setback routine. The program decides which of these three demands to be used and ramps the reactor power setpoint up or down to meet the demand. The calculation is done in two stages using the slow and fast programs, which are run every 2 and 0.5 seconds, respectively. The slow program determines the mode of operation and demand power increment which are used by the fast program. The fast program computes the demand power, demand rate and effective power error for the reactor regulating system.

**Reactor Regulating System:** This system is an integrated system which directly controls the reactor power. It comprises the reactivity control mechanisms, the reactor power measurement, the demand power routine, and the reactor power stepback and setback programs. The stepback routine monitors the plant parameters (reactor trip, turbine trip, HT pump trip, high HT pressure, high zone flux and rate log) and takes action to reduce the reactor power by dropping the four mechanical control absorbers if any one of parameters is met. The program may stop the rod dropping when stepback condition clears or the power reaches the pre-set power level. The setback mode is automatically initiated when any of the setback parameters exceeds its setpoints. The demand power is ramped down at a suitable rate until either the setback condition clears or the endpoint power is reached. The normal operating mode will be switched to the alternate mode by the setback routine when the endpoint power is reached. The power error between the actual reactor power (calculated from the point kinetics model) and the reactor power setpoint is used to control the following reactivity devices: light water zone control absorbers, adjusters, mechanical control absorber rods and the withdrawal of shut-off rods. The main physical parameters in this control system are the reactor power and the reactor thermal power. The power manoeuvring, pump trip or loss of coolant accident can be used to validate the function of this system. The stepback routine can be checked by simulating a turbine trip.

**Steam Generator Pressure Control System:** This system manipulates either the reactor power setpoint (normal mode of operation) or the turbine-governor reference setpoint (alternate mode of operation) to maintain the steam generator pressure at its setpoint. It also controls the opening of the atmosphere steam discharge and condenser steam dump valves to trim the steam generator pressure. The calculation of steam generator pressure setpoint depends on the operating mode of steam generator pressure control: warm-up, cool-down and hold mode. The pseudo poison prevent and poison prevent modes can be simulated by assuming a turbine trip with the stepback credited. When the unit control is in the normal mode, the steam generator pressure control program will calculate the required reactor power setpoint and send it to the demand power routine. When the unit control is in the alternate mode, the steam generator pressure control program will control the turbine load setpoint in response to the steam generator pressure error and the mismatch between the reactor power and turbine power. The turbine load setpoint demand rate is then processed by the turbine-generator electronic governor control system which manipulates the turbine governor valves. The speed of the turbine is a measure of the load requirement. The variation of turbine speed is used to vary the governor valve to control the steam flow to the turbine. The speeder setpoint signal is derived from a potentiometer driven by two constant speed motor. Correction action of boiler pressure error is initiated by speeder setpoint under the control of a pulse operated motor. The width of the pulses is proportional to the magnitude of the pressure error and pressure error rate. The steam generator pressures are measured at each main steam line. The control program reads all steam generator pressures and check the rationality and validity of the input before processing them to obtain the final measured steam generator pressure. The pressure error between the steam generator pressure setpoint and the measured steam generator pressure is the main control parameter. The control function can be tested with the cases of reactor warm up or cool down operation and manual reactor trip.

**Steam Generator Level Control System:** This system is designed to control the levels in all steam generators, by modulating the level control valves in the valve stations located in the feedwater lines. Three valves are arranged in parallel in each valve station: one small valve (20% capacity) and two large valves (100% capacity). Only one large level control valve is normally kept in service at any one time, the other large valve being on standby in case of failure. The control program runs in parallel in both digital control computers on a one second cycle time. The level in each steam generator is controlled individually using the same algorithm. The control algorithm consists of a single element, a three element and a default single element mode, depending on the valve lift required in the level control valves and the validation of the feedwater and steam flows. The main control parameters are the steam generator level, feedwater and steam flows, and reactor power. The simulation of manual reactor trip and steam line break can be used to test the control function.

**Heat Transport Pressure and Inventory Control System:** This system comprises a pressurizer (Bruce and Darlington NGS), bleed condenser, bleed cooler, two feed pumps, pressurizer steam bleed valves, bleed condenser level control valves, reflux control valves, spray cooling valve, HT feed and bleed valves, D2O storage tank, HT liquid (or pressure) relief valves, isolation valves, bleed condenser over-pressure protection system and associated piping. The heat transport pressure control is implemented via analog control. The HT pressure controller maintains the ROH pressure at its setpoint by modulating the pressurizer steam bleed valves and the pressurizer heaters. When the pressurizer is isolated from the HT circuit, the main circuit pressure is controlled by the wide range pressure controller via the feed and bleed valves. Under normal operating conditions, the HT inventory is controlled by the pressurizer level controller. When the pressurizer is isolated from the HT system under solid mode, the inventory control is done by the HT pressure controller. The bleed condenser pressure control is done by the operation of the reflux feed valve and spray valve controls.The main control parameters are the reactor outlet header pressure (highest of all reactor outlet header pressures), the pressurizer and bleed condenser pressures and the pressurizer level. The event of Class IV power failure is the best example to test this control function. The liquid in the bleed condenser is controlled to a level below the reflux tubes, but above the outlet nozzle to maintain effective reflux cooling and to prevent steam escape to the bleed cooler. The bleed condenser level control valves are controlled by the level controller and bleed cooler temperature controller. The main control parameters for the flow out to the purification system are the bleed condenser level and the fluid temperature at the bleed cooler. Simulation of the purification system against the plant upset operations such as loss of Class IV power or total loss of feedwater is the best example to test the function of bleed condenser level control.

The HT liquid relief valves are used for heat transport system overpressure protection device to relief coolant to bleed condenser. They are quick opening valves operated by ON/OFF controllers. All relief valves are opened when the highest pressure in all four ROHs exceeds the opening pressure setpoint and all valves are closed when the pressure falls below the setpoint. Overpressure protection of the bleed condenser is provided by two spring loaded shell side relief valves, connected to a heavy water recovery sump. One spring loaded relief valve provides overpressure protection of the tube side and associated piping.

**Safety Systems**

The safety systems of CANDU reactors are totally independent of the control systems. They are triggered automatically when certain system parameters exceed pre-determined setpoint levels. The primary objective of safety analysis is to determine whether the regulatory dose limits are exceeded or not under an accident. The initiation of the safety systems are important for CANDU reactors: the shut-down systems ensure that fuel sheath and pressure tube integrity are maintained prior to the initiation of the emergency cooling injection (ECI) system; ECI system ensures that adequate cooling is maintained; steam generator controlled cool-down provides an additional heat sink for the secondary side system; and emergency cooling system for steam generators provides emergency cooling for the secondary side.

In an emergency situation, two independent shut-down systems are available to rapidly reduce reactor power to the shut-down level. These systems cause the power reduction by dropping the shut-off rods into the core (SDS1), or by injecting a neutron absorbing solution (gadolinium) into the moderator (SDS2). The main parameters for the shut-down systems are the reactor outlet header pressure, inlet feeder flow, neutron power and log rate, pressurizer level and steam generator water level. The shut-down systems are automatically initiated when any of the parameters exceeds its pre-set value in the plant operation. In the safety analysis, only the back-up trip signal is credited for conservative reason. The simulations of manual reactor trip, loss of Class IV power and large LOCA can be used to test the actions of shut-off rods or poison injection.

In the primary side circuit, the program for the initiation of the ECI system checks the initiation conditions by comparing the initiation time, the header pressures and the inlet feeder flows. The ECI valves start to open when all conditions are met. The initiation time is used to simulate the LOCA conditioning signal time. The ECI initiation signal on header pressure is assumed to be registered if maximum ROH pressure is below the setpoint. The operation logic for the high pressure by-pass line valve and the recirculation line valves are modelled. The valves are opened if the initiation pressure is reached. The low pressure and high pressure pumps are assumed in full operation at the start of the transient.

In the secondary side circuit, the injection valves for the steam generator emergency cooling system start to open when the initiation conditions regarding initiation time, steam generator pressure and steam generator level are met. The initiation time is used to simulate the manual start-up by operator action. The steam rejection valves of the steam generators start to open when the initiation conditions of time, header pressure and pressurizer level are met. For Pickering NGS, the turbine steam flow signal is used in the steam generator rejection valves control algorithm, where the fraction of total steam rejection valve opening is the sum of feed back and feed forward signals.

**Discussion**

In the analog controllers, the transformation of the calculated fluid state into the measurements (e.g. steam drum pressure, steam flow, feedwater flow and inlet header temperature) uses the first-order delay equation by applying the time constants to the controller calculations:

$$\frac{dx_m}{dt} = \frac{[x - x_m]}{T} \qquad [1]$$

where $x_m$ is the measured signal of the quantity used by the controller, x is the actual value of the measured quantity and T is the time constant or relaxation time of the instrumentation.

Commissioning is a vital part of reactor operation. The initial operation of a new nuclear plant must be carried out in carefully planned stages and each component tested separately to ensure maximum reliability and safety. The aim of a commissioning program is to obtain accurate information about the reactor core and its components prior to carrying out any full-scale operation. The test results provide the important information about the characteristics of system components and control systems for the system analysis codes. For example, the control rods characteristics (position indicators, withdrawal times and drop times), the characteristics of control and safety valves, and the swelling curves of the pressurizer water level from the commissioning tests are usually implemented in the input data set of plant simulations.

Although the control routines in the code differ from one station to another, the code has been designed so that the control elements and the locations at which the input variables are measured are easily identified by means of location codes assigned on the input data for all CANDU reactors. Also, the malfunctions of components and controllers and the operator action can be simulated through the input data.

## 3. HEAT TRANSPORT AND TRANSFER SYSTEMS

**Thermal-hydraulics**

The modelling of two-phase flow is the most difficult part of reactor simulation. It constitutes the most profound differentiation between different codes. The two-fluid model implemented in the TUF code is discussed in Reference 9. In the CANDU reactors, two-phase flows occur only in certain areas of reactor piping and components during normal operating conditions. It has been illustrated by many one-fluid codes that operational transients near normal operating conditions can be predicted with engineering accuracy. Also, it has been reported in the literature that in some reactor simulations, the one-fluid or drift-flux model produced better transient tendencies than the two-fluid model. It indicates that without an appropriate modelling, increased thermal-hydraulic modelling complexity may not improve the reactor prediction. For the case with a LOCA, the entire spectrum from sub-cooled water to superheated steam is possible. Two-fluid effects becomes important either when the phase slip is compatible with the mixture velocity or after the activation of the ECI system. Detailed descriptions on the differences between the one-fluid and two-fluid models are presented in Reference 9.

**Operational Support Analysis:** Reactor operation is governed by the limitations and restrictions which may be placed on certain elements of the plant. In general, limitations will usually apply to fuel element temperature and rate of change of temperature, temperature differentials in certain parts of the plant, and coolant and steam pressures. The restrictions ensure that there is an adequate safety margin during operation so that as a result of the maximum credible accident there can be no possibility of a core melt-out or large release of activity into the surrounding area.

The important thermal-hydraulic parameters in this analysis are the correlations used for the wall frictional loss and the wall heat transfers in the heat exchangers and reactor channels. The flow induced pressure drop consists of wall skin friction and form (or minor) loss due to geometry. The Darcy friction coefficient depends on the phase velocities. For a turbulent flow, three correlations are available: the smooth pipe correlation, the Colebrook correlation and the analytical form of Colebrook correlation. There are several two-phase multiplier correlations available in the code. The selection of the correlation depends on the component type and the pipe diameter. Different correlations for the wall momentum and heat transfers are available in the code. Those correlations are well established for the liquid phase and the two-phase with a low vapour quality. The published validations of these correlations against experiments have increased confidence in the analysis and it lends support to the use of them for conditions inside the range for which they were determined.

Some aging mechanisms may create a weakened condition in a safety related components and affect the plant performance. The main areas that relate to thermal-hydraulics for aging reactors (for example Pickering and Bruce NGS) are the piping corrosion, pressure tube sagging in reactor channels and the steam generator tube degradation. The related physical parameters are: piping roughness due to corrosion, changes in flow length (pressure tube elongation), flow area change (tube blockage), and the fouling heat transfer coefficient due to coolant deposits on heat-transfer surfaces. In the code, the fouling resistance is incorporated into the analysis by a series of resistances. The fouling resistance for the inner (tube side) and outer (shell side) surfaces are expressed by a single value with a normalization constant which is used to modify the overall heat transfer coefficient to match designed heat exchanger performance data. It is apparent that the fouling resistance will usually increase with time until cleaning is necessary. However, in general the resistance ceases to increase after some operational time because the rate of deposition is balanced by the rate of removal by scouring.

**Safety Analysis:** Various phenomena governing the events of small and large LOCAs for light water reactors have been discussed in Reference 10. Since the stagnated critical break represents the most severe accident for a CANDU reactor, considerable design efforts and analysis have been devoted in Canada to ensure that the CANDU reactor design minimizes the consequences and reduce the risk to acceptable levels during a large LOCA.

In a large LOCA, there is a continuous transient consisting of three distinct phases: blowdown, feeder refill and channel refill. Each phase is governed by different dominant phenomena. In the blowdown phase, the important parameters are the peak in the sheath temperature transient and its timing. These parameters depend on the initial stored energy in the fuel, fuel thermal properties, gap heat transfer coefficient, sheath-to-coolant heat transfer coefficient, core power, and radial and axial peaking factors. The feeder refill phase begins when the ECI water starts to refill the feeders and ends when the water reaches the channels. In this phase, the sheath continues to heat up or to maintain a high temperature. The important physical phenomena are the timing of feeder refill, steam condensation, counter-current flow in the feeders, the entrainment and the possible flooding. The additional heat resulting from metal-water reaction and the pump is relatively insignificant due to adequate steam cooling when the ECI water is available. In the case of loss of ECI, the heat generated by the metal-water reactor may play a role in the channel heat up. The channel refill phase is the final stage of a large LOCA in which final quench in the channels is reached. This results in rapid cool-down of the core and a large axial temperature gradient in the sheath across the quench front. The physical phenomena that govern the channel refill phase are the steam condensation, the heat transfer mode in the core, and the flow regime transitions.

**Heat Conduction and Heat Transfer**

The heat conduction equations for fuel pins, pressure tube and calandria tube, heat exchangers and wall piping have been modelled in the code. Solution techniques on these heat conduction equations are well established. Finite difference and lumped parameter methods have been applied in the code to solve heat conduction equations. Emphasis is placed on the effect of flow regimes. The wall surface is separated into two distinct regions: vapour and liquid regions according to the vapour void fraction. In the stratified flow, different heat transfer coefficients are applied in each individual region. This approach provides the simplest logic to take into account the effect of flow structures. For the fuel pins in a channel, the power rating of each individual pin and the flow structure are the key parameters for the fuel pin temperature profile. In the large LOCA analysis, the important physical parameters are the gap conductance, the metal-water reaction and the contact between pressure tube and calandria tube. In the code, the gap conductance of the fuel pin comprises three components: conduction, convective and radiation. The deformation of the fuel and sheath due to thermal expansion and the pressure difference between gap pressure and coolant pressure is not considered in the current TUF versions. The contact criterion for the pressure and calandria tubes is calculated from the strain model developed by AECL Research at Chalk River.

A review of heat transfer correlations used in reactor safety studies has been presented by Groeneveld and Snoek (Reference 11). It has concluded that the most important heat transfer regimes (CHF, transition boiling and film boiling) still have a limited data base. In the code, different convective heat transfer correlations for sub-cooled boiling, nuclear boiling, transition boiling, film boiling, superheated steam convective and condensation regions are available. Several combinations of correlations in different heat transfer modes have been set up as options. Different correlations for critical heat flux (CHF) are used for Pickering (28 pins) and Darlington or Bruce (37 pins). The CHF correlations plays a role in the assessment of the maximum operating power since, in the event of an accident, the reactor must be shut down prior to the onset of dry-out on any fuel element. The CHF correlations were developed by AECL Research at Chalk River based on their experimental data. In the operational support analysis, the important physical parameters are the convective heat transfer correlations and the fouling heat transfer coefficient for steam generators. The validity of these correlations can be checked by comparing the code prediction with the plant data on the temperature difference between the inlet and outlet of the U-tubes. For the heat exchangers, the well-known film condensation heat transfer correlation is applied. In a large LOCA analysis, the film or post dry-out heat transfer coefficient plays a key role in the sheath temperature prediction. Also, the time period of flow stratification in the reactor channels significantly affects the sheath temperature transients of top and bottom pins.

The heat losses from the heat transport loops to the moderator and the outside of feeders have been considered in the code. These heat losses are not included in the safety analysis for conservative reasons. However, they may have to be included in the operational support analysis. The heat loss from the secondary side piping is usually small (about 1 MW) and it can be ignored in the simulation.

**Discussion**

Simulations of heat transport and heat transfer systems are the central parts for system analysis codes. Certain areas require special attention in the code development: water packing and empty node treatments, symmetric behaviour of solutions, numerical instability, uncertainty in two-fluid parameters, and wall heat transfer correlations.

Pressure spikes that cannot be traced to any physical origin sometimes are observed in the code simulations. These spikes usually result from numerical water packing or from interactions between spatial discretization and heat transfer. Also, a node with empty mass may occur in the simulation when a flow reversal happens and the time step size is too large. These pressure spikes or empty nodes generally do not affect the overall transient behaviour and, thus, are ignored. Nevertheless, these numerical artifacts must be removed by code or input data changes (for example the time step control parameters). In TUF, special treatments on these numerical artifacts have been adopted. The physical pressure spike either resulted from the interaction of two compression waves (for example produced by valve chattering) or induced by water hammer (for example due to vapour collapse at pipe dead-end) is one of main concerns in plant operations. While it is required for a stable solution, the numerical diffusion inherent in the finite difference scheme will affect the magnitude of predicted pressure spike.

Aside from possible mistakes in the input data, the round-off and truncation errors also exist in the applications of reactor system codes. Round-off errors stem from a finite number of digits in a computer word, while truncation errors are due mainly to finite approximations of limiting processes. When a decimal number which contains a fractional part is converted to its binary equivalent, a conversion error due to the finite word length of the computer may be introduced. The study of round-off errors and the control of its propagation are important in high-speed digital computations. The symmetric behaviour of the flow matrix equations will be destroyed if the propagation of round-off error is not controlled. A technique of using different precision levels in the calculation of the flow matrix equations has been implemented in the TUF code. As a result, the propagation of round-off errors during each time step has been eliminated.

A two-fluid code has two sets of governing equations that represent the conservation of mass, energy and momentum. The main uncertainty is in the interphase transport processes. Probably the key parameter in understanding these processes is the interfacial area. The interfacial area depends on the flow regime maps and the amount of entrained bubbles and droplets in each flow regime. The flow regime maps, which are usually based on void fraction and flow rate, are normally constructed from the steady state experimental data. The length or transient effect shown in experiments cannot be represented by any of the commonly used flow regime map.

While the two-fluid model is generally considered to be a state of the art method for modelling transient two-phase flows, it suffers from the fact that its time step size used is generally smaller than that used in the one-fluid model with a same numerical scheme. To reduce the computing cost, several stability enhancing methods, ranging from a simple two-step method to a fully implicit method, for two-fluid model have been suggested in the literature.

The uncertainty in the correlations of critical heat flux, transition boiling and film boiling is still large. As suggested by Groeneveld and Snoek (Reference 11), a combination of empirical correlations with appropriate transition regimes and extrapolating factors can improve the accuracy of the heat transfer correlations.

## 4. REACTOR CORE POWER AND PLANT COMPONENTS

### Reactor Physics

The formulation of neutron kinetics models for system codes is far less controversial than that of two-phase flow and component models. In the TUF code, two approaches are available for the reactor power calculation: using the point kinetics model or using channel and bundle powers data obtained from the reactor physics codes (for example SMOKIN code (Reference 12)). The first approach is normally used in the plant operating support and small LOCA analyses since the total reactor power can be accurately predicted by the point kinetics model. The second approach is employed in large LOCA cases of safety analysis because the conservative analysis for the coolant void reactivity (add 2.4 mk for uncertainty allowance) and the effect of flux tilt (limiting tilt of 20%) in the channels are required in the large LOCA analysis.

In the point kinetics model, six-delayed neutron groups are considered. The power delivered by the fuel due to decaying fission products is modelled by three decay heating groups. The Runge-Kutta integration technique is utilized for the point kinetics equations. The total power released in the fuel is the sum of the fission power and the power due to decaying fission products. In the point kinetics model, the reactivity change dk consists of the following components:

$$dk = dk(TF) + dk(TC) + dk(TM) + dk(F) + dk(C) + dk(B) \qquad [2]$$

where dk(TF) is the reactivity change due to fuel temperature, dk(TC) due to coolant temperature and density, dk(TM) due to moderator temperature, dk(F) due to refuel under reactor operating condition, dk(B) the reactivity change due to bundle movement resulting from a flow reversal in a large LOCA, and dk(C) due to control mechanisms from the reactor regulating and shut-down systems which is the sum of the following components: mechanical control rods, liquid zone absorbers, adjusters (or booster rods in Bruce NGS), and shut-down systems 1 and 2. The adjuster control is not normally used as short term power regulating. Its contribution on the reactivity change during the transient can be neglected. However, it can be used in the code to maintain a zero reactivity change during the normal operating conditions.

The four mechanical control absorber rods are used as auxiliary reactivity control devices of the reactor regulating system. These rods (separated into two banks) are normally withdrawn from the core. These rods can be driven gradually in or out of the core to provide a reactivity shim. Also, during a stepback, they are dropped under gravity to provide a sudden large decrease in power. The driving logic is determined by the reactor power error and the average zone power level. The reactivity contribution by each bank is calculated from the rod positions.

The light water zone controllers are spatially distributed in each power zone. Since the flux tilt control can not be included in the point kinetics model, the zone controllers are simulated as one group with an average level. The water level in the zone controllers is regulated by a valve which controls the inflow rate. The actual valve position is calculated from the following valve sizing equation:

$$\frac{d^2Z}{dt^2} + 2Dw\frac{dZ}{dt} + w^2 Z = w^2 Z_D \qquad [3]$$

where Z is the valve position, D is the damping coefficient, w is the undamped natural frequency and $Z_D$ is the demand position of the inlet valve which is a function of reactor power error. The relationship between the reactivity rate and the valve lift is given by the station data.

The reactivity increment due to coolant temperature and density changes in the channels will result in a large power pulse build-up in a large LOCA analysis. The power pulse build-up may be as high as 3 to 5 full power for a typical stagnated critical break in CANDU reactors. In the code, the fuel temperature and coolant density used to determine the reactivity changes are obtained by performing an axial neutron flux squared weighting within each zone and then performing a radial zone volume weighting. Using the average fuel temperature and average channel coolant density, the system reactivities are obtained by linearly interpolating in tables that are obtained from the physics codes. The reactivity changes due to fuel loading and moderator temperature change can be simulated through a time function.

For the shut-down system 1, the spring assistant gravitation fall of the shut-off rods (32 rods in Darlington NGS) are simulated as follows: The rod position as a function of time is calculated using a table look-up technique. The normalized reactivity insertion of the rods as a function of position is also calculated from a table. These tables are usually obtained from the design data or commissioning test data. In the safety analysis, three most effective rods are assumed unavilable. For the shut-down system 2, the normalized reactivity insertion by the injection of neutron absorbing solution into moderator (or moderator dump for Pickering A NGS) as a function of time after action starts is also calculated from a table look-up. In the safety analysis, the poison injection valves are assumed to open with a maximum allowable delay time after the trip signal is received.

Since the point kinetics model does not yield the spatial distribution of the neutron flux, the normalized axial flux distribution is simulated either by the cosine curve or by the data obtained from the reactor physics code. Axial and radial peaking factors, which describe the power density distribution in the core and also define the hot spots, are also input from the predictions of reactor physics codes. In the safety analysis, the peaking axial shape is assumed to yield licensing limit bundle power in hot channel. The following assumptions on the description of flux distribution are made: (1) during an upset condition the power distribution is the same as that under normal steady state operating conditions, and (2) the effect of dropping control absorbers or shut-off rods on the power distribution can be expressed as a function of the reactor power.

**System Components**

The following system components have been modelled in the TUF code: pumps, valves, pressurizer, bleed condenser, steam generators, turbine and accumulator. The engineering models are utilized in these components.

The pump model simulates a centrifugal pump under normal operating conditions (constant speed), run-down state or restart state. The solution of the pump dynamics equation for a run-down or restart pump is carried out simultaneously with the flow rate equations. The brake state of a run-down pump is also included in the model. Pumps can be restarted from either the brake or idle state. The pump characteristic is normally supplied by the pump manufacturer only for the first quadrant operation at design speed. For a run-down or restart pump, data for the additional operating quadrants are required. The pump characteristics for the reactor plants at OHN were developed from the pump tests performed at Ontario Hydro Technologies (OHT) and plant data (Reference 13). The total pump torque for a run-down pump is calculated by considering the hydraulic torque from the homologous pump curves and the pump frictional torque. The frictional torque consists of two components: dynamic friction and static friction torques. Pump performance under a two-phase condition is modelled using a head degradation parameter which is a function of void fraction.

Valves are a major component in power plants. Typically, valves represent about 10% of the plant maintenance budget. The types of valves used in CANDU reactors include check valve, gate valves, global valves, ball valves, butterfly valves, pressure relief valves and safety valves. In the code, control valves are modelled by the following six types: linear liquid valve, polynomial liquid valve, tabular liquid valve, steam valve with valve sizing coefficients, steam valve with tested valve characteristics and butterfly valve. For the liquid control valve, a test on the critical condition was made in the code. For a valve with a non-critical flow, the valve resistance is calculated from the valve coefficient and the valve position. The pressure relief and safety valves are generally designed for steam discharge conditions. Normally the flow through the valve is expected to be choked and the flow rate will be limited to a maximum or critical value. Under certain transients, these valves may be exposed to two-phase or liquid upstream conditions. The determination of the actual mass flow rate through a safety valve is prescribed by the ASME Boiler and Pressure Vessel Code. The following discharge mass flow rate for safety valves is usually applied:

$$W = 0.9\ C_v\ W(t) \qquad [4]$$

where the factor 0.9 is the ASME recommend multiplier, $C_v$ is the valve discharge coefficient and W(t) is the theoretical mass flow rate calculated.

The pressurizer is modelled as a single control volume with a distinct phase separation. The thermal non-equilibrium model is applied to the pressurizer. The interfacial heat transfer rate is dependent on the spray flow rate, entrained bubbles and droplets, and the operating condition of heaters. The void fraction of entrained bubbles in the liquid region is calculated from the level swell model. The model applied to the pressurizer has been verified against the NPD pressurizer test (Reference 8) and the Marviken top blowdown test (Reference 14). The total energy of the heaters (four ON/OFF and two variable heaters), which is located at the lower region of the pressurizer, is calculated from a first-order delay equation, Equation [1]. Similar to the pressurizer, the thermal non-equilibrium model is also applied to the bleed condenser. The condensate heat transfer coefficient was applied to the wall heat transfer mode of bleed condenser.

The boiler model comprises of heat transport U-tube side, steam-water shell side, riser, steam drum and downcomer. For the boiler inlet flow, which is in two-phase condition under a normal operating condition, the condensation heat transfer correlation based on the modification of the Dittus-Boelter correlation is applied. The steam-water shell side consists of preheater and boiling sections. In the preheater section, the Dittus-Boelter or other subcooled boiling heat transfer correlations (for example the Thom and Walker correlation) is used. In the boiling section, either the

Rohsenow, the Chen or other incipient boiling correlations can be used, depending on the user selection. The riser represents the two-phase flow portion of boiler above the top of the boiler tube bundles. The flow is close to homogeneous regime in this area under normal operating conditions. In the plant simulation, the extended steam drum model which combines with the downcomer is usually used in the input data. Two kinds of water levels in steam drums are calculated: collapsed and swelling levels. The collapsed water level is used in the steam generator level control. The swelling level is used in determining the separation capability of steam separators. The relationship between the steam drum water level and the drum volume is supplied by the user.

The steam produced from all the steam generators is combined in a common steam header before going to the turbine. The steam passes through the isolating valve which is normally open during operation and then through the emergency stop valve and governor steam valve which control the quantity of steam entering the turbine unit. The turbine unit contains one high pressure turbine, a moisture separator, a reheater and low pressure turbines (normally three in series for CANDU reactors). The steam passes through the high pressure turbine where the pressure and temperature of steam is reduced as work is done on the turbine rotor. The steam leaves the high pressure turbine and the moisture is removed in a moisture separation. The steam is then heated in a reheater which uses main steam at steam generators to heat the steam leaving the moisture separator. Before entering the low pressure turbines, the steam passes through the intercept valve which can shut off steam to the low pressure turbines under certain conditions. This intercept valve works in conjunction with a steam releases valve. After the steam passes through the normally open intercept valve, it enters the low pressure turbines, and then exhausted to the condenser. In the code, the turbine unit is modelled as a single steam control volume with the governor steam control valve. The piping associated with the turbine unit (for example the reheater drain) relating to mass and energy conservation laws of the secondary side system are included in the plant simulation. The relationship between the turbine load and the pressure differential across the turbine (or steam chest pressure) is represented by a linear function. The steam chest pressure is, therefore, calculated as a linear function of the steam flow to turbine. Steam chest pressure and reheater drain flow are treated as varying boundary conditions in the plant simulation. The reheater drain flow can be expressed as a function of governor valve steam flow. For a steady state with a reactor power less than 100% FP, the governor valve steam flow is a function of the reactor power and the reference turbine steam flow at 100% FP. Therefore, the reheater drain flow can be obtained by using these relationships.

The emergency coolant system, which is inactive but poised during normal operation, is activated automatically following the detection of a LOCA signal. The response of the ECI system can be divided into short and long term phases. For the Bruce station, the short term phase consists of high pressure injection from the accumulator water tank. On a LOCA signal, the accumulator gas tank isolation (butterfly) valves will open with a specified delay time. The modelling of the Bruce ECI accumulator and its validation against the plant data were presented in Reference 15.

**Discussion**

The actual maximum operating reactor power is 100% FP while it is 103% FP in the safety analysis for conservative pratice to include the uncertainty of measurements. The accuracy of the reactor core power and its distribution along the channels or bundles depends on the tables obtained from the reactor physics code for flux distribution, zone power distribution, peaking factors, fuel temperature and coolant void reactivities, and the characteristics of control and shut-down devices. To reduce the cost of externally iterative coupling between TUF and SMOKIN codes, a plan to combine SMOKIN and TUF is being considered at RSOAD. After the reactor is in a shut down state in most safety analysis, the reactor power transient is well predicted by the point kinetics model. The importance of an accurate reactor physics prediction becomes less in comparing with the modelling of thermal-hydraulics.

In the operational support analysis, the characteristics of valves and pressurizer play the most important role in the simulations. For example, there are three distinct pressure regions in the primary heat transport system of CANDU reactors under normal operating conditions: the high pressure in HT loops and pressurizer (about 10 MPa), the medium pressure region in bleed condenser (about 1.5 MPa) and the low pressure region of D2O purification system after the ion exchanger columns (about 0.1 - 0.5 MPa). Accurate descriptions of interactions among various pressure regions require appropriate control valve characteristics. Also, the pipe length of a safety relief valve is an important factor in the analysis. Safety valves employed in over-pressure protection systems are generally designed to open and close fast (within 50 ms). As the valve opens, pressure at upstream of the valve drops due to sudden expansion wave. As the expansion wave reaches the pipe end connected to the vessel (for example the bleed condenser), the wave gets reflected as a compression wave traveling back towards the valve. The frictional pressure drops caused by long pipe length may influence the safety valve performance by affecting the valve upstream pressure. The resulting acoustic wave may enhance the frictional effects leading to decreased stability of the valve during the opening and closing. If the combined acoustic and frictional pressure drop in the upstream pipe caused by the valve opening is large enough to prevent a full opening of the valve stem, it may start to close the valve and cause it to chatter. The valve chattering and the interaction between two compression waves are important in the process design analysis.

Similar to other pressurized water reactors, the pressurizer in CANDU reactors (except Pickering NGS) is a particularly important component. Several theoretical modellings of pressurizer have been presented in the literature. Most of the models require at least two control volumes in the representation of pressurizer. In the TUF code, the pressurizer is simulated as one control volume. A level swell model is applied to determine the amount of entrained bubbles in the liquid region. A series of tests on the pressurizer behaviour has been conducted at the Nuclear Power Demonstration in OHN in 1985. It has been found that the transient behaviour of two phase temperatures are different during the in-surge and out-surge processes.

In the event of Class IV power failure, the most important physical parameter is the run-down characteristics of the HT pumps, especially for the timing of low flow trip signal.

## 5. APPROACH FOR CODE QUALIFICATION

A mature system analysis code follows these activity cycles: user feedback, code validation and continuous code improvements. The following approach for the qualification of TUF code is currently being adopted: (1) cross code comparisons, (2) comparisons with available experimental data for separated and integrated tests, and (3) comparisons with plant data for normal and upset operations.

### Cross Code Comparisons

To check the numerical method implemented in the code, standard problems have been simulated (Reference 16) and compared with the so-called exact solutions obtained from the MECA code (Reference 17). This comparison provides the information about the degree of numerical diffusion that was inherent in the numerical method.

The fuel channel codes, SMARTT (Reference 18) and FACTAR (Reference 19) have been used in the safety analysis at OHN. It is logical to have a direct code-to-code comparison for the fuel channel calculations. Comparisons with the SMARTT code have been presented in Reference 20 for large LOCA simulations. Comparisons with the FACTAR code have been performed internally at RSOAD.

Since the SOPHT code has been used in the CANDU process design analysis at AECL Engineering (Reference 21) and safety analysis at OHN, the comparisons between the SOPHT and TUF predictions for different plant simulation events have been emphasized during the code development stage. These comparisons not only provide the necessary information about the two-fluid effects on the system responses, but also suggest the possible areas for code improvements.

### Comparisons with Experimental Data

The first and simplest types of comparisons are the separated effect experiments. These are small-scale tests in which complexities are held to a minimum and the governing physical parameters are accurately measured. The user effect on the input data can be eliminated in the simulations. The results of these analyses increase confidence in the individual and combined models utilized. This type of comparison was extensively used in the code development stage.

The second class of experiments are system effects tests, in which the interactions of various components must be described. In this class of comparison, the user effect on the simulations can not be eliminated (Reference 22). The TUF validation against this type of experiments is much more demanding since these comparisons can identify the weakness of the code and suggest the possible areas for code improvements.

The Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency of the OECD is an international committee made up of scientists and engineers who are responsible for nuclear safety research and nuclear licensing. The use of International Standard Problems (Reference 23) for code validation has been agreed by most OECD countries. Recently, a validation matrix for CANDU safety codes has been set up (Reference 24)in OHN. Most of the cases in this validation matrix have concentrated on thermal-hydraulics. There are two areas that can not be addressed in these validation matrices: the scaling effects and the interactions among various control systems and thermal-hydraulics.

**Comparisons with Plant Data**

Operating reactors clearly provide the only full scale and integrated tests for system analysis codes. It has been recognized that the thermal-hydraulics codes will be of little value unless that can predict the behaviour of the operating reactors. Satisfactory predictions by a system analysis code on the plant transients require two conditions: the appropriate models applied in the code and the appropriate input data. Simulation of plant upset transients as well as maintenance of the reference input data sets have been emphasized in OHN. During the past decade, SOPHT has been used to simulate many abnormal plant operations including the small break event (bleed condenser safety line fracture) that occurred at Pickering Unit 2 in December 1994 (Reference 25). Recently, TUF has been used to simulate the upset plant operations, for example the simulation of the Class IV power failure event in 1991 at Darlington NGS (Reference 26).

The interactions among the various control systems or components play a key role in the operational support analysis. The best example is the accident that occurred at Pickering Unit 2 in 1994. A spurious trip of one of the four pressure relief valves forced the heat transport system pressure to drop. It resulted in a reactor setback due to high water level in the bleed condenser. The boiler pressure control ran down the turbine and the reactor eventually tripped due to low HT pressure. The operator started the second HT feed pump and tripped the turbine to help restore the HT pressure. This caused a rapid increase in primary side pressure. The pressure recovered so high that the other pressure relief valves opened. As a result, the pipe leading to Relief Valve 5 of the bleed condenser broke and it produced a small LOCA for the heat transport system. Later, the ECI system activated according to the plant design. Finally the small LOCA was terminated after the pressure relief valves were manually closed. This example illustrates that the interactions among various control systems must be properly described in order to have an accurate prediction of the event.

**QA Program for TUF Code**

An issue that was raised by the regulatory authority (AECB) was the fact that TUF was not developed under a QA program. To satisfy this requirement, a QA team for TUF was set up. The major objectives of this QA team are (1) provide adequate documentation, (2) review of the coding with respect to the code specification document, (3) record and report code errors, perform impact assessment and notify all users, (4) assure coding changes following the QA requirement, and (5) construct new versions of TUF (either for the correction of coding errors, user requests for minor model modifications or new models) following the QA requirement. In additional to the QA team, a TUF users' supporting group has been set up. The main functions of this group are to provide the technical support to analysts and to document possible deficiencies of the code performance for further improvements. For example, certain areas of code performance have been addressed by some analysts: tracing of pressure wave propagation for water hammer analysis, density wave oscillation in the steam generator level control, plant aging effects, steam separation capability in steam generators, capability of maintenance cooling system, valve chattering criterion, effects of non-condensible gas on heat transport system, pump restart capability, and others. The development status of the TUF code was reported to all analysts during the users group meetings. This direct communication between the analysts and the code developers has significantly improved the code quality and predictions.

**Developmental Activities and Version Control**

The developmental activities are basically driven by two considerations: removing the code deficiencies and implementing the new models. As mentioned before, the uncertainties in the modules of thermal-hydraulics and system components are large. Therefore, the current developmental activities have been focused on these two areas. Normally, the engineering model is preferred over the complex physical model in the model development. For example, to obtain the film condensation heat transfer coefficient for a mixture of steam and non-condensible gas, the laminar boundary layer equation and the mass diffusion equation are solved simultaneously with an iterative technique to obtain the film interfacial temperature. In order to reduce the computing cost and to remove the possibility of divergence problem in the iterative procedure, a simplified correlation for the film interfacial temperature is derived and utilized in the calculation. As a result, the iterative technique is no longer necessary.

It should be noted that the main objective of a system analysis code is to predict the overall system response during an abnormal plant operation. Emphasis is placed on the averaged system or component behaviour rather than on detailed local information since the one-dimensional thermal-hydraulic equations are applied to all piping network and components. There is no attempt to make TUF a sub-channel or fuel channel code since the detailed channel information can be obtained from the other fuel channel codes.

In the past years, two versions of TUF have been developed: LOCA and water hammer versions. Recently, a unified TUF version has been generated for all types of analyses. Also, a configuration management for TUF code has been in place at RSOAD.

## 6. CONCLUSION

The TUF code has matured and the code is being routinely applied to a wide range of plant operating support and safety analyses in OHN. Currently, the merge (data transformation and model unification) of SOPHT code into TUF is under way, and eventually the SOPHT code will be phased out at OHN. The continuous code improvements in the uncertainties of two-fluid modelling, the numerical scheme applied in the thermal-hydraulic equations, and the enhancements of system components will improve the code performance. The use of TUF to predict the various plant upset operations will continue to enlarge the code qualification base.

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Table 1. Comparisons of various models between SOPHT and TUF codes

| MODELS | SOPHT | TUF |
|---|---|---|
| Plant Controllers | station dependent | same as SOPHT |
| Thermal-hydraulics | one-fluid model | one-fluid model and two-fluid model |
| Non-condensible Gas | not available | available |
| Channel Model | one-pin model | one-pin model and multi-ring multi-pin model |
| Wall Transfer Models | homogeneous flow model | homogeneous and separated flow model |
| Metal-Water Reaction | not available | available |
| Heat Losses to Moderator or Outside Feeders | not available | available |
| Model for PT/CT Contact | not available | available |
| Reactor Physics | point kinetics model with six groups or external coupling with reactor physics code | same as SOPHT |
| Piping Material | rigid pipe | rigid or elastic pipe |
| Pressurizer Model | thermal equilibrium or adiabatic | thermal non-equilibrium |
| Bleed Condenser Model | thermal equilibrium | thermal non-equilibrium |
| Pump Model | normal operating mode<br>run-down mode | normal operating mode<br>run-down mode<br>restart mode |
| Control Valve Model | liquid or steam valves | liquid or steam valves |
| Separation Model in SG | simple steam flow model or external coupling with STGEN code | level swell model |
| Numerical Method for Thermal-hydraulics | one-step semi-implicit | one-step semi-implicit and two-step implicit |