IAEA ACTIVITIES IN NUCLEAR REACTOR SIMULATION FOR EDUCATIONAL PURPOSES

Aurora Badulescu[1], Robert Lyon[2]

[1] Division of Nuclear Power, Department of Nuclear Energy, International Atomic Energy Agency, Vienna, Austria. Email: a.Badulescu@iaea.org
[2] Division of Nuclear Power, Department of Nuclear Energy, International Atomic Energy Agency, Vienna, Austria. E-mail: r.Lyon@iaea.org

ABSTRACT

The International Atomic Energy Agency (IAEA) has established a programme in nuclear reactor simulation computer programs to assist its Member States in education and training. The objective is to provide, for a variety of advanced reactor types, insight and practice in their operational characteristics and their response to perturbations and accident situations. To achieve this, the IAEA arranges for the supply or development of simulation programs and training material, sponsors training courses and workshops, and distributes documentation and computer programs. Currently, the IAEA has simulation programs available for distribution that simulate the behaviour of BWR, PWR and HWR reactor types.

INTRODUCTION

The International Atomic Energy Agency (IAEA) is assisting developing countries, by the use of reactor simulation, to gain insight in the operation of nuclear power plants. The principal objective is to make "desktop" simulators and training material available to Member States, to assist in their comprehension and training programmes for nuclear power.

The simulation codes provide insight and understanding of the designs as well as a clear understanding of the operational characteristics of the various reactor types. The simulators operate on personal computers and are provided for a broad-based audience of technical and non-technical personnel as an introductory educational tool. The application of the simulation programs is limited to providing general response characteristic of selected types of power reactor systems and they are not intended to be used for plant-specific purposes such as design, safety evaluation, licensing or operator training.

METHODS

This poster covers two simulation programs available for distribution: the Classroom-based Advanced Reactor Demonstrators package and the Advanced Reactor Simulator. The two simulators model the same three types of reactors, BWR, PWR and HWR, however they are somewhat different in nature and may be of interest for differing purposes. The Classroom-based Advanced Reactor Demonstrators package contains modules from an advanced full scope simulator, with an interaction capability reduced to only the necessary software required to demonstrate the general behavior of the plant. It is thus well suited to giving a rigorous demonstration of the behaviour of the reactor systems for a selected set of perturbations. The Advanced Reactor Simulator uses simplified models and provides a hands-on user interface that permits extensive user interaction with the simulation. It is thus well suited for use as an educational tool. The IAEA arranges for the supply or development of such simulation programs and training material, sponsors training courses and workshops, and distributes documentation and computer programs.

Classroom-Based Advanced Reactor Demonstrators

The Classroom-based Advanced Reactor Demonstrators package (CARDs), is a suite of nuclear power plant simulators developed by CAE Electronics Ltd., a Canadian company specializing in full scope flight, industrial, and nuclear plant simulators. The suite consists of PWR, BWR and HWR simulators and operates on a typical PC. The simulators are based on first principles and are modeled to the discrete component level. Control logic is based on the plant elementary diagrams and dynamic models of plant hydraulic circuits are incorporated. The simulators are fully calibrated against both design and plant data.

The CARDs simulators make extensive use of colour graphics to display data such as core flux, temperatures and voids, and to display the status of devices such as pumps and valves. Each of the three reactor models is referred to as a "CARD". The CARDs package serves as a demonstrator and not a training simulator. Although the models used in the CARDs are subsets of CAE's advanced full scope simulators, the package has been reduced to only the necessary software required to demonstrate the general behavior of the plant as seen from the nuclear steam supply system, with all the boundaries to this system emulated to provide the overall dynamic response.

The demonstrators feature CAE's advanced thermal-hydraulic model (ANTHEM™), and advanced reactor model (COMET™). ANTHEM is a non-equilibrium, non-homogeneous, enhanced five-equation, multi-phase, drift-flux model. COMET is a three-dimensional, multi-nodal, reactor kinetics model founded on the fundamental theory of time-dependent neutron diffusion.

The demonstrators consist of the complete neutronic and thermal-hydraulic model of the reactor and reactor coolant systems. The interfaces with the neighboring systems are emulated to provide a realistic feedback. The demonstrators employ a user-friendly, graphic-based man machine interface (TIGERS™) to manipulate the inputs, insert malfunctions, and display the behavior of the systems in a dynamic color-coded manner.

Standard windows *pull-down menus* are used to perform the demonstrator functions. The *Schematic* pull down menu contains three schematics for each CARD: *Overview* displays the reactor and reactor coolant system; *Slices* displays the reactor core flux and temperature distribution; and *Malfunctions* displays the malfunctions available. The *Plot* pull down menu has two functions: *New plot* opens the plotter; and *Plot parameters* opens a preset set of parameters to choose from. The *Options* pull down menu has five commands: *Run/Freeze* runs or freezes the simulation; *Reset* allows the user to reset to either full power steady state or a previously recorded snapshot; *Snapshot* allows the user to snap a certain condition for later use; *Readout* chooses to display or mask the readouts; and *Power Reduction* enables the user to reduce the power to a new level.

1. The BWR CARD

The BWR CARD is a simulation of a typical 670 MW GE BWR/6 vessel and core. In a BWR/6 plant, the re-circulation system flow rate is varied by a flow control valve located in the discharge line of the recirculation pump. While in a BWR/4, the flow is varied by varying the speed of the recirculation pump.

A BWR typically consists of the active core region, the lower plenum, the upper plenum, the separator, the dome and the down-comer and re-circulation system. In a "jet pump" BWR, a distinction is made between a driving flow and a driven flow. The re-circulation pump flow is the driving flow and the jet pump flow is the driven flow. The driving flow for the jet pumps is drawn from the bottom of the annular region surrounding the core into two symmetrically oriented piping loops. Each loop has a re-circulation pump installed below the vessel to provide the required NPSH. The pump discharge flow returns to the vessel at nozzles distributed on the vessel wall and through piping into the jet pumps inlet. Each jet pump consists of an inlet nozzle, a mixer section and a diffuser section. The driving flow and the driven flow mix in the mixer section and get diffused into the lower plenum. The flow then passes through an orifice into the fuel bundle. The steam/water mixture leaving the fuel bundles enters the upper core plenum where it is directed to the steam separators. The steam leaving the separators is dried in the dryer and then exits the vessel through the steam lines. Feedwater is added to the vessel near the top of the downcomer. The downcomer serves as the suction fluid for both the jet pumps and the recirculation pumps.

In the jet pump nozzle, the high static head developed by the centrifugal pumps is converted to a high velocity jet at a low static pressure. The low pressure at the nozzle discharge draws the surrounding fluid into the jet pump throat where it is mixed with the driving flow in the mixer section. A pressure rise occurs in the mixer section due to the velocity profile rearrangement and the momentum transfer in the mixing process. The fluid then enters a diffuser section which slows the relatively high velocity mixture and converts the dynamic head into a static head. Thus, the flow leaves the diffuser at a higher pressure than that of the downcomer region. The main resistance against which the jet pumps are working are the core drop and the steam separator pressure drop. Core flow is very nearly a linear function of the jet pump driving flow except at low flow rates where natural circulation becomes significant. The jet pump driving flow, and therefore the core flow, are varied by a variable recirculation pump speed (BWR/3/4) or flow control valve (BWR/5/6).

The BWR CARD simulates the neutronic behavior of the reactor core and the thermal-hydraulic behavior of the vessel from the feedwater inlet to the main steam lines. Feedwater, main steam and emergency core injection flows are emulated to provide adequate dynamics during transients. The logic is also functionally emulated. Power changes in the BWR CARD are made through varying the core flow through varying the valve position at the discharge of the recirculation pump. Simply put, increasing the flow results in increasing the power, and vice versa, decreasing the flow decreases the power.

Available *malfunctions,* that can be initiated from the specific malfunction window, are: main steam line break; loss of coolant accident with and without emergency core cooling; loss of feedwater; reactor scram; dual pump trip; and one pump trip.

2. The PWR CARD

The reactor coolant circuit of the PWR CARD consists of four heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and a steam generator. In addition, the system includes a pressurizer, a pressurizer relief tank, interconnecting piping, valves and instrumentation necessary for operational control. During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel, the steam generators and the reactor coolant loops. The water which serves as a coolant, moderator and solvent for boric acid is heated as it passes through the core. It then flows to the steam generators where heat is transferred to the main steam system, and returns to the reactor coolant pumps to repeat the cycle. The reactor coolant circuit pressure is controlled by the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays.

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The PWR CARD simulates the core neutronic behavior of the PWR core using 8 axial and 49 radial control volumes and the thermal-hydraulic behavior of the RCS and the steam generators from the feedwater inlet to the main steam lines. Feedwater, main steam, and emergency core cooling flows are emulated to provide adequate dynamics during transients. The logic is also functionally emulated. Power changes are made through varying the reactor power setpoint using a rate defined by the user.

Available *malfunctions* are: steam generator malfunctions (main steam line break and tube rupture); pressurizer relief valve stuck open; loss of feed water; reactor trip; reactor coolant circuit pump trip (individual or all); reactor coolant circuit loss of coolant accident (cold or hot leg); control rod drop; and miscellaneous malfunctions (inhibition of reactor trip or safety injection).

3. The CANDU 6 CARD

The primary heat transport system circulates pressurized heavy water through the reactor fuel channels to remove heat produced by the fission of uranium fuel. The heat is carried by the coolant to the steam generators where it is transferred to the light water side to generate steam which in turn drives the turbine generators. The main heat transport circuit consists of two loops. Each loop provides bi-directional flow through half the reactor core. In a typical CANDU 6, each loop consists of 190 fuel channels, two vertical steam generators with integral pre-heaters, two primary heat transport pumps, two reactor outlet headers, and two reactor inlet headers.

The fuel channels are horizontal. The headers, pumps, and steam generators are located above the reactor. This elevation difference promotes thermosyphoning flows on loss of power to the heat transport pumps which ensures a minimal heat removal capacity.

Each loop has two inlet and two outlet headers, one of each being located horizontally, at each end, above the reactor. Heavy water is fed to each of the fuel channels through individual feeder pipes from the reactor inlet headers and heavy water is returned from each fuel channel through individual outlet feeder pipes to the reactor outlet headers. Individual feeders and channels are sized to provide uniform coolant quality (~4%) at the exit of each channel, taking into account the non-uniform reactor heat profile. The reactor outlet headers at the end of the reactor are connected to a common pressurizer. Valves in the connecting lines enable isolation between the two loops, if required, as in the event of a loss of coolant accident. The heat transport system pressure and heavy water inventory are regulated by the pressurizer and a feed and bleed system.

The CANDU CARD simulates the core neutronic behavior of the CANDU core using 6 axial and 49 radial control volumes, the thermal-hydraulic behavior of the primary heat transport system and all the major surrounding systems such as feedwater, main steam, emergency core cooling, and the corresponding logic associated with these models. The major digital control computer programs emulated are: heat transport control; boiler level control; boiler pressure control; reactor regulating system; and moderator temperature control. Power changes are made through varying the reactor power setpoint in the reactor regulating system using a rate defined by the user.

Available *malfunctions* are: steam generator malfunctions (main steam line break and tube rupture); pressurizer steam break; loss of feed water; reactor trip; primary heat transport pump trip (individual or all); loss of coolant accident (Inlet or outlet header breaks); liquid relief valve failures; and miscellaneous malfunctions (inhibition of reactor trip or safety injection).

The Advanced Reactor Simulator

The Advanced Reactor Simulator was developed by Microsimulation Technologies, based on PCTRAN, a PC-based FORTRAN transient analysis code, developed in the United States. During the past seventeen years, the performance of PCTRAN has been documented against known data. The Advanced Reactor Simulator runs on a typical PC and models the reactor types: PWR, BWR and HWR in the 600MWe range. For the PWR models, plants with vertical inverted U-bend steam generators of Western design, plants with horizontal steam generators as designed in the former Soviet Union, and a next-generation PWR with passive safety features are included. The Simulator operates in real or accelerated time and covers the nuclear steam supply system, containment, control systems, and safety systems. Malfunctions and parameters can be selected to model normal and abnormal design basis conditions relevant to each reactor type. Conditions outside the design basis can also be simulated.

1. Simulation Models

Reactor core kinetics is based on a point neutron kinetics model with one delayed neutron group. Reactivity control is provided from external sources such as absorber rods and boron injection and reactivity feedback is included from moderator temperature, fuel temperature, coolant or moderator density and void fraction. Control can be either in the "reactor following" or "turbine following" mode. Following reactor shutdown, standard decay energy release rates are used for decay heat generation.

The reactor coolant system thermal hydraulics model is first-principle in mass and energy balance. A reduced-node approach (compared to the original PCTRAN simulator) is used. A fluid boundary separates saturated two-phase fluid volumes from subcooled liquid volumes. For example, for a PWR, the former is the pressurizer and the latter is the rest of the reactor coolant system. During a transient, the boundary is allowed to move upward or downward. Phenomena such as two-phase in the coolant loops, fractional core water level, and water-solid conditions can be reproduced. For a BWR, the reactor core and steam dome form the two-phase saturated region, while fluid in the remainder lower plenum and recirculation loops belongs to the subcooled region.

For forced circulation when the reactor coolant pumps are on, full (volumetric) rated flow is assumed. For accident situations, in the event that the pumps remain operating while the system is flashing, the volume occupied by the void reduces the flow. Once the pumps are tripped, the flow coasts down exponentially until natural circulation is established. The reactor hot and cold leg temperatures for the PWR models are calculated from the heat balance between the reactor core heat generation and the steam generators heat removal at the given loop flow rate.

When a break is introduced, the break discharge model used depends on the break conditions. If a break is within the subcooled region, a choking orifice flow model is used. If the break is located in the saturated region's liquid phase, a liquid critical flow model is used, and if in the vapor phase, the vapor critical flow is used.

Two *steam generators* are included in the PWR models. For PWR plants with more than two steam generators, the steam generators are lumped into two loops. The steam generator secondary side is modeled as a separate saturated two-phase volume in thermal contact with the primary. The wet surface area is a constant if all tube bundles are submerged which is the case for most operating conditions. During a loss of feedwater event, if the steam generator water level falls below the top of the tube bundles, heat transfer is taken to be proportional to the fractional water level height.

Fuel temperature is computed by a simplified heat transfer model that uses an overall heat transfer coefficient between fuel and coolant derived from reactor power and the average fuel and coolant temperatures during normal operation. Change in the heat transfer coefficient for impaired cooling is addressed, peak cladding temperatures are calculated and the extent of core damage is estimated from published correlations of percent of fuel rods with ruptured cladding vs. maximum core temperature. The steam/zirconium reaction is modeled together with the production of hydrogen and the effect of heat on containment conditions.

High and low pressure emergency core cooling systems are modeled as flow injections into the reactor coolant system. Flows are computed as functions of the corresponding pump's head curve and the back pressure of the reactor coolant. The different effects of spray systems and injection systems are modeled. A spray is more effective in pressure reduction, whereas an injection stream tends not to mix with the fluid in the receiving volume. Injection is mostly for coolant makeup and has a delayed effect on pressure and temperature. Adiabatic conditions are assumed for computing nitrogen pressure change in the accumulator tanks.

A dry *containment* model is used for PWRs, based on a mass and energy balance in a homogeneous, enclosed compartment, with participation of non-condensable air and hydrogen. For the BWR, a pressure suppression system consisting of a dry well and a wet well is modeled.

2. Program Operation

The models are programmed in FORTRAN and linked with graphics modules and a plant database into integrated simulation software that operates in the MS/DOS or Windows environment.

For each reactor, an on-screen "mimic" provides a user-friendly interface that facilitates control actions and diagnostics by the operator. Virtual control panels that are simplified versions of those used in a typical nuclear power plant of that type are included. Normal operation, power maneuvering, perturbations and accident situations can be induced by selection of initial conditions and malfunction states, and by actual adjustment of control points, valve positions, and equipment status via the mimic.

The *display* represents the controllable system as small panels with the main components shown as icons. Components such as power-operated-relief-valves and safety valves of the pressurizer and the steam lines, pressurizer spray valve and heaters, main steam isolation valves, turbine bypass (steam dump) valves, feedwater valves and reactor coolant pumps are displayed. Their status is indicated by color and can be overridden by the operator using the mouse to select objects for action (for example, push a button, turn on a pump). Keyboard access is only required for such actions as entering malfunction values, specifying a new initial condition and entering scale values for data trends. Control rod position and motion are displayed and pipe breaks are shown with flashing sprays at the break location with the leakage flow digitally displayed.

A typical run commences with selection from a set of initial conditions corresponding to various power, flow, and time-of-life conditions. During operation, the mimic dynamically displays the plant condition and the operator can initiate malfunctions that cover all categories analyzed in the plant's safety analysis report (and beyond for some cases). The severity, delay and ramp time of each malfunction is entered. The operator can trip the turbine or the reactor and can override the status of valves or pumps in the mimic, causing on/off status or partial failure at fractions of the full capacity. The operator can, for example, override the automatic initiation of emergency core cooling system pumps and take manual control. A set of malfunctions derived from the safety analysis report for each reactor has been prepared and can be selected, together with severity (for example, break size). Typical malfunctions include: loss of coolant; steam line break ; loss of feedwater; and loss of flow.

The status of the reactor protection system and safety feature actuation system is displayed. The reactor will trip automatically upon conditions exceeding any of the reactor protection system set points. The corresponding symbol will turn red and all control rods will be inserted.

Output variables can be viewed in "trend" graphs on the screen as the simulation progresses. Graphs can be printed at the end of the simulation. The operator can select to have the calculated transient parameters written into output files for detailed post-simulation analysis.

RESULTS

The Classroom-based Advanced Reactor Demonstrators package (CARDs), is a suite of nuclear power plant simulators developed by CAE Electronics Ltd., a Canadian company specializing in full scope flight, industrial, and nuclear plant simulators. The suite consists of PWR, BWR and HWR simulators and operates on a typical PC. The simulators are modeled to the discrete component level. Control logic is based on the plant elementary diagrams and dynamic models of plant hydraulic circuits are incorporated. The simulators are fully calibrated against both design and plant data but serves as a demonstrator and not a training simulator.

The Advanced Reactor Simulator (ARS) was developed by Microsimulation Technologies, based on PCTRAN, a PC-based FORTRAN transient analysis code, developed in the United States. During the past seventeen years, the performance of PCTRAN has been documented against known data. The ARS runs on a typical PC and models the reactor types: PWR, BWR and HWR in the 600MWe range. For the PWR models, plants with vertical inverted U-bend steam generators of Western design, plants with horizontal steam generators as designed in the former Soviet Union, and a next-generation PWR with passive safety features are included. The Simulator operates in real or accelerated time and covers the nuclear steam supply system, containment, control systems, and safety systems. Malfunctions and parameters can be selected to model normal and abnormal design basis conditions relevant to each reactor type. Conditions outside the design basis can also be simulated.

DISCUSSION

The CARDs program simulates the neutronic behavior of the reactor core and the thermal-hydraulic behavior of the primary heat transport system and all the major surrounding systems such as feedwater, main steam, emergency core cooling, and the corresponding logic associated with these models. The major digital control computer programs emulated are: heat transport control; boiler level control; boiler pressure control; reactor regulating system; and moderator temperature control. Available malfunctions are: steam generator malfunctions; loss of coolant accident; loss of feedwater; liquid relief valve failures; and other miscellaneous malfunctions.

The ARS program provides an on-screen "mimic"; a user-friendly interface that facilitates control actions and diagnostics by the operator. Virtual control panels that are simplified versions of those used in a typical nuclear power plant of that type are included. Normal operation, power manoeuvring, perturbations and accident situations can be induced by selection of initial conditions and malfunction states, and by actual adjustment of control points, valve positions, and equipment status via the mimic.

ACKNOWLEDGEMENT

The author is grateful to Mr. M. Chatlani of CAE Electronics Ltd., and to Dr. L.C. Po of Microsimulation Technologies for kind permission to derive material for this paper from their respective user manuals.

CONCLUSIONS

Nuclear reactor simulation computer programmes operate on personal computers and simulate responses of a number of reactor types (BWRs, PWRs and HWRs) to operating and accident conditions. The objective is to provide training tools for university professors and engineers involved in teaching topics in nuclear energy. The tools are also supplied directly to students, junior engineers, and senior engineers and scientists interested in broadening their understanding of the topic.

Currently, the IAEA has two simulation programs available for distribution: the Classroom – Based Advanced Reactor Demonstrators package and the Advanced Reactor Simulator, which model BWRs, PWRs and HWRs. However, they are somewhat different in nature and may be of interest for differing purposes. One is well suited to giving a rigorous demonstration of the behaviour of the reactor systems for a selected set of perturbations and the other well suited for use as an educational tool.