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PCTRAN: Education Tool for Simulation of Safety and Transient Analysis of a Pressurized Water Reactor

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Abstract

In Bangladesh, VVER type nuclear power plant is going to be operated by 2022/23 at Rooppur, Ishwardi, Pabna. In order to develop skilled manpower for Rooppur Nuclear Power Plant, it is important to impart practical training and education of the first generation of technical staff in Bangladesh. For practical training and education of the first generation of technical staff in Bangladesh, Personal Computer Transient Analyzer (PCTRAN) is a "live" simulation tool complementing classroom lectures. PCTRAN is a nuclear reactor transient and accident simulation tool that operates on a personal computer.Concepts of neutron multiplication, criticality, thermal hydraulic safety parameters, transient analysis, accidental dose estimation and Xenon poisoning and feedback, can be demonstrated in PCTRAN's nuclear reactor model.In addition to normal operation of start-up, power maneuver and shutdown, operational transients and accidents can be simulated.In this study, safety and transient analyses of a TRIGA pool type reactor and VVER type pressurized water reactor (PWR) using the PCTRAN simulator was performed. Various scenarios of transients and accidents likely to occur at any nuclear power plant were simulated. In addition, statistical analyses of the PCTRAN results were carried out. In this study the transient behavior simulation has been successfully benchmarked by using PCTRAN based on end-users input data with a reactor point kinetics model. The nuclear safety related concepts and phenomena could be easily explained with the simulators along with understanding of valuable technological differences in various designs. The simulations are very appropriate in the light of Bangladesh's plan to generate nuclear energy in the Rooppur of 1200 MWe from VVER nuclear power plant by 2022/23.

Keywords**:** *Transient analysis, Reactivity, Nuclear power, Decay power, Simulator, Nuclear reactor*.

1. Introduction

PCTRAN is the first PC-based nuclear power plant simulator developed by Micro-Simulation Technology [1]. It is used by International Atomic Energy Agency (IAEA), many government agencies and nuclear power plants all over the world [2-4]. PCTRAN is a software package that can simulate a variety of accident and transient conditions for nuclear power plants as well as research reactor. It operates in the Windows XP environment at a speed faster than real-time. A high-resolution color mimic of the Nuclear Steam Supply System (NSSS) and containment displays the status of important parameters and allows simulation of operator actions by interactive control. The source code of PCTRAN has been written based on Microsoft Visual Basic 6.0. Data input/output are in MS Office's Access database format.

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Reports and data can be transferred conveniently through all Windows-based software products over the entire exercise network. The simulator can be executed on a personal computer (PC), to operate essentially in real time, and to have a dynamic response with sufficient fidelity to provide nuclear power plant responses during normal operations and accident situations. The main purpose of this simulator is operator training and a dynamic test to validate the control logics in reactor regulating system (RRS). These simulators can provide basic understanding in the operational of NPP, control systems, safety systems and simulate the transients and accidents behavior of the common nuclear power plant. Operating on personal computers these software provide a thorough hands-on demonstration of the basic operational principles of various nuclear power plants by illustrating general concepts, and demonstrating fundamental safety processes in normal and transient/accident conditions. The aim of this study is to providing base information using the results from the simulation of a variety of operating, accident and transient conditions of a research and power reactor using the PCTRAN simulators and analyzing the resulting transient conditions as simulated. This simulation software was provided by the IAEA during an ICTP course on "Workshop on Nuclear Power Plant Simulator for Education".

2. Windows Based Simulation System

2.1 Lists of Simulators in PCTRAN

The International Atomic Energy Agency (IAEA) has established an activity in nuclear reactor simulation computer programsto assist its Member States in educationand training. The objective is to provide, for a variety of advanced reactor types, insight and practice in reactor operational characteristics and their response to perturbations and accident situations. For achieving the objective, the IAEA arranges for the supply or development of simulation programs and educational materials, sponsors workshops, and distributes documentation and computer programs. As part of these programs, the author has attended a Workshop on application and development of nuclear reactor simulators for educational purposesat ICTP. The ICTP provided a CD ROM to each course participant. The simulators based on the following nuclear reactor are available in the IAEA CD ROM for educational and training purpose[2-4]:

2.1.1 Pressurized Water Reactor (PWR) Simulators

- Generation II PWR
- 2-Loop Large PWR (Korean-OPR 1000)
- Russian-type PWR (VVER-1000/VVER-1200)
- Advanced Passive PWR (AP600/AP1000

2.1.2 Boiling Water Reactor (BWR) Simulators

- Advanced Boiling Water Reactor (BWR)
- Advanced Passive BWR (ESBWR)

2.1.3 Pressurized Heavy Water Reactor (PHWR)

- Conventional PHWR (CANDU-6)
- Advanced PHWR (ACR-700)

2.1.4 TRIGA Research Reactor Simulator

2.2 Theory and Solution Technique

2.2.1 Six-group Point Kinetics Model

A point kinetics equation with six delayed neutron groups and reactivity control from external sources and feedback was modeled. The model equations are expressed by [5]

$$\frac{dn}{dt} = \frac{\rho - \beta}{\tau} n + \sum_{i=1}^{6} \lambda_i c_i$$

$$\frac{dCi}{dt} = \frac{\beta i}{\tau} n - \lambda_i c_i$$

where, n = neutron density
ρ = reactivity = (k -1)/k
k = effective multiplication factor
β_i = delayed neutron fraction for the ith group
τ = neutron life time
λ_i = decay constant for the ith group
C_i = precursor concentration
S = neutron source

The six-group point kinetics equations are solved by finite difference method in the program. Reactivity is controlled by rod movement and boron concentration adjustment. Its feedback in moderator density and fuel temperature (Doppler) effects are described below:

2.2.2 Core Reactivity Corrections of Point Kinetics Model

The reactivity corrections in the model are calculated as:

$K_\infty = K_\infty^0 \, (1- \Delta K_B/K_\infty^0)\,(1-\Delta K_{DOP}/K_\infty^0)\,(1-\Delta K_{Xe}/K_\infty^0)(1-\Delta K_{Sm}/K_\infty^0)$

where,

K_∞ = the corrected K_∞ of core.

K_∞^0= the uncorrected K_∞ of fuels (combination results of rod controlled and uncontrolled K_∞^0) in the core.

ΔK_B = the reactivity correction of core due to boron.

ΔK_{DOP} = the reactivity correction of fuel temperature in the core.

ΔK_{Xe} = the reactivity correcti on of core due to Xenon-135.

ΔK_{Sm} = the reactivity correction of core due to Samarium-149.

The uncorrected $K_\infty 0$ of fuels in the core is retrieved from the following formula:

$K_{\infty}u = A1 + A2 * \rho + A3 * \rho 2 + A4 * \rho 3$ --- uncontrolled core fuel K_∞

$K_{\infty}c = A5 + A6 * \rho + A7 * \rho 2 + A8 * \rho 3$ --- controlled core fuel K_∞

$K_\infty^0 = K_{\infty u} + f_{rod} * (K_{\infty c} - K_{\infty u})$

where,

ρ is the moderator density of core (gm/cm^3).

f_{rod} is the fuel rod controlled fraction in the core.

2.2.3 Fuel and Clad Temperatures

A mechanistic model accounting for the temperatures of fuel and cladding has been constructed in PCTRAN. This model can simulate thermal power transmitted into the coolant in contrast to nuclear power generated by the fuel during startup and power operation. Fuel thermal heat transfer is described by

$$Q_{F\text{-}CL} = U_F\, A_{fuel}\, (T_F - T_{CL})$$

$$Q_{CL\text{-}Water} = U_{CL}\, A_{Clad}\, (T_{CL} - T_{Water})$$

where T_F is the average fuel temperature, T_{CL} the average clad temperature and T_{water} water temperature. The heat transfer coefficient U is established at steady state full power.

The Global Core reactivity ΔK_{Core} of the core is calculated as:

$K_{EFF} = K_\infty / K_{EIGEN}$

$\Delta K_{Core} = 1 - 1/K_{EFF}$

where

K_{EIGEN} is the core eigen value; the critical value in the core analysis calculation.

Neutron density is directly proportional to the nuclear fission power. Remainder of the core power is provided by fission product decay heat. The decay heatis given by an eleven group equation:

$$Q_{DH} = \sum_{i=1}^{11} E_j \, e^{-\lambda jt}$$

where,

E_j = amplitude of j-th term

λ_j = decay constant of j-th term

t = elapsed time since shutdown

Concentration of each decay group $\gamma_j$is represented by

$$\frac{d\gamma_j}{dt} + \lambda_j \gamma_j \;\; t = E_j(t)n(t)$$

The total power in the nuclear reactor core is given by

$$P(t) = P_o\{n(t)E_f + \textstyle\sum_{i=1}^{11} \lambda_j \gamma_j\}$$

where E_f is about 0.93, i.e. fission heat is about 93% of total core power.

In a pool reactor, the bulk of water is maintained near or slightly above the room temperature. If the clad surface is heated to above saturated temperature of water, i.e. 100°C, localized sub-cooled boiling may take place. The empirical correlation by Thom [6] was used to calculate the void production rate. It affects the local water density; that in turn feedback to the kinetics of neutron density calculation. Transient fuel temperature will be calculated by the imbalance between power deposited in the fuel and heat sink.

2.3. Simulator Display Screens

Basic mathematical algorithm is based on reactor point kinetics model with six groups of delayed neutrons [5]; the decay heat model uses a three-group approximation; 2-phase flow and heat transfer. Reactivity calculations include reactivity of control rods, fine motion control rods, and reactivity feedback effects due to Xenon, 2-phase voiding in channels; fuel temperature (Doppler) and moderator (light water) temperature. Operation follows strictly the Windows XP environment in graphic-user-interface (GUI). Data input/output are in Access database format. Selection of equipment malfunctions and accident types are from a drop-down menu. The following aspects are integrated into one simulation system [7-9]:

- On Basic Reactor Concept: From reactor physics, thermal-hydraulics to plant control

2.4 Simulation Analysis

PCTRAN has several modules corresponding to each nuclear reactor type including VVER-1000. It includes all possible disturbances to a nuclear reactor system [1,4]:

- Normal operation control – start-up, shutdown, power ramp
- Loss-of-coolant-accident (LOCA) or steam line break
- Loss of flow, single or two-phase natural circulation
- Turbine trip with or with bypass, station blackout
- Steam generator tube rupture (PWR)
- Feedwater transients (pipe break, loss of feed or loss of heating)

RadPuff is an atmospheric dispersion module which can process the radiation dose release generated by PCTRAN calculation [10, 11].

In this studyas benchmarking, TRIGA-Experimental Pool Reactor and VVER-1000 have been considered for transient analysisby using PCTRAN software. A 3 MW TRIGA Mark-II research reactor is available at AERE under Bangladesh Atomic Energy Commission (BAEC) and BAEC is planning to install a VVER type nuclear power plant at Rooppur, Pabna by 2023/2024 [12-14].This simulation will help the reactor operation people to get more information about the control of nuclear reactorbased on reactor point kinetics.

$$M_F C_{FP} \frac{dT_F}{dt} = Q_{Core} - Q_{F-CL}$$

where, M_F is mass of fuel in the core, Cp is its specific heat and T the temperature in average. Qis the power with subscript Core representing the core power and F-CL for heat transfer from the fuel to cladding.

- On Actual Plant Designs: From experimental pool reactor to advanced VVER, PWR and BWR
- On Plant Operations: From normal start-up, shutdown, power operation to transient and accident mitigation
- On Defence-in-depth Concept: From fuel element, reactor coolant and containment boundary to severe accident and offsite dose dispersion
- PC-based simulators provide a relative easy to learn educational tool.
- Simulators provide insight and understanding into basic Operational Characteristics, Reactivity Control Features, and Safety Systems.
- All students to explore and examine various plant responses to Transients, Malfunctions and Accidents.
- Anticipated transient without scram (ATWS)
- Containment failure (failed isolation or containment breach)
- Loss of AC power (loss of offsite grid and loss of diesel emergency power)
- Recirculation pump trip
- Station blackout or loss-of-load
- Inadvertent rod withdrawal or insertion
- Boron dilution transient
- Any combination of above

2.4.1. TRIGA-Experimental Pool Reactor Simulator

TRIGA (Training, Research, Isotopes, General Atomics) is a class of small nuclear reactor designed and manufactured by General Atomics. TRIGA is a pool-type reactor that can be installed without a containment building, and is designed for use by scientific institutions and universities for purposes such as undergraduate and graduate education, private commercial research, non-destructive testing and isotope production. Micro-Simulation Technology (MST) has prepared an experimental pool TRIGA reactor simulator for training and educational purpose. The main features and capabilities are briefly described below:

1. The software will simulate water moderated and cooled pool type reactor using a point kinetics model with 6 delayed neutron groups. The model will include simulation of the Iodine 135-Xenon 135 fission product poison decay chain with adjustable “fast time” capability (to establish steady state iodine and xenon levels as well as for student demonstration). An instructor adjustable neutron source will be provided to permit observation of sub-critical multiplication during the simulated approach to criticality.

2. The input parameter to the point kinetics reactor model will be the core average reactivity computed from the following:

a. An instructor adjustable base reactivity value;

b. Simulated control rod position;

c. Local (e.g. in-core) moderator density and voids;

d. Fuel centerline temperature (Doppler);

e. Xenon-135 Level.

3. A simplified reactor thermal-hydraulic model will be provided which computes fuel centerline temperature (for Doppler reactivity feedback) as well asin-core moderator temperature and density. The bulk (e.g. pool) temperature will be computed as well. The pool surface will be assumed to be at normal atmospheric pressure at all times, and bulk boiling of the pool will not be assumed. Simulation of core void will be limited to that void produced by sub-cooled and nucleate boiling within the core; transition and bulk boiling will not be simulated.

4. An instructor interface will be provided with the following capabilities:

a. The ability to “Freeze” and “R esume” the real-time simulation at any time;

b. The ability to “Snap” the existing simulator condition to multiple hard-disk files for later use as an initial condition(s);

c. The ability to “Reset” the simulator to a previously stored initial condition;

d. The ability to run the entire simulation in “slow time” (to permit observation of fast transients);

e. The ability to run the Iodine 135−Xenon 135 decay chain in “fast time” (to permit observation of Xenon transients);

f. The ability to change key reactor parameters including:

- Doppler coefficient;
- Moderator temperature coefficient’
- Basic core reactivity
- Neutron lifetime
- Decay neutron precursor fractions
- Delayed neutron precursor decay constants
- Control rod worth
- Initial pool water temperature
- Neutron source strength
- Control rod speed

Note that, in general, these parameters must be set with the simulated control rods fully inserted into the core in order to avoid unrealistic neutron flux transients.

g. A plotting package which permits viewing of key parameters on a “strip chart” display as well as the ability to transfer stored plot data to an Excel TM spreadsheet.

5. A “virtual” control panel graphic display will be provided with the following capability:

a. Display of critical reactor parameters including:

- bulk pool temperature
- local in-core moderator temperature
- source range monitor counts and rector period

- intermediate / power range monitor neutron flux and period

- core average reactivity
- control rod positions

b. The ability to insert and withdraw control rods at a fixed speed (adjustable via the instructor interface as previously described);

c. The ability to manually “scram” the reactor.

As part of our study, we chose a TRIGA type experimental pool reactor for the simulator model. The fuel is in cylindrical shape. Uranium-zirconium hydride UZrH of 19% enriched U-235 is the fissile material in the fuelassemblies. Water used for cooling is always in atmospheric pressure. The control rods containboron are used for neutron absorption. Combined they have the function of power regulation,chemical shim and safety shutdown. Graphite is used for moderation.There is a heat exchanger for heat removal during power operation. During source range andintermediate range operation with little core heat, the heat exchanger pumps could be shut off.The pool is maintained borated. A tank containing borate acid concentration higher than the pool’s is used for increase the pool concentration. Another tank with pure water is used for deboration.By opening or closing the corresponding valves and pumps the operator can adjust theboron concentration. It is a slower process than using the rods.During power operation, the clad temperature may approach boiling temperature of water.Localized or nucleate boiling may take place that reduces the average water density in the core. This effect is considered in addition to Doppler (fuel temperature) and boron reactivity in theneutronics calculation.

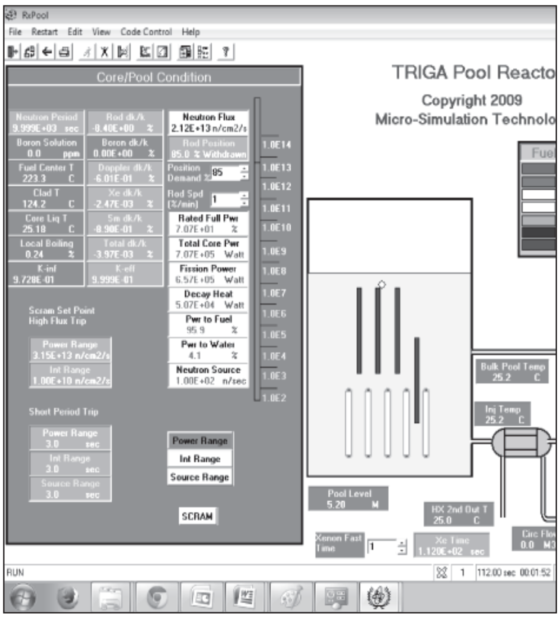

Fig. 1: PCTRAN Experimental TRIGA Pool Reactor Mimic.

The TRIGA model is of rated power at 250 kW thermal with neutron flux at 2×10^{10} n/cm^2/sec. It can be used for education of nuclear reactor principles on the concept of delayed neutron effect, multiplication factor, criticality, control by rods and boron concentration, feedback on fuel (Doppler) and moderator temperatures, xenon and samarium poisoning, etc. The screen shot of TRIGA model is shown in Fig. 1.

Some typical results for TRIGA reactor simulation are shown in Figs. 2-4.It is found from Fig. 2 that the reactivity attains stable value (k=1) after 300 sec. The time variation of reactor core power and decay heat power is shown in Fig. 3 due to insertion of positive reactivity. The neutron flux variation is shown in Fig. 4.After 500 sec, the neutron flux attains constant value.In reference to Fig. 3, we note that there is a very gradual increase in the average reactor core power starting at ~400 sec and rising at ~900 sec. Thus if corrective action has not been forthcoming at this juncture, rapidly escalating events going forward is very likely to negate any mitigating actions taken. As postulated there is a certain period of time within which damage can be averted under the situation created by this transient. Of all the transients simulated it is clear that residual heat removal failure without replenishment is likely to have the most severe outcome of any of the transients. The intent of the analysis was to gain greater understanding as to the types of events that have occurred during reactor operation, and to determine which characteristics of these events were important in terms of risk.

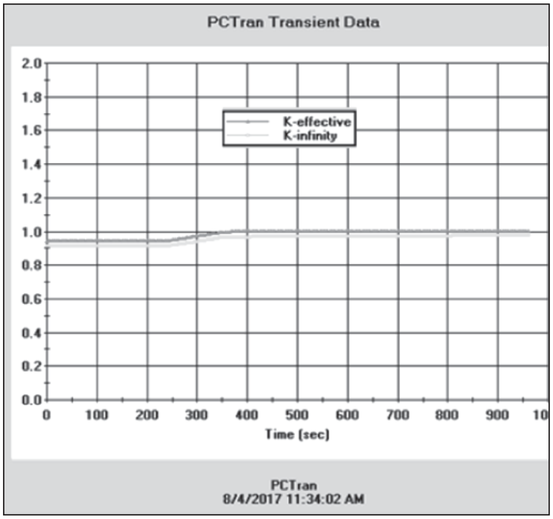

Fig. 2: Time vs. reactivity

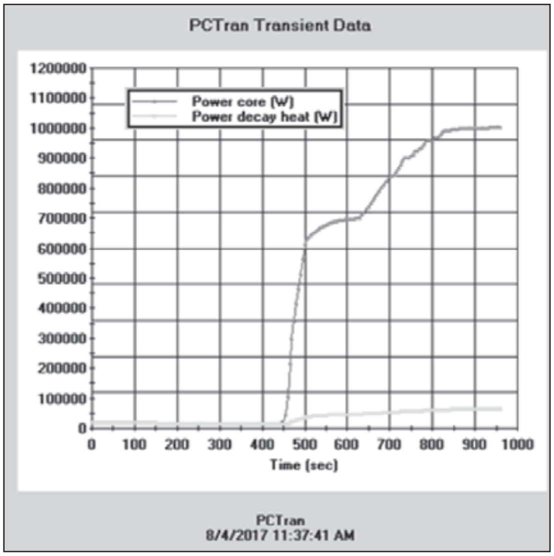

Fig. 3: Time vs. reactor core power and decay heat power

2.4.2 VVER-1000 simulator

VVER is an abbreviation for Water Water Energy Reactor. It is a pressure vessel type nuclear reactor with water used both as moderator and coolant, resulting in athermal neutron spectrum. The number following the reactor type usually indicates therated power of the unit. Thus, VVER-1000 designates a unit with 1000 MW electricalpower. Heat that is generated in the reactor core from the fission of

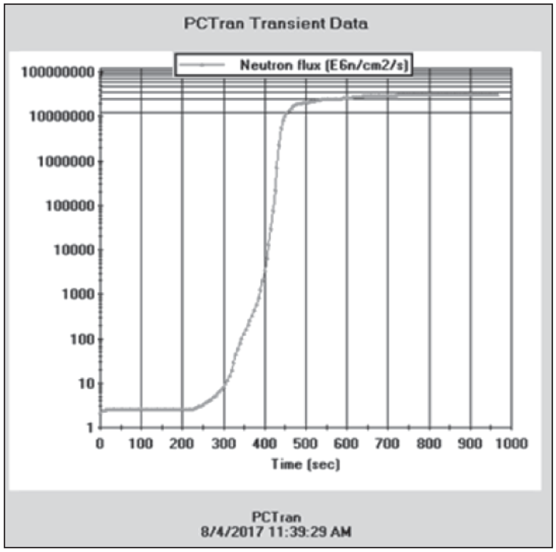

Fig. 4: Time vs. neutron flux.

nuclei in the fuel isremoved by the coolant (for NPP with VVER, the coolant is water or water-steam mixture). After leaving the reactor core, the coolant is transported along the part of theprimary circulation circuit called “hot leg” to the steam generator. The steam generator is a heat exchanger in which the heat from the primary circuit coolant transfers to feed water of the secondary circuit to form steam. After the steam generator, the coolant is transported along the part of primary circulation circuit called “cold leg” back to the reactor vessel. There are four circulation loops in the primary circuit of the NPP with VVER-1000 reactor. The coolant is pumped by four main circulation pumps, installed one in eachloop. The VVER-1000 reactor is a vessel-type Light Water Reactor where chemicallypurified water with boric acid serves as coolant and moderator. The reactor is intendedfor generation of heat within the NPP nuclear steam supply system. Regulation ofreactor power and suppression of the fission chain reaction is carried out by twosystems adjusting reactivity, which are based on different principles:

–Introducing solid absorbers -control rods system (control & protection system –CPS), and

–Injection of liquid absorber -boron regulating system.

Control rods are used for changing reactivity in maneuvering regimes and for reactor shutdown in normal and emergency operation conditions. Boron regulation is used for slow changes in reactivity. The boron concentration is changed during the life cycle.

2.5 VVER-100 Simulator Runs

PCTran has 23 possible malfunctions which can be applied separately or in combination with each other, some of them can also be adjusted in terms of the severity of the particular transient, for example, the size of the loss of coolant accident can be adjusted [2-4]. Six initial conditions (ICs) are saved as "protected" set that ranging from full power to intermediate range and shutdown conditions. At any time during a run the user can save additional IC's with index higher than 6. In the panel the reactivity and thermal-hydraulics variables are displayed inthe panel. To pull the rods, user can use the scroll bar to set a demand of withdraw percent and the rate in % per minute. If the rate is too fast, it will be witnessed by the shortening of the reactor period (i.e. efold multiplication factor). If the period is shorter than the scram set point of power range, intermediate range or source range respectively, the reactor will scram. For power range and intermediate range, there are also high flux scrams. So during the period of intermediate range on route topower range, the user should reset the monitor to power range by clicking the top button. Otherwise the high flux scram in the intermediate range would scram the reactor. At any time during startup, the operator should watch the neutron period closely against the short period scram. If it is getting close, he or she should reduce the rod speed or stop all together. If the reactor is scrammed and the operator decides to start it up by pooling the rods, the scram button will be cleared as soon as the rod position is greater than 1%.

By choosing one of the full power conditions, the operator should check the core neutronics and thermal-hydraulic condition. By pressing the Scram button the control rods will be inserted by gravity and the reactor becomes subcritical. There is decay heat from the core slowly decreasing with time. Feedback by Iodine and Xenon buildup will be indicated. It is relatively slow so fasttime should be used for observation this effect. There is a dedicated time-time multiple for this purpose. The indicated time post shutdown is displayed by its side. The source term is a user input imbedded in IC Thermo Data table. A number of operational transients have been analyzed to show the plant's control and protection system response without causing a reactor scram.For operational transients we have successfully simulated all the cases as outlined in section 2.3.In training on the fundamentals of plant transients it is paramount to understand how each and every system in the overall plant behaves and corresponds. Such systemic response behavior could be easily demonstrated with the help of the graphical user interface of the simulator.Here we show oneillustrative example [2-4] on training-by-doing with the goal to build an understanding of the fundamentals of a PWR plant response during a hot full power fast control rod withdrawal.In the FSAR (Final Safety Analysis Report), the fast withdrawal of a rod at power results in the reactivity insertion of 6 x 10^{-4} Δk/sec. This uncontrolled withdrawal results in an increase in core heat flux, which causes average RCS temperature to increase. The transient is terminated by a reactor trip at 118% neutron flux. The case is benchmarked in PCTRAN by selecting a beginning of life full power initial condition (IC #3). This IC uses a zero moderator coefficient, which was specified in the FSAR. Malfunction 12 was selected with a ramp time of 100 seconds and a reactivity insertion of 6% to produce the desired reactivity insertion rate. Neutron flux, RCS pressure, RCS average temperature, and DNB ratio are plotted for comparison to the FSAR. The PCTRAN simulation is comparable with those of FASR results [2-4].

There are many other transients, accidents and normal operation scenarios that could be simulated using the IAEA's simulators being therefore helpful in teaching safety related aspects of various types of power plants. A comparison of safety features across different technologies provides deeper understanding of the underlying safety designs of each type of the plants.

3. Future Study

BAEC is going to install a VVER-1200 type nuclear power plant at Roppur, Pabna by 2023/2024. The Russian VVER-1200 (or NPP-2006 or AES-2006) is the version currently offered for export. It is an evolution of the VVER-1000 with increased power output to about 1200 MWe (gross) and providing additional passive safety features. A VVER-1200 simulator has already been procured and to be installed in the Department of Nuclear

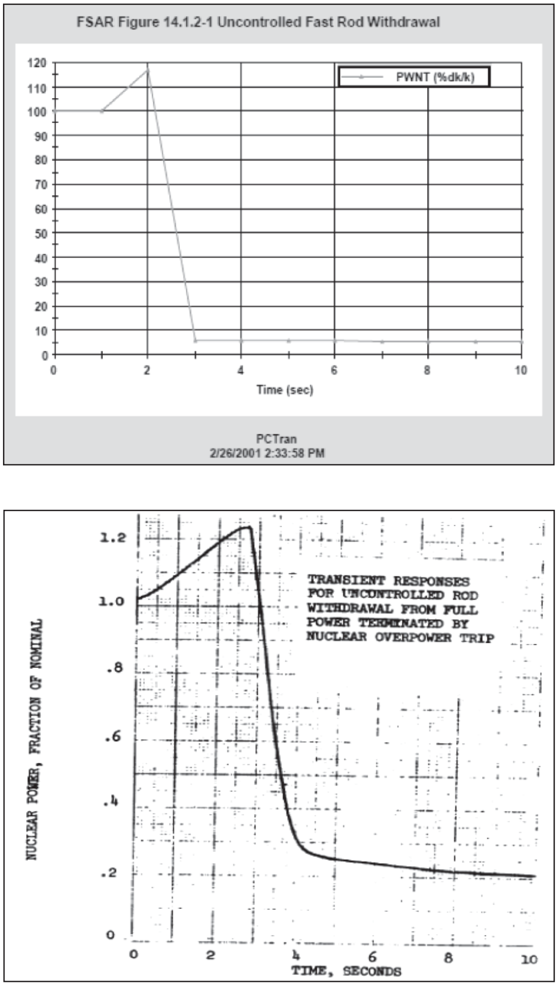

Fig. 5: Comparison of PCTRAN simulation and FSAR results for uncontrolled fast rod withdrawal.

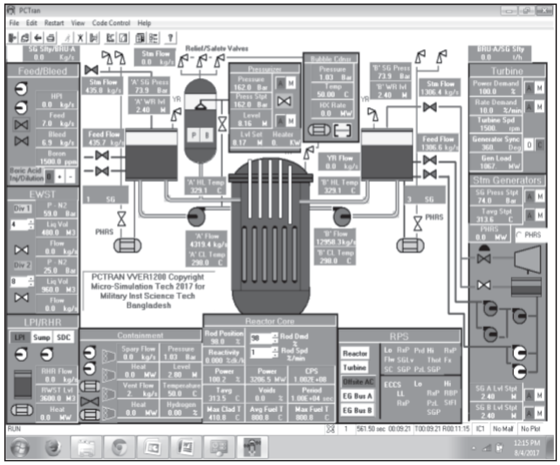

Fig. 6: PCTRAN VVER-1200 Mimic.

Science and Engineering at MIST for educational and training purposes. The screen shot of VVER-1200 model is shown in Fig. 6. Further studies for operational characteristics of VVER-1200 based on kinetics and thermal-hydraulics are in progress.

4. Conclusion

The main purpose of the PCTRAN reactor simulator is educational-to provide a teaching tool for university teachers, professional engineers and students involved in teaching topics in nuclear energy. As well, nuclear engineers, scientists and operators in the nuclear industry may find this simulator useful in broadening their understanding of nuclear reactor kinetics, transient behavior and power plant dynamics. PCTRANis family of codes form a complete teaching platform for nuclear technology.Education and training using the nuclear simulator either PC-based or task simulator is very useful for the capacity and capability building of human resources to support the nuclear power program in the country. In this study the transient behavior simulation has been successfully benchmarked by using PCTRAN based on end-users input data with a reactor point kinetics model.The IAEA's PC based basic principle simulators are excellent tools for early phase training and education on operation and safety aspects of various nuclear power plant technologies. A series of display pages showing different plant views provides the complete system information at one place. The nuclear safety related concepts and phenomena could be easily explained with the simulators along with understanding of valuable technological differences in various designs. .

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