A Dissertation on

Simulation and Comparison of Different Malfunctions of a Pressurized Water Reactor using PCTRAN

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By

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CERTIFICATE

This is to certify that the Major project (AP-811) report entitled "**SIMULATION AND COMPARISON OF DIFFERENT MALFUNCTIONS OF A PRESSURIZED WATER REACTOR USING PCTRAN**" *is a bonafide work carried out by* **Mr. Ankit Hanumant Janbandhu** *bearing Roll No.* **2K14/NSE/06**, *a student of Delhi Technological University, in partial fulfilment of the requirements for the award of Degree in* **Master of Technology** *in* "**Nuclear Science & Engineering**"*. As per declaration of the student, this work has not been submitted to any university/institute for the award of any degree/diploma.*

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DECLARATION

I declare that this written submission represents my ideas in my own words and where others' ideas or words have been included, I have adequately cited and referenced the original sources. I also declare that I have adhered to all principles of academic honesty and integrity and have not misrepresented or fabricated or falsified any idea/data/fact/source in my submission. I understand that any violation of the above will be cause for disciplinary action by the Institute and can also evoke penal action from the sources which have thus not been properly cited or from whom proper permission has not been taken when needed.

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Ankit H Janbandhu.

Abstract

Nuclear power has the characteristics of large unit capacity and high security requirements, therefore they are prone to have an impact, and even threaten the safe operation of the power grid since the nuclear power plants are connected to the grid. The nuclear power industry relies more heavily on theoretical analysis of design and safety features than does any other high-technology industry. Before the Three Mile Island accident, much of the safety analysis of commercial reactors focused on a hypothetical accident involving the rupture of a large pipe supplying cooling water to the reactor core. The design basis loss of coolant accident was thought to be worse than any event that would ever happen. Water and steam would be expelled rapidly at the break and the core would be left temporarily uncovered and poorly cooled.

Transient analysis of a pressurized water reactor (PWR) using the Personal Computer Transient Analyzer (PCTRAN) simulator was carried out. PCTRAN analyses consists a synergistic integration of a numerical model i.e. a full scope high fidelity simulation system which uses point reactor neutron kinetics model and movable boundary two phase fluid models to simplify the calculation of the program, so it could achieve real-time simulation on a personal computer. Here various types of malfunctions such as Inadvertent Rod Insertion, Loss of coolant Accident with Turbine Trip, Fuel Handling accident in Auxiliary Building, Loss of Coolant Accident with Locked Rotor, and Turbine Trip were simulated.

Furthermore, all these malfunctions were carried out at 100% power at the end of cycle and at 75% power at the end of cycle. Again, each malfunction has been compared in different power levels which are very useful for concluding, so that we could know that at which power level the power plant should be operated and is least affected by the above malfunctions. By performing these simulations, the upcoming fault analysis has already been done so that these can be prevented during the practical conditions and can ensure the safe and continuous working of the plant.

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Chapter - 1

Introduction

Nuclear power is clean, and has strong supply capacity. The development of nuclear power is very necessary to meet the current requirement of power. Since the current power system is largely based on thermal power plants due to which there many environment issues such as global warming. Also the quantity of coal is decreasing every day, so there is a need to have an alternate energy supplying system which can fulfill the increasing power demand of the world.

However there is a difference in opinion about the nuclear power generation. Some think that there is a need of nuclear power generation and it should be expanded, recognizing that the risks are not so devastating and others consider that this way of power generation should be abolished as it can be dangerous to humans and other living things. Moreover, things related to nuclear power generation are simply considered difficult to understand because it is heavily based on science.

Advanced nuclear reactors are very popular because of their passive safety components. Advanced reactors have different features, such as, reduced capital cost and shortened construction time, a simplified design which is more rugged, having higher availability and a longer operating life, reduced possibility of core melt accidents, higher burn-up reducing the need to refuel and the volume of waste produced, and many passive or inherent safety features were added which does not require any control or operational intervention to avoid the accidents during malfunctions. Hence, it is necessary to evaluate the safety performance of nuclear reactors.

Important part in nuclear power generation is the reactor which differentiates between the different nuclear power plants and the rest arrangement is same as the thermal power plants. In pressurized water reactor, boiling is suppressed in primary coolant system and steam is generated by the steam generator in the secondary coolant loop. The core of PWR consists of 200 tube assemblies containing ceramic pallets consisting of enriched uranium dioxide. The power is controlled by the insertion of control rods.

Nuclear power has large unit capacity, high security requirement, and restriction on fuel cycle, due to these the nuclear power plant connecting the grid is bound to have faults. The nuclear power generation depends more heavily on theoretical analysis for designing and safety features than any other power generating technology. Before the Three Mile Island accident, much of the safety analysis of commercial reactors was focused on a hypothetical accident which consists of the rupture of a large pipe supplying cooling water to the reactor core. The design basis loss of coolant accident was considered as the worse than any event that would ever happen. Water and steam would be ejected out rapidly from the breaks and the core would be left temporarily uncovered and poorly cooled.

In the last decade it has seen that there is an increase in the use of three-dimensional Computational Fluid Dynamics (CFD) codes, which can predict the steady state and transient flows in the nuclear reactors by focusing on the factors like pressurized thermal shocks, coolant mixing, and thermal striping cannot be predicted by one-dimensional system codes with the required accuracy and spatial resolution. A CFD code has models for simulating turbulence, heat transfer, multiphase flows, and chemical reactions. Such models have to be checked before they are applied be used in nuclear reactor safety applications. A must validation is performed by comparing model results against measured data. However, in order to achieve a reliable model result, CFD simulations for validation purposes must satisfy a strict quality criterion according to the Best Practice Guidelines.

For the purpose of nuclear safety, a tool has been used in many nuclear applications. The criticality safety index (CSI) was developed by Nuclear Regulatory Commission (NRC) as a number assigned to a package type container containing fissile material, so that there is control over the collection of containers.

The occupational safety index (OSI) was developed by the process industry, so that managers and workers can monitor their safety performance and give a warning signal whenever there is a probability of errors.

A safety index had been developed by Korea Atomic Energy Research Institute (KAERI) so that it can quantify plant status. It is only to concern about the reducing information loss resulting when simplifying the safety state of the plant into a binary state. But none of these were considered as the safety to nuclear power plants

The Overall Sustainability Index was evaluated using a Fuzzy Logic tool box (Matlab), as proposed by Lotfi A. Zadeh, which has the ability to deal with complex concepts, which are not responsible to a straightforward quantification, such as sustainability. Another important aspect of fuzzy logic is that it uses lexical variables, which means computation using words.

By improving the safety performance of nuclear reactors with the use of passive safety system instead of active safety systems, improvement to environment sustainability index can be achieved. Therefore, the safety performance of a conventional reactor which uses active safety systems, such as a current PWR was compared with an advanced reactor which uses passive safety systems, during a designed accident, such as Loss of Coolant Accident (LOCA). This was conducted using the PCTRAN.

A nuclear power plant is a very complex system with a large number of built-in regulators and security systems due to which it almost impossible to find by reasoning that what can occur due to plant changes, different transients in the system. To avoid interruptions in the power production it is therefore necessary to know that what will happen to the system if any fault occurs before the practical implementation and also need to take measures to rectify it. Simulation is the process by which it is predicted that how will system behave under different operating conditions in a nuclear power plant. If a simulating model is used which can describe the real system, one can do different tests without the risk of disrupting the power generation and harming components.

PCTRAN is a reactor transient and accident simulation software program that operates on a personal computer. The windows based graphic user interface [GUI] has completely revolutionized the simulation technology. The transient change characteristics of Nuclear power plant key parameters under various conditions were analyzed. The fault touching the nuclear reactor protection system can cause the reactor shutdown.

Chapter - 2

Literature Review on Pressurized Water Reactor

2.1 Introduction

Many different systems are there in a nuclear power plant to generate electricity. To generate electricity there are different functions such as monitoring of a plant parameter and controlling of the main reactor. In this chapter we will discuss about components used in pressurized water reactor and their purpose.

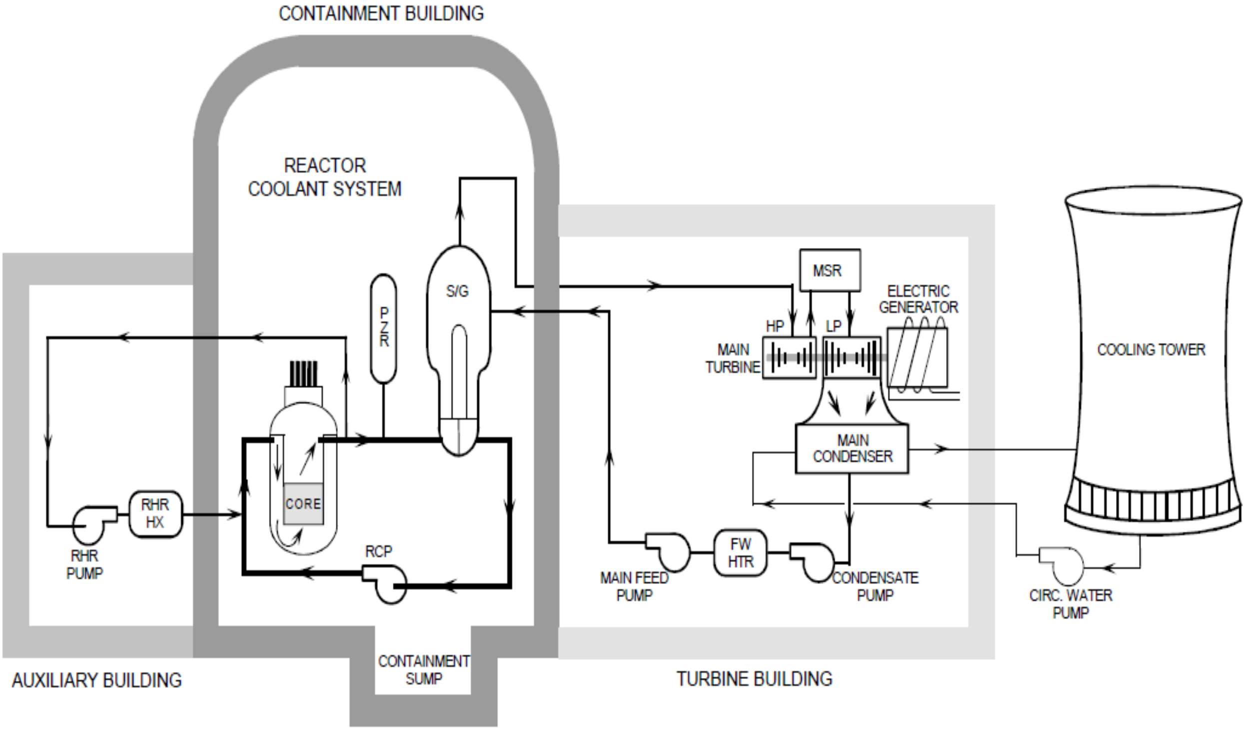

Figure 1 Schematic diagram of pwr

For converting the heat generated in the fuel into electrical power, two major systems are used. The primary system is transferring the heat from the fuel to the steam generator, and the steam from the steam generator is transferred to the main turbine through the secondary system and hence the electricity is produced. As the steam passes through the low pressure turbine, the steam is directed towards the main condenser which removes the excess heat from the steam, as there is cool water flowing through the tubes of the condenser. Further, the condensed water is pumped back to the steam generator for reuse.

The primary system which is also called as Reactor Coolant System consists of the reactor vessel, the steam generators, the reactor coolant pumps, and a pressurizer. The basic function of the reactor coolant system is to transfer the heat from the fuel to the steam generators and it is used to contain fission products.

2.2 Reactor Vessel

The reactor vessel is a cylindrical vessel in shape with a removable top head for refueling of the reactor. There is an inlet nozzle and an outlet nozzle for cooling system. Manganese molybdenum steel is used for the construction of reactor vessel and stainless steel is used to build the surfaces which came in contact of coolant, so that the corrosion resistance will be higher.

The core barrel has the fuel which is present inside the reactor vessel. At the bottom of the core barrel, there is a lower core support plate on which the fuel assemblies are present. The core barrel and all the internals hang inside the reactor vessel.

The reactor coolant enters the reactor vessel from the inlet nozzle and goes to the core barrel and then the core barrel forces the water to flow downward in between the reactor vessel wall and the core barrel. When the coolant reaches at the bottom of the reactor vessel, the flow is turned upward so that it can pass through the fuel assemblies. As the coolant flows through the fuel, the heat produced due to fission is removed. Now the hot water goes on to the steam generator through outlet nozzle.

2.3 Reactor Coolant Pump

The reactor coolant pump is used to provide force to the primary coolant so that it can absorb the heat generated due to nuclear fission process. The circulation of primary coolant will still be there if the pump is not used, but due to lack of pressure the removal of heat will not take place. But, when the plant is not critical natural circulation will be enough to remove the heat.

From the outlet of the steam generator, the coolant enters the suction side of the pump. At the discharge of coolant pump, the reactor coolant pressure will be more than that in the inlet side. The reactor coolant will further enter into the reactor vessel as soon as it leaves the pump and more heat is collected by the coolant as it passes through fuel assemblies. This hot coolant is now sent to the steam generator.

Motor, hydraulic section and the seal package are the major components of the reactor coolant pump. The motor is a large, air cooled electric motor. To provide the required flow of coolant, so that heat can be removed high power motors are used having ratings from 6000 to 10000 horsepower. Seal package is present between the motor and the hydraulic section, so that the water would not leak to the shaft. The hydraulic section of the pump is the impeller and the discharge volute.

2.4 Steam Generator

The reactor coolant flows from the reactor to the steam generator where the hot coolant flows through number of tubes. The feedwater, which is also called as secondary coolant flows outside of those tubes, so that it can absorb the heat from primary coolant. Since the heat is absorbed by the feedwater, it boils and then the steam is formed.

2.5 Pressurizer

The pressurizer is used in the reactor coolant system for controlling the system pressure. Electrical heaters, pressurizer spray, power operated relief valves, and safety valves are used to control the pressure. The pressurizer operates with a mixture of steam and water in equilibrium. When the pressure is deviated from its desired value, the various components will actuate to bring pressure back to its normal operating value. The main cause due to which the pressure is not in its normal value is because of change in reactor coolant temperature. When the temperature of coolant system is increased, the coolant density decreases so that water can take large area. As the pressurizer is connected to the reactor coolant system through the surge line, therefore the water will goes up to the pressurizer and due to this the steam in pressurizer will be compressed and there is increase in pressure. When the coolant temperature decreases, the water will be denser and acquire less area, due to this the pressure in the pressurizer will decrease. Whenever the pressure will increase will increase or decrease, the main purpose of the pressurizer is to bring the pressure back to its normal set value.

For example, when the pressure goes above the desired point which is set in pressurizer, the spray line will allow the same amount of cold water from the reactor coolant pump to be fed into the steam space. Then the cold water will convert the steam into water due to which the pressure will be reduced. Again, if the pressure increases furthermore, the pressurizer valves will get open and steam will be dumped into the relief tank and if somehow the valves would not opened up then the safety valves will lift and discharge it into relief tank.

The relief tanks are large tanks which contain water with atmospheric nitrogen. Discharged steam from the relief or safety valve is condensed by the water and the nitrogen is used to prevent the hydrogen so that it could not create any explosive environment.

Whenever the pressure starts to decrease, the electrical heaters will come into action and they get energized. Due to this more water will be boiled and converted into steam and therefore the pressure will increase. Further, the pressure continues to decrease, and reaches a certain set point, than the protection system will trip the reactor.

The main steam system and the feedwater system are the two major secondary system of pressurized water reactor.

The main steam system of the steam generator starts at the outlet. The steam first goes to the high pressure main turbine and then the steam goes towards the moisture separator and reheaters. In this, the steam is dried with moisture separators and it is reheated by using another steam as a heat source and this steam goes further to the low pressure turbines. After passing through the low pressure turbines, the steam goes to the main condenser, which is operating in vacuum to allow the greatest removal of energy by the low pressure turbines. Then by circulating the water through the condenser tubes steam is condensed.

In the feedwater system, the condensed steam gets collected into the hotwell present in the main condenser. Condensate pump increases the pressure of the water inside the hotwell and then, it goes through a cleaning system to remove the impurities of the water. If it is not removed, the heat transfer capability of the steam generator will be affected. The condensate then passes through some low pressure feedwater heaters and due to this the condensate temperature will be increased. The condensate then moves towards the main feedwater pump due to which the pressure of the water increases to a point so that it can go into the steam generator. The moving liquid rate of the feedwater is controlled as it moves in the steam generators. The feedwater shortly passes over a fit of high charge feedwater heaters, which are livid by extraction fume from the high pressure turbine.

Chemical and volume control system (CVCS) is known as the support system in the reactor coolant system. Purifying the coolant system using filters, boron addition and removal, maintaining the pressurizer level at setpoint are the main function of the system.

The heat produced during fission is removed by the reactor coolant and conveyed to secondary coolant. This coolant is further boiled and is conveyed to main turbine.

Heat produced due to decay of fission product after the reactor shutdown is also high and is sufficient to cause damage if not removed. So, the system must be capable of conveying the heat into the atmosphere even after the shutdown of the plant. During the maintenance, the pressure and temperature of the reactor coolant system should be low.

2.6 Emergency Core Cooling Systems

The two main purpose of the emergency core cooling system are as follows

1. To provide core cooling for minimum fuel damage
2. To provide extra neutron

The former can be attained by inserting maximum amount of cold and borated water inside the reactor coolant system. The later can also be achieved by same borated water so that there are extra neutrons for keeping the reactor in shutdown condition. The cold and borated water source is known as the refueling water storage tank. To act the function putting maximum amount of borated water there are four systems which are used in the emergency core cooling system. The high pressure injection system, the intermediate pressure injection system, the cold leg accumulators, and the low pressure injection system are used for converting high pressure to low pressure. These system should be able operate even when the supply of the plant is lost. Therefore, they are powered with the emergency power system i.e. from the diesel generators.

Pumps are used by the injection system which is at high pressure for volume and chemical control system. The system automatically starts to take water from the storage tank when the emergency signal is received. At the time of emergency water is provided by the high pressure system so that the coolant pressure will remain high. For this kind of emergencies, intermediate pressure injection system is designed so that the pressure should be high enough.

Large amount of borated water with pressurized nitrogen are present in the tanks, when the pressure of the primary system goes below a certain set point, the borated water will be out of the tank due the force from the nitrogen and goes directly to the reactor coolant system. The tanks are designed in such a way that they will provide the water to the reactor coolant system at the time of emergencies i.e. when the pressure of the primary drops very quickly, like during large primary breaks.

The low pressure injection system used for residual heat removal is designed in such a way that it should inject the water from the refueling water storage tank to the reactor coolant system, when the pressure decreases very rapidly such as during large breaks. The residual heat removal system has qualities which permit to take water from the containment sump. This water then goes through the heat exchanger for cooling purpose and then back to the reactor. When the primary system breaks and the storage tank goes empty this method is used for cooling. This is method is known as the long term core cooling or recirculation mode.

The reactor coolant system is located inside the containment building and it is designed in such a way that it should withstand the temperatures and pressures from fluids which are released at high energies. For long interval of time this exposure will certainly damage the building so, if any damage occurs in the primary system, the released coolant will be radioactive which is dangerous. If there are any cracks, there will be a threat to the environment as the radioactive material will come out of the building. Steel liners are used for covering the inner surface of the containment building to avoid any environmental threat. The liner act as a vapor proof membrane so that no gas can escape from the cracks developed in the building.

Two systems are designed for reducing the containment temperature and pressure when there is an accident in building. Fan cooler system and the containment spray system, former use heat exchanger for cooling by circulating air and later is used to reduce temperature and pressure of the building. The containment spray pump will take water from the refueling water storage tank and pump it into spray rings which are present in the upper part of the containment building.

The water droplets which are much cooler than the steam will absorb the heat and the steam will be condensed. Therefore, there will be reduction in the pressure inside the building and temperature will also be reduced. The containment spray system has the capability to take water from the containment sump when the refueling water storage tank goes empty.

Chapter - 3

Literature Review on PCTRAN

3.1 Introduction

IAEA has contracted Micro-Simulation Technology for preparing a series of PC-based nuclear power plant simulation software which can be presented in its annual workshop. Originally there was DOS-based software which than upgraded to Windows in 2003 and the PWR model was introduced to us. In 2009 the two-loop PWR model has been upgraded to include serious accidents which consist of core-melt and containment failure. Radiological discharge source term is produced for offsite dose prolongation and residue study.

The main purpose is to create a desktop simulator and a training substantial available to all, to boost in their approach and training programs for nuclear power. This thing will provide discernment and understanding of the reactor designs as well an understanding of the operational characteristics of the different kind of reactors. An accent is required for basic principles of the operation of each reactor type.

Micro-Simulation Technology (MST) in 1985, the company practiced to the advancement of the AP1000 plant severe accident scrutiny and simulation of small software. It is a personal computer by which the reactor transients and accidents on the authority of the simulation software. PCTRAN is previously used primarily to compile the FORTRAN language, and later on Windows as it is Microsoft's popular Windows platform, so the reworked version of windows is early used only for few simple analysis module interface. In it only a simple output was consistently increased the number of analysis modules, an easy interface but also by the previous development of the DOS interface to today's Windows graphical user interface (GUI). Due to this inception, MST has produced multiple versions of this simulator for different reactor types, so the PCTran software in nuclear power plant consists accidents that may have modeling and analysis, resulting in quicker the accident reasonable prediction, but also it is also used for evaluating the reference nuclear power plant accident analysis.

PCTRAN is a reactor transient and accident simulation software program which works on a Personal computer. Micro-Simulation Technology has been constantly upgrading the performance of pctran and also expanding its capabilities. Numbers of types and plant models have been installed in various countries around the world.

Upgrading in modern 32-bit microprocessors and the windows-based graphic user interface (GUI) has completely reformed the simulation technology. It is now easy as pie to prepare the work for designing and also actual exercise can be done on a desktop computer. Since 1998, the source code of PCTRAN has been changed into Microsoft Visual Basic 6.0. Data input/output are in MS Office's Access database format. Reports and data can be shipped conveniently over all Windows-based software commodities over the entire operating network.

The plant model present here is a generic two-loop PWR with upside down U-bend steam generators with dry containment system. One loop with the pressurizer is shaped differently from the other loop.

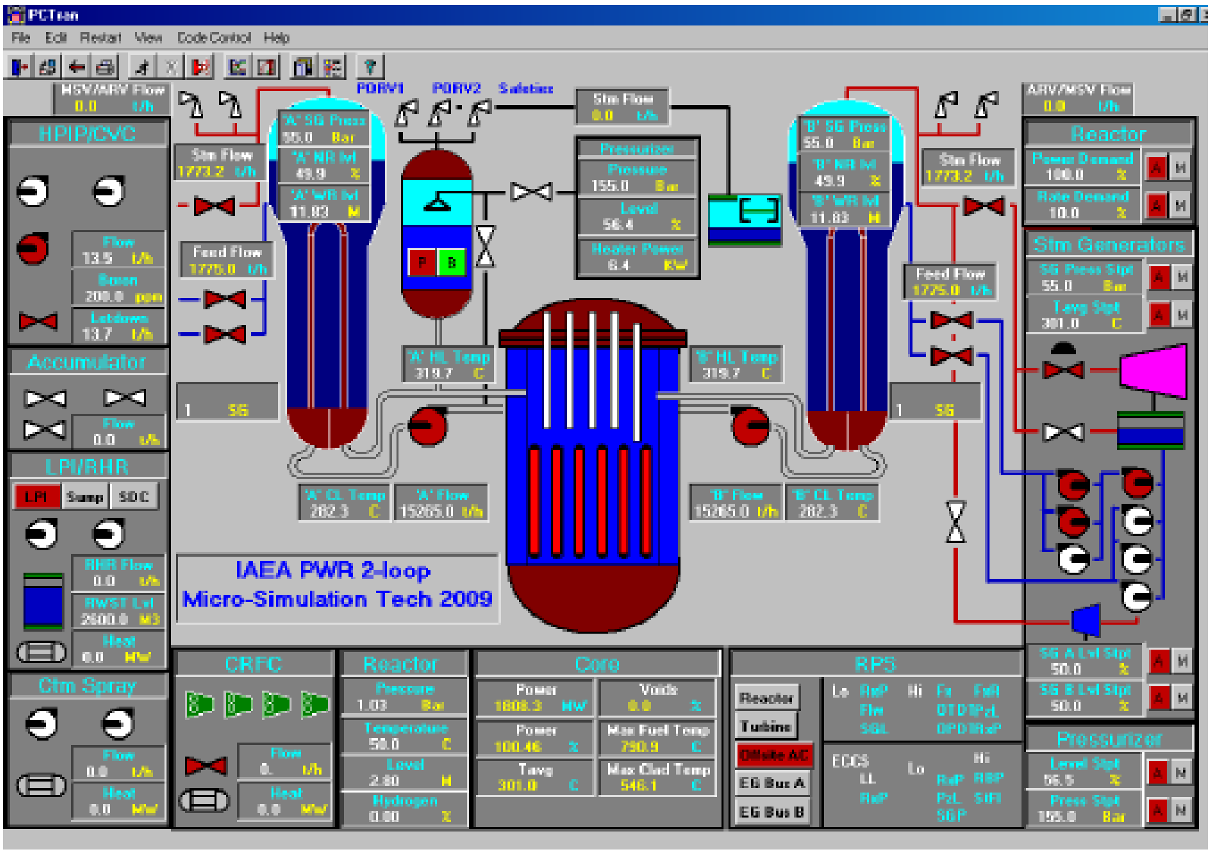

Figure 2- PCTRAN windows mimic

In a PWR's primary coolant system, boiling is abolished and steam is produced by using steam generators in the trivial coolant loops. A pressurizer is used to cultivate the primary coolant in auxiliary-cooled condition and the pressure convenient to a constant. Reactor coolant pumps are used to disperse the primary coolant over steam generators; at which steam is generated at the trivial side to run the turbine.

3.2 Reactor Power Control

PWR has refined automated control systems. The control rod system and dissolved boron regulate the core neutron flux. The Chemical and Volume Control System (CVCS) regulate the primary coolant index and water chemistry. On the trivial side, steam output is regulated by the turbine control faucet and steam dump system. The steam generator water level is regulated through the feedwater system. At the time of the automatic control mode, they endeavor in an integrated way so that transition to balanced conditions will be achieved evenly.

3.3 Pressurizer Pressure Control

All PWRs are constructed to operate adjacent to a constant pressure approximate of 15 MPa all along a power operation. The blunder between the system pressure and set point drives over a controller circuit. If the error is too high, the spray is turned on. When the pressure increases much more at the time of a transient, there are relief valves and safety valves set clear to alleviate the pressure. If the pressure decreases and a negative pressure error come in existence, the proportional heater starts and goes up linearly to its full power. If the difference develops more, the backup heater starts. Detail of the heater action is design definite. The PCTRAN/U model is universal and thus it is calculated for elemental principle instruction only.

3.4 Pressurizer Water Level Control

The charging pumps adopting an error of pressurizer level to the level setpoint which controls pressurizer water level. Backing up on a reactor's arrangement, the level set point can be a constant or prioritized as a function of the unit power. Setback is turned on at the time the pressurizer level goes above the setpoint. The charging and setback system also curbs the reactor coolant's chemical architecture. Therefore, it is also called the Chemical and Volume Control System (CVCS). When the pressurizer level is very much low, the setback is deserted and the heaters are turned down.

3.5 Steam Generator Control

3.5.1 Steam Header Pressure Control

The steam header pressure is restrained at a set pressure by the use of turbine control valve. It can either be a continual or programmed as an affair of the unit power lying upon the reactor's architecture. The Steam Dump System curbs the turbine circumvent valves' opening. When the pressure is high, there are eco dumps and safety valves set to comfort over-pressure in the steam lines.

3.5.2 Steam Generator Water Levels and Feedwater Control

At the time of normal operation, feedwater pumps gives water to the steam generators. The feedwater control valve is coordinated by the addition of two errors: steam generator water level analogous to the level set point, and feedwater to steam flow imbalance. By a proportionalplus-integral controller, the valve adjusts the feedwater discharge until any conversion is balanced and the errors reduced. When the prime feedwater pumps are not working, there are common turbine and motor driven ancillary feedwater pumps. The drivers will commenced them on a low water-level signal or manually.

3.6 Reactor Protection System

At any time the reactor's performing parameters outpace certain characterized safety limits; all control rods are discarded by gravity into the core to abolish the chain reactor. The following trip actions are common for a PWR:

1. High reactor pressure or pressurizer water level
2. High neutron flux
3. Over-temperature delta-T
4. Over-power delta-T
5. High RC outlet temperature
6. Low reactor pressure and/or pressurizer water level
7. Low SG water level
8. Low loop or core flow
9. Containment pressure

The over-power delta-T trips and over-temperatures are the temperature differences amid the reactor coolant inlet and outlet for core preservation. Liquid boron injection is used to add negative reactivity when all rod insertion functions break down.

3.7 Emergency Core Cooling Systems

PWR is armed with superfluous trains of ECCS for core heat elimination at the time of difficulty. They are mainly consists of:

3.7.1 High Pressure Safety Injection (HPSI) System

As it have superfluous trains of centrifugal pumps that works on compulsion diesel power and works on high operating pressure. It is initiated on a low reactor coolant pressure and low pressurizer level signal, or high building pressure signal. The purpose is to give a artificial coolant loss on a small LOCA after the as usual charging system's effectiveness.

3.7.2 Accumulators (ACC)

Tanks are full with pressurized nitrogen and borated water. For a LOCA not rectifiable through the HPSI, valves associating the ACC and the coolant system are opened up to 4 MPa. They will be not working when the pressures are even up so that nitrogen is interrupted from getting in the RCS.

3.7.3 Low Pressure Safety Injection (LPSI) System

It contains unnecessary trains of centrifugal pumps which have to be initiated on Safety Feature Actuation System (SFAS) signals. Their valve head has to be remark lower than the HPSI's. But the flowing rate should be much greater. It has a scope to refill the reactor vessel totally ensuing a dominant LOCA to the mark of break. LPSI usually catches its consumption from the Borate Water Storage Tank (BWST).If the water is about to drained, the driver will shift the suction from the containment sump and run from the heat exchangers ahead of adding back to the reactor. For a few plants the cloned pumps used for LPSI and then they are used for decay heat removal at the time of the cooldown interval after a usual closedown. The heat exchangers and the piping liners are associated to the Residual Heat Removal (RHR) or decay heat removal system and therefore is not the part of the ECCS.

3.8 Containment System

To avert over-pressure in the containment building following a LOCA, PWR is armed with an emergency fan coolers and a containment spray system. Intake for the spray pumps is taken from the Borate Water Storage Tank (BWST). It can further be diverted to recirculation mode when water supply is dead. Heat exchangers are again used to cut out the heat content of the containment building to outside atmosphere.

3.9 Transients and Accident

This part will show the kind of feedback given by PWR plant to each transient who is given in the simulator menu Input parameters which are useful to a transient will be conferred. In association to the feedback of the major system parameters, appropriate operating actions to alleviate the accident will be told.

3.9.1 Uncontrolled Rod Bank Withdrawal

A rod cluster control assembly (RCCA) withdrawal accident is characterized as an undisciplined inclusion of reactivity to the reactor core created due to withdrawal of RCCAs. It leads to a power expedition. Such kind of transient can be generated by a malfunction of the reactor rod control systems. This could developed with the reactor in either subcritical or at hot zero power (HZP). In PCTRAN, by choosing an initial condition at HZP (#7) and choosing Malfunction #12 for rod withdrawal will run this transient.

3.9.2 Moderator Dilution

This event is earlier described as a Chemical and Volume Control System Malfunction. It is described by an inadvertent addition of unborated water into the reactor coolant system through the reactor makeup part of the chemical and volume control system. This has the reaction of adding reactivity to the reactor. If the reactor is in automated control then the control rods will be inserted in reply to the increase in reactivity. The low insertion limit alarm would alert the operator to start borating the water.

Also, it would be important to separate the unborated water source. In manual control, the power and temperature will until the high pressure scram setpoint is reached in approximately 90 seconds. The dilution accident in this case is actually similar to the uncontrolled rod withdrawal transient. The reactivity rate is much less than what was pretended in that analysis.

In PCTRAN, a moderator dilution transient is started by selecting initial condition #1 and malfunction #14.

3.9.3 Reduction in Feedwater Enthalpy

The reduction in feedwater enthalpy is one more means of bringing up the core power above full power. Such increments are attenuated by the thermal capability in the secondary plant and in the RCS. An example of this is the accidental opening of the low pressure heater bypass valve which deviates the flow around the low pressure feedwater heaters. The lower feedwater enthalpy will process to lower inlet temperatures, which than gives a negative moderator coefficient which will again result in an increment in power.

In PCTRAN, changing the parameter HMFW can simulate a reduction in feedwater enthalpy.

3.9.4 Excessive Load Increase

An enormous load increase circumstance could result from either an administrative infraction such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control. The reactor control system is constructed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase without reactor trip in the range of 15 to 100 percent of full power. They are the ones which can occur with the reactor in an automatic control. The End of Life (EOL) condition consists processing the transient with a heavy negative moderator coefficient means maximum feedback. The Beginning of Life (BOL) case gives a zero moderator coefficient means minimum feedback. Due to the feedback, the reactor reacts more quickly to the increase in demand because of the contraction in core inlet temperature.

The transient is run in PCTRAN by setting the reactor power demand to 110%, with all other controls in auto. Initial condition #1 is selected for the EOL case and #3 is selected for the BOL case. No malfunctions are needed for this transient.

3.9.5 Loss of Reactor Coolant Flow

A loss of coolant flow circumstance can result from an electrical or mechanical failure in a reactor coolant pump or in a fault in the power supply for the pumps. When the reactor is at power, the actual effect of loss of coolant flow is a accelerated increase in coolant temperature. This boost could result in escape from nucleate boiling with consecutive fuel damage if the reactor is not tripped directly.

Two cases are benchmarked, 1 pump trip and 2 pump trip. In each case, the core is considered to be at 102% power with a zero moderator temperature coefficient. The accident is processed in PCTRAN by tripping the RCP on the left for the 1 pump trip case, and by tripping both the right and the left pump for the 2 pump trip case.

3.9.6 Locked Rotor

A locked rotor action is described by the instantaneous seizure of a reactor coolant pump rotor. Flow by the concerned reactor coolant loop is drastically reduced which leads to a reactor trip on a low flow signal. Before the trip, there is accelerated power and pressure increase. A zero moderator temperature coefficient was considered in the process. Malfunction #7 initiates a locked rotor accident. Initial condition #3 was selected for a full power, zero moderator coefficient simulation.

3.9.7 Loss of Normal Feedwater

A loss of feedwater transient can be started by tripping all crucial feedwater pumps, closing all feedwater valves which are in the malfunction list. All would result in a reactor trip on high RCS pressure or low SG level. RCS temperature and pressurizer level will increase due to a reduction of heat transfer until a secondary heat sink is established. This is established by the Emergency Feedwater (EFW) automatically starting on low SG level, or by manually clicking on the EFW pumps.

FSAR describes a loss of feedwater event with the turbine bypass valves not available. Also, only one motor driven auxiliary feedwater pump is available, with a flow rate of 200 gpm. In PCTRAN, selecting malfunction #5 and manually disabling the turbine bypass valve simulate the transient.

3.9.8 Steam Generator Tube Rupture

This accident is started by an entire rupture of a single steam generator tube which will result in a loss of reactor coolant to the trivial side. The loss of index will lead to RCS pressure and pressurizer level will decrease. Prolonged loss of inventory leads to a reactor trip on low pressurizer level. Safety injection directly displaces the reactor trip in array to maintain the pressurizer level. The safety injection signal removes normal feedwater and starts auxiliary feedwater inclusion. The plant trip naturally shuts off steam quantity which is going to the turbine and if outside power is possible then the steam dump valves open allowing steam dump to the condenser. In the process of station shutdown, the steam dump valves will close to rescue the condenser. The steam generator pressure would briskly increase which results in steam liberation to the atmosphere from the steam generator safety valves. The accident is considered to take place with one percent defective fuel rods. In PCTRAN, a SG tube rupture action is initiated by choosing malfunction #10 for an A-side break.

3.9.9 Large Break without ECCS

A cold leg serious accident case was started by disarming the accumulators which are HPI and LPI pumps. The core was briskly exposed and then starts to disappear collapsed and melt along the vessel bottom. The containment building pressure grows briefly when the vessel collapsed. Hydrogen quantity in the containment built grows up but it did not burst because inflated containment pressure which raised the ignition concentration. In the dose mimic the red area which comes inside the melt corium in the reactor cavity floor indicates corium-concrete-interaction (CCI) along with aerosol production in the containment. At this moment we should select Malfunction 14 for the containment failure.

3.9.10 Station Blackout

A station blackout drops on both the offsite AC and onsite diesel power. The only turbine-initiated AFW pump is working to administer feedwater to the steam generators. Here it is also assumed to be not available. Only DC-operated pressurizer is working to assuage the pressure. The steam generators are blain dry which follows by the loss of all feedwater. The elementary coolant pressure goes up to cycle the pressurizer. When the pressurizer level is at the top than the discharge swings into two-phase that breaks the drain tank and pressurizes the containment. Further it is continued than the coolant loss exposes the core to melt and there is vessel failure.

3.10 Theory and Models

The elemental PCTRAN mathematical model was selected for the framework of the PWR project. The consecutive sections characterize the basic theory and thermal hydraulic models in detail. A reduced-node path compared to the full-scope simulator was used to model the primary coolant system. For PWRs, a non-equilibrium pressurizer model handles its natural controls by the spray, heater and relief valves. It also grants sudden changes and acute conditions like "water solid" in the pressurizer and two-phase in the reactor core. The steam generators are shaped as homogeneous equilibrium two-phase volumes. Heat transfer from the primary to the trivial is treated anxiously during both the forced and the natural circulation. Kinetics model for the reactor power calculation has included a containment model. By adding fuel and containment condition simulation in addition to the original model, PCTRAN is a complete nuclear plant simulator.

In PCTRAN, the loop flow model was adjusted to accommodate individual pump trip and possible reversed discharge. Modeling of the important plant control systems and enhanced heat transfer interaction for the steam generators are added.

The fluid discharge rate from a break uses regular critical flow models. A mechanistic model of the coolant flow covering both the forced and the natural circulation provides temperature distribution in the primary coolant. The containment actions are calculated based on a homogeneous equilibrium model with support of non-condensable air and hydrogen. At the time of a severe accident with the core being disclosed to steam for protracted period of time, the core may become overheated. Zirconium in the cladding may react with steam and than the hydrogen will be produced. Simulation of clad failure and hydrogen content in the containment will be added in PCTRAN.

The mass and energy balance equations with co-relations in momentum and heat transfer are solved for all control volumes simultaneously. Transient progress is handled by using Euler integration over every time step increment. Key plant parameters are then displayed graphically on a mimic.

3.10.1 Severe Accident Degraded Core Model

An interpreted model accounting for the temperatures of fuel and cladding has been constructed in PCTRAN. This model can simulate:

1) Thermal power transmitted into the coolant in contrast to nuclear power generated by the core during normal operation
2) Fuel and cladding heatup during accident conditions. Core thermal power

Transient fuel temperature will be determined by the imbalance between nuclear power and thermal power. When the PCTRAN calculated core water level is under the top of the fuel than the temperature of the unprotected sector of fuel rod will increase as the steam cooling is much less compelling than liquid contact at the fuel surface. The peak clad temperature is a considerable concern for reactor safety analysis. It is usually designed not to exceed 1000°C and after that the water-zirconium reaction will becomes possible.

The hydrogen generation is modeled in PCTRAN. The heat and hydrogen production will add to the reactor vessel and containment condition computations.

This interpreted model will display an increase in peak fuel temperature and clad temperature when the core is uncovered and lowering, and a gradual decrease in temperature when ECCS refills the 177 core. If the core is totally reflooded than the cladding temperature will be as same as the water temperature.

From the calculated peak clad temperature and reactor coolant condition, there are published relations for estimating the extent of core damage. In the USNRC's "Severe Reactor Accident Incident Response Training Manual", there is possible core damage as function of core temperature and percent of fuel rods with damaged cladding with maximum core exit temperature. In PCTRAN, the calculated peak cladding temperature was assumed as the maximum core exit temperature, and the correlation was integrated in tabulated form for calculating the percent of fuel rods with failed cladding.

This calculation should be used with care and may be considered as only gross indicators of core damage conditions. Great uncertainties present as a phenomenon of severe accidents, and fuel damage are still under intensive research. However, for an emergency exercise, the calculated amount of core damage may serve as an estimate of fission gas and core radioactivity release source term. Combined with the calculated coolant discharge in the containment, containment dose rate can be estimated. Furthermore, PCTRAN also generates leakage rates through the containment. Therefore, radioactivity release outside the containment and offsite dose can be estimated according to dispersion models.

A severe accident is described as beyond design-basis with compelling core damage and containment failure. A borderline is generally set in probability safety assessment (PSA) Division 2 for top of the fuel (TAF) uncover, and Division 3 for fission product relocation.

The previous models have actually considered core heatup and hydrogen formation by metal-water reaction. The current addition is concentrated in core-melt.

The core is shaped into six vertical nodes. Each one will produce a portion of the decay heat. If the boundary heat elimination rate is less than the core heat than the core node is heated up to the point of melting. Molten fuel may crash into the lower bottom of the vessel. The vessel lower head may be then heated up to the melting point, too. The molten trash may drop into the containment building cavity floor. At the time of the fuel damage procedure, first the fission gas in the clad may cracked out. Later if the fuel and cladding goes on than the degraded fuel isotopes will release also. In inclusion to iodine and noble gases, there are alkali metals, tellurium, barium, cerium, lanthanides, etc. The increased concentration of these radioactive isotopes would find their path from the vessel break, relief valves, and containment leakage into the atmosphere.

In conducting a test run, one can initiate a large break LOCA, hot leg break for the PWR or recirculation line break for the BWR. With automated initiation of ECCS, the core will be reflooded very quickly and no important clad damage is expected. So we deliberately disable all ECCS trains. The water level in the vessel then drops very fast and soon discloses the fuel. One should click the "View" button in the top of the menu bar and choose "Dose mimic".

3.11 PCTRAN Shortcomings and Inadequacies of the Software

It cannot simulate the evaporator water pipe fracture accident and cannot precisely reflect the evaporator by a huge impulse in the dynamic response when not in use during the process parameters such as real-time adjustment. System scalability is another vital indicator of software design. PCTran scalability is not good because the software has to achieve the design features needed for the recognition of other functions that can be composed for a certain room for expansion. As PCTran has modules of data and internal package which makes it impossible to direct the data from external access therefore, PCTran scalability is not good. Again, because the design of PCTran not modular and also for functional improvement and expansion of the system it is difficult. The PCTran software only runs on a single screen and all the parameters and initial conditions by the users and fault simulation parameters are not conducive to train the users to determine the cause of the failure and their own capability to process.

Chapter - 4

Method and Materials

Inadvertent Rod Insertion, Loss of coolant Accident with Turbine Trip, Fuel Handling accident in Auxiliary Building, Loss of Coolant Accident with Locked Rotor, and Turbine Trip in Normal full power and 75% power are simulated by software and been compared.

The simulations for the above malfunctions were performed using PCTRAN, which simulates the Pressurized Water Reactor at different power level. For the reactor to be at 100% power at the end of cycle [EOC], IC1 was activated and for the reactor to be at 75% power at the EOC, IC6 was activated.

Chapter- 5

Result and Discussion

5.1 Simulation of Inadvertent Rod Insertion

For this simulation, Inadvertent Rod Withdrawal [IRW] malfunction was also considered with delay time 100 sec, ramp time 30 sec, and failure percentage 50% and Inadvertent Rod Insertion [IRI] with delay time 15 sec, ramp 25sec, and failure percentage 100%. The values considered were same for both IC1 and IC6.

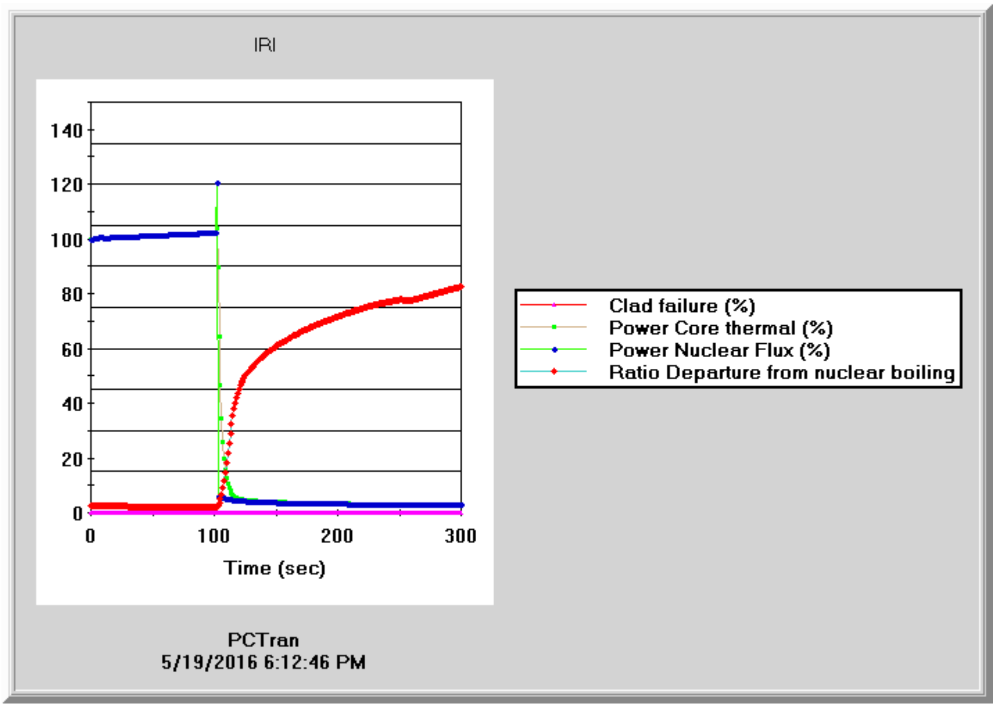

Fig.3 Inadvertent Rod Insertion (100% power)

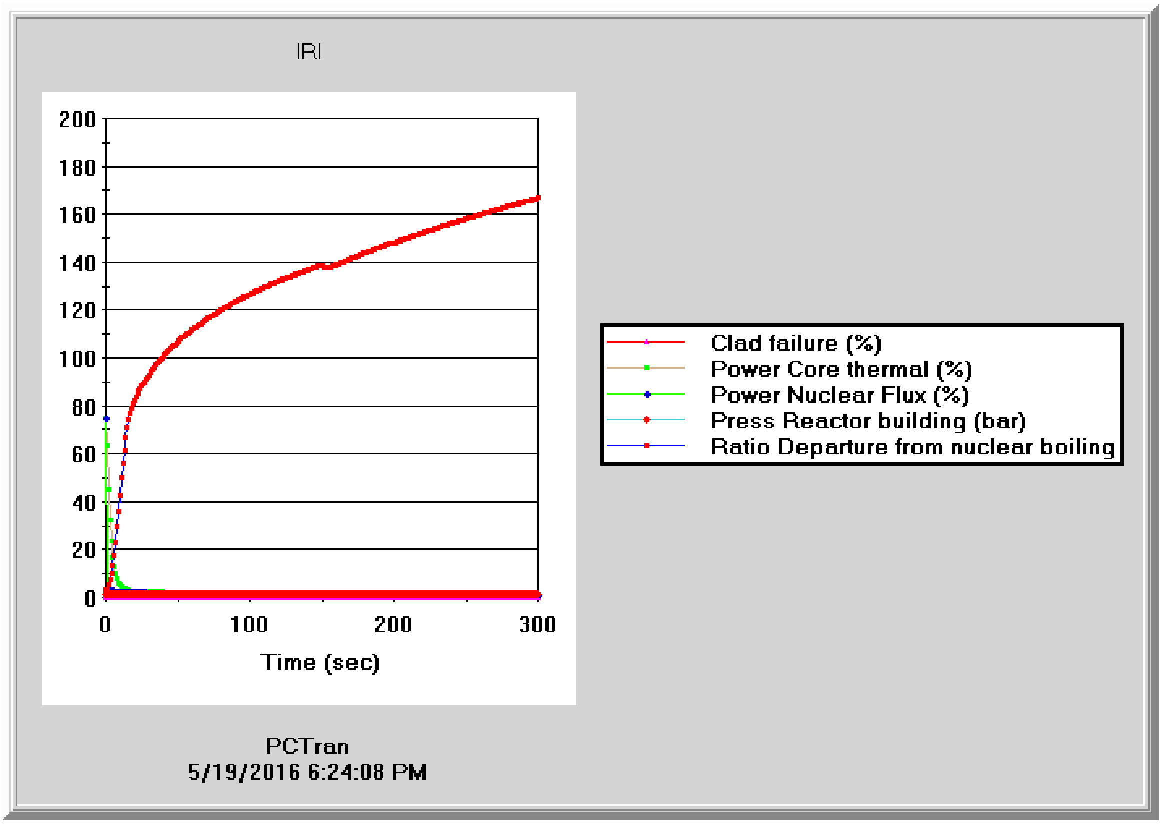

Fig.4 Inadvertent Rod Insertion (75% power)

According to fig.3, due to IRI there is decrease in percentage power core thermal and percentage nuclear flux while percentage clad failure remains constant as a result of low neutron absorption capacity of the material. There were also great departures from nucleate boiling.

As seen in fig.4, same things were occurring but the time interval was very less as compared to the 100% power and also departure is much higher due to nucleate boiling.

Hence, we can say that shut down occur much quicker at 75% power compared to 100% power at EOC.

5.2 Simulation of LOCA with Turbine Trip

For this simulation, Loss of Coolant Accident (LOCA) malfunction was considered the parameters used for moderator dilution are delay time 30 sec, ramp time 20 sec, and percentage failure 75. Turbine Trip with delay time 15 sec, and ramp 20 sec. Anticipated Transient without Scram (ATWS) also considered with delay time 25 sec and ramp time 0 sec. Same values were considered for both 100% and 75% power.

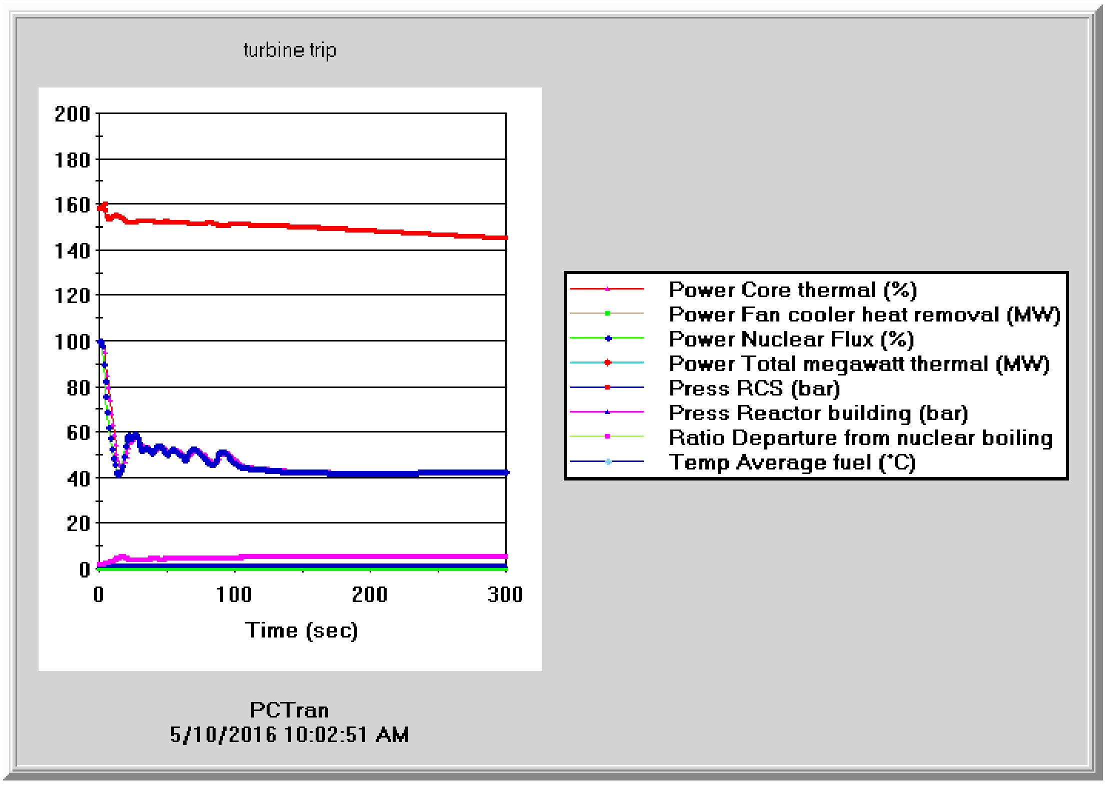

Fig. 5 LOCA with Turbine Trip (100% power)

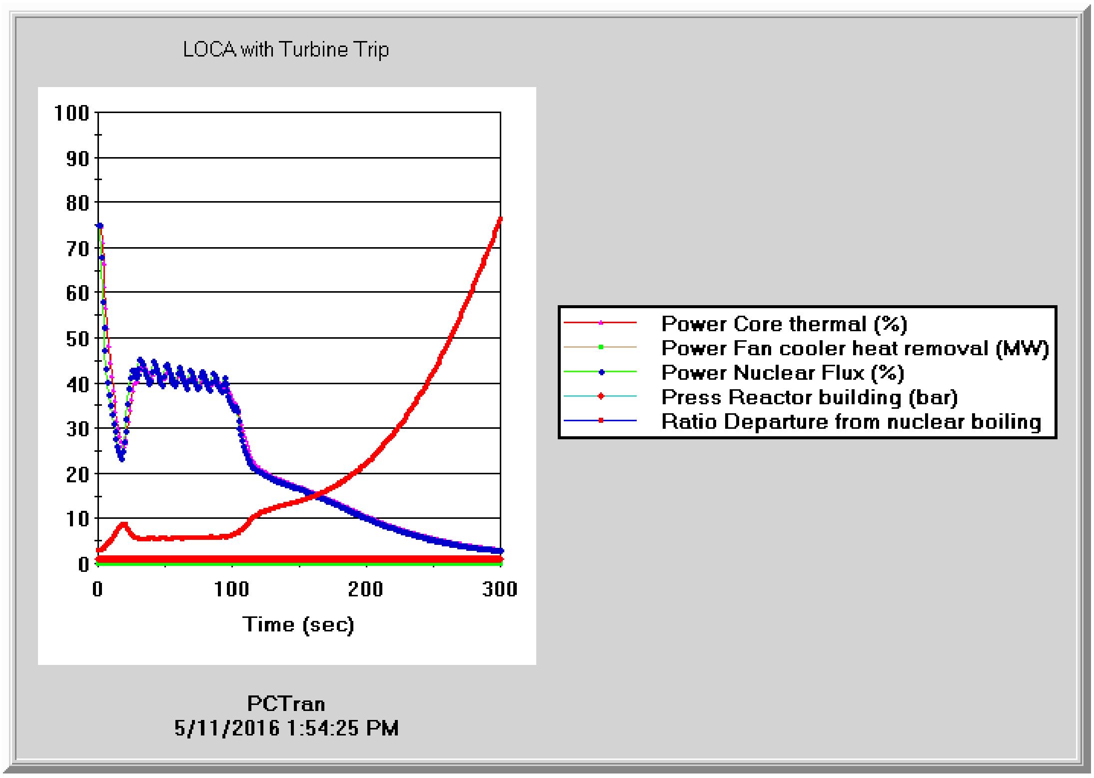

Fig. 6(a) LOCA with Turbine Trip (75% power)

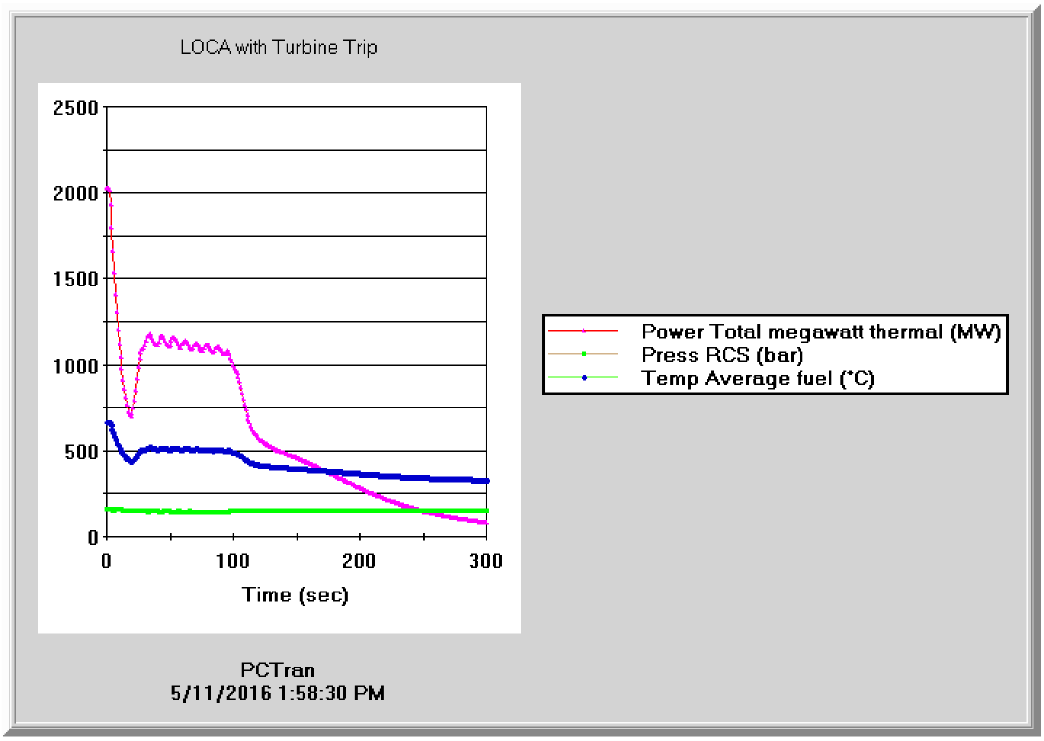

Fig. 6(b) LOCA with Turbine Trip (75% power)

According to fig.5, there is a decrease in total power of the plant and also sharp decrease in pressure due to which there is departure from nucleate boiling. The pressure of the reactor building increased and this is a kind of threat to the plant. Average fuel temperature, Percentage power core thermal, and percentage power nuclear flux all are decreased.

As seen in Fig. 6 (a) & (b), Total power is decreased but the pressure of reactor building is increased rapidly which is a serious threat to the power plant. Hence, operating at 75% power for this malfunction is a serious threat as compared to 100% power for the power plant.

5.3 Simulation of Fuel Handling Accident in Auxiliary Building

In this simulation, FHAIAB malfunction is considered with delay time 10 sec, ramp time, 20 sec percentage failure 50%. The values are same for both the power conditions.

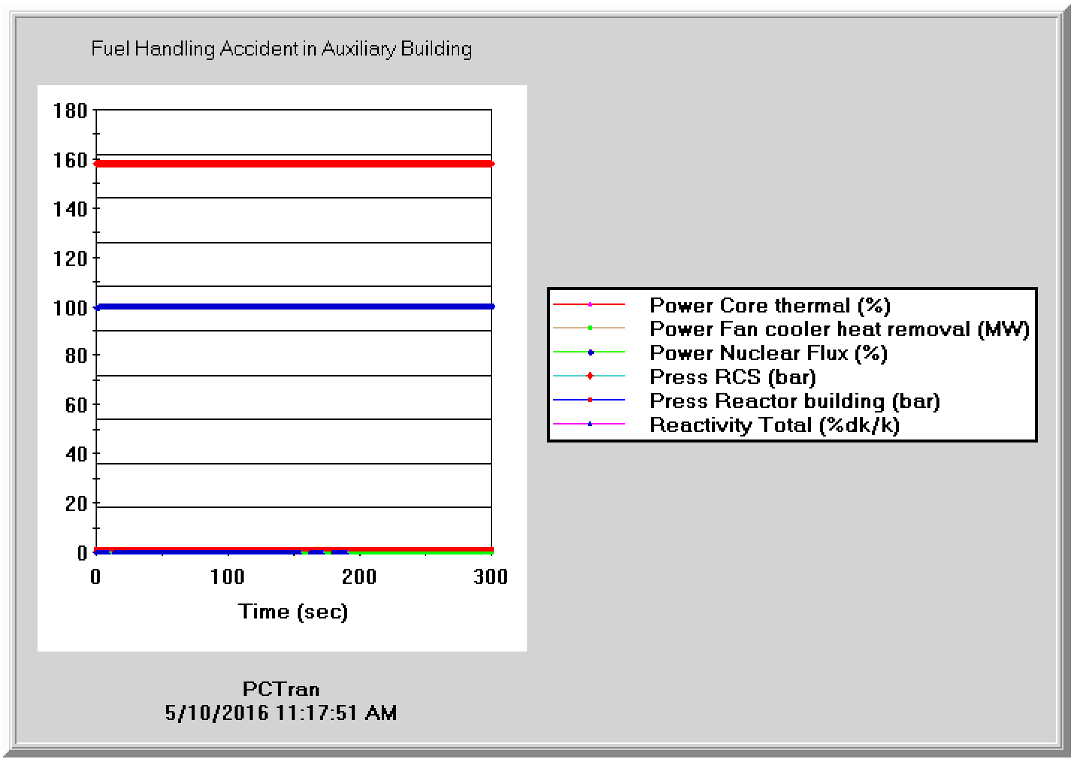

Fig. 7 FHAIAB (100% power)

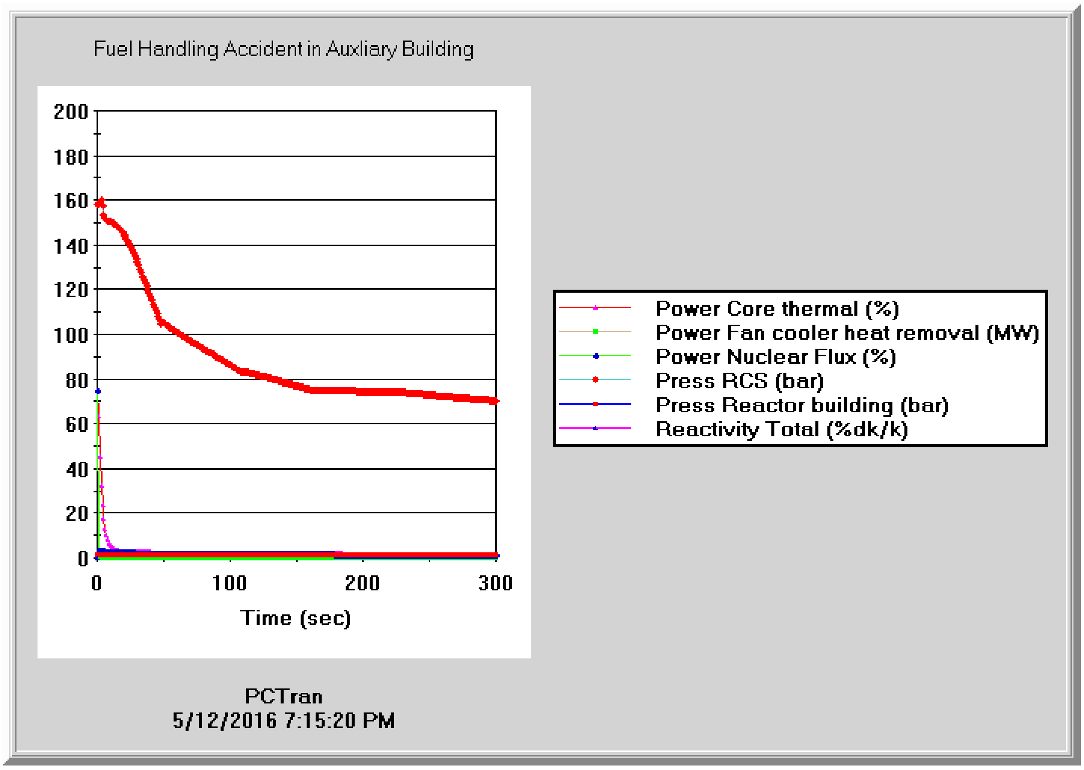

Fig. 8 FHAIAB (75% power)

As the fig.7 shows the straight line, therefore Fuel handling accident has no effect on the operation of Pressurized Water Reactor. But according to fig.8 there is decrease in percentage core power, percentage nuclear flux, and pressure of reactor building. This shows that at 75% power due to FHAIAB fault shut down occurs as power core decreases and there no effect when power is 100%.

5.4 Simulation of LOCA with Locked Rotor

In this simulation, LOCA with delay time 10 sec, ramp time 20 sec, and percentage fault 50% is considered. The Locked rotor malfunction with delay time 10 sec and ramp time 30 sec. The values are same for both power levels.

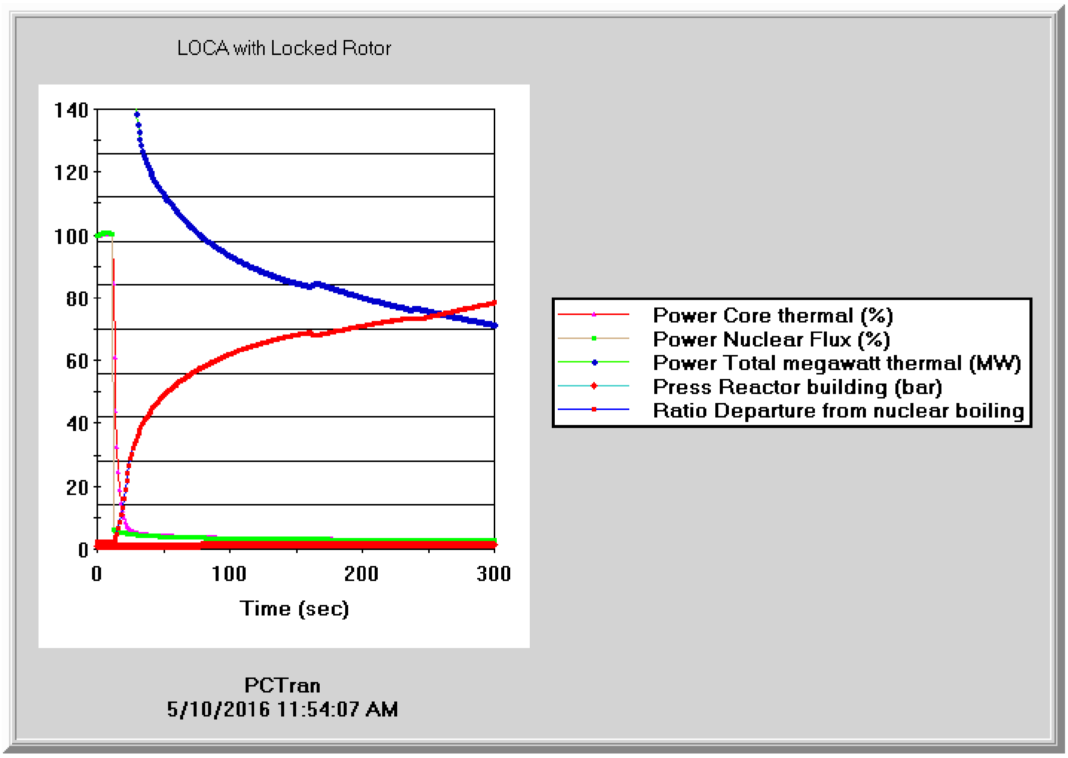

Fig.9 LOCA with Locked Rotor (100% power)

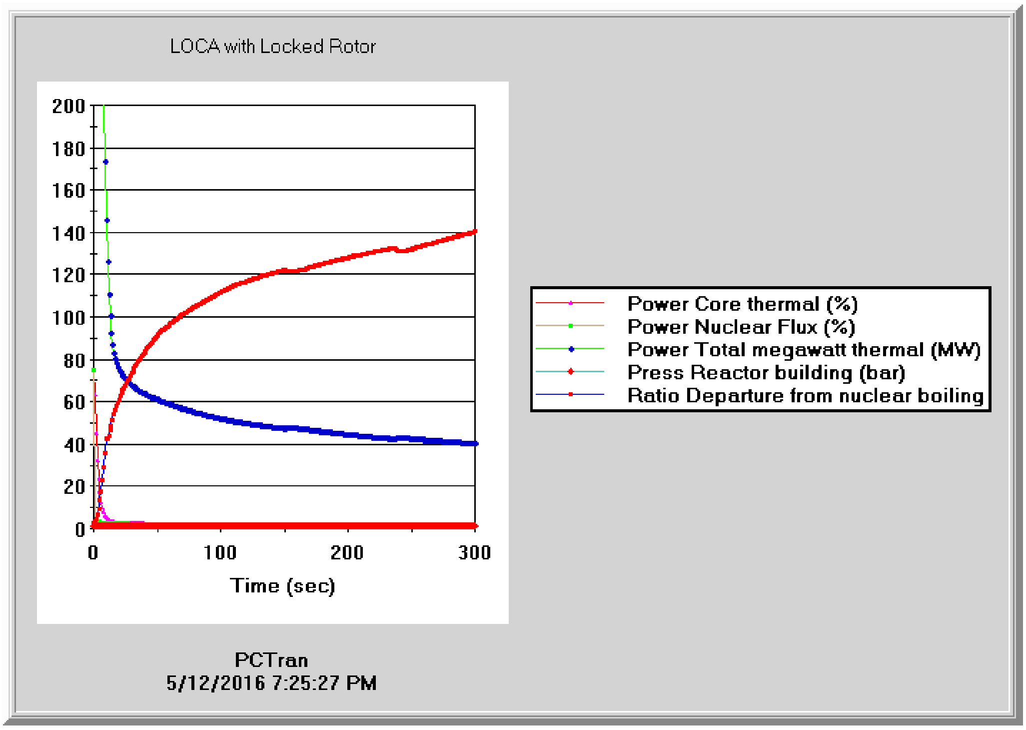

Fig.10 LOCA with Locked Rotor (75% power)

As shown in fig.9, Rotor building pressure increased during this fault and therefore, total power generation decreased. Percentage power core thermal and percentage power nuclear flux initially increased and then decreased drastically. While as in fig.10 same events are occurring but in less time as compared to 100% power.

5.5 Simulation of Turbine Trip

Under this fault delay time is 15 sec and ramp time is 20 sec for both the power levels.

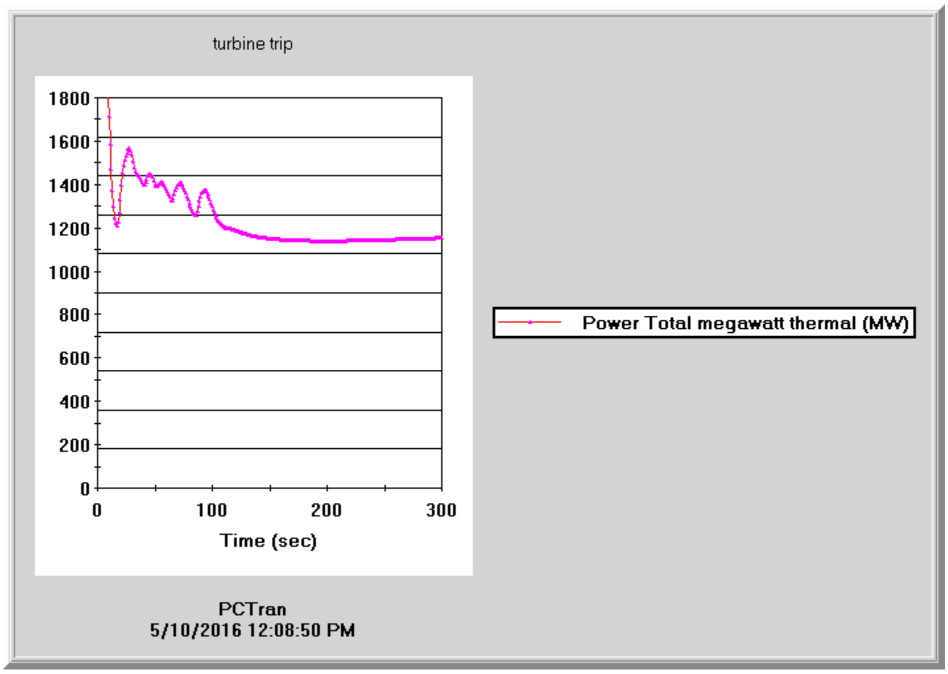

Fig. 11(a) Turbine Trip 100% power

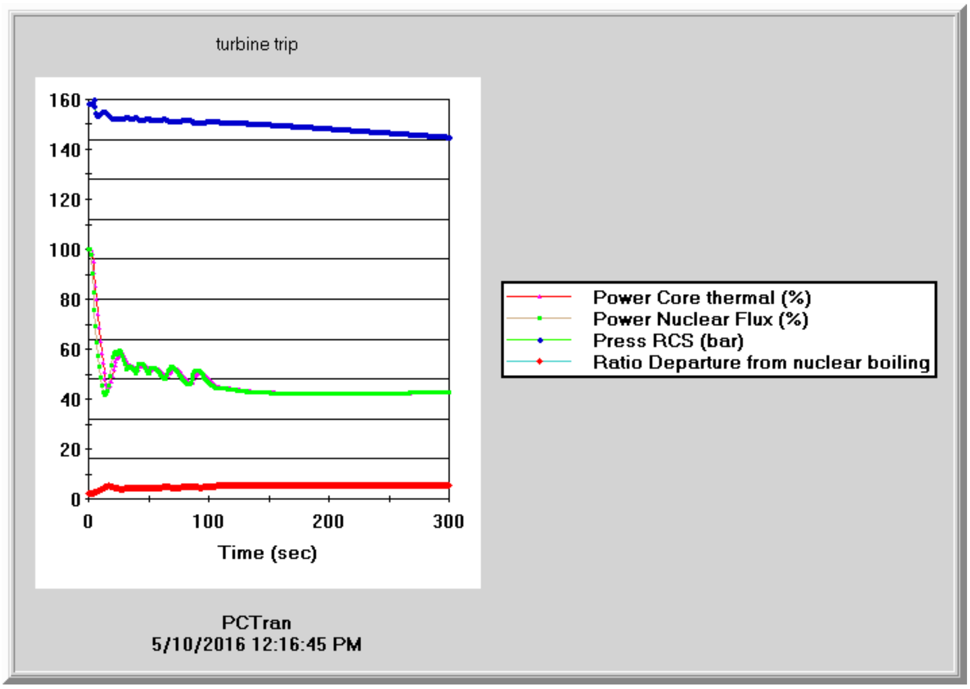

Fig. 11(b) Turbine Trip 100% power

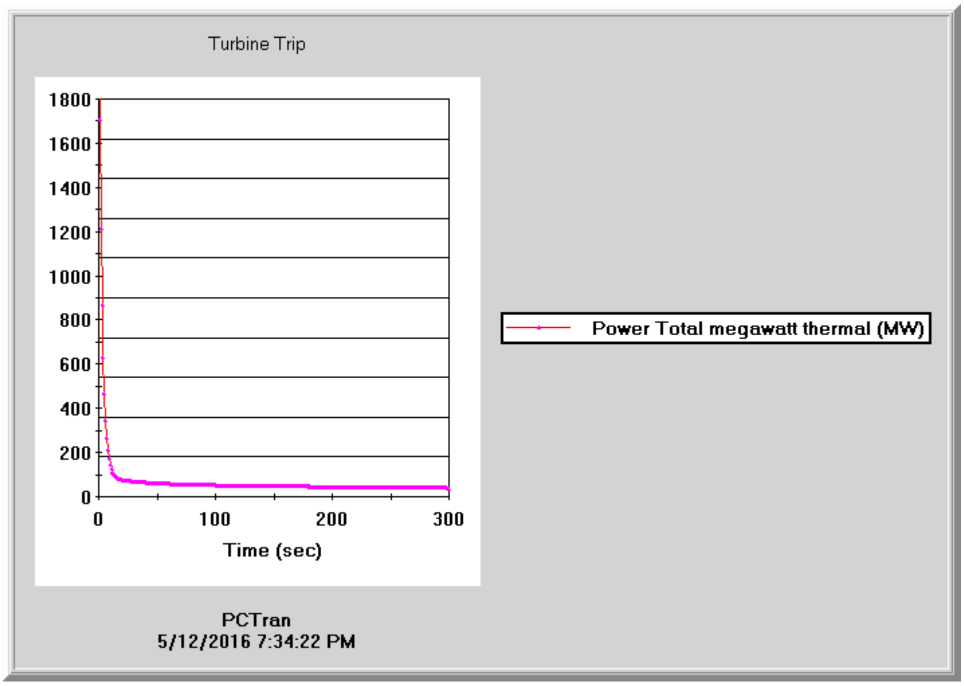

Fig. 12(a) Turbine Trip 75% power

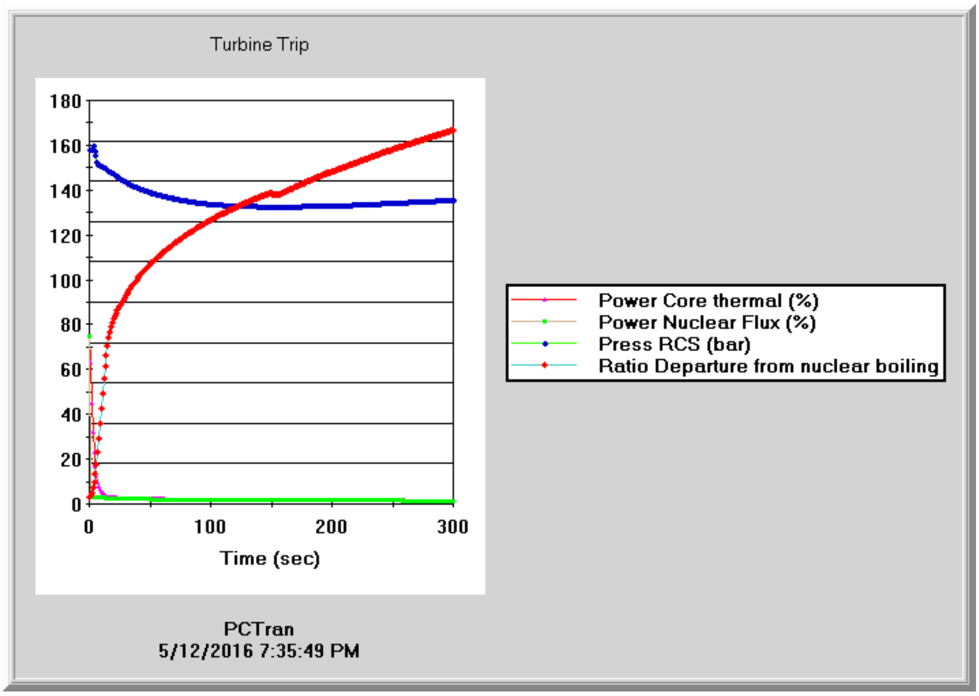

Fig. 12(b) Turbine Trip 75% power

As in fig.11 (a) & (b) total power thermal decreases sharply and then remain constant and the pressure of reactor cooling of system decreases and fault occurs. While in fig.12 (a) & (b) the total power decreases rapidly to zero and departure is there due to nucleate boiling.

Conclusion

Nuclear power plant simulation and analysis of serious accidents is an important research direction in nuclear power by using nuclear power plants based on common computer simulation software. Here, PCTRAN as a model for the nuclear power plant can be used for the PC simulation software analysis and research. PCTRAN can be used to form an emergency response system by using real time values.

Various given malfunctions have been simulated in this report for 100% power and as well as for 75% power at end of cycle and is compared by analyzing the generated graph for different factors after each simulation..

Hence from these simulations it has been concluded that the accidents or malfunctions occurring at 75% power at the EOC is more severe and devastating as compared to power plant operating at 100% power.

As every malfunction has been compared in two different power levels which are very useful for concluding, so that we could know that at which power level the power plant should be operated and is least affected by the above malfunctions. By performing these simulations, the upcoming fault analysis has already been done so that these can be prevented during the practical conditions and can ensure the safe and continuous working of the plant.

References

1. Sunday J. IBRAHIM, Daniel R. EWIM, and Okibe A. EDEOJA, Simulation of Safety and Transient Analysis of a Pressurized Water Reactor using the Personal Computer Transient Analyzer, Leonardo Electronic Journal of Practices and Technologies, 22, January-June 2013 p. 93-105.

2. YOSHIAKI ICHIHARA, SENIOR MEMBER, IEEE, A Perspective on Nuclear Power Generation in the Future Electric Power Industry—For Nonspecialists in the Electric Power Related Industries, IEEE, VOL. 89, NO. 12, DECEMBER 2001.

3. Dr. Li-Chi Cliff Po, Personal Computer Transient Analyzer for a two-loop PWR and TRIGA Reactor for International Atomic Energy Commission, 12 - 23 October, 2009.

4. Michael G. Stevenson and James F. Jackson, Primer on Reactor safety analysis, Los almos science.

5. Reactor Concepts Manual, Pressurized Water Reactor Systems, USNRC Technical Training Center.

6. Lam S. K., Advanced Pressurized Water Reactor Simulator Manual, Vol. 1, 2009.

7. Lam S. K., Advanced Pressurized Water Reactor Simulator Manual, Vol. 2, 2009.

8. Li-Chi Cliff Poa and Juyoul Kim, PCTRAN/KSNP1000 – A PC-based Simulator for KSNP1000, Transactions of the Korean Nuclear Society Spring Meeting Pyeongchang, Korea, May 27-28, 2010.

9. Akinfaderin A., Analysis of Transient Processes in Nuclear Reactor Loops, Atomnaya Energiya, 2010, 80(5), p. 261-263.