Evaluation of an advanced fault detection system using Koeberg nuclear power plant data

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It all starts here

ABSTRACT

The control and protection system of early nuclear power plants (Generation II) have been designed and built on the then reliable analog system. Technology has evolved in recent times and digital system has replaced most analog technology in most industries. Due to safety precautions and robust licensing requirements in the nuclear industry, the analog and digital system works concurrent to each other in most control and protection systems of nuclear power plants. Due to the ageing, regular maintenance and intermittent operation, the analog plant system often gives faulty signals. The objective of this thesis is to simulate a transient using a simulator to reduce the effects of system faults on the nuclear plant control and protection system, by detecting the faults early. The following steps will be performed:

- validating the simulator measurements by simulating a normal operation,
- detecting faults early on in the system

These can be performed by resorting to a model that generates estimates of the correct sensors signal values based on actual readings and correlations among them. The next step can be performed by a fault detection module which determines early whether or not the plant systems are behaving normally and detects the fault. (Baraldi P. et al, 2010)

Keywords: advanced fault detection, PCTRAN simulator, validation, steady state, fault detection, protection and control system, transients

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NOMENCLATURE

ABBREVIATIONS AND DEFINITIONS

AOO	Anticipated Operational Occurrences
ATWS	Anticipated Transients Without Scram
C&I	Control & Instrumentation
DNBR	Departure from Nucleate Boiling Ratio
GCT	Turbine by-pass System (NSSS- CI System)
GDC	General Design Criteria
FDD	Fault Detection and Diagnosis
FDS	Fault Detection System
INSAG	International Nuclear Safety Advisory Group
KKO	Event Recorder System
LOCA	Loss of Coolant Accident
NPP	Nuclear Power Plants
NRC	National Regulatory Commission
NSSS	Nuclear Steam Supply System
PWR	Pressurised Water Reactor
RC	Manual Setpoint

RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant System
SCRAM	Safety Control Rod Axe Man
SG	Steam Generator

DEFINITIONS OF TERMS

Anticipated Operational Occurrences: Conditions of normal operation that are expected to occur one or more times during the life of NPP and includes but not limited to the loss of power to all recirculation pump, tripping of turbine generator set, isolation of the condenser and loss of offsite power.

Anticipated Transient Without Scram: AOO's followed by the failure of the reactor trip portion of the protection system specified in general design criteria 20, because of common-mode failure.

Common-Mode Failure: The result of an event which causes coincidence of failure, states of components in two or more separate channels of redundancy systems leading to the failure of the defined system to perform its intended function.

Design Basis: Information that identifies a specific function to be performed by a structure, system or component of a facility and the specific values or range of values chosen for controlling parameters as reference bounds for design.

Design Basis Accidents: Postulated accidents that are used to set design criteria and limits for the design and sizing of safety related systems and components.

Fault Detection: Detecting the existence of an abnormal and unexpected disturbance in the system. (Cilliers, et al 2011).

Loss of Coolant Accident (LOCA): A postulated accident that results in the loss of reactor coolant at a rate in excess of the replacement capability of the reactor coolant makeup system.

Over-Pressurisation: The condition brought about by pressure exceeding the design pressure of the component by more than 10% in accordance with ASME codes.

Plant Computer System: It provides computational data processing and data presentation service for the plant. Flow maps and instrumentation diagrams may be called up and data logged to allow sequence analysis after an event. (ESKOM, 1985)

Postulated Accidents: Unanticipated conditions of operation which are not expected to occur during the lifespan of a NPP.

Reactor Protection System: The protection system is designed to initiate automatically the operation of appropriate reactivity control system, to ensure that specified acceptable fuel design limits are not exceeded as a result of AOO's and to sense accident condition and to initiate the operation system and components important to safety. It protects the reactor core and the NSSS by monitoring operating parameters and initiating safeguards actions on the detection of abnormal conditions.

Single Failure: An occurrence that result in a component's loss of capability to perform its intended safety function.

CHAPTER 1

Introduction

The control and protection system technology in all other electronic industries has evolved in recent years. In the Nuclear industry, this has never been exploited due to stringiest safety and licensing requirements, hence there has been little implementation of advanced system. There exists several fault detection methods in the electronic industry. Fault detection and diagnosis (FDD) is the process to detect, and isolate faults in a system (Jianping M., et al, 2010). In any nuclear system, the safety is of paramount importance in order to promote public confidence and protect the environment from any undesired incidences.

There exists digital reactor protection system in some generation II nuclear reactor technology type, like the spin line technology (Rolls Royce) and Teleperm XS (Areva), the reactor protection system used in most of generation II is still analog, hardwired circuitry. The analog circuits has common problem such as drift, degradation and component obsolescence. The system which uses algorithm is required not to replace the analog but operate alongside the system (analog) so as to improve the reliability of the plant protection and control system.

The control system in a nuclear power plant is designed to counter any change in transients of the plant condition. The sensors will detect a fault that will change condition and destabilise the plant system, and the control system will change some protection systems to counter the fault such that the plant will become stable again. Some faults are severe in that the control and protection system will not be able to transform the plant condition to stabilise it, especially when the operating parameters are at their operating set-points thus the plant will initiate a safe shutdown condition which is undesirable for generation of electricity but necessary for the safe operation of a Nuclear Power Plant (NPP). This is undesirable because it will take long to bring the plant to full power again.

Different components in a system fail or give faulty outputs due to their characteristics (semi-conductors). In a typical nuclear power plant system, a sensor is installed to monitor any transients that could change the integrity of the plant and will trigger the necessary plant corrective system to counter such undesirable condition. Those sensors can also be subjected to faults and malfunctioning, which leads to system failure hence there is a need for an early detection system in NPP similar to Koeberg. When a fault occurs in a system of a plant whose measurements are used for the control of an industrial process, a corrective action must be promptly initiated since the use of incorrect information by the controller could compromise the correct functioning of other systems, with potential fall-backs, both operating and safety of the plant. In this context, on-line monitoring methods can provide an indication of the health of the sensors and supply an early warning of developing faults. This enables the assessment on the reliability of the measurement and to conveniently plan the sensor maintenance. Additionally, for continuing operation while reparation is performed, the erroneous measurements should be substituted by accurate estimates of the signal’s true values (Baraldi, et al, 2010).

Fault diagnostic system (FDS) is implemented to reduce human error, which may lead to plant accidents and to increase plant efficiency. FDS is regarded as a compensator in control theory. Operators use it to help make informed, timely and correct decisions. FDS are considered for implementation so as to reduce the operators load and to support their decisions in operating the plant (Lee, J.S., et al, 2006). In newer nuclear plant, including Generation IV nuclear plants, it was proposed that to have less maintenance down-time, an integrated approach for monitoring, control, fault detection and diagnosis of plant components such as sensors, actuators, and control devices has to be developed Modern computer controlled industrial systems contain databases that are used to characterise the underlying dynamic processes (Kim, et al 1998).

Koeberg Nuclear Power Plant has been installed with analog control and instrumentation (C&I) systems which are increasingly faced with intermittent operation, frequent failures, obsolescence and high maintenance expenses. It is recommended that nuclear industry adopt modern digital and computer technology innovation to improve NPP safety (Hashemian, 2010). Most undetected errors occurring in the system lead to the system failure. Error detected before the system eventually fails is critical for a reliable fault tolerant system design (Rochester Gas, 2004).

The reactor control and protection system at the Koeberg NPP is designed to automatically initiate the operations of appropriate systems, including the reactivity control systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and senses accident conditions and initiates the operation of systems and components important to safety. Cilliers, et al. (2011), in the publication, have developed a simple model reference control system theoretically by using real time simulators of nuclear power plants. It is done by continuously monitoring and comparing simulated data with the actual measured data from the plant. This mini-dissertation is based on that theory and its main objective is to detect faults in the system (plant) early.

With the ageing of the NPP, there is a need to have a system that can detect and diagnose faults early in the system. Failure of some NPP components is prevalent in NPP which are beyond 20 years in operation. This would have a beneficial value to the NPP as the regular maintenance of such components would be undertaken before the total failure of the system could be experienced, which would affect the plant availability factor.

The introduction of the advanced fault detection system will introduce the benefits of such a system to changes in plant parameters by the nuclear plant by comparing the real-time data of the NPP and the simulator. Ideally, the simulator will be operating in parallel with the nuclear power plant. The proposed model reference fault identification system would improve the dependability of the system. This paper will show that combining real time plant simulations with measurement equipment data in protection and control systems would result in a higher dependability of the system and in turn would result in longer plant up-times and higher plant efficiencies in case of Koeberg NPP (Cilliers, et al 2011).

The simulator to be used against the real plant data is the PCTRAN, which is reactor transient and accident simulation software that operates on a personal computer. The plant model is a 3-loop PWR with inverted U-bend steam generators and dry containment system. The nuclear industry has begun the transition from traditional time-directed, hands-on, and reactive maintenance procedures to condition-based, risk-informed and automated maintenance strategies. This is partly because the current generation (2nd generation) of nuclear power plants has passed its mid-life and increased monitoring of plant health is critical to their continued safe operation.

The operating license renewal of nuclear power plants has accelerated, allowing some plants to operate up to 60 years. Furthermore, many utilities are maximizing their power reactor power output through fuel enrichment change and retrofits. This puts additional demand and more stress on the plant equipment such as the instrumentation and control (I&C) systems and the reactor internal components making them more vulnerable to the effects of aging, degradation and failure. In the nuclear industry, a great responsibility is put upon the simulation of systems to verify and ascertain other aspects of the plant. Thus, there is a need to make the simulator reliable.

1.1 Problem Statement

The NPP has a control and protection system that is designed to oversee and protect the plant against any irregular operation of the primary circuit. The system measures all parameters in the plant against a particular legend (setpoints) and any variations are highlighted by the internal control system. The control and protection system is designed to re-adjust and re-align itself automatically (remotely) to any changes in transients by changing other plant parameters and systems within the plant by countering the parameter changes.

Following the aftermath of Chernobyl nuclear reactor accident, an INSAG (International Nuclear Safety Advisory Group) series of publications were released to contribute to the safety of nuclear power plants and the concept of safety culture was introduced. Safety culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that as an overriding priority; nuclear plant safety issues receive the attention warranted by their level of significance.

The NPP control system has a built-in characteristic, once a fault occurs or any change in plant parameters, the control system will change the other plant parameters to counter and correct the earlier fault. When the fault cannot be corrected by the control system, that is setpoints have been exceeded, then it (control system) can safely initiate the reactor shutdown process of the plant by triggering other systems. In this case, the reactor operator might not know and correctly diagnose the fault that triggered the eventual shutdown of the entire plant. The plant shutdown sequence is a lengthy and costly process which can be avoided in this case by detecting any changes in the system early and avoiding the plant to eventually shutdown.

To encourage and enhance safety in NPP, there is a need for system that will correctly detect faults in the plant early. It is therefore proposed that a NPP simulator runs or operates alongside (real-time) the plant.
The simulator would have input parameters similar to those of the plant and when faults are detected in the system (plant), and then similar faults are envisaged in the plant simulator. The fault parameters would be recorded and analysed to correct the faults in the plant in future. (Cilliers et al., 2011). This system needs to be qualified for a live NPP.

1.2 Aims and Objectives

The protection and control system of a nuclear power plant is a vital part of a plant to promote safety, integrity and availability of the plant. The use of advanced digital technology in the control and instrumentation (C&I) in other industries has been applied and implemented with great benefits. Due to intense safety precautions and stringent licensing requirements, the nuclear industry has been very slow to introduce and exploit the digital technology. Although, some utility has this digital technology licensed, it process is lengthy and time consuming. In an effort to design for redundancy, the digital control system will be allowed to function alongside the existing analogue system.

Some faults that the plant experiences leads to reactor SCRAM but if detected early, they (faults) can be mitigated and avoided if the plant control system didn’t compensate and restored desired functionality. These faults could be isolated and system restored to do its primary functions.

This mini-dissertation will assemble, evaluate and analyse data from specific transients on Koeberg NPP. The results will be compared and evaluated with results from simulated transients. The use of the early fault detection system on the control and protection system of the Koeberg NPP will be evaluated for possible implementation in the nuclear industry so as to increase the reliability of the plant. This procedure will detect faults in the system as they occur. This will show that the introduction of an early detection system in a plant similar to Koeberg NPP will be beneficial.

The proposed system will only run alongside a live plant to detect any faults in the plant system. This is not meant to stop an automatic shutdown of a plant as it may compromise safety of the plant, but to emphasise that the faults in a plant can be detected earlier during normal operation. The simulator will be qualified as a reliable tool firstly, before further analysis can be done. This will help establish confidence in the system. The results of the plant will be compared to that of the simulator running at a steady state condition and later with the results of the simulator running alongside a plant with a fault.

CHAPTER 2

Background

When an NPP gets a plant-life extension, improved life maintenance becomes very important. Past corrective maintenance practices are not practical. Many top performing plants are moving towards condition-based maintenance practices when technology permits. This will allow a plant to optimize their performance by performing maintenance only when the condition requires. These techniques require robust and reliable estimates of the plant condition, which in many cases requires the use of simulators to process the plant data to infer condition.

The plant life management of a nuclear power plant raises several major issues which amongst others involve the aging management of the key components of the plant, both from a technical and an economic point of view. As the NPP is ageing, most components used become obsolete due to high maintenance cost. Most manufacturers of components used on the old NPP ceased the manufacturing of such components due to high production and labour costs. It is difficult to use any off-the-shelf components in NPP as they need to be certified and qualified to be used in NPP environment. Decision-makers are thus faced with the need to define the best strategy in order to achieve the best possible performance while meeting all regulatory requirements.

With the rapid development of digital technology, the analog-based Control and Instrumentation (C&I) systems in some nuclear power plants (NPPs) have been replaced with modern digital based C&I systems. Upon the shifting away from analog to digital systems, safety assessments remain an important factor. However, the different characteristics of these systems make such assessments very difficult. A key difference between digital and analog systems is in the architecture. Analog systems generally do not share hardware elements between redundant channels, and a desired level of system reliability is achieved through replication of the needed number of independent channels.

However, digital systems rely mostly on semi-conductor components to process or transmit multiple signals. The failure characteristics of the two systems are also different, owing to differences in system architecture. In analog-based C&I systems, system failure occurs by degradation of components in the system (Lee et al., 2006).

There is research work regarding the development of next generation prognostics systems which allow condition-based maintenance to take simultaneously into account monitoring data (for early fault detection), and time-dependent aging models. The knowledge-based systems can help top level decision-makers get a transverse, long-term view on how a life-management investment strategy translates into plant availability, avoided costs and improved component durability (Just et al., 2005).

The nuclear power industry is working to reduce generation costs by adopting condition-based maintenance strategies and automating testing activities. These developments have stimulated great interest in on-line monitoring (OLM) technologies and new diagnostic and prognostic methods to anticipate, identify and resolve equipment, process problems and ensure plant safety, efficiency, and immunity to accidents (Hashemian, 2010b).

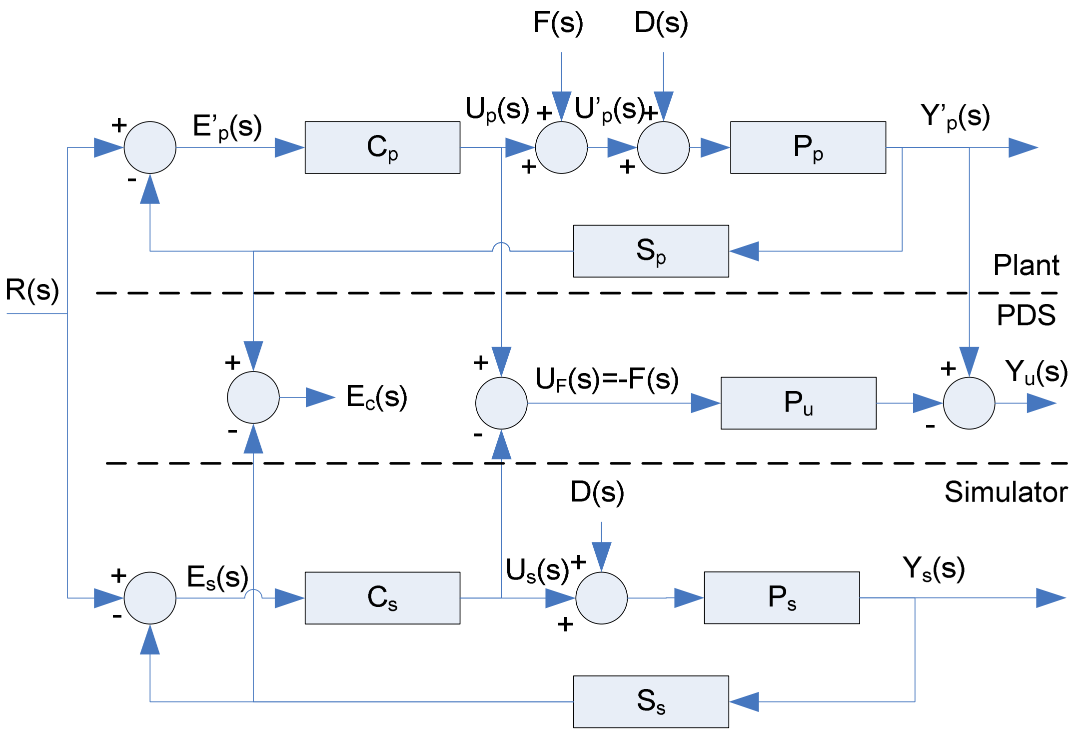

Figure 2.1: Measurement Generation Diagram (Cilliers and Mulder 2012)

Where: R(s) – reference output

$C_P(s)$ – control system

$P_P(s)$ – plant

$S_P(s)$ – measured output (sensor)

$E_P(s)$ – error between reference and measured value

D(s) – disturbances

In figure 2.1 above, the plant and simulator are running parallel to each other. The plant in a steady state condition would operate within its operating envelope (boundaries). The control system output is fed into a system that will detect any fault introduced into the system by the operating system itself. The protection system will access the fault introduced and measure it against the normal and boundary limits of the operating system. The output signal is fed back to the system to measure the error between the reference and measured value so that the control system can adjust so as to counteract the error differences.

For small errors in the system,

$$E_C(s) = 0 \quad \ldots\ldots 1$$

The control and protection system will compensate for the small faults and can be remedied by online maintenance.

When the error difference is large enough, then the control system will not be able to mask the error introduced,

$$E_C(s) \neq 0 \quad \ldots\ldots 2$$

Then it will initiate the reactor shutdown sequence to protect the plant from undesired transients and accidents. This reactor trip is not always desirable from the generation point of view as no more electricity will be generated but safety is always a priority in a nuclear power plant hence the reactor shutdown in case safety might be compromised. The system proposed here, will help detect the potential small errors in the system that grow large enough to trip the reactor when the fault can no longer be compensated by the plant control and protection system.

This is done by introducing the comparison point at $U_F(s)$ which has a feed of information from the output of control systems of both the plant and the simulator as depicted in the figure above. This will act as our plant diagnostic system (PDS). The output of the simulator at this point is expected to be as standard as the input throughout the plant in a non-fault condition (steady state).

The simulator output is not expected to change due to the simulator being a fixed apparatus (system) with standard pre-determined parameters (Cilliers & Mulder, 2012).

The control systems that were simulated includes nuclear power control, average coolant temperature control, pressuriser pressure control, pressuriser water level control, steam generator water level control and steam dump control system. Temperature and pressure are the most critical parameters of the core. The simulator used in this demonstration needs to be qualified and declared reliable. The fault can be detected with great confidence if the accuracy of the simulator outputs are known during all conditions and the tolerance of measuring instruments are determined.

2.1 Control and Protection System

The control and protection system in a nuclear power plant has a safety related function. According to the NRC, Criterion 13, the Control and Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences (AOO), and for accident conditions as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges (Comper, 2003)

C&I is provided to monitor and control the neutron flux, control rod positions, temperatures, pressures, fluid flow and levels so as to ensure that adequate safety can be maintained. Instrumentation is provided in the reactor coolant system, steam and power conversion system, the containment, engineered safety systems, radiological waste systems and other auxiliaries. Parameters that must be provided for the operators under normal operating and accident conditions are displayed in the control room in proximity to the pertinent control devices for maintaining the indicated parameter in the proper range. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the unit. The reactor control system is designed to maintain automatically a programmed average temperature in the reactor coolant during steady-state

operation and to ensure that plant conditions do not reach reactor trip settings as the result of a transient caused by load change.
Overall reactivity control is achieved by the combination of soluble boron and rod cluster control assemblies. Long term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short term reactivity control for power changes is achieved by the reactor control system, which automatically moves rod cluster control assemblies. This system uses neutron flux, coolant temperature and turbine load input signals. The pressurised pressure control system limits pressure excursions that might produce reactor trip, changes in reactivity and actuation of the relief valves.

A wide spectrum of measurements is displayed for operator information, many of which are processed to provide alarms. These measurements provide notification and allow correction of conditions having the potential of leading to accident conditions. Typical indication measurements are rod position, rod deviation, insertion limit, rod bottom, rod control system failure, in-core flux and temperature, protection system faults and protection test mode. Pressuriser pressure level and reactor coolant system are monitored and alarmed to ensure that the reactor coolant system pressure is maintained within design operating limits. Containment pressure is monitored and alarmed to enable the operator to operate the containment vacuum system as needed to maintain the design operating pressure inside the containment. Instrumentation monitoring containment pressure, pressurizer pressure level, steam flow and pressure (Comper, 2003).

2.1.1 Protection System

Similarly, according to the USNRC criterion 20, the protection system shall be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to sense accident conditions and to initiate the operation of systems and components that are important to safety (Comper, 2003).

The reactor protection system equipped with appropriate redundant channels (3 channels, 2/3 logic) is capable of coping with transients where insufficient time is available for manual corrective action. The design basis is in accordance with international standards. The reactor protection system will automatically initiate a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the predefined set-points. The set-points provides for an envelope within which a safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded. Reactor trip is initiated by removing power to the rod drive mechanisms of all the full-length rod control assemblies. This will lead to the control rods to fall by gravity into the core and consequently they would absorb neutrons and stop fission reactions which would reduce the reactor power output. The reactor protection systems also include the engineered safety features actuation systems which automatically initiate emergency core cooling and other safeguard functions when sensing accident conditions. Redundant analog channels measuring diverse variables are used. Manual actuation of safeguards systems may be performed when enough time is available for operator action.

A circuit that is diverse from the reactor trip system automatically initiates a reactor trip through the opening of the RAM breakers and initiates a turbine trip under conditions indicative of an Anticipated Trip Without Scram (ATWS) (Comper, 2003).

2.2 Fault Detection Theory

This mini-dissertation is based on the theory developed by Cilliers et al (2011) on the principle of early fault detection system. The primary objective of the research is to verify the theory developed on a real plant data. In this theory, Cilliers et al. (2011) has proved the authenticity of the theory successfully using only the simulator. This dissertation will verify the theory using the real plant data to detect faults early in the system together with the simulator. The plant diagnostic system (PDS) is introduced between the plant and the simulator. Its role is to continuously compare the measured value of the plant parameters and the pre-determined value of the simulator.

PDS acts as an interface between the plant and the simulator. The measured values are fed into the PDS continuously and if any difference in the values is recorded outside the predefined value, then fault is detected. The control system operates over a range of values or operating envelope, outside of which the protection system will be actuated to counter the status of the plant by initiating an appropriate action. The simulator is operating in parallel with the plant as depicted in figure 2.1. The effect of the control system in a plant is to measure the plant parameters against known reference values. When a different value is measured then the control system will oppose the change in the plant by changing other plant parameters to maintain the steady state condition. (Cilliers et al (2011).

The reference value is acceptable in a particular range; say 6.7% of the reference value is acceptable as operating margin. Once the control system measures the value exceeding the reference margin, then the control system can initiate safe shutdown of the system or SCRAM the plant. The small changes that can be measured are usually not detected because they are overridden by the control system when it re-adjusts the plant parameters. The large deviation from the plant parameters are easily detected and identified as the protection system will be actuated.

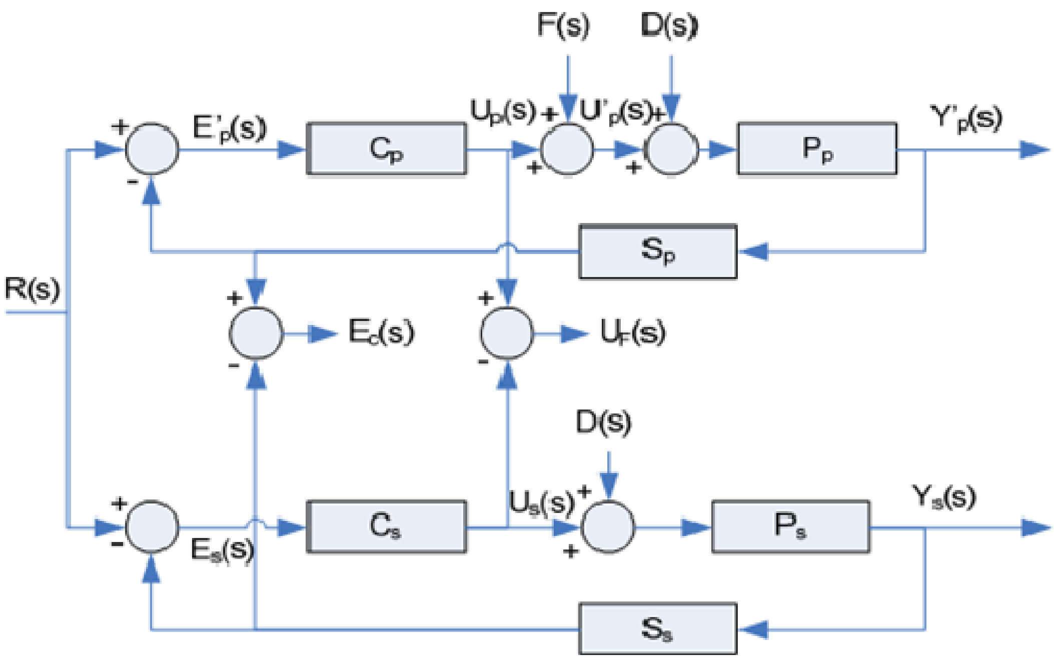

Figure 2.2: Fault detection diagram (Cilliers & Mulder, 2012)

The plant on the top part of the diagram is operating in a closed-loop principle. S_P on the diagram acts as a feed-back control system. $E_C(s)$ is introduced in the system to detect faults. It cannot detect all the faults, especially the small faults as they are masked within the control system. An additional point, $U_F(s)$, is introduced to compare the control system outputs. At point $E_C(s)$, the control outputs of the plant and the simulator are compared. Both outputs are assumed to be the same during the steady state condition (Cilliers and Mulder 2012).

When the control output of the plant and the simulator are constantly compared, then if any deviation in the value is noticed then a fault is detected in the system.

$$U_F(s) = U_P(s) – U_S(s) \qquad \text{2.1}$$

In a steady state condition, the disturbance introduced in the plant, $D_S(s) = 0$. It can be shown mathematically that $U_S(s)$ is approximately 0, then it follows that:

$$U_F(s) = - F(s) \qquad \text{2.2}$$

There is no disturbance acting on the simulator and thus the control system output should be 0. The condition is valid, only if the control and protection system remains within the designed operating region. In large transient condition, the plant operates close to its trip point. Any fault F(s) that is in phase with expected disturbance D(s) would lead to the protection system detecting the fault and initiating reactor shutdown sequence (Cilliers & Mulder, 2012).

If $E_C(s)$ is approximately 0, it is possible there is a fault in the system and might not be detected by the system as a fault as it is small enough to be detected. Also,

$$E_C(s) > 0 \qquad \text{2.3}$$

Then, the fault can be detected by the plant diagnostic system. The control system is operating at its maximum capacity to reduce effect of unbalance in the system and would (control system) be trying to return the plant to the reference value. From the analysis above, it can be shown in general that the equation for fault detection is:

$$Y_U (s) = Y'_P (s) – P_U (s) (U_P(s) – U_S (s)) \qquad \text{2.4}$$

The above equation has been derived in reference to figure 2.1. (Cilliers & Mulder, 2012).

CHAPTER 3

Nuclear Plant and Simulator Theory

3.1 Nuclear Power Plant

Figure 3.1: Koeberg Nuclear Power Plant in Cape Town, South Africa (ESKOM, 2005)

The Koeberg Nuclear Power Plant is a Pressurised Water Reactor (PWR) type, generation II plant with 3 loops consisting of 2 units operating adjacent to each other. It is situated along the western coast of Cape Town in South Africa as shown in figure 3.1. The plant has a thermal power output of 2785 MWth. Each unit is designed to produce net output of 921 MWe. Unit 1 started commercial operation in July 1984 and unit 2 in November 1985.

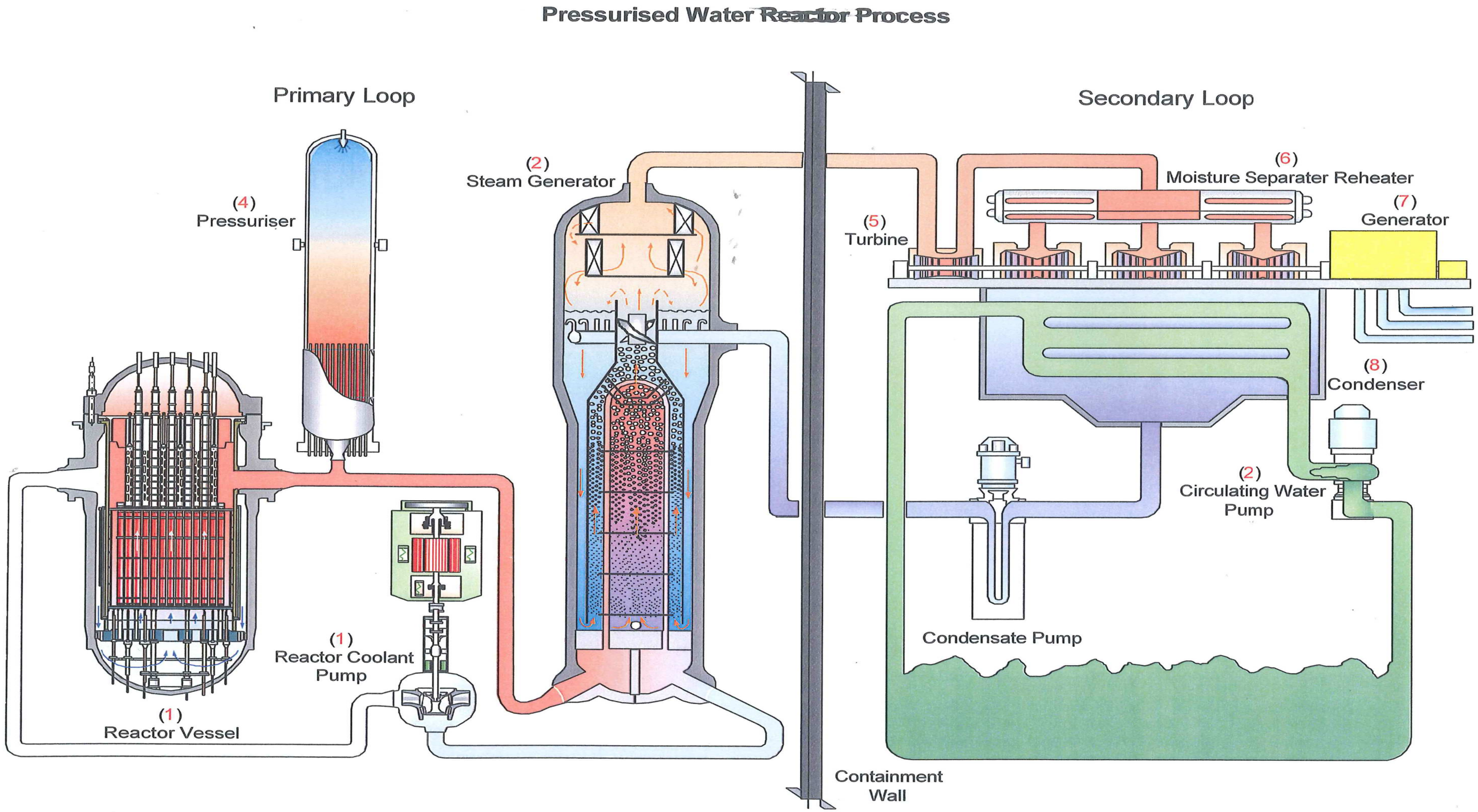

Figure 3.2: PWR major system components (ESKOM, 1985)

The overview of major system components of a typical PWR plant is shown on figure 3.2. Both the primary and secondary loop is interfaced to deliver the electricity at the generator site. These are the most basic major components. Some of these major components and their roles in the plant will be discussed in the coming sections. The KNPP uses water as a coolant and moderator.

KNPP has the following characteristics:

- Nuclear island consisting of 2 reactor buildings each housing a NSSS, 2 fuel buildings, nuclear auxiliary building common to both units and connecting buildings
- Shared turbine building housing turbine generators and their auxiliaries
- 5 diesel generators buildings each housing one emergency diesel generator, 2 are assigned to each unit and one (1) can be assigned to either unit
- Shared electrical building
- One pumping station for the conventional island cooling water
- One pumping station for the nuclear island cooling water
- 2 condensate polishing plants and a water treatment building
- Miscellaneous buildings for auxiliary equipment
- Workshops and service buildings

3.1.1. Major Systems

Each unit has the following systems:

(i) NSSS which includes engineered safety feature systems required for reactor trip, containment integrity, core cooling, containment spray and heat removal, post-accident radioactivity removal and component cooling. The NSSS is designed to withstand all transients anticipated during the service life of the plant with 80 % load factor.

(ii) The containment which houses the NSSS is of concrete lined with steel. It is designed to:

- Prevent release of radioactive products to the environment during normal operation or after LOCA,
- Withstand pressure and temperature following break of line, and
- Withstand a small LOCA,

(iii) Auxiliary systems which includes:

- Gaseous, liquid and solid waste treatment systems,
- Fuel handling and storage system, and
- Nuclear island ventilation systems

(iv) Secondary systems (steam and power conversion)

This is constituted of a turbine generator, condenser, feedwater plant and turbine bypass to the condenser. The condenser steam dump is designed in such a manner that during a transient, the steam produced by the NSSS can be removed. This will allow rapid turbine generator load rejection. Reactor power is decreased by rod control to match the reduced turbine load. Thus, the condenser steam dump acts to prevent reactor trip and lifting of safety valves.

The reactor coolant system and associated control and instrumentation consist of:

- Low-alloy steel reactor vessel with internal stainless steel cladding, containing the reactor core. The core consists of 157 fuel assemblies containing slightly enriched sintered UO_2 pellets. Each fuel rod is enclosed in a leak-tight cladding.
- The three reactor coolant loops containing water at a pressure of 15,5 MPa. Each loop includes one reactor coolant pump, one steam generator and loop piping. Steam at 4,8 MPa is produced on the steam generator shell side, passed through moisture separator dryers and routed to the main turbine.
- A pressurizer which is connected to one of the loops through a surge line which maintains constant reactor coolant pressure. See figure 3.2.
- Instrumentation channels which allows continuous monitoring of NSSS parameters. The signals generated provide inputs to the reactor protection system which prevents safety limits from being exceeded.

- 48 RCCA which controls reactivity by inserting or withdrawing rods from the core (reactivity control is also ensured by varying the concentration of soluble poison in the reactor coolant).

3.1.1.1. Nuclear Steam Supply System (NSSS)

The NSSS control system has mainly the following functions:

- It maintains the operating parameters at values as close as possible to the optimal values for dependable and economic operation established by design studies, during the steady state operation.
- It enables the NSSS to deal with a certain number of normal transients dictated by operating requirements and the manoeuvrability desired, bearing in mind the characteristics of the grid.
- It enables the plant parameters to be maintained within a range acceptable for proper operations of the whole of the installation so as to avoid actuation of the protection system (ESKOM, 1985).

3.1.1.1.2 PWR Primary Circuit Control System

PWR uses 10 primary control systems to keep the plant operating in a steady state and within design safety limits. The control systems are:

- Pressuriser pressure control
- Pressuriser level control
- Reactor average temperature control
- Atmospheric steam dump control
- Steam generator level control
- Steam generator feed water flow control
- Steam pressure control
- Power control
- Turbine speed control
- Generator voltage control

3.1.1.1.3 Parameters Controlled

*(*a) Reactor Coolant Temperature

The temperature of the reactor coolant is regulated by variations in the temperature of the saturated steam within the steam generator. The control rod temperature regulating systems for small transients and by condenser steam dump for large transients. The design operating temperature is approximately 343 ^{o}.

(b) Reactor Coolant Pressure

The reactor coolant pressure must remain above a value which would produce excessive boiling at the output of the hottest channel. The pressure must remain below the designed primary system pressure so as to avoid the risk of damaging the components in the system. The pressure is adjusted to a constant value of 15.5 MPa. The error in pressure signal is given by the difference between the pressure, P measured by a pressure sensor and the setpoint pressure, P_{ref}. This error signal is processed by the pressuriser pressure controller by PID action whose transfer function is given by

$$K_{21}\ (1 + 1/T_{21}\ s) + (K_{21}\ T_{22}\ s)\ /\ (1 + 1/\lambda \times T_{22}\ s \qquad \ldots\ldots\ldots\ldots\ldots\ldots 3.1$$

$\lambda \equiv$ transient gain

The last part in the equation (1/ T_{21} s), gives the correction needed during the slow transients. The output signal from the controller that compensated pressure error signal operates the following:

- The relief valve
- The heater
- The 2 spray control valve

(c) Pressurizer level

During normal operation, the reactor coolant is constantly renewed by the Chemical and Volume Control System so that it regulates the quality of the borated water and adjusts the concentration. The pressurizer fluid level control system ensures that the constant mass of water is maintained within the main reactor coolant pressure boundary.
The level controller has proportional plus integral type. The transfer function is: K_{22} (1 + 1/T_{23} s)……………………………………………………………..3.2

The reactor coolant water at constant inventory changes volume in relation to cold leg and hot leg, core and steam generator temperatures. Good approximation to the level can be obtained on the basis of the average temperature alone. Hot leg and cold leg temperatures are derived from average temperature and power.

(d) Steam generator level

The steam generators transfer the core thermal power towards the turbo-alternator and thus ensure the proper heat transfer from the primary system to the secondary system and normal operation of the steam separator and dryers, the mass of the fluid within the steam generator must be controlled by adjusting the fluid level in the steam generator.

(e) Secondary steam pressure

During power operation, the secondary pressure is not controlled. For a NSSS with steam generators using natural circulation, the reactor coolant system average temperature and the secondary pressure are inter-correlated as a function of the power level during steady state operation. During a transient, the secondary steam dump system is used temporarily to create an artificial load by releasing steam taken upstream of the turbine inlet valves, either to the condenser or to the atmosphere. The system is also used for cooling the reactor and for evacuating residual heat when the turbine is shutdown. The system comprises of condenser dump and atmospheric dump.

The turbine by-pass valves are used to control either secondary steam pressure or primary temperature. During large transients such as load rejection, turbine or reactor trip, the condenser steam dump is used in addition to rod insertion to control primary average temperature. It allows temporary evacuation of steam refused by the turbine while the control rods progressively decrease the reactor power.

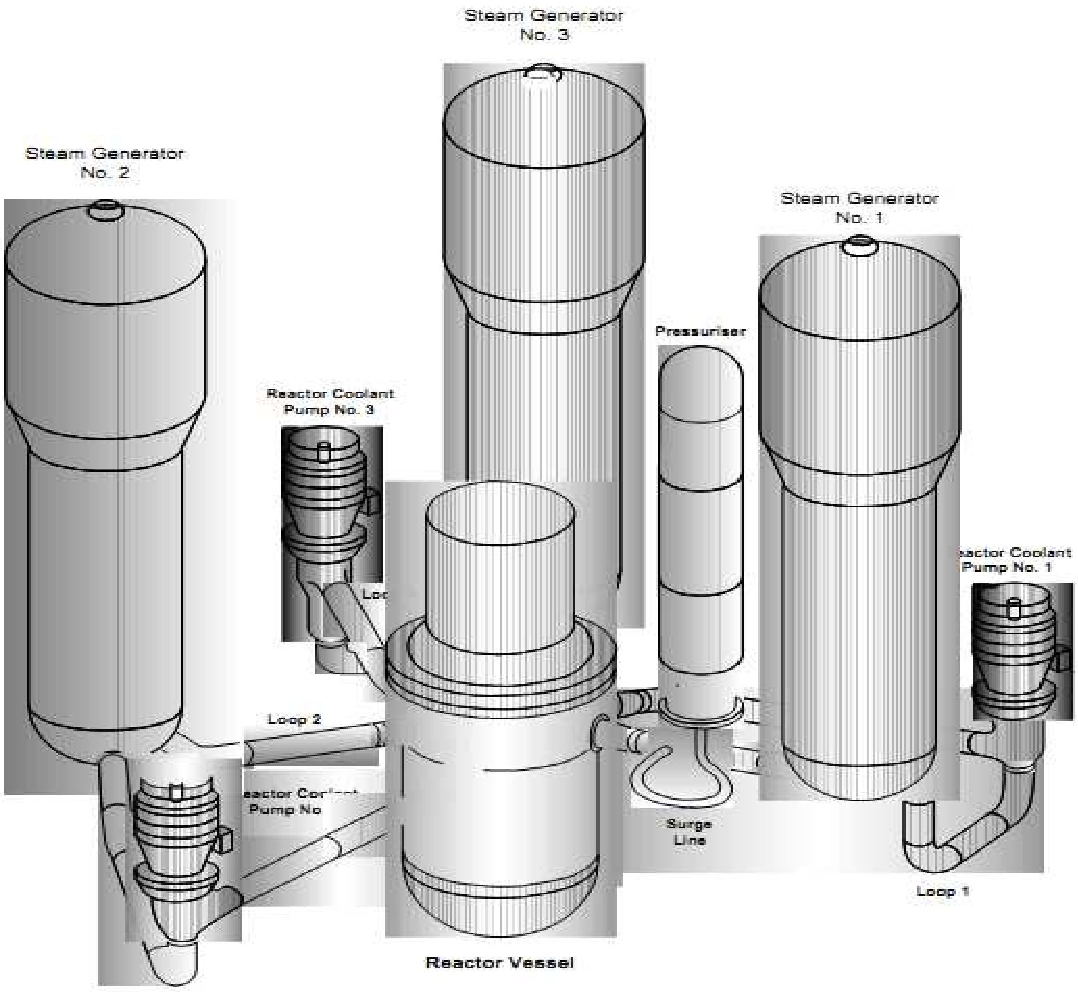

Figure 3.3: Reactor Coolant System (RCP) (Comper, 2003).

The figure 3.3 above shows some main components of the primary system. The primary coolant is designed to transfer heat generated in the reactor core through fission process to the steam generators when operating at full power and to remove the core decay heat during the reactor shutdown. During normal operation and reactor transient conditions, the primary coolant pumps maintain the flow of the primary coolant (water) through the core.

The heat is transferred by reactor coolant through three independent closed loops to the steam generators (SG). The SG's will transfer the heat in the form of steam to drive the turbine.

The main components of the RCP include:

- Reactor Pressure Vessel;
- 3 Steam Generators;
- 3 Primary Coolant pumps;
- Pressuriser; and
- Pressuriser Relief Tank (ESKOM, 1985).

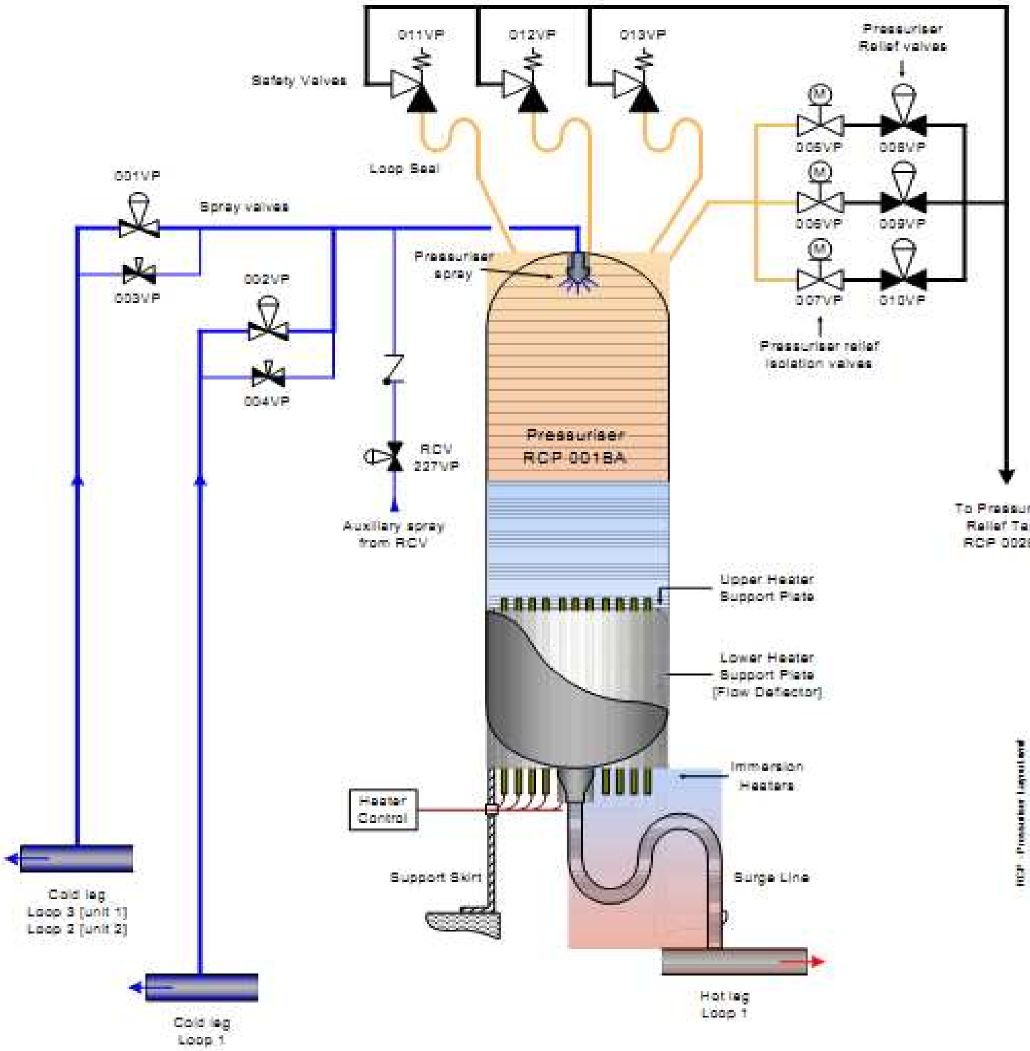

Figure 3.4: Pressuriser General Layout and associated components (ESKOM, 1985).

The pressuriser (figure3.4) is a vessel that forms part of the primary circuit and is connected to loop 1 hot leg through a surge line. The liquid and vapour are maintained in equilibrium in the pressuriser, under saturated condition for pressure control purposes. The purpose of the pressuriser is to maintain pressure at the set-point value of 15,4 MPa so as to avoid boiling in the primary coolant. It acts as a surge tank for the primary system by absorbing volume changes as the temperature changes. It relieves the high pressure steam in emergency conditions by opening and closing the pressure relief valves. It also protects the RCP against high pressure gradients during transients (Cilliers et al., 2011).

Pressure is controlled by increasing power to the heaters to advance the saturation conditions thereby increasing pressure of the system or it can be reduced by spraying water into the steam space to condense some steam and reduce saturation conditions.

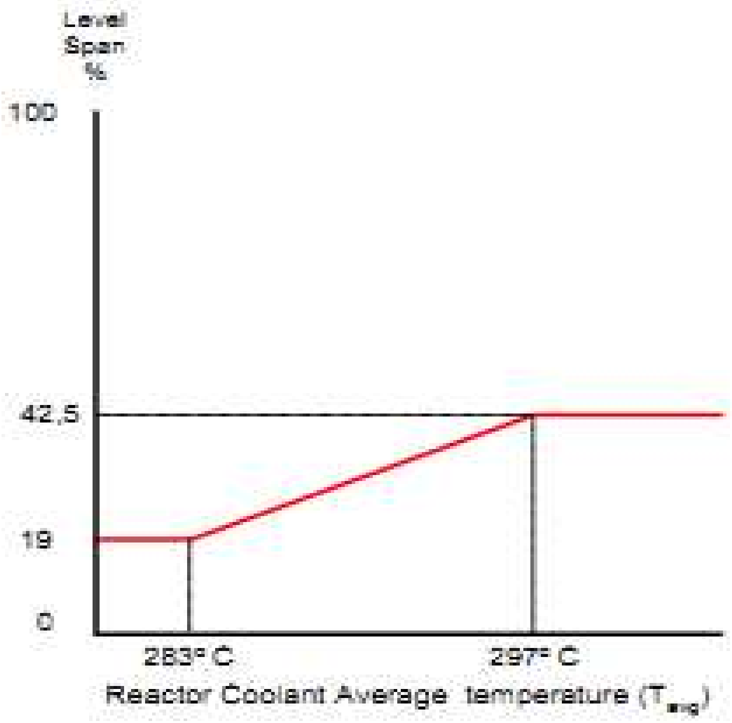

Figure 3.5: Pressuriser level control (Cilliers et al (2011).

The level in the pressuriser is set to change with actual power. When the reactor coolant average temperature (T_{avg}) increases, then the volume of water increases due to thermal expansion and this will lead to the level in the pressuriser increasing. The temperature range of the coolant is shown in figure 3.5 above (Cilliers et al (2011).

3.1.2. Nuclear Auxiliary Building

The nuclear auxiliary building is shared between the units and houses systems necessary for unit operation and safety. Some of the systems are:

- High head safety injection sub-system components;
- Chemical and volume control system components;
- Reactor boron and water make-up system components;
- Gaseous, liquid and solid waste treatment systems;
- Component cooling water system components; and
- Ventilation system.

3.1.3. Electrical Building

The electrical building contains the control rooms and all electrical equipment required for plant control and instrumentation. The system includes all the required characteristics of independence, redundancy, operation, and testing, so that the successful operation of the safeguard systems can be ensured. The separation between off-site and on-site power supplies is made at the 6.6 kV medium voltage busbar level. For house-loading, each unit may be operated so that the generator is separated from the preferred source supply (400 kV) network, the line circuit breaker being in the open position. The auxiliaries of the unit are supplied through the (24kV) generator circuit breaker and the unit transformer.

The main features of the distribution network are as follows:

The generator circuit breaker in the 24 kV busbars between the generator and the generator transformer enables the generator to be synchronised to the network.

The auxiliaries for each unit are divided into groups depending on their safety function and operational function, including consideration of all the different operational situations and foreseeable abnormal situations. The two units are electrically independent to each, in case of an accident; the faulty unit must be able to respond to the accident regardless of the condition of the other unit. For this reason, each unit has two independent diesel generator sets.

A fifth standby diesel generator (9 LHS) acts as a replacement for any one of the other four emergency diesel generator sets should one of them be unavailable or undergoing maintenance. The fifth diesel may be connected manually to any one of the safeguard boards as required.

3.2. Simulator

The PCTRAN is a simulator whose concept is based on a 3-loop PWR system. It is a reactor transient and accident simulation software programme. It is the plant model PWR with inverted U-bend steam generators and dry containment system. For a Westinghouse designed PWR of 3000 MW_{th} (900 MWe), a single loop with the pressurizer is modelled separately from the other two loops lumped together. Plant data parameters controlled by the users would define the model to represent a specific plant.The major plant systems simulated will be described. In a PWR, the primary coolant system, the water is not allowed to boil and steam is generated through the steam generators in the secondary loop. The pressuriser is used to maintain the pressure of the primary coolant to a constant value.

The development of PCTRAN has focused more on abnormal transients and accidents than normal operation. This was motivated by the aftermath of the TMI accident, the operators and industry as a whole had shown weakness in handling complex and multiple failures. PCTRAN does have sophisticated control systems for rods, primary and secondary pressure and level controls. Without that the plant cannot be operated properly. The neutron flux of the core is controlled by the control rod system and soluble boron. The chemical and volume control system maintains the primary coolant inventory and water chemistry.

Steam output is controlled by the turbine control valve and steam dump system. The feedwater system controls the steam generator water level. The PCTRAN simulator has built-in design basis accident and incidents that can be simulated during the normal running of a plant. The KNPS is operating at the pressure of 15 MPa. The error between the system pressure and set-point is routed through the controller circuit and will be recorded. If the error is higher, then the spray will be turned on. If the pressure increases further during a transient, there is a relief valve and safety valves set to open to relieve the pressure. If the pressure decreases during a transient and negative pressure error exists, the heater will be turned on.

The makeup pump using an error of pressuriser level to the level set-point controls pressuriser water level. Let-down is turned on when the pressuriser level exceeds the set-point. The makeup and let-down system also controls the reactor coolant chemical composition. When the pressuriser level is too low, the let-down is isolated and the heaters are turned off. During normal operation, feedwater pumps provide water to the steam generators. The feedwater control valve is regulated by the sum of two errors: steam generator water level relative to the level set-point and feedwater to steam flow mismatch. The valve controls the feedwater flow until any transition is stabilised and the error diminishes.

When the operating parameters of a reactor exceed some pre-defined safety limits, all the control rods are dropped by gravity into the core to SCRAM the reactor. The following trips functions are typical of PWR's:

- High neutron flux;
- Over power delta;-T
- High reactor pressure and pressuriser water level;
- Low reactor pressure and pressuriser water level;
- High temperature delta-T;
- High RC outlet temperature; and
- Containment pressure.
- Low SG water level;
- Low loop and core flow;

The over-temperature and over-power delta-T trips are temperature differences between the reactor coolant inlet and outlet for core DNBR protection. Liquid boron injection is used to provide negative reactivity if the rod insertion functions fails. The PWR has redundant trains of Emergency Core Cooling System (ECCS) for core heat removal during emergency situations. They are composed of the following systems:

(i) High Pressure Safety Injection (HPSI) system.

It consists of redundant trains of centrifugal pumps that can run on emergency diesel power and can operate on high pressure. This will start on a low reactor pressure and low pressuriser level signal or high containment pressure signal. The aim is to make up coolant loss on a small break LOCA beyond the regular makeup system's capability.

(ii) Accumulators (ACC).

The tanks are filled with borate water and pressurised nitrogen. For a LOCA not recoverable by the HPSI, valves connecting the ACC and the reactor coolant system are opened at 4 MPa. They will be closed when the two side pressures are equalized so that nitrogen is prevented from entering the RCP.

(iii) Low Pressure Safety Injection (LPSI) system.

It has redundant trains of centrifugal pumps to be started on Safety Feature Actuation System (SFAS) signals. Their shutoff head (1.0 – 1.5 MPa) is lower than the HPSI but the flow rate is much higher. It has the capacity to completely refill the reactor vessel following a major LOCA to the point of break. LPSI normally takes its suction from the Borate Water Storage Tank (BWST).

When the water is exhausted, the operator will switch the suction from the building sump and run through heat exchangers before injecting back to the reactor. Some plants use the same pumps in LPSI which are used to decay heat removal during the cool-down period after a normal shutdown.

(iv) Containment System.

To prevent the over-pressure in the containment after a LOCA, PWR is fitted with a containment spray system and emergency fan coolers. The suction of the spray pumps is from the BWST. This is also switched to recirculation mode when water supply is exhausted. The heat exchangers are used to remove the heat inside the containment to outside ambiance (MST, 2009).

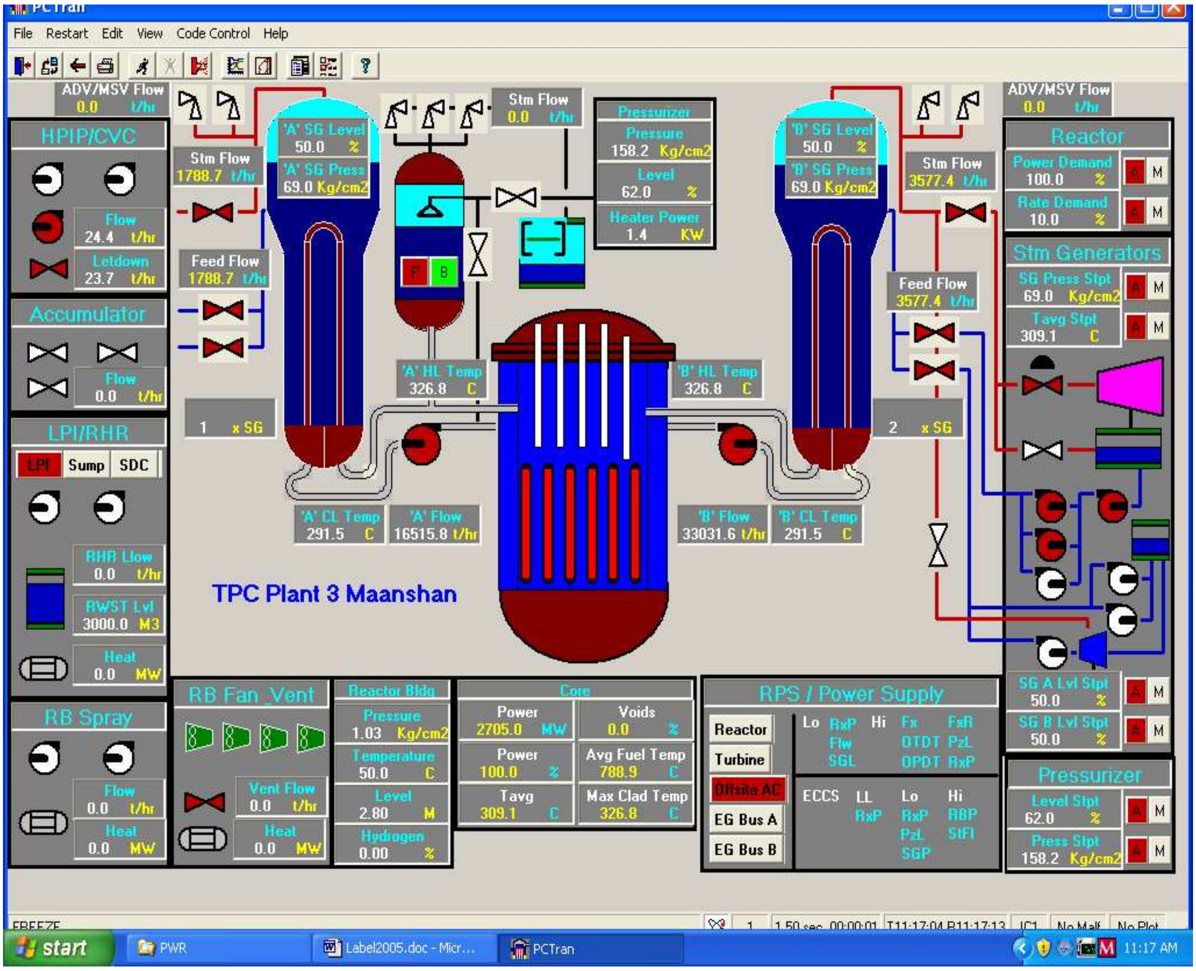

Figure 3.6: PCTRAN Simulator (MST, 2009).

The figure 3.6 above shows, all the major components of the reactor core. All the parameters of the simulator are also shown online. In PWR's primary coolant system, boiling is suppressed and steam is generated via steam generators in the secondary coolant loops. A pressurizer is used to maintain the primary coolant in sub-cooled condition and the pressure close to a constant. Reactor coolant pumps are used to circulate the primary coolant through steam generators. Steam is generated at the secondary side to drive the turbine.

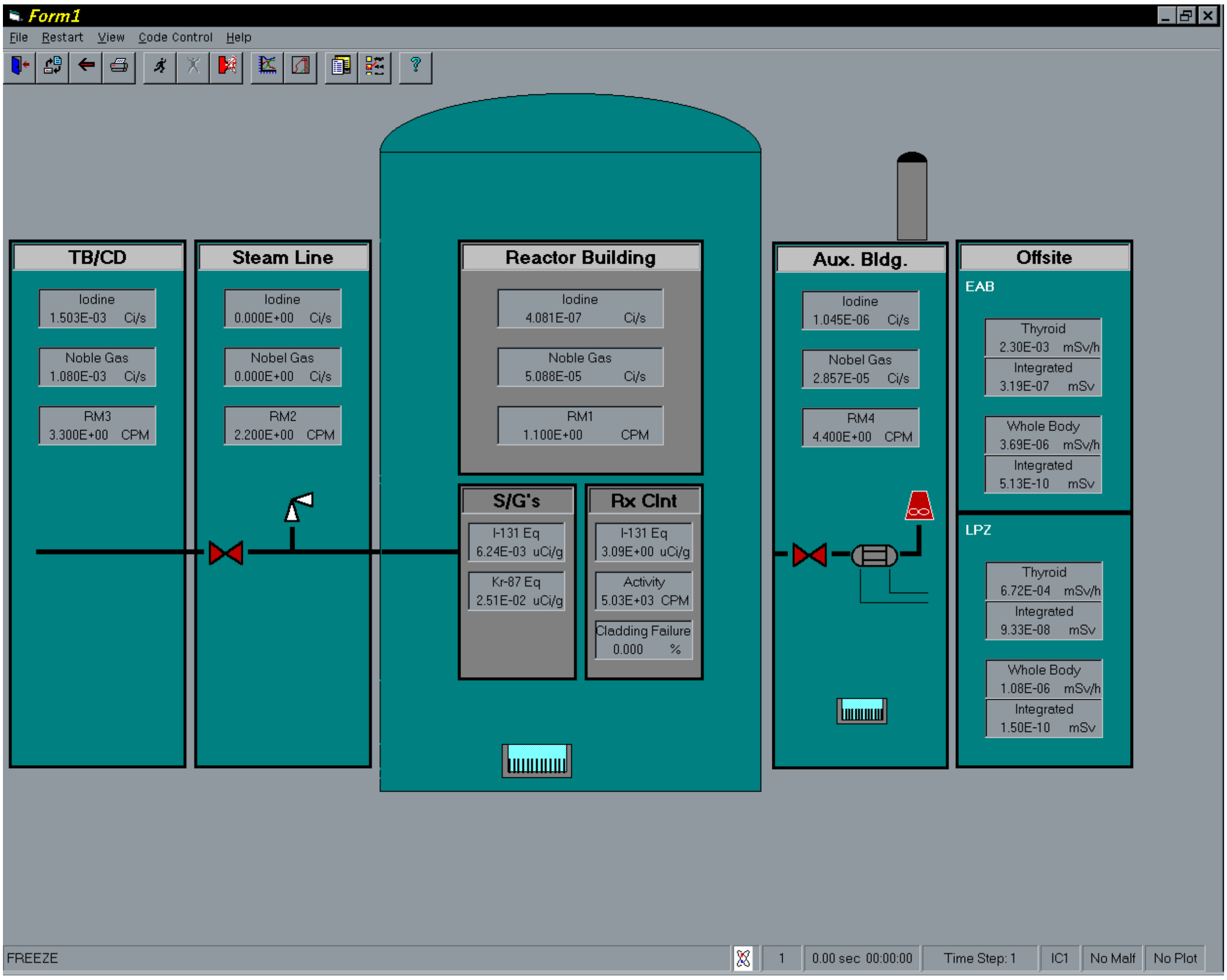

Figure 3.7: PCTRAN Radiation Monitoring System and Source Term mimic (MST, 2009).

3.2.1 System Operation

The operation of PCTRAN systems is based on the user-friendly concept using point-and-click mouse control in pull-down menus. The Westinghouse 3 loop PWR with inverted U-bend steam generators and dry containment system will be explained.

Figure 3.6 and 3.7 is the mimic of the PWR model for a three-loop PWR of 2785 MW. Its net electric output is about 900 MW. A single loop designated as "A" is at the left side and the other two loops are combined as "B" at the right side.

The display also represents the controllable system as small panels with the important equipment shown as icons (i.e., pumps, valves, and heat exchangers). Selection of the panels and equipment displayed in the mimic is consistent with the description of plant systems in Section 3.2. The real control room of a nuclear power plant has hundreds if not thousands of instruments and controls: gauges, displays, strip charts, knobs, switches, dials, push buttons, etc. They are reduced to an absolute minimum in order to fit into a PC's screen. The basic principle and characteristics of the simulated advance reactor will still be demonstrated by the selected mimic display.

For a major control system where complicated automatic operation logic is involved, e.g., rod assemblies, pressurizer level and pressure, steam generator level and pressure, operation is defaulted to the automatic mode. At any given time if the operator decides to take one of the control systems into manual operation, he/she just clicks at the corresponding "M" button and a window will show up. By entering a new set point, activating the manual action and closing the window, the reactor will then run into a manual mode. These panels are located at the top-right hand of the mimic. If you need to change the set point again, click the “A” button and “M” button again to open the window. Also the malfunctions can be activated in this way.

Critical components such as the Power-Operated-Relief-Valves (PORV) and safety valves of the pressurizer and the steam lines, pressurizer spray valve and heaters, Main Steam Isolation Valves (MSIVs), Turbine Bypass (Steam Dump) Valves, Feedwater Valves, Reactor Coolant Pumps, etc. are displayed locally. Their status is indicated by colour and can be overridden by the operator [0]. Control rod position and motion is displayed by motion of simulated control rods. Pipe breaks are shown dynamically by flashing sprays at the break location with the leakage flow digitally displayed.

The system controls for normal operation is as follows:

3.2.1.1 Reactor Control

Pwr Dmd	=	Power demand (%)
Rate	=	Ramp rate (%/min)

3.2.1.2 Steam Generator Control

Turb P	=	Turbine header pressure (bar)

Tavg	=	RC T_{avg} control (C)
SG Lvl	=	SG narrow range (%)

There are turbine-driven and electric-driven auxiliary feedwater pumps. They will be started on a low water level signal in the steam generators. There are two types of controls during power operation according to the steam header pressure/steam dump control diagram: Tavg control and pressure control. After a MSIV closure at one SG, when it is reopened, the operator should switch to the pressure control mode by clicking the "M" button of the "SG press StPt" panel. Even when there is no set point change, just close the window and PCTRAN will switch from the normal Tavg control to pressure control mode. The reactor will slowly recover to proper power distribution (1: 3) left to right. It can be checked by plotting the SG power removal A and B.(MST, 2009)

3.2.1.3 Pressurizer Control

Lvl	=	Pressurizer water level (%)
Press	=	RC pressure control (by heaters and spray)

To return to Auto mode, click at "A". The set point field shown in the window will not be used for auto mode; just activate and close the window it will return to auto operation. The charging pump and let-down valve is used for actual control of the level. Their status is displayed in the upper right panel. The operator can override it by using the right mouse button.

The transfer function of the pressuriser pressure is:

$$K_{21} \left(1 + 1/T_{21}\, s \right) \qquad 3.3$$

3.2.1.4 RPS and ECCS Manual Control

At the bottom of the mimic, status of the Reactor Protection System (RPS) and Safety Feature Actuation System (SFAS) are displayed. Reactor will be tripped automatically upon conditions exceeding any of the RPS set points. The corresponding symbol will turn into red. For example, if the reactor pressure is below the set-point for low-pressure trip, 127 bar, the symbol RC P Lo and the reactor trip button RX T will turn into red. It is followed by all control rods insertion. The turbine stop valve will be closed and the turbine's colour will turn from pink to blue [0].

Other trips in the panel include:

- High reactor pressure (RX P Hi) at 167 bar
- Low steam generator level (SGL Lo) at 28 percent
- High steam generator turbine trip (SGL Hi) at 82 percent
- Anticipatory reactor trip at turbine trip (Tb Ant)

The SFAS signal starts HPSI and LPSI. The panel includes the following signals:

- High Reactor Building Pressure (RBP Hi) at 2.6 bar
- (RBP HH was not used for this model)
- Low-low Reactor Pressure (Rx P LL) at 123 bar
- (Rx Lo for simultaneous pressurizer low level not used in this model)

The reactor/turbine can also be tripped manually by moving the mouse and clicking at the button. On the left-hand side, panels for the HPI, ACC and LPI are displayed. Operators can override the automatic initiations of any ECCS pumps and take manual control.

- HPI and CVCS

Two of the four HPI pumps will start on the SFAS signal, the other two are spares. The positive displacement pump and the let-down valve are part of the CVCS and controlled by the pressurizer level control logic.

- Accumulators

Two valves connecting to the accumulators will be opened at RC pressure below 48 bar. They will be closed when the liquid is exhausted.

- LPI/RHR

Two of the LPI pumps will be started on SFAS signal also, but no flow will be injected into the RCS until the pressure is below the pump shutoff head at 11.4 bar. A large flow will be shown and the water level in the Refuel Water Storage Tank will start to decrease. When it is about to be empty, the operator should re-align the suction from the building sump by clicking at the "Smp" button. Then water will be routed through the heat exchanger and a heat removal rate will be shown. The same pumps are used for shutdown cooling by the Residual Heat Removal (RHR) system during normal cool down. This can be conducted by clicking at the "SDC" button.

A Refuel Water Storage Tank (RWST) object is added in the RHR/LPI panel. As HPSI, LPSI and building spray water draws water from the RWST to the minimum, suction will be switched automatically from the building sump and routed through the respective heat exchangers. The heat exchanger removal rates will be displayed.

The nominal RHR heat exchanger rate QRHR0 in MW is also added into the table.

- Reactor Building Spray

  Reactor Building Spray is started on RB high pressure at 2.6 bar.

- Reactor Building Vent

  The normal Reactor Building Vent will be closed on SFAS signal for containment isolation.

- Fan Coolers

  Fan Coolers are started on high RB pressure also. The containment air is routed through heat exchangers and cooled by external service water to remove the containment heat.

- P/T Saturation Diagram

  As a result of the TMI-2 accident, PWR control rooms have been equipped with RC pressure to temperature P/T diagram showing the sub-cooling margin. Two dots in red and green in the diagram represent the two hot legs, pressure and temperature. Their horizontal distances to the saturation curve show the sub-cooling margin. [0]Shutdown and Cooldown in Simulator

The following are transient plots:

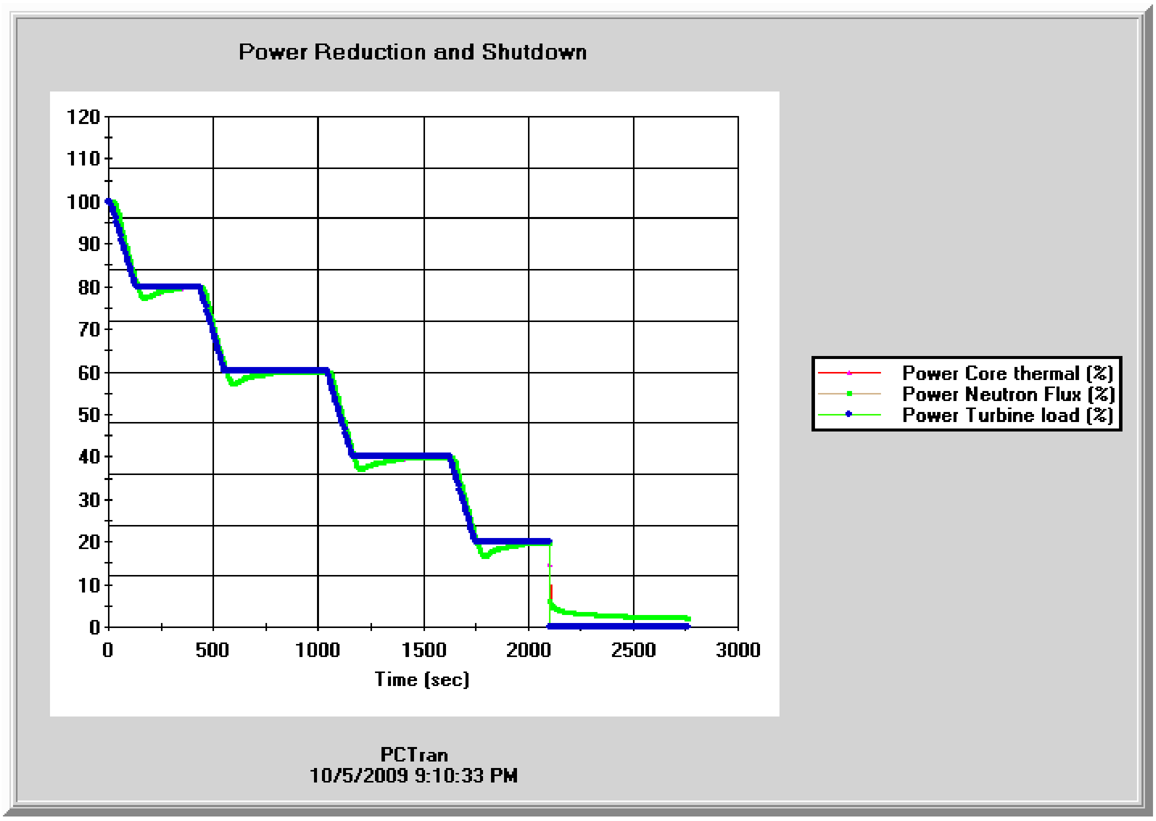

Figure 3.8: Power reduction and shutdown (MST, 2009).

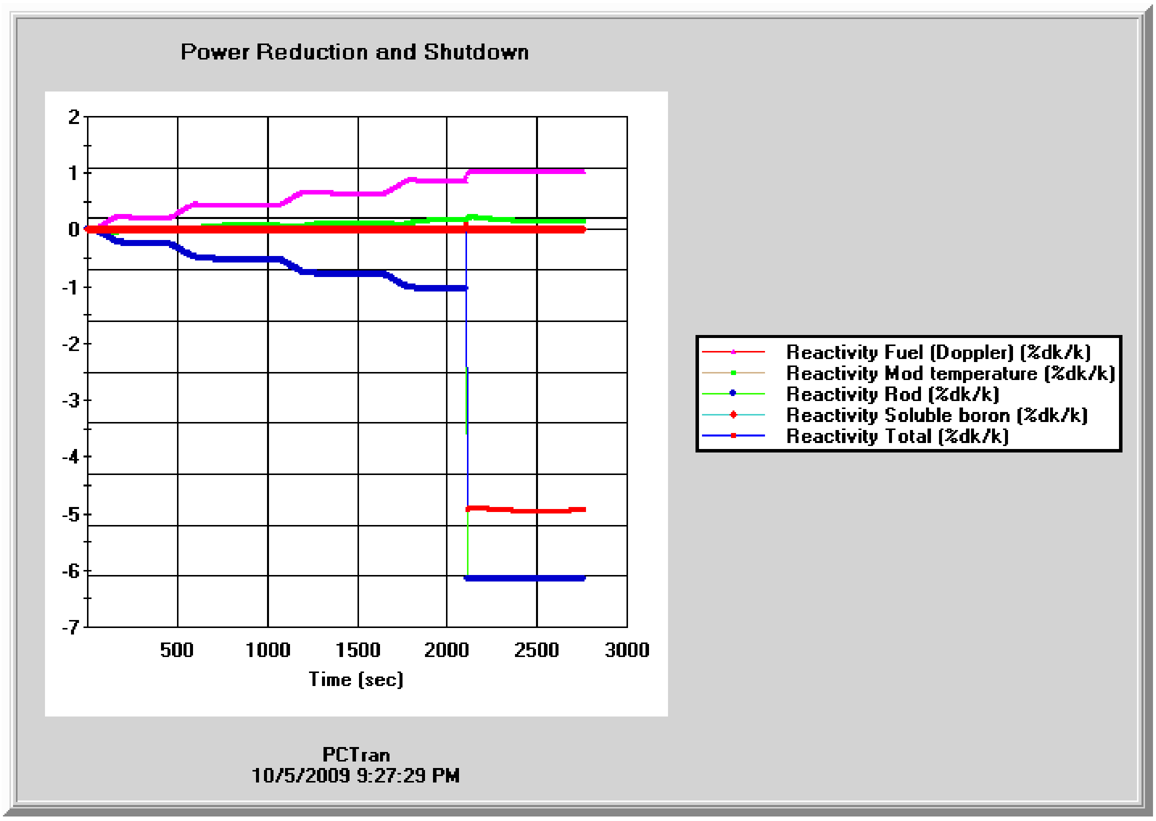

Figure 3.9: Reactivity control during shutdown (MST, 2009).

The figure 3.8.and figure 3.9 above shows the power reduction to cold shutdown in a simulator at a reduction rate of 20 percent per unit of time. The reactivity control in respect of fuel, rods and boron insertion are also illustrated here.

CHAPTER 4

Methodology

The process will be conducted in 2 legs, one being that the NPP simulator needs to be qualified against the NPP and secondly, the simulator will be used to simulate record and evaluate real-time data as recorded from the input plant parameters. The accuracy of the NPP simulator needs to be evaluated by using data from Koeberg on specific expected and normal operation condition and parameters (100 percent power down to 80 percent power); all parameters that changes are recorded. The similar data is used on the simulator to simulate these specific transients and the changes observed are recorded, compared and analysed.

The acquired information from the simulator will be recorded and compared to the real-time plant parameters. If the results of the anticipated plant parameters are similar to the results from the simulator, then we can qualify and validate the simulator to be reliable. It is anticipated that there might be a certain variation from the simulator data to the real plant data. This will help to calculate the percentage error of the simulator so as to help us validate the data from the simulator in un-anticipated fault evaluations. Percentage error band will be calculated from the simulator results.

The simulator can be used from then on, to simulate un-expected transients that have previously occurred with parameters that were not recorded due to the plant control and protection system re-aligning and compensating for changes in the plant behaviour. As it is depicted in figure 2.1, the simulator runs in parallel alongside the NPP. Only the output of the plant is monitored and compared with the output of the simulator. The resultant will be analysed for any drifting from the steady state condition results.

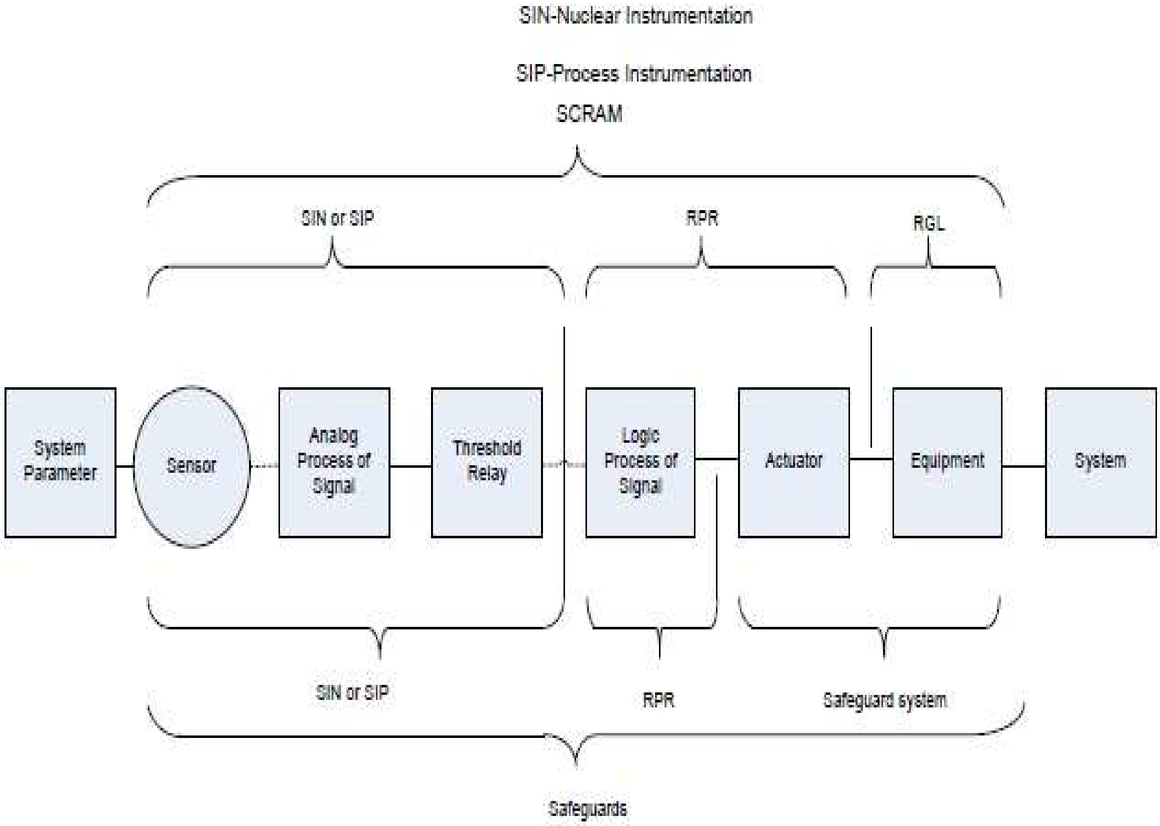

Figure 4.1: Control and Protection System (ESKOM, 1985).

The figure 4.1 shows the block diagram of the control system of the plant. The safeguard systems are shown, mainly the nuclear instrumentation and process instrumentation.

The reactor protection system automatically initiates a reactor trip when any variable monitored by the system, or combination of monitored variables exceeds the normal operating range. Set-points are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded. Reactor trip is initiated by interrupting the power to the control rod drive mechanisms of all the full length control rod assemblies. This causes the control rods to fall by gravity into the core and thus, rapidly reducing the reactor power output. The adequacy of the protection system characteristics has been verified by analysis of the various anticipated transients

With the simulator running alongside the plant when such faults are detected, the simulator cannot compensate itself to correct the fault; hence the fault can be traced back to a specific system.

The following sequence was followed in operating the simulator to track the plant data. The power is reduced in a similar to ascension by stepwise operation according to the following steps:

1. In the upper right “Turbine Drive” panel click the “M” for manual and enter 40 % demand for turbine power. The default ramp rate is 5,7 % /min. The turbine load will be reduced to 40 percent. The control rod assemblies; pressurizer heats and sprays; charging and let-down in Chemical and Volume Control System; turbine header pressure and feedwater control will all coordinate to meet the power reduction requirement.

2. When most parameters and systems settle down to the desired condition, then we shall repeat the above by typing in 40 % for the turbine power. Note: the pressurizer level is lower, the steam generator level is the same, RCS pressure is the same, SG pressure is higher and Tavg is lower for a lower power level. All these follow the plant design diagram.

3. After the plant condition becomes stabilized, operator could take the turbine off-line and scram the reactor. This is done in PCTRAN by simply clicking the “Reactor” scram button.

4. After reactor trip the TBV will automatically set to the pos-trip pressure set point at about 1000 psia. Feedwater and TBV flow will be regulated to match the core decay heat.

In the following section, the plant data which has a fault in the system will be simulated with PCTran. The results are compared to each other to see if there are any differences. This will further establish confidence in the simulator. Thereafter, to illustrate the fault detection method, the simulated plant data with a fault is fed through a Plant Diagnostic System together with plant data within which a transient is introduced into that system. This will show different results that would show the effectiveness of the early fault detection system as proposed.

CHAPTER 5

Results and Analysis

5.1 Simulator Validation

The validation of the simulator is necessary to qualify the simulator as reliable. This was done by taking data from a transient as it occurred in the plant and using the same data in the simulator and compares the outcomes. If the trends in the data collected for both plant and simulator are similar then we can qualify the simulator and declare it to be reliable.

When the simulator is running at full power, the results depict a nuclear plant operating at its optimal level with steady state condition. When the fault is detected by the protection system of the plant, the control system will try to change other parameters to counteract the effect of the fault introduced. If the fault is large enough and outside the operating limits of the protection system, then the reactor trip will be initiated to protect the plant from possible damage. KNPP unit 1 tripped during operation due to a faulty electronic component. Some of the data from the transient was obtained and simulated in the PCTRAN to produce similar results. The results are presented here and compared to establish any relationship for the proposed fault detection system.

The results obtained here is only for some critical primary systems and only the simulation of such systems have been performed.

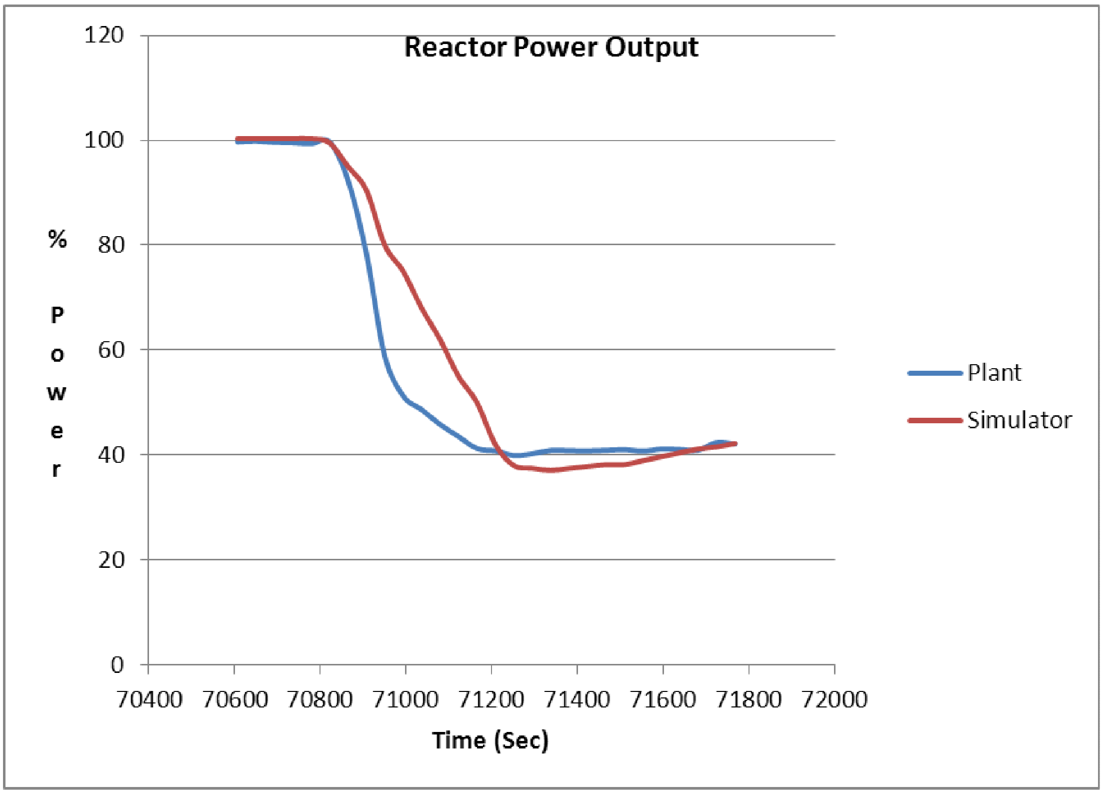

Figure 5.1: Reactor Power (%) Vs time (sec).

The plant has been operating on a steady state condition as depicted in the figure 5.2 below. At a particular time, the plant is scrammed due to a fault in the control system. The protection system duly initiates a reactor scram to protect the plant from further incidents. The reactor power output reduces from 100 % power to a minimum of 40 % at a rate of approximately 5 % per minute before it stabilises again. Due to this reduction in power of the reactor, all other parameters in the reactor core are affected and will also change with respect to that.

The simulator was also set at the similar rate of reduction of power as it tracks the plant data as depicted by the red line in figure 5.1. The power of the plant is reduced from full power to approximately 40 % power and due to the limitations on the simulator; it cannot be scrammed at the similar rate and still maintains operation. The transfer function between the simulator and the plant is different as explained earlier in the theory. But it is evident from the data that the simulator gives closer results to the plant data. The difference in plant data and simulator is minimal.

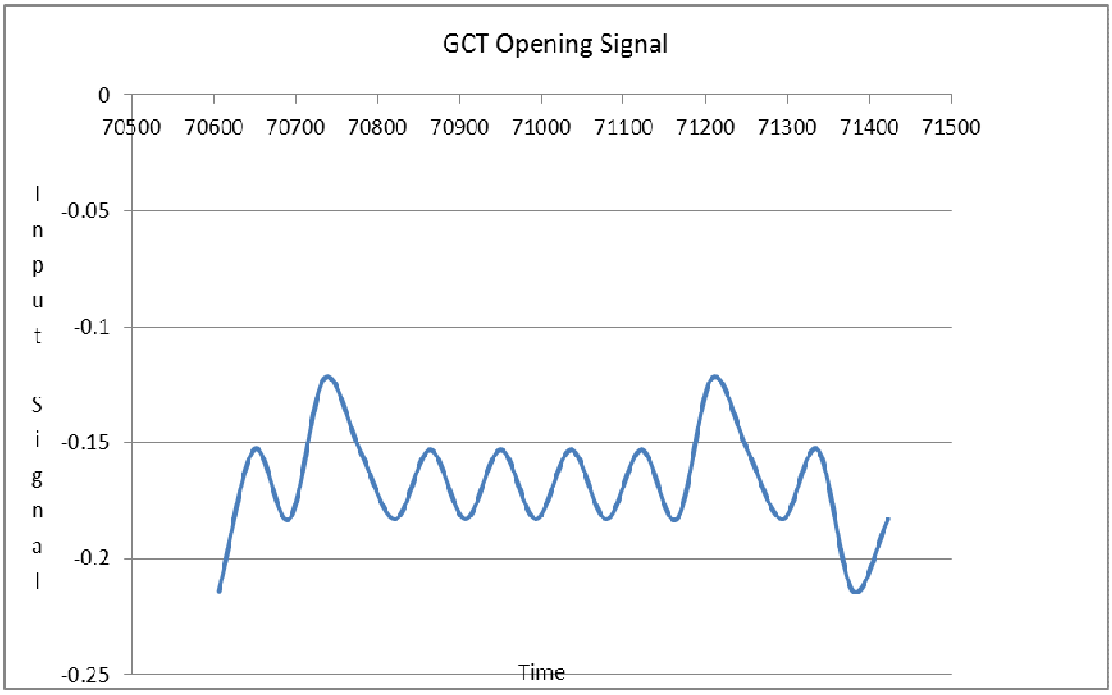

Figure 5.2**:** GCT Opening Signal (Plant)

During the steady-state operating condition, the thermal energy produced in the steam generators is transferred to the main turbine and auxiliary systems. When the energy produced by the reactor exceeds the energy removed by the turbine, provision is then made to remove excess energy from the system by the turbine by-pass steam dump system (GCT). Also, during the turbine load reduction, the excess energy produced must be removed from the system or in case of hot shutdown.

In Figure 5.2 above, the reactor trip which triggered the turbine trip will cause an increase in the steam generator pressures. The pressure will rise momentarily and the signal will be sent to open the steam dump valves. These are opened and closed corresponding to the pressure being maintained at the desirable reference point. The opening of the safety valves that may follow is avoided by the signal that prompts the opening of the steam dump valves as shown by the figure and helps bring the primary system temperature back to the zero-load value.

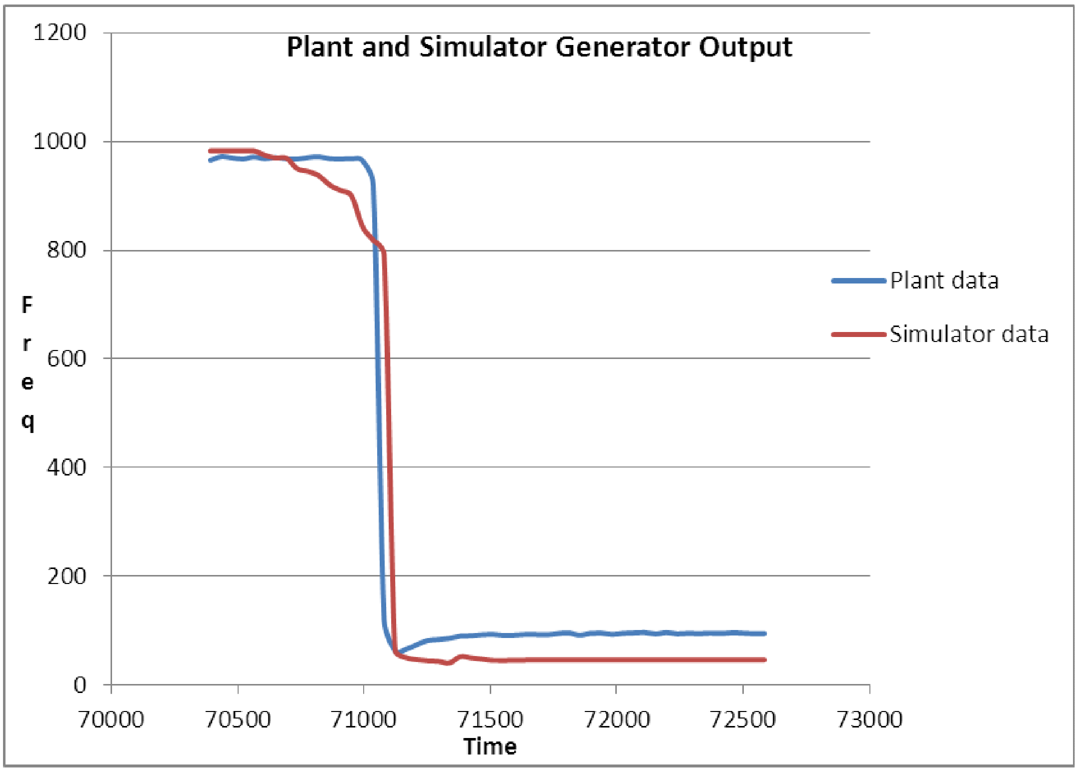

Figure 5.3: Generator Output (Plant)

During a steady state operation, electrical energy produced by the Turbo-alternator is transmitted through the generator transformers to the grid network. Some electrical and mechanical parameters on the units are recorded and stored by the Event Recorder System in case an incident occurs. The measurements are important for the supervision of the turbine and the alternator. The KKO is equipped with the oscillo-perturbograph which records and stores the main electrical parameters for the supervision of the alternator (generator) in an event of an incidence. Also, it is equipped with the tachy-perturbograph which records and stores the main mechanical parameters for the supervision of the turbine in an event of an incidence.

The data recorded can be used for root cause analysis and fault finding. There are several measurements that are recorded by the KKO system. Both analog and digital data are recorded. In this instance, the generator output frequency has been recorded for analysis. The output frequency of the generator has been operating at the optimal frequency until a disturbance was introduced into the system. The generator frequency output dropped close to a zero.

The figure 5.3 shows the frequency as it is recorded from the generators from both the reactor plant and the simulator. The difference in the frequency in two sets of data is due to transfer function being different as explained in the background theory of the two. The plant uses PI derivative and the simulator just PI. The derivative is introduced in the plant so that there are oscillations during plant transients for compensations. The simulator need not have the oscillations and its functions will reduce linearly.

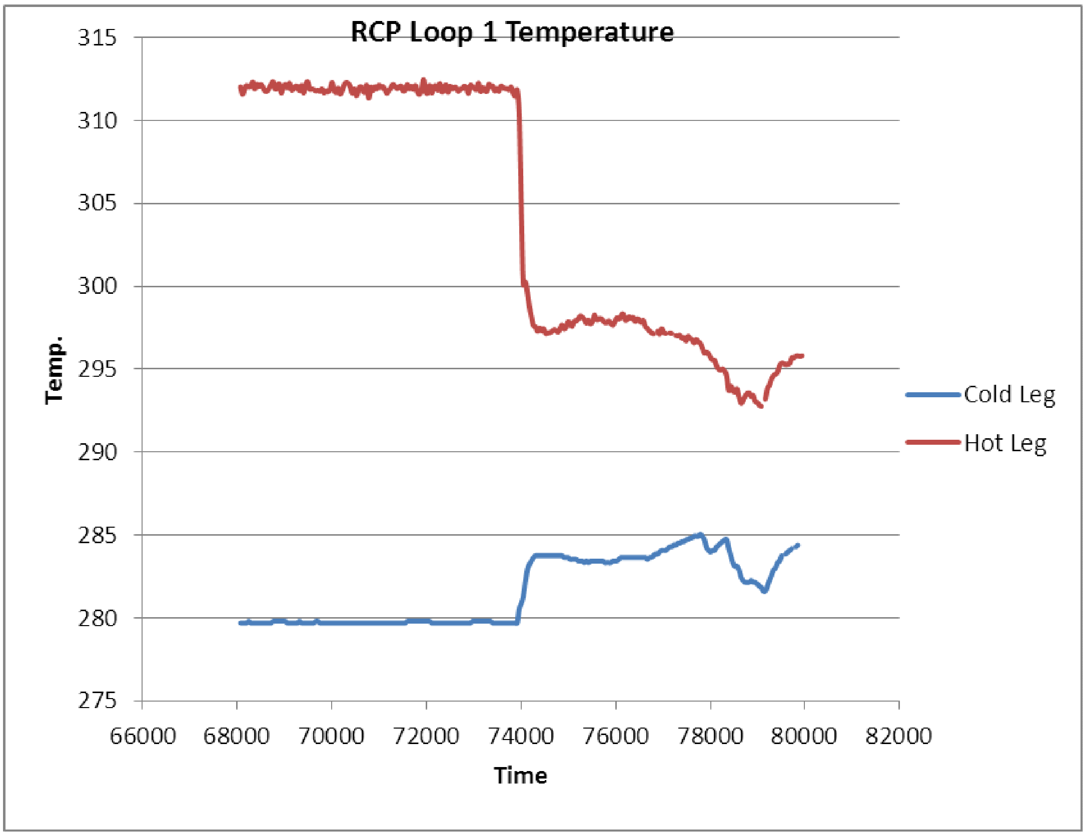

Figure 5.4a: RCP Loop Temp (Plant)

Reactor coolant system (RCP) is designed to transfer heat generated by the fission reaction from the reactor core to the steam generators when the reactor is at maximum power and to remove the decay heat during reactor shutdown. The primary coolant pumps maintain a continuous flow of primary coolant through the reactor core during steady state ad transient conditions. The heat is transferred by the reactor coolant through 3 closed loops to the steam generators where the heat exchanger takes place. On figure 5.4a, the plant data is plotted for the RCP loop 1 temperature of the hot and cold leg. Similarly, the simulator data is plotted for the RCP loop 1 temperature (figure 5.4b) and it also took the same output as for the plant data.

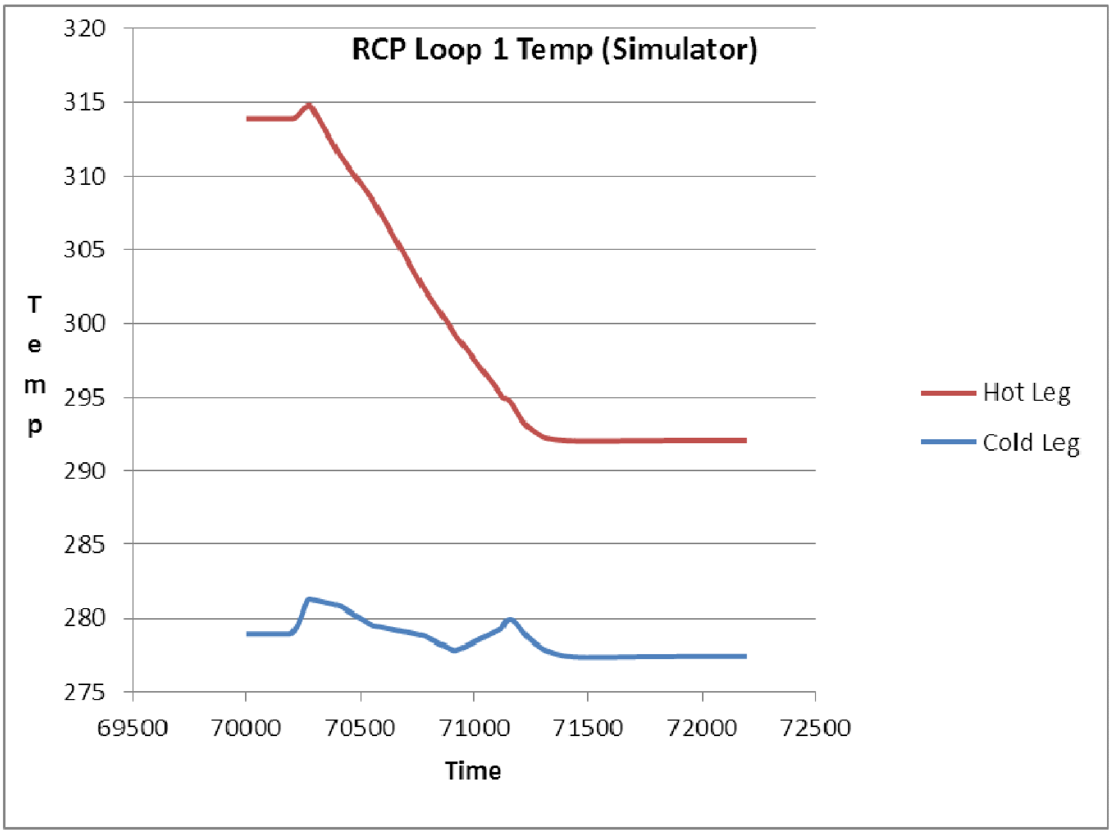

Figure 5.4b: RCP Loop Temp (Simulator)

From both sets of data, the plant had a constant hot leg temperature of approximately 312°c before the transient took place (at 100 percent reactor power output). The simulator had a constant hot leg temperature of approximately 314℃ before the transient was introduced. The difference in temperature between the two sets of data is that the simulator operates at optimum operating parameters, whereas in the real plant there are external factors that are affecting other parameters of the plant that can lead to the drop in temperature as it is evident in the results. The hot leg temperature drops gradually towards the cold leg temperature.

The plant has been operating for the past 27 years, and most of the main plant have not been changed or upgraded ever since. They will normally degrade in their operation; hence some output values are lower than design values. This will also require that a safe operating envelope can be calculated to determine when the reactor can scram once the values have been exceeded. The Koeberg NPP technical personnel adopted the advent of ORT programme (operating at reduced temperature), which also contributes to the lower than normal design operating temperatures.

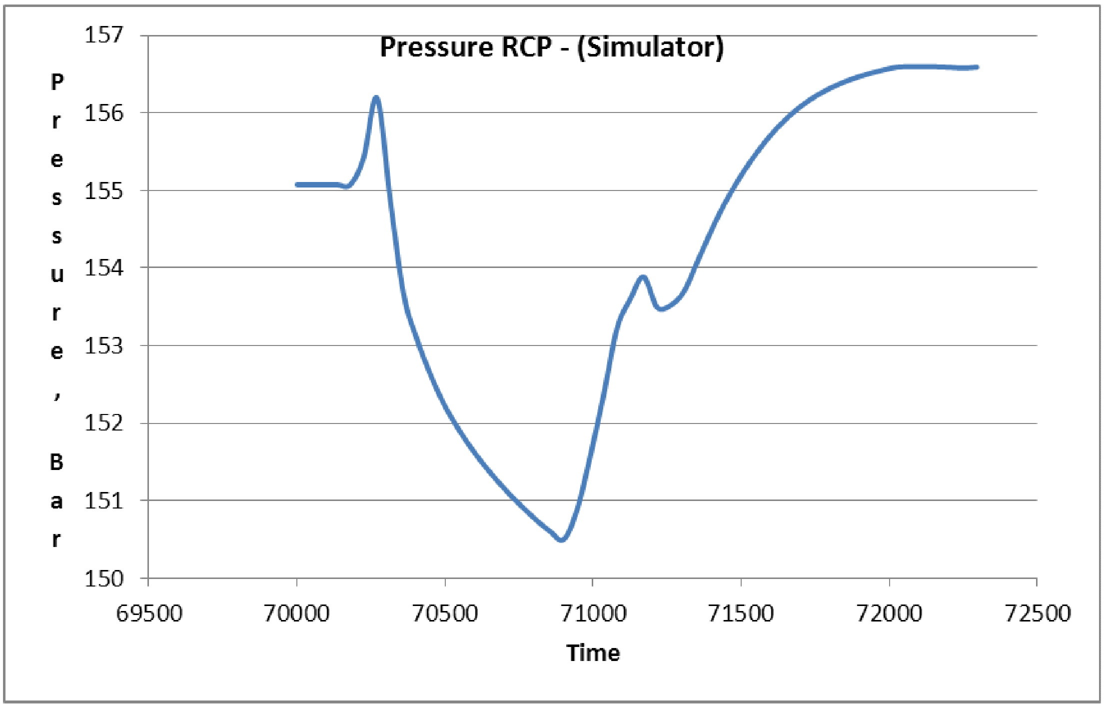

Figure 5.5: RCP pressure (Simulator)

The operating pressure of the RCP is 15,5 MPa during steady state operation. In figure 5.5 above, the pressure spiked to the maximum at reactor scram. The Protection and control system of the plant will counter the effects of the transient by initiating the plant safeguards systems. The pressuriser at this stage will try to maintain pressure within the operating range and to a minimum. The pressure will reduce linearly until the reactor stabilises. In the other set of data, that is not included here, can be illustrated that the heaters started so as to counter the drop in pressure (low pressure) and try maintaining a balance within the reactor to a desired operating pressure.

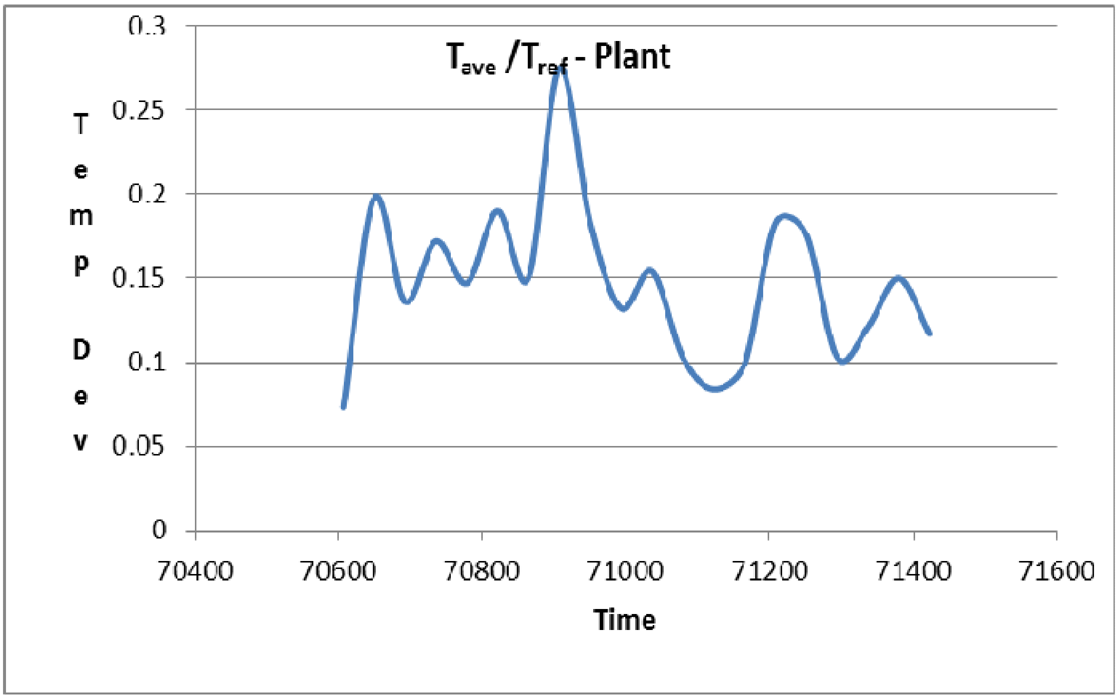

Figure 5.6a: T_{ave} /T_{ref} of the plant

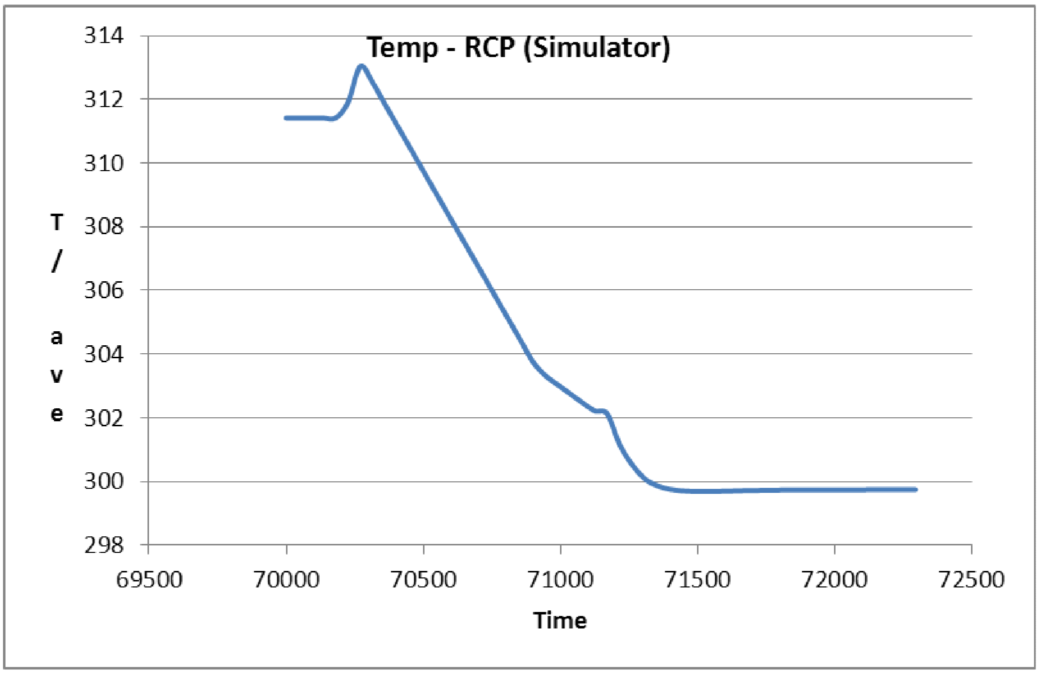

Figure 5.6b: RCP Temperature (Simulator)

The change in pressure of the primary system will affect the temperature of the system (T_{avg}). The temperature will follow the trend in pressure (figure 5.5 and figure 5.6a&b). Figure 5.6a depicts the deviation in T_{ave} /T_{ref} of the plant. When the pressure spiked, similarly the temperature of the system followed by increasing and it immediately reduced as the pressure of the primary system was maintained below operating pressure so as to suppress boiling during this transient. T_{avg} is the difference between the reactor coolant inlet- and outlet temperatures. The constant T_{avg} is maintained at all power levels. T_{avg} is reduced to a minimum in correspondence to the system pressure.

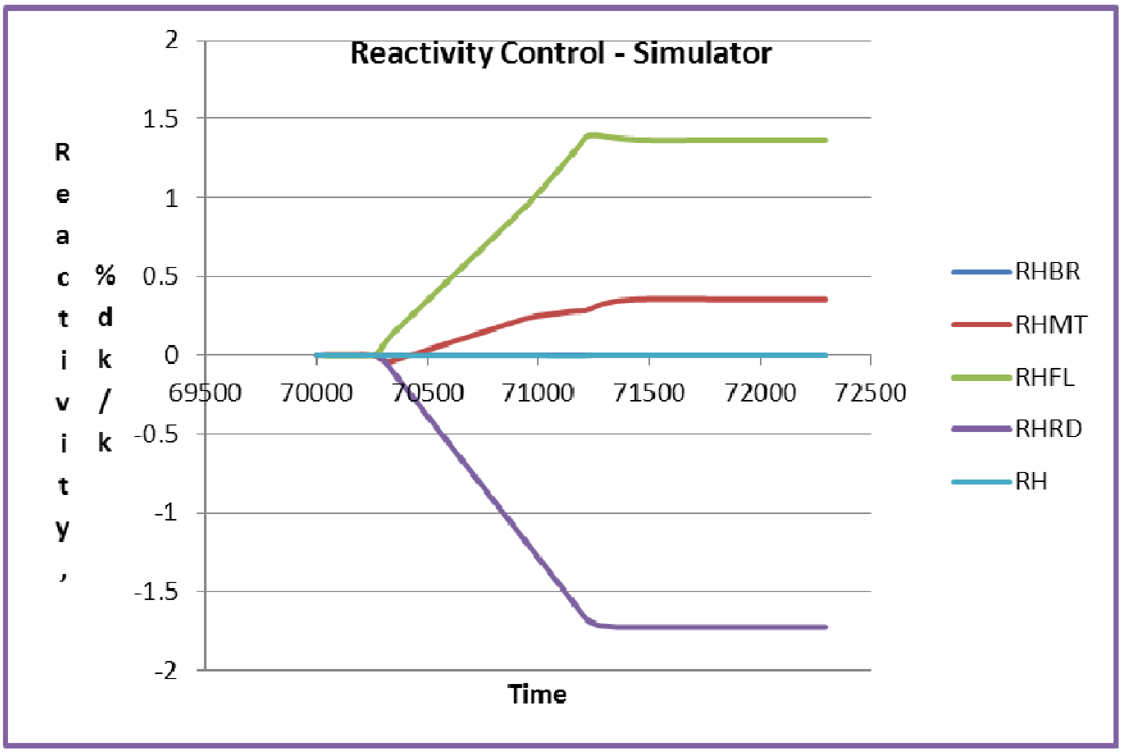

Figure 5.7: Reactivity Control - Simulator

RHBR: Reactivity soluble boron

RHMT: Reactivity mod temperature

RHFL: Reactivity fuel (Doppler)

RHRD: Reactivity rod

RH: Reactivity total

The reactivity control in the figure 5.7 above shows the balance that the reactor strikes to control the reactivity in the core. This is done to reduce the neutron flux of the core. The RCV (Reactor Chemical and Volume Control system) and REA (Boron Control System) are responsible for primary systems make-up, boration, dilution and chemical injection. These function are summarised by figure 5.7 in which the reactivity is reduced by these processes. The total reactivity of the system is zero.

All the simulator systems that were simulated using the KNPP data have shown reliable relationship with the plant data. That is to say, the simulator parameters are closely similar to the plant data. When the plant tripped, all the parameters responded in a manner as to shut down the reactor in a safe shutdown condition. Following the two set data from both simulator and plant being similar, the analysis drawn here will trace the fault for the reactor trip beyond the secondary of the plant. It can be concluded that even though the plant tripped and required to be shut down, the fault was not on the primary or immediate secondary systems.

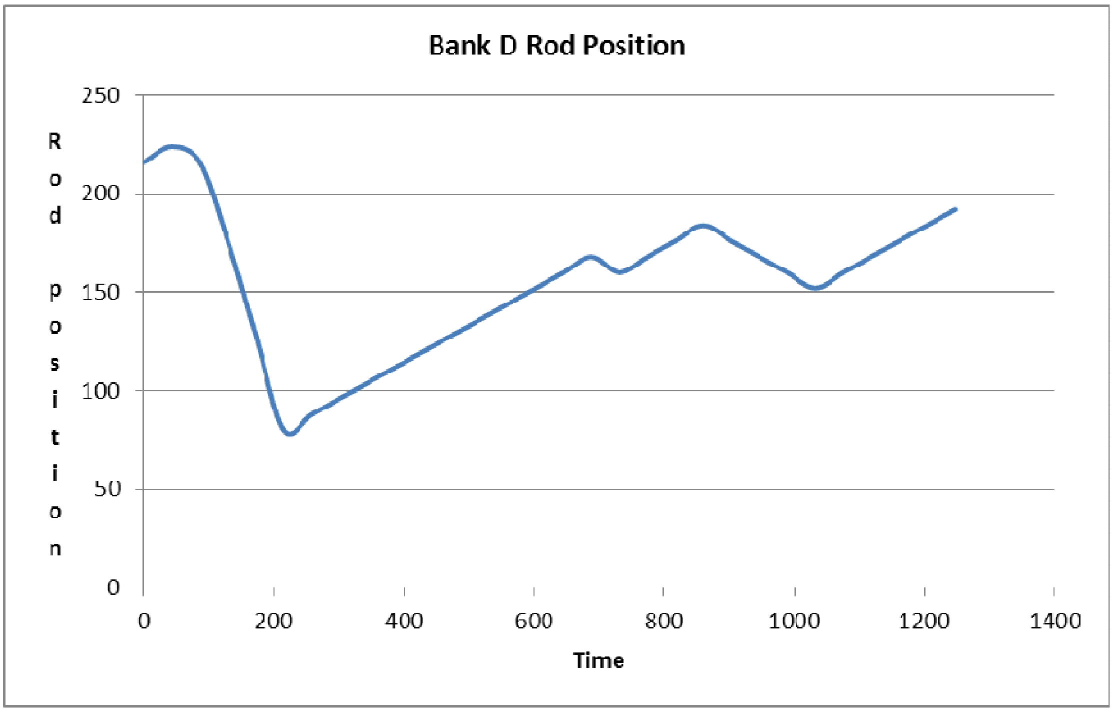

Figure 5.8: Bank D Control Rod Position (Plant)

The control rod drive mechanism in the core will be actuated so as to drive the control rods at a desired rate into the reactor core such that it changes the reactivity of the core. During SCRAM, rods can be dropped into the reactor due to gravity. In this instance, the control rods are driven by the CRDM (Control Rod Drive Mechanism) which moves at pre-determined rate until the required reactivity is achieved in the core. The control rods are inserted in the core to absorb neutrons causing fewer neutrons to fission and eventually to reduce core thermal power.

The control rod clusters consist of four control banks namely, A, B, C and D and two shutdown banks namely, SA and SB [0].The Control Rod Position Monitoring equipment which has provided the information in the figure 5.8 above provides the operator with the position of the control rods bank D during the transient. It provides continuous data on the position of each rod control cluster assemblies (RCCA) and also provides annunciator signal. Figure 5.8 provides the data of the position of the Bank D control rod cluster during the transient.

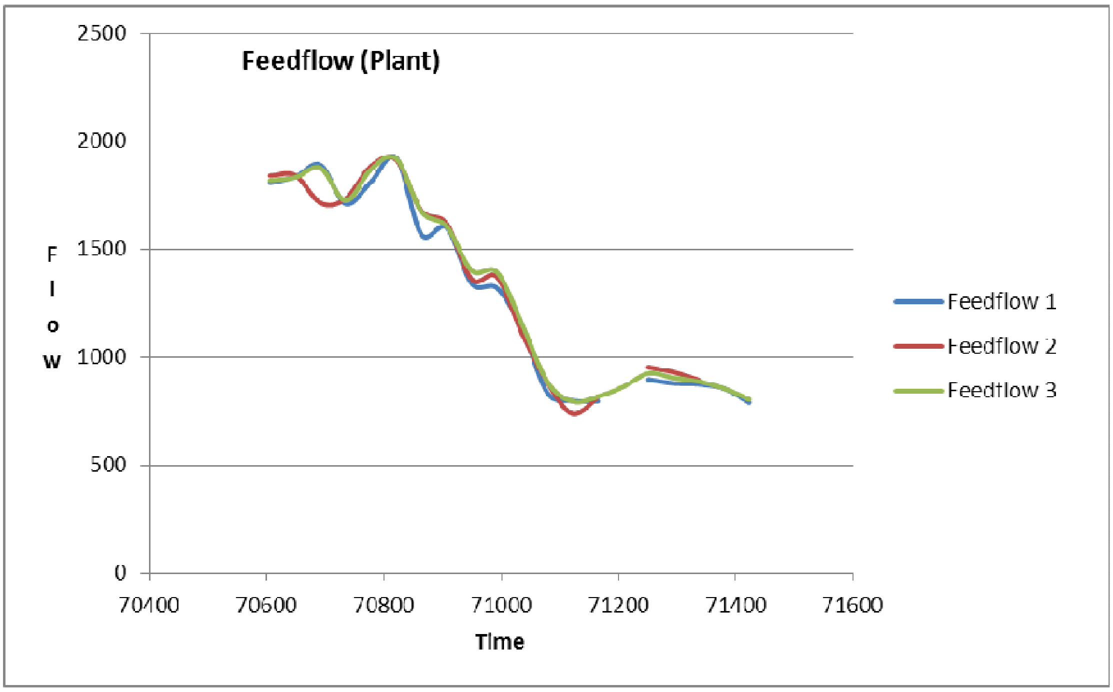

Figure 5.9: Feedflow (Plant)

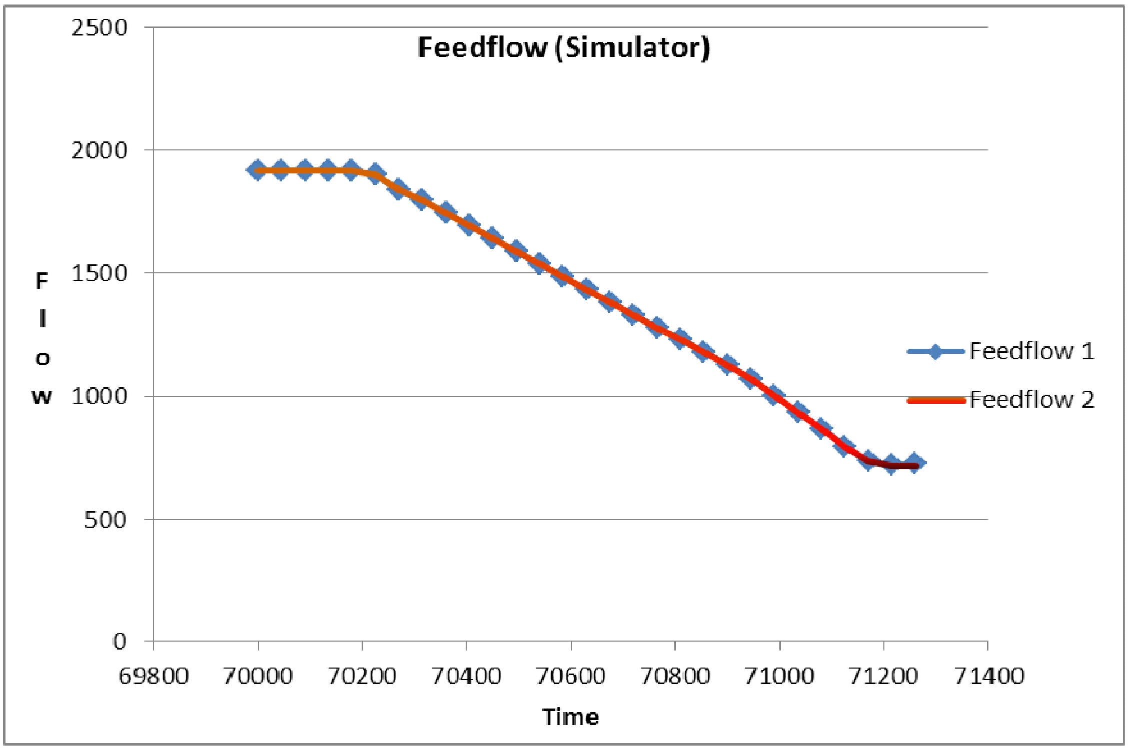

Figure 5.10: Feedflow (Simulator)

The reactor protection signals affect the feedwater flow control system (ARE). The purpose of the ARE system is to control the steam generator (SG) water level in a particular range by controlling the feedwater flow rate that enters the SG. When steam is extracted from the SG, their water level also changes. The feedwater flow signals are supplied by the two differential pressure sensors on the main supply lines to the steam generators. The readings are taken for all 3 steam generators. The trend on both figure 5.9 and figure 5.10 are similar. This is indicative of volume of water level in the SG. There will be a change in steam flow demand due to effects of shrink and swell in SG. The steam flow decreases and the level will increase. The level will initially decrease as SG pressure increases. Feedwater flow 3 on figure 5.10, is averaged as the loop 2 and 3 are lumped together.

5.2 Conclusion

The simulator has been validated to be reliable using the plant data based on the analysis of different plant parameters as they are shown in the results of this report. The plant was going through a fault. The fault was traced and identified for purposes of analysis. The same fault was introduced in the simulator by tripping it, with the same condition as the plant at that point in time when the transient started in the plant. The trend in the parameters analysed confirms the simulator to have the same trends as the plant. In some instances there is slight difference in the two sets of data due to the transfer function. That of the plant have oscillations introduced whereas with the simulator, there is none. The difference is within tolerance and wouldn't have any significant effect on our analysis. The simulator can therefore be declared valid and competent for further analysis of plant behavior.

CHAPTER 6

6.1 A: Fault Detection Process

In the following chapter, we will use the plant data used in the previous section and the simulator to detect a fault early in the system. The simulator will be operating in a steady state throughout the analysis whereas the fault is introduced in the plant operation. This will show that if there are no other external circumstances that influence the plant, there will be no changes in any parameters of a plant. The simulator data will be set in such a way that it is not influenced by any external factors and it remains steady throughout.

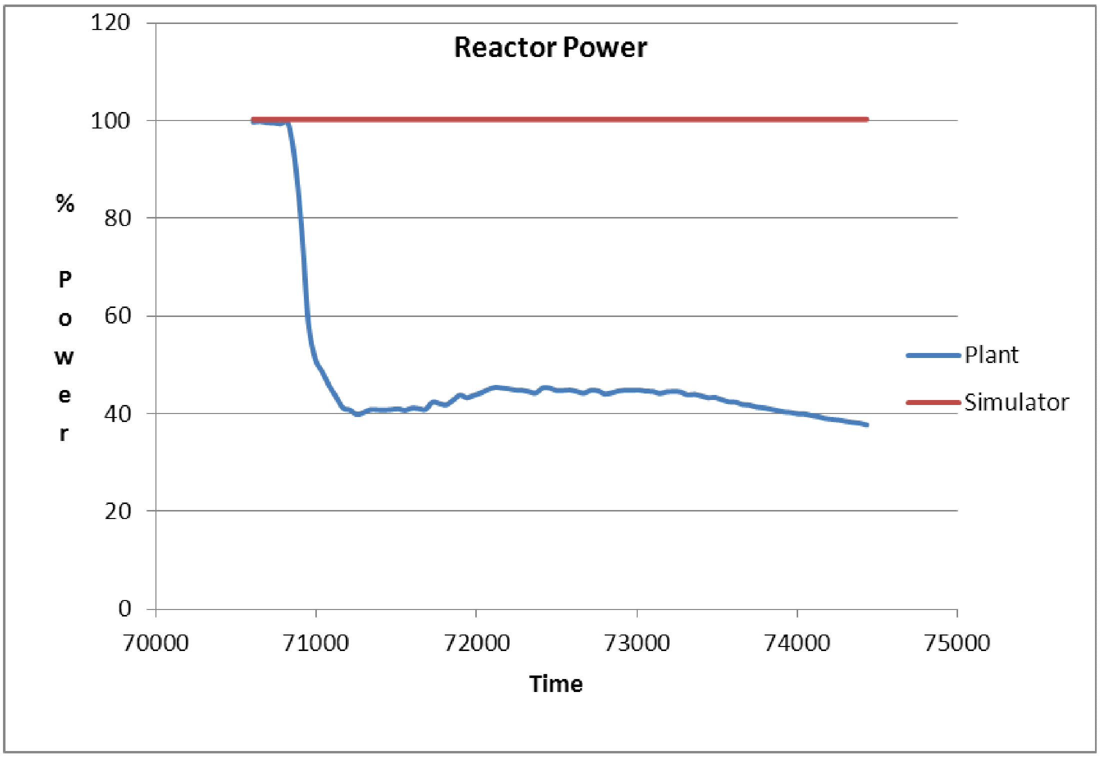

Figure 6.1: Reactor Power

In the figure 6.1 above, the plant has been operating steadily with the power output of 100 percent until the power started decreasing instantly. The plant is designed to operate within the design envelope of less than 10 percent of design value of a particular parameter before the reactor protection and control system can initiate a shutdown process. Both the power output of the plant and simulator are recorded at

100 percent until the plant experiences a dip in power and reduces to almost 40 percent power output before it momentarily stabilises again. The plant reactor power is reduced by 5 percent per minute.

On the simulator side, which had no external influences that the plant might be experincing, it (simulator) continues to operate at 100% power. At the moment the plant reaches 93.3 percent power, we can confidently assume the plant to be undergoing a transient.

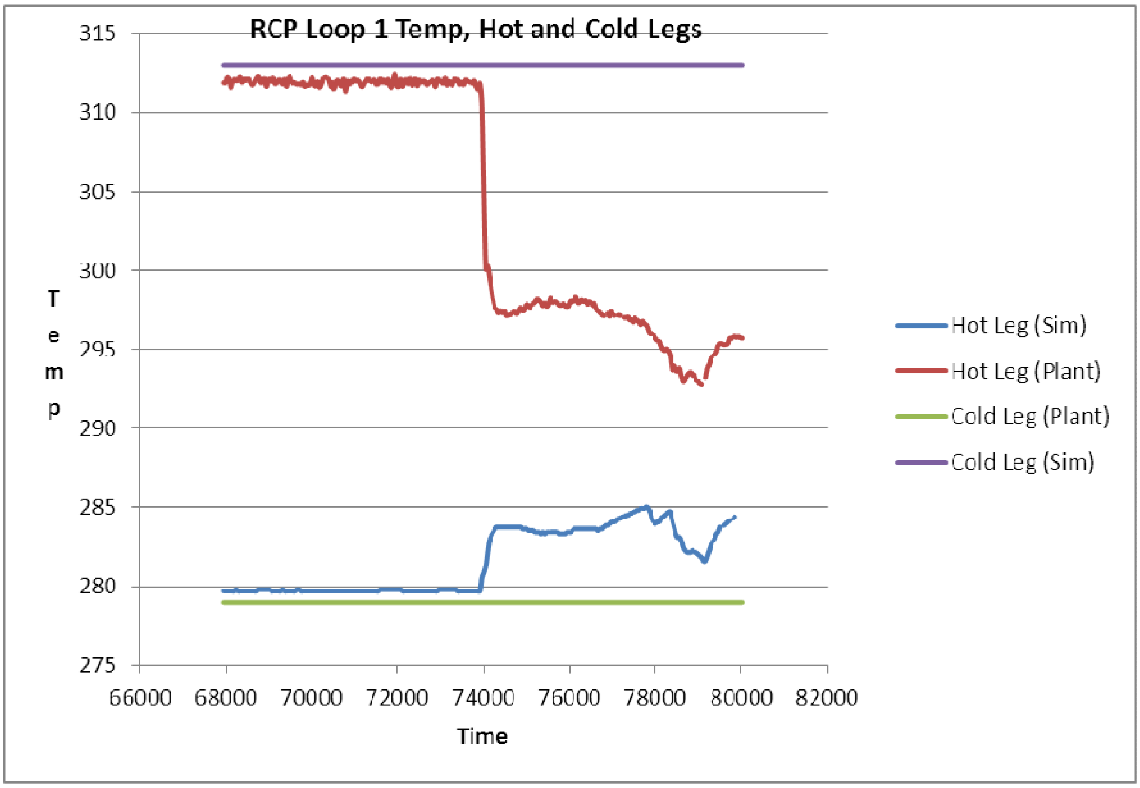

Figure 6.2: RCP Loop 1 Temp (Plant & Simulator)

The reactor power is controllled by controlling the average temperature. The plant is operating at the cold leg temperature of about 279°C and hot leg temperature of 312℃. At the instant, when power decreases to a mi nimum, the hot leg temperature also drops and cold leg temperature will increase. This will result in a decrease in average temperature. The plant control and protection system maintains the average temperature of 295ºC for the effectiveness of the plant. Once the T_{avg} narrows as is the case here, the effectiveness of the plant will be reduced. It can be noted that the cold and hot legs temperatures of the simulator are constant throughout.

From the figure 6.2, it can be noted that as the cold leg and hot leg temperatures narrows the T_{ave}, it can be deduced that the plant is in a transient.

$$T_{ave} = ½(T_{cold} + T_{hot})$$

Reactor can trip with low T_{ave}..For Koeberg NPP, reactor will trip if T_{ave}< 290ºC. 0.

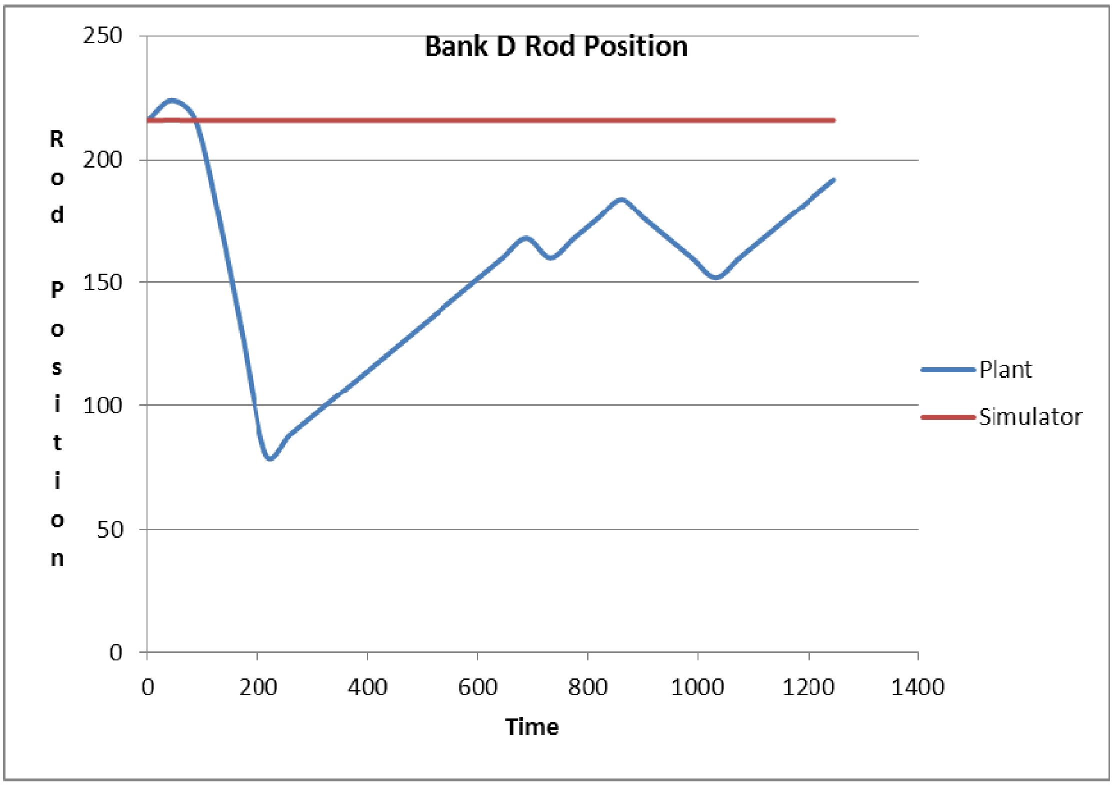

Figure 6.3: Bank D Rod position, (Plant & Simulator)

The control rod bank D are inserted inside the reactor core of the plant to change the reactivity of the reactor. Bank D are slightly inserted in the reactor core and are the first cluster to be inserted in the core during reactor shutdown. In the simulator, bank D is slightly inserted in the reactor core as per operating procedure. It can be fully inserted when the plant needs to limit the reactivity of the core as is the case in figure 6.3. Bank D is inserted to reduce the power output of the reactor.

Both temperature and pressure are the most important parameters that needs to be constantly monitored in the plant. The reactor power, when changed as is the case by inserting control rods, it will also have an effect on the loop temperatures of the plant. The temperature will change.

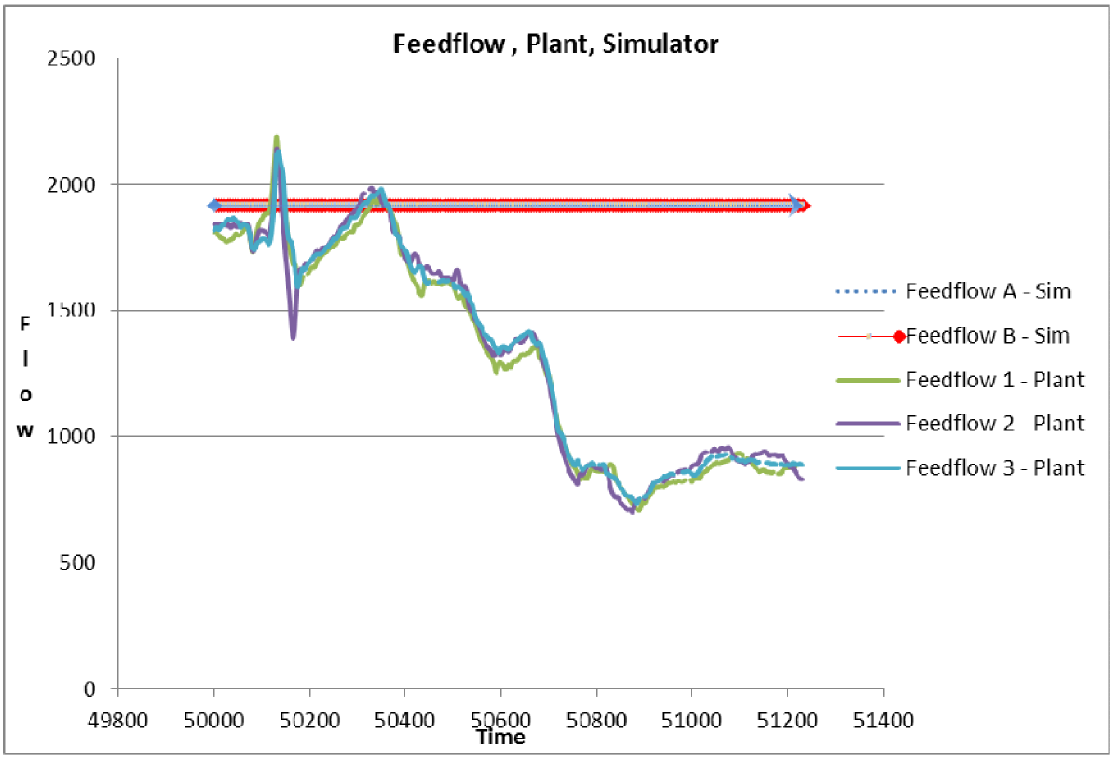

Figure 6.4: Feedflow, Plant & Simulator

The steam generators are continuosly supplied with feed water. This supply has a potential to over-cool the RCP system. As depicted in the figure 6.4, the feedwater decreases gradually in the plant representation. Decreased feedwater flow increases the feedwater pressure, which will create a steam differential pressure which differs from the set point. Feedwater flow control level signals are also used for generating some of the reactor protection system (RPR) logics. The feedflow water for the early detection system is constant and that of the plant decreases in time to drift away from the stable flow. It can e concluded that there exist a fault in the plant.

6.1.2 Analysis

The fault has been detected in the plant when the plant was operating at its maximum design operating limits values. Once these values were exceeded then the plant control and protection system initiated a plant shutdown process so as to counter the effects of the changes in the conditions of the plant. The objective of this work has been achieved by this analysis.

Table 5.1: Koeberg NPP Transient data (**Data used in plotting the transients**)

	Reactor Power	Gen Output Power	Tave-Tref Deviation	Reactive Gen Power	RCP Loop Temp Hot leg	RCP Loop Temp Cold leg	RGL Insertion Limit	Bank Rod position	Feedflow 1	Feedflow 2	Feedflow 3
Time	A_1RPN 416EU	A_1KKO00 1CE	A_1RGL4 11CA	A_1KKO00 2CE	A_1RCP030 MT	A_1RCP029 MT	A_1RGL413 CA	A_1RGL032 EUM	A_1ARE043 MD	A_1ARE044 MD	A_1ARE045 MD
70606	99.702	968.498	0.073	-74.725	312.03	279.808	121.868	216	1807.536	1840.732	1818.669
70649	99.849	970.696	0.198	-84.066	311.816	279.808	123.538	176	1835.901	1842.924	1832.82
70692	99.614	968.132	0.136	-79.121	311.688	279.808	123.399	128	1890.074	1714.464	1876.785
70735	99.527	967.766	0.172	-79.67	312.115	279.808	123.538	80	1710.221	1734.131	1724.326
70778	99.351	970.696	0.147	-81.868	311.346	279.808	124.165	88	1811.998	1884.513	1869.889
70821	99.702	972.161	0.19	-86.264	311.987	279.701	124.583	96	1923.944	1914.69	1920.164
70864	92.38	968.498	0.15	-79.67	311.795	279.808	125.209	104	1568.375	1678.292	1674.922
70907	78.379	967.766	0.275	-82.418	312.008	279.808	124.722	112	1600.479	1618.537	1606.018
70950	58.785	968.132	0.183	-76.374	311.902	279.808	125.348	120	1340.197	1359.339	1400.298
70993	51.257	970.33	0.132	-89.011	312.115	279.808	124.583	128	1320.167	1371.166	1393.942
71036	48.65	967.033	0.154	-101.099	311.987	279.808	124.443	136	1130.392	1108.029	1150.919
71079	45.809	112.454	0.103	-115.385	312.03	279.808	124.026	144	834.546	883.423	885.249
71122	43.525	62.637	0.084	-114.835	311.838	279.701	124.931	152	798.959	738.024	795.922
71165	41.24	65.202	0.099	-110.989	311.624	279.701	124.443	160	796.935	817.934	817.934
71208	40.801	73.626	0.183	-130.22	311.816	279.701	124.095	168	796.935	817.934	863.082
71251	39.922	81.319	0.176	-126.923	311.987	279.701	124.026	160	897.027	953.739	925.381
71294	40.244	83.15	0.103	-133.517	312.158	279.701	123.678	168	879.759	931.469	904.199
71337	40.859	85.348	0.121	-132.418	312.03	279.701	123.817	176	875.159	894.323	887.981
71380	40.83	89.377	0.15	-130.22	311.923	279.701	124.443	184	855.566	894.323	855.566
71423	40.771	90.11	0.117	-125.824	312.008	279.701	123.817	176	790.833	769.095	805

6.2 B: Transient Analysis

The reactor protection system safeguards all the safety-related systems in the plant. These systems are set at a specific pre-set limits which when exceeded will initiate a safe plant shutdown. In the following section, it will be illustrated when those limits are exceeded how the protection system reacted to new conditions. During normal operating conditions, the thermal energy produced in the steam generators is transferred to the main turbine and auxiliaries. Provisions are also made available for removal of this energy, when the energy produced by the reactor exceeds the energy removed by the turbine. Consider cases where the reactor is producing thermal energy that is not being removed by the turbine. These may be caused by:

- a sudden load reduction of the turbine,
- turbine trip.

In the following illustrations of this section, the plant data in which the plant had a fault and was forced to shutdown due to the persisting fault. The analysis will show that the fault was tracked from the secondary side of the plant and eventually tracked to the primary side as it progresses. The figure 6.6 below, shows 3 independent sets of data from the plant, simulator and reference simulator data.

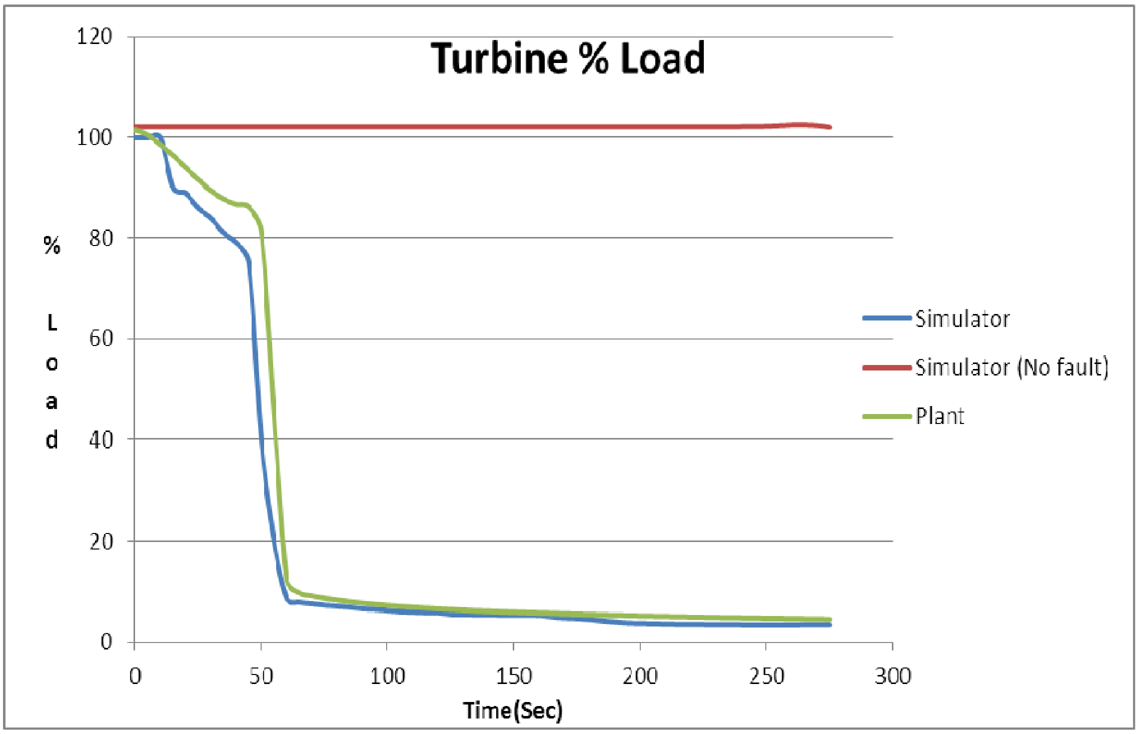

Figure 6.5: Turbine Load

In figure 6.6 above, the line illustrating the plant shows that the turbine tripped with a full load and reduced towards no load. A simulation was done to track the plant data as illustrated by the red line and similarly a reference data was done with which the simulator was running with no fault. The overall graph shows that at the origin where all the 3 sets of data were at 100 percent (maximum) then with the early detection system it could have immediately alerted the plant operators that the plant was under a transient when the blue plant line was further moving away from the orange reference lines. The plant(blue line) and tracked sim (red line) have the same shape but not super-imposed due to a slightly different tranfer function of the respective system. This can be seen in the figure below of a turbine and steam dump control for the simulator.

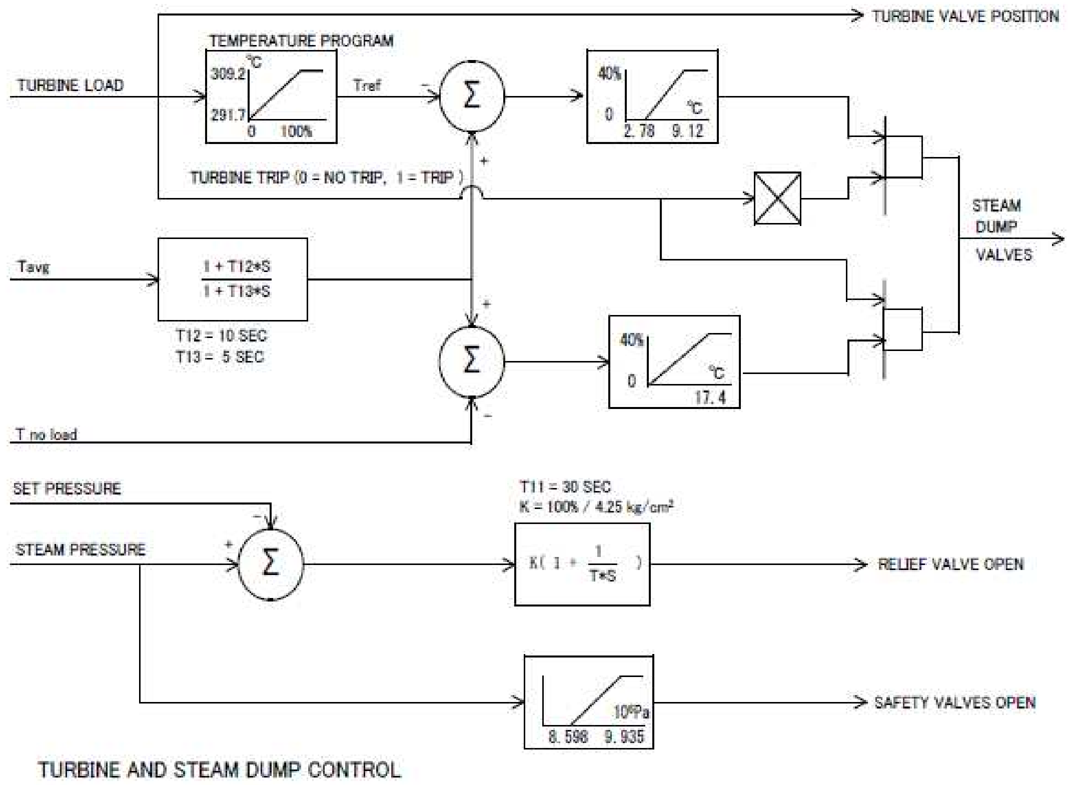

Figure 6.6: Turbine and Steam Dump Control (MST, 2009)

In a transients of turbine trip or sudden load reduction of the turbine, provision must be made to remove the energy mismatch between the primary and secondary systems. Figure 6. Above illustrates this concept. Also the transfer functions are given with respect to the simulator. The plant was tripped due to a turbine related fault that necessitated the turbine trip in a plant. This trip, eventually affected the primary side which also ultimately tripped.

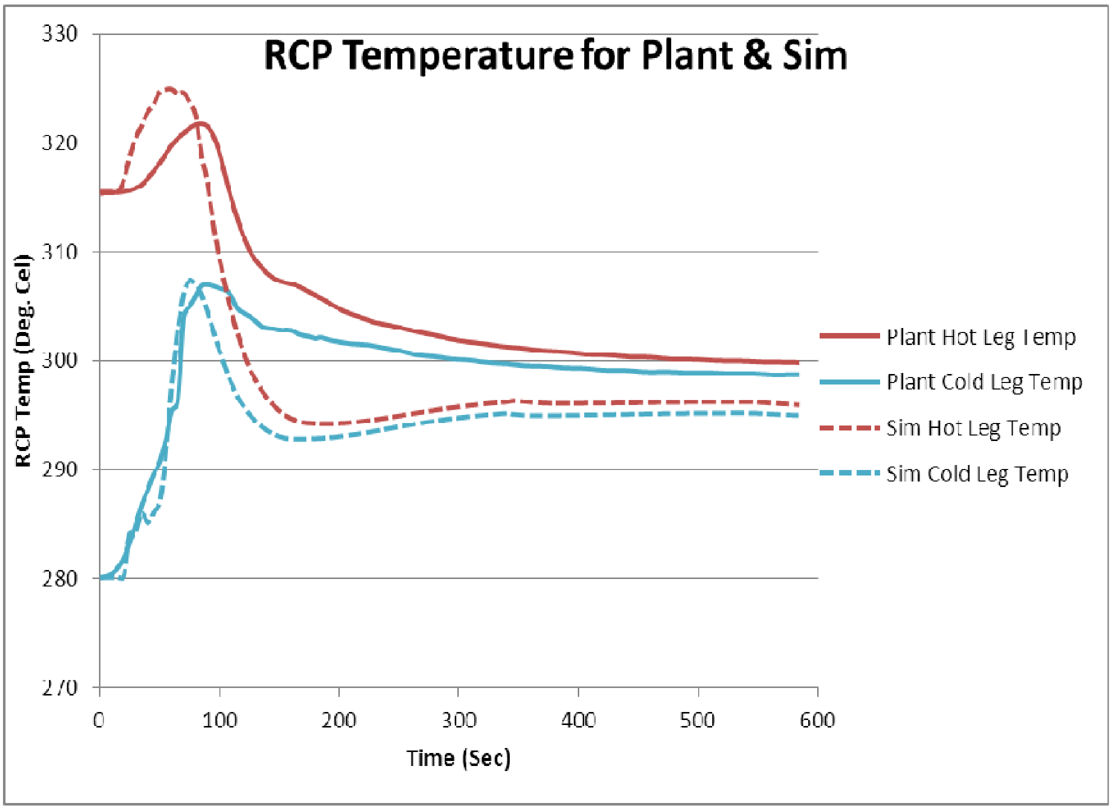

Figure 6.7: RCP Temperature for Plant and Simulator

The figure 6.7 above shows the temperature of both the simulator and plant. The plant was going through a transient. The temperature and behaviour of the plant was tracked with the simulator. A similar shape was obtained. It must be noted that the temperatures are not the same due to plant running on reduced temperatures. The temperatures spiked up wards and immediately decreased until they converged at a certain point towards the same outlet temperature. This also signifies the increase in pressure as it will be shown later on this analysis. The difference in temperature between any of the two legs (hot or cold) of plant and simulator is about 5ºC. This is a negligible value.

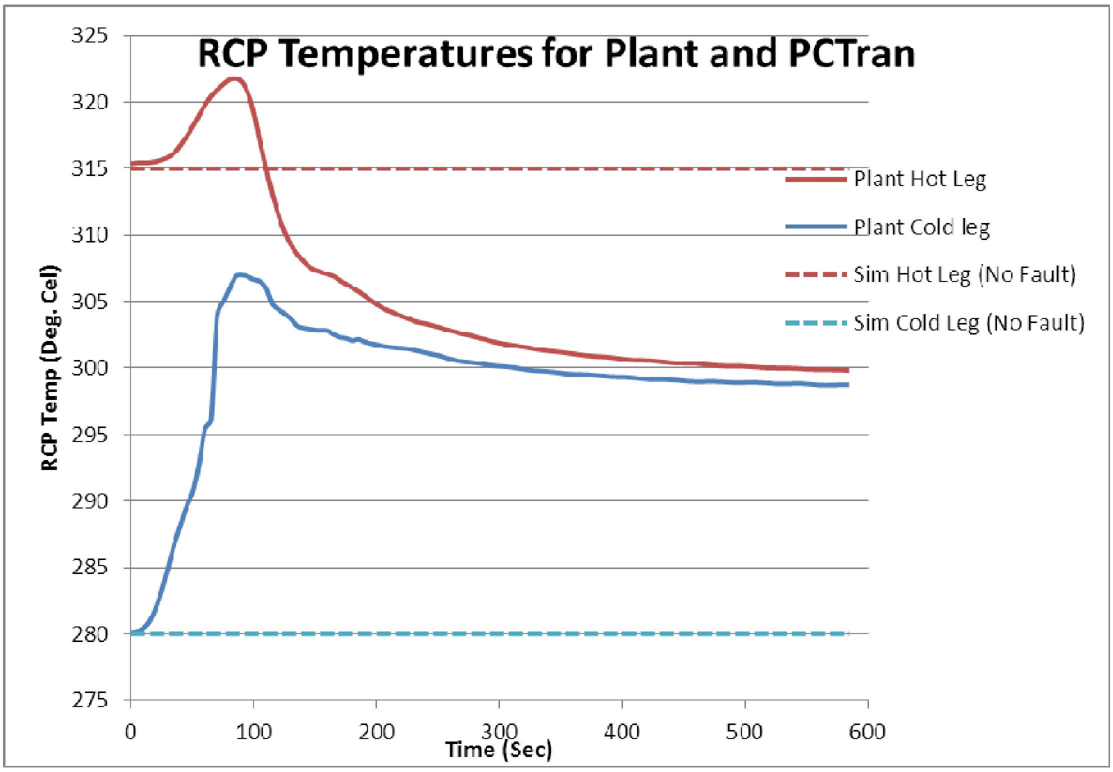

Figure 6.8: RCP Temperature for Plant and Simulator (Steady State condition)

In the figure 6.8 above, the reference temperature is the constant temperature of the simulator with no built in fault which is a reference temperature. This illustrates that, the plant temperature was moving away from the reference due to transient. The RCP reached and exceeded its operating limits and reactor trip was initiated. The reactor protection system tries to maintain the operation of the plant within the safety limits all the time. The temperature in the primary system increased due to not enough heat being removed from the primary system. This led to unstable condition in the primary system.

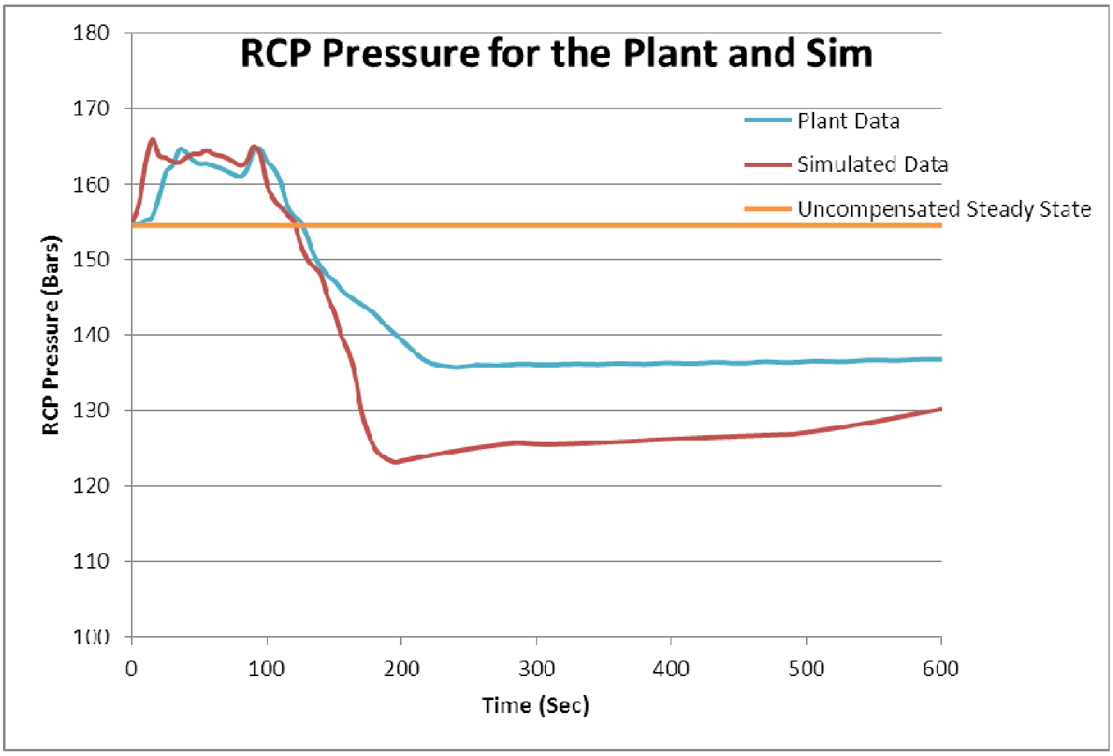

Figure 6.9: RCP Pressure

The purpose of the pressuriser in the reactor is to maintain the primary system at its set-point value of 155 Bars in order to avoid any boiling and it prevents the RCP against high pressure gradients during transients. The RCP system has high pressure settings. The power operated valves (PORV's) will open at set-point of pressure reaching 161 Bars and the safety valves lift at design pressure of 170 Bars. In the figure 6.9 above, the pressure spiked to 165 Bars before it was reduced momentarily and increased to same pressure again.

Pressure is a function of temperature, as it has been illustrated by figure 6.7, 6.8 and 6.9 that when temperature increases, the pressure in the vessel will also increase. The pressuriser pressure control system maintains the primary system at 154 Bars regardless of the reactor power level and coolant average temperature. The Limiting Conditions of Operation (LCO) for the Koeberg plant specifies that a high signal to Reactor Protection System (RPR) of 164.5 Bars will initiate a reactor scram. This is true as shown by figure 6.9. The reactor was scrammed.

The turbine stop valve will be closed and this causes over-heating in the primary side. The reactor pressure, pressuriser level and coolant average temperature increase until the reactor is tripped as it has been shown on the above figures and below.

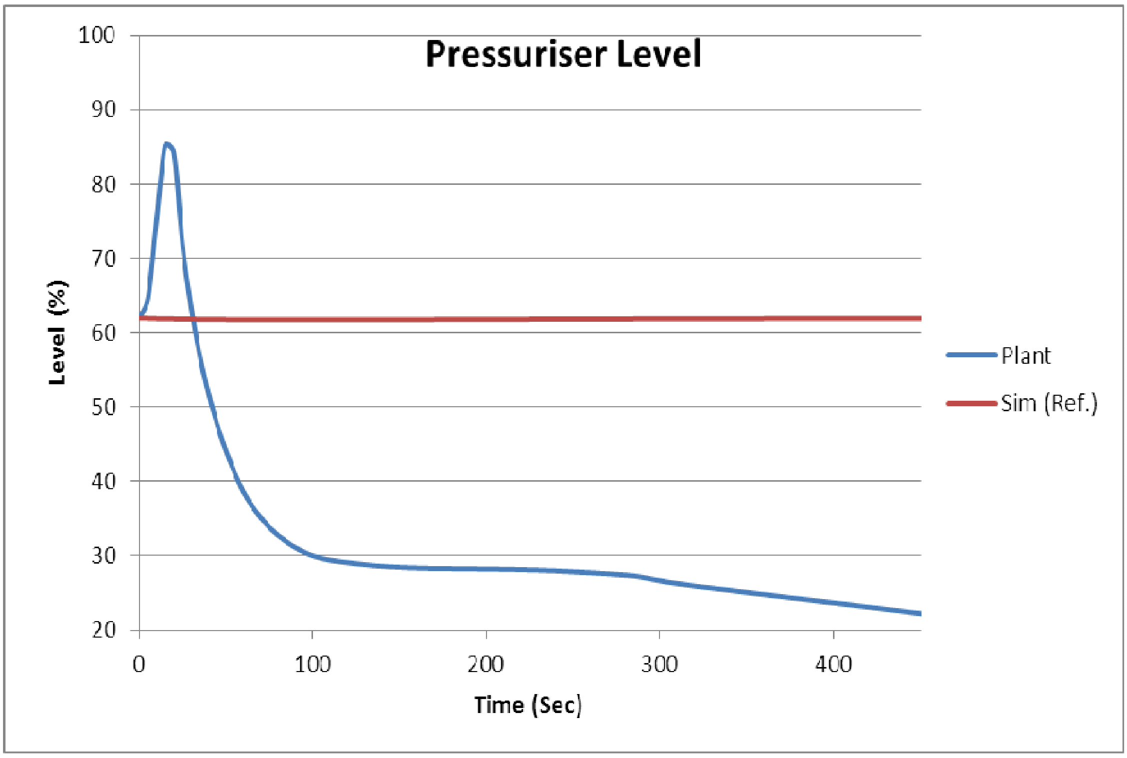

Figure 6.10: Pressuriser level

The pressuriser contains water and steam in saturation so that the correct liquid-vapour ratio exists so as to control pressure of the RCP during normal and transients plant operations. To make the water inventory in the reactor coolant system as constant as possible, the pressuriser level need to be maintained. This is maintained as a functiion of reactor coolant average temperature. The PORV's and safety valves release steam from the pressuriser steam to lower or to prevent further increase in the RCP system pressure. In figure 6.10, the pressuriser level of the plant decreases as the pressuriser releases steam to help maintain RCP pressure to within set-points. Also, the reference pressuriser level from simulator data shows a constant level inventory. The pressuriser level is directly related to Tave.

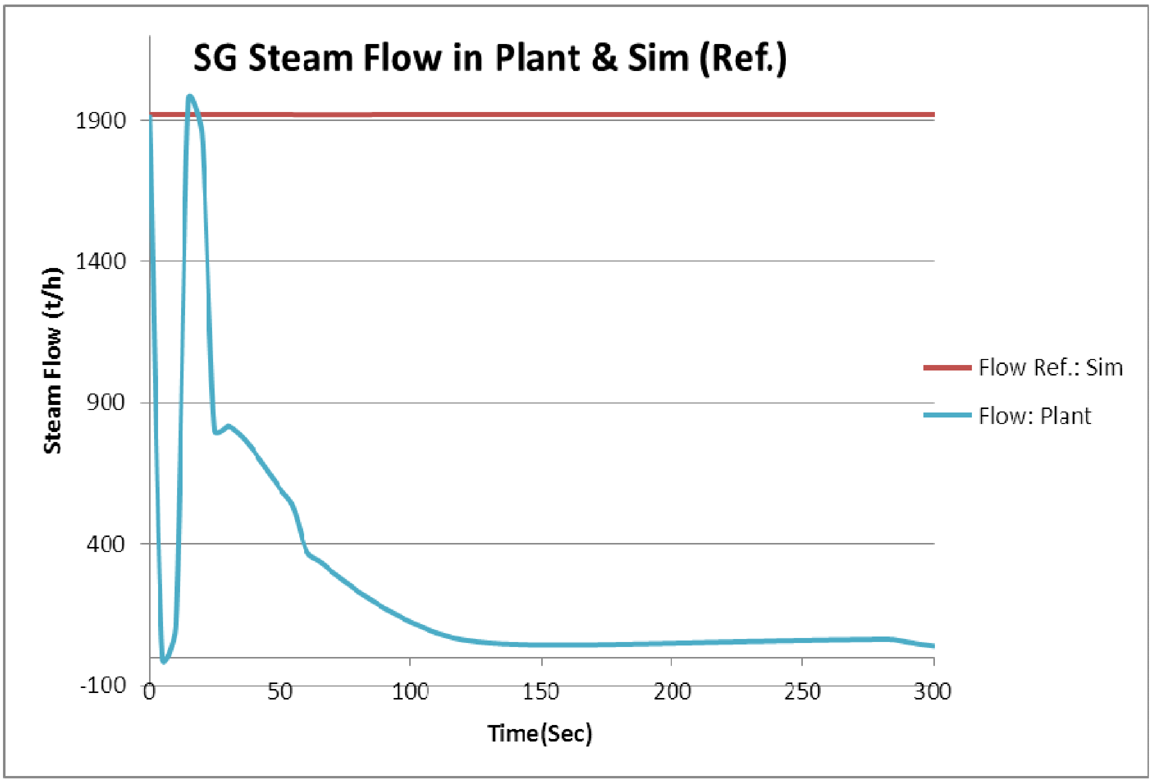

Figure 6.11: SG Steam Flow in Plant and Simulator

The Steam Generator (SG) produces steam at a required pressure, quality and flow to drive the main turbine. The steam flow rate from the SG to the turbine system is shown by the figure 6.11 above. From the plant data, the flow rate decreases from the normal of about 1800 t/h. this is due to the turbine trip. The demand of steam to the turbine is reduced. The reactor trips by steam/feedwater flow mismatch of 726 t/h which is 40% of rated flow. The pressure drop means a drop in Tsat. The mass flow in the riser of the SG will reduce. The shrink and swell are transition changes in the SG level due to change in average density of steam-water mixture. As steam flow changes, pressure changes as well. The mass flow of the reference (simulator) is constant. The plant mass flow line moves away from the reference line, which signifies a plant going through a transient.

After reactor trip, the SG level drops especially when the reactor was running at full power.

6.3 C: Transient Fault Detection Process

The transient was introduced in the system so as to illustrate the effectiveness and reliability of the early fault detection method. With the initial fault still in the system as in the previous section (turbine trip), then a transient was additionally introduced in the plant system. This was done with confidence as the reliability of the simulator was established in above sections. Therefore the transient fault was introduced in the simulator as it will act as a plant for this analysis. Both the graphical representation of the initial fault and transient malfunction are presented hereafter. The loss of coolant to the cold leg was simulated as a transient introduced.

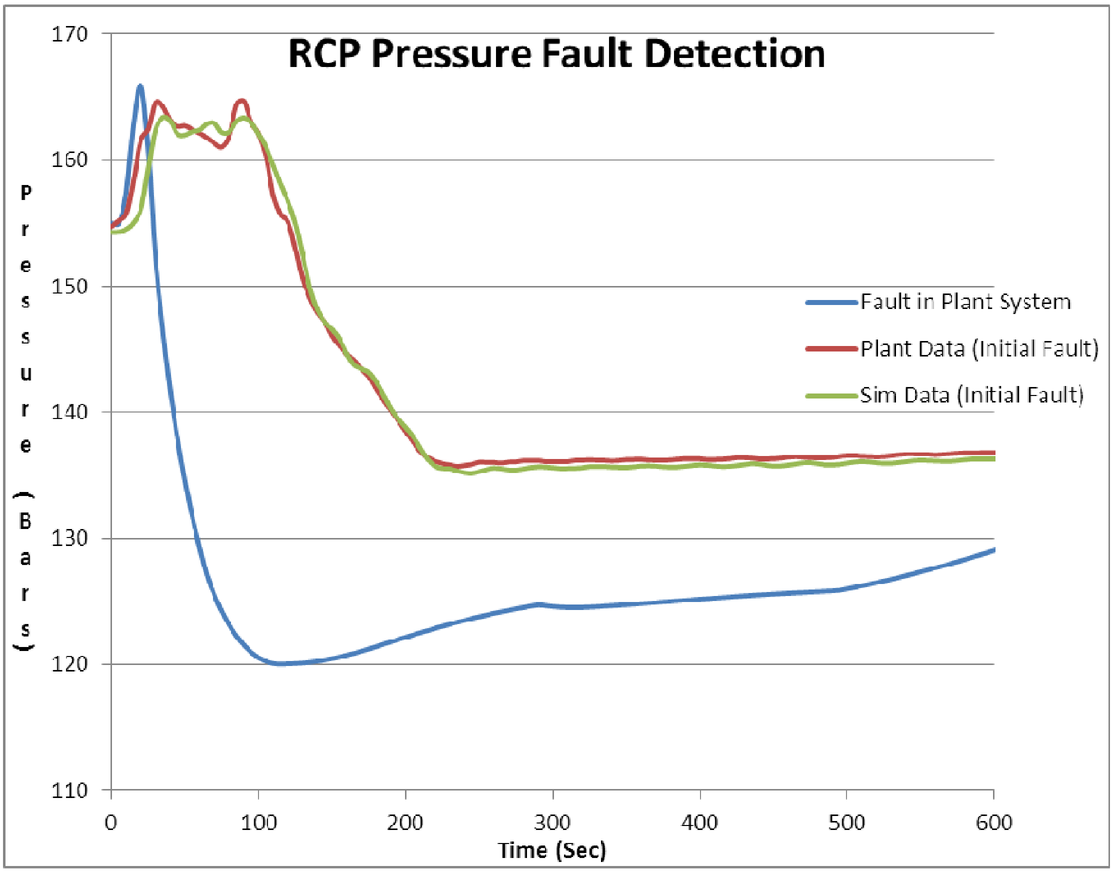

Figure 6.12: RCP Pressure for Plant with initial conditions and transient.

The above figure 6.12 shows the fault detection method. The plant and the simulator on the which have graphs similar to figure 6.10 are shown with an aditional graph when the plant was undergoing an additional transient. The blue line on the figure depicts a change in plant condition. The change is seen to be showing earlier than the intial fault. The drifiting of the blue transient pressure line from the initial line, shows the presence of a transient in the plant. The operators would be monitoring all the displays in the control room. This clearly illustrates an additional condition that has changed in the plant that necessitates the control system to react in a different manner. It shows that, there is a an additional transient that the plant is responding to. The presure on initial graph are similar and tracks each other as explained previously but additionally the fault has been introduced that emphasises the effectiveness of the early fault detection method. It is clear that the pressure is way outside the safe operational band of the plant. This illustrates another fault in the system which according to plant response, there was a loss of coolant in the primary system of the plant.

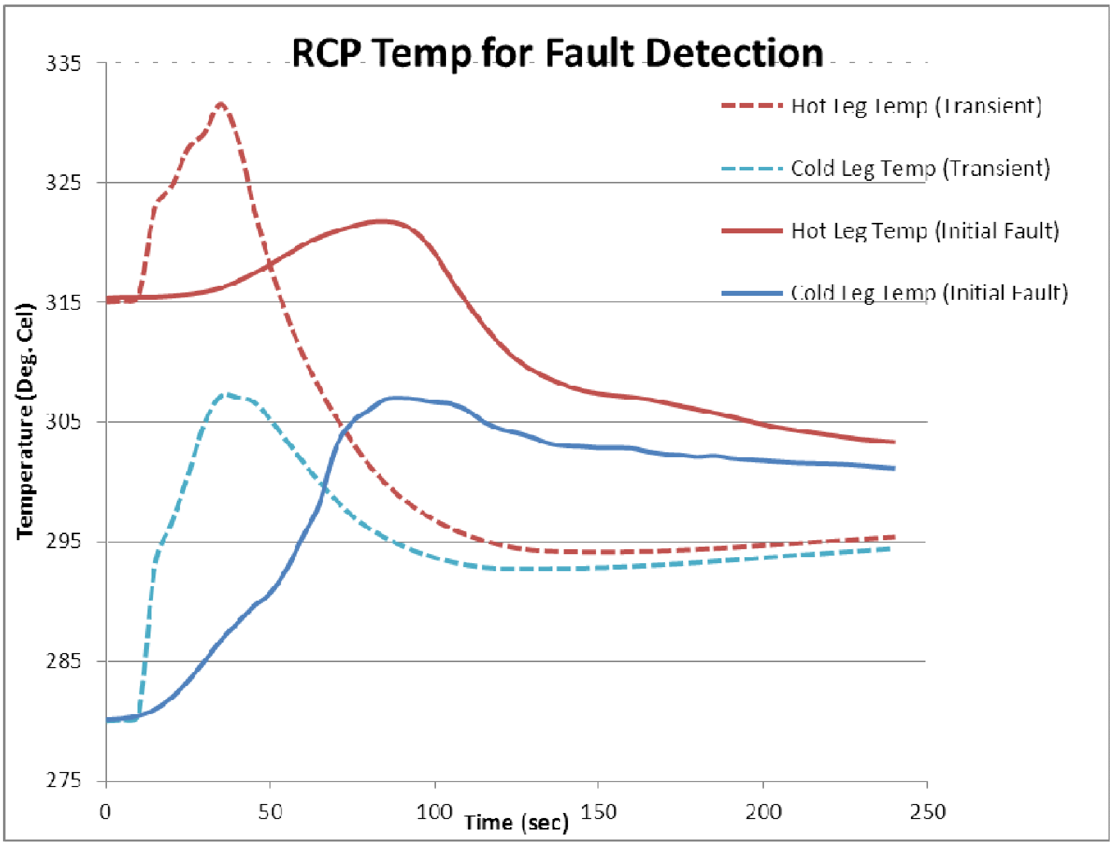

Figure 6.13: RCP Temperatures in a system with additional transient

The RCP temperature which is closely related to pressure changes in the system is following the same trend as figure 6.8. in the figure 6.13 above, the RCP temperature rises sharply and rapidly when the transient is introduced. The sharp rise confirms that the system is having an additional transient. Depending on how the system pressure is maintained and changes, then temperature of the system will follow those changes.

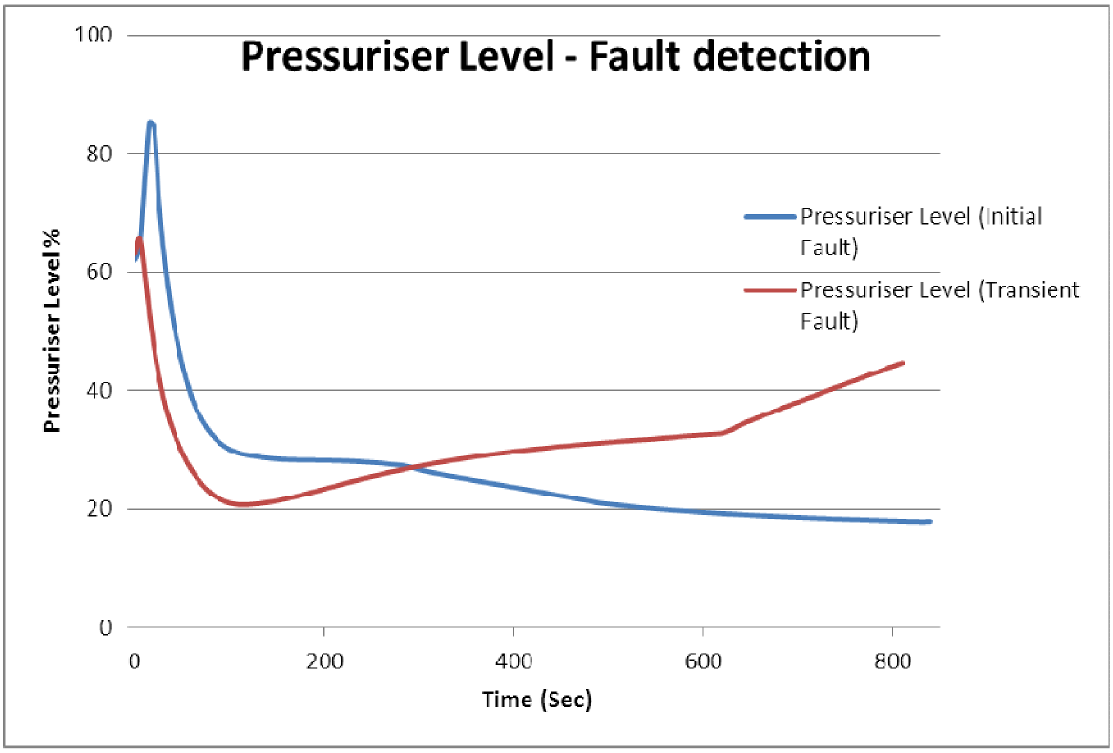

Figure 6.14: Pressuriser Level in a system with a transient

The pressuriser level in a system changes rapidly with the initial fault that has an additional transient like a LOCA (red line) as illustrated by figure 6.14 above. This shows that during a LOCA, other safety systems will immediately be actuated into action. When the level changes by more than 5 percent of the set pressuriser level, then the alarm is triggered for the attention of plant operators. The Control System will try to maintain the stability although the condition of the plant is changing rapidly. In eventuality, the reactor will scram after it has been operating outside of the safe operating band conditions.

The proposed system is introduced not to replace any of the existing plant protection system but it is an additional redundant system. The plant could respond in a way as to be in a fail-safe condition and the early detection system's role and objective only being to timeously and very early identify faults and or transients in the system.

6.3.1 Analysis

The above analysis illustrates the transient that initiated in the secondary part of the plant but due to interfaces in the plant it has affected the primary side of the plant. The failure of the secondary side to remove excess heat has ended with the slow heat removal causing the rise in temperature in the primary side. This rise in temperature has caused direct rise in pressure of the RCP which has initiated a reactor trip.

Due to the initial turbine trip, the heat removal from the primary side was not adequate. The fault in the system was traced to beyond the turbine. The turbine tripped and subsequently, the reactor was scrammed.
With the proposed system, the simulator operating alongside the plant will run independently from the plant. From the control room, the operators can monitor any deviation from the reference(simulator) and take any necessary action to avert plant shutdown especially when the fault could be traced beyond the secondary side.

Recalling the equation:
$Y_U(s) = Y'_P(s) – P_U(s)\ (U_P(s) – U_S(s))$,

Of which it was derived with help of figure 2.1. The above analysis for fault detection was shown to follow the exact theoretical concept as explained previously. The measured pressure from the plant compares it with a reference value through a comparator device. The error difference is fed into a control system that will determine which system to action for countering the on-going transient or fault depending on the fact that pressure is increasing or decreasing.

Whilst the plant was undergoing a fault, an additional transient was introduced to illustrate the early fault detection method. From the pressure and temperature of the RCP, it is shown that the plant was reacting vigorously to an additional transient so as to maintain safe operating condition. On persistent of this faults and operating outside the safe limits, the plant initiated a reactor scram. The plant initial condition is fed into a comparator along with the new condition (transient) and when the error is detected, it is then analysed for the plant to respond in a way that could promote safety. With the operators monitoring all the displays in the control room, wherein the proposed early detection system would be stationed, they would have a timely and prompt intervention to any persisting changes to the system parameters.

Table 6.1: Fault Groups with initiating measurements (Cilliers 2012)

Fault group / Measurement indications	Uncontrolled control assembly withdrawal in subcritical cold condition	Uncontrolled control assembly withdrawal in subcritical cold condition	Inadvertent reduction in RCS boron concentration	Loss of reactor coolant pumps	Primary loss of coolant	SG tube rupture	Leaks and breaks in the MS system	Leaks and breaks in the Feed Water system	Loss of off-site power	Loss of main heat sink	Inadvertent closure of secondary loop isolation valve
Intermediate range neutron flux > max	x										
Intermediate range neutron flux doubling time > max	x										
Power range neutron flux rate of change > max		x	x								
Reactor thermal power > max		x	x								
RCP speed < min for > 1 loop				x					x		
PRZ level > max	x	x	x	x				x	x	x	x
PRZ pressure > max	x	x	x	x				x	x	x	x
PRZ pressure < min					x	x					
SG pressure > max				x					x	x	x
SG pressure drop > max							x				
SG pressure < min							x		x		
SG level A < min							x	x	x		
SG level A > max						x					
Containment pressure > max					x		x	x			

For a loss of primary coolant transient, as shown on the table 6.2, the fault will only show in the pressure and temperatures of the primary system. Some of the parameters will change due to the changing pressure such as the level in the pressuriser. The mass flow rate will also reduce.

The loss of coolant in the primary side was introduced in addition to the initial fault condition. The response of the plant (simulator) as shown by the rapid changes RCP pressure and temperatures has shown that the early detection system as described and proposed is functional and proved to be working well. The simulator and plant results are mostly affected by the difference in the transfer function of the respective systems. These differences are in most cases small and can be neglected. In the last section, it is shown that the fault or transient in the system can be shown early on. These illustrations was shown by keeping the simulator steady and compare it with the plant which is going through a transient, as a result the plant parameters are continually changing as was shown in the section.

CHAPTER 7

Conclusions and Recommendations

7.1 Conclusions

The illustration represented here is applicable to generation 2 nuclear reactors similar to Koeberg NPP and would benefit the operation of such to be used effectively and efficiently. The operators of such plants will enhance confidence in their safety systems and the public in general. The proposed system will improve the operation and reliability of the entire control and protection system. The simulator was proved to be reliable and giving a proper desired and expected results. The system introduced here is a useful tool for diversifying the safety systems in the plant and augers well with defence in depth as is the safety culture in nuclear power plants.

All kinds of plants go through transients all the time. The Reactor Protection System and Plant Control system have an objective to counter the effect of any transient when operating limits are reached and exceeded. It acts swiftly to reinstate the balanced condition of the plant without compromising any safety to all other safety related systems. The simulator has been validated in other unrelated studies using RELAP code, MAAP code and other nuclear related codes.

The simulator results follow exactly what was obtained in the plant during the steady state and transient condition as it has been shown in this analysis. The simulator can be qualified as reliable and represents the actual plant data and behaviour during accident (transient condition). The simulator (PCTRAN) can run parallel, alongside the plant Control and Protection system as proposed in figure 2.1 without any disturbance to the real plant systems. This system should not have any bearing or influence on existing safety related systems in a plant but becomes an additional system that could only be a reference to plant operators and personnel.

This study has achieved its objective by initially qualifying the simulator (PCTRAN) as a reliable tool. It further shown that it could be used along-side a real plant as a reference as it exhibited similar behaviour to a live plant as shown in chapter 5 of this report. With confidence, the simulator could show a plant going through a transient whilst at the same time, the Control System is responding to a fault. This was the main objective that would show a fault in the system early for a timely response from plant operators. The advanced fault detection system as proposed has also shown reliability in steady state conditions. It follows all other trends in output of the plant when the plant parameters change due to a transient. The most important parameters monitored being pressure of the RCP and temperature. This analysis can lead to detection of faults in the primary system of the plant as this study has shown.

The fault is detected in the system when the expected outcome on the simulator is at a constant value whereas in the plant data the outcome drifts away at a certain rate. The plant can be said to be undergoing a transient when the value exceeds the constant steady state value by about more than 10 % of set value.

The early detection system will also limit the intervention of the nuclear operators on critical systems which will eventually eliminate human factors/human errors, which has contributed to most undesired accidents in nuclear power plants. The proposed system has shown that the simulator data (parameters) will only change as prompted by the simulator operator as the plant parameters drifts away due to on-going transient. There is a need to emphasise that this proposed system will not replace any current system in the plant but it (PCTRAN simulator) is operated parallel to the real plant. This will be an additional system to existing systems. This is to enhance confidence in the operator action, input and response into the plant transient situation.

There were some certain limitations in this dissertation. The KNPP has been operating for number of years already and some system has started degrading and no longer operating at their optimum design potential (values). This is due to life expectancy of some components which are approaching their end of life in their lifespan. It is important to note that KNPP is operating at reduced capacity so that the life-expectancy of the steam generators that are currently used in the plant is increased so the plant is operating with the concept known as ORT, Operation with Reduced Temperature and FMS, Fuel Management Strategy.

According to ORT, the inlet temperature into the core is 276,8 ºC and the outlet temperature of the working fluid to the SG's is 312,4 ºC, which effectively translates to reduction of 10 ºC in working fluid temperatures. The ORT modifies the secondary steam characteristic with respect to pressure; flow rate and moisture content. The objective of the ORT was to slow down the SG tubing degradation due to intergranular stress corrosion cracking in high residual stress areas. It is important to mention that the thermal power level remains unchanged at 100% full power.

The FMS refers to the change enrichment of the nuclear fuel. The uranium fuel, U_{238}, was enriched to 3,9 % initially and now has been increased to 4,4 %. All this factors have an overall effect on the normal operation of the plant as it is no longer operating at design parameters and that would factor in differences in the output of the proposed simulator. There have been numerous modifications on other safety related systems to enhance their performance and their safety. For example, the pressuriser at Koeberg and similar plants has been modified with the addition of the PORV's (Power operated relieve valves).

The current model of the simulator might also need to be upgraded and updated so as to be in par with the current plant on those critical and simulated systems. The design engineers of the plant continuously seek ways to advance the performance of the plant by modifying and installing other performance enhancing system and cost effective approaches to maximise the outputs from the plant. The KNPP has been designed for lifespan of 40 years. Currently, Eskom is under-going a licensing process to extend the life of the plant by another 20 years, to an effective 60 years. Since most components were only designed for 40 years, then the nuclear plant operator (Eskom) will need to change other components and perform some modifications as it may be required by the nuclear regulator.

The nuclear industry requires that primary verification and validation (V&V) on a system be independently performed. This V&V is critical for licensing procedures. It is done to ascertain the results or outcomes with a different approach and methodology completely different from the initial one but with similar results. Eskom will change the steam generators from 2015 which explains the fact that the coolant temperature had lower than design values. This affects the overall efficiency of the plant which currently stands at 32,6 percent. This intervention was necessary to limit the corrosion of the current steam generators.

The in-corporation of this proposed advanced fault detection system will be beneficial to the plant which is currently approaching its end of life operation or life extension licensing processes to proof to the regulator that safety is enhanced and human/operator intervention is less. The other limitation identified is the data made available from the Koeberg NPP which was not sufficient for security reasons. Koeberg NPP is regarded as national key point as documented in South African law and thus most data is not available in the public domain and can only be accessed and used by personnel within the organisation but would not be made available to the general public. The data that was generated from the simulator though, is sufficient to make an informed conclusion and recommendation.

7.2 Recommendations

It is recommended that the advanced fault detection system and the results of this study be presented to the plant and design basis engineering department in charge of Koeberg NPP. This would be to demonstrate the benefits of introduction of a similar system to the plant. Once the benefits of the system can be justified then it can also be introduced and demonstrated to the National Nuclear Regulator (NNR) for possible in-corporation of the system to the operation of the plant and possible licensing of the system. There should be another method independent of these presented here to verify the results before these authorities mentioned above can approve it as reliable. This is a standard requirement for Verification and Validation (V&V) in NPP.

Further studies in regard to other plant outputs can be done to ascertain the outcomes of this research. As the Koeberg NPP is ageing, it will experience a lot more transients of which the parameters of the transients of the plant can be modelled using the simulator to statistically declare it reliable and relevant for the application of early detection fault diagnostics system. Matlab/Simulink (Mathworks, 2000) software as a powerful research and simulation tool can be used to simulate the NPP main control system parameters. This (software) can also act as a V&V standards requirement (Lin et al., 2010).

This study can be taken further to fault characterisation, where the fault will be detected, analysed and identified. When the early fault detection system is implemented, the configuration of the simulator must be changed to match that of the plant with all known parameters. The modifications that were done on all the major plant components must be changed and adopted into the PCTran so as to have reliable results on outputs especially on transients that the plant would try to counter so as to stabilise and maintain normal operation.

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ANNEXURE A: PAPER PRESENTATION

"Evaluation of an Advanced Fault Detection System using Koeberg Nuclear Power plant"

The South African Nuclear Human Asset and Research Programme (SANHARP) Postgraduate Conference

12 – 14 October 2011

iThemba LABS

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