Bangladesh University of Engineering and Technology (BUET)

Design Basis Accident Analysis of Pressurized Water Nuclear Reactor

by

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Declaration

I, hereby, declare that, except where specific reference is made to the work of others, the contents of this dissertation are original and have not been submitted in whole or in part for consideration for any other degree or qualification in this, or any other university. This dissertation contains fewer than 14200 words including appendices, bibliography, footnotes, tables and equations and has fewer than 50 figures.

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Abstract

Nuclear Power is the cleanliest of all other power sources available with comparatively low fuel cost and zero Carbon-di-Oxide (CO_2) and other Green House Gas (GHG) emissions. Nuclear Power Plant (NPP) utilizes the heat generated from nuclear fission reaction of Uranium-235 to produce steam which then rotates "Turbine" and generates electricity. These NPPs are equipped with "state-of-the-art" safety equipment and protection systems to check any kind of harmful radiation to the outside environment. Despite these safety systems, an accident can happen in a NPP due to various reasons i.e. human error, equipment failure, natural disaster etc. So, to assess the integrity of different components of NPP in the event of Nuclear Accident, it is a good practice to study those transients before putting a NPP in operation. With the advancement of computing power and resources, it becomes both economical and feasible to study these transients in a more efficient way providing a good insight of what actually is happening inside the reactor during accidents. In this work, Personal Computer Transient Analyzer, abbreviated as PCTRAN, is used to simulate 4 different Design Basis Accidents (DBA) named Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB), Loss of Coolant Accident (LOCA) and Loss of Offsite Power (LOOP). A plant specific PCTRAN model for VVER-1000 Pressurized Water Reactor has been used. The results obtained from the simulation show that, VVER-1000 active and passive safety systems is capable to mitigate the consequences of these accidents hence reducing the possibility of radioactivity release into the atmosphere.

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Introduction

1.1 Background

Bangladesh is a densely populated country. Our country has comparatively low GDP. In 1970 the population was about 70 million. But within 46 years the number of population has been more than doubled. Population and Economic condition is the main factor for the development of any sector in a country.

Electricity is the major source of power for most of the country's economic activities. Bangladesh's total installed electricity generation capacity (including captive power) was 15,351 MW as of January 2017. [1] As 2015 92% urban population and 67% rural population have the access to the electricity for their source of light. Average 77.9% population have the access to the electricity in Bangladesh [2]. Problems in the Bangladesh's electric power sector include high system losses, delays in completion of new plants, low plant efficiency, erratic power supply, electricity theft, blackouts, and shortages of funds for power plant maintenance. Overall, the country's generation plants have been unable to meet system demand over the past decade. Noncommercial energy sources, such as wood fuel, and crop residues, are estimated to account for over half of the country's energy consumption

1.2 Energy Scenario of Bangladesh

As on February-2016, total installed capacity is 12,071 MW (Public Sector 6,440 MW, Independent Power Producers IPP, SIPP and Rental 5,131 MW and Power Import 500 MW) of Power Plants located at different parts of the country. The main fuel used for power generation is indigenous gas. In Fy-2014-2015, total 22,163 GWh gross energy was generated in the public sector power plant under BPDB. In addition, total 19,255 GWh of energy was purchased by BPDB from Independent Power Producers and 3,380 GWh from Power import in the private sector. The total numbers of grid substations are 131 and the total capacity is 24,670 MVA as on June-2015

FUEL TYPE	CAPACITY	TOTAL (%)
COAL	250	1.82
F.OIL	0	0
GAS	8727	63.68
HFO	2690	19.63
HSD	1158	8.45
HYDRO	280	2.04
IMPORTED	13705	4.38
TOTAL	13705	100

Installed Capacity of BPDB Power Plants as on August 2017

1.3 Why Nuclear Power is necessary?

No Emission of Carbon-di-Oxide:

Necessity of Nuclear Power Plant in Bangladesh in now well recognized. Nuclear power is an environment friendly and cost effective option for electricity generation. In this process there is no emission of carbon-di-oxide, a major factor for global warming and climate change that poses a killer threat to the whole of humanity. Nor it produces any harmful chemical that causes disasters like acid rain, depletion of ozone layer affecting bio-diversity. That is why this option is safe. From a nuclear reactor the amount of radiation released in the surroundings in 2000 years is almost equal to the amount of radiation absorbed by a patient exposed to X-ray machine for the purpose of medical diagnosis.

Low Fuel Cost:

Nuclear Power Plants involve very low fuel cost compared to the fossil fuel plants. The new generation nuclear reactors are more reliable and efficient than the earlier ones. New technologies have made them safer.

Dependable Option of Electricity:

Generation of electricity from nuclear power is a dependable option. In this process a stable and easily changeable fuel-Uranium (135U) is used. The amount of free energy contained in nuclear fuel is million times higher than that of chemical fuel like gasoline. In Nuclear Power Plant is possible to generate more than 2000 kWh electricity by using only one gram of Uranium. Whereas, several tons of coal is required for the same. Moreover, cost of electricity generation in this process in much lower than other conventional options. Though the initial investment appears to be high, it is a cheaper option in the long term.

Pollution Free:

In Bangladesh, generation of electricity by using nuclear power will make a revolutionary change in the economic sectors and improve the life standard of the people as well. As a clean energy it will help protect environment from pollution.

1.4 Is Nuclear Power safe?

- Decay heat is the principal reason of safety concern in Light Water Reactors, and by this reason nuclear power stations has several safety and safeguard systems to make sure that even in the worst case scenario the fuel does not become uncovered (or uncooled by water). These safety systems are subdivided into other safety systems. There are also systems that can work off of steam pressure in the reactor/steam generators, even if the most powerful power sources are lost. Of course, those systems are operated through various sources of power such as batteries, diesel generators, backup generators, and the external power line.
- If one system or component fails, there are additional systems (some of them are passive systems) that prevent a nuclear reactor from overheating. That is how safe a nuclear reactor actually is. In addition, new reactors such as the VVER-1200 and the AP-1000 can exist, some days, without any kind of external power (neither AC nor DC).
- Another thing that may concern people about a nuclear reactor is the massive amount of power that it can produce. It is obvious that an uncontrolled chain reaction will increase the reactor's power production by a factor of hundreds, melting it within

seconds. To prevent this from happening, there are two things methods of control: control rods and nuclear poison.

- Control rods are present in all kind of reactors and are designed to shut down the reactor within two seconds.
- The second method of control is nuclear poison (boron), which is only present in pressurized water reactors. (This system and the system used to control the power of the reactor aren't the same thing).
- Both of them are made of elements (or an element) able to absorb neutrons: Boron and cadmium in the first system and solely boron in the second one. Both of them are designed to terminate the reaction within a couple of seconds.

There is a clear need for new generating capacity around the world, both to replace old fossil fuel units, especially coal-fired ones, which contribute a lot of CO2 emissions, and to meet increased expectations for electricity in many countries. There are about 127,000 generating units worldwide, 96.5% of these of 300 MW or less, and one-quarter of the fossil fuel plants are over 30 years old. There is scope for both large new plants and small ones to replace existing units 1:1, all with near-zero CO_2 emissions.

Chapter 2

Nuclear Power Reactor

2.1 Introduction

A Nuclear Power Plant (also known as Nuclear Power Station) looks like a standard thermal power station with one exception. The heat source in the “Nuclear Power Plant” is a “Nuclear Reactor” where fission reaction takes place and huge amount of heat is generated from splitting of Uranium atoms. As is typical in all conventional thermal power stations the heat is used to generate steam which drives a steam turbine connected to a generator which produces electricity.

2.2 Types of Nuclear Reactor

From the physics point of view, the main differences among reactor types arise from differences in their neutron energy spectra. In fact, the basic classification of nuclear reactors is based upon the average energy of the neutrons which cause the bulk of the fissions in the reactor core. From this point of view, nuclear reactors are divided into two categories:

- ***Thermal Reactors:*** Almost all of the current reactors which have been built to date use thermal neutrons to sustain the chain reaction. These reactors contain neutron moderator that slows neutrons from fission until their kinetic energy is more or less in thermal equilibrium with the atoms ($E < 1$ eV) in the system.
- ***Fast Neutron Reactors:*** Fast reactors contains no neutron moderator and use less-moderating primary coolants, because they use fast neutrons ($E > 1$ keV), to cause fission in their fuel.

Conventional Nuclear Reactors that are currently in commercial operation throughout the world are categorized as the following:

- ***Pressurized Water Reactor:*** Pressurized water reactors use a Reactor Pressure Vessel (RPV) to contain the nuclear fuel, moderator, control rods and coolant. They are cooled and moderated by high-pressure liquid water (e.g. 16 MPa). At this pressure water boils at approximately 350°C (662°F). Inlet temperature of the water is about 290°C (554°F). The water (coolant) is heated in the reactor core to approximately 325°C (617°F) as the water flows through the core. The hot water that leaves the pressure vessel through hot leg nozzle and is looped through a steam generator, which in turn heats a secondary loop of water to steam that can run turbines and generator. Secondary water in the steam generator boils at pressure approximately 6-7 MPa, what equals to 260°C (500°F) saturated steam. Typical reactor nominal thermal power is about 3400MW, thus corresponds to the net electric output 1100MW. Therefor the typical efficiency of the Rankine cycle is about 33%.

- ***Boiling Water Reactor:*** A boiling water reactor is cooled and moderated by water like a PWR, but at a lower pressure (7MPa), which allows the water to boil inside the pressure vessel producing the steam that runs the turbines. The BWRs don't have any steam generator. Unlike a PWR, there is no primary and secondary loop. The thermal efficiency of these reactors can be higher, and they can be simpler, and even potentially more stable and safe. But the disadvantage of this concept is that any fuel leak can make the water radioactive and that radioactivity can reach the turbine and the rest of the loop.

- ***Pressurized Heavy Water Reactor:*** The CANDU reactor design (or PHWR – Pressurized Heavy Water Reactor) has been developed since the 1950s in Canada, and more recently also in India. These reactors are heavy water cooled and moderated pressurized water reactors. Instead of using a single large reactor vessel as in a PWR or BWR, the nuclear core is contained in hundreds of pressure tubes known as ***"calandria"***. PHWRs generally use natural uranium (0.7% U-235) oxide as fuel, hence needs a more efficient moderator, in this case heavy water (D2O).

- ***Advanced Gas-cooled Reactor:*** An advanced gas-cooled reactor (AGR) is a British design of nuclear reactor. AGRs are using graphite as the neutron moderator and carbon dioxide as coolant. AGRs were developed from the Magnox type reactor. These are the second generation of British gas-cooled reactors. AGRs are operating at a higher gas temperature for improved thermal efficiency, thus requires stainless steel fuel cladding to withstand the higher temperature. Because the stainless steel fuel cladding has a higher neutron capture cross section than Magnox fuel (magnesium non-oxidising alloy), low enriched uranium fuel is needed.

2.3 How Nuclear Power Plant Works?

Nuclear Power Plants, unlike conventional Thermal Power Plant where fuel is burnt, produce electricity by boiling water into steam. The steam produced then turns turbines to generate electricity. The difference is that NPP do not burn anything. Instead, they use Uranium fuel, consisting of solid ceramic pellets, to produce electricity through a process called fission. Fuel consists of two types of Uranium, U-238 and U-235. Most of the Uranium in nuclear fuel is U-238, but U-235 splits—or fissions—easily. In U-235 atoms, the nucleus, which is composed of protons and neutrons, is unstable. As the nuclei break up, they release neutrons.

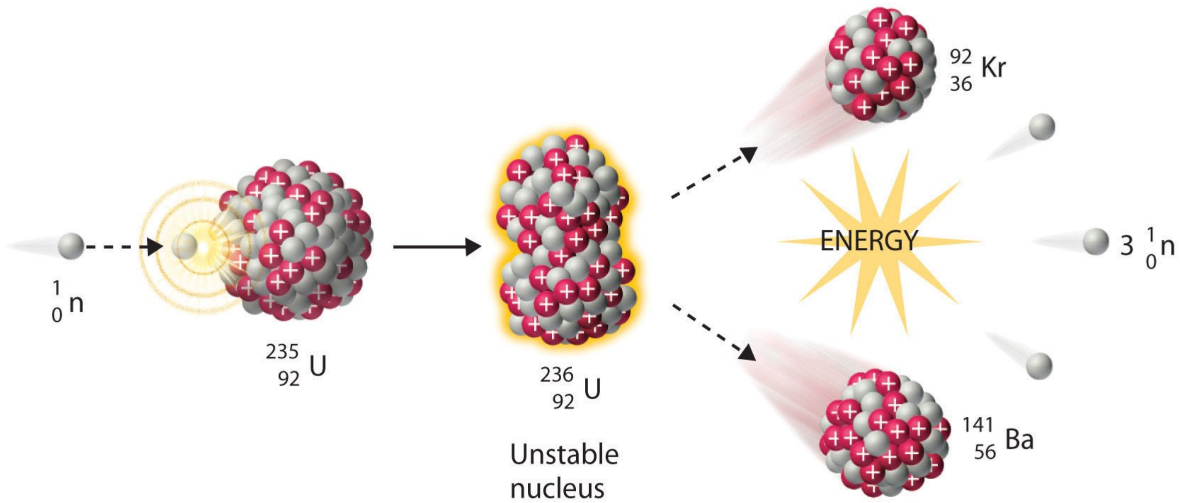

Figure 2. 1 Nuclear Fission Reaction

When the neutrons hit other Uranium atoms, those atoms also split, releasing neutrons of their own, along with heat. These neutrons strike other atoms, splitting them. One fission triggers others, which triggers still more until there is a chain reaction. When that happens, fission becomes self-sustaining. Control Rods (made of Neutron absorbing material) inserted among the tubes holding the Uranium fuel control the nuclear reaction. Control rods, inserted or withdrawn to varying degrees, slow or accelerate the reaction. Cooling water separates fuel tubes in the reactor. The heat produced by fission turns this water into steam. The steam drives a turbine, which spins a generator to produce electricity.

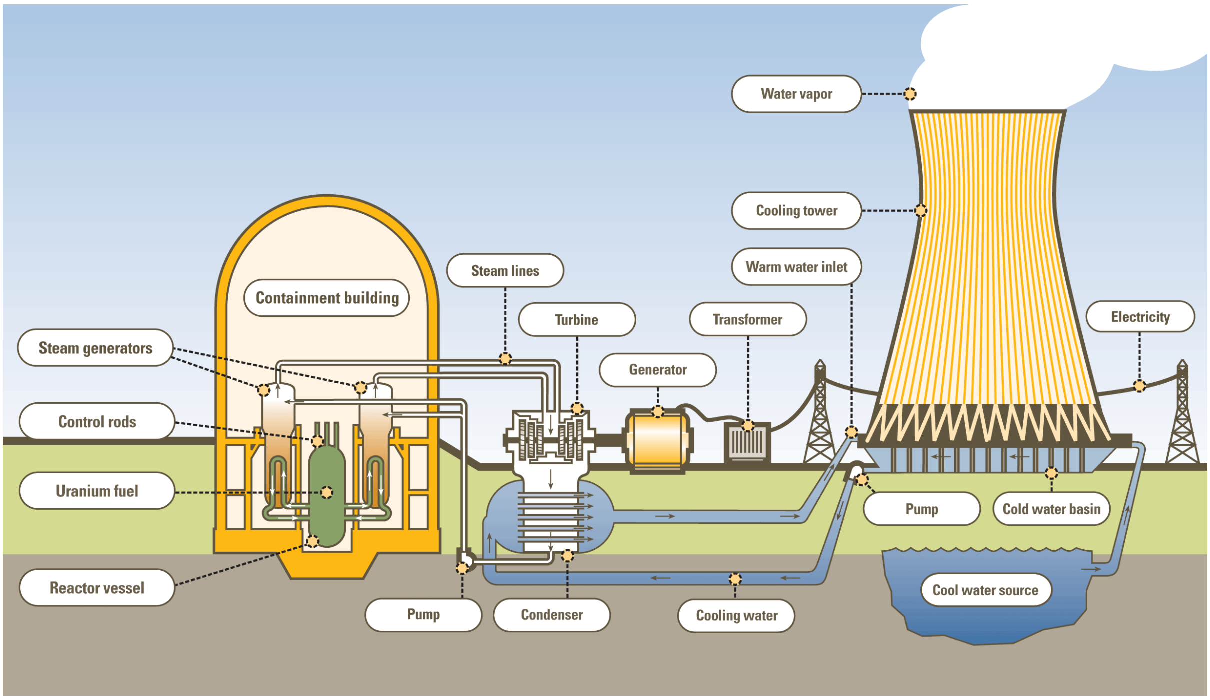

Figure 2. 2 Main features of Nuclear Power Plant

2.4 Nuclear Power Plants in Operation throughout World

On December 20, 1951, at the Experimental Breeder Reactor EBR-I in Arco, Idaho, USA, for the first time electricity - illuminating four light bulbs - was produced by nuclear energy. EBR-I was not designed to produce electricity but to validate the breeder reactor concept.

On June 26, 1954, at Obninsk, Russia, the Nuclear Power Plant APS-1 with a net electrical output of 5 MW was connected to the power grid, the world's first Nuclear Power Plant that generated electricity for commercial use. On August 27, 1956 the first commercial Nuclear Power Plant, Calder Hall 1, England, with a net electrical output of 50 MW was connected to the national grid.

As of November 28, 2016 in 31 countries 450 Nuclear Power Plant units with an installed electric net capacity of about 392 GW are in operation and 60 plants with an installed capacity of 60 GW are in 16 countries under construction. [3]

Table 2. 1 Nuclear Power Plants in the World

Country	In Operation		Under Construction	
	Number	Net Electric Output (MW)	Number	Net Electric Output (MW)
Argentina	3	1.632	1	25
Armenia	1	375	-	-
Belarus	-	-	2	2.218
Belgium	7	5.913	-	-
Brazil	2	1.884	1	1.245
Bulgaria	2	1.926	-	-
Canada	19	13.524	-	-
China	36	31.402	20	20.500
Czech Republic	6	3.930	-	-
Finland	4	2.752	1	1.600
France	58	63.130	1	1.630
Germany	8	10.799	-	-
Hungary	4	1.889	-	-
India	22	6.225	5	2.990
Iran	1	915	-	-
Japan	43	40.290	2	2.650
Korea, Republic	25	23.133	3	4.020
Mexico	2	1.440	-	-
Netherlands	1	482	-	-
Pakistan	4	1.005	3	2.343
Romania	2	1.300	-	-
Russian Federation	36	26.557	7	5.468
Slovakian Republic	4	1.814	2	880
Slovenia	1	688	-	-
South Africa	2	1.860	-	-
Spain	7	7.121	-	-
Sweden	10	9.651	-	-
Switzerland	5	3.333	-	-
Taiwan, China	6	5.052	2	2.600
Ukraine	15	13.107	2	1.900
United Arab Emirates	-	-	4	5.380
United Kingdom	15	8.918	-	-
USA	99	98.868	4	4.468
Total	**450**	**391.915**	**60**	**59.917**

*Nuclear Power Plants world-wide, in operation and under construction, IAEA as of 27 November 2016

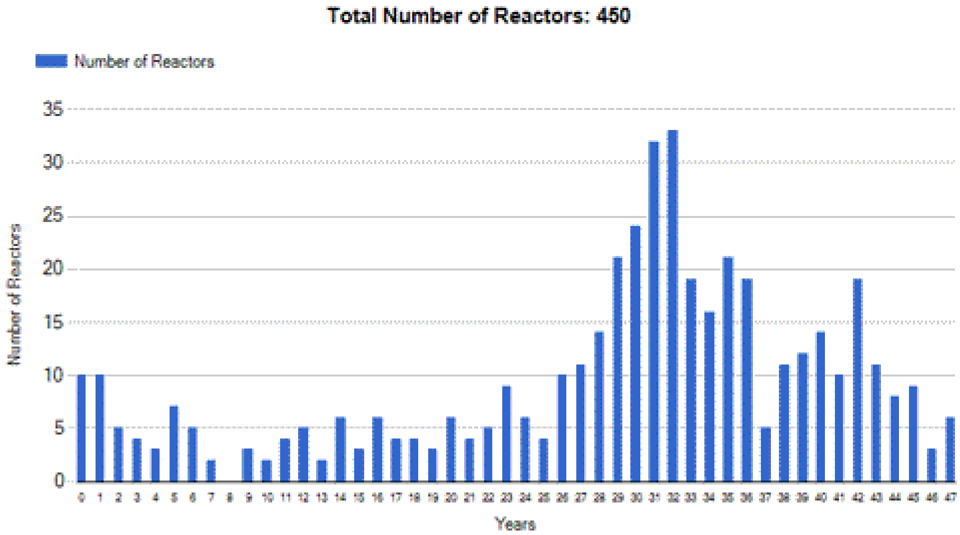

Figure 2. 2 *Operational Years of Existing Nuclear Reactor*

2.5 Worldwide Power Generation from Nuclear Power Plant

There are over 450 commercial nuclear power reactors operable in 31 countries, with over 390,000 MWe of total capacity. About 60 more reactors are under construction. They provide over 11% of the world's electricity as continuous, reliable power to meet base-load demand, without carbon dioxide emissions. 55 countries operate a total of about 250 research reactors, and a further 180 nuclear reactors power some 140 ships and submarines.

Sixteen countries depend on nuclear power for at least a quarter of their electricity. France gets around three-quarters of its power from nuclear energy, while Belgium, Czech Republic, Finland, Hungary, Slovakia, Sweden, Switzerland, Slovenia and Ukraine get one-third or more. South Korea and Bulgaria normally get more than 30% of their power from nuclear energy, while in the USA, UK, Spain, Romania and Russia almost one-fifth is from nuclear. Japan is used to relying on nuclear power for more than one-quarter of its electricity and is expected to return to that level. Among countries which do not host Nuclear Power Plants, Italy and Denmark get almost 10% of their power from nuclear.

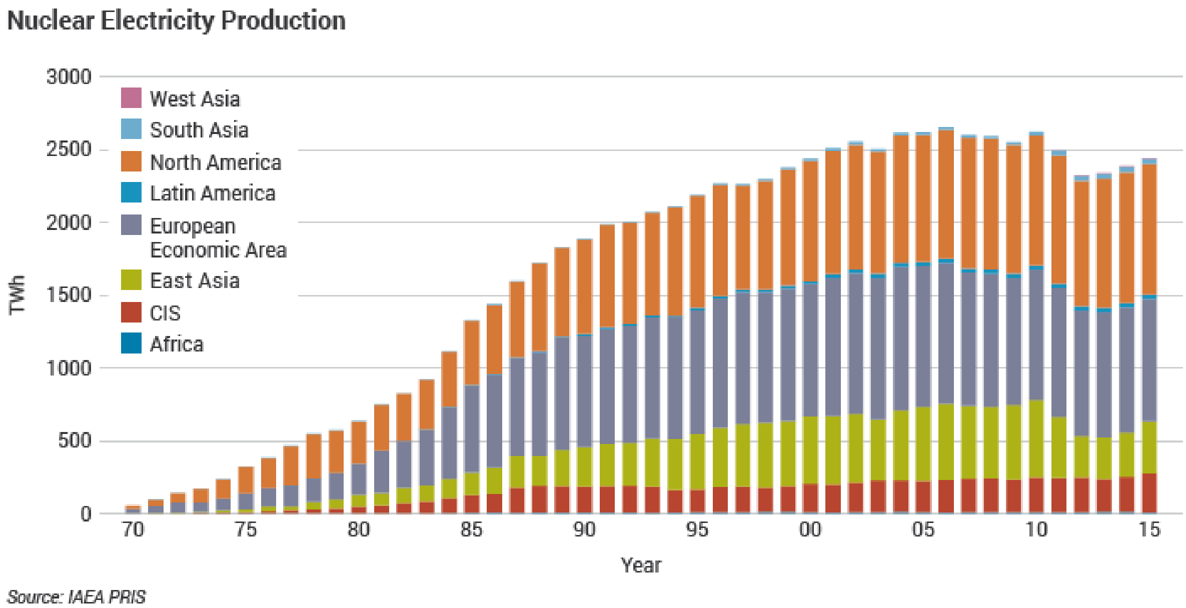

Figure 2. 3 *Worldwide Power Generation Scenario from NPP*

2.6 Nuclear Power in Bangladesh

To meet the rapidly increasing demand of electricity due to industrialization and reduce dependence on Natural Gas as a primary fuel for Power Generation, Bangladesh plans to have two large Russian VVER-1200 Nuclear Power Reactors in operation, the first from 2023. [4]

	Type	**Capacity**	**Construction start**	**Commercial operation**
Rooppur 1	AES-2006/V-523	1200 MWe	2017	2023 or 2024
Rooppur 2	AES-2006/V-523	1200 MWe	2018	2024 or 2025

Building a Nuclear Power Plant in the west of the country was proposed in 1961. Since then a number of reports have affirmed the technical and economic feasibility. The Rooppur or Rooppur site in Pabna district about 160 km northwest of Dhaka was selected in 1963 and land was acquired. The government gave formal approval for a succession of plant proposals, then after independence a 125 MWe Nuclear Power Plant proposal was approved in 1980 but not built. With growth in demand and grid capacity since then, a much larger plant looked feasible, and the government in 1999 expressed its firm commitment to build this Rooppur plant. In 2001 it adopted a national Nuclear Power Action Plan.

In May 2010 an intergovernmental agreement was signed with Russia, providing a legal basis for nuclear cooperation in areas such as siting, design, construction and operation of power and research nuclear reactors, water desalination plants, and elementary particle accelerators. Other areas covered included fuel supply and wastes – Russia will manage wastes and decommissioning.

In February 2011, Bangladesh reached an agreement with Russia to build the 2,400 megawatt (MW) Rooppur Nuclear Power Plant with two reactors, each of which will generate 1,200 MW of power. The NPP will be built at Rooppur, on the banks of the Padma River, in the Ishwardi subdistrict of Pabna, in the northwest of the country. The RNPP is estimated to cost up to US $12 billion, and start operating by 2023.

Chapter 3

VVER Nuclear Reactor

3.1 Introduction

The VVER (Water-Water Energetic) Reactors designed and constructed by ROSATOM (Russian State Atomic Energy Corporation) are among the world's most widely-used reactors. VVER plants have proved their reliability over more than 1300 "Reactor Years" of operation. Since the commissioning of the first VVER power unit in the 1960s, the technology has been providing safe and affordable electricity throughout the world.

3.2 A brief history of VVER

i) First VVER Units:

A total of 67 VVER have been constructed since the 1960s. The first VVER unit was commissioned in 1964, at Novo Voronezh Nuclear Power Plant, in the Voronezh region, Russia. The first unit was called the V-210, the second one is V-365 which produce 210 MW and 365 MW of electrical power.

ii) *VVER-400*

VVER-440 was the first of the VVERs to be constructed on a serial basis. VVER-440 units have been safely operating in many European Union countries: Slovakia (Bohunice 1-4, Mohovce 1-2), Hungary (Paks 1-4), Bulgaria (Kozloduy 1-4), Czech Republic (Dukovany 1-4), and Finland (Loviisa 1-2). Design of the Finnish Loviisa was completed in 1971-72 taking into account the General Design Criteria for Nuclear Power Plants, issued by the US AEC in 1971. After that, all VVER plants were designed to meet these safety principles.

iii) *VVER-1000 – V-320*

The VVER-1000 was a milestone not only in terms of generating capacity, but also because of the many safety innovations it incorporated. The VVER- 1000 is the most common VVER design worldwide, 31 units are in operation, and have amassed about 500 reactor-years of operation.

iv) *VVER-1000 – AES-91 & AES-92*

Drawing on the significant body of experience gained with the well-established VVER-1000/V-320, the AES-91 (or VVER-1000/V-428) was developed by Saint-Petersburg Atomenergoproekt and the AES-92 (or VVER-1000/V-412 and 466) by Moscow Atomenergoproekt. Along with upgraded technology and improved economics, these designs deployed the concept of BDBA (Beyond Design Basis Accident) management based on a balanced combination of passive and active safety systems.

v) *VVER-1200 – AES-2006*

The AES-2006 design is the latest evolution in the long line of VVER plants. It meets all the international safety requirements for Gen III Nuclear Power Plants. The first AES-2006 units are now under construction in Russia; two units in Sosnovyi Bor (Leningrad II), two units in Novovoronezh (Novovoronezh II) and two units in the Kaliningrad Region (Baltic project). In addition, construction contracts have been signed and site preparation is ongoing for four units in Turkey and two units in Belarus. It is also proposed for Temelin 3-4 (Czech Republic) and Hanhikivi 1 (Finland).

3.3 Features of VVER-1000 Reactor

The VVER is a pressurized-water reactor (PWR), the most common type of nuclear reactor worldwide, employing light water as coolant and moderator. However, there are some significant differences between the VVER and other PWR types, both in terms of design and materials used. Distinguishing features of the VVER include the following: [6]

- Horizontal Steam Generators
- Hexagonal Fuel Assemblies
- Avoidance of bottom penetrations in the VVER vessel
- High capacity Pressurizers

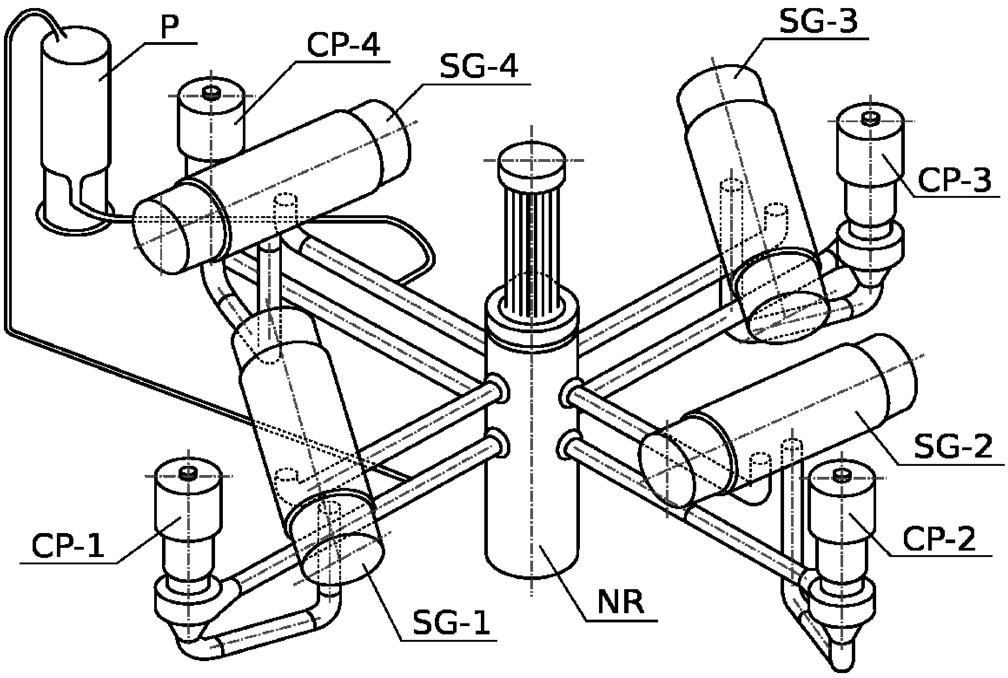

Figure 3. 1: VVER-1000 Nuclear Power Plant Outline

CP: Coolant Pump; SG- Steam Generator; P- Pressurizer; NR- Nuclear Reactor

3.3.1 Reactor Pressure Vessel

The reactor vessel internals include core barrel, core baffle protective tube unit and in-core instrumentation system parts in the upper plenum of the vessel, the core itself, control rods and ICID (in-core instrumentation detector) sensors. The reactor vessel head is integrated structurally with the upper assembly. The CPS (control and protection system) drive housings are installed on the reactor vessel head. The core barrel, core baffle and protective tube unit are kept from lifting under normal operating conditions by their weight plus hold-down assemblies employing elastic components made of thermo-expanded graphite. This performs better than the materials used in the V-320 reactor, with a service life of at least four years without replacement.

3.3.2 Main Coolant Piping

The main coolant (primary) pipework connecting the reactor, steam generators and reactor coolant pumps has a nominal diameter of 850 mm, with a service life of 60 years. The nominal external diameter is 990 mm and the nominal wall thickness is 70 mm. The total length is 146 m. The main coolant piping is designed to meet all necessary conditions of the "leak-before-break" concept: material properties, stress analysis, in-service inspections and leak monitoring.

3.3.3 Reactor Coolant Pump

The reactor (main) coolant pump is of the GCNA-1391 type. The RCP is equipped with a flywheel providing smooth main coolant circulation rundown during accident scenarios with loss of power. This permits adequate reactor cooling until the reactor is shut down and decay heat has dropped to level where it can be safely removed by natural circulation. The design incorporates experience gained from the GCN-195M type pumps employed in the V-320 reactor and from the fist GCNA-1391 pumps, which have operated reliably at Tianwan since 2007.

Basic data for the GCNA-1391 pump is as follows:

- Capacity: 22 000 m^3/h
- Head : 0.588 MPa
- Nominal Suction Pressure: 16.02 MPa
- Speed: 1000 RPM
- Supply Current Frequency: 50 Hz
- Weight (without motor): 75.5 ton
- Service Life: 60 years

3.3.4 Steam Generator

The steam generators employed in the VVER-1000 are of the PGV-1000MKP type. As well as being horizontal the steam generator also uses a “corridor” layout for the heat-exchange tubes in the tube bundle. Proven design with incremental improvements such as effective sludge removal from the steam generator bottom, the use of secondary side ethanolamine water chemistry and elimination of copper-bearing components on the secondary side, enable an expected service life of 60 years to be achieved.

3.3.5 Pressurizer

In VVER reactor plants, high volume pressurizers have always been used, thus assuring a high level of reactor safety owing to the large coolant inventory in the primary circuit. In the VVER-1000 design, a modernized system for even better pressure control under transients is used. The new system has an additional line of controlled water injection to the steam space.

3.3.6 Reactor Core and Fuel Assemblies

The reactor core comprises 163 fuel assemblies. Reactor output is controlled using the 121 control rods of the control and protection system, by burnable neutron absorber in the fuel rods, and by change of boric acid concentration in the primary circuit water.

In the VVER-1000 design hexagonal fuel assemblies are used with these characteristics:

- 13 spacer grids (including the anti-vibration one);
- located with a gap of 340 mm;
- height of fuel column 3.73 m;

- weight of fuel UO_2 – up to 534 kg;
- enrichment – up to 4.95 %;
- quickly removable fastening of the top nozzle with the use of collets enabling quick disassembly and assembly for the replacement of leaky fuel rods;
- collet fastening of fuel rods;

3.3.7 Turbine

The VVER-1000 is offered with two steam turbine variants: Russian (LMZ) design full speed turbine manufactured by Power Machines .

Characteristics:

- Types of Turbines: K-1200-6.8/50
- Number of Turbine Sections per unit (e.g. HP/MP/LP): 1/0/4
- Turbine Speed: 3000 rpm
- HP Turbine Inlet Pressure: 6.8 MPa
- HP Turbine Inlet Temperature: 283.8 °C

3.4 Working Principle

The VVER reactor is a pressurized-water nuclear power reactor with a pressure vessel, working with thermal neutrons. It is used for production of thermal energy released by fission of nuclear nuclei in fuel. The equipment of the reactor circuit is intended for dry saturated steam production for turbine generators, where the thermal energy of steam transforms into electrical energy.

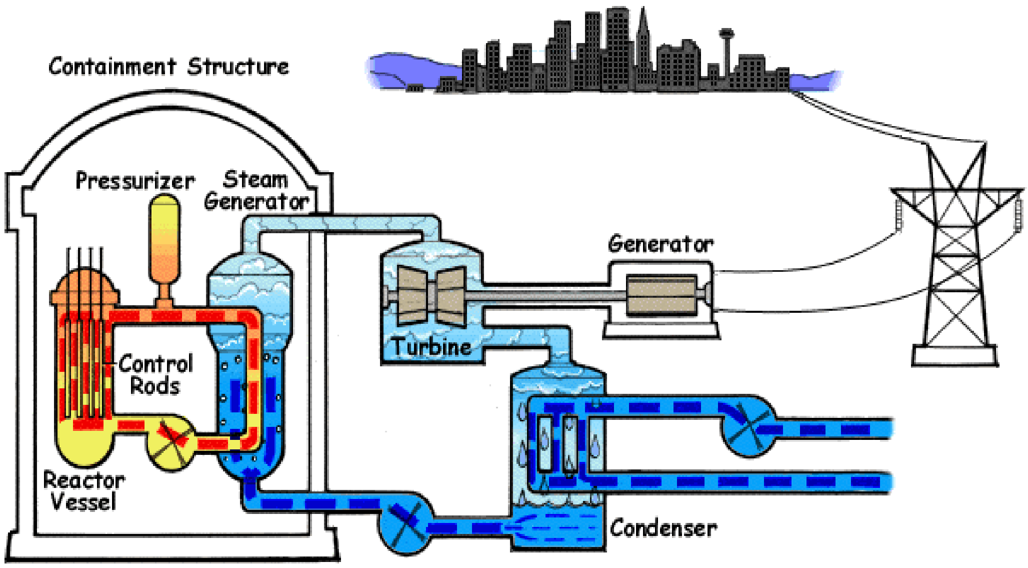

Figure 3. 2 Pressurized Water Reactor Working Principle

The energy production process begins with the fission reaction of enriched Uranium atom inside the reactor core which contains Nuclear Fuel Assemblies .Each of these assemblies contains several hundred sealed tube containing palette of enriched Uranium (high proportionate Uranium-235. During the fission reaction, enormous amount of heat is generated within the fuel tubes. These tubes then transfer heat to the primary circuit water. After leaving the reactor core, the coolant water is transported along the part of the primary circulation circuit called "hot leg" to the SG. The SG is a heat exchanger in which the heat from the primary circuit coolant transfers to feed water of the secondary circuit to form steam. They play a significant role in the safe and reliable operation of VVER power plants by determining the T-H response of the primary coolant system during operational and accidental transients. After the steam generator, the coolant is transported along the part of primary circulation circuit called "cold leg" back to the reactor vessel. Most of the steam formed in the SG is sent to the turbine, with a much smaller part to feed water heating. High pressure steam rotates these turbines and generates electricity.

3.5 Safety Features

The VVER-1000 plant was designed to meet the Russian general safety requirements issued in 1997, which were consistent with the IAEA's International Nuclear Safety Group (INSAG) recommendations. The INSAG group recommendations led to the development of what were called 'Generation (Gen) III' Nuclear Power Plants, and the current IAEA safety standard on Nuclear Power Plant design safety, issued in 2012, builds on the same principles. The Russian general safety requirements are also consistent with the safety objectives specified by WENRA (Western European Nuclear Regulators Association) in 2010 for new Nuclear Power Plants.

The VVER-1000 design takes account of Design Extension Conditions (DEC), in accordance with the current IAEA safety standard. Thus all new VVER-1000 plants under construction already have design features that take fully into account the main 'Fukushima lessons learned including: [8]

- Long-term cooling of reactor core without electrical power
- Long-term decay heat removal that does not rely on primary ultimate heat sink (sea, river, cooling tower)
- Protection of reactor containment integrity with dedicated systems after a core meltdown accident.

The safety systems are designed to have the capability for stable operation under adverse conditions due to natural phenomena, such as; earthquakes, floods, storm winds, hurricanes, snowfalls, tornadoes, low and high extremes of temperature, as well as man-induced events, including; **aircraft crash** or **impact from aircraft parts, air shock wave, fire** and **flooding** caused by water pipe breaks. The main principles include:

• The inherent safety principle, that is, the ability of the reactor to ensure safety based on natural feedback processes and characteristics.

• **Defence in Depth** principle, that is, use of successive barriers that prevent the release of ionising radiation and radioactive substances to the environment, as well as a system of technical and organizational measures for protection of these barriers.

The main concept for providing fundamental safety functions are:

• **Passivity**: Passive means are used to deal with 'design extension conditions' and 'beyond design basis accidents' (passive SG cooling system, passive containment cooling system) and provide back-up for active safety systems.

• **Multiple train redundancy:** The plant utilizes four trains for safety systems and for their control systems.

• **Diversity:** The back-up systems for the systems providing basic safety functions use different equipment from the backed-up safety system and, if possible, also a different operating principle.

• **Physical separation**: All four trains of safety systems and their control systems are physically separated, which addresses common mode failures due to fire, aircraft accident or terrorist act. Unit control rooms (main control room and emergency control room) are also physically located in separate rooms/buildings.

Chapter 4

Personal Computer Transient Analyzer (PCTRAN)

4.1 Introduction

It is now a common practice to use computer to simulate different accident scenarios that may happen in a Nuclear Power Plant prior to their installations. The data obtained from the simulations are then cross-checked against the available experimental data. With the advancement of computer memory and processing power, it has become both economical and time saving to simulate these accidents virtually and implement those data in real world applications. There are numerous Nuclear Accident Simulator available depending on the type of reactor. These simulators have capabilities to mimic different accident scenarios typically occurred in a Nuclear Power Plant. Some of the common simulators that are currently used in the industry are mentioned below:

- RELAP3D
- PCTRAN
- NOTRUMP

4.2 What is PCTRAN?

PCTRAN is a software package that can simulate a variety of accident and transient conditions for Nuclear Power Plants. A high-resolution color mimic of the Nuclear Steam Supply System (NSSS) and containment displays the status of important parameters and allows simulation of operator actions by interactive control. It is specifically designed for many different plant types, including PWR, BWR, advanced AREVA EPR, Westinghouse AP1000, GE ABWR, and ESBWR. PCTRAN's simulation scope extends to severe accidents of BWR5 and RadPuff for dose dispersion from all above. Since its first introduction in 1985 by Micro-Simulation Technology, PCTRAN has been selected by the IAEA as the training platform for its annual IAEA Simulators Workshop. Plant-specific models have been installed at Nuclear Power Plants and institutions all over the world for practical application in training, analysis, probabilistic safety assessment, and emergency exercises. [7]

PCTRAN is most powerful in its versatile and interactive control. By using graphic icons and pull-down windows, plant control is conducted by an intuitive point-and-click of the mouse. The user can at any time manually trip the reactor or the pumps, open or close a valve, override the ECCS, or change the operational set points. The system has the ability to freeze, back-track, snap a new initial condition, change the simulation speed, trend plot selected variables, etc. for the convenience of conducting a training session or engineering analysis. The design of the man-machine interface is similar to, but more powerful than, a typical instructor's station with a full-scope simulator.

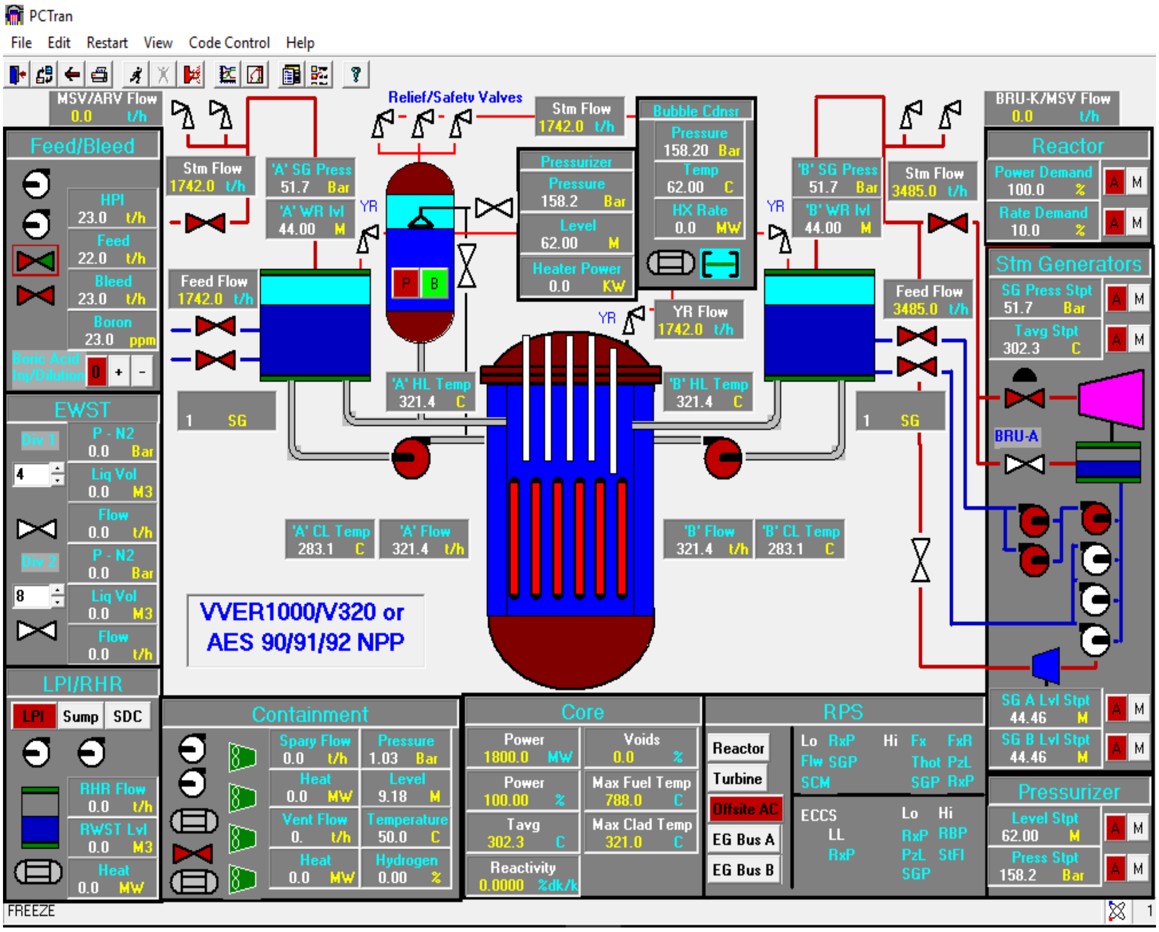

Figure 4.1: PCTRAN Nuclear Steam Supply System GUI

4.3 Simulation Scope of PCTRAN

PCTRAN can perform a variety of Design Basis Accidents (DBAs) occurred in a Nuclear Power Plant. Transients that have been successfully simulated with PCTRAN and validated with real world data are listed in the table below.

- Normal Operation Control - Startup, Shutdown, Power Ramp
- Loss-of-Coolant-Accident (LOCA)
- Main Steamline Break
- Loss of Flow or Recirculation Pump Trip

- Turbine Trip With/Without Bypass
- Station Blackout or Loss-Of-Load
- Inadvertent Rod Withdrawal or Insertion
- Boron Dilution Transient
- Steam Generator Tube Rupture (PWR)
- Feedwater Transients
- Anticipated Transient without Scram (ATWS)
- Any Combination Of Above

4.4 Methodology

The user can select from a set of initial conditions corresponding to various power, flow, and time-of-life conditions of the plant. By using the mouse, the user can choose from auto or manual mode of plant operation. He or she can also initiate malfunctions that encompass all possible disturbances to a plant and all categories analyzed in the plant's FSAR. For example:
A plant mimic as shown in the figures will be displayed on the screen as computation is progressing. Selected key plant parameters are digitally displayed in real-time or faster, if selected. Water levels dynamically rise and fall as calculated and coolant void fraction is reflected by changing colors. Other dynamic indicators show the status of pumps, valves, trips, and alarms, as well as control rod movement.

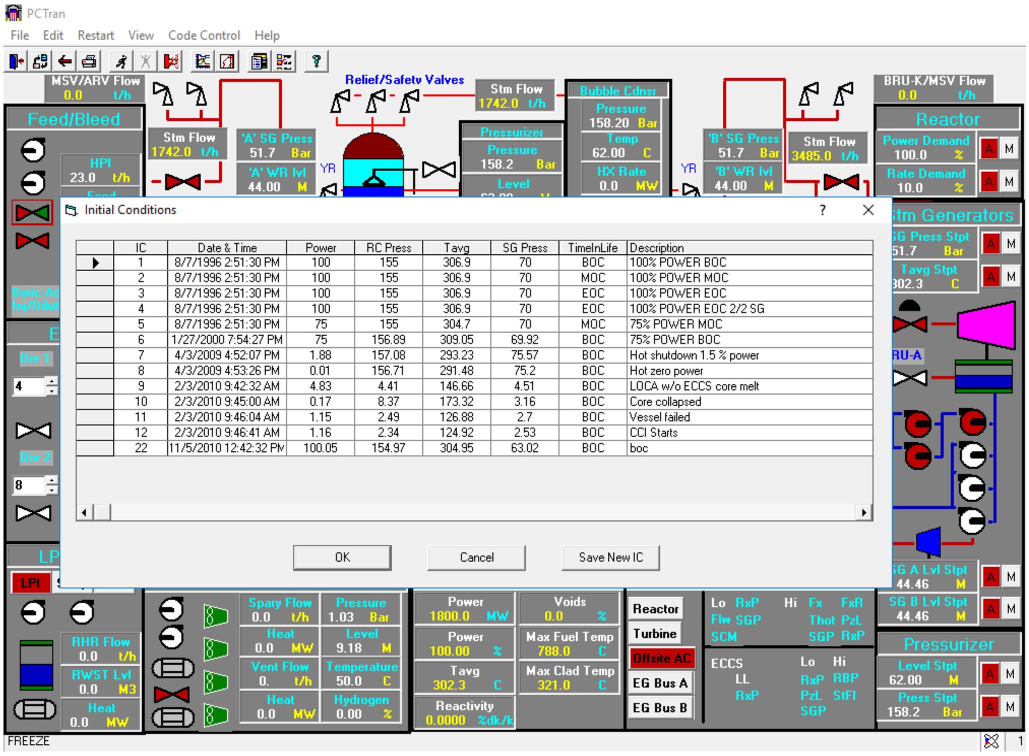

Figure 4.2: Setting up Initial Conditions

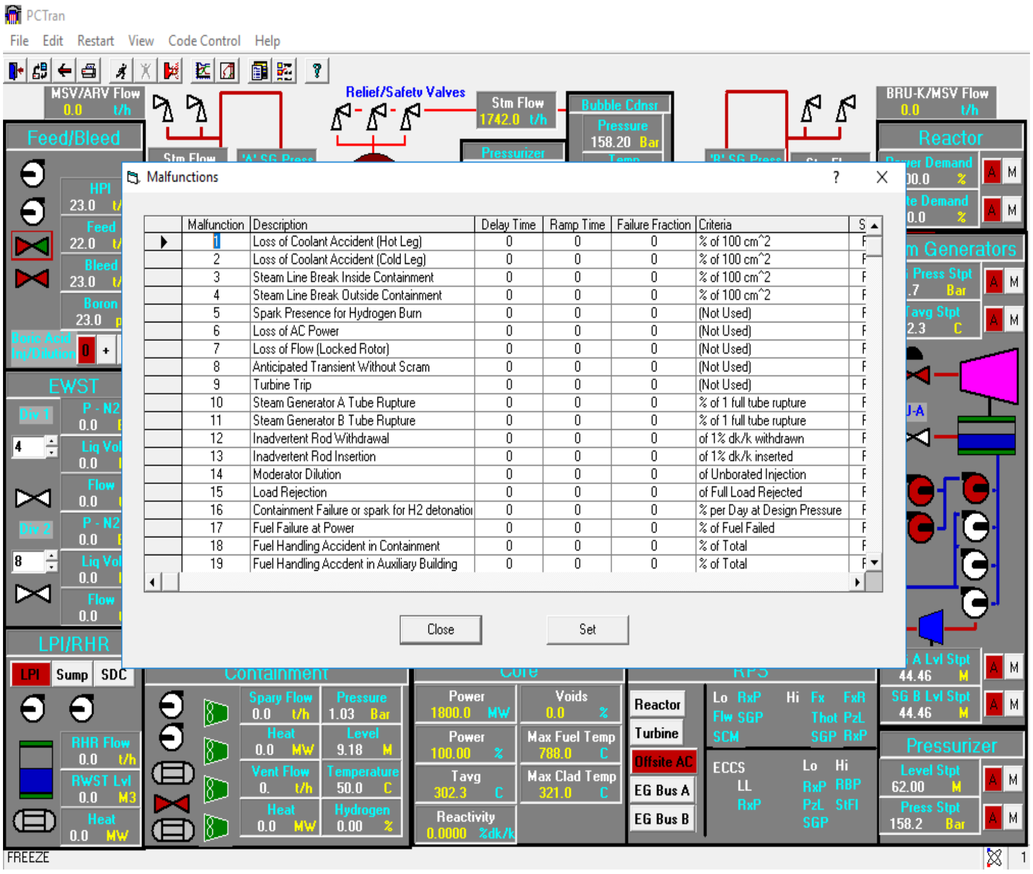

Figure 4.3: Activating Transients or Malfunctions in PCTRAN

These desktop simulator codes provide insight and understanding of the designs as well as a clear understanding of the operational characteristics of the various reactor types. The simulators operate on personal computers and are provided for a broad audience of technical and non-technical personnel as an introductory educational tool. The preferred audience, however, are faculty members interested in developing nuclear engineering courses with the support of these very effective hands-on educational tools. The application of the simulator programs is limited to providing general response characteristics of selected types of power reactor systems and they are not intended to be used for plant-specific purposes such as design, safety evaluation, licensing or operator training.

Chapter 5

Acceptance Criteria for Design Basis Accidents in a PWR

5.1 Introduction

The applicable acceptance criteria used in accident analysis typically depend on the objective of the analysis (e.g. design or licensing), on the frequency of initiating events, on the related safety aspects and on the existing high level limits for accident consequences. Thus, different criteria exist for AOOs, for DBAs and for BDBAs or SAs. Acceptance criteria for accidents are further differentiated for LOCAs, for ATWS and for other types of accident.

5.2 Acceptance Criteria for Transients

For transients it has to be demonstrated that the intrinsic features of the design and the systems automatically actuated by the instrumentation, particularly the reactor trip system, are sufficiently effective to ensure that:

- The probability of a boiling crisis anywhere in the core is low. This criterion is typically expressed by the requirement that there is a 95% probability at the 95% confidence level that the fuel rod does not experience a departure from nucleate boiling (DNB).The DNB correlation used in the analysis needs to be based on experimental data that are relevant to the particular core cooling conditions and fuel design.
- The pressure in the reactor coolant and main steam systems is maintained below a prescribed value (typically 110% of the design pressure).
- There is no fuel melting anywhere in the core.

5.3 Acceptance Criteria for Design Basis Accidents

For DBAs it has to be demonstrated that the design specific engineered safety features are sufficiently effective to ensure that: [9]

- The radially averaged fuel pellet enthalpy does not exceed the prescribed values (the values differ significantly among different reactor designs and depend also on fuel burnup) at any axial location of any fuel rod. This criterion ensures that fuel integrity is maintained and energetic fuel dispersion into the coolant will not occur (specific to RIAs).
- The fuel rod cladding temperature does not exceed a prescribed value (typically 1480°C). This criterion ensures that melting and embrittlement of the cladding are avoided.
- Fuel melting at any axial location of any fuel rod is limited (typically, no fuel melt is allowed or a maximum 10% melt of the fuel volume at the hot spot is

accepted). This criterion ensures that substantial volumetric changes of fuel and a release of radioactive elements will not occur.

- The pressure in the reactor coolant and in the main steam system is maintained below a prescribed value (typically 135% of the design value for ATWSs and 110% for other DBAs). This criterion ensures that the structural integrity of the reactor coolant boundary is maintained.
- Calculated doses are below the limits for DBAs, assuming an event generated iodine spike and an equilibrium iodine concentration for continued power operation, and considering actual operational limits and conditions for the primary and secondary coolant activity.

In addition to criteria described in the previous section, particularly for design basis LOCAs, short term and long term core coolability should be ensured by fulfilling the following five criteria:

- The fuel rod cladding temperature should not exceed a prescribed value (typically 1200°C); the value is limiting from the point of view of cladding integrity following its quenching and is also important for avoiding a strong cladding–steam reaction, thus replacing criterion (5) which is valid for other accidents.
- The maximum local cladding oxidation should not exceed a prescribed value (typically 17–18% of the initial cladding thickness before oxidation).
- The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed a prescribed value (typically 1% of the hypothetical amount that would be generated if all the cladding in the core were to react).

- Calculated changes in core geometry have to be limited in such a way that the core remains amenable to long term cooling, and the CRs need to remain movable.
- There should be sufficient coolant inventory for long term cooling.

5.4 Selection of Initial and Boundary Conditions

There are a number of input parameters for analysis, corresponding to the NPP status prior to the accident, which have a major influence on the results; they are often called key parameters. Among them, plant initial conditions and possible bounding values for some process variables play an important role. Initial and boundary conditions have to be selected, either realistically or conservatively (depending on the approach selected), from a range depending on the plant operational mode and on the parameter uncertainties. These uncertainties include technically acceptable tolerances, calculation uncertainties or measurement.

A typical set of initial conditions, to be specified for thermo-hydraulic and neutronic calculations, is shown in Table 5.1. For each parameter, the conservative direction (maximum/minimum) is indicated for two basic types of analysis: for conservative analysis of core cooling (leading to a low value of the DNBR) and for conservative analysis of pressure in the primary system (leading to a high value of the system pressure). The results of an analysis depend also on the neutronic characteristics of the core, which determine the reactor power behavior during the course of accidents. The conservative approach typically aims to overestimate the reactor thermal power, which is strongly dependent on reactivity feedback coefficients and on the change of core parameters. Reactivity feedback depends on the direction of the change (increase or decrease) of the parameter under consideration. The direction may change during the course of the accident, and therefore the influence of feedback coefficients may also vary during the process.

Parameter	Conservative direction	
	Core cooling	System pressure
Reactor power	Max.	Max.
Reactor residual heat	Max.	Max.
Reactor coolant flow	Min.	Min.
Reactor core bypass	Max.	Min.
Reactor coolant temperature	Max.	Min.
Reactor coolant pressure	Min.[a]	Max.
Steam generator level	Min.	Min.
Steam pressure	Max.	Max.
Feedwater flow	N/A (consistent with power)	N/A (consistent with power)
Pressurizer level	Min.	Max.
Power peaking factor	Max.	Max.
CR worth available for reactor scram	Min.	Min.

[a] For LOCA analysis, a maximum value should be selected. For ATWS and SA analysis, best estimate plant initial conditions are typically acceptable, even for design and licensing type analyses.

Table 5.1: Typical Initial Conditions for Plant Accident Analysis

Chapter 6

Design Basis Accident Analysis with PCTRAN

6.1 Introduction

Design Basis Accidents or DBAs are postulated accidents to which a Nuclear Power Plant, its systems, structures and components must be designed and built to withstand loads during accident conditions without releasing the harmful amounts of radioactive materials to the outside environment. Any DBA is controlled by the reactor safety systems with insignificant off-site consequences, but may require long shutdown for correction or repair.[9]

6.2 Why DBA Analysis is so important?

The concept of DBA is very useful both for normal and abnormal operation. Design basis accidents are used in the design of a Nuclear Power Plant to establish the performance requirements for reactor structures, systems and components.

Design basis accidents are classified by type of initiating events. For example:

- Reactivity initiated accidents cover such initial events as single rod withdrawal or voiding of control rod channel cooling system.
- Single or multiple main circulation pump trip are examples of loss-of-flow transients.
- Loss-of-coolant accidents include full or partial breaks of group distribution header, breaks of main steam line in different locations, etc.
- Loss of preferred power or turbine trip can be referred to accidents initiated by equipment failure.
- Fuel handing accidents cover failures during refueling machine operation.
- Flooding, fire, earthquakes are examples of the last group of accidents referred as other accidents (external events)

More serious accidents that may involve significant core degradation and/or pose the real danger of a significant release of radiation to the environment are classified as Beyond Design Basis Accidents (BDBAs). These accidents have an extremely low probability of occurrence, so low that full control of them is not considered in the design.

6.3 PCTRAN Simulation of Design Basis Accidents

PCTRAN will be used to simulate different Design Basis Accident. The accidents that will be covered in this chapter are listed below:

- Steam Generator Tube Rupture (SGTR)
- Main Steam Line Break (MSLB)
- Loss of Coolant Accident (LOCA)
- Loss of Offsite Power (LOOP)

6.3.1 Steam Generator Tube Rupture (SGTR) Analysis

6.3.1.1 Introduction

Steam Generator is an integral part of Nuclear Steam Supply System (NSSS) performing an important role of producing steam by transferring heat from primary coolant system to secondary coolant system. In a typical PWR Steam Generator, the high pressure primary coolant exchanges heat with a low pressure secondary coolant and forms steam which drives the turbine. As a result, steam generator reliability and performance bear significant concerns in the safe operation of a PWR power plant.[13] During continuous and long term operation, these steam generators experience problems due to corrosion, flow induced vibration, stress, fatigue failure etc. which degrades the structural integrity of U-tubes posing a great risk of radioactivity release into the atmosphere. To prevent the release of radionuclides, the steam generator tubing must be essentially free of cracks, perforations, and corrosion resistant. A number of Steam Generator Tube Rupture (SGTR) events have occurred in Pressurized Water Reactors (PWR), including those at Point Beach, Surry, Mihama, Oconee, Prairie Island and Gina. [10] SGTR causes a direct flow path between primary and secondary system and results in the release of radioactive materials into the environment.

The following sections of this paper include a brief description SGTR Event Scenario, setting up Initial Conditions in PCTRAN, results and discussion.

6.3.1.2 SGTR Event Scenario

SGTR is categorized as "Class 5" grade accident as stated in NRC Accident Classification [11].In the event of SGTR, automatic reactor trip could occur on either low pressurizer pressure or over-temperature delta-T and subsequently, safety injection (SI) signal would be generated on low pressurizer pressure. Reactor coolant system (RCS) makeup would be performed by high pressure injection (HPI). For smaller tube failures, reactor trip and/or HPI actuation may be done manually. Main Feedwater (MFW) is tripped on SI signal and Auxiliary Feedwater (AFW) is automatically actuated to supply coolant to all SGs.[12] The operators are required to identify the ruptured SG and isolate it by closing the main steam isolation valve (MSIV) and other potential leak paths. In addition, the operators should perform depressurization of RCS by secondary cooling with use of AFW/MFW and steam dump or the pressurizer power-operated relief valves (PORVs) to minimize the leak flow. After the RCS depressurization is completed, the operators should secure HPI to prevent or limit the RCS repressurization. Long-term decay heat removal is carried out by secondary cooling, high pressure recirculation (HPR) or residual heat removal (RHR). Prior to the RHR operation, RCS should be cooled down to the RHR initiation pressure.

Table 6.1 SGTR Event Sequence

EVENT	TIME (s)
Pressurizer Spray Valve #1 Position Change: 0%	
Pressurizer Proportional Heater Capacity Change: 0%	0.0
Tube Rupture Initiation (Failure Fraction = 100%)	5
Pressurizer Backup Heater Capacity Change: 100%	10.5
Low Pressurizer SCRAM (146.0 bar)	16.5
Reactor SCRAM	
TCV Valve #1 Position Change: 0%	
Turbine Trip	19.5
TBV Valve #1 Position Change: 100%	
TBV Valve #1 Position Change: 0%	20.0
TCV Valve #1 Position Change: 100%	22.0
Safety Relief Valve #1 Position Change: 100%	27.0
Safety Relief Valve #2 Position Change: 100%	33.0
Pressurizer Backup Heater Capacity Change: 0%	
Pressurizer Proportional Heater Capacity Change: 0%	37.0
HPI Pump #1 Position Change: 100%	
HPI Pump #2 Position Change: 100%	
Letdown Valve #1 Position Change: 0%	
HPI Pump #1 Position Change: 0%	47.0
Containment Vent Valve #1 Position Change: 0%	
Safety Relief Valve #2 Position Change: 0%	54.0
Safety Relief Valve #1 Position Change: 0%	60.5

6.3.1.3 Initial and Boundary Conditions

Steady state condition of the plant before the accident is as follows:

- Reactor Power: 3000 MW (100%)
- RC Pressure: 155 bar
- Core Average Temperature : 306.9 °C
- SG Pressure: 70 bar
- Time in Life: BOC (Beginning of Cycle)
- SG Water Level: 2.45 m
- Pressurizer Level: 6.96 m
- Max. Fuel Temperature: 788.9°C
- No interaction of operators during the accident.

For this analysis, 100% rupture of a single U-tube of steam generator has been taken into consideration. The duration between the initiation and complete rupture of the tube is 5 seconds. After setting up the initial and boundary conditions, they are used to provide a necessary conservatism of calculation results in terms of releases from the affected steam generator.

6.3.1.4 Analysis of SGTR Simulation Results

The tube rupture is assumed to be initiated at 5 seconds in one of the two Steam Generators (in this case SG 'A') when the reactor is operated at full power (100%). The simulation is performed for about 300 seconds to study the various transient behavior of plant parameter during the accident.

After the initiation of SGTR, the Primary Pressure drops instantaneously from 15.5 MPa to about 8 MPa due to opening and closing of Relief Valve (Figure 6.1). This rapid depressurization in the primary system was followed by a “Reactor Trip”. The “Reactor Trip” tends to decrease “Average Temperature” (Figure 6.2) of the reactor coolant while primary pressure drop is caused due to rupture of SG Tube. Pressurizer Heaters were turned on to increase the pressure on Primary side. Pressurizer pressure and water level (Figure 6.4) continued to decrease. The pressure in both SG at first increased in the early phase of transient followed by closing of turbine throttle valve during “Turbine Trip” at 19.5 seconds. The pressure in SG ‘A’ was slightly higher than pressure in SG ‘B’ as shown in (Figure 6.1). Since the “Primary Side” pressure was higher than “Secondary Side” pressure, there will be a considerable amount of coolant leakage as reactor coolant passes from “Hot Leg” to “Cold Leg” within the SG.

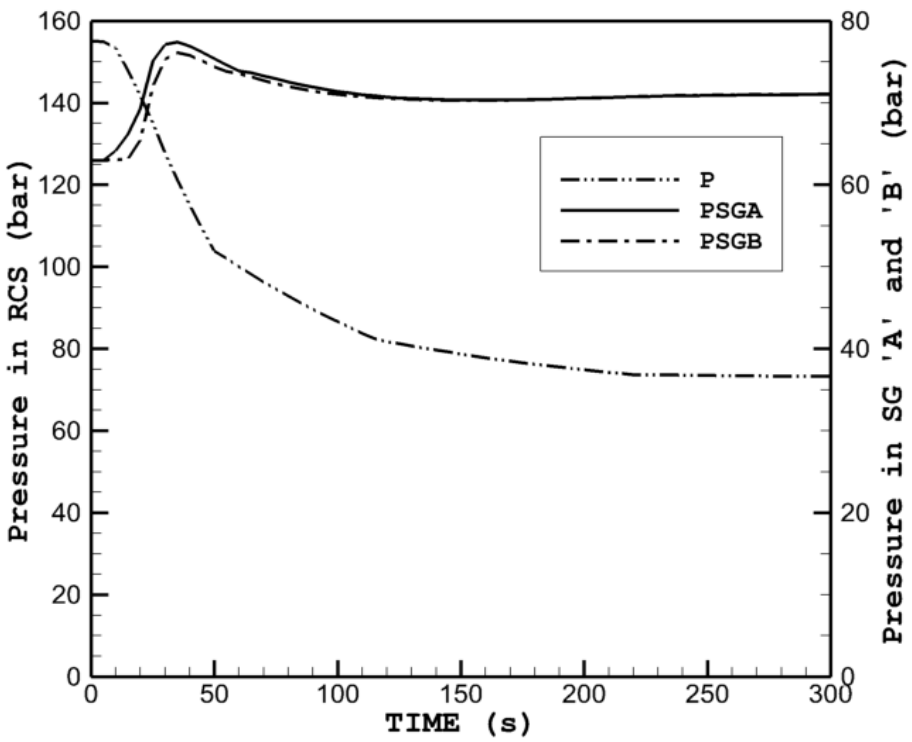

Figure 6. 1 Pressure in Reactor Coolant, Primary and Secondary System (bar)

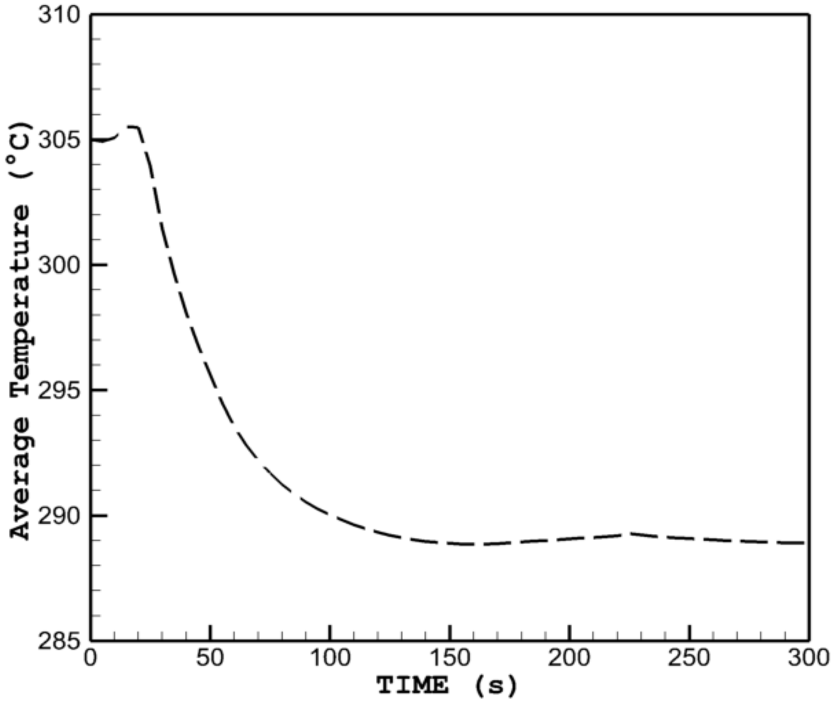

Figure 6. 2 Average Temperature (°C) in Reactor Core

Figure 6.3 shows the break flow rate of SG 'A' after the accident. The primary coolant leakage peaks about 3000 ton/hr (833.33 kg/s) instantaneously following tube rupture at about 15 seconds before PORV opened but the after PORV is opened,[14] leakage dropped to 500 ton/hr.(138.89 kg/s) after 300 seconds and will be eventually dropped to zero due to equalized pressure between both system.

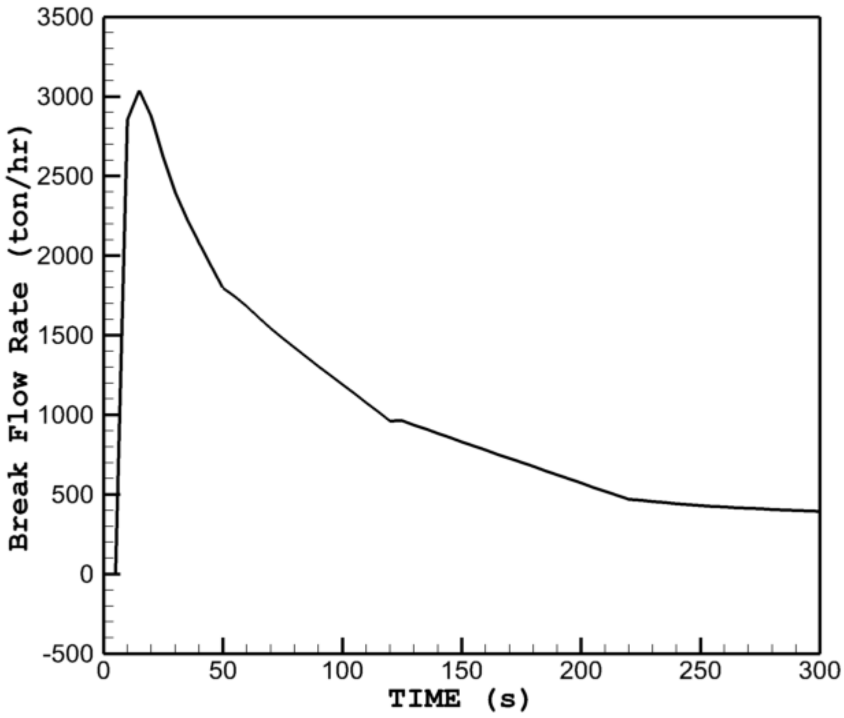

Figure 6. 3 Break Flow rate (ton/hr.) in affected SG 'A'

Figure 6.4 and 6.5 shows the water level in the Pressurizer and both Steam Generator. As Figure 6.4 shows, pressurizer water level drops rapidly from the initial value of 6.96 m and is emptied at about 50s following the transient. Both SG 'A' and 'B' mixture level drops instantaneously at the beginning of accident (Figure 6.5). This is caused by collapsing of vapor bubble due to loss of heat source following reactor trip. [15] SG 'A' level increased later due to rupture flow from the primary side (SG 'B' is isolated) and due to increase of secondary pressure, caused by turbine trip.

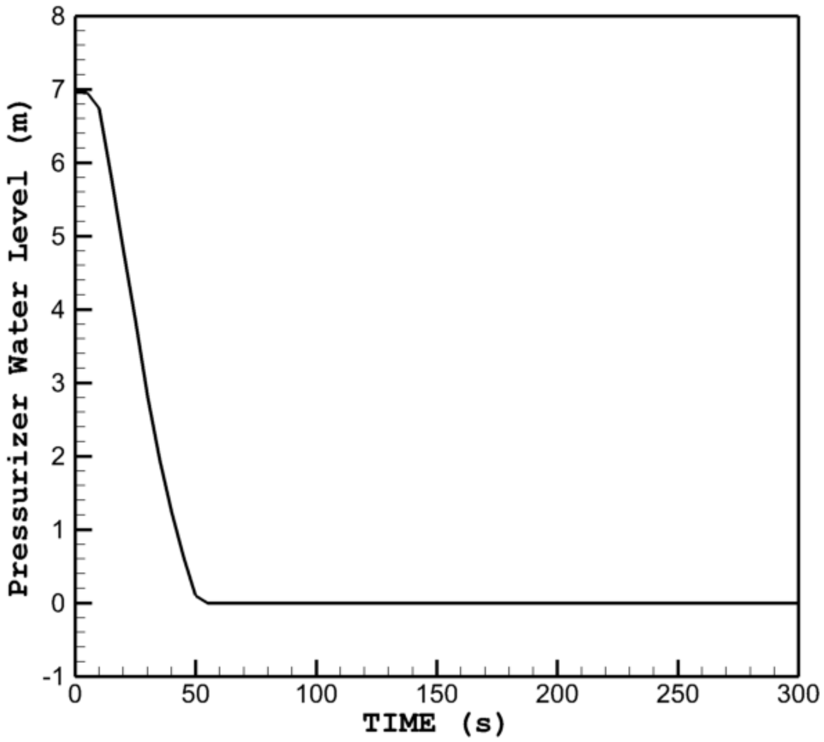

Figure 6. 4 Water Level in Pressurizer (m)

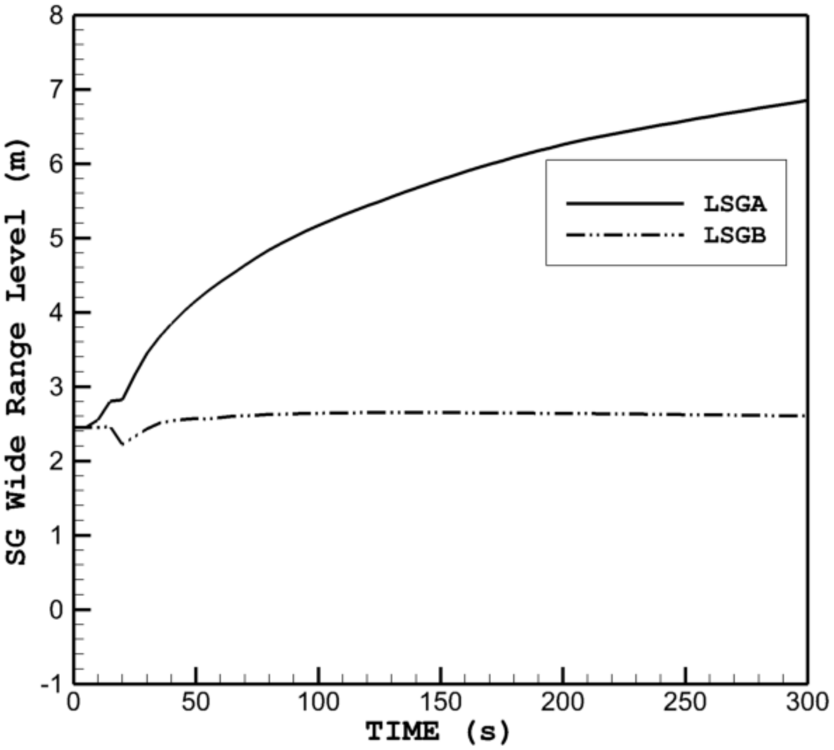

Figure 6. 5 Steam Generator Wide Range Level (m)

The temperature in “Hot” and “Cold” leg of affected SG ‘A’ is shown in Figure 6.6 which indicates the decrease of temperature at both leg although the temperature in ‘Cold’ leg increases at the initiation of the accident. ECCS, comprising of High Pressure Injection, Low Pressure Injection is provided after the pressure drops below a certain value. In this particular case, at about 100 seconds when the coolant pressure drops to approximately 80 bar (8 MPa) the ECCS is activated to prevent the core from being damaged due to excessive heat generation. This is due to the fact that following loss of coolant pressure, the primary coolant may get boiled at lower temperature and creates water vapor around the reactor core which has lower heat transfer capability in comparison to water. As a result, the heat produced within the reactor core can’t be transferred to coolant resulting in the heating up the core.

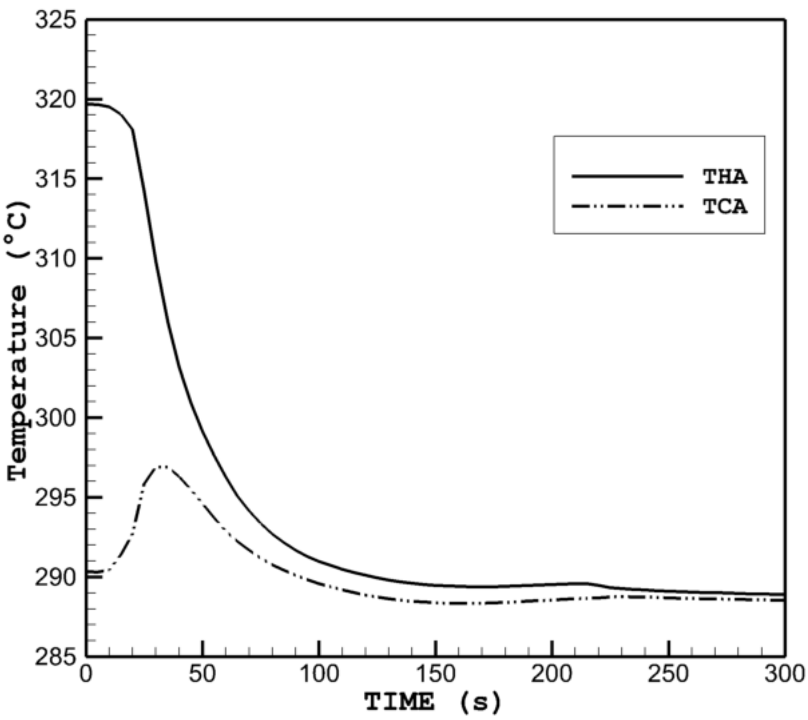

Figure 6. 6 Temperature in “Hot” and “Cold” Leg of SG ‘A’

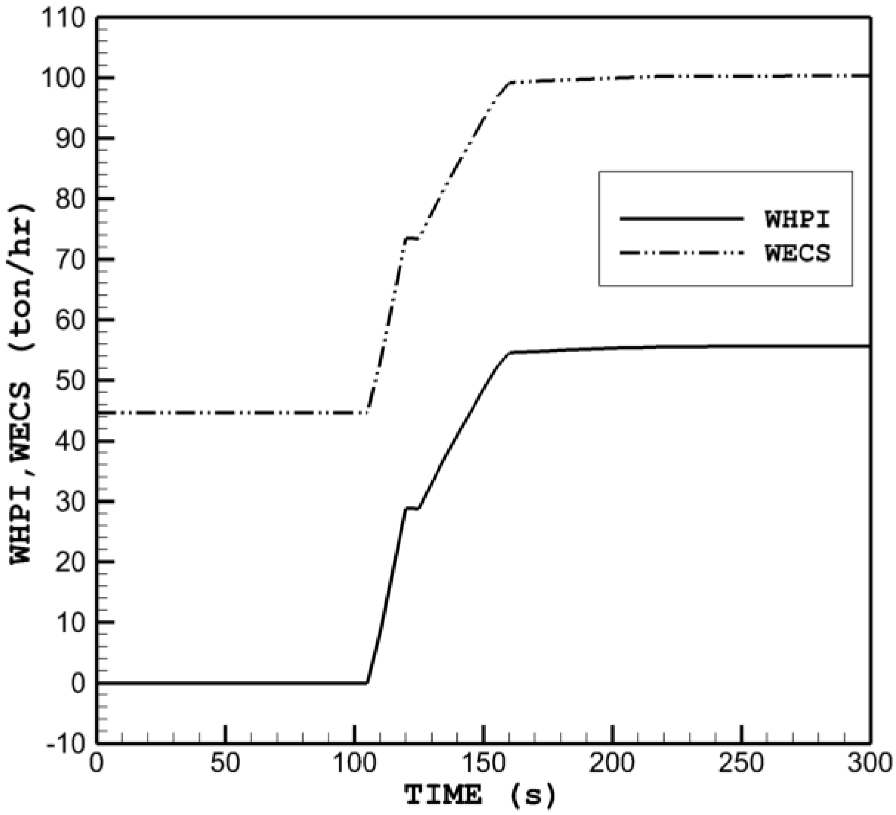

Figure 6. 7 HPI and ECCS Flow Rate (ton/hr.)

As shown in Figure 6.8, the supply of feedwater into affected SG 'A' reduces rapidly and is eventually stopped within 200 seconds of the transient. The production of steam follows the same manner (Figure 6.9). As the temperature of "Hot" leg approaches to "Cold" leg temperature, there will be no net produced steam.

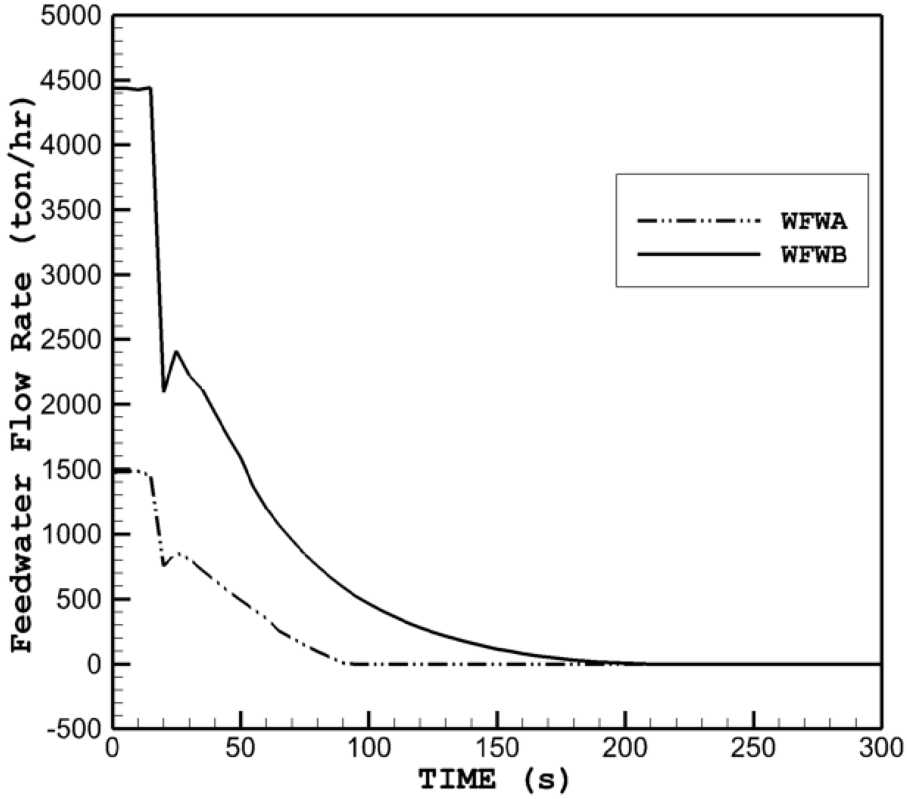

Figure 6. 8 Feedwater Flow Rate in SG 'A' and 'B' (ton/hr.)

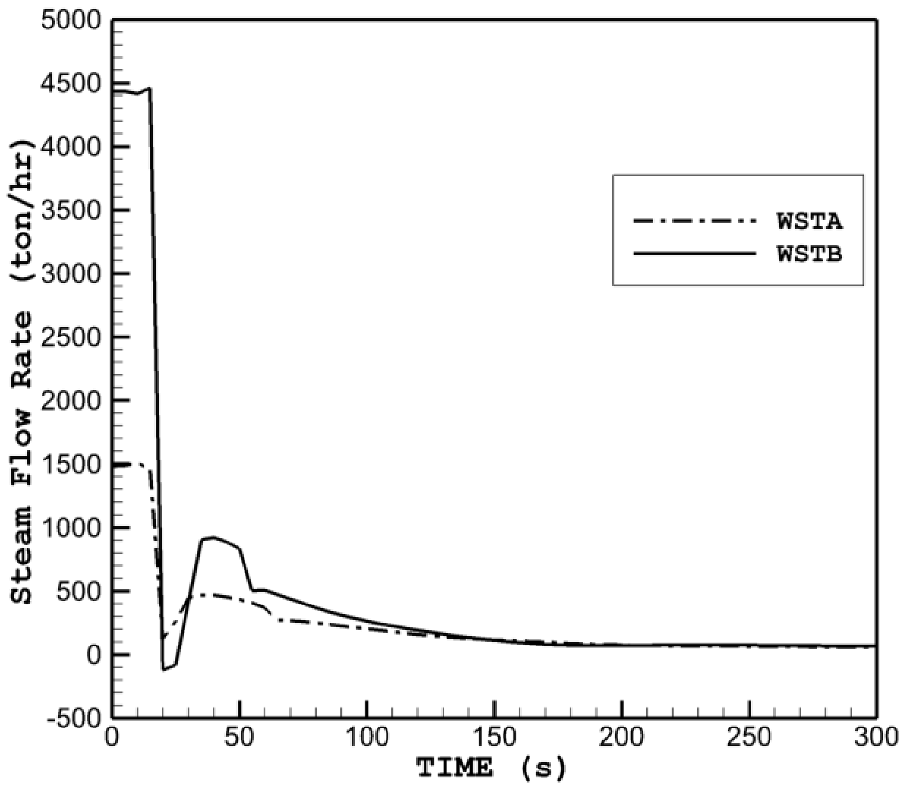

Figure 6. 9 Steam Flow Rate in SG 'A' and 'B' (ton/hr.)

Void Fraction and DNB Ratio has been shown in Figure 6.10 and 6.11 respectively on a percentage basis. Void fraction is the fraction of vapor present in a certain volume of two phase mixture (in this case “reactor coolant-vapor”). Figure 6.10 indicates that the amount of vapor in the coolant started to increase at about 50 seconds after the transient. To prevent the leakage of coolant into the secondary system, primary system is depressurized in the early phase of the transient. As a result the coolant pressure is decreased so as its saturation temperature. This leads to boiling of the coolant to some extent forming vapor within the primary system. However, void formation can be controlled by restoring the primary pressure to its nominal value.

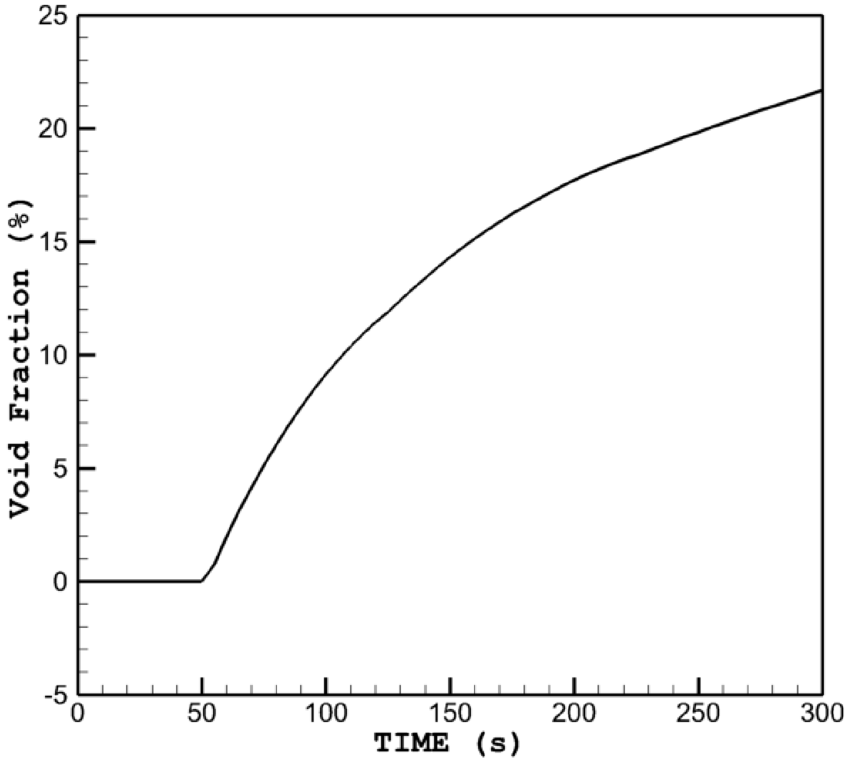

Figure 6. 10 Void Fraction in Reactor Coolant (%)

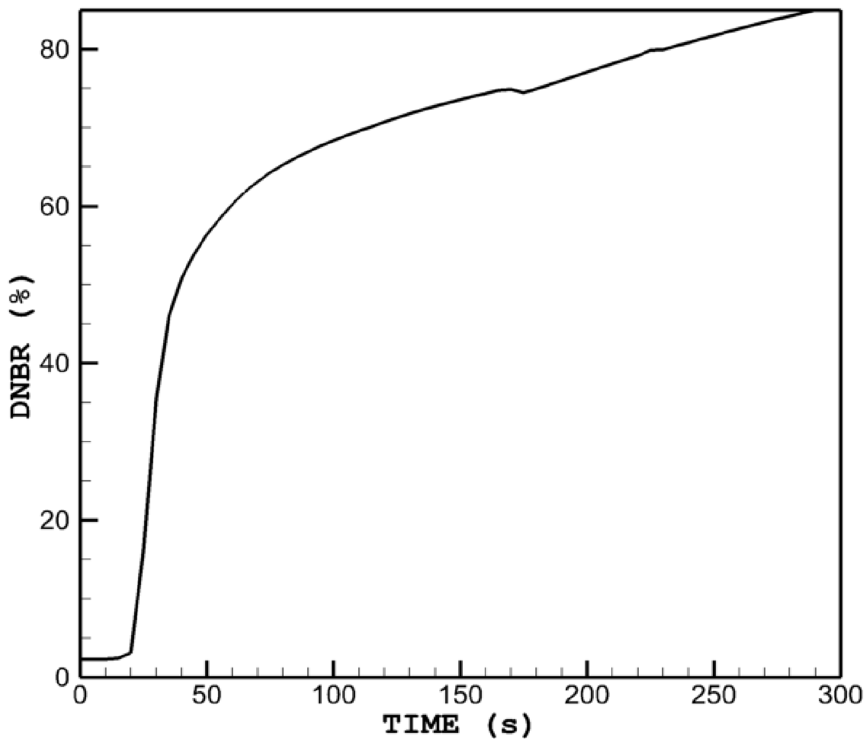

Figure 6. 11 Departure from Nuclear Boiling Ratio (%)

The fuel surface and cladding temperature are shown in Figure 6.12. After reactor shut down, reactor power stays in nominal level with in 20 s from the beginning then in a short time it will decrease from nominal value to decay heat (Figure 6.13). Since, the fuel temperature is proportional with heat generation in fuel surface unit, there for it will reduce with decrease of reactor power.

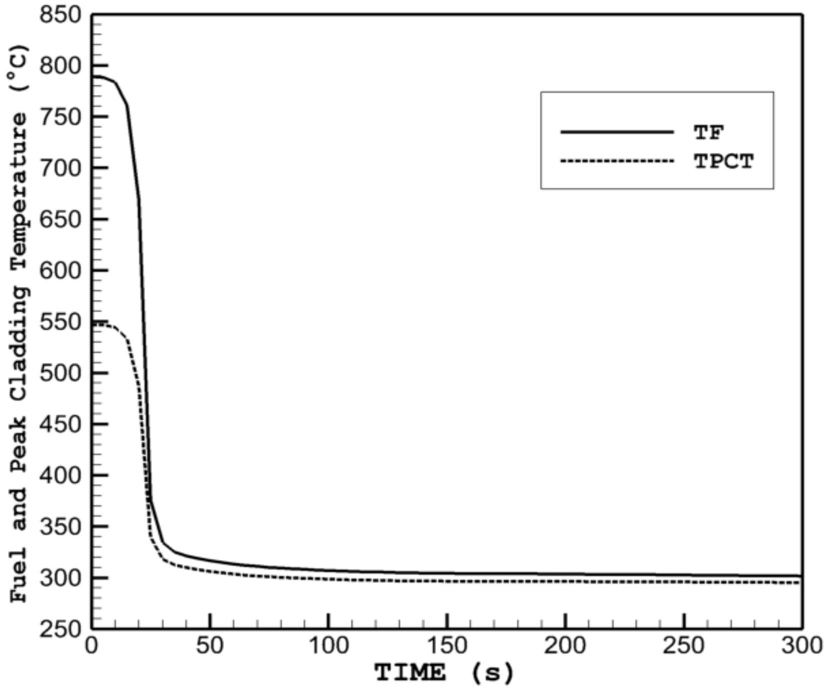

Figure 6. 12 Fuel and Cladding Temperature (°C)

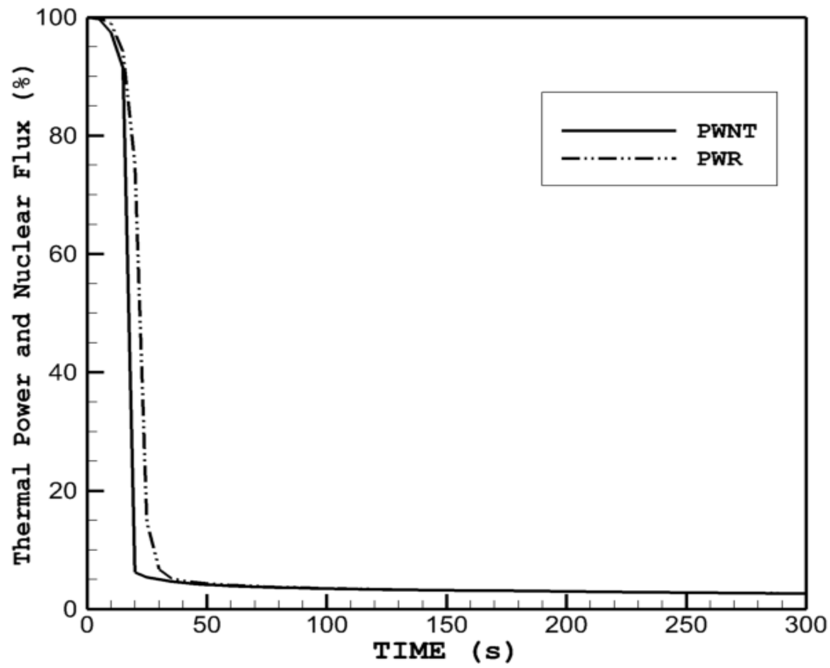

Figure 6. 13 Core Thermal Power and Nuclear Flux (%)

6.3.1.5 Discussion

The thermal-hydraulic response of Pressurized Water Reactor during SGTR accident is presented and analyzed in this section. A plant specific PCTRAN model (VVER-1000) was used for simulating the system behavior during the transient. The simulation was performed assuming single tube rupture in the secondary system. The results obtained from the calculation are in good agreement with experimental data conducted by Lin, Wassel, Kalra and Singh [10]. It is found from the study that, the plant safety system (both active and passive) is capable to restore the plant in steady state condition without operator interactions and thus maintaining the acceptance criteria for any Design Basis Accidents (DBA) as prescribed in Safety Report Series [9] by IAEA . As it is seen from the simulation results:

- The pressure in the primary coolant and in the main steam system is maintained below a prescribed value (typically 110% for DBAs). This criterion ensures that the structural integrity of the reactor coolant boundary is maintained. The pressure in the primary coolant system decreases from 17.5 MPa to approximately 8 MPa during the whole transient. Thus, the ultimate value of coolant pressure of the primary side is not reached, and the acceptance criterion is met.
- The reactor is ensured to be filled with cooling water during the transient by activating ECCS. This protects the fuel rod from being uncovered and prevents melting of core.
- The peak fuel cladding temperature (788.9°C) doesn't exceed the prescribed value of 1200°C which ensures that the melting and structural deformation of core integrity can be avoided.

It is expected to provide useful information in understanding the plant responses to SGTR event and evaluating the effectiveness of existing safety system and operator actions to mitigate the consequence of the transients.

6.3.2 Main Steam Line Break Analysis

6.3.2.1 Introduction

The main purpose of this analysis is to investigate the reactor responses due to accidental Main Steam Line Break. The consequences of a MSLB considerably depend upon several system parameters: initial power level; location of break; size of break; safety systems that are operational; control systems that are operational; and possible other failures that could occur. It could also be followed by reactor criticality, which is studied carefully.

6.3.2.2 Main Steam Line Break Event Sequence

Main Steam Line Break (MSLB) is identified as a Design Basis Accident (DBA) occurred in NPP. This MSLB Accident is initiated by a full size break of one of the main steam line at outlet of one SG which may occur either inside or outside the containment. In MSLB, very large amounts of steam are removed from the broken steam generator that significantly reduce the primary coolant temperature. This could increase power due to the effect of the negative moderator temperature coefficient of reactivity. The accident is supposed to occur at end of cycle with the core at hot zero power condition. The limiting size for a break (typically located outside the containment) is rupture of the main steam header (if relevant) up to its full size break. [16] The accident leads simultaneously to depressurization (cooling down) of the secondary circuit and loss of the secondary coolant, leading also to overcooling of the Reactor Core. Typically, the effect of cooling down dominates, leading to a non-symmetrical cooling of the Reactor Pressure Vessel (RPV) wall, to a positive reactivity insertion, to a potential criticality and to a reactor power increase in some situations regardless of SCRAM.

Table 6.2: MSLB Event Sequence

Event	Time (sec)
PZR Spray Valve #1 Position Change: 0%	0
PZR Proportional Heater Capacity Change: 0%	0
Malfunction # 3 Fraction = 100.0 %	0.5
PZR Backup Heater Capacity Change: 100%	32
Scram Low SG Press 49.00 MPa	98
MSIV #1 Position Change: 0%	98
Reactor Scram	98.5
TCV Valve #1 Position Change: 0%	98.5
Turbine trip	98.5
HPSI start high RB Press 1.30 psia	176.5
HPI Pump #1 Position Change: 100%	176.5
HPI Pump #2 Position Change: 100%	176.5
Letdown Valve #1 Position Change: 0%	176.5
HPI Pump #1 Position Change: 0%	176.5
Ctmt Vent Valve #1 Position Change: 0%	176.5
RBS Pump #1 Position Change: 100%	176.5
RBS Pump #2 Position Change: 100%	176.5
Ctmt Spray Starts 1.3 bar	176.5

6.3.2.3 Initial and Boundary Conditions

Plant data parameters controlled by the operators would define the model to represent a specific plant. For this analysis, initial conditions and plant parameters have been defined as follows:

- Reactor Power: 3000 MW (100%)
- Reactor Core Pressure: 155 bar
- Core Average Temperature: 306.9 °C
- SG Pressure: 70 bar
- Time in Life: BOC
- SG Water Level: 2.45 m
- Pressurizer Level: 6.96 m
- Max. Fuel Temperature: 788.9°C

No interaction of operators during the accident. Reactor system respond without operator actions.

6.3.2.4 Analysis of MSLB Simulation Result

Main Steam Line Break accident has been simulated with a delay time of 5 sec and ramp time of 5 sec. Failure fraction has been set to 100% of total rupture. The simulation is conducted for 300 seconds. Thermal hydraulic response has been plotted and shown in the following figures.

Figure 6.14 shows the change of Pressure in Reactor Coolant System and both Steam Generator ‘A’ and ‘B’. Initially, Reactor Coolant System Pressure is 155 bar. After the accident is initialized, the pressure began to drop moderately. Then after 160 seconds the pressure drops suddenly to approximately 143 bar and continued to increase until it reaches 147 bar before SCRAM. Both SG ‘A’ and ‘B’ experience instantaneous pressure drop following the accident. But after Reactor SCRAM at 98.5 seconds approximately, both system restore the pressure in its nominal value to stop leakage between primary and secondary system. As shown in Figure 6.15, the primary coolant temperature experiences linear reduction initially after the transient and exponentially after SCRAM at 98.5 seconds.

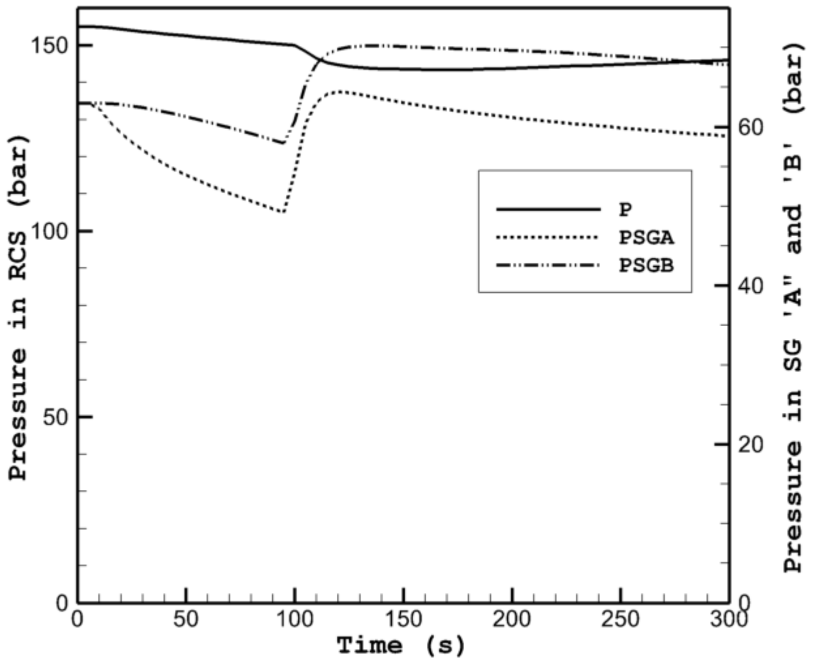

Figure 6. 14 Pressure in Reactor Coolant, Primary and Secondary System (bar)

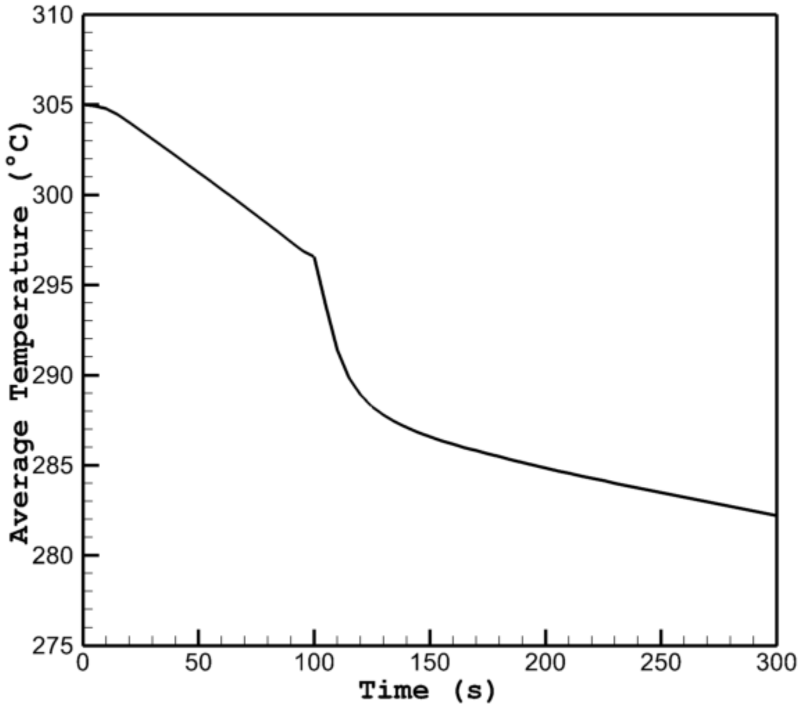

Figure 6. 15 Average Temperature (°C) in Reactor Core

Figure 6.16 and 6.17 shows the behavior of Pressurizer and SG water levels. Before the MSLB accident, the water level inside the Pressurizer is 6.9 m. With the initiation of accident, due to decrease in pressure (inside the Pressurizer), the water level decreases moderately for the first 165 seconds due to reactor scram and cooldown of RCS by faulted SG.[17] The reason for rising of water level comes from Emergency Protection (EP) make-up system.

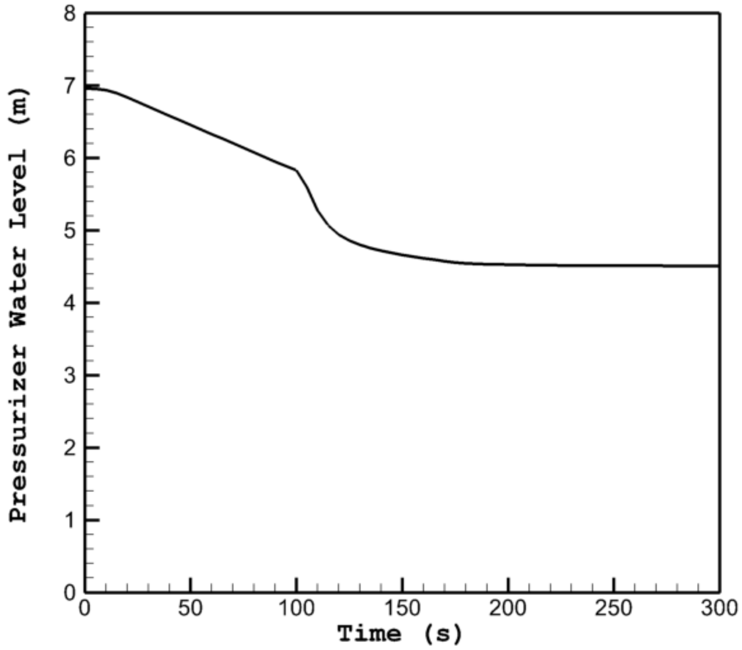

Figure 6. 16 Water Level in Pressurizer (m)

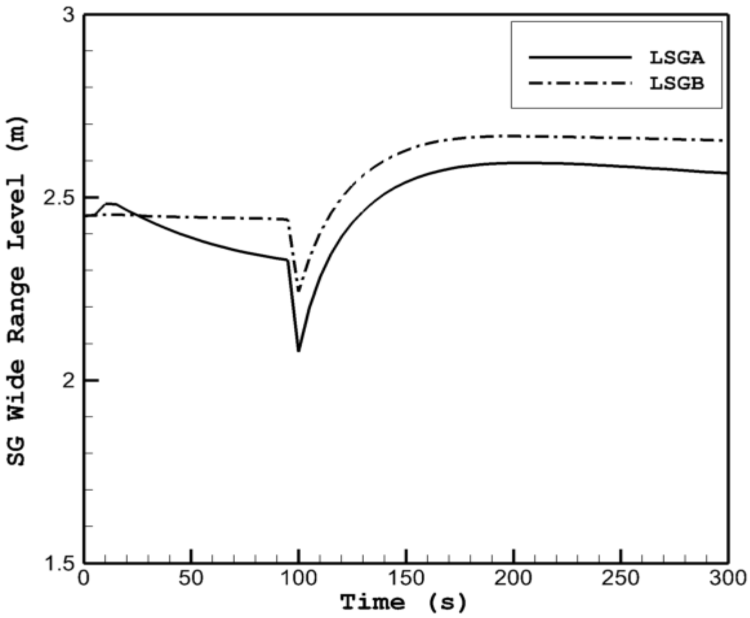

Figure 6. 17 Steam Generator Wide Range Level (m)

As shown in Figure 6.18, the production of steam starts to decrease slowly and after SCRAM the production collapses to nearly zero because of absence of temperature difference between two system.

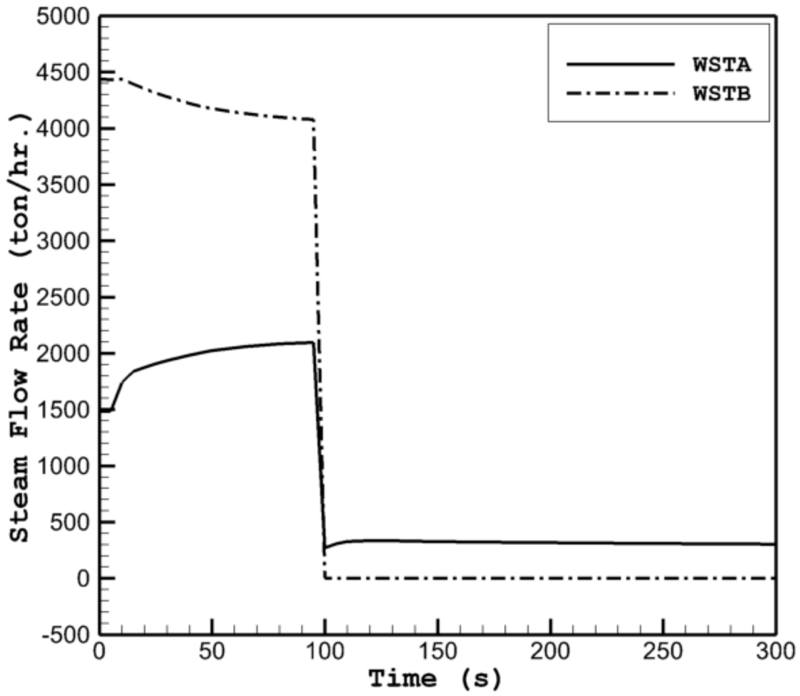

Figure 6. 18 Steam Flow Rate in SG 'A' and 'B' (ton/hr.)

Figure 6.19 indicates that the amount of vapor in the coolant started to increase at about 50 seconds after the transient. The fluctuation of void fraction is due to continuous change of pressure within the reactor coolant system which is also dependent on the opening and closing of Pressurizer Power Operated Relief Valve (PORV).

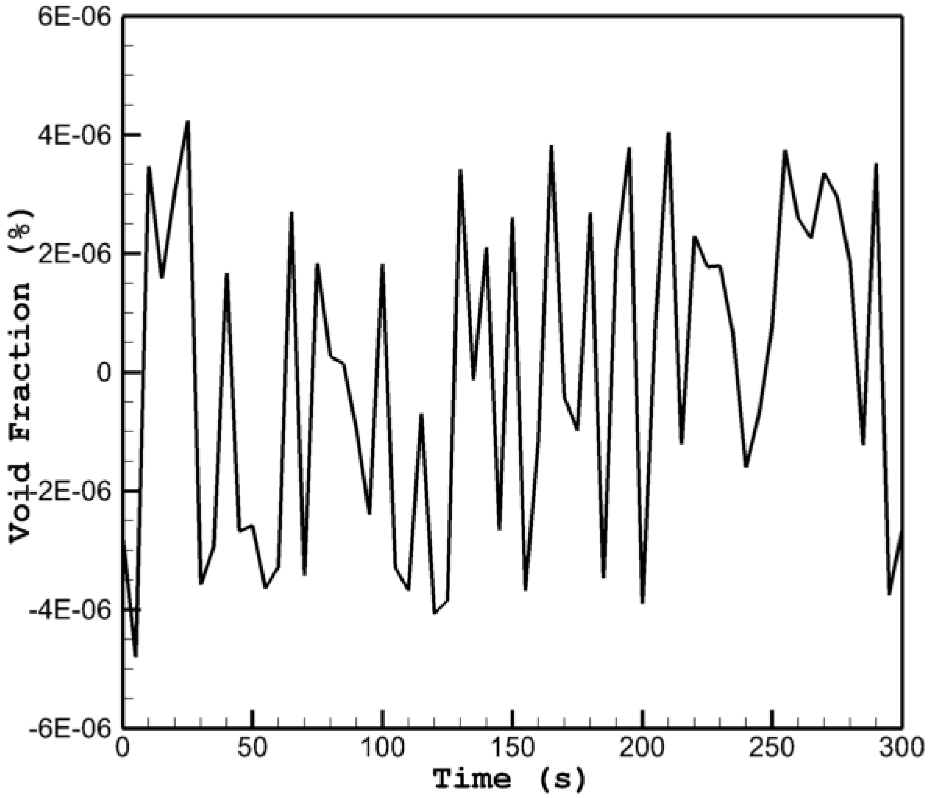

Figure 6. 19 Void Fraction in Reactor Coolant (%)

Figure 6.20 and 6.21 show the Fuel-Cladding Temperature and Reactor Power after MSLB accident. The reactor follows a steady decrease in power until time reaches about 160 seconds. After 160 seconds, the reactor power instantly drops down nearly to zero because of depressurization which reduces primary coolant temperature. As the primary system temperature drops, the heat transfer to the faulted SG and the primary system cool down rate will be reduced. This causes the decreases in reactor power. Since, the fuel temperature is proportional with heat generation in fuel surface unit, there for it will reduce with decrease of reactor power.

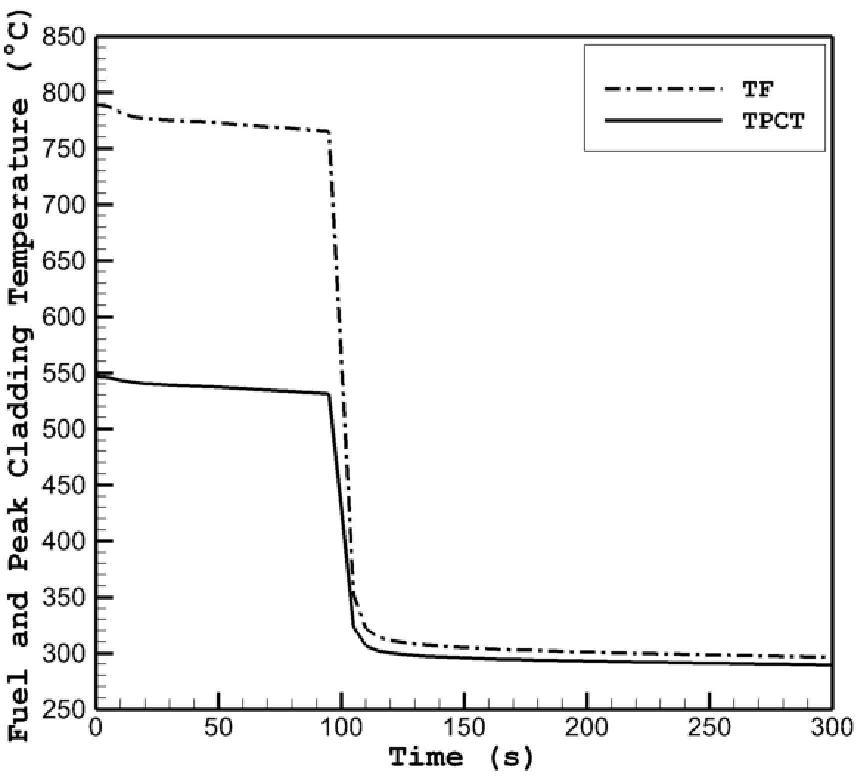

Figure 6. 20 Fuel and Cladding Temperature (°C)

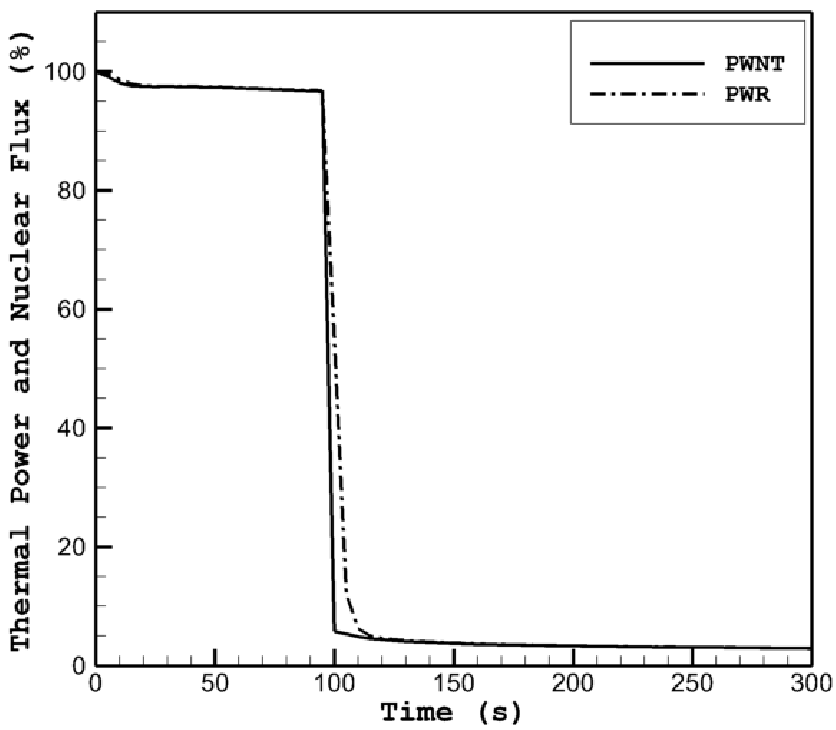

Figure 6. 21 Core Thermal Power and Nuclear Flux (%)

6.3.2.5 Discussion

The simulation study conducted by the analysis of several safety aspects of MSLB and SGTR accidents, with no possible violation of acceptance criteria as follows:[18]

- There is no possibility of reactor power increase or reactor criticality after its shutdown due to a substantial decrease of the core inlet temperature, typically in one section of the core adjacent to the affected loop. In the case of a coincident loss of power supply, the DNBR is reduced and a boiling crisis can occur in the reactor core.
- Loss of secondary coolant is not so significant due to steam outflow; No significant or rapid non-symmetrical reduction of coolant temperature at the RPV inlet as followed by high pressure emergency coolant injection, potentially affecting the vessel integrity.
- Containment pressure and sub-compartment differential pressure did not increase, leading to pressure loading of the containment structure (in the case of a break inside the containment). [19]
- Loss of secondary coolant can lead to SG Tube bundle to uncover, primary circuit temperature increment and high pressure injection system (HPIS) activation. This situation did not lead to an increase of primary pressure, filling of the pressurizer and over pressurization of the primary circuit.

6.3.3 Loss of Coolant Accident Analysis

6.3.3.1 Introduction

Loss of Coolant Accident are those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system. Safety analysis and accident management of nuclear reactor systems during a Loss of Coolant Accident (LOCA) require a better understanding of the Thermal-Hydraulic phenomena related to the reactor transients.

6.3.3.2 Loss of Coolant Event Scenario

Table 6.3: LOCA Event Scenerio

Time (s)	Events
0	PZR Spray Valve #1 Position Change: 0%
0	PZR Proportional Heater Capacity Change: 0%
0.5	Malfunction # 2 Fraction = 100.0 %
11	PZR Backup Heater Capacity Change: 100%
16.5	Scram Lo RX Press147.0 bar
19.5	Reactor Scram
19.5	TCV Valve #1 Position Change: 0%
19.5	Turbine trip
19.5	TBV Valve #1 Position Change: 100%
20	TBV Valve #1 Position Change: 0%
25	TBV Valve #1 Position Change: 100%
34	Safety Relief Valve #1 Position Change: 100
34	PZR Backup Heater Capacity Change: 0%
34	PZR Proportional Heater Capacity Change: 0%
34.5	Safety Relief Valve #2 Position Change: 100
41.5	HPI Pump #1 Position Change: 100%
41.5	HPI Pump #2 Position Change: 100%
41.5	Letdown Valve #1 Position Change: 0%
41.5	HPI Pump #1 Position Change: 0%
41.5	Ctmt Vent Valve #1 Position Change: 0%
51.5	Safety Relief Valve #1 Position Change: 0%
51.5	Safety Relief Valve #2 Position Change: 0%
52.5	RBS Pump #1 Position Change: 100%
52.5	RBS Pump #2 Position Change: 100%
52.5	Ctmt Spray Starts 1.3 bar
239.5	TBV Valve #1 Position Change: 0%

6.3.3.3 Initial and Boundary Conditions

Steady state condition of the plant before the accident is as follows:

- Reactor Power: 3000 MW (100%)
- RC Pressure: 155 bar
- Core Average Temperature : 306.9 °C
- SG Pressure: 70 bar
- Time in Life: BOC (Beginning of Cycle)
- SG Water Level: 2.45 m
- Pressurizer Level: 6.96 m
- Max. Fuel Temperature: 788.9°C
- No interaction of operators during the accident.

The duration between the initiation and complete rupture of the tube is 5 seconds. After setting up the initial and boundary conditions, they are used to provide a necessary conservatism of calculation results in terms of releases from the affected steam generator.

6.3.3.4 Analysis of LOCA Simulation Result

Figure 6.22 shows the pressure behavior calculated during the transient. After the initiation of LOCA, the Primary Coolant Pressure drops gradually from 15.5 MPa to about 7 MPa due to opening and closing of Relief Valve. This depressurization in the primary system was followed by a "Reactor Trip".[20] Pressurizer Heaters were turned on to increase the pressure on Primary side. The pressure in SG "A" at first increased in the early phase of transient followed by closing of turbine throttle valve during "Turbine Trip" at 19.5 seconds.(Figure 6.23).

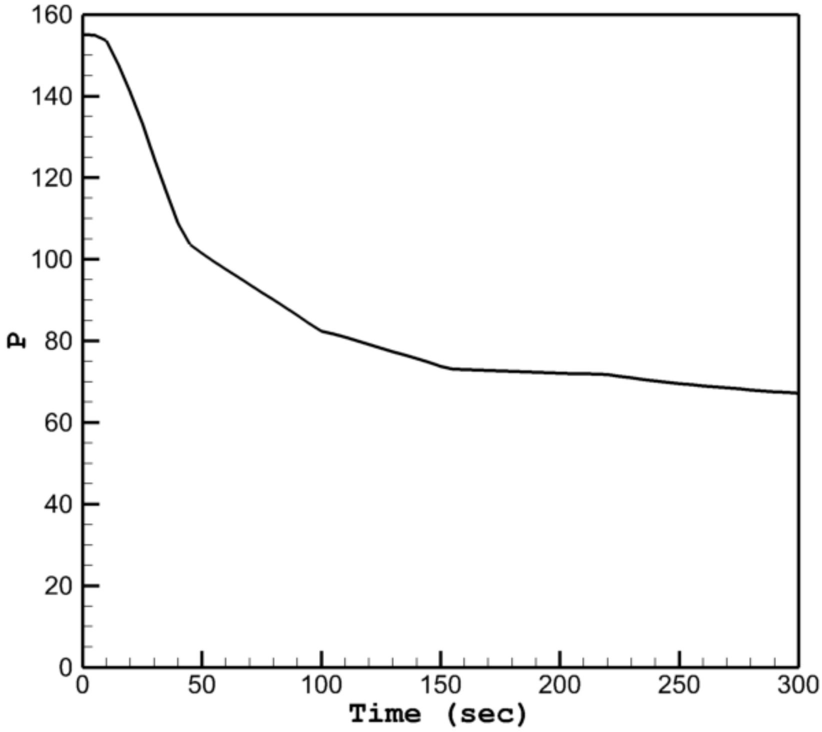

Figure 6. 22 Pressure in Reactor Coolant (bar)

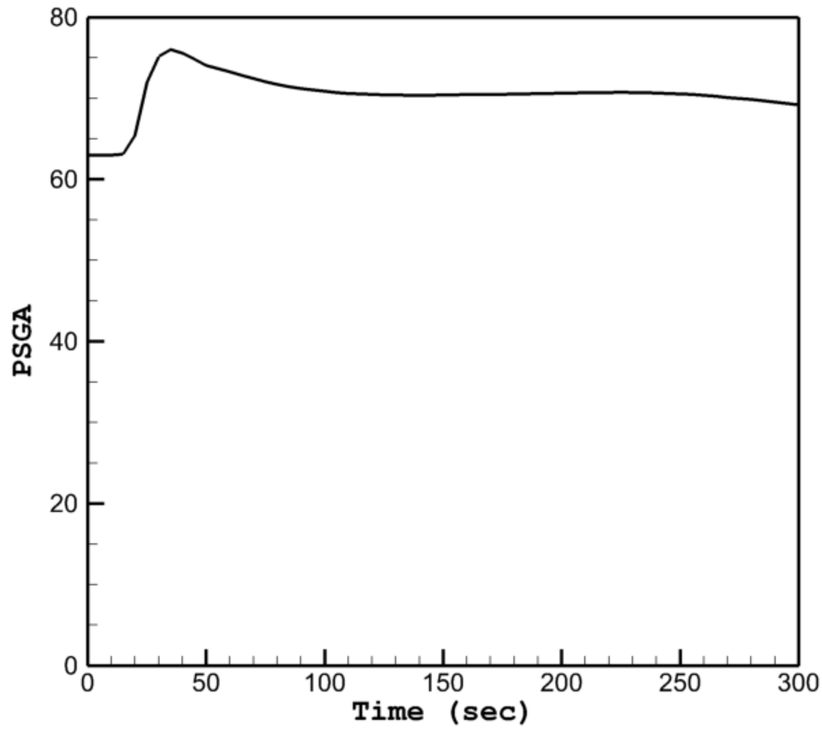

Figure 6. 23 Pressure in Primary System (bar)

The “Reactor Trip” tends to decrease “Average Temperature” (Figure 6.24) of the reactor coolant. The temperature in “Hot” and “Cold” leg of SG ‘A’ is shown in Figure 6.25 which indicates the decrease of temperature at both leg although the temperature in 'Cold' increases at the early phase of the accident. At about 41.5 seconds when the coolant pressure drops to approximately 80 bar (8 MPa) both High Pressure Injection (HPI) Pump is activated to prevent the core from being damaged due to excessive heat generation. This is due to the fact that following loss of coolant pressure, the primary coolant may get boiled at lower temperature and creates water vapor around the reactor core which has lower heat transfer capability in comparison to water. As a result, the heat produced within the reactor core can't be transferred to coolant resulting in the heating up the core.

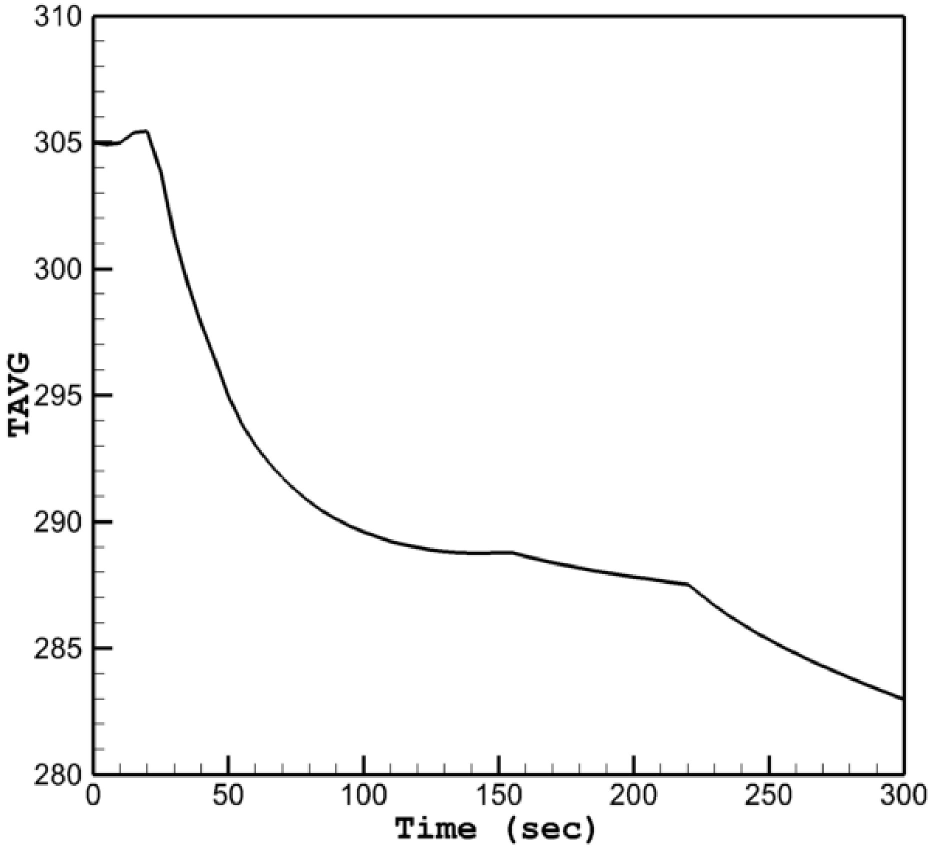

Figure 6. 24 Average Temperature (°C) in Reactor Core

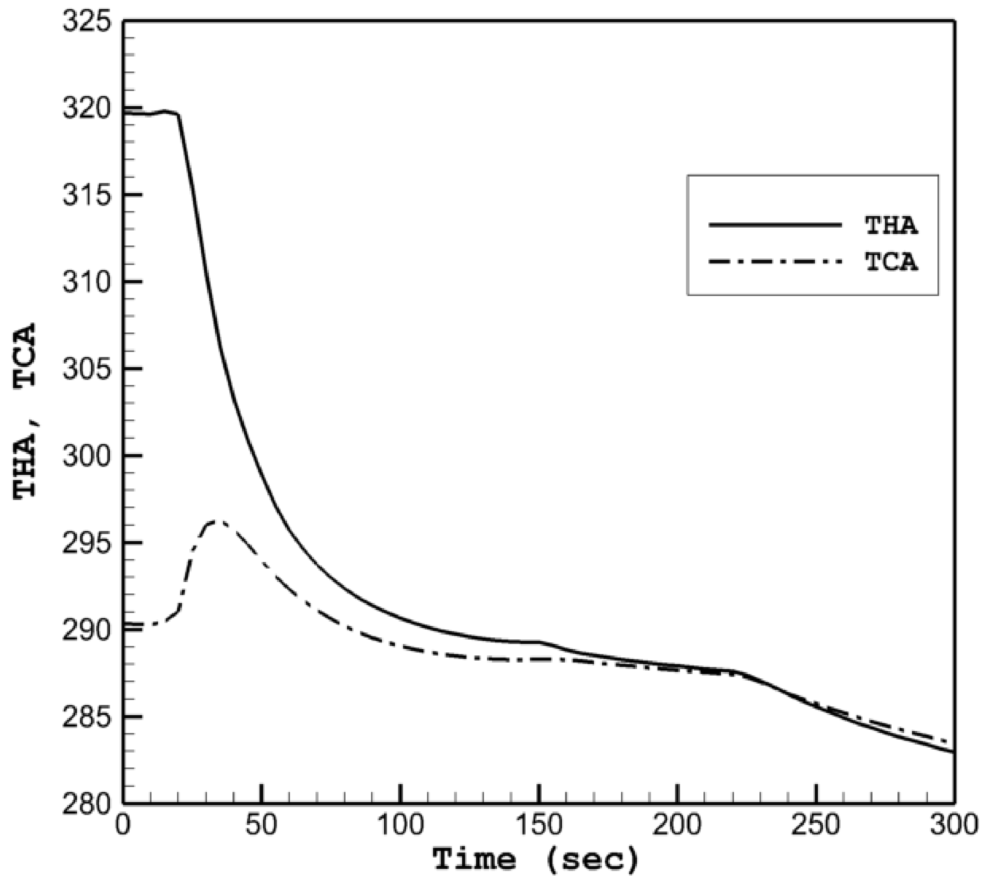

Figure 6. 25 Temperature in "Hot" and "Cold" Leg of SG 'A'

Figure 6.26 & 6.27 shows,the water level in the Pressurizer and both Steam Generator. As Figure 6.26 shows, pressurizer water level drops instantaneuosly from the initial value of 6.96 m and is emptied at about 50s following the transient. Both SG 'A' and 'B' mixture level drops instantaneously at the beginning of accident (Figure 6.27). This is caused by collapsing of vapor bubble due to loss of heat source following reactor trip.

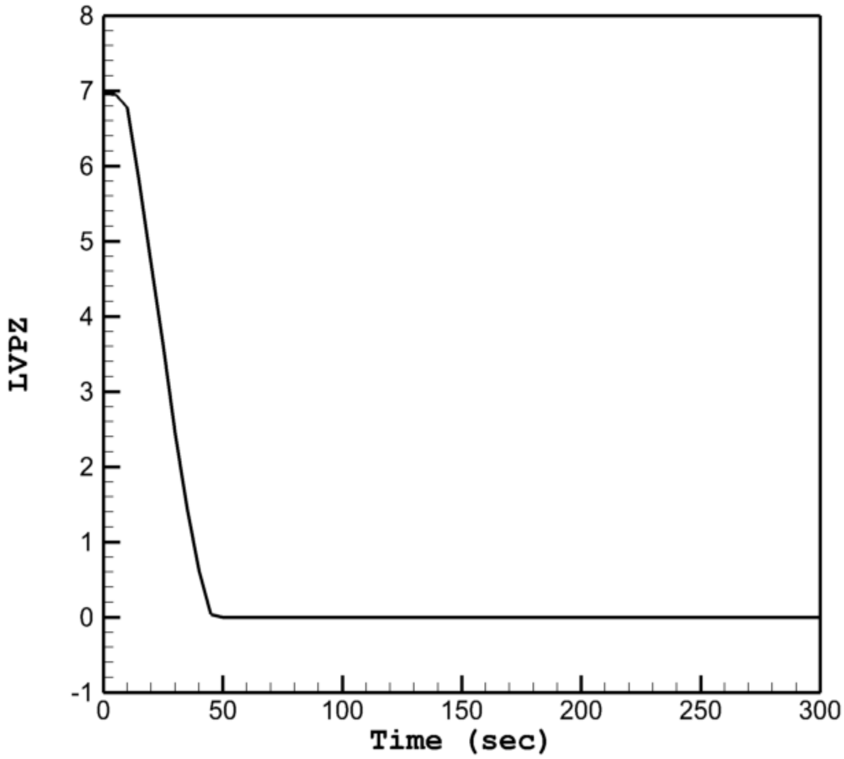

Figure 6. 26 Water Level in Pressurizer (m)

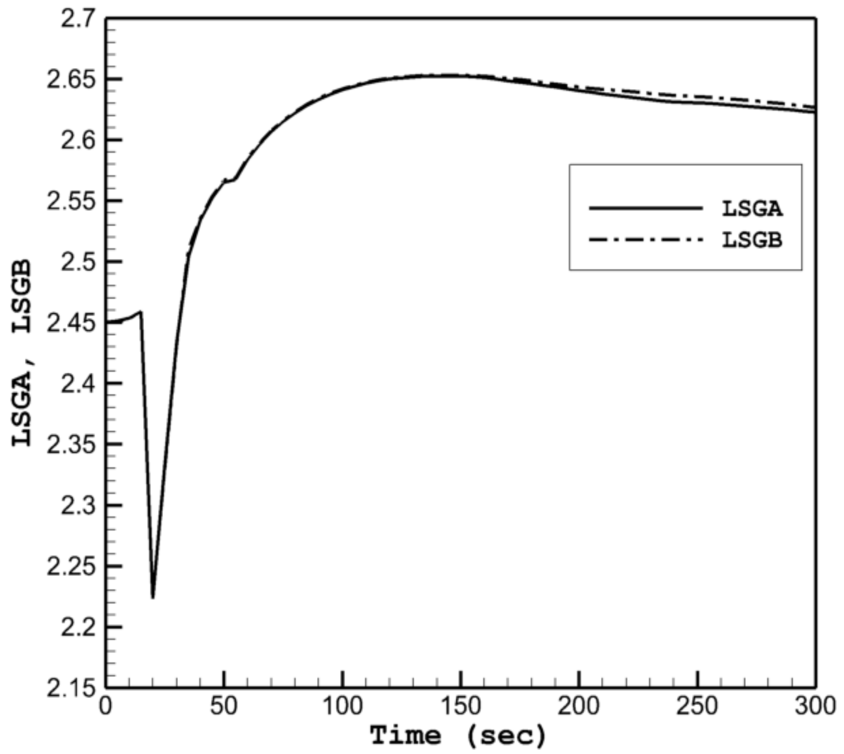

Figure 6. 27 Steam Generator Wide Range Level (m)

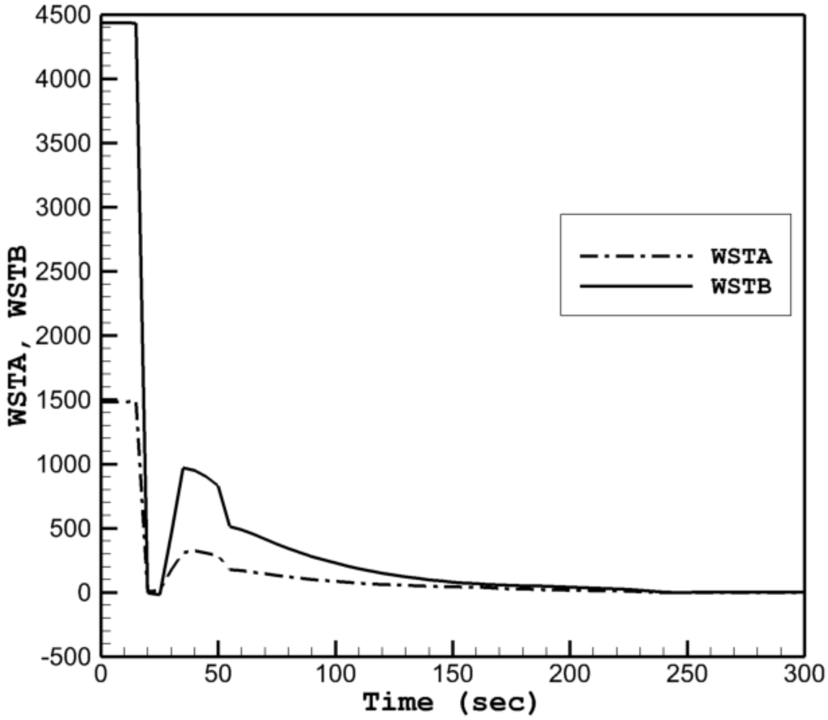

Figure 6. 28 Steam Flow Rate in SG ‘A’ and ‘B’ (ton/hr.)

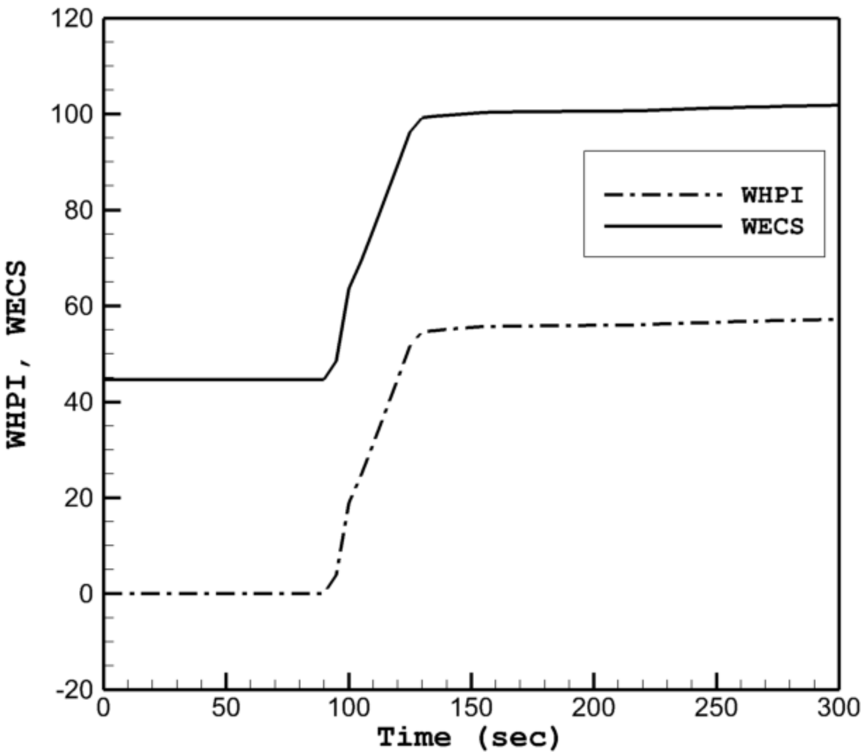

Figure 6. 29 HPI and ECCS Flow Rate (ton/hr.)

Void Fraction has been shown in Figure 6.30 on a percentage basis. Figure 6.30 indicates that the amount of vapor in the coolant started to increase at about 50 seconds after the transient. To prevent the leakage of coolant into the secondary system, primary system is depressurized in the early phase of the transient. As a result the coolant pressure is decreased so as its saturation temperature. This leads to boiling of the coolant to some extent forming vapor within the primary system. However, void formation can be controlled by restoring the primary pressure to its initial value. [21]

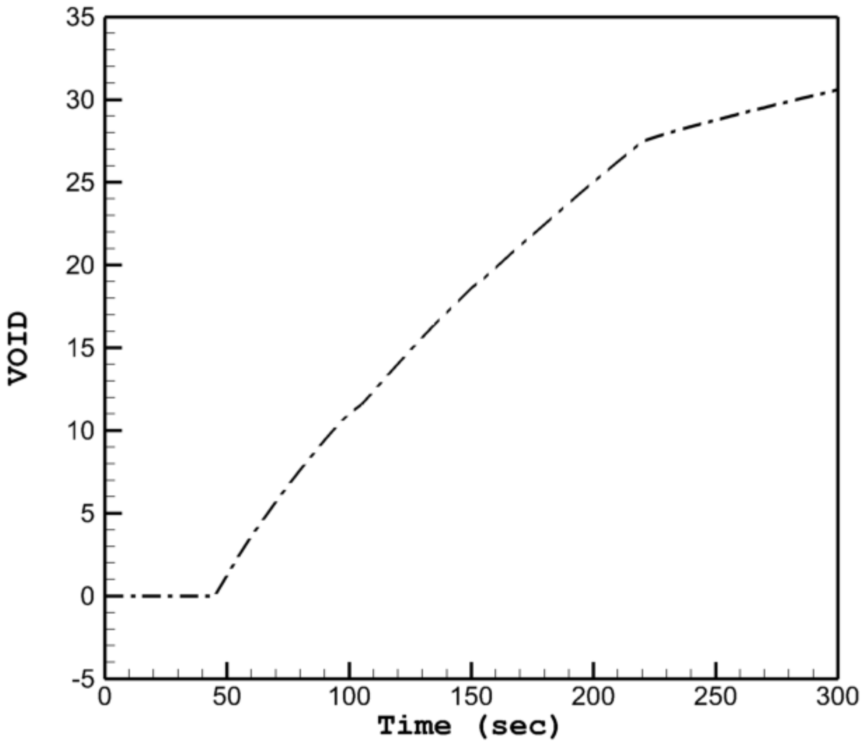

Figure 6. 30 Void Fraction in Reactor Coolant (%)

The fuel surface and cladding temperature are shown in Figure 6.31. After reactor shut down, reactor power stays in nominal level with in 20 s from the beginning then in a short time it will decrease from nominal value to decay heat (Figure 6.32). Since, the fuel temperature is proportional with heat generation in fuel surface unit, there for it will reduce with decrease of reactor power. As Figure 6.31 shows, the fuel surface temperature reduces linearly. This behavior is due to the fact that heat generation of decay heat has the same linearly reduction in a day.

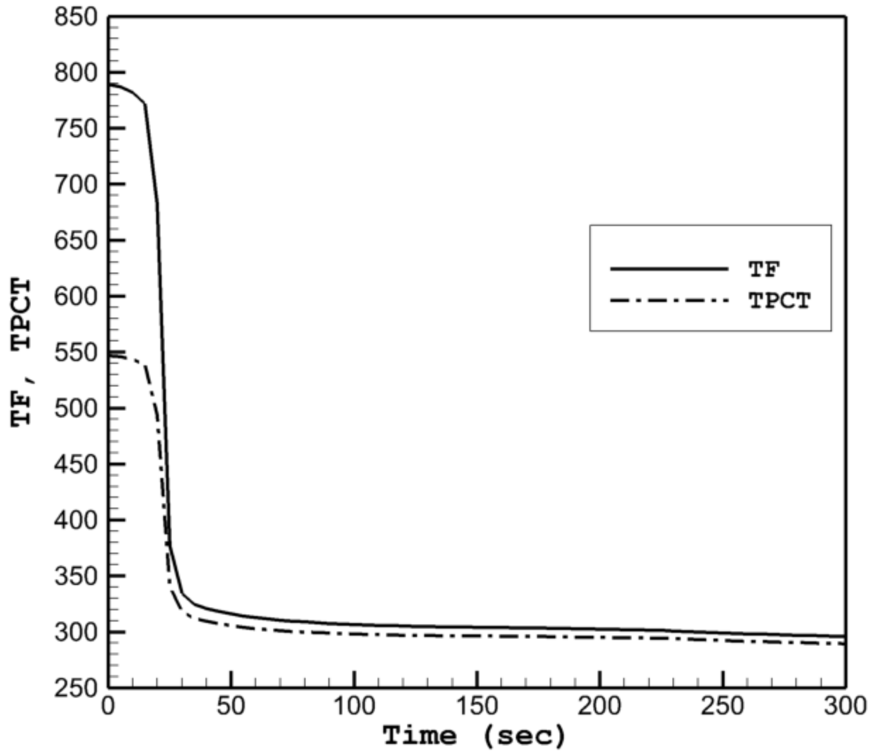

Figure 6. 31 Fuel and Cladding Temperature (°C)

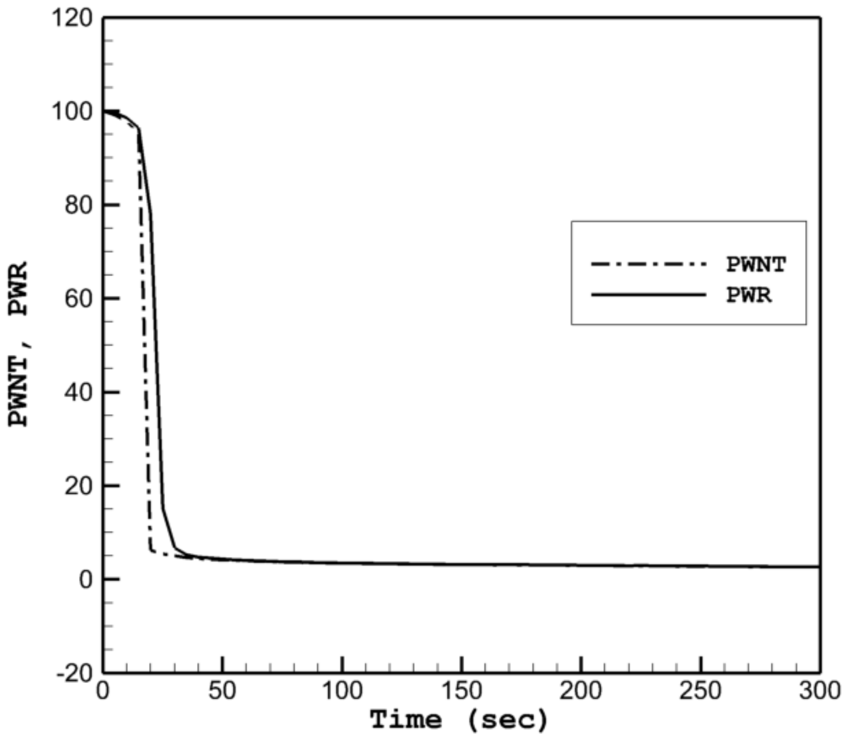

Figure 6. 32 Core Thermal Power and Nuclear Flux (%)

6.3.3.5 Discussion

Thermal hydraulic phenomena of VVER-1000 Nuclear Reactor during loss of coolant accident has been analyzed in this section. Loss of Coolant in Primary system results in continuous heating of the core. In case of LOCA, the safety system requires that it will provide sufficient cooling to the reactor core so as to avoiding melting of the core. As it is seen from the simulation results, the primary system pressure, coolant temperature, fuel cladding temperature doesn’t exceed the safe value which ensures the effectiveness of safety equipments and protection systems of VVER-1000 Reactor.

6.3.4 Loss of Offsite Power (LOOP) Analysis

6.3.4.1 Introduction

Commercial Nuclear Power Plants rely on alternating current power supplied through the electric grid for both routine operation and accident recovery. While emergency generating equipment is always available onsite, a loss of offsite power (LOOP) can have a major negative impact on a plant's ability to achieve and maintain safe shutdown conditions. Risk analyses have shown that LOOP can represent a majority of the overall risk at some plants. Therefore, LOOP events and subsequent restoration of offsite power are important inputs to plant Probabilistic Risk Assessments (PRAs). These inputs must reflect current industry performance so PRAs accurately estimate the risk from LOOP-initiated scenarios.

6.3.4.2 Loss of *Offsite* Power Event Scenerio

LOOP is the simultaneous loss ofelectrical power to all plant safety buses (also referred to as emergency buses, Class 1E buses, and vital buses), requiring all emergency power generators to start and supply power to the safety buses.[22] There are several possible SBO scenarios which could lead to a severe accident, for example:

- Neither the grid nor stand-by AC power sources could successfully be restored within the coping time;
- The alternate AC power source cannot be started and/or connected to the safety bus within the coping time
- The alternate AC power source was able to connect within the coping time, but it fails to operate in long-term (e.g. insufficient diesel fuel, damage due to extreme external events, etc.)

- If the entire grid, standby and alternate AC power sources are unavailable, and mobile means were unable to deploy with the coping time.

The impacts of a LOOP depend upon whether the plant is critical or shut down. If the plant is critical when a LOOP occurs, then a reactor trip generally occurs, challenging various safety systems designed to bring the plant to a safe shutdown. Most ofthe safety systems require ac power, so emergency diesel generators (or other emergency ac power sources) must start and run to supply this power until offsite power is restored to the safety buses. If the emergency ac power sources fail, the plant is still designed to shut down safely via portions of safety systems that can function for a limited period of time without ac power (e.g., turbine-driven pumps for coolant injection). Even if the plant is shut down when a LOOP occurs, emergency ac power must be supplied to the residual heat removal systems.

Table 6.4: Loss of Offsite Power Event Scenario

Time (s)	Event
0	PZR Spray Valve #1 Position Change: 0%
0	PZR Proportional Heater Capacity Change: 0%
0.5	Malfunction # 6 Fraction = 00.0 %
6.5	AFW Turbine Valve #1 Position Change: 100%
6.5	TDAFW Pump Position Change: 100%
6.5	Condensate Pump #1 Position Change: 0%
6.5	PZR Proportional Heater Capacity Change: 0%
6.5	PZR Proportional Heater Capacity Change: 0%
6.5	PZR Backup Heater Capacity Change: 100%
6.5	PZR Backup Heater Capacity Change: 0%
6.5	PZR Proportional Heater Capacity Change: 0%
7	RCP-A trip
7	RCP-B trip
7	RCP #1 Capacity Change: 0%
7	RCP #2 Capacity Change: 0%
7	All MFW Pumps trip
7	Feed Pump #1 Position Change: 0%
7	Feed Pump #2 Position Change: 0%
7.5	TCV Valve #1 Position Change: 0%

7.5	Turbine trip
9	Scram Low RC Flow 87.0 %
66	D/G A Starts 60.0 Sec Delay
186	Safety Relief Valve #1 Position Change: 100%
186	Safety Relief Valve #2 Position Change: 100%
202	Safety Relief Valve #1 Position Change: 0%
202	Safety Relief Valve #2 Position Change: 0%
233	Safety Relief Valve #1 Position Change: 100%
233	Safety Relief Valve #2 Position Change: 100%
249	Safety Relief Valve #1 Position Change: 0%
249	Safety Relief Valve #2 Position Change: 0%
281	Safety Relief Valve #1 Position Change: 100%
281	Safety Relief Valve #2 Position Change: 100%
296.5	Safety Relief Valve #1 Position Change: 0%
296.5	Safety Relief Valve #2 Position Change: 0%

6.3.4.3 Initial and Boundary Conditions

Steady state condition of the plant before the accident is as follows:

- Reactor Power: 3000 MW (100%)
- RC Pressure: 155 bar
- Core Average Temperature : 306.9 °C
- SG Pressure: 70 bar
- Time in Life: BOC (Beginning of Cycle)
- SG Water Level: 2.45 m
- Pressurizer Level: 6.96 m
- Max. Fuel Temperature: 788.9°C
- No interaction of operators during the accident.

6.3.4.4 Analysis of *LOOP* Simulation Result

At the event of Loss of Offsite Power, the Reactor Coolant System (RCS) pressure starts to increase from its nomial value of 15.5 Mpa until “Reactor Coolant Trip” occurs at 7 seconds as shown in Figure 6.33. This RC Trip tends to increase the coolant pressure to avoid boiling in case of pressure loss. The pressure in Steam Generator initially decreases before "Turbine Trip" occurs at 7.5 seconds. After at about 186 seconds, there is a fluctuation of SG pressure due to successive opening and closing of Safety Relief Valve as shown in the figure.

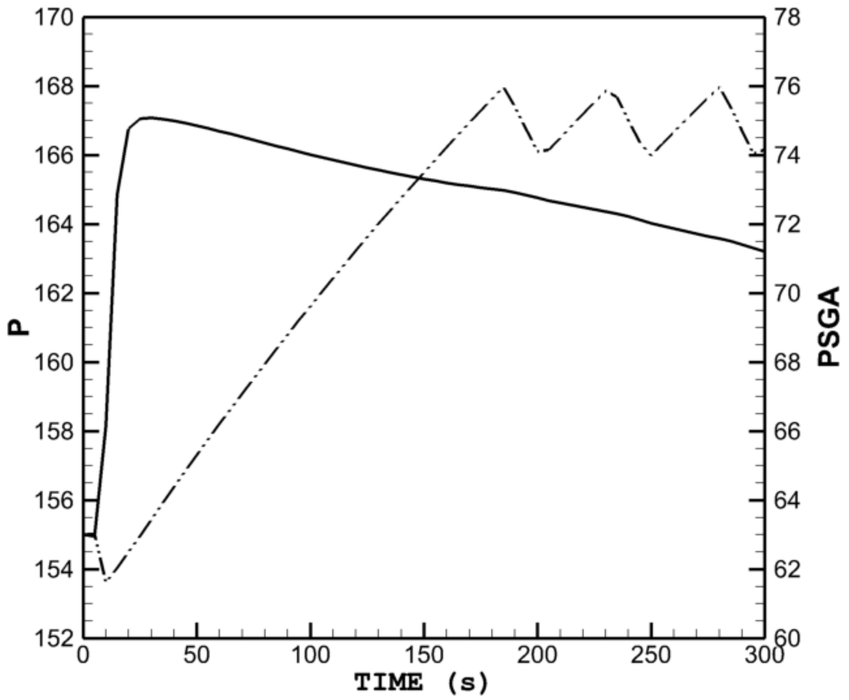

Figure 6. 33 Pressure in Reactor Coolant, Primary and Secondary System (bar)

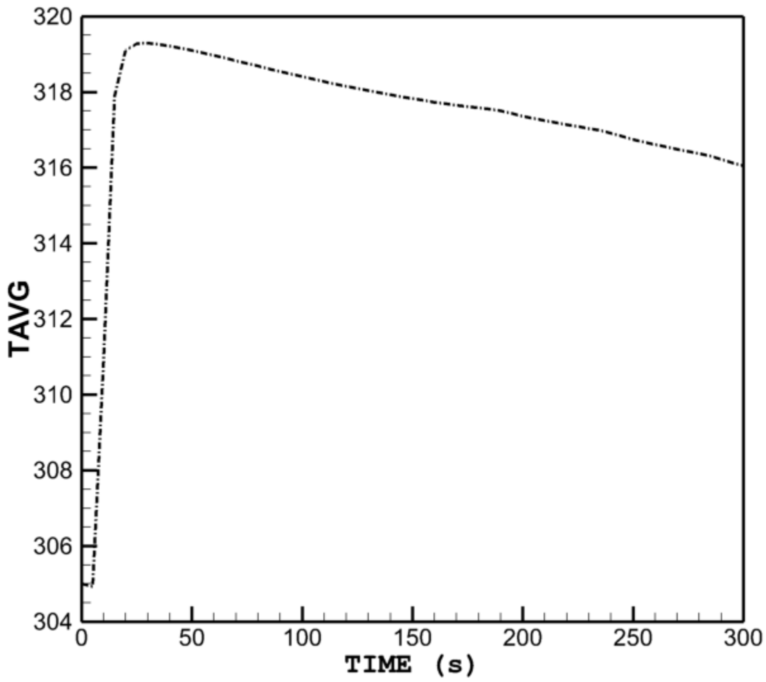

Figure 6. 34 Average Temperature (°C) in Reactor Core

Figure 6.35 shows the water level in the Pressurizer and both Steam Generator Wide Range Level.Water level in Pressurizer increases rapidly from its initial value of 6.96 m to approximately 9.3 m then decreases gradually. On the other hand, both SG Wide Range Level decreases (from 9.5m to 2.1m) instantaneously following LOOP event. Both SG level starts to increase after "Turbine Trip" with continuous fluctuation due to opening and closing of Safety Relief Valve.

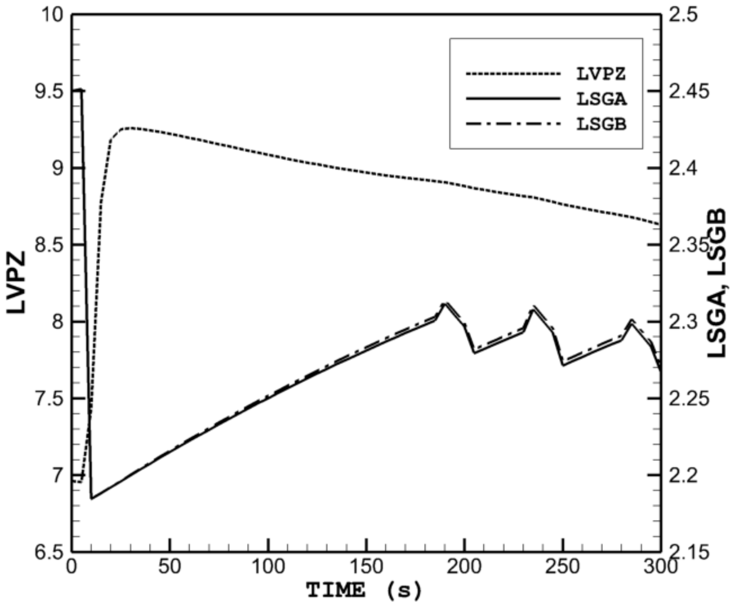

Figure 6. 35 Water Level in Pressurizer and Steam Generator Wide Range Level (m)

The temperature in "Hot" and "Cold" leg of both SG loop 'A' and 'B' is shown in Figure 6.36 and 6.37 respectively which indicates the decrease of temperature at both leg although the temperature in 'Cold' increases at the initiation of the accident. ECCS, comprising of High Pressure Injection, Low Pressure Injection is provided after the pressure drops below a certain value. In this particular case, at about 100 seconds when the coolant pressure drops to approximately 80 bar (8 MPa) the ECCS is activated to prevent the core from being damaged due to excessive heat generation. This is due to the fact that following loss of coolant pressure, the primary coolant may get boiled at lower temperature and creates water vapor around the reactor core which has lower heat transfer capability in comparison to water. As a result, the heat produced within the reactor core can’t be transferred to coolant resulting in the heating up the core.

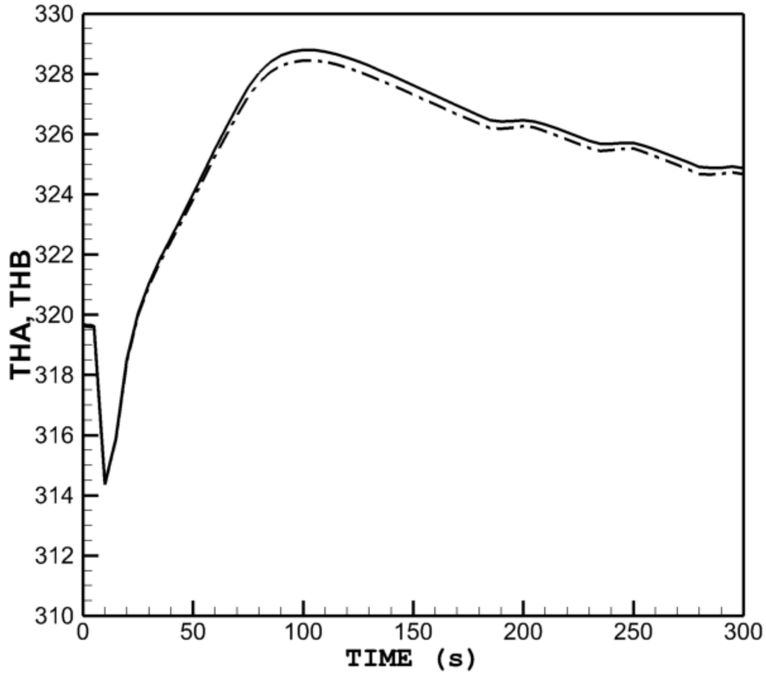

***Figure 6. 36 Temperature in “Hot” Leg of SG ‘A’** and 'B'*

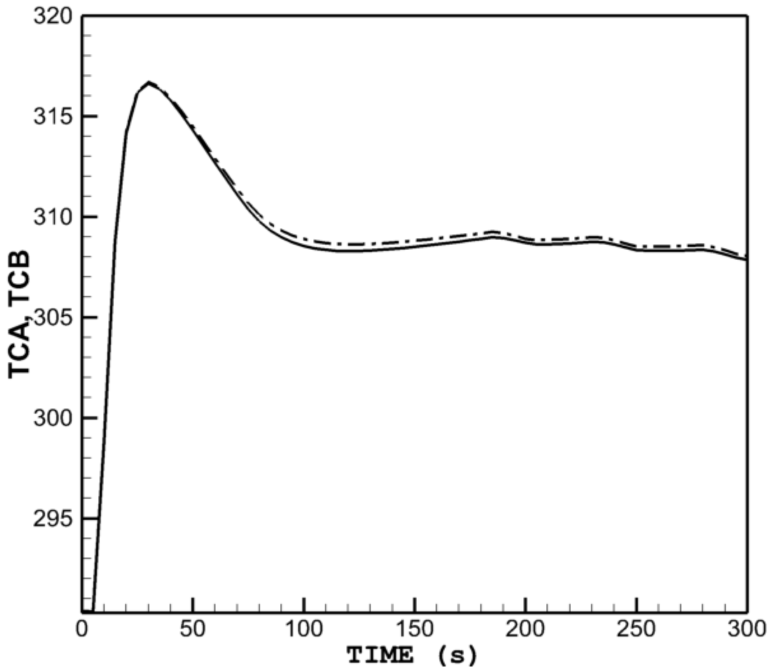

***Figure 6. 37 Temperature in “Cold” Leg of SG ‘**B’*

The fuel surface and cladding temperature are shown in Figure 6.38. After reactor shut down, reactor power stays in nominal level with in 20 s from the beginning then in a short time it will decrease from nominal value to decay heat (Figure 6.39). Since, the fuel temperature is proportional with heat generation in fuel surface unit, there for it will reduce with decrease of reactor power.

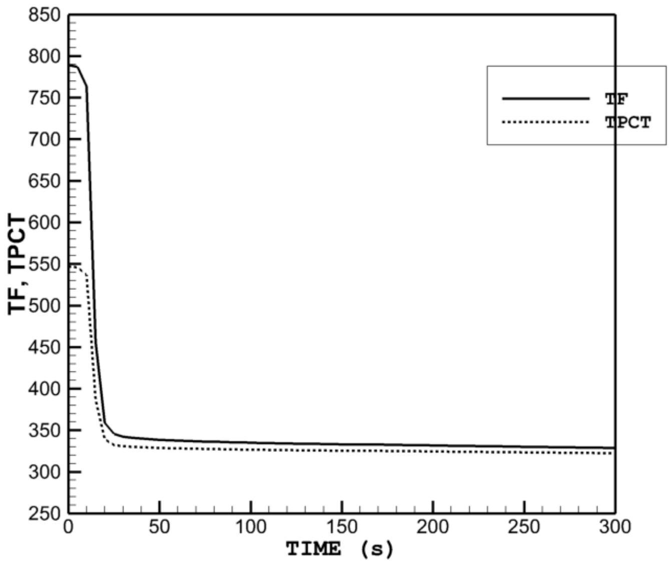

Figure 6. 38 Fuel and Cladding Temperature (°C)

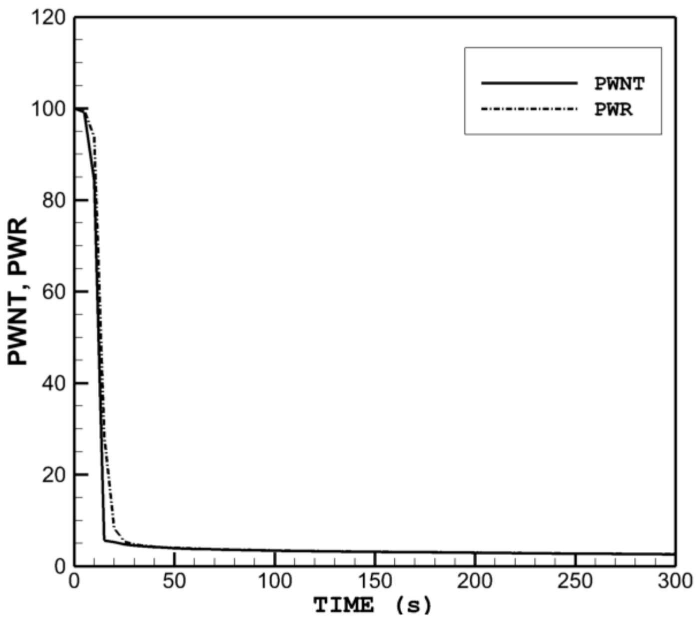

Figure 6. 39 Core Thermal Power and Nuclear Flux (%)

6.3.4.5 Discussion

Loss of Offsite Power may result in Station Black Out (SBO) which poses a great risk for the safe operation of a Nuclear Power Plant. So, every NPP design requires that it should have certain safety features that would be activated in response to the SBO event. The "SBO Coping Time" determines whether the effective countermeasures can be implemented to prevent the core damage. For some NPP design the SBO coping time is very short (e.g. less than one hour), which makes it difficult to implement effective measures to either restore the power supply or ensure the heat removal function. Besides that, plant SBO can challenge performance of the systems and components; e.g. integrity of reactor coolant pump seals, loss of equipment due to loss of ventilation, etc.

Conclusion

The Thermal-Hydraulic Analysis of a VVER-1000 Pressurized Water Reactor during different Design Basis Accidents have been simulated and analyzed in this work. The simulation result shows that:

- The pressure in the reactor coolant and main steam systems is maintained below a prescribed value (typically 110% of the design pressure).This criterion ensures that the structural integrity of the reactor coolant boundary is maintained.
- The maximum fuel cladding temperature during those accidents is found to be 788.9°C which is within the safety limit (less than 1200°C) as prescribed in Acceptance Criteria.
- Emergency Core Cooling System (ECCS) along with other passive safety features prevents the reactor core from melting by providing sufficient cooling hence reducing the possibility of radioactivity release into the atmosphere.

So, it is reasonable to come into conclusion that, VVER's "state-of-the-art" safety systems plays an important role in mitigating the consequences of Nuclear transients to a greater extent and checking radioactivity realease into the environment.

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