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TRANSIENT ANALYSIS OF VVER–1200 NUCLEAR POWER REACTOR IN THE EVENT OF AC POWER FAILURE

***M.G Zakir, A S M Nasim, A Islam, M A Hossain Md.Rosaidul Mawla & M A R SARKAR**
Department of Nuclear Science & Engineering, Military Institute of Science and Technology, Bangladesh
paban@gmail.com, nasimbappee@gmail.com, altab76@gmail.com, mawla71@yahoo.com drmarsarkar@gmail.com

Abstract- The purpose of this paper is to analyze the transient characteristics of VVER-1200 reactor parameters in the event of AC power failure. VVER reactors have diversified and redundant active and passive safety features. In case of AC power failure, passive systems play a vital role in restoring the reactor in a safe and steady state condition. The result shows that VVER-1200 reactors have safety features to restore the reactor into steady state in the event of an AC power failure

Keywords: VVER-1200, AC power failure, transient analysis, safety, PCTran simulator

1. INTRODUCTION

The function of electric power system in a Nuclear Power Plant (NPP) is to provide safe and reliable electricity for the equipment in normal operation or accident conditions, and to provide emergency power for nuclear safety related systems and equipment to maintain the safety of NPPs [1]. The Fukushima plant was cut off from the national grid which left the power plant dependent on back up diesel generators for AC power .The tsunami flooded most of the electrical equipment and knocked out the back up AC power for cooling the reactors.

This paper presents thermal hydraulic analysis in the reactor core in the event of alternating current (AC) power failure [2].

2. OBJECTIVE

The objectives of this paper are as follows:

i)To study the transient parameters such as neutron flux, thermal power, peak cladding temperature, peak fuel temperature, steam generator pressure, reactor coolant system pressure, flow rate in main coolant circuit and feed water flow in steam generator .

ii)To analyze the transient parameters of the VVER-1200 nuclear reactor .

3. BACKGROUND

PCTRAN is a PC-based nuclear power plant simulator for all types of light water nuclear reactor developed by Micro-Simulation Technology (MST) [3]. This software package can simulate variety of accident and transient condition for nuclear power plants. It is used by government agencies and nuclear power plants all over the world, education institutions and International Atomic Energy Agency (IAEA) [4].The user may select from a set of initial conditions corresponding to various power, flow, and time-of-life conditions of the plant. PCTRAN simulation program incorporates the knowledge of reactor physics, thermal hydraulics, control system, etc. into solution techniques with the assistance of modern computer graphics that enable interactive operation on a PC. Since 1998, the source code of PCTRAN has been converted into Microsoft Visual Basic 6.0. Data input/output are in MS Office's Access database format. Reports and data can be transferred conveniently through all Windows-based software products over the entire exercise network. The verification of the simulator is conducted against Final Safety Analysis Reports and validated with respect to real plant data [5].

4. INITIAL CONDITION

5. Table 1: Initial condition before the power failure

Rated Thermal Power (MW)	3200
RCS initial pressure (Bar)	162
RCS initial average temperature (°C)	313.55
Total core flow rate (t/hr)	62200
Average fuel temperature at full power (°C)	800

5. METHODOLOGY

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.

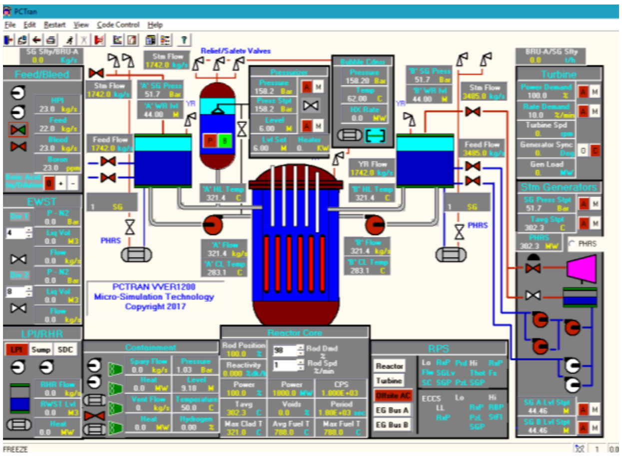


Fig 1: Reactor vessel and steam generator model

Table 2: Power failure chronology

Time(s)	Event
2.5	Turbine trip, 60% load rejection, feed pump trips
10	Pressurizer Safety Relieve Valve (PSRV)-1 opens
11	Pressurizer Safety Relieve Valve (PSRV)-2 opens
22.5	Scram and thermal power decreases to 5%
40	Thermal power changes from 5% to 0%

6. RESULTS & DISCUSSION

6.1 Thermal Power

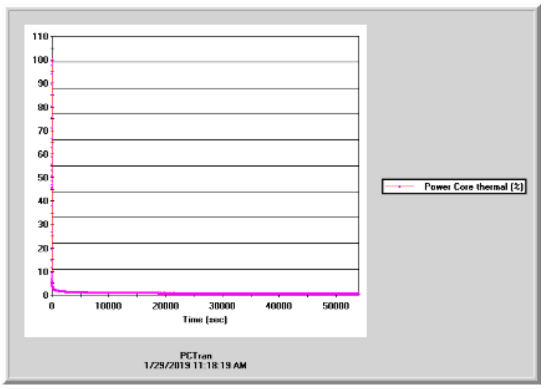


Fig. 2 Thermal power

Thermal power remains 100% until turbine trips which is 0 sec to 2.5 sec. At 2.5 sec, turbine trips and 60% load rejection happened. Thus, thermal power gradually decreases to 45% until the reactor scrams which happens at 22.5 sec after event. After 22.5 sec, reactor scrams, thus thermal power decreases to 5% until the end of 40 sec After 40 sec, thermal power changes very slowly from 5% to 0% until the end of calculation, in this case which is 54000 sec.

6.2Power Nuclear Flux

At the moment of power loss, proportional heater and backup heater lose their capacity to 0%. At 2 sec after power loss,

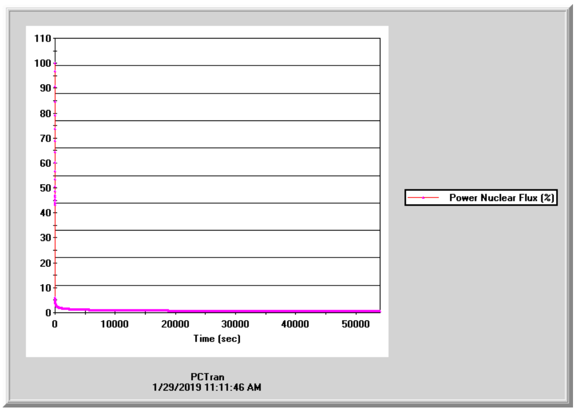


Fig 3: Power nuclear flux

reactor coolant pumps trip, main feed water pump trips, at 2.5 sec turbine trips, due to turbine, 60% load rejection occurs.

Due to these events, control rod position changes, thus nuclear flux changes from 100% to 44% at 2 sec to 19.5 sec after power loss. Due to steam generator pressure increases to 88 bar, control rod moves again and neutron flux goes from 44% to 47% at 19.5 sec to 22.5 sec. At 22.5 sec after power loss, reactor scrams, nuclear flux immediately falls to 5% from where it decreases to almost 0% until the end of calculation.

6.3 Reactor Coolant System Pressure

At 1.5 sec, after power failure, due to proportional and backup heater capacity changes in pressurizer, reactor coolant system pressure starts to increase from 160 bar to 175 bar from 1.5 sec to 11 sec. At 10 sec and 11 sec Pressurizer Safety Relieve Valve, PSRV – 1 and PSRV – 2 both opens and pressure starts to decrease to 130 bar. At 22.5 sec after the event, reactor scrams and both PSRV are close, pressure decrease very slowly to 120 bar. From this position pressure level decreases very slowly with the increase of time till the end of the calculation.

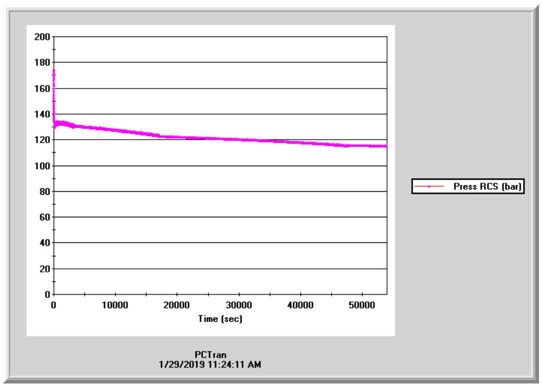
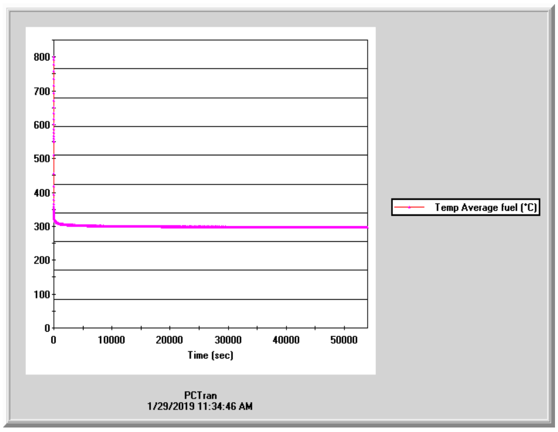


Fig. 4 Reactor coolant system pressure

6.4 Average Fuel Temperature

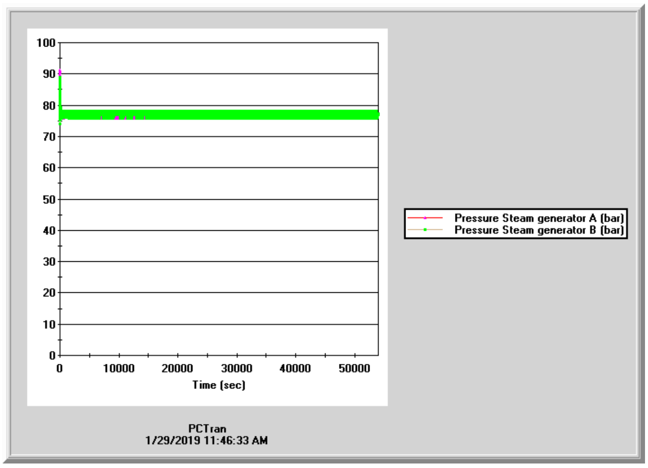


Due to reactor coolant pump trips, main feedwater

Fig. 5 Average fuel temperature

pump trip and turbine trip, control rod position changes and neutron flux and power level reduces. Thus average fuel temperature, gradually decreases from 800 °C to 550 °C at 2 sec to 22.5 sec after the event. After reactor scram at 22.5 sec, neutron flux drastically falls thus fuel temperature falls from 550 °C to 330 °C from 22.5 sec to 40 sec. After 40 sec, temperature reduces very slowly to 300 °C and stay steady until the end of calculation.

6.5 Steam Generator Pressure



Pressure in the steam generator starts to increase just at the time of malfunction because all the Reactor Coolant Pump (RCP) and all the main feed water pump (MFW) trip within 2 second of malfunction. This pressure keeps increasing until reactor scram which is at 22.5 sec after malfunction. After reactor scram SG pressure starts to decrease. At 44 sec SG initiate a signal of low SG level, then at 53.5 sec after malfunction SG Steam Release Valve 3 and 4 both closes and SG pressure increases very slightly and decrease very slowly. Until the end of calculation, SG SRV opens and closes several times and keeps SG Pressure within the design limit.

6.6 Flow in Reactor Coolant Loop

All the Reactor Coolant Pump (RCP) trip just after 2 sec of malfunction. As the RCPs stop, it was supposed to stop the flow immediately. But there is a flywheel installed at every RCP, so even when the RCP trips, flywheel rotates due to inertia of motion. This phenomenon continues for 90 seconds which is quite enough to bring the heat in the RPV in a state from which natural circulation can keep the reactor in steady state through thermo siphoning. After 90 sec flow decreases very slowly because thermo siphoning is a slow process and becomes constant eventually.

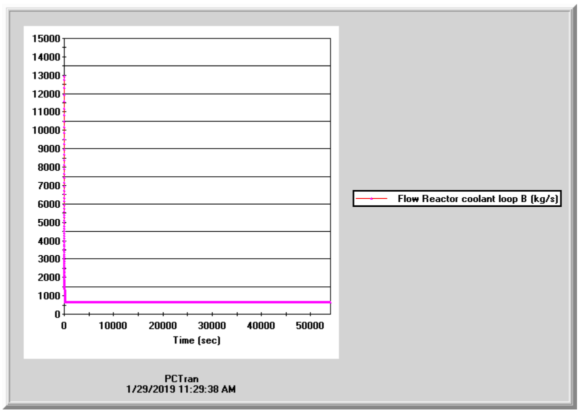


Fig 7. Flow reactor coolant loop

6.7 Steam Generator Feedwater Flow

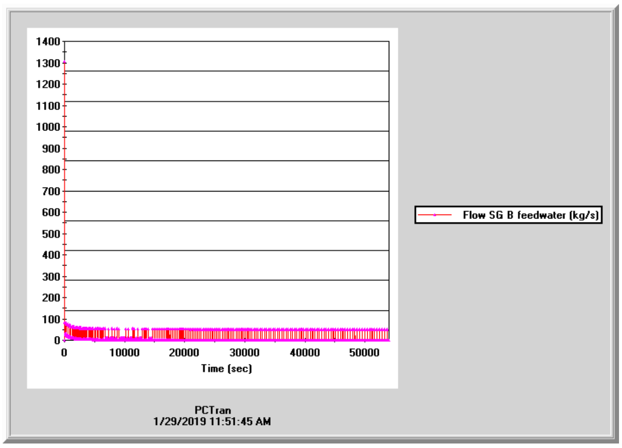


Fig 8. Flow steam generator feed water

Just after 2 sec of malfunction, all the feed water pump trip. Unlike RCP, there is no flywheel installed with feed water pumps, that's why flow of SG feed water stops immediately. It shows flow increases again after 61.5 sec because that's when Turbine Driver Auxiliary Feed Water pump (TDAFW pump) 1 and 2 starts to work. This flow continues for 1 minute and decrease again due to changing of state of SG SRV 1 and 2. Periodical increase and decrease continues until the end of the calculation.
Different types of active and passive safety systems are available to remove the decay heat. One of these passive safety features is the Passive Rasidual Heat Removal System (PRHRS) which removes the residual heat in the event of station blackout.
As result of loss of all AC power supply, the turbine generator stop valve closes leading to increase the pressure in the secondary circuit and then causing the steam dump to atmosphere to open which immediately decrease the pressure in the secondary circuit and release the residual heat to atmosphere. By the end of this process the Passive Rasidual Heat Removal System (PRHRS) reaches its nominal capacity which leads to further decrease in the pressure of secondary circuit until the valve closes. Then the pressure is stabilised due to the PRHRS is working properly in the SG pressure maintaining mode. There will be a sharp decrease in the pressure due to SCRAM, then it will increase due to the decay heat. But the pressure will eventually stabilize due to heat removal through the PRHRS.
The fuel temperature is also important to analyse in order to avoid any fuel failure and to make sure that radioactivity is confined and no release of fission products.When the reactor is scracmed, there is no power produced by the fuel and all the power in the reactor is due to the decay heat, the fuel temperature will eventually decrease since no heat is generated from fission and there is sufficient cooling to cool the cladding, the cladding temperature is stable for a period of time when the pump is still working due to its inertia, however when the pump stops the fuel temperature slightly increase, and then stabilize as a result of PRHRS operation.
The steam generated comes into the passive heat removal system where steam is condensed on the internal surface of the tubes that are cooled on the outside surface by the water stored in the demineralized water tank outside the containment. The water inventory in this tank is sufficient for the long-term heat removal (at least 24 hours) .

7. LIMITATION

Due to the data overflow error of the simulator, the simulation was done for 15 hours only.

8. CONCLUSION

The thermal-hydraulic response of Pressurized Water Reactor during AC power failure accident is presented and analyzed in this paper. A plant specific PCTRAN model (VVER-1200) was used for simulating the system behavior during the transient.

It is found from the study that, the plant safety system 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 [6] by IAEA .

9. ACKNOWLEDGEMENT

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