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# Analysis of LOCA in Cold Leg and Risk Evaluation of Pressurized Water Reactor based NPP

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## ABSTRACT

Transient response of a Pressured Water Reactor (PWR) based nuclear power plant parameters and safety systems during a Loss of Coolant Accident (LOCA) for cold leg has been carried out by using PCTRAN and COMSOL. The whole study is done with a consideration of Nuclear Power Plant (NPP) with two loops of a PWR. The plant model used in this work is a generic two loop with inverted u bend steam generator and dry containment system with a thermal output of 1800 MWt. The simulation result of this entire study is conducted by PCTRAN software, COMSOL Multiphysics software and Fuzzy Expert System (FES) installed in Nuclear Engineering Laboratory at MIST. PCTRAN results show that the Loss of Coolant Accident (LOCA) has caused a rapid drop in coolant pressure while the turbine trip accident has showed a rapid drop in total plant power. Few neutronics parameters have been studied using PCTRAN, thermal behavior during Cold LOCA has been studied using COMSOL, and finally accident prediction has been simulated using FES. The results obtained from the study are compared for getting a precise and accurate result. All the rated parameters regarding this type of reactor are set as default. The main focus of this work is comparing the behavior of some important parameters in case of the selected accident which are basically design basis accidents (DBAs) and the plant is capable to withstand such accident to a certain level varying the type of situations.

Keywords: Pressured Water Reactor, Loss of Coolant Accident, Design basis accidents, Safety systems, Risk evaluation.

## 1. Introduction

Pressurized Water Reactor (PWR) is a type of nuclear reactor which is used to generate electricity. They use light water as the coolant and neutron moderator. Majority of world's nuclear reactors are Pressurized Water Reactors with the others being the Boiling Water Reactors (BWR) and the Super- Critical Water-Cooled Reactors. Bangladesh plans to set up two large nuclear power reactors with the help of Russia to meet the increasing demand of electricity and the model that will be used is VVER-1200 (vodo-vodyanoi enyergeticheskiy reactor or water water energetic reactor) which is a type of PWR. VVER-1200 is the modified form of VVER-1000 which provides optimized fuel efficiency and additional safety features [1]. The new safety system includes not only hydrogen recombine but also a core catcher to contain the molten reactor core and aircraft crash protection. Computational Fluid Dynamics (CFD) code is widely used to predict normal operation and transient state in nuclear power reactors over the last decade. CFD codes contain models for various physical and chemical conditions. Before using these model in nuclear reactor safety applications, they needs to be validated. The required validation is performed by comparing the model with the referenced data [2]. At steady state operation the process parameters do not vary with respect to time but under transient condition or in case of an accident scenario certain parameters deviate with respect to time. These deviations can be analyzed with the help of different simulation software. The risk associated with the malfunction can be predicted by obtaining a better knowledge about the deviation of parameters [3, 4]. The severity of the risk needs to be assessed for ensuring safety of the nuclear power plant [5]. However, the secondary flow was too small to be accurately measured. The lack of full understanding of complex mechanisms connected with the dynamic behavior of a nuclear reactor core still challenges the design and the operation of nuclear reactors. Meanwhile, due to the current enlarged exploitation of nuclear Research Reactors for research and other commercial purposes, the concern of their safety issues has increased [6]. The characterization of the kinetic and thermal-hydraulic parameters that govern the dynamics of the core under off normal operating conditions is one of the most important aspects considered in the present work. There exists different simulation software in order to analyze different transients and accident conditions in a nuclear power plant. Personal Computer Transient Analyzer (PCTRAN) is reactor simulation software specifically designed not only for design-basis accidents, but also security events such as terrorist attack resulting in containment or fuel pool failure. Subsequently, COMSOL Multiphysics software can be used to analyze the steady state, transient and accident conditions in a nuclear reactor [7]. COMSOL Multiphysics is a cross-platform finite element analysis, solver and Multiphysics simulation software [8]. Different parameters such as heat flux, coolant velocity and pressure impact along reactor core channel and sub-channel can be analyzed by COMSOL as well to ensure safe PWR operation [9]. The risk associated with the transient and accident condition needs to be assessed for ensuring safety of the nuclear reactor and the risk frequency can be predicted with the help of Fuzzy Expert System (FES) [3,10]. FES takes inputs and process them based on the pre specified rules to produce the outputs. Both the inputs and outputs are real valued, whereas the internal processing is based on

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fuzzy rules and fuzzy arithmetic formulas. In case of decision making during an accident scenario FES is a very useful tool [11].

## 2. Methodology

One transient is considered in this work. The simulations of this study is conducted by PCTRAN, COMSOL CFD and Fuzzy Expert System software. Due to the loss of coolant accident (LOCA) there is a rapid drop in coolant pressure in the primary circuit of the reactor. Some neutronics parameters have been studied using PCTRAN, thermal behavior during Cold LOCA has been studied using COMSOL, and finally accident prediction has been simulated using FES. These initial conditions contain the basic plant geometry, physics, trip set point and other characteristic data. Fig. 1 shows the flow diagram of the work.

2.1 Modelling of LOCA in cold led in PCTRAN

A single loop has been chosen for operation. LOCA in cold leg and turbine trip has been studied by PCTRAN that is installed in the Modelling and Simulation Laboratory of Nuclear Science and Engineering Department, MIST.

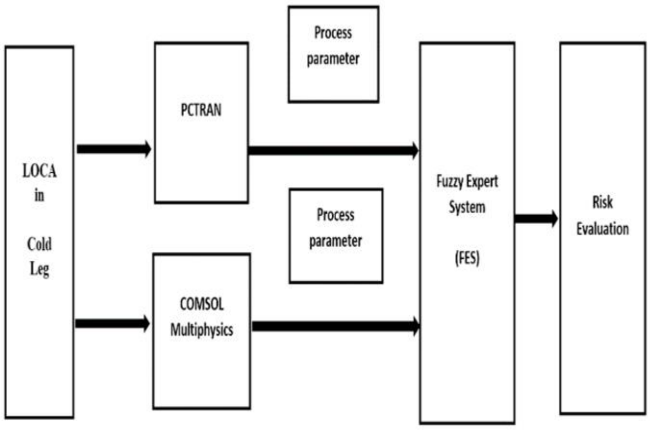



**Fig. 1** Flow chart of the work.

2.2 Modelling of LOCA in cold led in PCTRAN

In this work, Thermal hydraulics behavior of coolant is studied with a Multiphysics software "COMSOL Multiphysics 5.0" by CFD modelling. Geometrical design of a primary coolant pipe is presented in Fig. 2. Length of the pipe is considered as 5 meter and diameter is as 500 mm and thickness is 2 mm. Stainless steel is considered as the material of the object. At 5 m length a circular hole as a break is created as shown in Fig. 2 in which diameter of the break is 250mm. Turbulent Flow (k-ε) and Heat transfer in Fluids are added to the model builder as physics input. K-epsilon (k-ε) turbulence model is used for this work. Meshing is done for the simulation purpose. This work is done with physics-controlled mesh and normal element size. The matrix is a non-symmetric matrix. Study is solved for time independent solver.

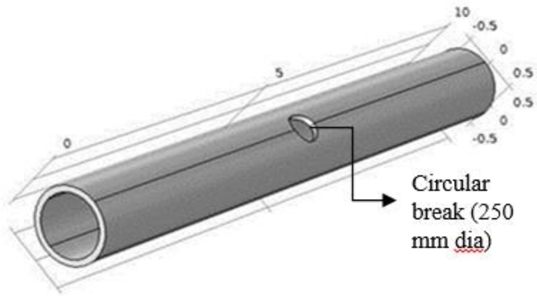



**Fig. 2** Design of primary coolant pipe.

2.3 Modelling of Cold Leg LOCA in Fuzzy Expert System

In order to evaluate the risk level in cold leg LOCA, using the valve obtained from PCTRAN in Fuzzy Expert System, three important parameters are considered as input. These parameters are: a) Neutron flux b) Pressure c) DNBR. FES is used for the prediction of the trip of the reactor. Table 1 shows the membership functions used in modeling for LOCA in cold leg. The different ranges of value are considered for different the membership function. Three Membership functions are used for each case of input parameters. Values are considered from the normal and abnormal (transient) operation in a PWR. For example on neutron flux, the ranges of value of small, medium and large membership functions are 0-60, 40-160 and 140-200, respectively.

**Table 1** Parameters of various membership functions for LOCA in cold

| | Name of the variables | Membership Function and Range |
|---|---|---|
| Input | Nuclear Flux (%) | Small (0-60)<br>Medium (40-160)<br>Large (140-200) |
| | RCS Pressure | Small (12-15.5)<br>Medium (15.5-17.5)<br>Large (17.5-20) |
| Output | DNBR | Small (0.5-1.5)<br>Medium (1-2.5)<br>Large (2-3) |
| | Reactor Trip | Medium (-5)<br>Trip 1 (5-50)<br>Trip 2 (50-70)<br>Trip 3 (70-100) |

For analyzing cold leg LOCA, using the valve obtained from COMSOL CFD analysis in FES, two important parameters are considered as input. These parameters are: a) Velocity b) Pressure. FES is used for the prediction of the trip of the reactor. The different wide ranges of value are considered for different the membership function. Three Membership functions are used for each case of input parameters. These membership functions are small, medium, and large, respectively. Trip is considered as the output of the variables. For analysis, three membership functions are considered. These membership functions are trip1, normal, and trip2, respectively.

## 3. Results and Discussion

3.1 Analysis of Cold led LOCA using PCTRAN

For analyzing LOCA in cold leg by PCTRAN three important parameters have been studied: neutron flux (%), Departure from Nucleate Boiling Ratio (DNBR) and pressure in Reactor Coolant System (RCS). Figure 3 shows the change of neutron flux (%) with respect to time. It decreases as the break occurs and depressurization causes rapid loss of power. Since, water is used as coolant and moderator, a LOCA event may cause a massive loss of moderator as well .Thus, a negative feedback arises and neutron flux reduces almost to zero (Figure 3).

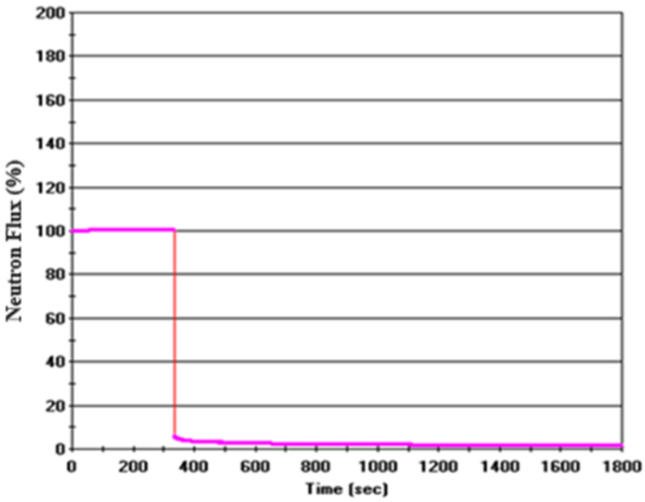



**Fig. 3** Change of neutron flux (%).

Figure 4 represents the changes in departure from nucleate boiling ratio. It increases rapidly after the break close to 500 sec. The reason of rapid increase is the massive loss of coolant in RCS.DNBR is very important in case of reactor safety.

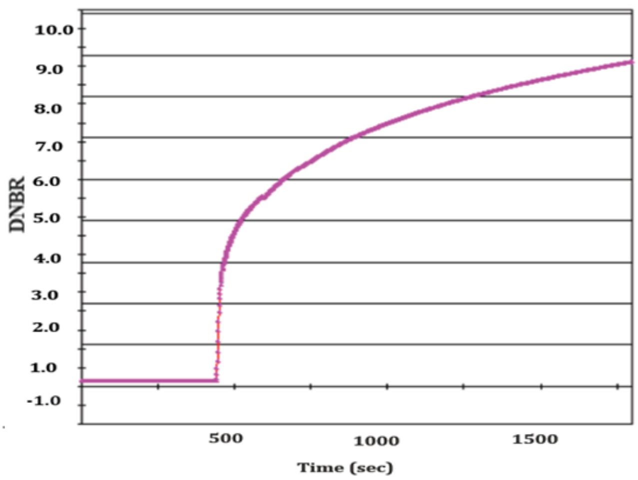



**Fig. 4** DNBR change.

Figure 5 represents the change of pressure in RCS with respect to time. After the initiation of Cold Leg LOCA pressure decreases slowly up to 300 seconds because of initial depressurization from LOCA. Slow depressurization occurs since the reactor was on sub cooled condition and Main Feed Pump was still in operation and reactor was not scrammed yet. Then a faster decrease of pressure is observed from 300 to 400 seconds since reactor been scrammed and all of a sudden neutron flux falls down to zero. After 700 seconds RCS pressure becomes constant. System pressure restores to the steady state condition after initiation of different safety system.

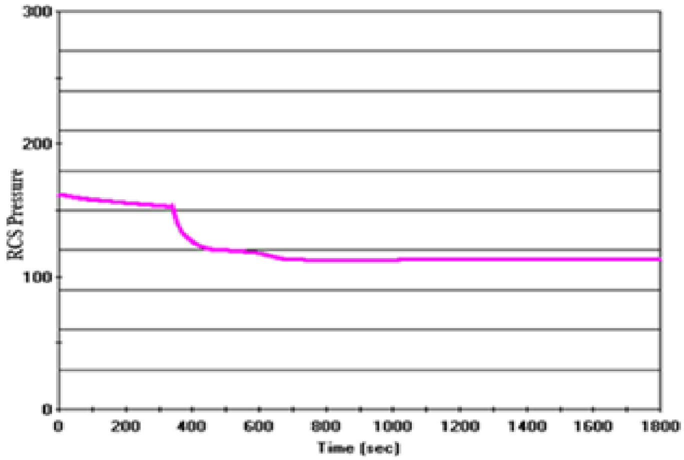



**Fig. 5** RCS Pressure change with time.

3.2 Analysis of LOCA using COMSOL Multiphysics

For analyzing thermal hydraulic behavior during LOCA in COMSOL three important parameters have been studied: pressure, flow of coolant and turbulent kinetic energy. Figure 7 represents flow of coolant in the pipe with break. From the above graphs it is seen that the outlet velocity decreases as the break occurs. The decrease occurs from the starting of the break as the coolant begins to leak from the main coolant pipe.

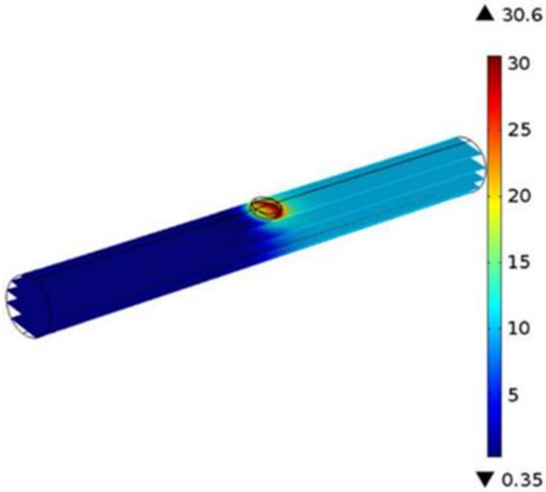
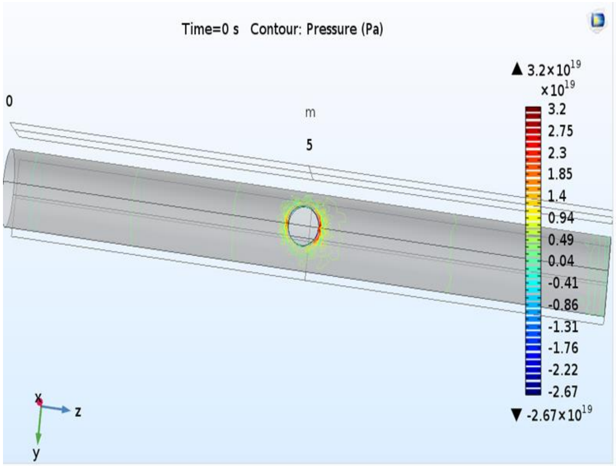
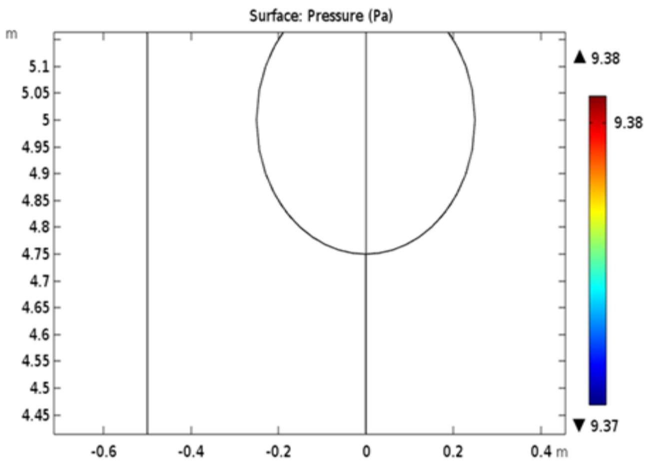



**Fig.6** Flow of Coolant in Pipe with Break.

Figure 7 (a) represents graphical view pressure couture of a pipe with break. From the graph it is seen that after the break pressure is decreased as the coolant is leaving the pipe through the break. There is a fixed pressure in cold leg and it is almost 15MPa.But due to the leakage occurring in pipe the possible consequence is pressure lose. Figure 7 (b) represents 2D view of pressure change. From the graph it can also be said that after the break the kinetic energy of the coolant is decreased as the coolant is leaving the pipe through the break. This initiates the boiling phenomena. High pressure in primary coolant circuit prevents the phage change of coolant. But due to depressurization pressure starts to reduce.




(a)



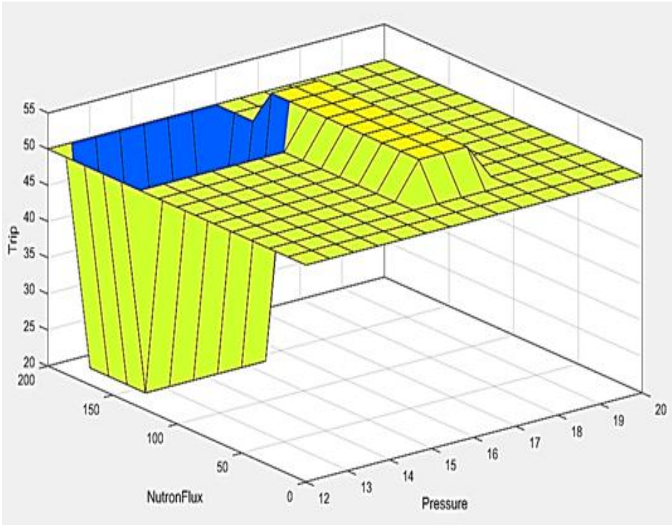

(b)

**Fig.7** Pressure changes (a) graphical view (b) 2D view.

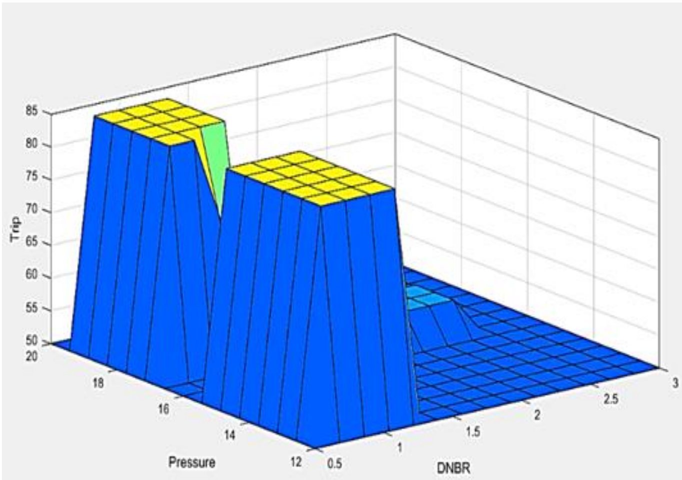
Pressure reduction in the cooling circuit initiates boiling of the core which leads to nuclear meltdown. Hence pressure needs to be maintained.

3.3 Risk Prediction of Cold Leg LOCA in FES

For the input and output parameters, a fuzzy associated memory is created as decision rules based on expert knowledge. After all the rules being set, the output result is shown in rule viewer (Fig. 8). The rule viewer actually displays a simulation depiction of each of the variables through all the rules, an illustration of the combination of the rules, and a demonstration of the output from the defuzzification in numeric values. This value changes along with the changes of input parameters (neutron flux, pressure, DNBR).




(a)

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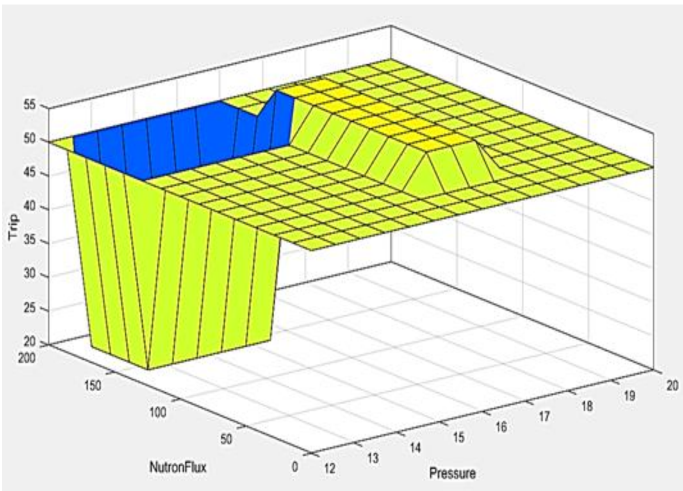


(c)

**Fig.8** Surface viewer: input vs (a) pressure and neutron flux (b) DNBR and neutron flux (c) DNBR and pressure.

For analyzing the model an example is taken. For example, if neutron flux is 100 of, pressure is about 16 MPa and the DNBR is taken as1.78. All three of input parameters are in High range of their respective membership functions. So, according to one of the rules, if output is almost medium and it is about 55%. Therefore, it is normal operation according to the output membership function. If neutron flux is 82.5 of, pressure is about 13.4 MPa and the DNBR is taken as 0.94. So, according to one of the rules, if output is almost high and it is about 85%. Therefore, it is trip 2. Hence the risk assessment model gives the result for risk level indication give correctly.

3.4 Comparative Analysis of Cold Leg of LOCA

Table 2 represents the comparative analysis reactor trip in percentage in the cold leg LOCA. Input 1 represents the parameters obtained from PCTRAN, input 2 indicates the parameters from COMSOL. Output 1 & 2 represents the trip of the reactor in percentage. After simulating the scenario with PCTRAN deviation of almost all key parameters were found after 1800 seconds. Therefore, it can be said that the deviation of each case is comparatively very small. In this comparison only five rules are considered for cold leg LOCA and the comparison is done based on pressure. Neutron flux, DNBR and mass flow are added to identify the trip at the targeted pressure. If more rules are added in fuzzy expert system, this will create a better comparative prediction of the reactor trip. Similarly, if more membership functions are added in the fuzzy expert system, a more accurate trip prediction can be obtained.

**Table 2** Comparative Analysis of the cold leg LOCA

| No. of event | Input 1 (PCTRAN) | | Input 2 (COMSOL) | | | Out-put 1(% trip) | Out-put 2(% trip) | Deviation (%) |
|---|---|---|---|---|---|---|---|---|
| | Pressure MPa | Neutron flux (%) | Pressure MPa | DNBR | Mass flow | | | |
| 1 | 13 | 170 | 13 | 1.75 | 20000 | 20 | 19.5 | 2.5 |
| 2 | 13.5 | 100 | 13.5 | 1.0 | 84000 | 85 | 85 | 0 |
| 3 | 18.5 | 150 | 18.5 | 1.75 | 15000 | 85 | 84 | 1.1 |
| 4 | 18.6 | 100 | 18.6 | 0.60 | 17000 | 90 | 88 | 2.2 |

## 4. Conclusion

Accident analysis is an integral part of nuclear reactor safety. The simulation of these accident scenarios has been performed by using PCTRAN and COMSOL software. In case of COMSOL the deviation of key thermal hydraulic parameters was observed after modelling the scenarios with valid inputs and mathematical formula. The deviation of parameters obtained from both PCTRAN and COMSOL is used as input of Fuzzy Inference System to predict the risk level involved with the malfunction. The entire work can be concluded as follows:

a. Graphical representations are obtained for some key parameters of the reactor transient condition: Cold Leg LOCA.

b. Geometrical modelling of Cold Leg LOCA has been designed and thermal hydraulics parameters have been analyzed by using COMSOL.

c. The severity of the risk associated with the transients are evaluated with the help of Fuzzy Expert System. For different transient's different risk level is estimated. For Cold Leg LOCA the severity of risk is.

The analysis of these scenarios shows that the safety systems of PWR can put the plant in safe state after any of the combination for these three types of events. From the analysis of malfunctions suitable results have been found and risk level has been predicted to ensure safety in a nuclear power plant.

## 5. Acknowledgement

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