A New Methodology for Assessing Passive System Reliability

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Abstract: In this paper, we present a methodology known as APSRA (Assessment of Passive System ReliAbility) for evaluation of reliability of passive systems. The methodology has been applied to the boiling natural circulation system in the Main Heat Transport System and the Passive Containment Isolation System (PCIS) of the Indian AHWR concept. In the APSRA methodology, the passive system reliability is evaluated from the evaluation of the failure probability of the system to carryout the desired function. The methodology first determines the operational characteristics of the system and the failure conditions by assigning a predetermined failure criterion. The failure surface is predicted using a best estimate code considering deviations of the operating parameters from their nominal states, which affect the passive system performance. Since applicability of the best estimate codes to passive systems are neither proven nor understood enough, APSRA relies more on experimental data. APSRA proposes to compare the code predictions with the test data to generate the uncertainties on the failure parameter prediction, which is later considered in the code for accurate prediction of failure surface of the system. Once the failure surface of the system is predicted, the cause of failure is examined through root diagnosis, which occurs mainly due to failure of mechanical components. The failure probabilities of these components are evaluated through a classical PSA treatment using the generic data. Reliability of the system is evaluated from the probability of availability of the components for the success of passive feature (e.g. natural circulation) in the system.

Keywords: Passive System, Natural circulation, Containment Isolation, Reliability.

1. INTRODUCTION

Advanced nuclear reactor designs incorporate several passive systems in addition to active ones, not only to enhance the operational safety of the reactors but also to eliminate the possibility of hypothetical severe accidents. Unlike the active systems, the passive system does not need external input such as energy to operate. Passive systems are simpler in design besides avoiding human intervention in their operation, which increases their reliability as compared to the active ones. However, their performance is always correlated with the system geometry and the operating parameters. Normally, the driving head of passive systems is small, which can be easily influenced even with a small change in operating condition. This is particularly true for the passive systems classified as “Type B” by IAEA [1], i.e. those with moving working fluid; for example a natural circulation system. Such systems rely on natural forces arising due to gravity or buoyancy. The driving force is created by the buoyancy action due to change in density of fluid across the heated/cooled sections. For steady state operation, the buoyancy force is balanced by the resistive frictional force in the system. Since the driving force is due to buoyancy, its magnitude can be easily altered due to any disturbance either in operating parameters or geometry. Because of this, there has been growing concern amongst the nuclear engineers about their reliability not only during transients and accidents but also at normal operation.

Due to the low driving force of passive systems, sometimes the flow is not fully developed and can be multi-dimensional in nature. Besides, there can be existence of thermal stratification particularly in large diameter vessels wherein heat addition or rejection takes place. In such systems, the high density of fluid may settle at the bottom of the vessel and the low density fluid sits at the top allowing kettle

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type boiling when heat addition takes place. Besides, the heat transfer and pressure loss laws for natural convection systems may be quite different from that of forced convection systems. In the absence of plant data or sufficient experimental data from simulated facilities, the designers have to depend on existing 'best estimate codes' such as RELAP5 or TRACE or CATHARE, etc. for analyzing the performance of these systems. However, it is difficult to model accurately the characteristics of these passive systems using the above codes. As a result, there could be large scale uncertainties in simulation of several phenomena of these systems, particularly

- low flow natural circulation;
- natural circulation flow instabilities;
- critical heat flux under oscillatory condition;
- condensation in presence of non-condensables;
- thermal stratification in large pools, etc.

These uncertainties can significantly influence the prediction of natural circulation characteristics and hence assessment of reliability of such passive systems with natural circulation as mode of heat removal [2]. In view of the above, assessment of reliability of passive safety systems is a crucial issue to be resolved for their extensive use in future nuclear power plants. Several physical parameters affect the performance of a passive safety system, and their values at the time of operation are a priori unknown. The functions of many passive systems are based on thermal-hydraulic principles, which have been until recently considered as not subject to any kind of failure. Hence, large and consistent efforts are required to quantify the reliability of such systems.

In late 1990s, a methodology known as REPAS has been developed cooperatively by ENEA [3], the University of Pisa, the Polytechnic of Milan and the University of Rome. This methodology is based on the evaluation of a failure probability of a system to carry out the desired function from the epistemic uncertainties of those physical and geometric parameters which can cause a failure of the system. The REPAS method recognizes the model uncertainties of the codes. The uncertainties in code predictions are evaluated by calculations of sensitivities to input parameters and by code-to-code comparisons. The methodology has been applied to an experimental natural circulation test loop by Jafari et al. [4]. Zio et al. [5] applied this methodology for reliability evaluation of an Isolation Condenser System. However, it was later identified that to assess the impact of uncertainties on the predicted performance of the passive system, a large number of calculations with best estimate codes were needed. If all the sequences where the passive system involved are considered, the number of calculations could be prohibitive. In view of this, another methodology known as Reliability Methods for Passive Safety Functions (RMPS) was developed within the 5th framework programme of the EU (Marques et al. [6]). This method considered the identification and quantification of uncertainties of variables and their propagation in thermal hydraulic models, and assessment of thermal hydraulic passive system reliability. Similar approach is followed by Pagani et al. [7] to evaluate failure probability of the gas cooled fast reactor (GFR) natural circulation system. However, they used simpler conservative codes to evaluate the failure of a system. In addition to this, the above methodologies are yet to be applied to real systems of innovative reactors and the true reliability number for each of the passive system needs to be worked out. On the otherhand, preliminary calculations at MIT have suggested that the reliability of passive natural circulation systems can prove to be lower as compared to an active system.

The RMPS approach adopts a probability density function (pdf) to treat variations of the critical parameters considered in the predictions of codes. To apply the methodology, one needs to have the pdf values of these parameters. However, it is difficult to assign accurate pdf treatment of these parameters, which ultimately define the functional failure. Moreover, these parameters are not really independent ones to have deviation of their own. Rather deviations of them from their nominal conditions occur due to failure/malfunctioning of other components. Hence, assigning arbitrary pdf for their deviations appears illogical.

In this paper, we present a different methodology known as APSRA (Assessment of Passive System ReliAbility) for evaluation of reliability of passive systems. In this approach, the failure surface is generated by considering the deviation of all those critical parameters, which influence the system performance. Then, the causes of deviation of these parameters are found through root diagnosis. It is attributed that the deviation of such physical parameters occurs only due to a failure of mechanical components such as valves, control systems, etc. Then, the probability of failure of a system is evaluated from the failure probability of these mechanical components through classical PSA treatment. Moreover, to reduce the uncertainty in code predictions, BARC will use in-house experimental data from integral facilities as well as separate effect tests. The methodology has been applied to the natural circulation system and the Passive Containment Isolation System (PCIS) of the Indian AHWR concept as examples.

2. THE APSRA METHODOLOGY

In the APSRA methodology, the passive system reliability is evaluated from the evaluation of the failure probability of the system to carryout the desired function. In principle, in a natural circulation system, the operational mechanism of buoyancy driven pump should never fail as long as there is a heat source and sink with an elevation difference between them. However, even though the mechanism does not fail, it may not be able to drive the required flow rate whenever called in, if there is any fluctuation or deviation in the operating parameters even though the system geometry remains in tact. In the case of an AC driven pump, the head vs. flow characteristics is not so much susceptible to a slight change or fluctuation in operating parameter to cause the failure of the system unless there is any mechanical failure of the pump itself. Hence, its performance characteristics are well known and can be simulated accurately while assessing the overall safety of the plant. On the other hand, the characteristics of buoyancy driven pump can not be accurately predicted under all operational conditions or transients due to the inherent complex phenomena associated with natural convection systems as discussed before. Since applicability of the best estimate codes to passive systems are neither proven nor understood enough, hence, APSRA relies more on experimental data for various aspects of natural circulation such as steady state natural circulation, flow instabilities, CHF under oscillatory condition, etc. APSRA compares the code predictions with the test data to generate the uncertainties on the failure parameter prediction, which is later considered in the code for prediction of failure conditions of the system. A detailed discussion of the APSRA methodology is given in the following section.

2.1 The Methodology

Fig. 1 shows the structure of the methodology for the calculation of reliability of passive system. The methodology has been described in details by Nayak et al. [8].

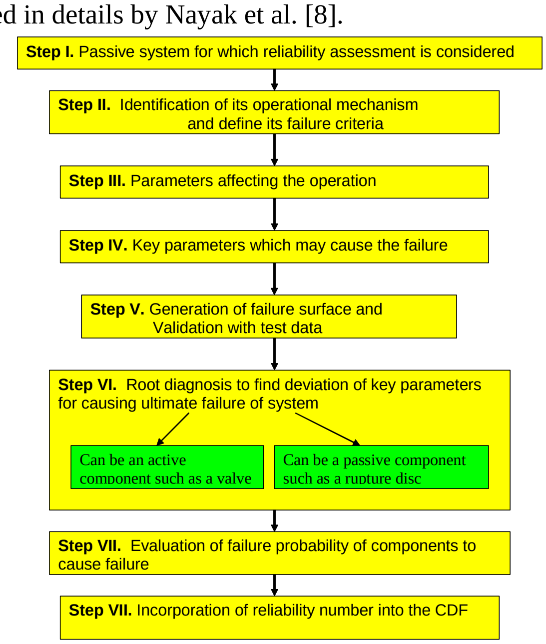


Figure 1: The APSRA methodology

3. APPLICATION OF APSRA METHODOLOGY

3.1. EXAMPLE-1: To the Boiling Two-Phase Natural Circulation System of AHWR

3.1.1 Natural Circulation path in the MHT of the AHWR

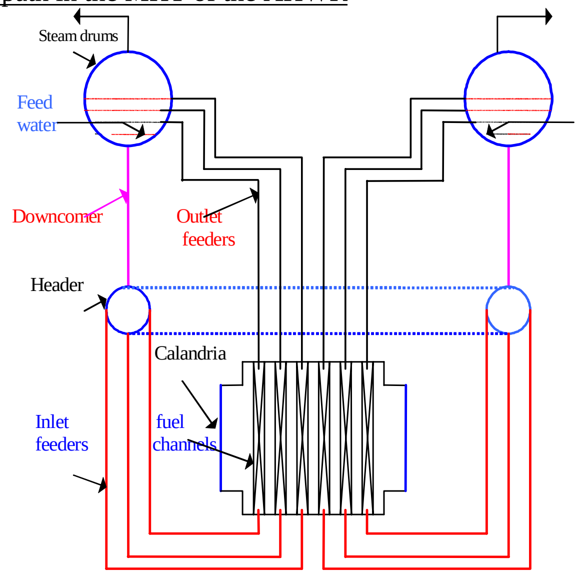


Figure 2: Schematic of MHT System of AHWR

Fig. 2 illustrates the schematic of the Main Heat Transport (MHT) system. The geometric details of the reactor can be seen in reference [9]. The MHT system consists of a common reactor inlet header from which 452 inlet feeders branch out to an equal number of fuel channels in the core. The outlets from the fuel channels are connected to tail pipes, 113 of which are connected to each of the four steam drums. From each steam drum, four downcomer pipes are connected to the common inlet header.

During normal operating conditions, the steam drum pressure is maintained at 7.0 MPa. The collapsed level of water in the steam drum at nominal operating conditions is 1.85 m. The two-phase mixture leaving the core is separated into steam and water in the steam drum. The steam-water separation in AHWR steam drum is achieved naturally by gravity without the use of mechanical separators which affect natural circulation flow rate due to their additional flow losses. At the normal operating condition, 407.6 kg/s of steam, separated in the steam drums, flows into the turbine and an equal mass rate of feed water enters the steam drum at 130^{o}C. For proper mixing of the feed water with the saturated water coming from the riser portion of the steam drum, the feed water is entered in the form of jets through a number of J-tubes located on the feed water header. Hence, the feed water properly mixes with the saturated water in the downcomer portion of the steam drum without causing thermal stratification. The flow rate and temperature of feed water shall change depending on the load conditions. The outlet temperature of the water from the steam drum is about 260.5 ^{o}C at nominal full power conditions assuming complete mixing of feed water with the saturated water in the steam drum.

3.1.2 Stepwise Reliability Calculation Procedure

Step I: Passive system considered – Natural circulation in the MHT System of the AHWR.

Step II:
Some of the critical parameters which influence the natural circulation flow rate in the MHT of the AHWR are

- System pressure;

- Heat addition rate to the coolant;
- Water level in the steam drum;
- Feed water temperature or core inlet subcooling;
- Presence of non-condensable gases;
- Flow resistances in the system.

Step III:
To understand the natural circulation characteristics of the AHWR, the natural circulation flow rate as a function of different parameters has been predicted using the simple code TINFLO-S [10]. Examples of key parameters affecting the operation are given below:

(a) System pressure
Normally, with rise in pressure from very low pressure condition, the flow rate in the system increases due to reduction in two-phase frictional resistance for the same heat addition rate. However, at high pressures, the flow rate reduces with increase in pressure due to reduction of void fraction or buoyancy force. This implies that the pressure is one of the key parameters which can influence the performance of the system.

(b) Heat generation rate
The heat generation rate in the core has a direct bearing on the buoyancy force and hence the natural circulation flow rate. An increase in the parameter from very low value increases the buoyancy force due to increase in void fraction which results in increase in flow rate. However, at power more than about 40 % value, the rise in buoyancy force is compensated by a corresponding increase in frictional resistance which results in that the flow rate more or less remains constant beyond 40 % of full power. This further implies that the buoyancy pump behaves more or less same as that of a centrifugal pump above 40 % full power. However, if the power rises beyond 100 % full power, the rise in frictional resistance in two-phase region becomes dominant, which may result in reduction in flow rate.
(c) Feed water temperature/core inlet subcooling
The core inlet subcooling is directly dependent on the feed water temperature, which can influence the natural circulation flow rate by changing the density of fluid in cold leg. Besides, a smaller inlet subcooling may induce the CHF in the system.

(d) Non-condensable gases in the MHT system
The sources of non-condensable gases in the coolant can be from the ECCS accumulator, clad failure, radiolytic decomposition of coolant, etc. of course, the non-condensable gases, unless continuously added, should get out of the system through the turbine off-gas system along with the steam. Hence, these gases may not affect natural circulation at normal operational states.

(e) Flow resistances in the system
Once the system geometry has been fixed considering the desired flow rate to be generated at the nominal power of the reactor, deviation due to manufacturing tolerances should not have any affect on the performance of the system. Hence, this parameter need not be considered for reliability analysis of natural circulation systems unless one considers channel flow blockage due to generation and deposition of corrosion products in the system resulting in failure of fuel pins.

(f) Water level in stem drum/downcomer
A reduction in level in steam drum is possible due to malfunction of the level control valve or feed pumps. With reduction in water level in the steam drum or if it further falls to the downcomer, the driving head would continuously fall. This would cause a reduction in natural circulation flow rate.

Identification of natural circulation failure:

For normal operational states, natural circulation failure in AHWR occurs if

- Clad surface temperature rises above 400 ^{0}C (673K), or/and
- CHF occurs with or without flow induced instability

Step IV: Key parameters causing the failure
The key parameters to cause the failure of the system are

- fission heat generation rate;
- level in steam drum;
- pressure in the system;
- feed water temperature/subcooling.

Effect of some of the above parameters on the failure of natural circulation is discussed below.

(a) High fission heat generation rate
Figs. 3 (a) to (c) show an example of the effect of increase in channel power on the natural circulation flow behaviour at the rated pressure condition, but at a subcooling of 5 K. The results have been predicted using the best estimate code RELAP5/MOD 3.2. With rise in power to 180% FP, the clad surface is found to rise above 400 ^{o}C (Fig. 3(b)) and the system is declared to fail. There is no occurrence of CHF or flow instability in the system at such conditions (Figs. 3(a) and (c)).

A different effect is observed at higher subcooling (Figs. 4(a) to (c)). In this case, the subcooling has been increased to 30 K instead of 5 K as shown in Fig. 4. With rise in power above 155% FP, flow instabilities were observed and at 160% FP, the amplitude of oscillations were large enough to reduce the thermal margin thereby causing both CHF and rise in clad surface temperature above 400 ^{o}C.

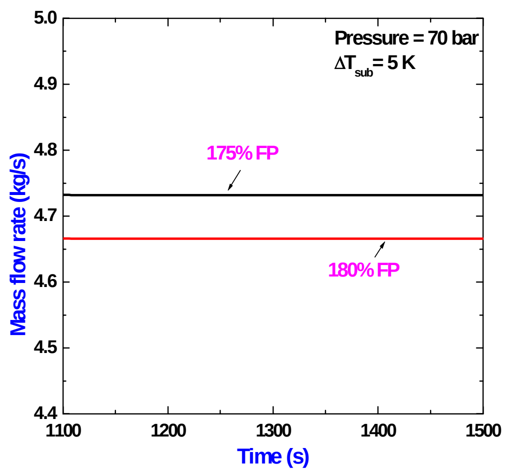


Figure 3(a): Effect of channel power on flow rate under low subcooling

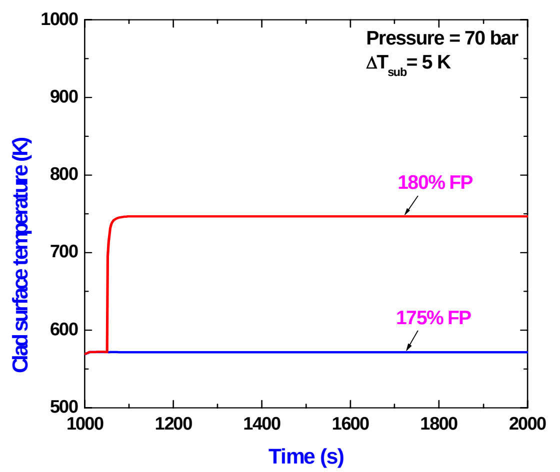


Figure 3(b): Effect of channel power on clad surface temperature under low subcooling

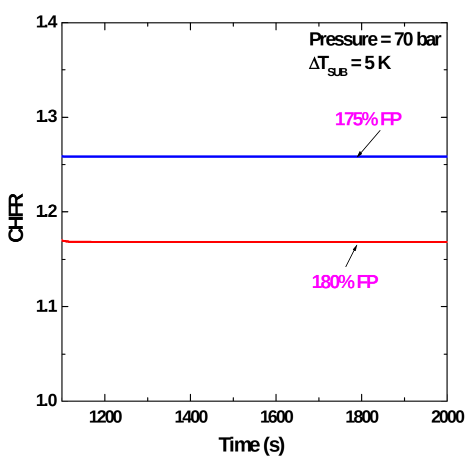


Figure 3(c): Effect of channel power on CHFR under low subcooling

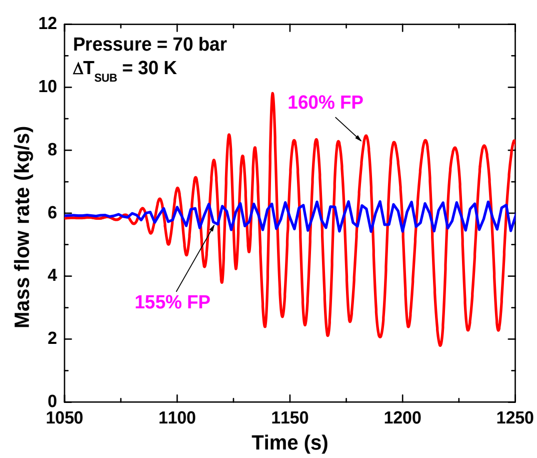
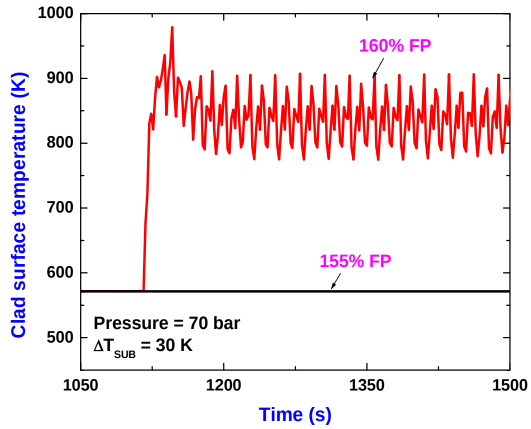


Figure 4(a): Effect of channel power on flow rate under high subcooling



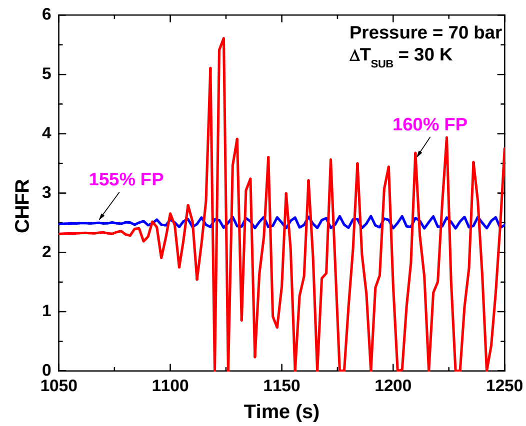


Figure 4(b): Effect of channel power on clad surface temperature under high subcooling

Figure 4(c): Effect of channel power on CHFR under high subcooling

(b) Behaviour due to sudden reduction in system pressure and other operating conditions

At low pressure conditions, the passive system is found to fail due to flow instabilities even occurring at low subcooling (5K) as seen in Figs. 5 (a) to (c). Divergent flow oscillations are found to occur at 160 % FP with rise in clad surface temperature due to occurrence of CHF.

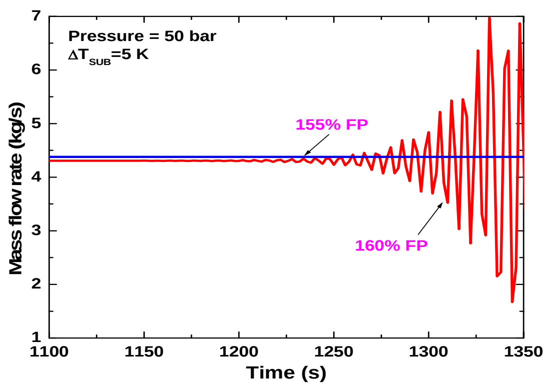


Figure 5(a): Effect of channel power on flow rate under low pressure and low subcooling

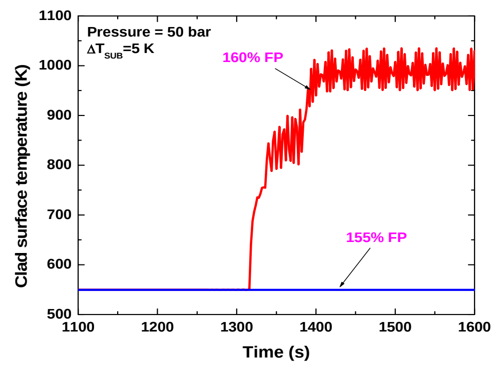


Figure 5(b): Effect of channel power on clad temperature under low pressure and low subcooling

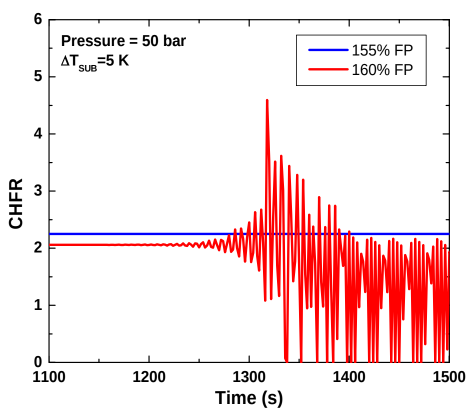


Figure 5(c): Effect of channel power on CHFR under low pressure and low subcooling

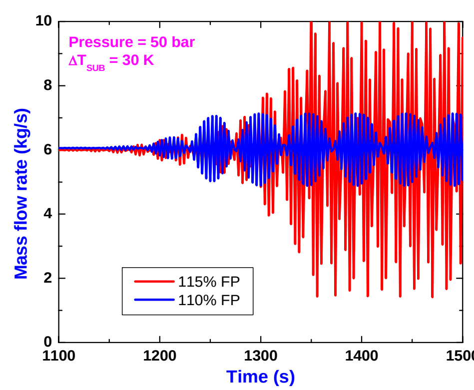


Figure 6(a): Effect of channel power on flow rate under low pressure and high subcooling

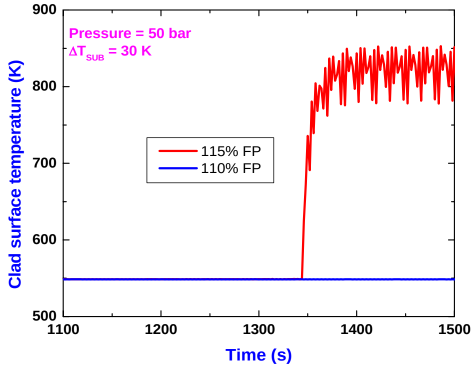


Figure 6(b): Effect of channel power on clad temperature under low pressure and high subcooling

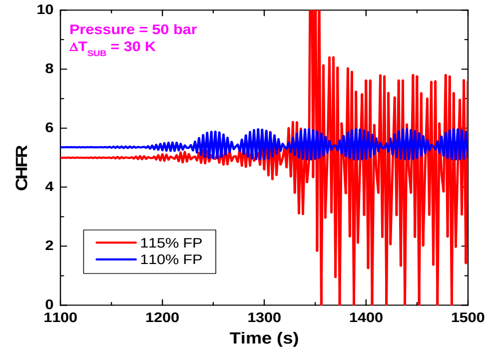


Figure 6(c): Effect of channel power on CHFR under low pressure and high subcooling

Similar behaviour is also observed in Figs. 6 (a) to (c) at high subcooling conditions (30K). However, the power at which the system is found to fail reduces with rise in subcooling.

Step V: The loci of all failure points so generated can be joined to generate the failure surface as shown in Fig. 7.

It may be mentioned here that the effect of water level in the steam drum on natural circulation failure is not shown in Fig. 7. The water level was varied in the steam drum until the low level trip set point, however, natural circulation failure was never observed. However, it may be noted that the applicability of the best estimate codes to evaluate the flow instability under natural circulation is still not well understood and a lot of uncertainty exists. Added to that the behaviour of the CHF under oscillatory condition needs to be studied through experiments in order to validate the best estimate codes and evaluate their uncertainties.

Step VI: After establishing the domain of failure surface, next task is to identify the causes for the deviation of key parameters. The deviation of key parameters are either caused by failure of some active components such as valves, pumps, instruments, control systems, etc., or, due to failure of some passive components such as rupture disc, check valves, passive valves, etc.

Step VII: The failure probability for the system to reach the failure surface has been worked out using the generic data for the failure of active/passive components. A typical example of fault tree is shown in Fig. 8.The results are calculated considering the rise in power, deviation in subcooling and pressure from the nominal conditions. The failure probability of the natural circulation system is found to be ~ 3 x 10^{-9}/yr.

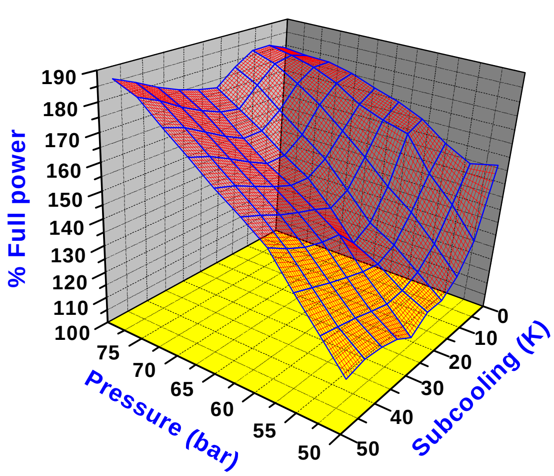


Figure 7: Failure surface for natural circulation

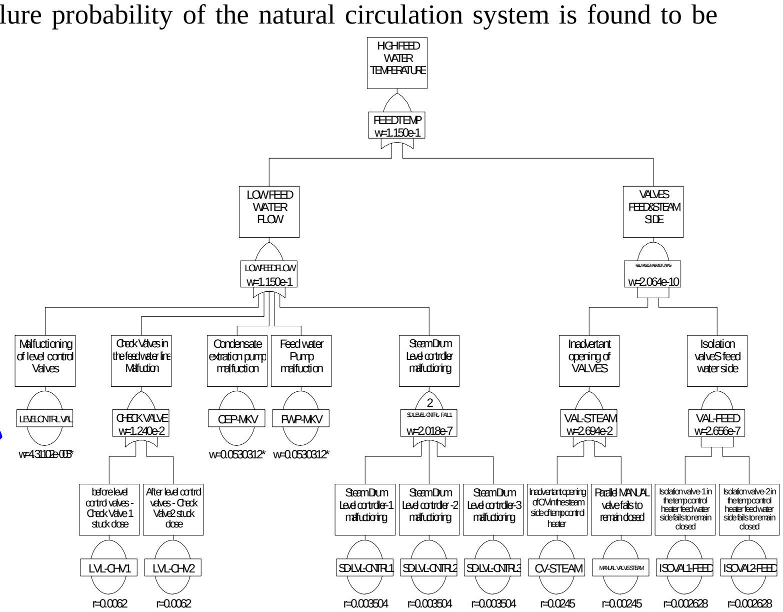


Figure 8: Fault tree for high feed water temperature

3.2. EXAMPLE-2: To the Passive Containment Isolation System of AHWR

Following is the application of the methodology to another passive concept employed in AHWR, to isolate the containment form external atmosphere following a LOCA.

3.2.1 Passive Containment Isolation System of the AHWR

AHWR employs a double containment system that consists of a primary containment enveloped by secondary containment. Primary containment comprise of a high enthalpy zone (V1 enveloping mainly the reactor core and main heat transport system) and a low enthalpy zone (V2 enveloping rest of the systems including suppression pool). Under normal operating conditions, V2 region is in communication to external atmosphere through ventilation system that consists of ventilation duct, blower and filters whereas V1 region remains isolated from the V2 region through suppression pool (connected by partial submerged vent shaft) and blow-out panels. Under the accident conditions like LOCA, involving release of high enthalpy fluid, pressure in the V1 region rises. As the pressure raises enough to overcome the hydrostatic head in the partial submerged vent pipes, vent clearing takes place resulting in release of air-steam mixture into the suppression pool (GDWP). Steam gets condensed in the pool water and non-condensable accumulate in the V2 region resulting in pressure rise with little delay. However, during accidents like large break LOCA, the pressure differential of V1 and V2 rise very fast such that blowout panel opens and the direct communication between V1 and V2 gets established. Schematic of flow path between V1 and V2 is shown in Fig.9. During the events leading to pressurization of V2 region, the containment needs to be isolated from external atmosphere by curtailing the flow from V2 to atmosphere.

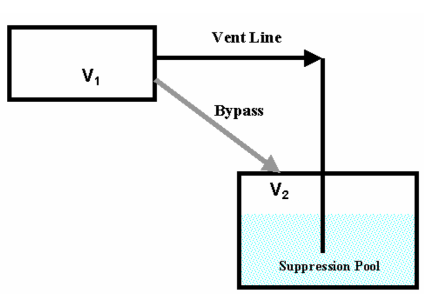


Figure 9: Schematic of flow path between V1 and V2 zone

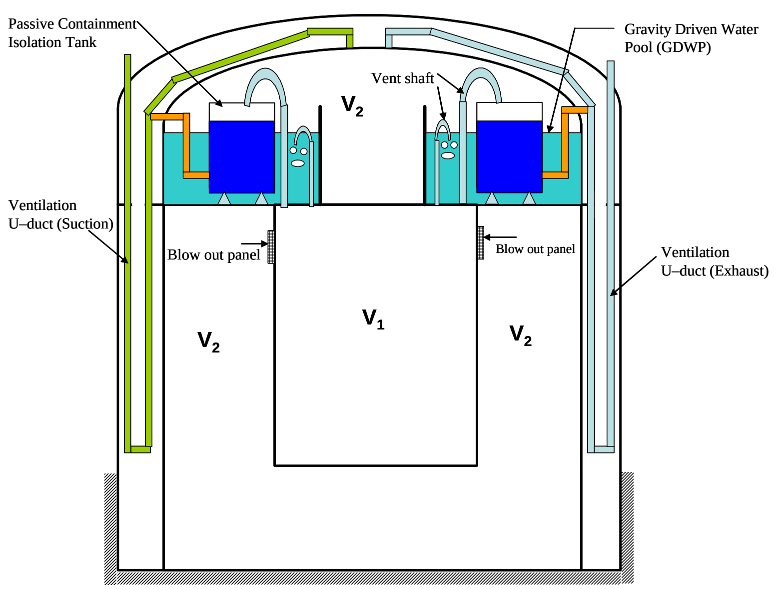


Figure 10: Schematic of Passive Containment Isolation System (PCIS) of AHWR

A passive system is designed to isolate the containment by establishing a liquid seal in the U-shaped ventilation duct. Passive containment isolation system of AHWR, as shown in Fig.10, consists of tanks located in GDWP with an exit pipe connected to ventilation duct which is in communication to V2 region. Vent shafts from V1 region is connected to the top of tank. Thus water in the tank would experience the V1 pressure and in the exit pipe V2 pressure. Under accident conditions the V1 pressure would raise more and faster than V2 and thus water from the tank displaced to spill into the U-shape ventilation duct and hence establishing a liquid seal that isolates the containment from external atmosphere. A separate system each for suction and exhaust duct is considered.

3.2.2 Stepwise Reliability Calculation Procedure

Step I: Passive system considered – Passive Containment Isolation System (PCIS) of the AHWR.
Step II:
Identification of the parameter critical to the performance of the system
- Water inventory in the isolation tank
- Break size (of LOCA)

Step III:
To understand the behavior of PCIS of AHWR, the performance of system is analyzed for above mentioned parameters over a wide range using best estimate code RELAP5/Mod3.2.

(a) Water inventory

Containment isolation is achieved passively by spilling water from a tank (containment isolation tank) into the ventilation duct which is provided with a U-shape leg to establish a liquid seal. Sufficient quantity need to be spilled such that hydrostatic head of the differential liquid column could overcome the pressure differential between V2 and atmosphere. Hence, availability of water inventory (water level) in the tank is a critical parameter.

(b) Break size

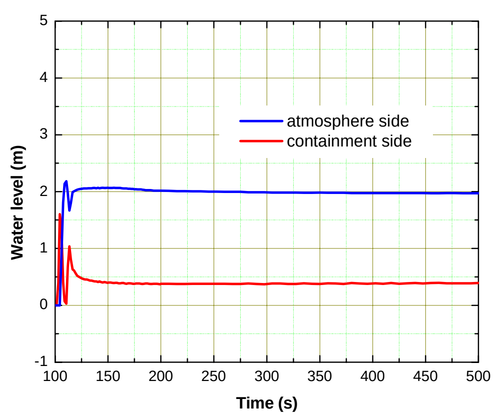
Water to be spilled from the tank to the ventilation duct is driven by the differential pressure in the V1 and V2 zones of the primary containment. This differential pressure is a function of break size in the main heat transport system of the reactor.

Identification of containment isolation failure:

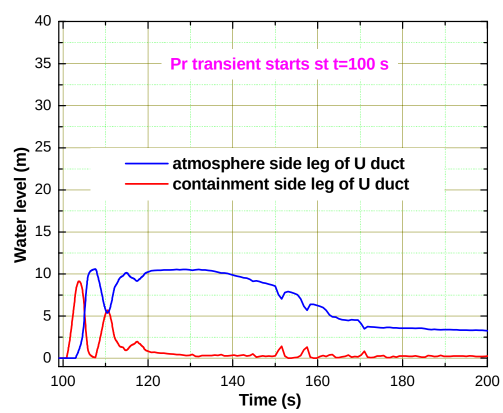
Containment isolation is considered to have failed, if a sustained liquid seal fails to form in the U-duct. This is established by measuring the flow through stack/filters following a LOCA.
Step IV: Key parameters causing the failure are identified. A best estimate analysis is performed to investigate the failure conditions. Following assumption are made to simulate the system behaviour:

- Containment pressure transients are known a priori based on LOCA analysis.
- Containment pressure transient is based on assumption of active isolation system availability. This leads to conservative estimate as the V1-V2 pressure differential would be high if the active isolation of containment fails to occur.
- Only air is considered for pressurizing the tank and exit pipe as during initial part of the LOCA containment atmosphere would be predominantly air.

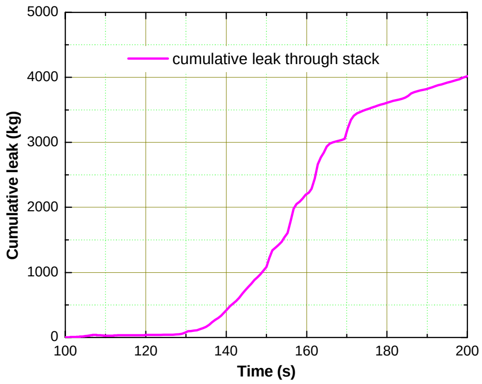
Some of the typical failure conditions are as shown below:



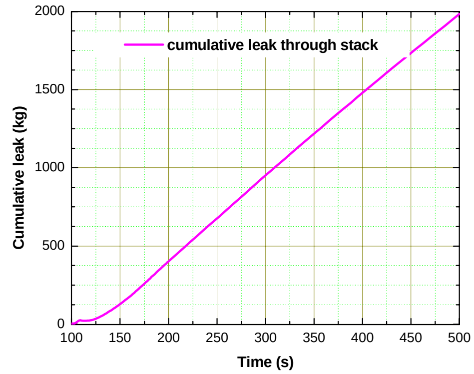
(11a) Variation of water level in the U-duct for large break LOCA



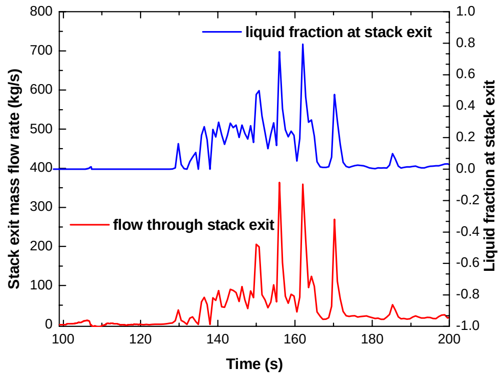
(12a) Variation of water level in the U-duct for small break LOCA



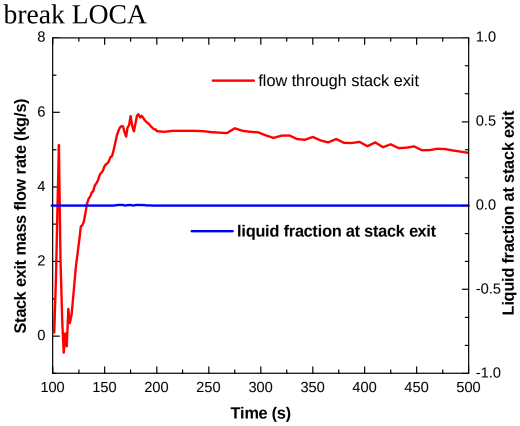
(11b) Cumulative leakage through stack for large break LOCA



(12b) Cumulative leakage through stack for small break LOCA



(11c) Variation of flow and liquid fraction at stack for large break LOCA



(12c) Variation of flow and liquid fraction at stack for small break LOCA

Figure 11: Performance of PCIS for large break LOCA (200% break size)

Figure 12: Performance of PCIS for small break LOCA (10% break size)

A typical performance under large break LOCA (200% break size) at reduced water level of 30% is as shown in Fig. 11a-c. Similarly a typical performance under small break LOCA (10% break size) at reduced water level of 85% is as shown in Fig. 12a-c. It is observed that under degraded conditions corresponding to 30% water inventory for 200% break size and 85% water inventory for 10% break size the liquid seal fails to form and there is continuous discharge of gas from containment to the external atmosphere.
Step V: The failure conditions so obtained are used to establish the failure region (Fig. 13) and to identify the different modes/cause of failure.

Step VI: The causes of deviation of the critical parameters that led to failure are identified in this step. For the PCIS of AHWR, water inventory in the tank is the critical parameter leading to failure. The inventory may deviate from design value due to following plausible reasons:

- Water tank drain valve malfunction
- Water tank make-up system malfunction

It is interesting to note that a sustained liquid seal may fail to establish even if there is enough water in the tank and adequate amount has been spilled following LOCA. Such a failure may result due to malfunction of the ventilation duct drain valve.

Step VII: The failure probability for the system to reach the failure surface has been worked out using the generic data for the failure of active/passive components as shown in Fig. 14. The failure probability of the passive containment isolation system is thus found to be ~ 9.12 x 10^{-6}/yr.

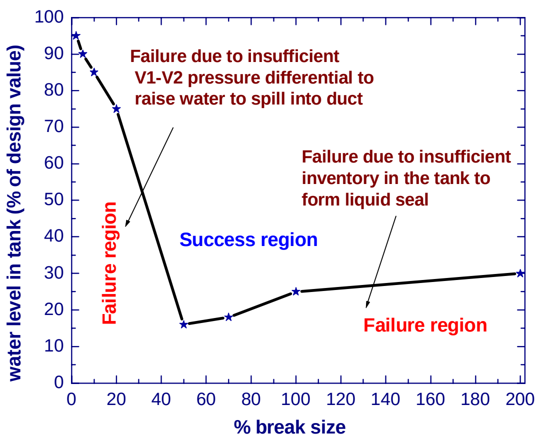


Figure 13: Failure curve for PCIS

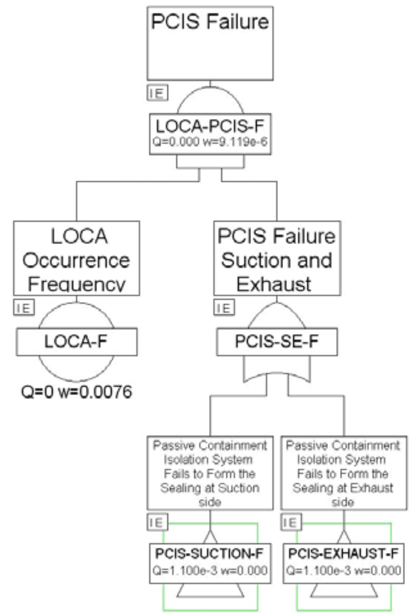


Figure 14: Typical fault tree for PCIS

4. CONCLUSION

Evaluation of passive system reliability is a challenging task. It involves a clear understanding of the operation and failure mechanism of the system which the designer must do before prediction of its reliability. Besides, applicability of the so called ‘best estimate codes’ to the reliability of passive systems are neither proven nor understood enough due to lack of sufficient plant/experimental data. That also creates another problem in assessing the uncertainties of the best estimate codes when applied to passive system safety analysis.

In this paper, we have proposed a methodology known as APSRA to analyze and evaluate the reliability of passive systems. The methodology first determines the operational characteristics of the system and the failure conditions by assigning predetermined failure criteria. The failure surface is predicted using a best estimate code considering deviations of the operating parameters from their nominal states, which affect the natural circulation performance. Once the failure surface of the system is predicted, the cause of failure is examined through root diagnosis, which occurs mainly due to failure of mechanical components. The failure probability of these components is evaluated through a classical PSA treatment using the generic data. At present, the model has been applied to the boiling natural circulation occurring in the Main Heat Transport System (MHTS) and Passive Containment Isolation System (PCIS) of the Indian AHWR concept. The failure probability of natural circulation in the system is found to be ~ 3 x 10^{-9}/year and that for PCIS is ~ 9.12 x 10^{-6}/year. However, to reduce the uncertainty in the failure surface prediction, the code predictions will be compared with the test data for certain conditions in near future. For this purpose, experiments are being carried out in various in-house facilities to evaluate the data relating to failure of natural circulation systems.

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