Large Break LOCA Uncertainty Quantification for Boiling Natural Circulation Reactor Using Latin Hypercube Sampling

A. K. Trivedi [1], A. Srivastava [3], H. G. Lele [3], Ashok Khanna [2] and P. Munshi [1]

[1]Nuclear Engineering and Technology Programme
[2]Department of Chemical Engineering
Indian Institute of Technology, Kanpur 208016, India
[3]Reactor Safety Division, Bhabha Atomic Research Centre,
Mumbai 400085, India
E-mail: akhanna@iitk.ac.in

Abstract

This work deals with uncertainty in Peak Clad Temperature (PCT) for double-ended rupture in the primary coolant circuit of Indian natural circulation reactor. This rupture is identified as a critical break size leading to maximum clad temperature from best estimate code RELAP5. Based on initial sensitivity studies, six important parameters are selected for their significant impact on PCT. Latin Hypercube Sampling (LHS) is used to create 500 sets of six independent variables based on their probability distribution and LOCA calculations have been performed. The 95th percentile value of PCT is 1287 K and it is significantly below the core coolability criteria of 1477 K.

1. Introduction

The Indian Natural Circulation Reactor (NCR) [1] is being designed to achieve large-scale use of thorium for the generation of commercial nuclear power. It is a 920 MWt, vertical, pressure tube type, boiling light water cooled, and heavy water moderated reactor. The fuel cluster of (Th-^{233}U) O_2 and (Th-Pu) O_2 pins are designed to generate maximum energy out of ^{233}U. One of the important passive design features of this reactor is that the heat removal is achieved through natural circulation of primary coolant at all allowed power levels with no primary coolant pumps. A unique feature of this design is a large tank of water on top of the primary containment vessel, called the Gravity Driven Water Pool (GDWP). This reservoir is designed to perform several passive safety functions. Several innovative passive safety systems have been incorporated in the design, for decay heat removal under shut down conditions. The Main Heat Transport System (MHT) [2] of the proposed reactor consists of reactor core, core inlet, core outlet, bottom extensions, inlet feeders, tailpipes, steam drums, downcomer and inlet header as shown in Figure 1. The reactor core consists of 452 pressure tubes each houseing a 54-element single fuel bundle. Emergency Core Cooling System (ECCS) [2] is provided to limit the fuel temperature rise within acceptable limits in the event of Loss of Coolant Accident (LOCA). The ECCS consists of (a) High pressure injection from advanced accumulator (b) Low pressure injection from Gravity Driven Water Pool (GDWP) (c) Long term core cooling by recirculation & cooling of reactor cavity water.

2. Main Heat Transport (MHT) System

The MHT system nodalization is shown in Figure 1 and thoroughly discussed in the reference [2]. It models 452 channels and considers 4 parallel paths in the core. Each parallel path i.e. one-fourth core path comprises of one hot channel and 112 lumped channels. There are 4 steam drums and one parallel path of the core is connected to single drum. Sixteen downcomers coming from 4 steam drums are represented by 4 parallel paths and are connected to inlet header. Discretization scheme used for simulating the reactor coolant loop consists of RELAP5 [3] specific models for the reactor core, core top and bottom extensions, inlet feeders, tail pipes, steam drums, downcomers and inlet header. Component numbers 100, 101, 102 and 103 represent average channels and 121,122,123 and 124 represent hot channels. Forty heat (20 average and 20 hot) slabs are connected to these core volumes with appropriate axial power profile. Flow coming from inlet header is divided into 8-flow paths-four flow paths are connected to average channel and the other four are connected to hot channel. The nodalization of ECCS is shown in Figure 2 and details of the different component numbers are given in Table 1.

3. Methodologies in Safety Analysis

It is stated in IAEA safety requirements (for design of NPPs) that "safety analysis should be conducted using both deterministic and probabilistic methods". Deterministic methods show that the values of important parameters lie below the acceptance limits while probabilistic method is applied to calculate risk from a nuclear plant. For applying deterministic method, various computer codes have been developed. These codes involve lots of uncertainties in the area of safety analysis. They are represented by the uncertainty [2] of the code (associated with the code models, correlations, model options, data libraries or deficiencies of the code) and representation uncertainties (accuracy of the complex facility geometry, 3D effects, scaling, control and system simplifications) and plant data uncertainties (unavailability of some plant parameters, instrument errors and uncertainty in instrument response). US NRC (United States Nuclear Regulatory Commission) issued a revised Emergency Core Coolant System (ECCS) rule in 1988 [4] which allow the use of best estimate computer codes for safety analysis, <u>provided uncertainty of the calculations is quantified</u>.

3.1. Conservative v/s Best Estimate Approach

The conservative approach uses codes based on conservative models. Initial plant parameters, availability of safety control components and systems and operator action are selected over the nominal values in order to predict critical parameters conservatively. **Best Estimate** methodology is characterized by applying the best estimate codes and realistic input in contrast to the conservative approach. Introduction of realistic inputs makes it imperative to perform uncertainty analysis.

Basic advantage of the BE methodology over the conservative approach is the quantification of the safety margin between the realistic value of the calculated parameters and the defined safety limits. A conservative approach can successfully show that the facility's parameters are well below the safety limits, but it fails to predict exactly how far. The use of best-estimate codes instead of conservative codes is motivated by both economical and safety reasons: a).economical reason- it is expected that the use of best-estimate codes will allow relaxation of unnecessary technical specifications and operating limits set up by conservative codes (b) safety reason- due to the presence of numerous counter-reactions, it is difficult to prove the conservatism of conservative codes. Moreover the use of best-estimate codes allows improvement of accident management procedures due to better understanding of accident progress.

4. Methods in Uncertainty Analysis

They consist of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). These may be treated differently by different methods. Underlying idea [5] is that analysis results $y(x) = [y_1(x), y_2(x),..y_n(x)]$ are functions of uncertain inputs $x = [x_1, x_2, ., x_n]$. Uncertainty in y(x) is driven by uncertainty in x. The objective of “uncertainty” analysis is to calculate uncertainty in y(x) given the uncertainty in x and the objective of “sensitivity” analysis is to identify the importance of the individual elements of x with respect to the uncertainty in y(x).

Various countries have developed different methodologies to deal with LBLOCA uncertainty. The leading method (developed by USNRC) is CSAU (i.e. Code Scaling, Applicability, and Uncertainty) and it is reported by Boyack (1990) [6]. It was developed and demonstrated to support revised ECCS rule. The methodology is structured, traceable, and practical, as is needed in the regulatory arena. It addresses in a comprehensive and systematic manner questions concerned with following points:

1. Whether code is capable to scale-up the process from test facility to full-scale nuclear power plant (NPP).
2. Code applicability to safety studies of a postulated accident scenario in specified NPP.
3. Quantifying uncertainties of calculated results.

CSAU methodology emphasizes a practical engineering approach that can be used to quantify code uncertainty for LOCAs. It can be most easily conceptualized as consisting of fourteen primary steps which can be grouped in three key elements:

- **Requirement and capabilities** in which scenario modeling requirements are identified and compared against code capabilities to determine code’s applicability to a particular scenario and to identify potential limitations.
- **Assessment and ranging** of parameters in which code’s capabilities (to calculate processes important to the scenario) are assessed against experimental data to determine code accuracy, scale-up capability and sensitivity studies.
- **Sensitivity and uncertainty analysis** in which the effects of individual contributors to the total uncertainty are obtained, for which the propagation of uncertainty through the transient is properly accounted.

4.1. Direct Monte Carlo Simulation Using LHS

The best way to find 95th PCT value is Monte Carlo method using simple random sampling. This technique is impractical due to very large number of RELAP5 calculations (~10^6) and the high cost of each RELAP5 run. To overcome this limitation, LHS is used instead of simple random sampling. LHS is a stratified sampling technique where random variable distributions are divided into equal probability intervals [7]. It generates a sample size N for n variables. A 1/N probability is randomly selected from within each interval that is partitioned into N non-overlapping ranges for each basic event. It is recommended [8] that good results can be obtained if the LHS sample size is between (4/3) × (number of parameters) and 5× (number of parameters). According to this formula for six input parameters, LHS sample size of 30 will be enough. The selected number runs in our case are 500 which are consistent with above recommendation.

5. Results and Discussion

This work follows from the reference [2] except for the method of uncertainty quantification.

5.1. Acceptance Criteria of ECCS

According to the document 10 CFR part 50 appendix K [4] acceptance criteria for ECCS performance for reactors are the following:

1. Clad temperature to be less than 1204 ^{o}C (1477 K).
2. Oxidation criteria of the clad surface to be less than 17%.
3. Hydrogen generated should not be greater than 0.01 times the hydrogen generated due to total zircaloy present in the core.
4. Coolable core geometry should be maintained.
5. Long term cooling should be provided.

It is discussed in the literature [6] that significant oxidation and hydrogen generation was never encountered within the threshold values of clad temperature. Thus, Peak Clad Temperature **(**PCT) can be considered governing safety criteria for LBLOCA. An input deck is prepared according to the standard RELAP5 manual and nodalization diagram shown in Figure 1. We have considered the case of double ended 200% rupture in the inlet header [2] for the uncertainty quantification. Nominal values of all independent parameters are chosen and various thermal hydraulic properties are predicted.

5.2 Total Reactor Power

Reactor power variation is shown in Figure 3. A 200% guillotine rupture in the inlet header is assumed at time equal to zero second. Reactor scram is initiated on sensing high dry well pressure signal. Control rods begin their inward travel and are completely inserted into the core after receiving the scram signal. Total reactor power of the core decreases sharply to a value of 90 MWt within 4.6 seconds because of negative void reactivity feedback and later on stabilized to 44 MWt.

5.3 Inlet Header Pressure Transients

Pressure at inlet header is characterized by initial sharp drop due to large break flow owing to sub-cooled condition in the header. This sudden depressurization leads to two-phase condition in the inlet header, thus reducing critical flow rate. This sharp pressure drop is due to initial non-equilibrium condition between the two phases. Pressure continues to decrease smoothly afterwards as shown in Figure 4.

5.4 Break flow transient

Severed pipe begins discharging pressurized coolant into containment with the initiation of break. Steam discharge behavior is depicted in Figure 5. Maximum discharge rate of 26194 kg/s occurs during initial blowdown period. Sudden discharge through break depressurizes the reactor system which in turn reduces discharge rates. Break discharge comes down to a low value 2075 kg/s at the end of the transient.

5.5 Clad Temperature Transient

Figure 6 shows fuel clad temperature variation for the average channel, the hot channel and the hot pin. Break is initiated at t=0 second and clad temperature starts rising sharply. Temperature rise is driven by stagnation of flow, flow reversal, increase in reactor power and poor heat transfer coefficient. This leads to loss of coolant from the primary system. Maximum clad temperature in hot pin is 975.59 K, which occurs at 21 sec after the break.

5.6 Results of Uncertainty Analysis

Results of the nominal case show that maximum value of PCT occurs during the blowdown phase. Uncertainty (in the value of PCT) is mainly governed by fuel stored energy in such cases as reported by Wulf (1990) [9]. Six important parameters [2] have been considered for uncertainty analysis and reported in Table 2 with their statistical properties as widely discussed in literature [6]. All the parameters are varied either within the range of ±3 times the standard deviation ($\pm 3\sigma$) or within minimum and maximum admissible values. Values of PCT are computed for variations in individual variables using RELAP5 MOD3.2 code [3]. This work involves Latin Hypercube Sampling (LHS) [10] for generating the values of input variables. PCT values obtained from 500 RELAP5 runs are used directly for Monte Carlo simulations. MATLAB [11] is used for implementation of LHS in place of random sampling. The CDF of input variables are plotted to check whether they preserve the distribution and cover the whole sample space or not. This requirement is fully satisfied. Figures 7 and 8 show the shape of PDF and CDF of PCT indicating that the 95% probability value of PCT is **1287 K** and the 50% probability value is 1027 K.

6. Conclusion

Uncertainty analysis for PCT during large break LOCA for NCR is performed. It involves direct Monte Carlo simulation using Latin Hypercube Sampling. A direct Monte Carlo method using LHS is proposed and used for uncertainty propagation. This method is free of any bias and approximations compared to other methods. The 95 percentile value of PCT is 1287 K with mean value 1027 K. The 95th percentile value is fairly below the acceptance criteria of 1477 K with a margin of 190 K. It can be inferred that reactor will be safe during LOCA (with the considered uncertainties in input variables) and ECCS would be able to compensate coolant in the core when it is getting lost due to LOCA.

7. References

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[11] MATLAB, www.mathworks.com

Component Number	Description of control volume
100-01, 100-02,101-01,101-02,102-01,102-02, 103-01,103-02	Inlet feeders in 4 different flow paths – 100, 101, 102, 103
100-03, 101-03, 102-03, 103-03	Core bottom volume in 4 different flow paths
100-04 to 100-08,101-04 to 101-08,102-04 to102-08, 103-04 to103-08	5 active core volumes in 4 different flow paths
100-09, 101-09, 102-09, 103-09	Core top extension in 4 different flow paths
100-10, 100-11,101-10,101-11,102-10,102-11, 103-10,103-11	2 volumes in tail pipes in 4 different flow paths
109-01	Inlet header volume
104-01, 106-01, 406-01, 506-01	Subcooled volumes in steam drums
150-01 to 150-10, 105-01 to 105-010, 405-01 to 405-10, 505-01 to 505-10	Steam drum volumes
351 to 356	Steam piping
107, 108, 119 & 110	4 downcomers pipes clubbed for 4 steam drums
111, 113, 114, 115	Time dependent volumes representing constant feed
112 and 120	Time dependent volumes for turbine steam flow
511 and 513	Time dependent volumes for safety valve
512 and 514	Time dependent volumes for relief valve
350	Time dependent volumes for containment
900, 902, 904 and 906	Accumulators
700, 716, 717 and 718	ECCS header
704, 705, 706, 707	Piping between accumulator to ECCS header
701-01 to 701-05, 702-01 to 702-05, 703-01 to 703-05, 708-01 to 708-05	ECCS feeders and water tubes
709-01 to 709-05	GDWP
720–724	GDWP piping
710	GDWP header

Table 1 Description of Hydrodynamic Components

Name of parameter [symbol]	Distribution	Mean value (μ)	Nominal value	Standard deviation (σ)	Range covered in analysis
Reactor power **[x_1]**	Uniform	NA[a]	100%	NA	97% to 113%
Fuel conductivity **[x_2]** (as fraction of mean value)	Normal	1	1	0.1	0.7 to 1.3 (±3σ)
Gap conductance **[x_3]** (as fraction of nominal value)	Uniform	NA	1	NA	20% to 180%
Sub-cooled discharge coefficient **[x_4]**	Normal	1	1	0.042	0.874-1.126 (±3σ)
Two phase discharge coefficient **[x_5]**	Normal	1	1	0.062	0.814-1.186 (±3σ)
Decay power **[x_6]** (as fraction of nominal value)	Uniform	NA	1	NA	0.9-1.1 (±10% about nominal value)

[a] not applicable

Table 2 Statistical Properties of Independent Parameters

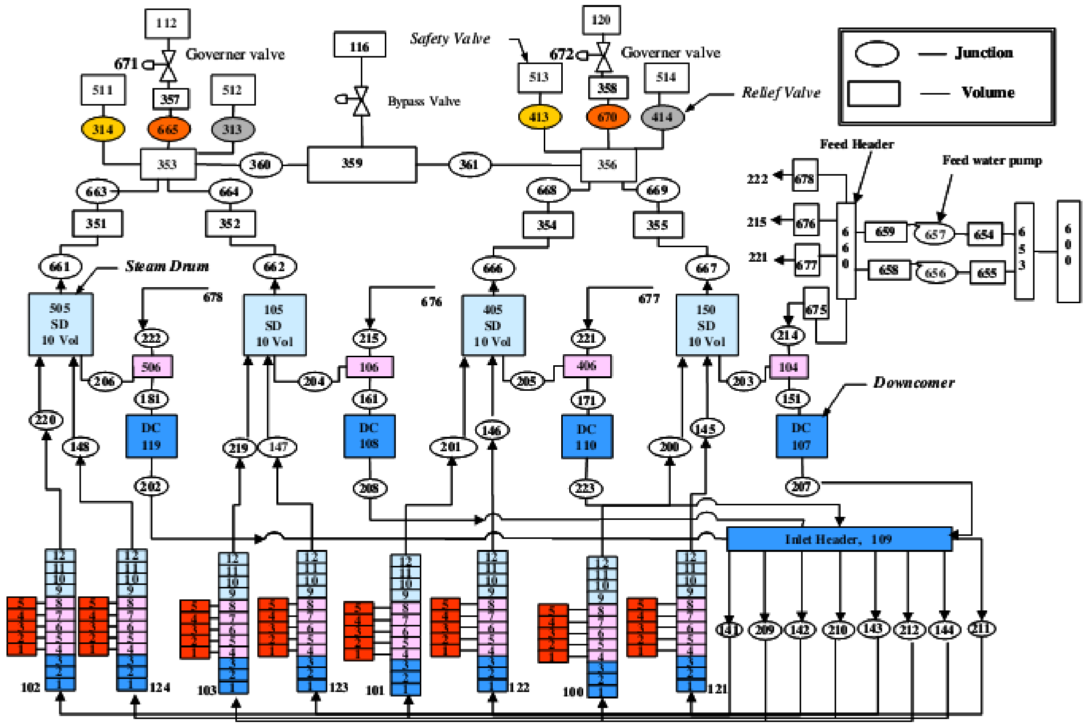

Figure 1 Nodalization Scheme of MHT for Natural Circulation Reactor

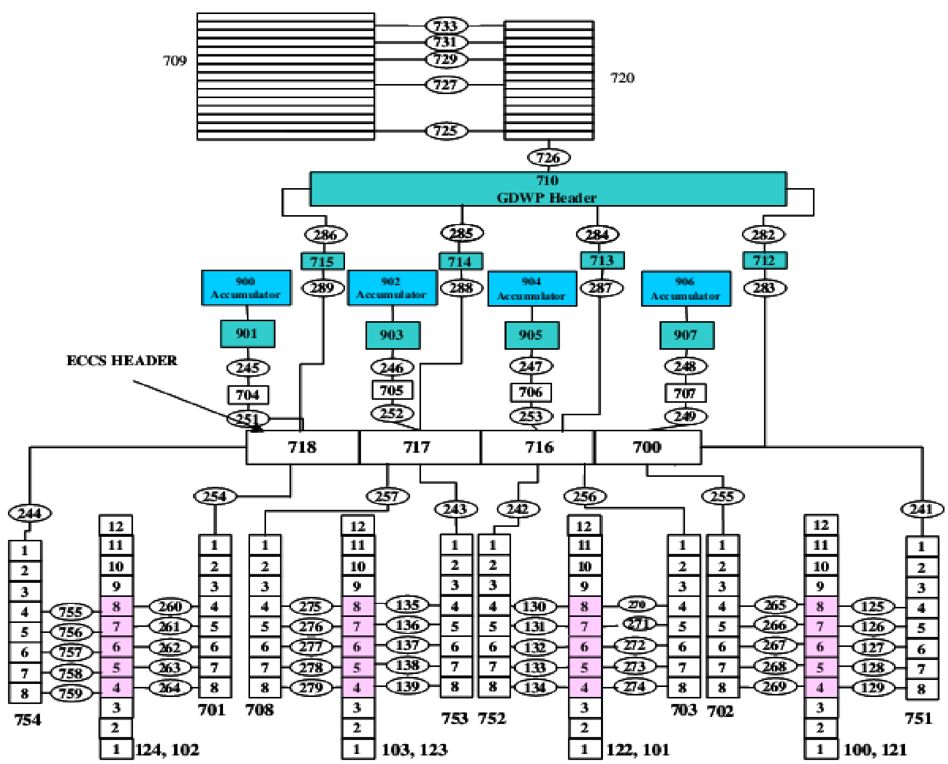

Figure 2 Nodalization of ECCS

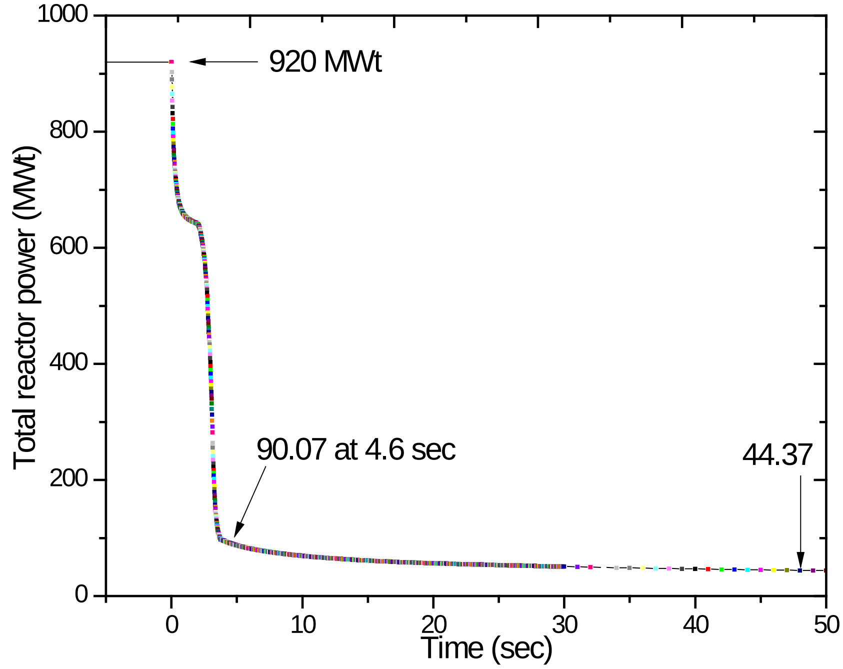

Figure 3 Total Reactor Power Transient

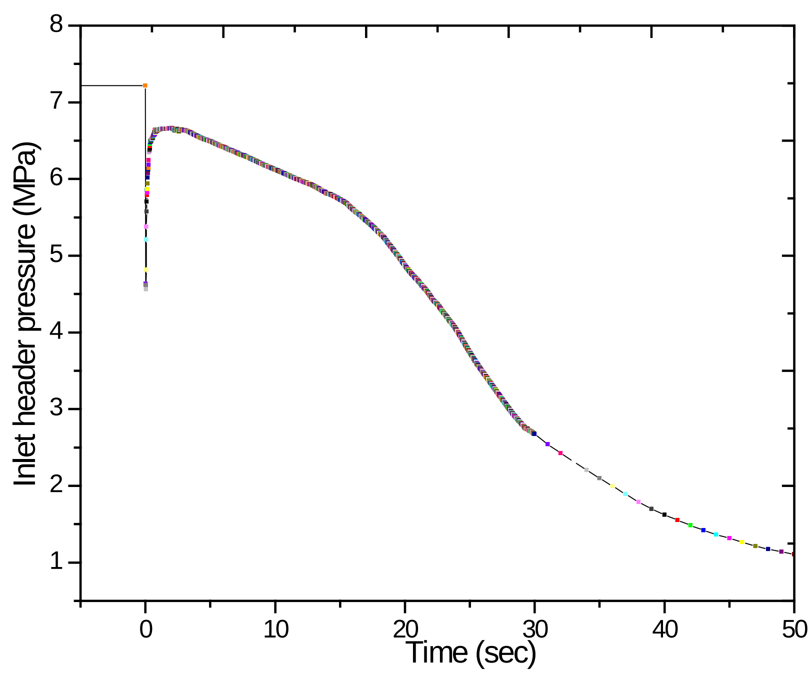

Figure 4 Inlet Header Pressure Transient

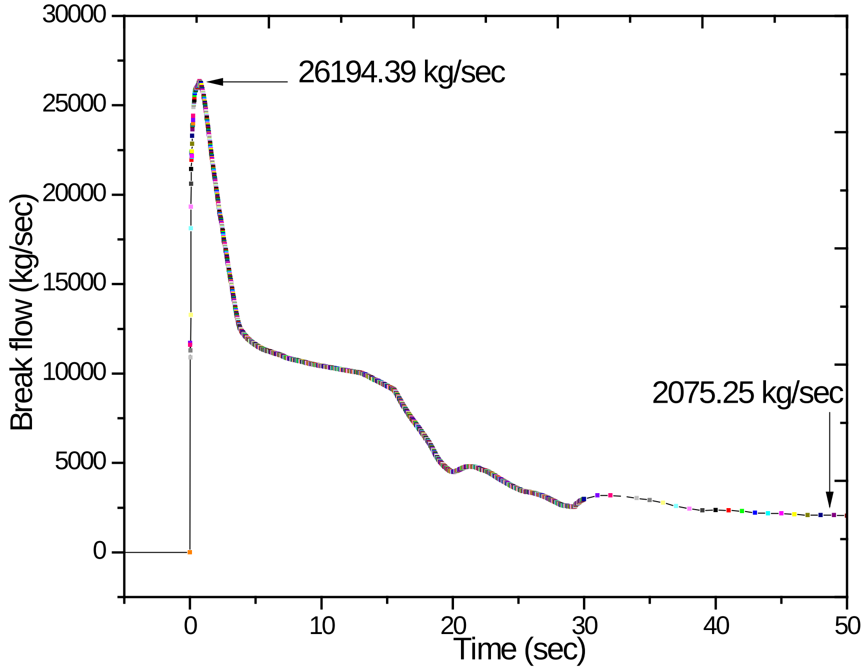

Figure 5 Break Flow Transient

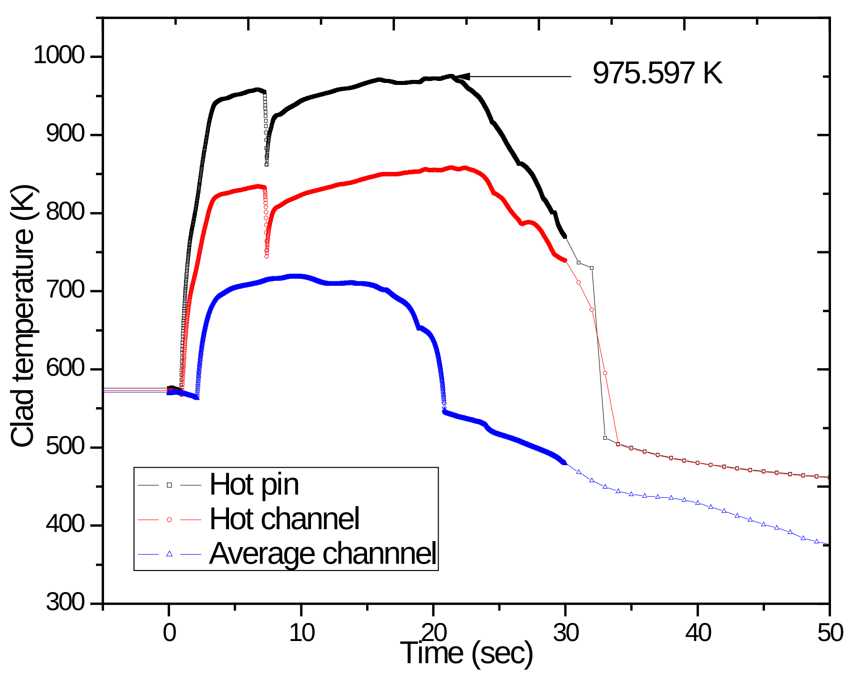

Figure 6 Clad Temperature Transient

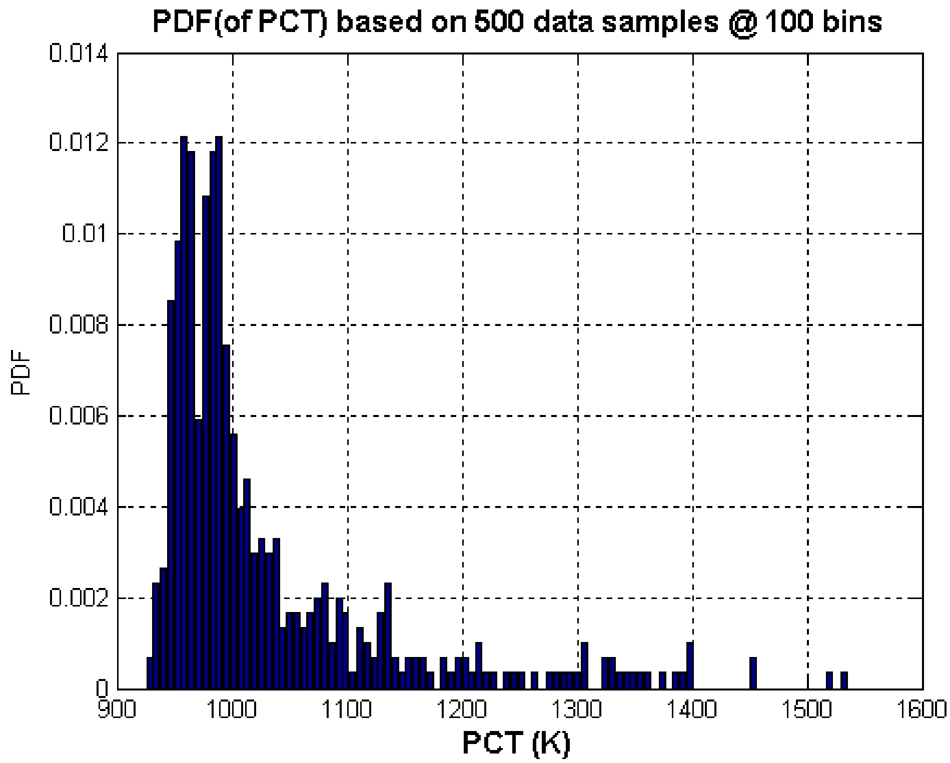

Figure 7 PDF of PCT

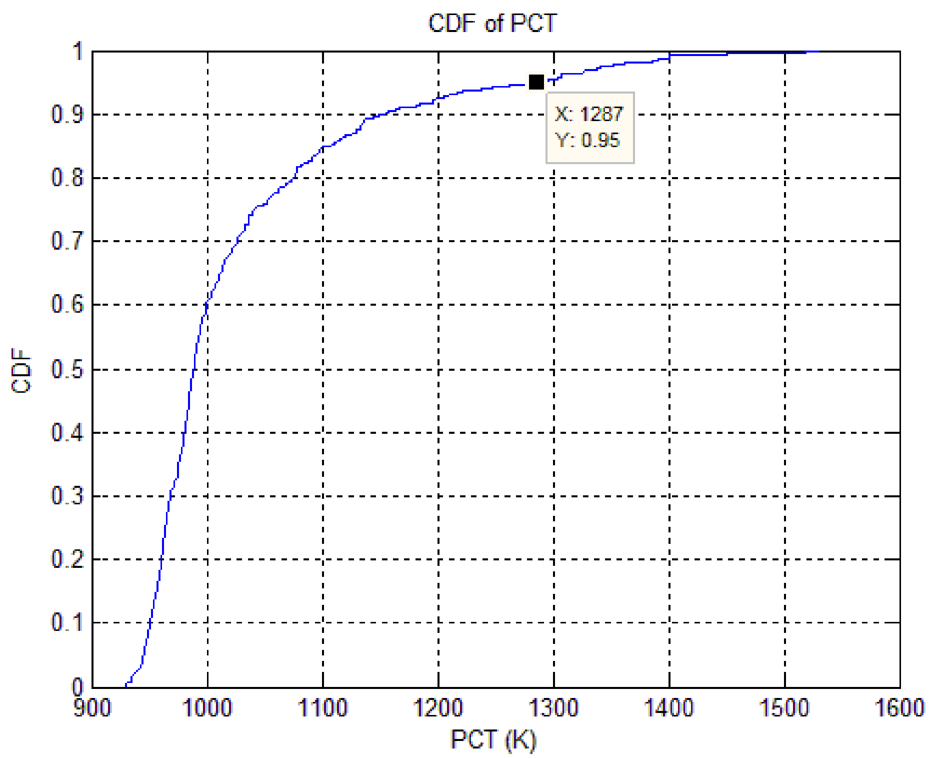

Figure 8 CDF of PCT