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Uncertainty and Sensitivity Analysis of Hot Leg Loss of Coolant Accident Simulated by RELAP5

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ABSTRACT

Among the risks that threaten the integrity of the reactor pressure vessel is the possible destruction due to pressurised thermal shock (PTS). PTS can occur during several postulated accident scenarios, including loss of coolant accidents. The purpose of this study is to perform the uncertainty and sensitivity analysis for RELAP5 simulation of small break loss of coolant accident, located in the hot leg. Namely, the results of thermal hydraulic calculations are needed for further structural analysis. The reactor selected was two-loop pressurized water reactor (PWR), for which verified and validated input deck was available. The RELAP5 developmental version 33lj from 2022 with built-in code uncertainty parameters has been used in this study. In total 15 uncertain input parameters have been considered. For uncertainty and sensitivity analysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.5 has been used. For uncertainty and sensitivity analysis 130 runs have been performed with RELAP5. The main figures of merit in uncertainty analysis were reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection. The results showed that figures of merits for reference case are bounded by the results of 130 runs. The sensitivity analysis showed that the most influential parameters are high pressure safety injection system temperature and flow, initial pressurizer pressure, form loss coefficient and thermal non-equilibrium coefficient for Henry-Fauske choke flow model.

1 INTRODUCTION

The reactor pressure vessel of water-cooled reactors is one of the most important and non-replaceable nuclear power plant components. Among the risks that threaten the integrity of the reactor pressure vessel is the possible destruction due to pressurised thermal shock (PTS). PTS can occur during several postulated accident scenarios like primary side pipe breaks (small to large diameter), stuck-open valves on the primary side, main steam line breaks, stuck-open valves on the secondary side, feed-and-bleed, and steam generator tube rupture. In the frame of Advanced PTS Analyses for LTO project (APAL), where LTO abbreviation means long term operation, loss of coolant accident in hot leg has been selected in German design 1300 MW four-loop pressurized water reactor (PWR) [1].

In 2021 PTS has already been studied for two-loop PWR, Westinghouse type, under loss of coolant accident (LOCA) scenarios in the cold leg [4]. In this study same two-loop PWR has been selected for uncertainty and sensitivity study of LOCA in the hot leg. The selection of input uncertain parameters was based on the results obtained in the frame of APAL project [1]. It should be also noted that in the APAL the phenomena identification and ranking table (PIRT), which is basis for selection of input uncertain parameters, was obtained by expert judgement of the project partners after considering the PIRTs developed for other transients, such as those presented in Refs. [2] and [3].

2 CODES, TOOLS, INPUT MODEL, SCENARIO, INPUT UNCERTAIN PARAMETERS AND UNCERTAINTY METHOD DESCRIPTION

In this section first codes and tools are described. Then RELAP5 input model is presented and scenario description for reference case. Next, selected input uncertain parameters with distributions are given. Finally, uncertainty and sensitivity analysis method is described.

2.1 Codes and Tools Used

U.S. Nuclear Regulatory Commission RELAP5/MOD3.3 developmental version 3.3lj from May 2022 has been used for thermal-hydraulic calculations [5]. This version added 50 new uncertainty quantification (UQ) sensitivity parameters. The parameters relate to interfacial heat transfer (27 variables) and wall heat transfer (23 variables).

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for the Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.5 has been used for sampling, uncertainty and sensitivity analysis [6]. SUSA tool has been developed to facilitate the performance of uncertainty and sensitivity analyses based on Monte Carlo simulation. It combines well established methods from probability calculus and mathematical statistics with a comfortable graphical user interface. First identification of the input parameters which represent the main uncertainty sources of the result of the applied computer code and the formulation of the uncertainties of the identified parameters is done. Further analysis steps are sampling of parameter values, and the calculation of statistics indicating the uncertainty and sensitivity of the computational result. In addition, Jožef Stefan Institute (JSI) in-house Microsoft Excel macros have been developed for uncertainty analysis.

2.2 RELAP5 Input Model Description

The RELAP5 input model for a two-loop PWR is shown in Figure 1. The primary side includes the reactor pressure vessel (RPV) and the two primary loops (loops 1 and 2), both with a reactor coolant pump (RCP) and steam generator (SG) U-tubes. The pressurizer (PRZ) vessel is connected to its spray lines – two power operated relief valves (PORVs) and two safety valves – and to the primary loop 1 through the surge line (SL). The two trains of emergency core cooling system (ECCS) comprise active high pressure safety injection (HPSI) and low pressure safety injection (LPSI) pumps and accumulators (ACCs). The secondary side consists of the steam generators with main feedwater (MFW) and auxiliary feedwater (AFW) systems, and main steam lines with SG relief valves and main steam isolation valves. Finally, the break is modelled with two valves connected to the hot leg (and volume after the valve to collect discharged mass), which give possibility to model single and double-ended breaks. In our case one valve was opened at accident initiation.

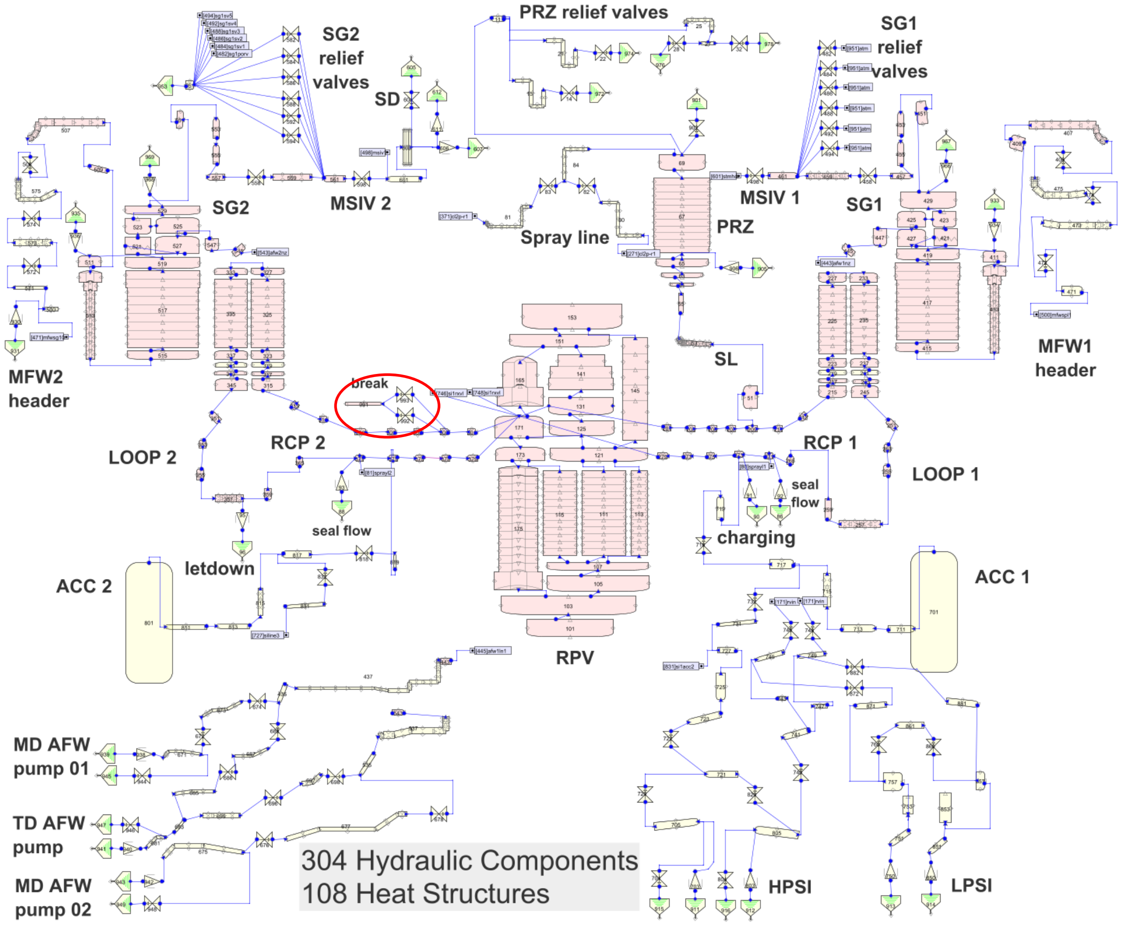

Figure 1: RELAP5 input model of a two-loop PWR (hydraulic components view)

2.3 Scenario Description

Initiating event is 45.6 cm^2 (7.62 cm - 3” diameter) hot leg break loss of coolant accident (HL LOCA) at 0.01 s from full reactor power. Initial and boundary conditions are shown in Table 1). A loss of alternate current (AC) has been assumed to occur at the same time as the break occurrence. Therefore, the reactor coolant pumps (RCPs) trip immediately. One train of active emergency core cooling systems is available (one high pressure injection pump, one low pressure injection pump) and both accumulators. Both motor driven auxiliary feedwater pumps were assumed available.

Table 1: Initial and boundary conditions

Parameter	Two-loop PWR	RELAP5
Core power (MW)	1994	1994
Steam generator power (MW)	1000	996.5 / 1002.5
Pressurizer pressure (MPa)	15.51	15.51
Steam generator pressure (MPa)	6.28	6.44 / 6.44
Cold leg temperature (K)	559.2	559.51 / 559.32
Hot leg temperature (K)	596.9	596.79 / 596.79
Feedwater temperature (K)	492.6	492.5
Pressurizer level (%)	55.7	55.8
Steam generator narrow range level (%)	69.3	69.3 / 69.3
Steam mass flow (kg/s)	544.5	541.3 / 544.5

After break occurrence the reactor trips on (compensated) low pressurizer pressure (12.99 MPa), which further causes the turbine trip. The safety injection (SI) signal is generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal active safety systems like high pressure safety injection (HPSI) pump, low pressure safety injection (LPSI) pump and both motor driven (MD) auxiliary feedwater (AFW) pumps start. HPSI pump start to inject with 10 s delay on SI signal. When primary pressure drops below 4.96 MPa, both accumulators start to inject. When primary pressure drops below 1.13 MPa, LPSI pump start to inject.

2.4 Input Uncertain Parameters with Distributions

In total 15 input uncertain parameters have been selected based on the results obtained in the frame of APAL project [1]. The following four input uncertain parameters have been considered not much significant in our study for selected HL LOCA: secondary side pressure, ACC initial nitrogen volume, and HPSI and LPSI pump pressure curve multipliers. The reference case consists of values shown in column four (reference or best estimate values).

Table 2: Input uncertain parameters with distributions

Par. No.	Parameter Name	Unit	Ref. Value / Best estimate	Distribution Type	Distribution Parameter1	Distribution Parameter2	Minimum	Maximum
1	Core power	W	1994E6	Normal	1994E6	19.94E6	-infinity	infinity
2	Pressurizer pressure	Pa	15.512E6	Normal	15.512E6	0.15512E6	-infinity	infinity
3	Decay heat	-	1	Uniform	0.9	1.1	0.9	1.1
4	Timing of SIS actuation	s	5	Uniform	0	20	0	20
5	ACC injection temperature	K	322	Uniform	312	332	312	332
6	ACC initial pressure	Pa	4.9277E+06	Uniform	4.7277E+06	5.1277E+06	4.7277E+06	5.1277E+06
7	HPSI temperature	K	310	Uniform	295	325	295	325
8	HP pump flow curve	-	1	Normal	1	0.1	-infinity	infinity
9	LP pump flow curve	-	1	Normal	1	0.1	-infinity	infinity
10	Initial pressurizer level	%	55.7	Uniform	48.34	63.06	48.34	63.06
11	Thermal-nonequilibrium coefficient for Henry-Fauske model	-	0.93	Weibull	7	1	0	1.5
12	Single-phase liquid to wall HTC	-	1	Log. Uniform	0.8	1.2	0.8	1.2
13	Single-phase vapour to wall HTC	-	1	Log. Uniform	0.8	1.2	0.8	1.2
14	Wall-drag coefficient	-	1	Log. Uniform	0.5	2	0.5	2
15	Form-loss coefficient	-	1	Log. Uniform	0.9	1.1	0.9	1.1

2.5 Uncertainty and Sensitivity Analysis Method

In the uncertainty analysis the input uncertain parameters are propagated through the computer code model [6]. Multiple simulations of the transient scenario produce multiple sets of simulation output, each set of output is the result of a unique combination of randomly-chosen values for the input parameters. The different values finally obtained for the computational result can then be analysed by statistical methods in order to derive appropriate indicators for the uncertainty of the result. Approach of Wilks has been used.

The following five figures of merit (FOM) were used in the uncertainty analysis: (a) primary pressure, (b) liquid temperature at reactor vessel cold leg (CL) inlet, (c) liquid temperature below reactor vessel CL inlet, (d) reactor vessel wall temperature at CL inlet, and (e) reactor vessel wall temperature below CL inlet. For primary pressure in each time step maximum of 130 runs is selected, while for temperatures minimum in each time step is selected. Namely, higher pressure and lower temperature are more challenging for PTS.

In sensitivity analysis Spearman's rank correlation coefficient has been used, which takes into account a degree of association between two random parameters (e.g. input uncertain parameter and output uncertain parameter).

3 RESULTS

The results are shown in Table 3 and Figures 2 through 6. The sequence of events for reference case is shown in Table 3.

Table 3: Sequence of events for reference case.

Event	Time (s)
Break occurrence	0.01
Reactor trip signal	2.37
Turbine trip	2.37
Safety injection signal	18.89
Main feedwater pump trip	18.9
High pressure safety injection	23.89
Auxiliary feedwater start	23.9
Accumulator injection	905
Low pressure injection	2660

The results for reference calculation are shown in Figure 2. The reference case value for thermal-hydraulic non-equilibrium constant in Henry-Fauske (HF) choke flow model was not selected to be default value 0.14, because it is very low probability to be sampled this value of parameter having Weibull distribution. Therefore value 0.93 has been selected, which represents approximately 50 percentile of Weibull distribution. However, it should be noted that this change of value has quite large influence on the results shown in Figure 2. The break flow is initially higher (see Figure 2(c)), therefore the pressure and temperature drop is faster as shown in Figure 2(a) and Figure 2(b), respectively. Due to faster pressure drop, there is earlier and/or earlier injection of HPSI pump, accumulator and LPSI pump shown in Figures 2(d), 2(e) and 2(f), respectively.

Figure 3 shows the simple random sampling produced distribution and distribution characteristics of the results for primary pressure. Namely, SUSA can calculate distribution characteristics at each time step like minimum labelled as 'min', maximum labelled as 'max' and mean labelled as 'mean'. The reference (best estimate - BE) calculation is labelled 'BE'. The 'min', 'max' and 'mean' present the minimum, maximum and mean value of 130 runs (at least 124 runs are needed to quantify scalar uncertainty for three figures of merit [7] when independent) in each time step. These values were calculated also in Microsoft Excel for validation of results obtained by SUSA. Similarly, Figure 5 and Figure 6 simple random sampling produced distribution and distribution characteristics of the results for liquid temperature at reactor vessel CL inlet and for reactor vessel wall temperature at CL inlet, respectively.

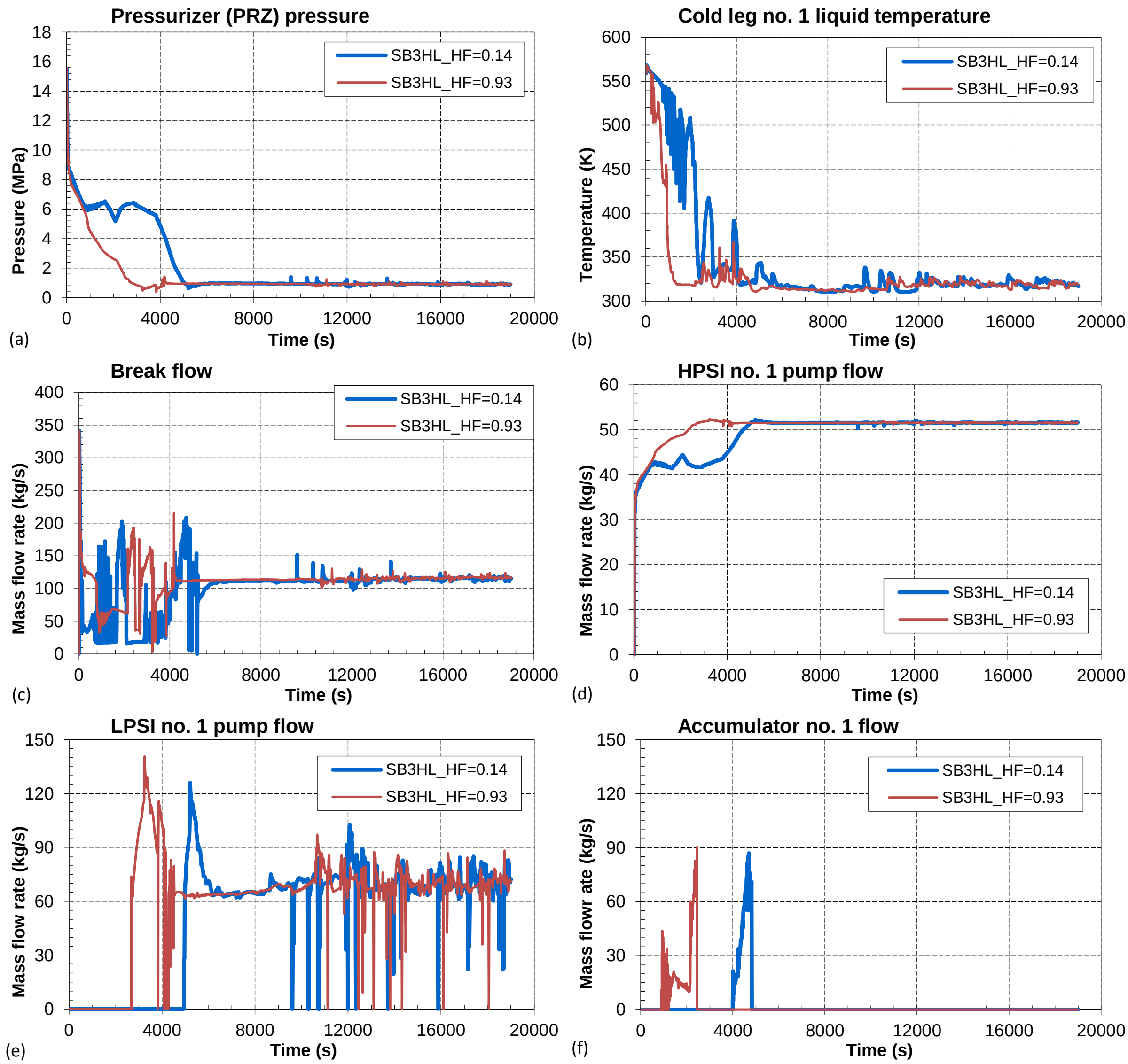

Figure 2: RELAP5 results for two reference cases

In each three figures showing 130 runs the BE calculation is bounded by minimum and maximum curve, while there is difference between BE calculation and mean curve.

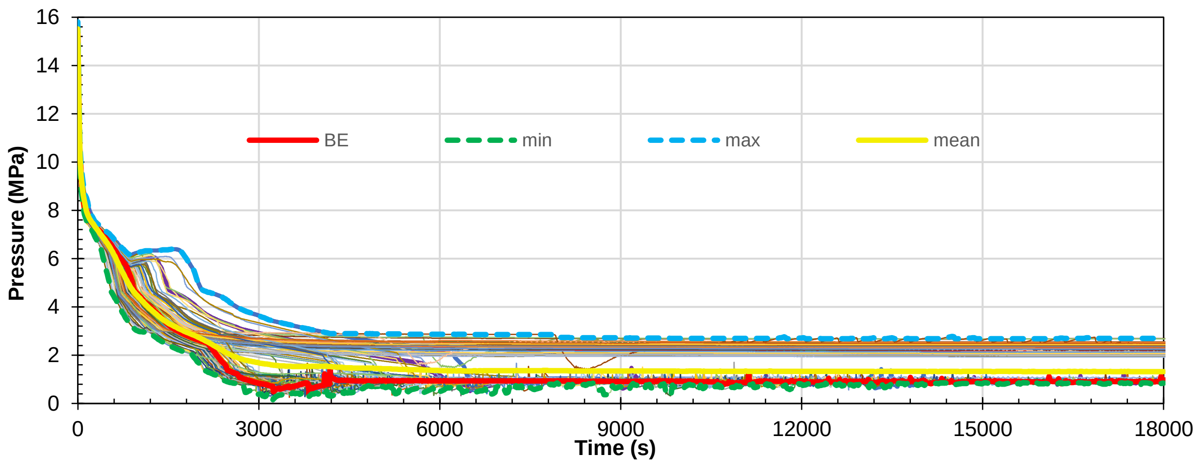

Figure 3: Simple random sampling produced distribution and its characteristics for primary pressure

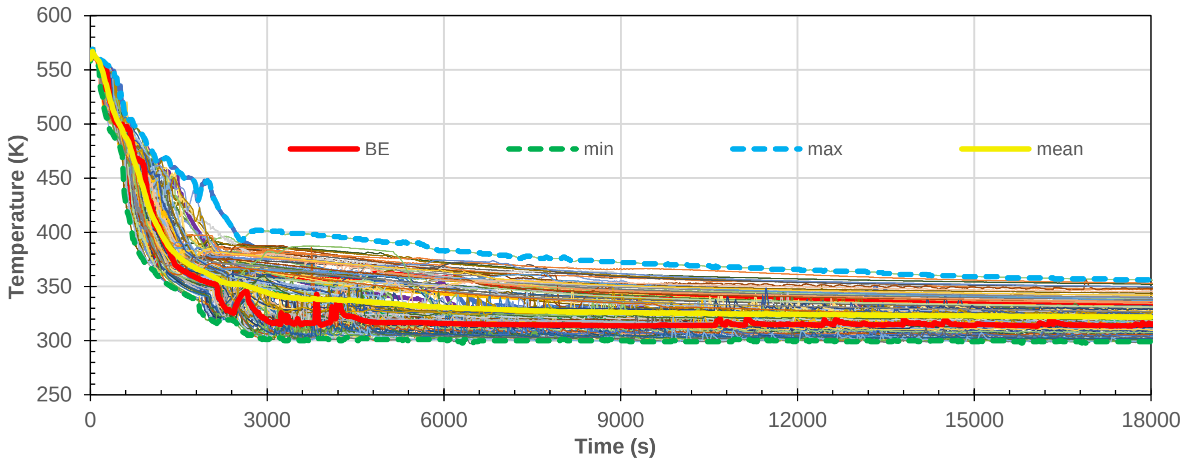

Figure 4: Simple random sampling produced distribution and its characteristics for liquid temperature at reactor vessel CL inlet

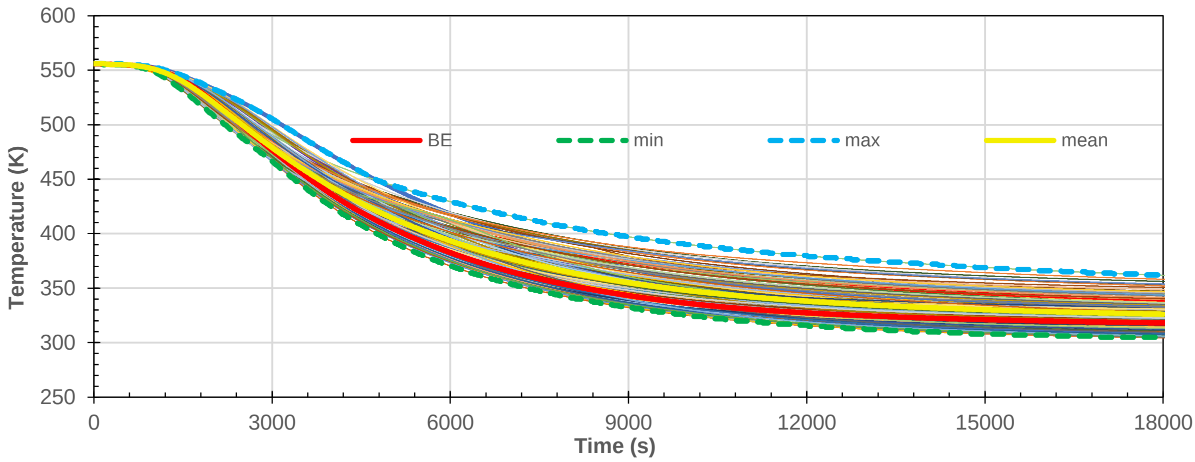

Figure 5: Simple random sampling produced distribution and its characteristics for reactor vessel wall temperature at CL inlet

Finally, Figure 6 shows results of sensitivity analysis for reactor vessel wall temperature at CL inlet. Spearman rank correlation coefficient (SRCC) measures the correlation between the inputs and output on their rank data instead of their actual data, it is thus effective even when inputs and outputs differ greatly in magnitude. Values from SRCC range from −1 to +1. Positive and negative signs indicate positive and negative correlations between the inputs and the output. A larger absolute value means a greater dependency between the inputs and the output.

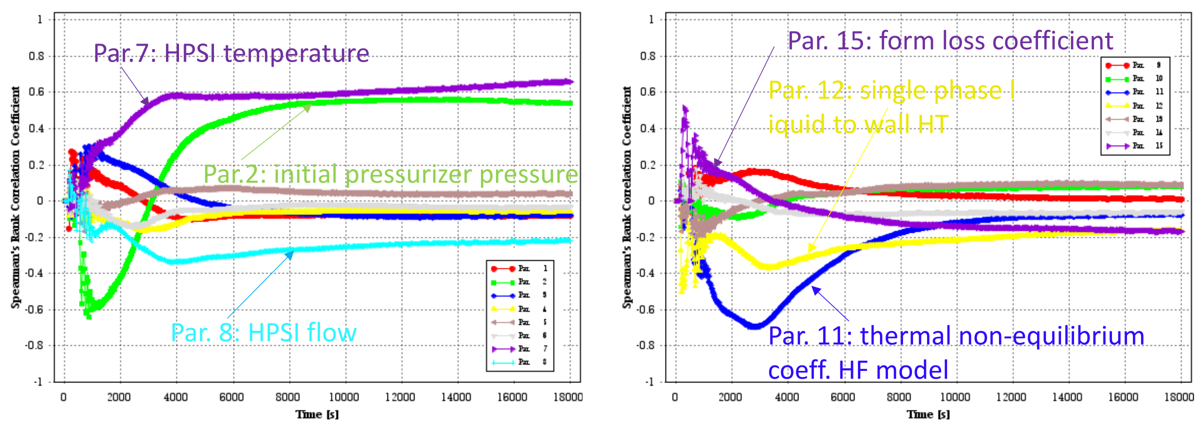

Figure 6: Sensitivity analysis for reactor vessel wall temperature at CL inlet

In the early phase the reactor vessel wall temperature at CL inlet has greatest dependency (negative sign) on the initial pressurizer pressure ('Par. 2') at around 1000 s and thermal-nonequilibrium coefficient for HF model ('Par. 11) at around 3000 s, while after 4000 s the greatest dependency (positive sign) is on HPSI temperature ('Par. 7'). Finally, after 5000 s reactor vessel wall temperature at CL inlet has great dependency (positive sign) also on initial pressurizer pressure ('Par. 2').

4 CONCLUSIONS

Uncertainty and sensitivity analysis of hot leg loss of coolant accident in two-loop PWR, Westinghouse type, has been performed for RELAP5 version 3.3lj calculation. The selection of input uncertain parameters and distributions has been based on the results of APAL project. In total one best-estimate run and 130 sampled runs have been performed. In the uncertainty analysis minimum and maximum values in each time step were determined, representing lower and upper uncertainty bound, respectively. In the sensitivity analysis the Spearman rank correlation coefficient sensitivity measures for output uncertain parameters were used.

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