

UNCERTAINTY ANALYSIS OF THE 35% REACTOR INLET HEADER BREAK IN A CANDU 6 REACTOR USING RELAP/SCDAPSIM/MOD4.0 WITH INTEGRATED UNCERTAINTY ANALYSIS OPTION

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Abstract

The RELAP/SCDAPSIM/MOD4.0 code, designed to predict the behavior of reactor systems during normal and accident conditions, is being developed as part of an international nuclear technology Software Development and Training Program (SDTP). RELAP/SCDAPSIM/MOD4.0, which is the first version of RELAP5 completely rewritten to FORTRAN 90/95/2000 standards, uses the publicly available RELAP5 and SCDAP models in combination with (a) advanced programming and numerical techniques, (b) advanced SDTP-member-developed models for LWR, HWR, and research reactor analysis, and (c) a variety of other member-developed computational packages. One such computational package is an integrated uncertainty analysis (IUA) package being developed jointly by the Technical University of Catalonia (UPC) and Innovative Systems Software (ISS).

RELAP/SCDAPSIM/MOD4.0(IUA) follows the input-propagation approach using probability distribution functions to define the uncertainty of the input parameters. The main steps for this type of methodologies, often referred as to statistical approaches or Wilks' methods, are the ones that follow:

1. Selection of the plant;
2. Selection of the scenario;
3. Selection of the safety criteria;
4. Identification and ranking of the relevant phenomena based on the safety criteria;
5. Selection of the appropriate code parameters to represent those phenomena;
6. Association of uncertainty by means of Probability Distribution Functions (PDFs) for each selected parameter;
7. Random sampling of the selected parameters according to its PDF and performing multiple computer runs to obtain uncertainty bands with a certain percentile and confidence level;
8. Processing the results of the multiple computer runs to estimate the uncertainty bands for the computed quantities associated with the selected safety criteria.

RELAP/SCDAPSIM/MOD4.0(IUA) calculates the number of required code runs given the desired percentile and confidence level, performs the sampling process for the user-selected parameters including code embedded correlations and models such as the wall-to-fluid heat transfer package, executes the code runs to propagate the uncertainty through, and processes the output data to derive the tolerance interval defining the uncertainty region.

This paper includes a:

1. Brief description of RELAP/SCDAPSIM/MOD4.0(IUA) and associated CANDU 6 base input model for the 35% reactor inlet header (RIH) break scenario,

2. Discussion of the safety criteria, relevant phenomena, and associated code parameters used in the uncertainty analysis,
3. Discussion and conclusions for the RIH transient and associated uncertainties.

Keywords: Thermal hydraulics, Uncertainty analysis, CANDU, RIH

Introduction

The present paper describes the uncertainty analysis of the 35% reactor inlet header (RIH) break scenario simulated in a CANDU6 reactor. The main elements of the work here presented are a complete input deck file for the CANDU6 reactor and the uncertainty analysis package implemented in the RELAP/SCDAPSIM/MOD4.0 code version.

The paper is divided into four sections. The first section provides a brief description of the RELAP/SCDAPSIM/MOD4.0 uncertainty package and the basic principles of the BEPU (Best Estimate Plus Uncertainty) methodology that the package follows. The second section is devoted to describe the input decks and base case simulation of the CANDU6 35% RIH break scenario. The third section describes the application of the RELAP/SCDAPSIM/MOD4.0 package to evaluate the uncertainty in the maximum cladding temperature during a 35% RIH break. The last part provides the concluding remarks of the work.

1. RELAP/SCDAPSIM/MOD4.0 uncertainty analysis package

The RELAP/SCDAPSIM/MOD4.0 uncertainty analysis package follows the probabilistic “input uncertainty propagation” approach of the BEPU methodologies (see detailed description in [1]). The characteristic steps of such approach are the use of a best-estimate code to simulate the scenario, the selection of a set of input parameters based on their impact to the safety criteria, the determination of their associated uncertainty in terms of probabilistic distribution functions (PDF's), the random sampling of the parameters based on the defined uncertainty, the multiple execution of the code using the generated random samples, and the application of order statistics theory to determine the uncertainty regions for the restrictive output quantities, *i.e.* the safety criteria. In addition to that, a main element of the probabilistic BEPU methods is the Wilks' formula, which determines the number of required code runs to obtain the estimation of the i^{th} percentile of the output quantity given a certain confidence level, and independently of the number of selected input parameters.

A description of the RELAP/SCDAPSIM/MOD4.0 uncertainty package has been already presented in NUTHOS7 and NURETH13 conferences (see [2] and [3]), in which a PWR LBLOCA uncertainty analysis was performed. However, a short description follows to illustrate the required steps to execute it. The application of the uncertainty package is divided into three consecutive phases:

- The first one, hereafter referred to as *setup* phase, consists in the determination of the number of code runs and the generation of the input random samples required to execute each code run. The code required information is related to the Wilks' formula and to the uncertainty associated to the selected input parameters. The information is implemented in a separate input file specific to the *setup* phase that includes the base case input deck in order that the code cross-checks the user supplied data. As it will be clearly illustrated in section 3 of the present paper, the input parameters are classified into two types depending upon their modification can be implemented from a regular input deck (parameters labelled as “input treatable parameters”) or need for source code modification (parameters labelled as “source

correlation parameters”). A main feature of the package is the availability of the latter type for modification from the *setup* input file, *i.e.* without the need of modifying the source code.

- The second phase, hereafter referred to as *execution* phase, consists in the multiple execution of the base case modified by the random samples generated in the *setup* phase. The *execution* phase does not require any additional file: each code run uses the same base case input file and modifies it by loading the data generated in the previous phase.
- The third phase, hereafter referred to as *post-processing* phase, consists in the analysis of the multiple code runs results’ to build the uncertainty limits. The generated uncertainty limits are unilateral tolerance limits defined by the percentile, confidence level and order of Wilks’ formula application specified in the *setup* input file. To be more precise the code generates an upper and a lower uncertainty limit based on the information associated to the upper one. The required data for the *post-processing* phase are the output quantities to be analysed and the code runs from which the data is to be extracted. The information for this phase is implemented in a separated file specific to the *post-processing* phase.

Summarizing, the main features of the package are the generation of the input random samples, the automated code execution and the processing of the code results to obtain the uncertainty limits for the desired output quantities.

2. CANDU6 35% RIH break

2.1 Short CANDU description

CANDU is a pressurized water reactor developed by AECL in Canada. It uses heavy water in a closed primary circulation loop to transport heat to the steam generators. In a second closed loop the light water steam acts as the working fluid. The main difference between a CANDU and a PWR plant is the reactor assembly. At CANDU the primary water is heated by the fuel in several small horizontal fuel channels connected by individual feeders to headers, whereas at PWR this heating takes place in a large pressure vessel, where all fuel is located. Outside the reactors themselves, the two power systems are very similar. CANDU6 plant has two identical loops connected to a common pressurizer in a figure eight pattern.

The high-pressure emergency core cooling (HPECC) system design uses gas-driven accumulator tanks. The following ECC stages use pumps to provide emergency core cooling. These pumps are firstly supplied with dousing water and then with water recovered from the reactor building floor.

Following a LOCA, when the primary loop system pressure reaches 5.42 MPa (g), the loop isolation valves (D2O feed valves, pressurizer isolation valves and primary loop purification system valves) close to prevent the transfer of coolant from one primary loop to the other. Each loop becomes isolated from the pressurizer, feed and bleed system and the purification system. In addition, steam generators are rapidly cooled in order to depressurize the heat transport (HT) system by opening the main steam safety valves (MSSV).

A high-pressure accumulator system containing one gas tank and two water tanks ensures the high-pressure emergency core cooling to the reactor. The tanks are located in the high-pressure ECC building outside of containment. The total quantity of water in the two water tanks is more than three times the inventory of one heat transport loop. The ECC gas tank, normally pressurized to 4.14

MPa (g) is isolated from the ECC water tanks by two parallel pneumatically operated valves. The ECC water tanks and the piping downstream are normally pressurized to 275 kPa and the water temperature is maintained at 21°C by an external heater and a recirculation pump. Two parallel isolating valves known as the HP injection valves are provided to isolate the high-pressure system from the injection piping downstream. These valves are normally closed. Two check valves, in series with and downstream of the HP injection valves provide passive protection to ensure that the HP accumulator system does not spuriously become over-pressurized by the HT system. Since the ECC system is a light water system, two rupture discs, one in each of the common injection lines are used to provide a positive interface between H₂O and D₂O. Each of the injection lines to the reactor headers contains two motorized valves in parallel, named D₂O isolation valves. They are normally closed and serve to isolate the heat transport system from the ECC system.

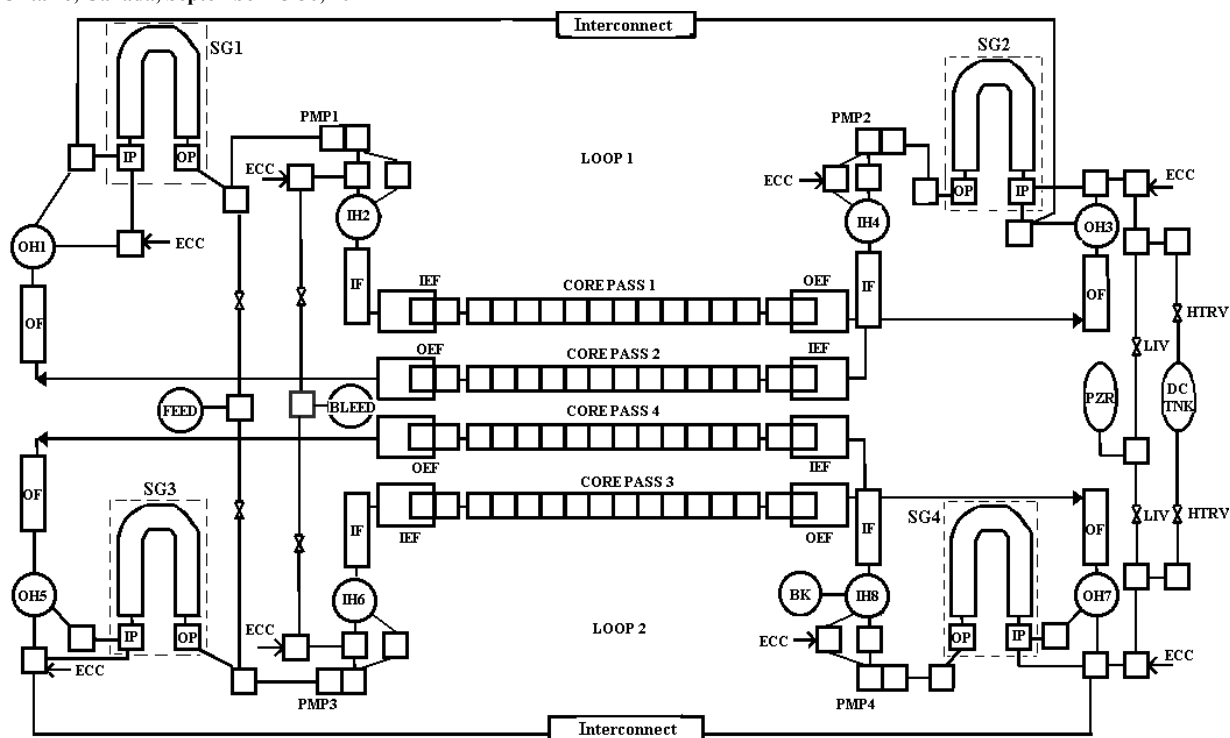
2.2 CANDU model

RELAP5 was successfully used to thermal hydraulic analyses of design basis accidents for CANDU reactors [4-6] and to the modelling of CANDU like experimental facilities [7-9]. These assessments make reliable the utilization of RELAP5 to simulate the thermal-hydraulic response of CANDU plant during LBLOCA.

A CANDU6 RELAP5 input model was developed, and it is shown in Figures 1, 2 and 3. The model consists of 390 volumes and 413 junctions. It includes the representation for the two loops, four fuel channels (one per core pass), inlet headers, feeders and end fittings, outlet headers, feeders and end fittings, four steam generators (SG) and associated reactor coolant pumps, pressurizer and the high pressure ECC injection stage. A simplified model for the secondary side has been developed with the turbine and condenser represented by time-dependent volumes and the feed water train modeled as time-dependent volume and time-dependent junction. More details can be found in [4] and [5].

The fuel channel was divided in 12 volumes, one for each fuel bundle. The 37 fuel pins of the fuel bundle are combined into a single fuel pin heat structure maintaining the surface area, mass and equivalent heated perimeter. The fuel pin is radially discretized in ten regions simulating the different layers. Heavy water properties for all systems connected to primary loop, including ECC, were used. A trip valve in the RELAP5 input deck simulates a reactor inlet header break. The break is assumed to occur at RIH 8, see Figure 1. Thus the loop 2 becomes the broken loop and the core pass 4, downstream broken inlet header, which would be the most affected core pass becomes the critical core pass.

Generally, for CANDU LOCA analysis the power pulse after the break due to positive void reactivity is calculated with a 3D finite difference neutronic code coupled with a thermal hydraulic code for the first seconds after the LOCA. Then the reactor power is tabulated and in this form is used in the present study. Thus, the void reactivity feedback was not quantified in the present study. This will be done in a future work.



LEGEND

SG - Steam Generator
IF - Inlet Feeder
OF - Outlet Feeder
IP - SG Inlet Plenum
OP - SG Outlet Plenum

PMP - Pump
IEF - Inlet End Fitting
OEF - Outlet End Fitting
IH - Inlet Header
OH - Outlet Header

ECC - Emergency Core Cooling Injection Point
PZR - Pressurizer
DC - Degaser-Condenser
LIV - Loop isolation Valve
HTRV - Heat Transport Relief Valve

Figure 1 CANDU 6 Heat Transport System nodalization

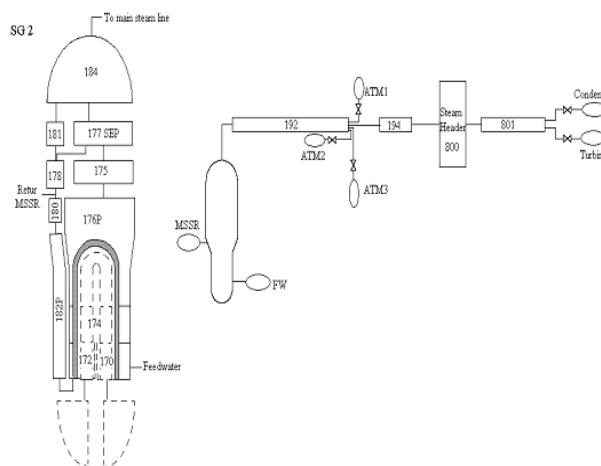


Figure 2 Steam generator and main steam line nodalization (similar for the other three SG)

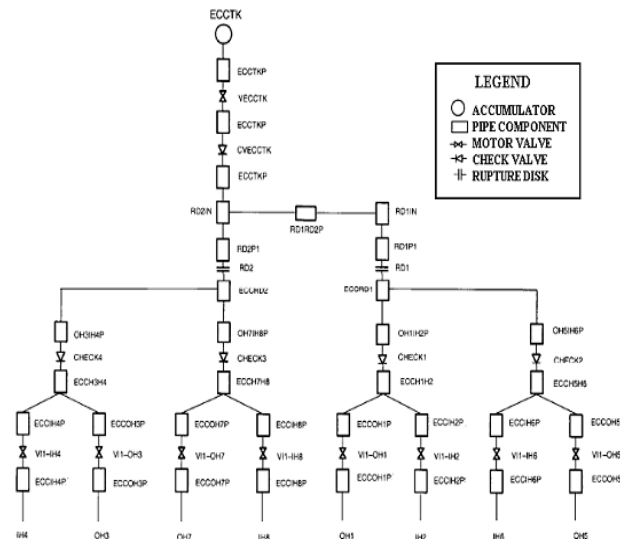


Figure 3 HPECC nodalization

2.3 Base case

Usually three break locations are analyzed in CANDU safety analysis: the reactor inlet header break, the pump suction pipe break and the reactor outlet header break. For each break location a survey to determine the representative stagnation break is completed. In this study the 35% reactor inlet header (RIH) break was chosen. This break size is identified in CANDU safety analysis as the break that produces the highest fuel sheath temperature.

The main assumptions used in this study are:

- All heat transport pumps are assumed to run until tripped by protective system (during the running time of transient the pumps are not tripped).
- Pressurizer is connected prior loop isolation.
- The D2O feed and bleed system is modeled and is assumed to function according to normal inventory control logic.
- Emergency core coolant injection is available.
- Feedwater control valve is available in order to control the steam generator level.

After the break is initiated, the broken inlet header pressure decreases rapidly, as shown in Figure 4. When the reactor pressure drops below the LOCA and loop isolation signal, the high-pressure ECC injection valves begin to open and the loop isolation valves begin to close. Twenty seconds after the loop isolation signal, the broken loop is isolated from the intact loop. Thirty seconds after LOCA, the steam generator crash cool begins (automatically opens of the steam discharge valves to depressurize at the maximum rate the secondary side). However, this is not of particular importance in a large break LOCA. When the broken loop pressure falls below the ECC injection pressure, ECC injection begins to the broken loop. Nevertheless, the presence of rupture discs on the injection line,

used to provide a positive interface between H₂O and D₂O, delay the ECC injection until a rupture differential pressure of 0.38 MPa is attained.

A short period of stagnation is observed soon after the accident in the channels downstream the break when the break force is balanced by the pump, Figure 5. Consequently, fuel sheath heats up as shown in Figure 6. After the stagnation is ended due a continuous change in pump head and break flow, the fuel sheath temperature decreases.

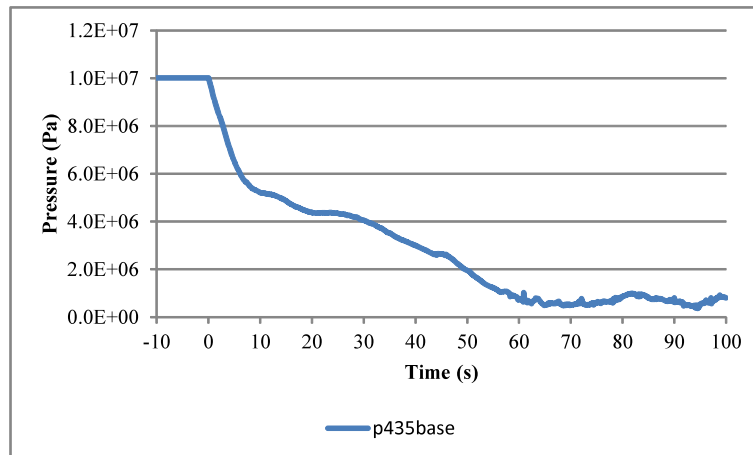


Figure 4 CANDU6 35% RIH break - Broken channel pressure

Figures 5 and 6 show the break mass flow and the maximum cladding temperature time evolution during the transient.

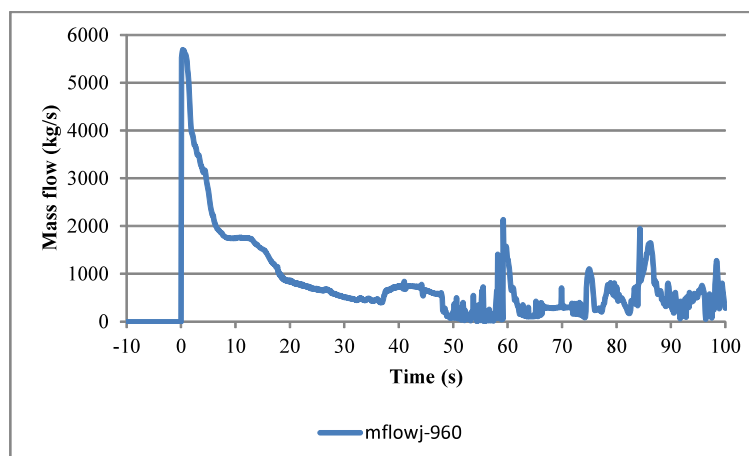


Figure 5 CANDU6 35% RIH break - Break mass flow

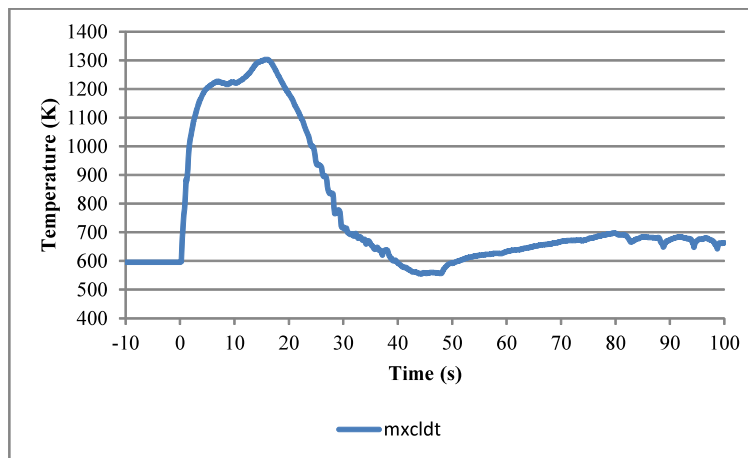


Figure 6 CANDU6 35% RIH break - Maximum cladding temperature

3. Application of the RELAP/SCDAPSIM/MOD4.0 uncertainty package to a CANDU6 35% RIH break.

The uncertainty analysis was performed at second order of Wilks' formula to derive the estimations of the 5th and 95th percentile with a 0.95 confidence level of the maximum cladding temperature reached during the 35% RIH break.

A total of 28 parameters were selected for the uncertainty analysis. They cover initial and boundary conditions, pressure drops, material properties and heat transfer correlations including critical heat flux.

Tables 1 and 2 list the selected input parameters with uncertainty associated for the 35% RIH break scenario. The tables provide the phenomena, the associated code parameter, the PDF type to describe the uncertainty and, when needed, additional comments to clarify how the uncertainty was applied. The parameters are classified as "input treatable parameters" (see Table 1) and "source correlation parameters" (see Table 2) depending upon their modification can be done from the regular input file or need for a source modification. The information required by the code for the first type is the input card and word number holding the base case value of the parameter, while the latter type requires the specification of a code given name (see column "code notation" in Table 2) that identifies the correlation coefficient to be modified. The uncertainty of the parameters was determined according to code documentation [10], phase 5 of BEMUSE programme [11], plant instrumentation information, and expert judgment when no related information could be found.

Table 1 List of "input treatable parameters" with uncertainty associated

INPUT TREATABLE PARAMETERS				
#	Phenomena	Parameter	Distribution ^{*1}	Comments
01	Power	Initial core power	UD	-
02		Power after scram	ND	Multiplier applied to the time-power table for the power after scram.
03		Peaking factor	UD	Multiplier applied to the central core nodes, which have the larger coefficients of the cosine power shape. The rest of the core nodes are multiplied by a normalization factor to keep the sum of the coefficients to 1.0.
04	Fuel	Thermal conductivity	ND	Temperature<2000K
05	thermal		ND	Temperature>2000K

06	behavior	Specific heat	ND	Temperature<1800K
07			ND	Temperature>1800K
08	Initial mass flow	Steady-state pump velocity	ND	-
09	Core	Form loss coefficients	UD	Multiplier applied to the junctions of the pipe modeling the core.
10	ECC ^{*2}	ECC friction form loss coefficients in the injection line	LD	Multiplier applied to the junctions of the volumes modeling the injection line.
11		Pressure	ND	Initial accumulator pressure
12		Temperature	ND	Initial accumulator temperature
13	Pressurizer	Friction form loss coefficients in the pressurizer line	LD	Multiplier applied to the junctions of the volumes modeling the pressurizer line to both channels.
14		Initial pressure	ND	-
15	Flow rate at the break	Subcooled discharge coefficient	UD	Multipliers applied to the subcooled and two-phase coefficients of the Ranson-Trapp choked flow model.
16		Two-phase discharge coefficient	UD	
17		Containment pressure	UD	Multiplier applied to the time-pressure table setting the containment conditions
1* LD stands for log-normal distribution, ND stands for normal distribution, UD stands for uniform distribution				
1** ECC stands for Emergency Core Cooling.				

Table 2 List of "source correlation parameters" with uncertainty associated

SOURCE CORRELATION PARAMETERS					
#	Phenomena	Parameter	Distribution ^{2*}	Code notation	Comments
01	Gap	Gap thermal conductivity	UD	GAP.CON.AT	Gap thermal conductivity computed from gap conductance model
02	Fuel heat transfer	Single phase liquid	UD	HTC.02.101	Heat transfer coefficient
03		Subcooled nucleate boiling	UD	HTC.03.101	Heat transfer coefficient
04		Saturated nucleate boiling	TD	HTC.04.101	Heat transfer coefficient
05		Subcooled transition boiling	TD	HTC.05.101	Heat transfer coefficient
06		Saturated transition boiling	TD	HTC.06.101	Heat transfer coefficient
07		Subcooled film boiling	TD	HTC.07.101	Heat transfer coefficient
08		Saturated film boiling	TD	HTC.08.101	Heat transfer coefficient
09		Single phase vapor	UD	HTC.09.101	Heat transfer coefficient
10		Condensation when void is less than one	UD	HTC.10.101	Heat transfer coefficient
11	CHF ^{2**}	CHF multiplier	LN	CHF.LT.GR	Groeneveld lookup table method
^{2*} LD stands for log-normal distribution, ND stands for normal distribution, UD stands for uniform distribution					
^{2**} CHF stands for Critical Heat Flux					

According to Wilks' formula, the *execution* phase consisted in 93 code runs. Out of the 93 code calculations three of them failed. However, the failure occurred near the end of the simulation (100 seconds) and, more important, after the maximum cladding temperature occurrence. Therefore, all 93 code runs could be included in the uncertainty evaluation of the maximum cladding temperature.

According to order statistics theory, the 95/95 unilateral tolerance limit is given by rank number 92, *i.e.* the second largest value, and covers the 95th percentile of the output quantity with a confidence level of 0.95. On the other hand the 5/95 unilateral tolerance limit is given by rank number 2, *i.e.* the second smallest value, and covers the 5th percentile of the output quantity with a confidence level of 0.95. Figure 7 depicts the time evolution of the maximum cladding temperature obtained in the base case, the upper and lower limit, and the highest and lowest ranks obtained after the 93 code runs.

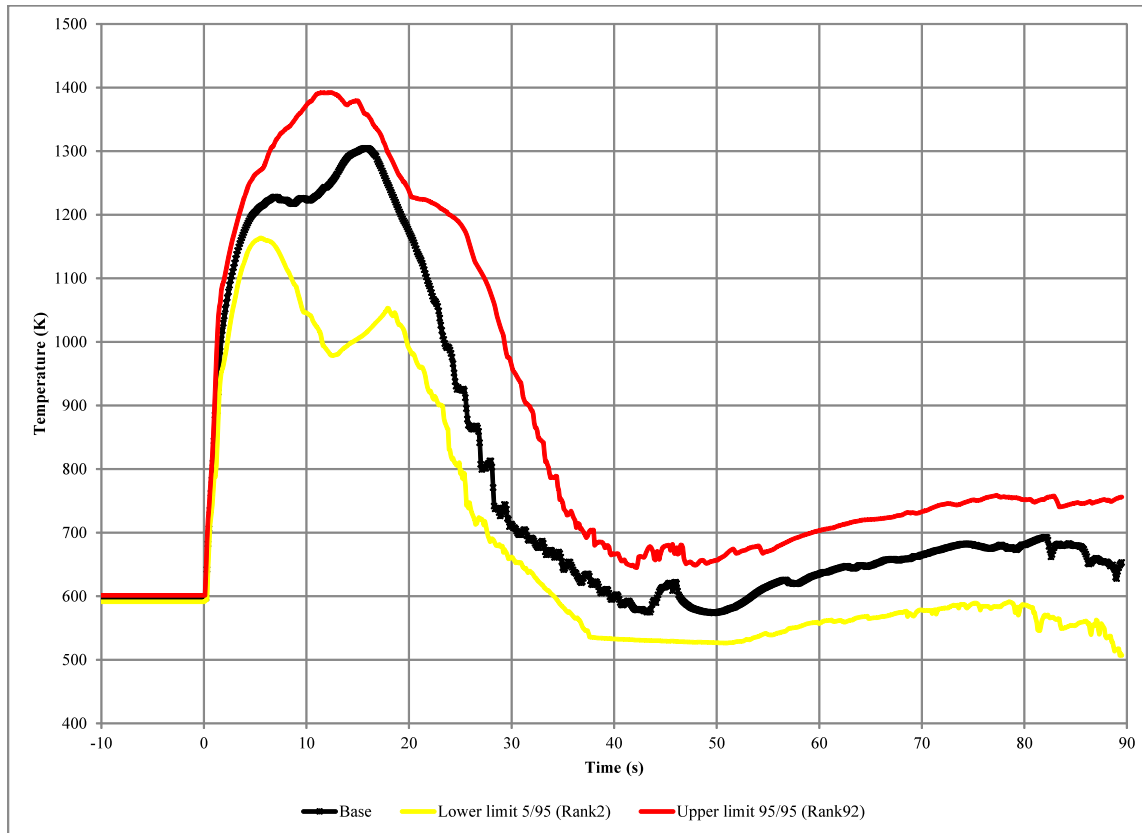


Figure 7 CANDU6 35% break uncertainty analysis - Maximum cladding temperature

In addition to the maximum cladding temperature, the primary pressure in the broken channel and the integral mass flow rate had been analysed (see Figure 8 and Figure 9, respectively).

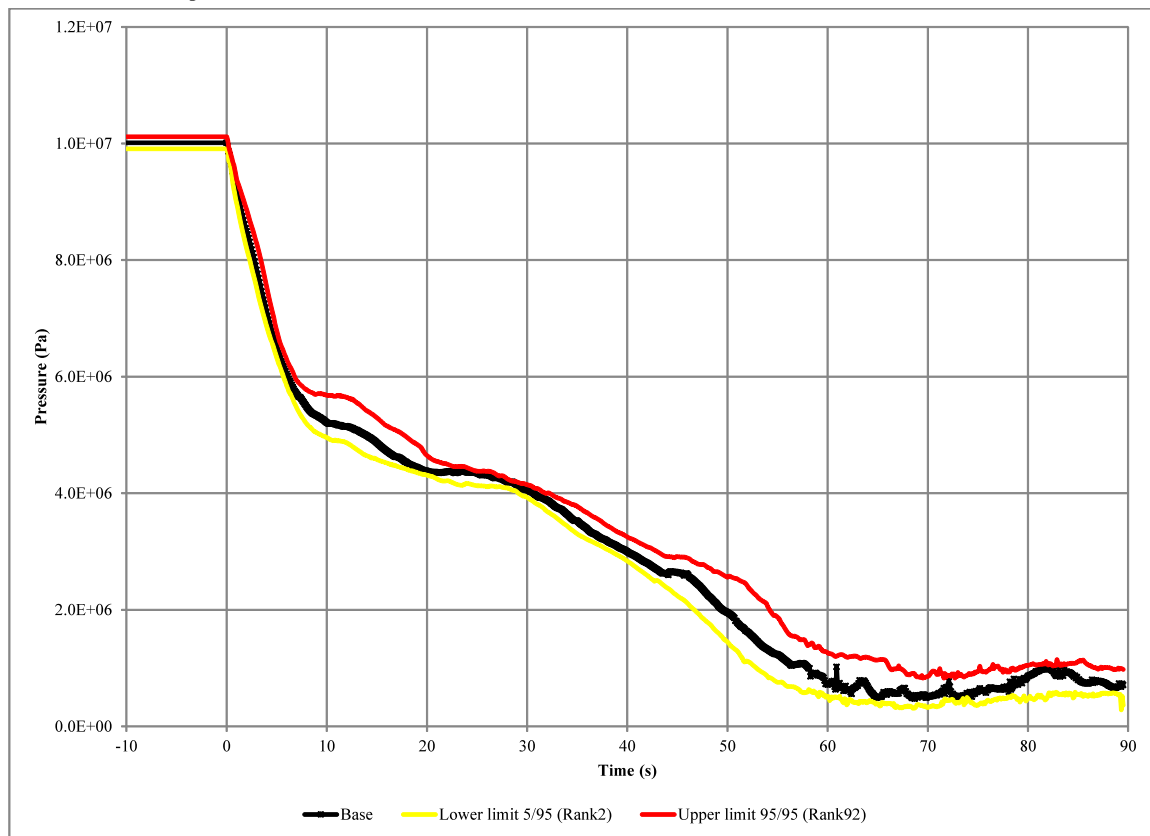


Figure 8 CANDU6 35% break uncertainty analysis - Pressure in the broken channel

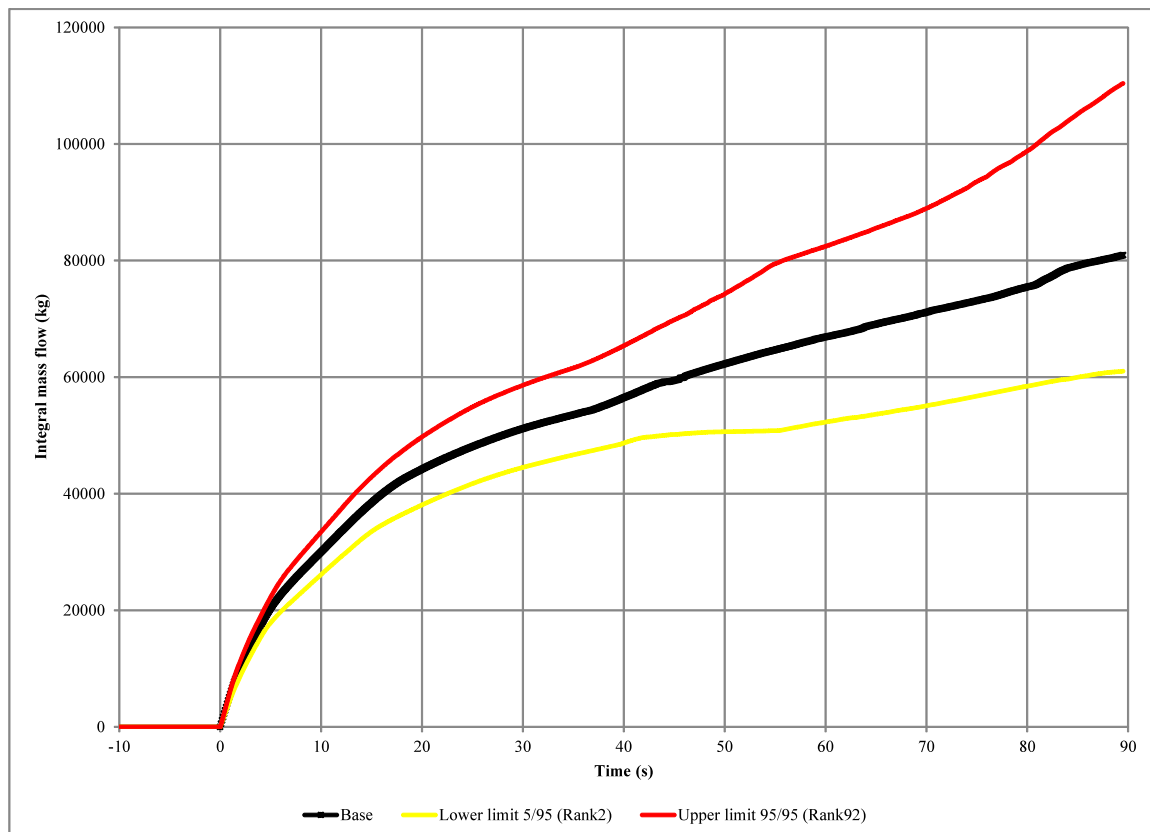


Figure 9 CANDU6 35% break uncertainty analysis - Integral break mass flow rate

4. Conclusions

A package to compute integrated uncertainty analyses has been developed and integrated to RELAP/SCDAPSIM/MOD4.0. The integrated tool allows performing different steps of the BEPU methodology minimizing data transfer and thus avoiding some error sources. The uncertainty packages have been applied successfully to perform analysis of Large Break LOCA scenario in CANDU6 plant.

The maximum sheath temperature computed has a value of 1392.05 K and is produced at 11.5 s after the break. This results is about 94 K higher than the maximum sheath temperature computed when assumption usually considered in LBLOCA analysis of CANDU reactor are used[4]. The order of magnitude of the uncertainty bands obtained for maximum sheath temperature and broken loop pressure are reasonable.

The results discussed in the present paper are preliminary. The purpose of this study was not to produce final results, but to demonstrate the capability of the integrated uncertainty analysis package of the RELAP5/SCDAPSIM/MOD4 code for the analysis of Large Break LOCA scenario in a CANDU6 plant.

5. References

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