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Original Article

Uncertainties impact on the major FOMs for severe accidents in CANDU 6 nuclear power plant

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ABSTRACT

In the nuclear safety studies, a new trend refers to the evaluation of uncertainties as a mandatory component of best-estimate safety analysis which is a modern and technically consistent approach being known as BEPU (Best Estimate Plus Uncertainty).

The major objectives of this study consist in performing a study of uncertainties/sensitivities of the major analysis results for a generic CANDU 6 Nuclear Power Plant during Station Blackout (SBO) progression to understand and characterize the sources of uncertainties and their effects on the key figure-of-merits (FOMs) predictions in severe accidents (SA).

The FOMs of interest are hydrogen mass generation and event timings such as the first fuel channel failure time, beginning of the core disassembly time, core collapse time and calandria vessel failure time. The outcomes of the study, will allow an improvement of capabilities and expertise to perform uncertainty and sensitivity analysis with severe accident codes for CANDU 6 Nuclear Power Plant.

1. Severe accidents in CANDU 6

The accidents with reactor core damage in CANDU reactors (which are Pressurized Heavy Water Reactors, PHWRs) are specifically known as Design Extension Conditions (DEC) as referenced by the IAEA[1], which include two approaches: a) accidents where the core geometry is preserved (fuel channels integrity is maintained, but fuel degradation may occur), so called Limited Core Damage Accidents (LCDA); and b) the severe accidents (SA) which may result from LCDA with loss of moderator during postulated initiating events as Loss of Cooling Accidents (LOCA) or Loss of Flow Accidents (LOFA) concomitant with the loss of multiple safety systems and support safety systems, which challenges the core integrity (fuel channels failure, core degradation and finally core collapse take place), so called Severe Core Damage Accidents (SCDA).

The SBO accident is initiated by a total loss of AC power both off-site and on-site power supply is lost, along with the unavailability of Standby Diesel Generators and Emergency Power Supply. The Mobile Diesel Generators (MDG) are not credited in the analysis. MDGs are normally designed as back-up power supplies for multiple nuclear power plants as post-Fulkushima Daichii accident improved measures (including also Cernavoda NPP).

2. Description of the severe accident code and model

2.1. Code identification and brief description

The RELAP/SCDAPSIM/MOD3.6 code was used to analyse the progression of the severe accident. RELAP/SCDAPSIM is a nuclear thermal-hydraulics and safety analysis tool [2], which was continuously developed through the SCDAP Development and Training Program (SDTP).

The thermal hydraulic models for the hydrodynamic systems are integrated in the RELAP5 part of the code. The code is based on a non-homogeneous and non-equilibrium model for the two-phase system that is solved by a fast, partially implicit numerical scheme.

Heat structures provided in RELAP5 permit calculation of the heat transferred across solid boundaries of hydrodynamic volumes. Modelling capabilities of heat structures are general and are assumed to be represented by one-dimensional heat conduction in rectangular, cylindrical or spherical geometry. The heat transfer model in RELAP5 is based on the boiling curve which is used to govern the selection of the heat transfer correlation.

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SCDAP code is used to describe the core components. Treatment of the core includes fuel rod heat-up, ballooning and rupture, fission product release, rapid oxidation, zircaloy melting, UO_2 dissolution, ZrO_2 breach, flow and freezing of molten fuel and cladding, and debris formation and behaviour. MOD 3.6 contains various improvements for CANDU core degradation model [3], which allows an improved confidence in prediction of HWRs important severe accident phenomena. These improvements consist of mechanistic deformation models for pressure tube and fuel channel including core disassembly and collapse model.

2.2. CANDU 6 plant model for SA

The nodalization scheme used for the Primary Heat Transport System (PHTS) is shown in Fig. 1. The core model consists of 4 hydrodynamic characteristic channels per pass. Each representative channel is subdivided into 12 control volumes corresponding to the 12 fuel bundles inside the channel.

The heat losses from the PHTS to the containment environment are modelled with RELAP heat structures with convective heat transfer as boundary condition.

Fig. 1 shows the nodalization scheme for the steam generators (SG) and main steam lines.

In a fuel channel, the 37 fuel elements of the fuel bundle, the pressure tube (PT) and the calandria tube (CT) are modelled using SCDAP core components, as represented in Fig. 2. All fuel rods which belong to the same fuel bundle are considered to have the same power. The PT, the CO_2 filling the annular space between PT and CT, and the CT are modelled using a shroud component with three layers of materials. The 12 bundles in a CANDU fuel channel were divided into 12 axial nodes to properly model the axial power distribution.

Fig. 3 shows the CV model. Calandria vessel is modelled as a pipe component with four sub-volumes having vertical orientation representing the moderator surrounding fuel channels. The four calandria pressure relief ducts are modelled as pipe components with three sub volumes having vertical orientation. Calandria over pressure rupture disk (OPRD) is modelled as a trip valve and connects CV with containment.

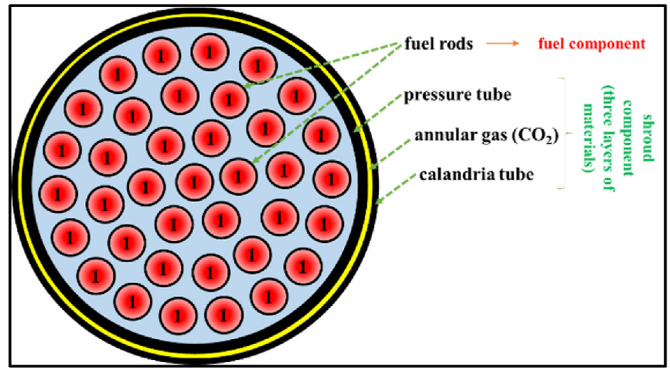

Fig. 2. Fuel channel model in SCDAP.

2.3. Failure criteria

The following failure criteria were used in the UPB analysis:

Fuel sheath failure. RELAP/SCDAPSIM/MOD3.6 calculates cladding ballooning and rupture using a mechanistic model to calculate the elastic-plastic deformation using the theory of Hill and the Prandtl-Reuss equations [4].

Fuel channel failure. The channel is assumed to have failed when a user input pressure tube failure strain was reached. For fuel channel with relatively uniform temperature distribution the value for the average creep strain of 20% for PT failure strain is considered [3].

Fuel channel disassembly. The MOD3.6 models the sagging of an entire fuel channel assembly. It is modelled as one beam with two fixed ends prior to channel contact with the lower assembly and after channel-to-channel contact, the interaction force between the two channels are considered [3].

Core collapse. A mechanistic criterion based on static beam loading calculation has been used [3]. When channels get in contact (due to the sagging of the upper channel and contact to the lower channel) the CT of the lower channel temperature increases as heat is transferred from the upper channel. The mechanistic core collapse criterion has been used, which is a function of unloaded length and ultimate tensile stress (UTS) of the CT.

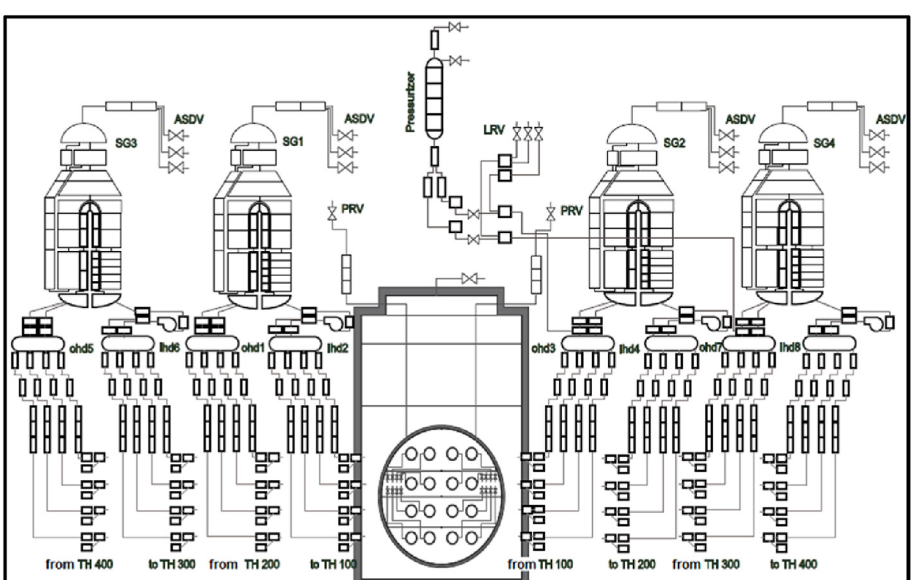

Fig. 1. RELAP5 PHTS nodalization.

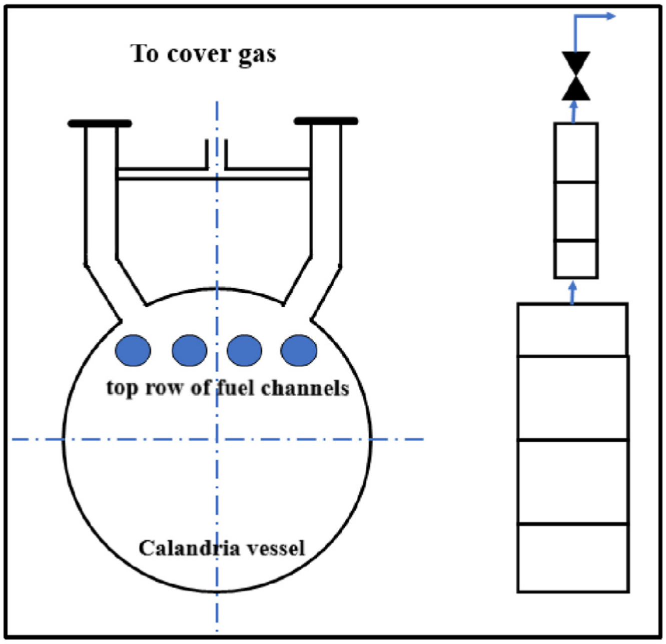

Fig. 3. Calandria vessel model.

2.4. Modelling assumptions

The modelling assumptions used for CANDU 6 SBO analysis are consistent with the assumption presented in Refs. [3,5] for the new mechanistic channel deformation models. Some of the key modelling assumptions are listed below.

- SBO accident is initiated at the beginning of the analysis, leading to the primary pumps trip, loss of moderator cooling, shield tank and end-shield cooling and the loss of Emergency Core Cooling (ECC);
- 4 channels/pass representing the core model, each channel dispaced at different vertical elevation (considering the specific CANDU6 lattice pitch). All fuel rods from the same fuel bundle produce the same amount of heat;
- the existing limits for the Zircaloy oxidation in RELAP/SCDAPSIM [6]: a) oxidation is terminated when the material is fully oxidized, b) oxidation is limited by the steam availability, c) oxidation is limited by the diffusion of water vapor;
- after fuel cladding rupture, in failed regions, the inside and outside of cladding oxidize at the same rates. Similar, after the PT and/or the CT is breached, inside and outside surfaces of the PT and the CT oxidize at the same rates;
- trip valves are used for the hydraulic connection between the fuel channel and moderator. They remain closed until fuel channel rupture occurs;
- bundle slumping and fuel relocation inside channel are not considered;
- the loads applied to the PT and to the fuel channel are assumed to be uniformly distributed;
- in the PT sagging model, the four garter springs are assumed to be evenly and located in the centre of the channel;
- PT-to-CT sagging contact and channel-to-channel contact is taken into account by assuming a constant contact area and a constant contact conductance applied to the location of contact;
- after channel failure the fuel bundles in the end stubs will fall out and be relocated to the CV bottom together with the suspended debris.

3. Uncertainty analysis methodology

3.1. Uncertainty propagation methodology

The uncertainty analysis methodology used in the uncertainty analysis performed by UPB was jointly developed by the Spanish regulatory body (CSN) and the Technical University of Catalonia (UPC), which is a statistical methodology where uncertainties are described through probability distribution functions (PDFs) and propagated through code runs by the variation of a set of input parameters. This methodology is based on the scheme proposed by the CSAU methodology, with three main features, as follows [7].

- “Requirements and code capabilities”, where the problem is defined referring to a specific nuclear power plant, the selected scenario, the prescriptive safety criteria, the selected computer code and its capabilities to perform the selected scenarios and related documentation;
- “Assessment and selection of the parameters” which consists in the assessment of the code capabilities to predict important processes and phenomena relevant to the scenario and to determine the uncertainty associated to code parameters;
- “Sensitivity and uncertainty analysis” consisting in setting up, executing and evaluating the uncertainty of the simulated scenario results.

As stated in Ref. [7], the input parameters are randomly sampled according to their PDFs simultaneously and a number of input samples are generated for the uncertainty calculations. The number of input samples is determined according to the Wilks' formula and it only depends on the percentile to be covered by the estimate and the confidence level of that estimate. The number of input parameters with uncertainty associated is independent of the number of code runs, thus there is no limitation on the number of input parameters.

As an example, the Wilks' formula establishes that the minimum number of code runs required to derive the one-sided 95/95 and 5/95 tolerance limits is 59 when the formula is applied to 1st order. To perform the uncertainty analysis of the SBO (in-vessel accident progression) for a CANDU 6 reactor the Wilk's formula of the 1st order was used. The 95/95 and 5/95 values indicate the 95th and 5th percentile, respectively, of the targeted output quantity with a confidence level of 95%.

3.2. Uncertainty analysis tool description

The RELAP/SCDAPSIM/MOD3.6 code has been selected for the uncertainty analysis of a Station BlackOut (SBO) accident in CANDU 6 reactor (in-vessel retention); the code benefits of a computational package to perform Integrated Uncertainty Analysis (IUA). The uncertainty evaluation capability of the IUA is implemented as an alternative run mode which allows the automatic execution of an

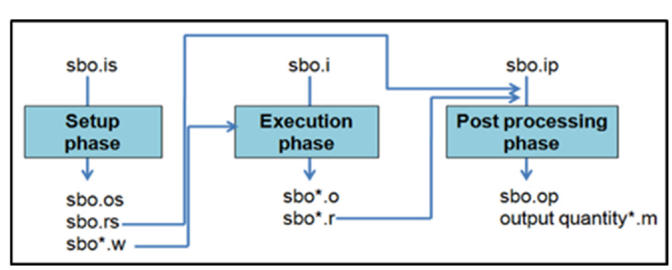

Fig. 4. Uncertainty analysis phases in RELAP/SCDAPSIM/MOD3.6(IUA).

uncertainty analysis based on the probabilistic approach [7]. A complete uncertainty analysis using RELAP/SCDAPSIM (IUA) code requires the execution of three related phases, as represented in Fig. 4.

The “setup” phase generates the total number of sampled values, also called “weights”, and information needed to build the tolerance bounds during the “post-processing” phase. The weights are used to associate uncertainty to code parameters by applying them as multipliers to the base values. During this phase the code also computes the required number of code runs.

The “simulation” phase consists of the base case run in which the simulation is done as if there were no uncertainty option available, and the set of uncertainty runs which have input and source modifications. Except for the base case run, each run of the simulation phase reads its corresponding weight file generated by the “setup” phase for that run. All simulation runs write information to be used in the “post-processing” phase into the plot records of the restart-plot file. The input data used for the simulation phase should be the same for the base run and each simulation run and is also the same used during the setup run.

The “post-processing” phase reads the restart-plot files written during the base case and the uncertainty runs and generates the rank matrices for the output quantities defined in the “post-processing” input file. The rank matrices contain the values for the output parameters sorted according to its rank and are used to determine the tolerance intervals.

3.3. Selected uncertainty parameters and associated references

For the uncertainty analysis 26 uncertain parameters were selected, as summarized in Table 1. From the 26 uncertain parameters, 9 parameters are referred as “source correlation parameters” (7th to 15th parameter from Table 1, which were applied to the heat transfer coefficients from the heat transfer correlations available in the code from wall structures to fluids, i.e. heat transfer from the steam generators U tubes to the secondary side water, heat transfer from the fuel bundles to the primary coolant), and the other 19 parameters are “input treatable parameters”, which allow the perturbation of the specific thermal hydraulic data for the core and different components, and also the fuel channels behavior during the core disassembly phase of the accident.

The sampling technique implemented in the code automatically generates vectors of sampled values. A vector contains one sampled value for each selected parameter and is saved in a data-file, basically coefficients for each uncertainty parameter based on the considered PDF (which is different for each vector). The code generates a vector file specific to each code run for the uncertainty analysis (59 code runs for one-sided statistical tolerance limits with a confidence level of 95%, independently of the number of uncertainty parameters and type of PDFs).

4. Sensitivity analysis methodology

4.1. Sensitivity analysis method description

The sensitivity analysis performed by UPB was based on two different correlations: Pearson's correlation coefficient and Spearman's correlation coefficient, both approaches being applied for the sensitivity study using the EXCEL specific functions.

Pearson's correlation coefficient methodology returns the Pearson Product Moment Correlation Coefficient [13], which is a dimensionless index that ranges from −1.0 to 1.0 inclusive and reflects the extent of a linear relationship between two data sets.

Spearman's correlation coefficient methodology can be defined as Pearson's correlation coefficient between the rank variables [14] rather than linear relationship between two parameters, as defined by the Pearson's correlation coefficient.

The influence of the uncertainty parameters based on the correlation coefficient is presented in Table 2, according to Ref. [15], states the strength of association of *r* coefficient (for absolute values).

4.2. Sensitivity analysis tool

The sensitivity analysis was performed using the PEARSON and SPEARMAN correlation coefficient functions available in EXCEL. The EXCEL tool allows the user to perform the sensitivity analysis for two set of data representing the set of a specific parameter coefficient from the uncertainty parameters list, resulted from the setup phase of the uncertainty analysis (extracted from the *sbo. os* file) and the set of values for the selected FOM (for example, H_2 mass generation for the in-vessel phase of the accident, resulted after the moderator depletion inside the CV).

5. SBO uncertainty analysis results

5.1. SBO uncertainty analysis results

The uncertainty analysis results from the 59 code runs (resulted from the Wilks' formula of 1st order of application) are summarized in Table 3.

The major FOMs selected as relevant from the uncertainty analysis are: H_2 mass generation inside the CV, events timing, such as: first fuel channel failure, beginning of core disassembly, core collapse, moderator depletion inside the CV.

The uncertainty analysis shows that the first fuel channel fails at 2.95−6.92 h from the initiating event, followed by the core disassembly beginning, which was estimated to occur in the time interval of 3.16−7.15 h for Loop 1, and 3.16−7.21 h for Loop 2 respectively. The estimated time interval for the beginning of core disassembly during the SBO is 3.16−7.21 h.

Continuous heat transfer from the hot fuel channels to the remaining moderator will lead to the CV level decrease (continuous moderator boiling), gradually uncovering next row of fuel channels. When the second row of fuel channels remains uncovered by the moderator, fuel channels heat-up and sag into contact with the lower channels located on the third row of equivalent fuel channels, transferring an important amount of heat to the channels below, leading to the formation of the suspended debris. Also, a sagged channel into contact with its lower channel will transfer its load to the supporting channel. The process continues up to the moment where the last row of fuel channels fails due to high temperatures of the CT (the melting point of the Zircaloy), or the longitudinal strain of a node exceeds a user-input specific value, or the channel sags more than a distance of two to three lattice pitches. The estimated time interval for the entire CANDU 6 core collapse resulted from the uncertainty analysis is 7.14−16.99 h from the initiating event.

The suspended hot debris is surrounded by the steam environment due to the continuous moderator boil-off. The environment condition is favorable to Zr-steam reactions. Once the suspended debris relocates to the bottom of the CV, it is quenched by the moderator and no longer contributes to the H_2 production. The total hydrogen produced immediately after the calandria vessel dryout is estimated at about 132−258 kg resulted from the uncertainty analysis, as depicted in Fig. 5.

The estimated timed interval of the moderator depletion (Fig. 6) is around 7.19−16.87 h, which shows that the moderator complete boil-off can occur slightly before the entire core disassembly moment.

Table 1
Uncertainty parameters for SBO analysis (in-vessel retention/CV dryout).

#	Phenomena	Parameter	PDFs	Comments
1	Fuel pellet behaviour [8]	Heat capacity when T ≤ 1800K	ND	Multiplier applied to the heat capacity when T ≤ 1800KMultiplier applied to the heat capacity when T > 1800KMultiplier applied to the thermal conductivity when T ≤ 1800KMultiplier applied to the thermal conductivity when T > 1800KMultiplier applied to the density when T ≤ 1800KMultiplier applied to the density when T > 1800K
2		Heat capacity when T > 1800K	ND	
3		UO_2 thermal conductivity when T ≤ 2000K	ND	
4		UO_2 thermal conductivity when T > 2000K	ND	
5		UO_2 density when T ≤ 1800K	ND	
6		UO_2 density when T > 1800K	ND	
7	Heat transfer from walls to fluids [8,9]	Heat transfer coefficient from HT source correlation for single phase liquid	ND	Multiplier applied to the HTC from HT source correlation when single phase liquid takes placeMultiplier applied to the HTC from HT source correlation when subcooled nucleate boiling takes placeMultiplier applied to the HTC from HT source correlation when saturated nucleate boiling takes placeMultiplier applied to the HTC from HT source correlation when subcooled transition boiling takes placeMultiplier applied to the HTC from HT source correlation when saturated transition boiling takes placeMultiplier applied to the HTC from HT source correlation when subcooled film boiling takes placeMultiplier applied to the HTC from HT source correlation when saturated film boiling takes placeMultiplier applied to the HTC from HT source correlation when single phase vapor takes placeMultiplier applied to the HTC from HT source correlation when condensation with void less than 1 takes place
8		Heat transfer coefficient from HT source correlation for subcooled nucleate boiling	UD	
9		Heat transfer coefficient from HT source correlation for saturated nucleate boiling	UD	
10		Heat transfer coefficient from HT source correlation for subcooled transition boiling	TD	
11		Heat transfer coefficient from HT source correlation for saturated transition boiling	TD	
12		Heat transfer coefficient from HT source correlation for subcooled film boiling	TD	
13		Heat transfer coefficient from HT source correlation for saturated film boiling	TD	
14		Heat transfer coefficient from HT source correlation for single phase vapor	UD	
15		Heat transfer coefficient from HT source correlation for condensation with void less than 1	UD	
16	Power after scram	Core thermal power	ND	Power scram, with best estimate value and the standard deviation calculated upon a variation of ±10% decay heat, according to [10,11].
17	Coolant pressure [8,9]	Form loss coefficients	UD	Multiplier applied to the junctions of the pipes modeling the core, calculated by the code through the uniform distribution coefficients resulted from minimum and maximum factors of variation of 1% (engineering judgement, due to lack of data). The assumption was made based on the cooolant pressure impact on the fuel channel failure time and accident progression.
18	Flow rate at LRVs	Discharge coefficient - subcooled	ND	Multiplier applied to discharge coefficient: subcooled discharge coefficient with best estimate value of 0.89 and standard deviation of 0.0349 as stated in [11]. Multipliers are calculated for each code run based on the distribution data introduced by the user, and randomly sampled by the code.
19		Discharge coefficient - two-phase	ND	Multiplier applied to discharge coefficient: two-phase discharge coefficient with best estimate value of 1.07 and standard deviation of 0.1189 as stated in [11]. Multipliers are calculated for each code run based on the distribution data introduced by the user, and randomly sampled by the code.
20	Moderator expulsion through CV rupture disks	Discharge coefficient - subcooled	ND	Multiplier applied to discharge coefficient: subcooled discharge coefficient with best estimate value of 0.89 and standard deviation of 0.0349 as stated in [11]. Multipliers are calculated for each code run based on the distribution data introduced by the user, and randomly sampled by the code.
21		Discharge coefficient - two-phase	ND	Multiplier applied to discharge coefficient: two-phase discharge coefficient with best estimate value of 1.07 and standard deviation of 0.1189 as stated in [11]. Multipliers are calculated for each code run based on the distribution data introduced by the user, and randomly sampled by the code.
22	Fuel channels behavior	PT failure strain	UD	Multiplier applied to the SCDAP shroud components: failure strain with best estimate value 22% and both minimum and maximum values of 19% and 24% respectively, as resulted from experiments referenced in [12]. For each code run from the uncertainty analysis, a specific multiplier for the failure strain to obtain a minimum value of 0.19 and a maximum value of 0.24 was derived from the sampling using a uniform distribution for this parameter. Since there are no information available on the failure strain spread of data, the uniform distribution was considered (engineering judgement).
23		Contact angle for channel-to-channel contact	UD	Multiplier applied to the SCDAP shroud components: contact angle for channel-to-channel contact with recommended best estimate value of 30° and minum and maximum values of 20° and 40° respectively. The multipliers are created automatically by the code sampling for each code run, thorugh the uniform distribution of the data with a mean value of 1.0 (normalization of the base case value to the best estimate value) and minimum and maximum factor of ±33% (engineering judgement on the use of PDF duel to lack of information).
24		Contact conductance for channel-to-channel contact heat transfer calculation	LN	Best estimate value recomended for SCDAP shroud components for the contact conductance of 5000 W/m^{2}K, according to [3]. Values considered between 2800 - 11000 W//m^{2}K. The maximum limit of 11 kW/m^{2}K predicts accurately the heat transfer calculated by MOD3.6 version of the code used in this analysis in comparison with experiments, as stated in reference [3]. The increased value of the contact conductance is not recommended due to the critical heat flux overprediction by MOD3.6. Lower values were considered for the contact conductance that indicated good agreement on heat transfer predictions of the code with experimental data, as stated in [3]. The log-normal distribution used here is based upon engineering judgment, which considers that the lower values of the contact conductance higher probability of occurrence, which supports the higher fidelity for the heat transfer prediction of MOD3.6 than the contact conductance higher values.

Table 1 (*continued*)

#	Phenomena	Parameter	PDFs	Comments
25		Contact angle for PT-CT sagging contact	UD	Recommended best estimate value of 20° according to [3], and minum and maximum values of 10° and 30° respectively. The multipliers are created automatically by the code sampling for each code run, thorugh the uniform distribution of the data with a mean value of 1.0 (normalization of the base case value to the best estimate value) and minimum and maximum factor of ±50% (engineering judgement on the use of PDF duel to lack of information).
26		Channel rupture area	UD	The best-estimate value is the flow area of an individual fuel channel. Multiplier applied to the valves modelling channel rupture to derive the opening area between 50% and 100% based upon a uniform distribution of the data. Flow area of an individual fuel channel is assumed 0.003475 m^2 as used in reference [5]. Since the hydrodinamic model of the fuel channels involves grouping of CANDU6 380 channels into 16 equivalent flow channels in the input model, the fuel channel failure mai result in a single core channel or multiple core channels (depending on the power group, whether is a high, medium or low power channel failure).

Table 2
Guidelines to interpreting correlation coefficient.

Strength of association	Coefficient, ***r***	
	Positive	Negative
No influence ("very weak")	0.00 to 0.19	−0.19 to 0.00
Low influence ("weak")	0.20 to 0.39	−0.39 to −0.20
Moderate influence ("moderate")	0.40 to 0.59	−0.59 to −0.40
High influence ("strong")	0.60 to 0.79	−0.79 to −0.60
Very high influence ("very strong")	0.80 to 1.00	−1.00 to −0.80

Table 3
SBO uncertainty analysis results.

Events	Timing (hrs)
Class IV and Class III Power loss	0
Reactor shutdown	0
CV bleed valves open	0.870–0.915
LRVs first opening	1.53–3.89
SG secondary side is dry Loops 1&2	1.40–3.45
Pressurizer empty	2.07–5.04
At least one channel is dry	1.95–6.01
PT and CT are ruptured	**2.95–6.92**
CV rupture disks #1–4 open	2.88–6.51
Beginning of the core disassembly	**3.16–7.21**
Core collapse	**7.14–16.99**
Water is depleted inside CV	**7.19–16.87**
Calandria vessel failed	43.75–61.72
H_2 mass production (kg)	**132.0–258.0 kg**

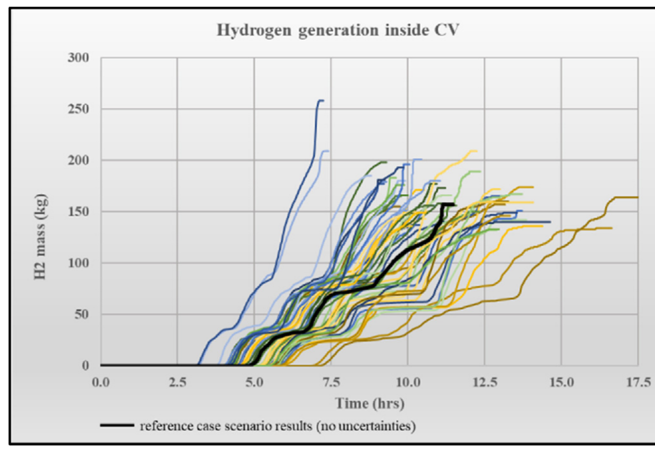

Fig. 5. H_2 mass production (in-vessel).

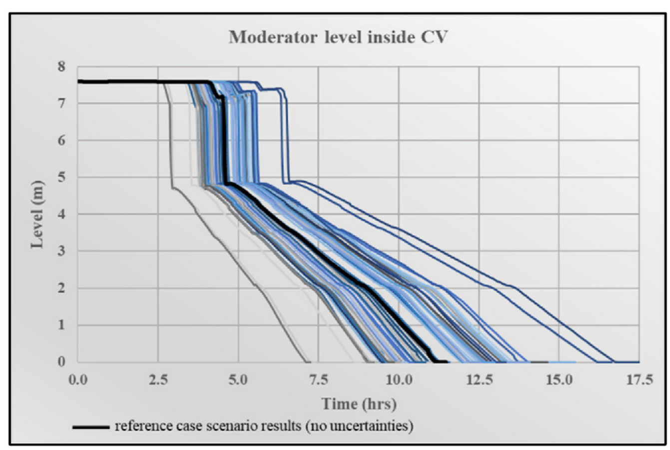

Fig. 6. Moderator level (depletion).

5.2. Sensitivity analysis

The sensitivity analysis was performed using EXCEL available functions to evaluate the influence of the selected uncertainty parameters on the major selected FOMs.

For the in-vessel H_2 mass generation, the entire list of 26 uncertainty parameters was considered as being influential, and the correlation coefficients obtained are graphically represented in Fig. 7.

Regarding the sensitivity analysis for the predicted time of the first fuel channel failure, from the 26 uncertain parameters, only 22 parameters were considered as being influential (fuel behavior specific parameters, heat transfer from heat structures to fluids, core power after accident initiation, core pressure loss, flow rate at LRVs, flow rate at CV rupture disks and the pressure tube failure strain). The Pearson's and Spearman's correlation coefficients for each uncertain parameter with the selected FOM are represented in Fig. 8.

A sensitivity analysis was performed for the timing of core disassembly phase of the accident (beginning of core disassembly time and end of core disassembly/core collapse considering the influence of the first fuel channel failure time, as resulted from the uncertainty analysis based on parameters 22–26 (according to Table 1), and the fuel channel rupture area. Fig. 9 represents the influence of the uncertain parameters on the beginning of core disassembly (CD), and the specific impact on the core collapse timing is depicted in Fig. 10. To be noted that the fuel channel failure time is almost perfectly correlated with the beginning of core disassembly and the core collapse timings.

The mean value of the hydrogen mass generated is 164.64 kg with a standard deviation of 22.91 kg. The uncertainty analysis

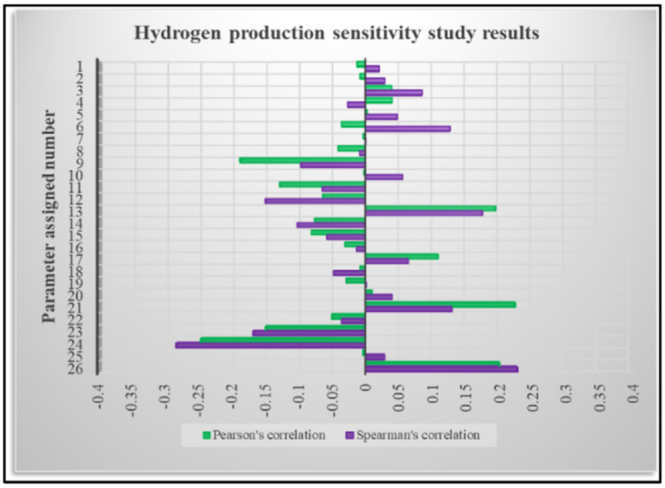

Fig. 7. Sensitivity analysis on the H_2 mass generation.

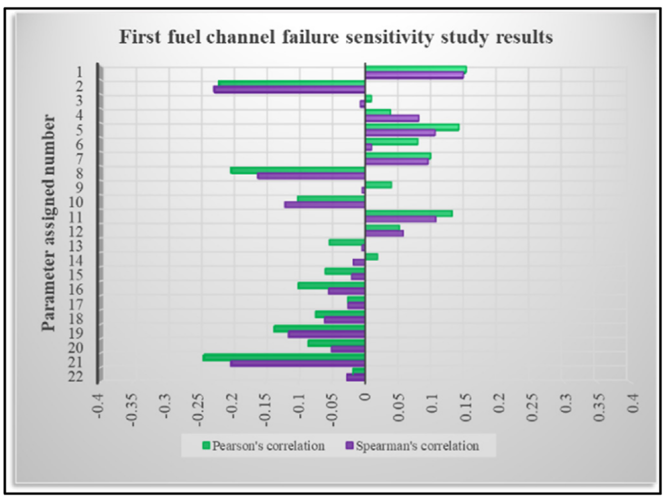

Fig. 8. Sensitivity analysis on the first fuel channel failure time.

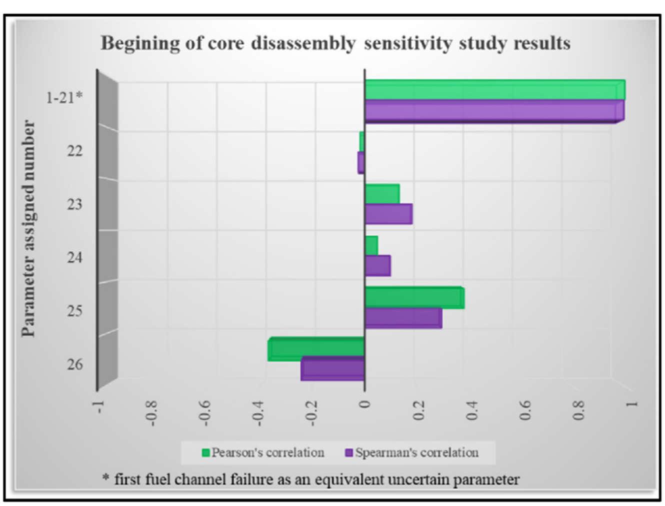

Fig. 9. Sensitivity analysis on the beginning of core disassembly timing.

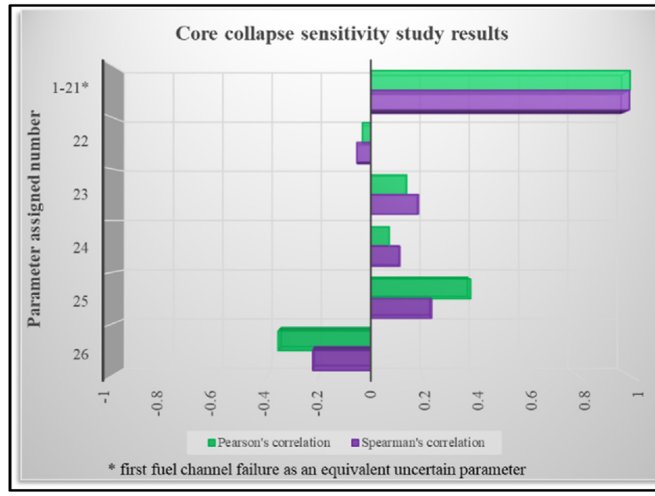

Fig. 10. Sensitivity analysis on the core collapse timing.

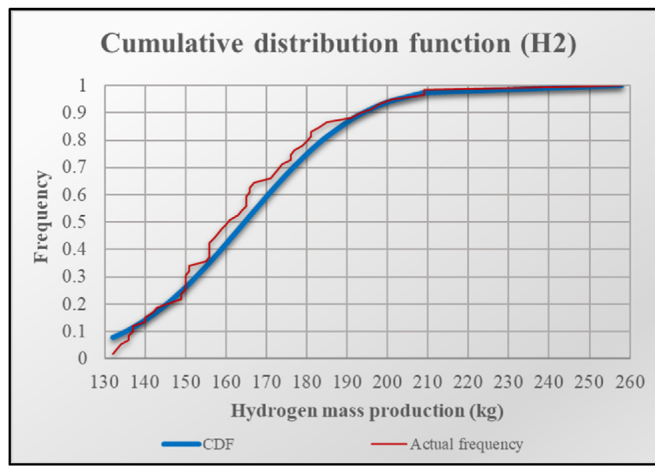

Fig. 11. CDF for in-vessel H_2 mass generation.

shows that the total hydrogen production during in-vessel phase is estimated between 132 and 258 kg. Fig. 11 shows the actual frequency of estimated hydrogen mass generated in the core, as resulted from the uncertainty analysis of the SBO, and cumulative distribution function (CDF) for the hydrogen mass generation during the in-vessel phase of the SBO.

6. Summary and conclusions

Based on the guidelines for interpreting the Pearson's correlation coefficients presented in Table 2 the sensitivity analysis results, also extended to the Spearman's correlation coefficients, showed that.

- for the H_2 mass generation resulted from the uncertainty analysis performed for a SBO accident (in-vessel) in CANDU 6, none of the uncertain parameters can be assumed as having moderate or high influence since the correlation coefficients obtained for both Pearson's and Spearman's calculation do not exceed the absolute value of 0.39. Thus, there are 3 uncertainty parameters showing low influence on the H_2 mass production as follows: the two-phase moderator expulsion through CV rupture disks, the contact conductance for channel-to-channel contact heat transfer calculation, and the fuel channel rupture area;

- regarding the first fuel channel failure time estimated in the uncertainty analysis 3 uncertainty parameters showed low influence: the fuel heat capacity when temperature exceeds 1800K, the heat transfer coefficient (from the source code correlation) when subcooled nucleate boiling occurs, and the two-phase moderator expulsion through CV rupture disks;
- the sensitivity on the beginning of core disassembly time based on the 5 uncertainty parameters considered as being influential (first fuel channel failure time represented thorugh th 1st to 21st uncertain parameters, and the other five specific core disassembly phase parameters) showed a perfect correlation with the fuel channel failure time and low influence of the contact angle for PT-CT sagging contact and fuel channel rupture area;
- the sensitivity on the core collapse time based on same uncertain parameters similar to the beginning of core disassembly time, resulted in similar trends (referring to the impact of the considered uncertainty parameters) as the beginning of core disassembly timing;

The main conclusion on the results from the sensitivity analysis of the selected FOMs and the uncertainty parameters listed in Table 1 is that the parameters showing low influence on the FOMs should not be neglected since the major FOMs can be strongly impacted by the relevant combination of the entire list of uncertain parameters, nor only by a single parameter.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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