BEPU ANALYSIS OF A STATION BLACKOUT IN CANDU REACTORS WITH RELAP/SCDAPSIM

R. M. Nistor-Vlad, D. Dupleac, I. Prisecaru
Politehnica University of Bucharest
Splaiul Independentei 313, Bucharest 060042, Romania
roxanamihaelanistorvlad@gmail.com; danieldu@cne.pub.ro; prisec@gmail.com

C. M. Allison, M. Perez-Ferragut, J. K. Hohorst, A. Stefanova
Innovative Systems Software
3585 Briar Creek Ln, ID 83406, Ammon, Idaho, USA
iss@srv.net, marinotka@gmail.com, jkh@srv.net, ant_stefanova@abv.bg

ABSTRACT

The worst nuclear accident after Cernobil (in august 1986), the Fukushima Daichii accident from March 11^{th}, 2011 has increased nuclear experts interest, all over the world, on severe accident management measures. Even though CANDU reactors are different from BWRs, the effect of an unmitigated SBO (Station BlackOut) accident could turn into a real challenge for the plant's components and systems, but also for the environment and population.

The purpose of this study is to determine the efficiency of severe accident management measure to be implemented, such as depressurizing the steam generators secondary side follow by the water injection from the dousing tank that could prevent the fuel channel failure during the early phase of a Station BlackOut analysis for a CANDU 6 reactor, when uncertainties are considered.

A SBO accident in CANDU reactors is initiated by a total loss of off-site AC power concomitant with the turbine trip and the unavailability of the Class IV and the backup power (loss of all on-site standby and electric power supplies, the diesel generators). The present paper includes BEPU (Best Estimate Plus Uncertainty) approach of a SBO using RELAP/SCDAPSIM best estimate tool with the integrated uncertainty package implemented in the code considering measures to mitigate the severe accident progression. Current trends refer to the BEPU (Best Estimate Plus Uncertainty) approach in the safety analysis of nuclear reactors. BEPU is a modern and technically consistent approach has been built upon best estimate methods including an evaluation of the uncertainty in the calculated results.

RELAP/SCDAPSIM is a best-estimate nuclear tool designed to analyze the behavior of reactor systems during normal and accident conditions. Three main versions of RELAP/SCDAPSIM are currently used by program members and licensed users to support a variety of activities. RELAP/SCDAPSIM/MOD3.4 is the version of the code used by licensed users and program members for critical applications such as research reactors and nuclear power plant applications. Even though the code was initially designed for LWRs (Light Water Reactors), Politehnica University of Bucharest demonstrated the applicability of the RELAP/SCDAPSIM code for CANDU (CANada Deuterium Uranium) reactors analyses, by simulating some of the most important postulated accident transients.

A complete uncertainty analysis using RELAP/SCDAPSIM/MOD3.4 code requires the execution of three related phases, namely the "setup" phase, the "simulation" phase consisting of several executions, and the

"post-processing" phase. The uncertainty data has to be supplied for the two types of parameters, the "input treatable" and the "source correlation" quantities. The required information is the probability distribution function and its characteristic parameters.

KEYWORDS
Severe accident, CANDU, RELAP/SCDAPSIM, Best estimate, Uncertainty

1. INTRODUCTION

One of the general attributes of a methodology to perform accident analysis of a nuclear power plant for the safety assessment is directly connected with the availability of qualified tools and analytical procedures suitable for this purpose. A modern and technically consistent approach has been built upon best estimate methods including an evaluation of the uncertainty in the calculated results (Best Estimate Plus Uncertainties approach) [1], [2]. The RELAP/SCDAPSIM code, designed to predict the behavior of reactor systems during normal and accident conditions, is being developed as part of the international SCDAP Development and Training Program (SDTP). Three main versions of RELAP/SCDAPSIM are currently used by program members and licensed users to support a variety of activities, including the CANada Deuterium Uranium (CANDU) reactor analysis. The version of the RELAP5/SCDAPSIM/MOD3.4 code used in this work is one developed by Innovate System Software (ISS) as part of the international SCDAP Development and Training Program (SDTP) for best-estimate analysis to model reactor transients including severe accident phenomena.

The Innovative Systems Software (USA) and Polytechnic University of Catalonia (UPC) joint efforts to develop and implement an uncertainty package to RELAP/SCDAPSIM code, which was designed to predict the behavior of reactor systems during normal and accident conditions [3]. The uncertainty package was initially implemented in RELAP/SCDAPSIM/MOD4.0 through the work as part of the international SCDAP Development and Training Program (SDTP). The SDTP is a cooperative program, including more than 60 organizations from 27 countries. RELAP/SCDAPSIM/MOD4.0, which is the first version of RELAP5 completely rewritten to Fortran 90 standards, uses publicly available RELAP3.3 and SCDAP models, developed by the US Nuclear Regulatory Commission (US NRC), in combination with [4]: (a) advanced programming and numerical techniques, (b) advanced SDTP-member-developed models for LWRs (Light Water Reactors), HWRs (Heavy Water Reactors), and research reactor analysis, and (c) a variety of other member developed computational packages.

Recently the uncertainty package was fully introduced in the MOD3.4 version of the code, which is the current production version and is designed specifically for “faster-than-real-time” simulations on typical Windows or LINUX PCs. The designation MOD3.4 is used to indicate that additional modeling options have been included relative to the original US NRC codes. These modeling options [3] include improved models for a detailed fuel rod, an electrically heated fuel rod simulator, and other SCDAP core components. Other modeling improvements include new models and correlations for air ingression transients and alternative fluids and cladding materials. Continued improvements in the coding and numerics also allow both MOD3.4 and MOD4.0 to run a wider variety of transients including low-pressure transients with the presence of noncondensable gases such as those occurring during mid-loop operations in LWRs, in pool type reactors, or in spent fuel storage facilities.

2. CANDU 6 CHALLENGES

Te unique features of the PHWRs (Pressurized Heavy Water Reactors) are the primary heat transport system and the moderator system, which are completely separated unlike the PWRs/BWRs (Pressurized Heavy Water Reactors / Boiling Water Reactors). The moderator system contains heavy water in the calandria vessel, at low pressure and low temperature (p_{mod}=0.1 MPa and T_{mod}=70°C) compared to the heavy water circulated through the fuel channels from the primary system, which is a high pressure and high temperature coolant ($p_{coolant}$=10 MPa, $T_{coolant}$=266-312°C). The moderator from the calandria vessel surrounds the fuel channels and serves as an alternative heat removal source in case of a severe accident scenario [5]. Figure 1 shows a representative of the both PHTS and the moderator system. The PHTS circulates pressurized heavy water coolant through the fuel channels to remove the heat produced by fission in the nuclear fuel. The D_2O coolant transports the heat to the steam generators, where it is transferred through the U tubes to light water from the secondary side of the SGs to produce steam. Ina CANDU 6 reactor the primary heat transport system has two loops interconnected through the pressurizer, designed to control the pressure. Each loop contains half of the total number of fuel channels of a CANDU 6 reactor, the additional feeders connecting the inlets and outlets of the fuel channels, 2 inlet headers collecting the cold heavy water from the SGs and 2 outlet headers collecting the hot heavy water from the core, 2 primary pumps operating in series, and 2 SGs. The coolant flow in adjacent channels is in opposite directions [6].

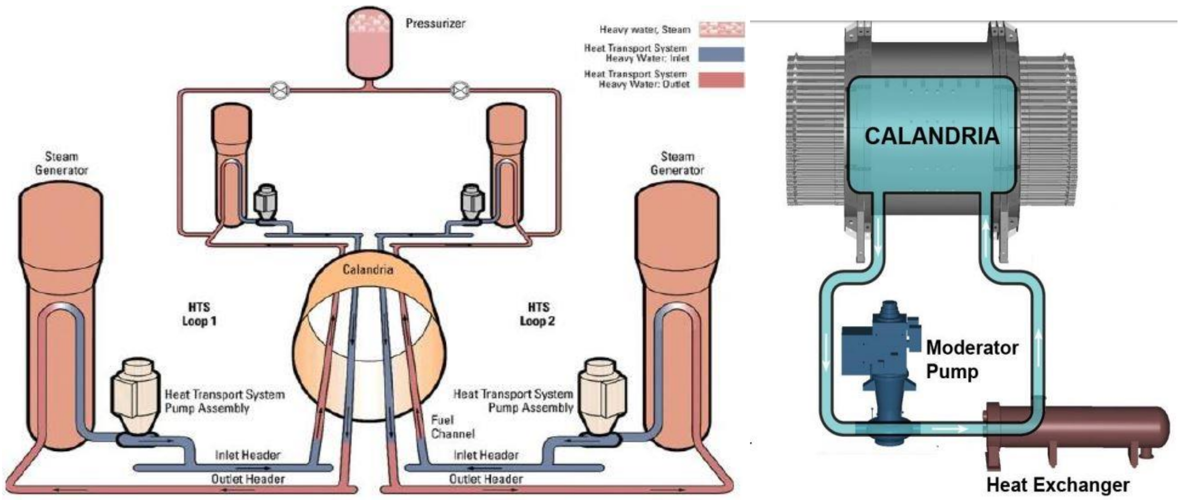


Figure 1. PHTS and Moderator System [5]

Since the RELAP/SCDAPSIM code was initially designed to analyse the behaviour of LWRs, Politehnica University of Bucharest started the investigation of the applicability of the code to CANDU 6 reactors analysis. Over the past years, experts from Politehnica University of Bucharest have developed a full plant model of a generic CANDU 6 reactor in RELAP/SCDAPSIM (as is shown in Figure 2), that being used to run the several calculations in order to demonstrate that the code adequately predict the behaviour of such a reactor type [7], [8].

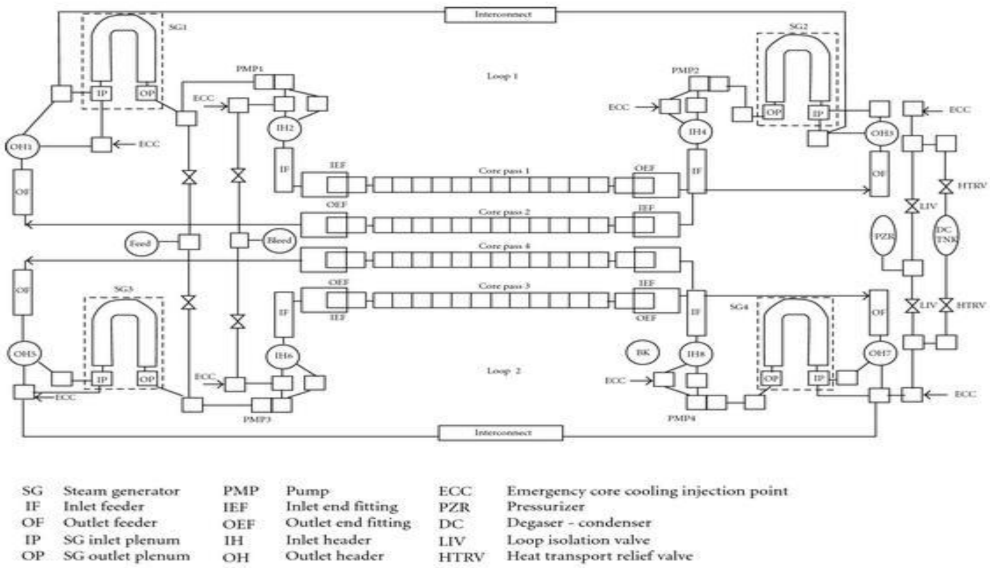


Figure 2. Generic CANDU 6 plant nodalization in RELAP5

RELAP/SCDAPSIM/MOD3.4 has been used at Politehnica University of Bucharest, Institute for Nuclear Research, and National Commission for Nuclear Activities Control in Romania to analyze a variety of transients in CANDU reactors, including: LOCA (Loss of Coolant Accident) transients [9], [10], [11], including the effect of different sizes of the break, the choked flow model employed, the emergency core cooling (ECC) performance, and the core nodalization during a reactor inlet header break (different sizes of the break, including a 35% inlet header break which was expected to produce the highest peak fuel cladding temperatures among all postulated break sizes, the results reported in [9] were compared with the ones from CATHENA code, a CANDU specific code, which have proved to be in good agreement), a 100% reactor outlet header break which had the highest potential for fuel failure and release of radioactivity, severe accident analysis, like a SBO (Station BlackOut) transient [12], or a SBO with voluntary depressurization of the PHTS (Primary Heat Transport System) [13], steam generators depressurization and water injection [14], [15], moderator drainage [16].

The RELAP/SCDAPSIM code was also used in the Coordinated Research Project (CRP) from the International Atomic Energy Agency for the "Benchmarking severe accident computer codes for heavy water reactor applications" [17]. The CRP scope included the identification and selection of a severe accident sequence (a selected SBO scenario), selection of appropriate geometrical and boundary conditions, conduct of benchmark analyses, comparison of all simulation results, evaluation of the computer code capabilities to predict important severe accident phenomena and core damage progression, and the proposal of code improvements and/or new experiments to reduce uncertainties. For the purpose of the present paper the detailed nodalization of the core was used (16 fuel channels describing the core) and a calandria vessel model was developed (as described in Figure 3), based on the input deck used in the the CRP study.

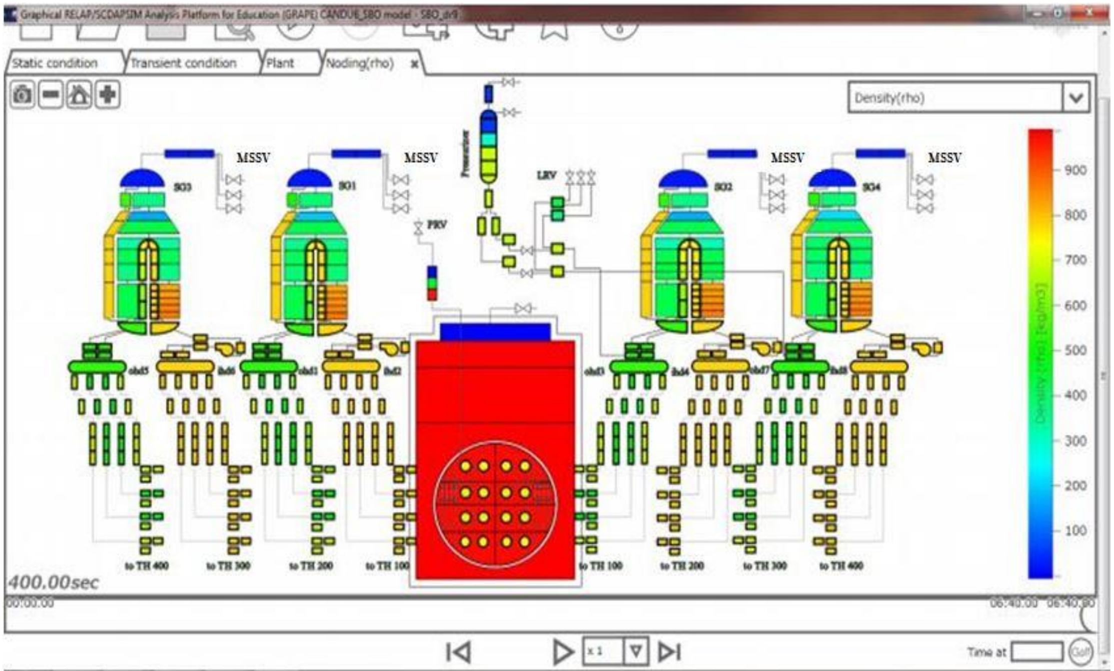


Figure 3. CANDU 6 model for the SBO accident analysis[1] [18]

3. UNCERTAINTY ANALYSIS OF A SBO WITH RELAP/SCDAPSIM(IUA)

3.1. Sensitivity Analysis of the SGs Nodalization during a SBO Transient

The steam generators play an essential safety function during the SBO accident scenario, serving as the primary function of removing the heat from the primary system by transferring the heat from the primary coolant (D_2O) circulated through the "U" tubes to the secondary side containing H_2O as a coolant. After the reactor trip, the decay heat from the fission products consists of about 7% of full power heat production. The most common method of removing the decay heat is through the steam generators. The heat removal function will be ensured by maintaining a minimum water inventory in the steam generators.

The main components of the steam generators modelled in RELAP/SCDAPSIM are: the hot and cold legs of the "U" tubes, the primary coolant inlet and outlet, the preheater section, the cyclone separator, the steam dome, the downcomer, the riser, and the shrouds (describing the walls of the pipes used). Most of the steam generators components were modelled using pipe components, single volumes, a separator, and single junctions describing the connections between the components. The proposed nodalizations of the SGs is hereby presented for 4 cases (Figure 4), and each case consists of:

- Case a) - the "U" tubes are discretized in 15 volumes, and separate volumes (single volumes) for the inlet and outlet plenum. The secondary side consists of the preheater, riser, separator, drum and downcomer volumes discretized in different number of nodes to properly model the steam generator performances. The feed water is injected in the preheater, a time-dependent volume

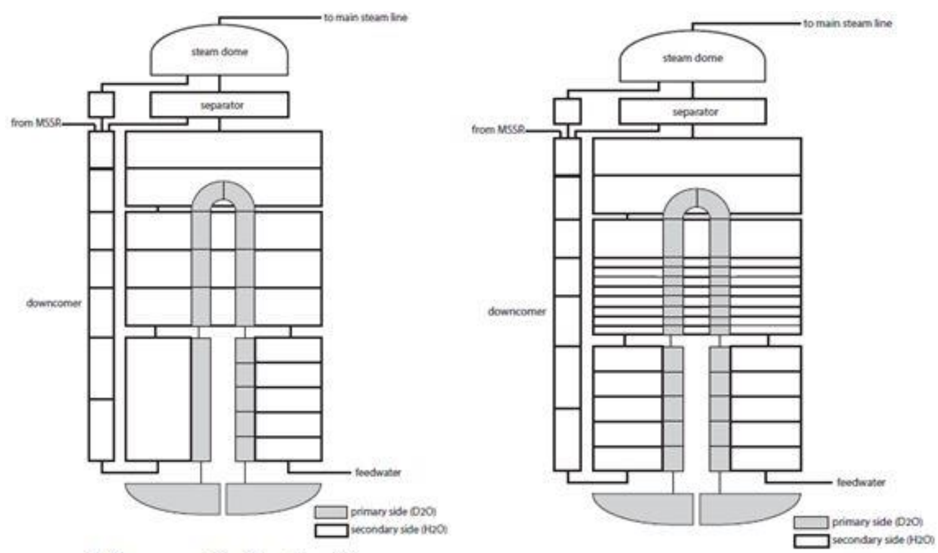
[1] Figure 3 is a result of the graphical display which was developed by the author using GRAPE (Graphical RELAP/SCDAPSIM Analysis Platform for Education and Engineering) [18] based on the input deck for the CANDU 6 SBO developed at UPB.

gives the temperature and a time-dependent junction is used as a boundary condition for the feed water flow rate.

- Case b) - with the variation of the number of volumes of the steam generator secondary side components (Figure 4) from 3 volumes of the riser to 9 volumes, and the preheater section from one volume to 5 volumes. The primary side of the steam generator was also divided in multiple nodes corresponding to the detailed nodding used on the secondary side.

- Case c) - starts from the model used in the base case analysis, with the variation of the number of volumes used to describe the steam generator secondary side components (Figure 5), from 3 volumes of the riser to 12 volumes, and the preheater section from one volume to 10 volumes. The primary side of the steam generator has been modelled based on the nodalization used for the secondary side.

- Case d) - a single volume describing the secondary side of the steam generator, the downcomer's nodding being maintained as it was in the previous proposed nodding schemes.



a) Base case Noding (used in the CRP)

b) SG Detailed Noding Scheme 1

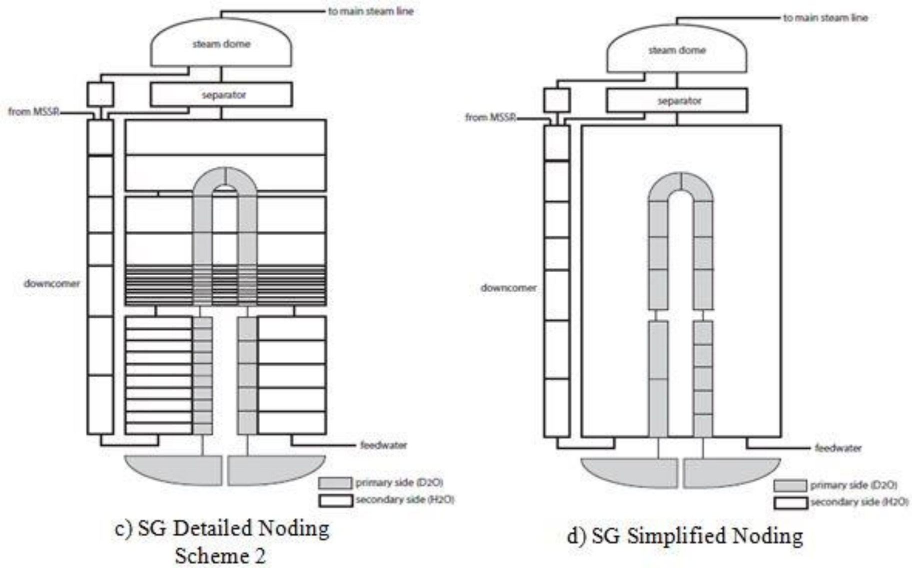


Figure 4. Proposed nodalization for the SGs in a SBO analysis

The heat transfer from the PHTS to the steam generators causes the water boil-off and the increase of the pressure in steam generators secondary side. When the steam generators secondary pressure reaches the set point for the opening of MSSVs, the steam is discharged from the secondary side to the environment outside the containment.

Following SBO primary pumps trip and the flow rate through reactor core decreases rapidly to the level of natural circulation. The SGs mass inventories and therefore the water level in the steam generators continuously decrease as a result of boil-off. When the SGs secondary side inventories are depleted, the SGs are no longer a heat sink to remove heat from PHTS and the natural circulation in PHTS is ceased.

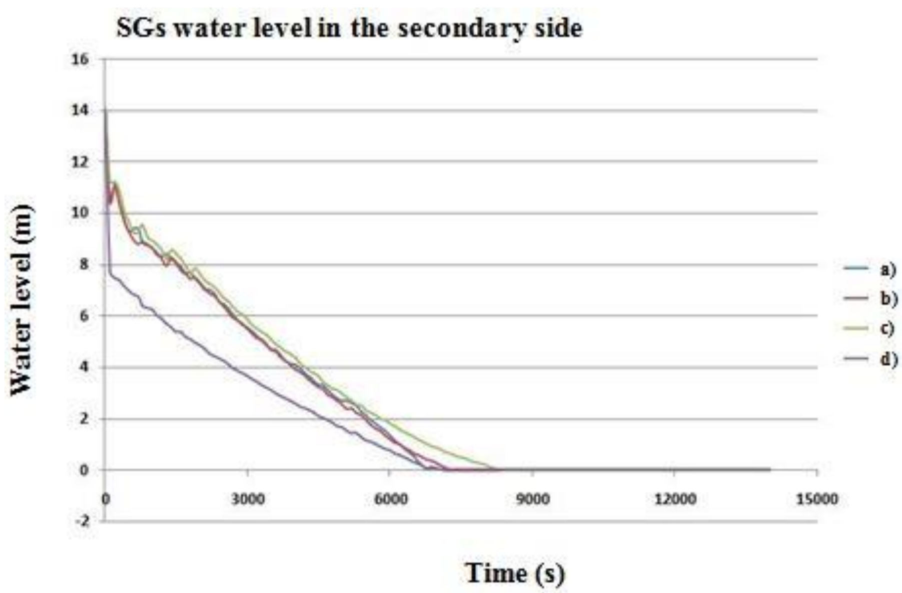


Figure 5. SG water level in the secondary side

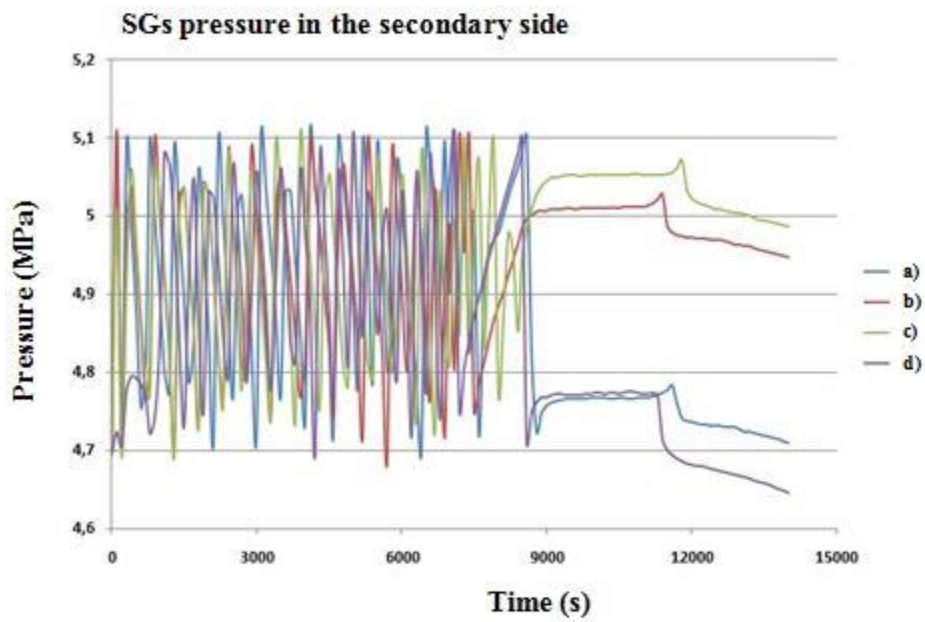


Figure 6. SG pressure in the secondary side

This modelling approach has intended to show the influence of the sensitivity studies through the variation in the number of axial volumes used to describe a major component of a CANDU reactor to some of the parameters of the plant during the early stages of a severe accident.

The event timing showed differences in the evolution of the water level in the secondary side of the SGs. What is noticeable here (in Figure 5) is that the earliest SG secondary side dryout moment occurs when the base case noding scheme is used; for cases b) no major difference in the evolution of the water level was observed, only a slower depletion of the water level near the estimated dryout moment. For cases c) and d) it is important to be underlined that the evolution of the water level is highly influenced by the nodalization (for example, for case c) the water level shows a very slow decrease up to the point where the dryout occurs, while for case d) a fast decrease of the water level has been observed right after the initiating event, that may occur due to the initial conditions which are distributed in the entire volume modelled, compared to the model containing multiple interconnected pipes where each component has different initial conditions regarding pressures, temperatures and mass flows).

Also, the evolution of the steam pressure from the secondary side (Figure 6) is unusual for cases b) and c) right after the dryout moment, and also for the last case, case d), after the initiating event. Taking in count these aspects, the selected nodalization for the SGs used to perform the uncertainty analysis for the early stages of a SBO in CANDU 6 remains the nodalization used in the base case (referenced as case a) in the analysis).

3.2. The Integrated Uncertainty Package

The uncertainty analysis of the early stages of SBO in CANDU 6 reactors was performed using the MOD3.4 version of code with the integrated uncertainty package. The uncertainty package application has to follow three main phases:

- the *setup* phase, which generates the required number of code runs and the input random samples required to execute each run. For this phase, the code requires informations related to Wilks' formula (order), and the uncertainty associated to the selected input parameters[2] [4];

- the *execution* phase, which consists of the multiple code runs (starting from the base case input deck, and following the modifications of it by the random samples generated in the previous phase);

- the *post-processing* phase to build the uncertainty limits based on the multiple code runs results (output quantities).

RELAP/SCDAPSIM/MOD3.4 with the integrated uncertainty package follows the probabilistic "input uncertainty propagation" approach of BEPU methodologies. The purpose of such approach is using a best-estimate code for the simulated SBO scenario, the selection of the input parameters based on their impact on the failure criteria[3] [19], the determination of their associated uncertainty in terms of PDFs (Probabilistic Distribution Functions), the random sampling of the parameters based on the defined uncertainty, the code executions using the random samples, and the application of order statistic theory (Wilks' formula) to determine the minimum or maximum limits (the uncertainty band) for the parameters of interest (relevant for the accident scenario to be analyzed).

3.2.1. The setup phase

For the uncertainty analysis a total number of 23 parameters were selected. The amount of parameters selected were organized into two groups depending upon their perturbation which could be applied directly in the regular input deck, the so-called "input treatable parameters", or had to be implemented in the source files, the so-called "source correlation parameters". Regardless of the parameter's nature, the uncertainty package in RELAP5/SCDAPSIM allows the perturbation of the two groups of parameters in a similar way, without the need of neither modifying nor re-compiling the code; the required information is a list of the selected input parameters and their uncertainty information in the format of a multiplier factor (for example, for a normal distribution it . The selected parameters and their associated PDFs (including the motivation for the selection of the parameters) are summarized in a previous paper of the author from entitled "Key parameters in the early stages of a Station Blackout analysis in a CANDU 6 reactor using RELAP/SCDAPSIM" [20], along with the steady state analysis of the plant behaviour, accident analysis of the base case (with no uncertainties involved).

3.2.2. The execution phase

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 code parameters selected in step "Identification of code parameters and determination of their uncertainty" are randomly sampled according to their PDF. The code is executed using the random input sets, i.e. all selected parameters are varied at a time, to generate simple random samples of results with a prescribed size N. In this way, the uncertainty is propagated from inputs to outputs. The number of code runs to obtain an upper (lower) tolerance limit that estimates a certain percentile with a certain confidence level β is determined by the Wilks' formula. The advantage of using

[2] According to [4] the code allows the user to perturb parameters described in the base case input deck (named "input treatable parameters"), and also source code modification (named "source correlations parameters").

[3] The SBO accident scenario will refer to the failure criteria and its effect [19].

the statistical formula rather than for instance a response surface technique is that the number of code calculations does not depend on the number of selected input parameters, but only on the percentile and confidence level of the desired uncertainty bound. The Wilks' formula establishes, given the order of application, which rank estimates the desired percentile with an associated confidence level.

For the purpose of this paper the Wilks` formula[4] of the 2nd order was used, which consisted of 93 code calculations, and it gives the non parametric tolerance bounds which are set from the multiple output samples obtained following the order statistics and Wilks' method, which basically consists in ordering the obtained data according to the rank (magnitude) from the smallest to highest value. The input parameters are randomly sampled according to their PDF 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.

3.2.3. The post-processing phase

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. The information required in the "post-processing" input file also includes the simulation runs to be used in the generation of the tolerance intervals

3.3. The SBO Accident in CANDU 6 Reactors

For the purpose of this paper, the SBO scenario in a CANDU 6 reactor has been selected, and it will be carried up until the pressure tube failure (since the present model for CANDU 6 build in RELAP/SCDAPSIM code becomes uncertain for the following stages of the accident, due to the fact that after the fuel channel failure a more detailed model of the core is requested, in order to observe the behavior of each row of individual fuel channels - refering to the axial position in the calandria vessel - remains uncovered due to moderator drainage. The present model uses the channel grouping based on the radial power distribution in the core and the core pass, and the axial position is neglected).

The Station Blackout (SBO) accident is one of the most challenging initiating event for any type of nuclear power plant, this including the CANDU 6 reactors. A SBO event is initiated by a total loss of off-site AC power concomitant with the turbine trip and the unavailability of the Class IV and the backup power (loss of all on-site standby and electric power supplies - diesel generators).

For an unmitigated SBO scenario, the main assumptions made were as follows:

- Class IV power and all onsite standby and emergency electric power supplies are unavailable;
- The unavailability of the Emergency Core Cooling System;
- Primary Heat Transport System loop isolation is not credited;

---

[4] The form of the Wilks' formula [21], [22] is the incomplete beta function. The code computes the number of needed runs by an iterative process that tests that the confidence level supplied in the "setup" input deck is greater or equal to the confidence level computed by the code using the incomplete beta function. The order of the Wilks' formula application increases the number of required code runs but will also obtain a more accurate estimation of the uncertainty bounds.

- SGs safety valves are available (with the set point for opening and closing to relieve pressure);
- Crash cool-down system credited;
- Air-operated atmosphere steam discharge valves are fail-closed;
- Pressurizer steam bleed valves are fail-closed;
- Moderator cover gas system bleed valves is assumed available;
- Some of the operator interventions are not credited (for example, opening the MSSVs manually to depresurize the secondary side of the SGs).

A recent uncertainty analysis [23] of this phase of the accident showed that the first fuel channel failure occurs in the time interval of 13,600-15,800 s.

The present analyses were focused on the impact of the uncertain parameters on the selected output parameters (pressure in the PHTS and SGs, water level in the SGs, channels temperatures) with the depressurization of the SGs secondary side (1h after the initiating event, at 4000 s from the beginning of the analysis for two of the analyzed cases, and also near the fuel channel failure first estimated time of occurence) followed by cooling water injection from the dousing tank. The crash cooldown will start with the depressurization of the SGs through the operator action by manually open the MSSVs (as shown in Figure 7) to relief pressure from the secondary side and locking the valves in open position (the depressurization consists in the release of steam in the atmosphere from the secondary side of the SGs through an operator action, as mentioned in [14], [15, [17]) .

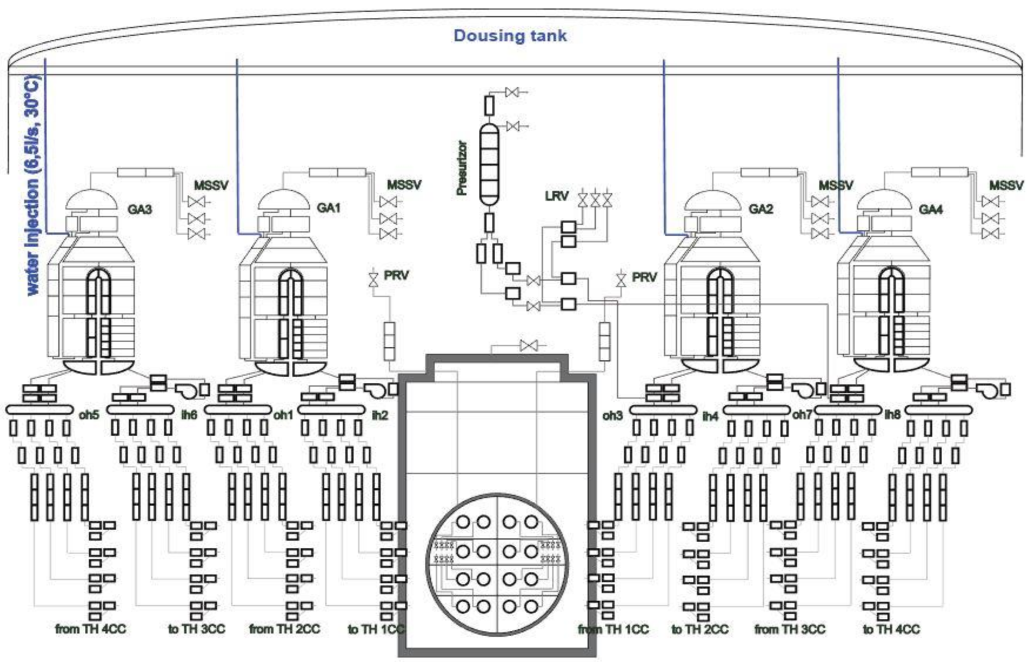


Figure 7. SGs depressurization and water injection model

The water injected into SGs is assumed to have 30°C, and the minimum flow rate that could be injected from the dousing tank is 30 l/s. A smaller amount of cold water (a total flow of 26 l/s) has been selected to be injected from the dousing tank by gravity into SGs (6.5 l/s per each SG) until the end of the analysis. The proposed scenarios regarding the time depressurization and water addition (shown in Figure 7) are as follows:

- case 1: depressurization of the SGs 1h after the initiating event (4000 s from the beginning of the analysis), and a constant cold water flow of 6.5 l/s per each SG short after the depressurization moment (100s after the depressurization time) until the end of the analysis;

- case 2: depressurization of the SGs 1h after the initiating event (4000 s from the beginning of the analysis), and a constant cold water flow of 6.5 l/s per each SG long time after the depressurization moment (3600 s after the depressurization time) until the end of the analysis;

- case 3: depressurization of the SGs before the estimated earliest moment of the first fuel channels failure (based on the uncertainty analysis of the early stages of the SBO with no operators interventions credited - unmitigated SBO), and a constant cold water flow of 6.5 l/s per each SG long time after the depressurization moment (100 s after the depressurization time) until the end of the analysis.

3.4. Results of the Early Phase of the SBO Accident considering SA Management Measures

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.

3.4.1. Case 1

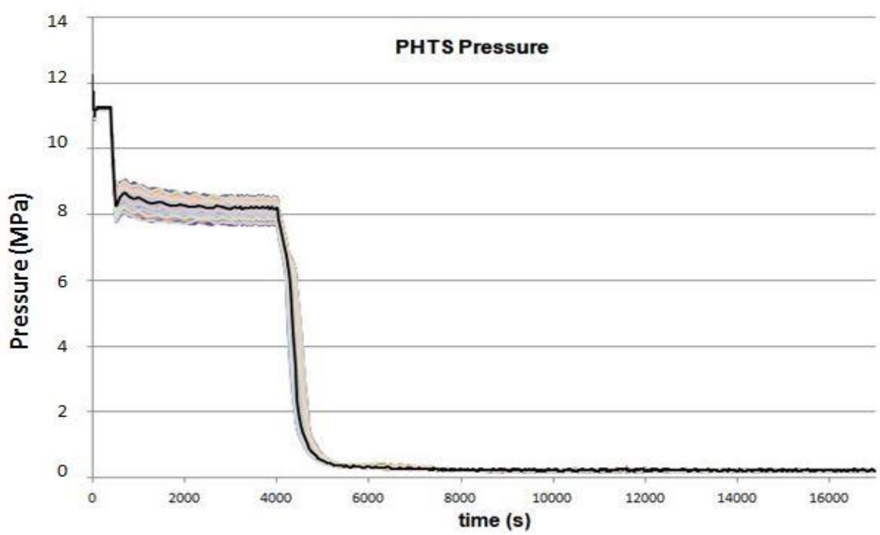


Figure 8. Pressure of the coolant in the primary system

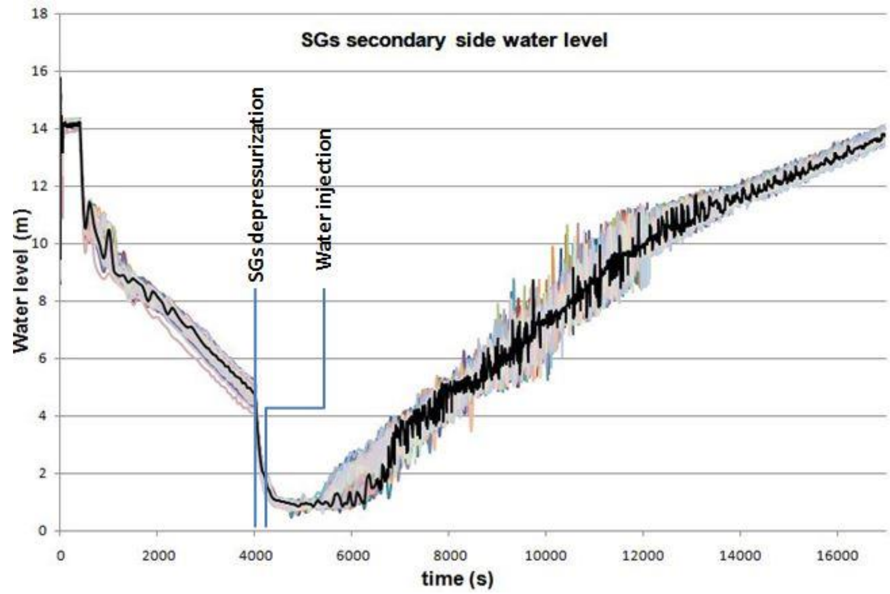


Figure 9. Water level in the secondary side of the SGs

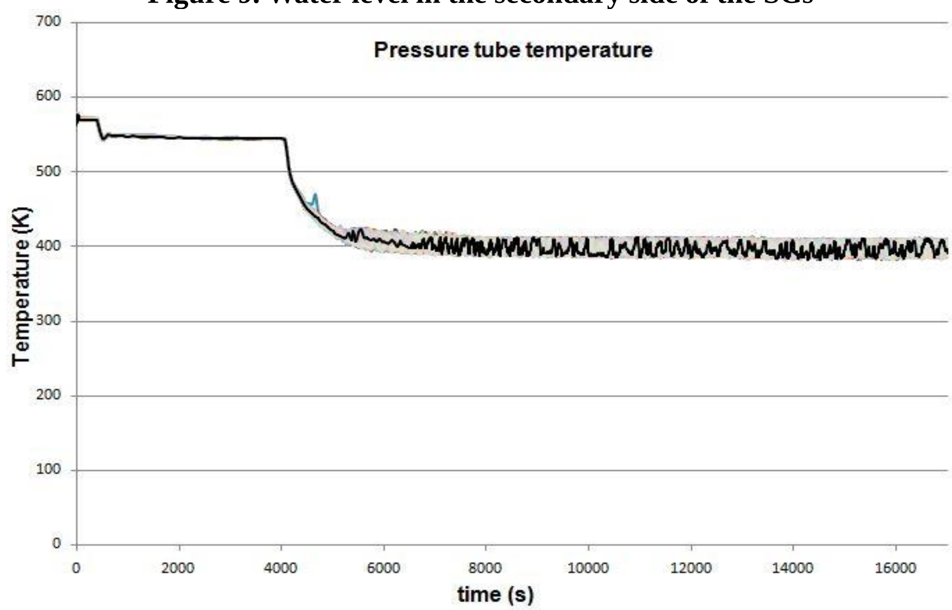


Figure 10. PT inside surface temperature

3.4.2. Case 2

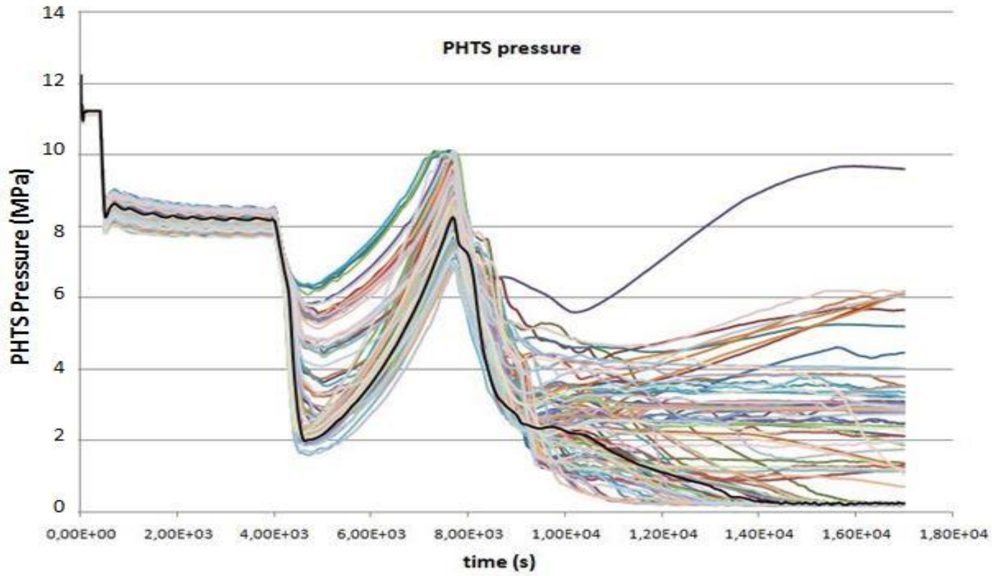


Figure 11. Pressure of the coolant in the primary system

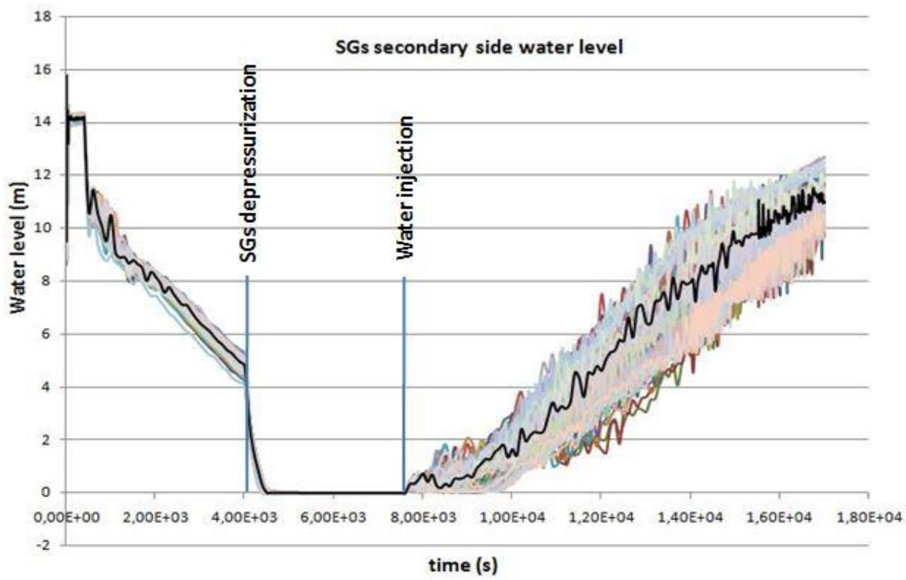


Figure 12. Water level in the secondary side of the SGs

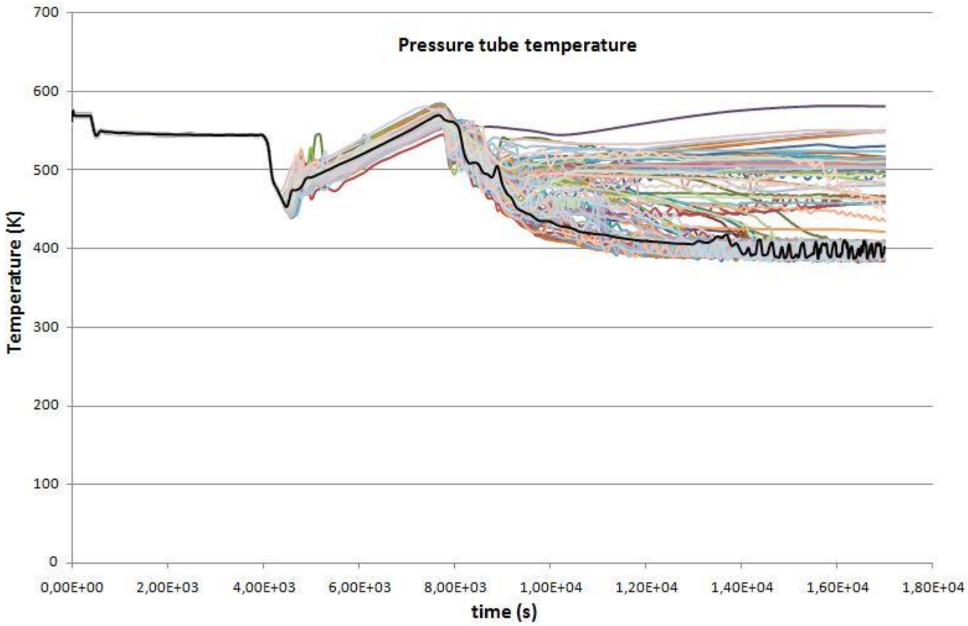


Figure 13. PT inside surface temperature

3.4.3. Case 3

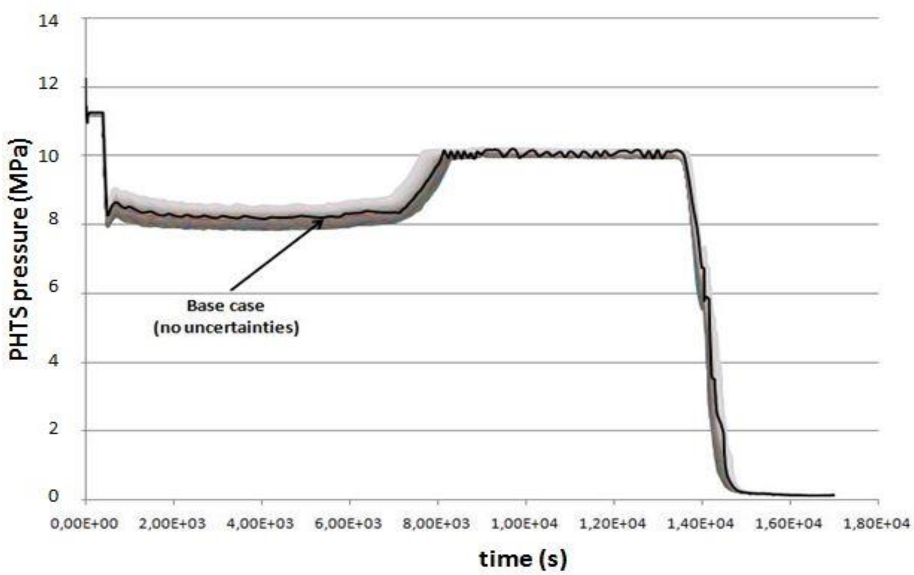


Figure 14. Pressure of the coolant in the primary system

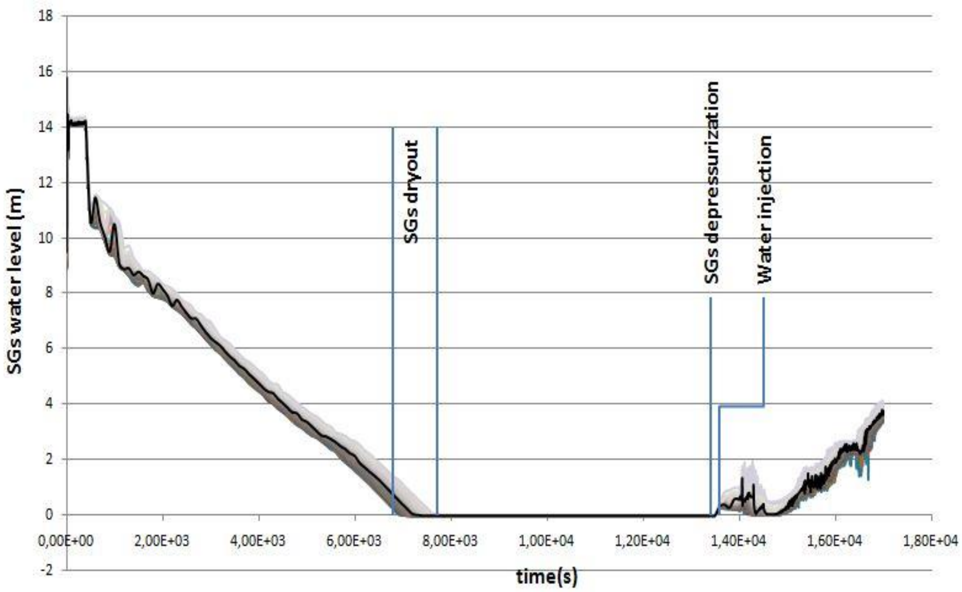


Figure 15. Water level in the secondary side of the SGs

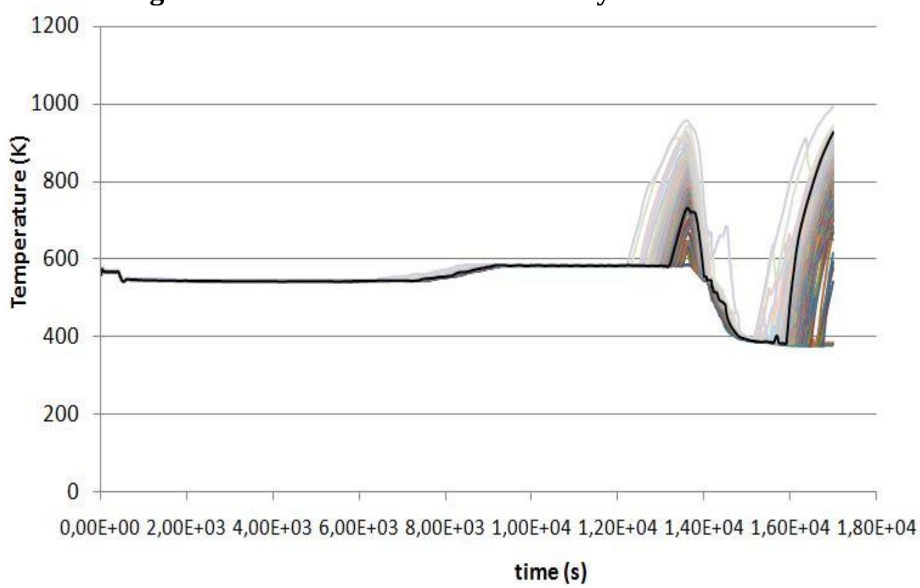


Figure 16. PT inside surface temperature

4. CONCLUSIONS

The uncertainty package in RELAP/SCDAPSIM was successfully applied to the CANDU 6 SBO analysis. The lack of experimental data for a SBO accident in a commercial CANDU 6 reactor only allows a qualitative approach of this analysis.

From the total amount of 23 parameters selected for the uncertainty analysis, only 10 of them are of source correlation type, and they are related to wall-to-fluid heat transfer (calandria tubes to moderator, fuel sheath to coolant, steam generators U tubes to secondary coolant, calandria walls and moderator etc.), critical heat flux, and fuel channel behaviour. As for the input treatable parameters, the form of the uncertainty is defined through a multiplier defined by a PDF and its characteristic parameters.

Following the prescriptions for CANDU 6 under severe accident conditions, maintaining the integrity of the safety barriers (i.e. fuel pellet, fuel sheath, PHTS limits - pressure tubes and calandria tubes, calandria vessel) is the key aspect to look for. The unmitigated SBO analysis shows that the fuel channel integrity has not been preserved. Furthermore, the uncertainty analysis gives a perspective of the failure in the time interval of [13,600-15,800 s], as well as for the calandria vessel disks.

The SBO analysis with water injection in the secondary side of the SGs was performed to demonstrate the efficiency of the safety measures regarding the integrity of the PHTS safety barrier - fuel channel. The uncertainty analysis including the implementation of the safety measures shown that the input parameters and also the source parameters have a significant impact after the fuel channel failure (t>15,000 s, as can be observed in Figure 8 to Figure 16) rather than on the early phase of the SBO, which is subjected to the purpose of this paper. The uncertainty analysis will be extended in future analyses by adding more specific parameters.

As for the case when depressurizing the SGs at about 13,400 s and water injection at 13,500 s, the PHTS integrity is not preserved, the pressure tube temperature reaches 1000K, so this measure could be implemented only if the time moment of SGs depressurization and water injection in the secondary side will be considered early before the fuel channel failure. An extended analysis on this section will be soon performed to establish the proper conditions for the implementation of this accident mitigation measure.

The obtained results could not be represented in terms of dispersion due to the time steps used for the analysis; it would be inconclusive for the present study, since the time steps was reduced at minimum near the fuel channel failure moment in order to avoid code failure with existent properties (if a fast change in pressure or temperature occurs - such as a sudden fast decrease of pressure in PHTS due to fuel channel rupture - the code will end the transient with properties failure) or near the SGs first estimated dryout moment.

REFERENCES

1. IAEA, "*Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation*", IAEA Safety Reports Series No 52, pp 1-162 Vienna, Austria (2008)
2. F. D'Auria, O. Manzzantini, "*The Best-Estimate Plus Uncertainty (BEPU) Challenge in the Licensing of Current Generation of Reactors*", IAEA-CN-164-4S08
3. C. M. Allison, J. K. Hohorts, "Role of RELAP/SCDAPSIM in Nuclear Safety", *Proceedings of the TopSafe2008*, Dubrovnik, Croatia (2008).
4. M. Perez, "Integration of a quantitative-based selection procedure in an uncertainty analysis methodology for NPP safety analysis", *PhD. thesis - UPC*, Barcelona, Spain (2012).
5. Gong Hongqi, R. S. Hart, "Moderator, Heat Transport and Steam Systems", *China Journal of Nuclear Power Engineering*, Vol. 20, No. 6 (1999).
6. AECL, "CANDU 6 Technical summary", Mississauga, Ontario, Canada (2005). Available online: https://canteach.candu.org/Content%20Library/CANDU6_TechnicalSummary-s.pdf
7. I. Prisecaru, D. Dupleac, L. Biro, **"**A study of RELAP5 capability to perform the analysis of a CANDU 6 reactors accidents", *Proceedings of the 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-11)*, Avignon, Frane (2005) INIS Ref. Number 37012968.+
8. M. Mladin, D. Dupleac, I. Prisecaru, "Evaluation of the RELAP5/SCDAP Accident Analysis Code Applicability to CANDU Nuclear Reactors", in UPB Scientific Bulletin, **Vol. 71**, ISSN 1454-234x (2009)
9. I. Prisecaru, D. Dupleac, P. Ghițescu, L. Biro, "A Parametric Study of a Large Break in Reactor Inlet Header of CANDU 6 Reactors using RELAP5 Code", *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP)*, Reno, France (2006). ISBN: 0894486985;978-089448698-2.

10. D. Dupleac, I. Prisecaru, M. Mladin, G. Neguț, P. Ghițescu, "Investigation on Header Manifold Model Effect at CANDU6 Large Loss of Cooling Accident", *Proccedings of the16th International Conference on Nuclear Engineering (ICONE),* Orlando, USA (2008). ISBN: 0-7918-4817-5.
11. D. Dupleac, I. Prisecaru, P. Ghițescu, G. Neguț, "Thermalhydraulic analysis of Candu 6 100% reactor outlet header break using RELAP5 code", *Proceedings of the International Congress on Advances in Nuclear Power Plants (ICAPP),* Nice, France (2007). ISBN: 0894480588;978-089448058-4.
12. D. Dupleac, M. Mladin, I. Prisecaru, "Generic CANDU 6 plant severe accident anlysis employing SCDAPSIM/RELAP5 code", *Nuclear Engineering and Design*, , **Vol. 239**, pp. 2093–2103 (2009). ISSN: 0029-5493.
13. E. Dincă, D. Dupleac, I. Prisecaru, "RELAP/SCDAP simulation results for CANDU 6 accident management measure: PHTS voluntary depressurization following a SBO", *UPB Scientific Bulletin*, **Vol. 78** (2016). ISSN 2286-3540.
14. E. Dincă, D. Dupleac, I. Prisecaru, "Verification by analytical means of the efficiency some accident management measures for SBO at a CANDU-6 NPP", *International Nuclear Safety Journal*, **Vol. 4** (2015). ISSN 2285 – 8717.
15. E. Dincă, D. Dupleac, I. Prisecaru, "Analysis of the CANDU-6 plant behavior in case of very late steam generators depressurization and water injection following a Station Black-Out accident", *Proceedings of the 7th International Conference on Energy and Evironment (CIEM),* Iasi, Romania (2015).
16. D. Dupleac, "Analysis of CANDU6 Reactor Station Blackout Event Concomitant With Moderator Drainage", *UPB scientific Bulletin*, **Vol. 78** (2016). ISSN 2286-3540.
17. IAEA, TECDOC-1727, "Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Application", Vienna, Austria (2013). ISBN:978-92-0-114413-3.
18. D. E. Moraru, D. Dupleac, R. M. Nistor-Vlad, C. M. Allison, J. K. Hohorst, "Development and study of a SBO on CANDU-6 plant model using GRAPE", paper presented at the *8th Conference on Severe Accident Research* (ERMSAR2017), Warsaw, Poland (2017)
19. IAEA, TECDOC 1594, "Analysis of Severe Accidents in Pressurized Heavy Water Reactors", Vienna, Austria (2008). ISBN 978–92–0–105908–6
20. R. M. Nistor-Vlad, D. Dupleac, I. Prisecaru, "Key parameters in the early stages of a Station Blackout analysis in a CANDU 6 reactor using RELAP/SCDAPSIM", *Proceedings of the International Conference on ENERGY and ENVIRONMENT (CIEM)*, Bucharest, Romania (2017).
21. S. S.Wilks, "Order statistics", *Bulletin of the American Mathematical Society*, 541:pp 6-50, 1948.
22. S. S. Wilks, "Determination of Sample Sizes for Setting Tolerance Limits", *Annals of Mathematical Statistics*, 121:pp 91-96, 1941.
23. R. M. Nistor-Vlad, D. Dupleac, I. Prisecaru, M. Perez, C.M. Allison, J. K. Hohorst, "CANDU 6 Accident Analysis using RELAP/SCDAPSIM with the Integrated Uncertainty Package", in *Proceedings of the 2018 26th International Conference on Nuclear Engineering (ICONE26)*, London, UK (2018)