UNCERTAINTY ANALYSIS CONSIDERING THE IMPLEMENTATION OF DELAYED ACCIDENT MANAGEMENT MEASURES FOR CANDU REACTORS

Roxana-Mihaela NISTOR-VLAD[1], Daniel DUPLEAC[2]

After the Fukushima accident from March 11th, 2011, the nuclear safety studies have been extended to the verification and implementation of severe accident management measures in order to mitigate the severe accident progression, preventing the core damage. In a CANDU reactor (which is a Pressurized Heavy Water Reactor) one of the potential ways severe accidents are initiated is by a total loss of power (total loss of off-site AC power Class IV concomitant with loss of on-site standby Class III and the Emergency Power Supply). In the nuclear safety studies, a new trend refers to the evaluation of uncertainties as a mandatory component of best-estimate safety analysis; this modern and technically consistent approach being known as BEPU (Best Estimate Plus Uncertainty).

The purpose of this study is to determine the efficiency of delayed severe accident management measures to be implemented, such as depressurizing the steam generators secondary side followed 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.

Keywords: CANDU reactors, severe accident, RELAP/SCDAPSIM, Best-Estimate Plus Uncertainty.

1. Introduction

CANDU 6 reactors are Pressurized Heavy Water Reactors, which have particular features compared to LWRs (Light Water Reactors) such as the completely independent systems for the primary coolant heavy water and moderator. The Primary Heat Transport System (PHTS) circulates heavy water through the 380 horizontal fuel channels (each containing 12 fuel bundles) connected to large volume pipes called headers (located above the core) by a complex system of feeders (small diameter pipes connected to the inlet and outlet of each fuel channel) as shown in Fig. 1. The fuel channels are immersed in the heavy water moderator contained in a cylindrical horizontal vessel (Fig. 2), known as the calandria vessel. The main difference between these two separated

[1] Assistant prof., Dept. of Power Production and Usage, University POLITEHNICA of Bucharest, Romania, e-mail: roxanamihaelanistorvlad@gmail.com, roxana.nistor2810@upb.ro

[2] Prof., Dept. of Power Production and Usage, University POLITEHNICA of Bucharest, Romania, e-mail: danieldu@cne.pub.ro

systems, is that during normal operating conditions the PHTS circulates heavy water at high pressure (~10 MPa) and temperature (~310˚C), while the moderator system circulates heavy water at low pressure (~0.1 MPa) and temperature (~70˚C). Particularly, the moderator system acts as an alternative heat sink for the fuel bundles for both design basis accidents and beyond design basis accidents.

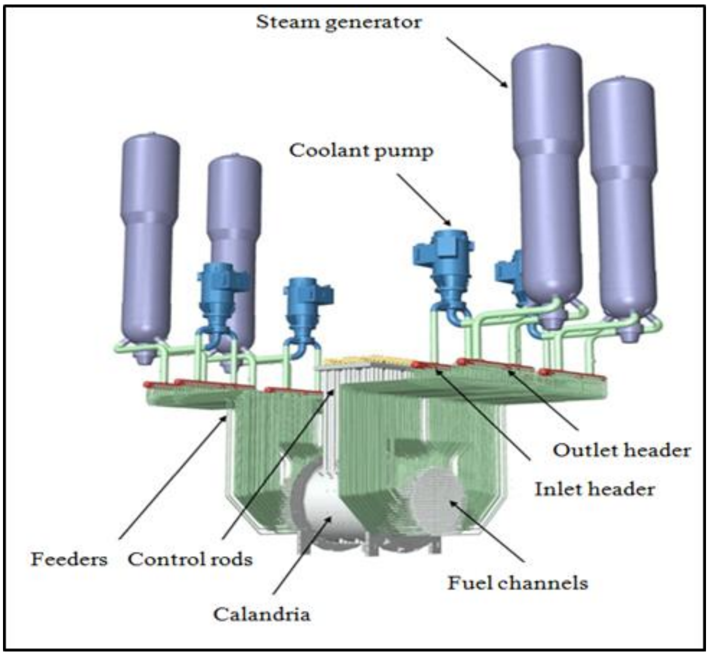

Fig. 1 Primary heat transport system [1]

The heavy water coolant transports the heat produced in the fuel assemblies to the steam generators (SGs), where it is transferred through the U tubes walls to light water from the secondary side of the SGs to produce steam. In CANDU 6 reactors the PHTS 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 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 in each loop operating in series, and 2 SGs per loop. The coolant in adjacent channels flows in opposite directions (Fig. 2).

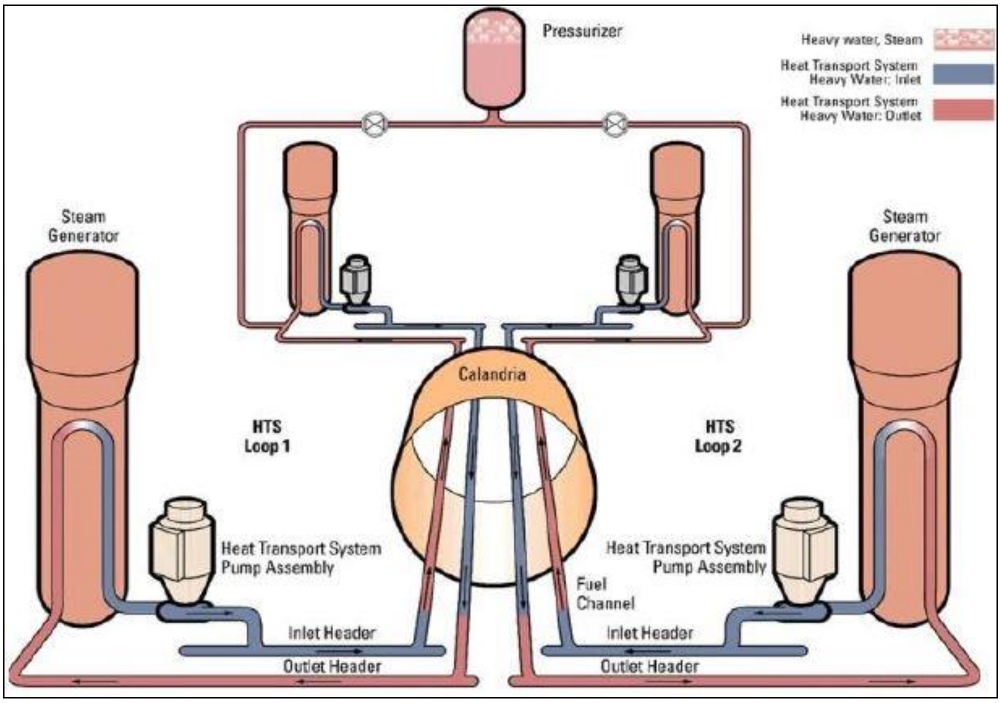

Fig. 2 A schematic figure of the CANDU loops [2]

2. Severe accident in CANDU 6

The accidents in PHWRs that could lead to the core damage are generally named by the IAEA [3] Design Extension Conditions, being divided into two classes, as follow:

- Limited Core Damage Accidents (LCDA), for which the integrity of the fuel channels is preserved (no significant core damage results from the accident progression);
- Severe Core Damage Accidents (SCDA) or Severe Accidents (SA), for which the core geometry is not preserved, due to the fuel channels failure and core collapse.

Typically, a SCDA results from a LCDA with loss of moderator as heat sink during a postulated Loss of Coolant Accident (LOCA), or from Loss Of Flow Accidents (LOFA) associated with the loss of multiple safety systems, including the loss of support safety systems.

The Station BlackOut Accident (SBO) in a CANDU reactor is initiated by a total loss of AC power (loss of off-site power supply and emergency power supply Class IV are lost, on-site power Class IV is lost, with the unavailability of Standby Diesel Generators and Emergency Power Supply. Mobile Diesel Generators (MDG), which were designed for multiple nuclear power plants after the Fukushima Daichii Accident (including also Cernavoda NPP), are not credited as available.

According to IAEA [4], the SBO progression was divided into four phases, as follow:

- Phase I: starting from the initiation of the SBO event and ending with the calandria rupture disk out, concomitant with a significant group of channels failure;
- Phase II: from the blow out of the calandria disk until core collapse;
- Phase III: starts with the core collapse and ends with the calandria vessel failure due to creep;
- Phase IV: which covers the phenomena in the reactor vault, including the corium relocation in the reactor vault floor, quenching of the corium with hydrogen production up to the fission products release in the environment.

For the purpose of this study, the analysis covered the early phase of the accident, up to the point where a significant group of fuel channels rupture occurs, based on the fuel channel failure criterion described in TECDOC-1727 [4]. The PHTS pressure is the main component that defines the mechanistic fuel channel failure. At high pressure in the PHTS ($p \geq 1MPa$), the fuel channel is assumed to fail due to the non-uniform circumferential temperature distributions, causing the pressure tube ballooning (phenomenon considered to occur when the temperature of the pressure tube inner surface reaches 1000K). At low pressure in the PHTS ($p<1MPa$) the fuel channel is considered to fail by sagging when the outside surface of the calandria tube reaches 1473K.

The Fukushima Accident, from March 11th, 2011, has increased the greatest challenge for nuclear power plants safety around the world. It consisted in the need to accelerate the study of different types of nuclear reactors behavior in the event of a severe accident, as a lesson-learned from the analysis of the event that led to the nuclear accident at Fukushima Daiichi as well as from the analysis of the response to this accident. This lesson refers both to the need for a more realistic analysis of a nuclear power plant behavior in the event that leads to a severe accident, including one in which several nuclear units are located on the same site, but also when analyzing the actions to be taken by implementing the appropriate accident management measures. These measures must be validated both by analyzing the accident, for the different scenarios considered, but also by actions taken to verify the feasibility of the actions considered, within the available time limits and with the available resources. Severe accident management measures should be taken to reduce the potential risk to the population and the environment due to the deterioration of the reactor core.

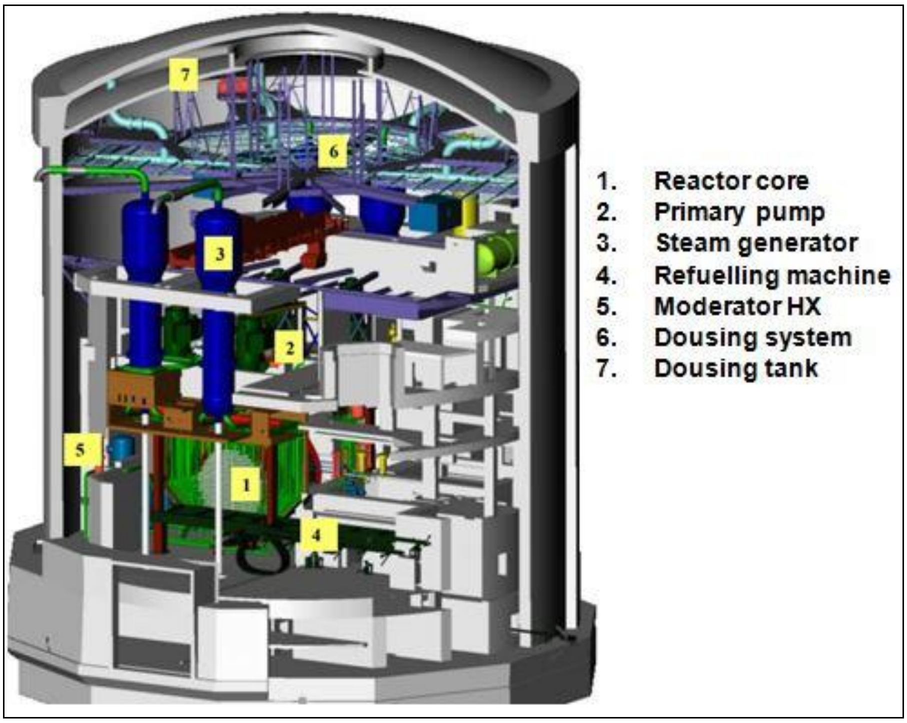

Fig. 3 Essential components of a CANDU 6

A set of management measures to be taken in case of a severe accident in CANDU 6 reactors consists in the depressurization of the SGs followed by the water injection from the dousing tank located above the reactor core (Fig. 3). If the Auxiliary Feedwater (AFW) is unavailable, an alternative source of water (demineralized) can be provided to SGs secondary side from the water inventory available from the dousing tank. The water will be injected gravitationally in the SGs once the pneumatic isolating valves open and the SGs are depressurized to atmospheric pressure. The minimum available water inventory from the dousing tank is about 2000 m^3, according to [5]. The gravitational water flow to all steam generators is about 43 l/s, considering the maximum water level in the dousing tank.

From the late 2009 University Politehnica of Bucharest investigated the severe accident progression in CANDU 6 reactors with RELAP/SCDAPSIM [6], to demonstrate the code capability to predict the plant behavior during a SBO through a comparison in the timing of the major events with the results from MAAP4-CANDU (a CANDU specific code for safety analyses). A detailed PHTS thermal-hydraulic model was developed for this study, which demonstrated that the development and verification of severe accident management programs could be successfully performed with RELAP/SCDAPSIM code.

IAEA launched in 2009 an international Coordinated Research Project conducting a benchmark exercise on SA computer codes for PHWRs analysis, which was published in 2013 in the IAEA technical documents [4]. The purpose of the project was the comparison of the integrated effects of codes models, understanding of codes limitations, assessment of the uncertainties, and the increase of the confidence in SA code predictions. PUB was part of the research team, using RELAP/SCDAPSIM to simulate the SBO accident progression.

The research activities on SA analysis for CANDU 6 was extended by E. Dinca [7], covering a complex deterministic safety analysis for different SBO scenarios with/without severe accident management measures.

Starting 2018, a new approach in the safety analysis was considered, such as the evaluation of the uncertainties in the calculated results of the early phase SBO accident scenario using the BE computer code RELAP/SCDAPSIM. The results of the uncertainty analysis [8] were focused on the evaluation of the time intervals of the relevant phenomena specific to this phase of the accident, such as the SGs dryout (1.75 – 2 hrs. from the initiating event) and fuel channel failure under high pressure in the PHTS (3.66 – 4.28 hrs. from the initiating event). Based on the results summarized in [8], the investigation of the plant behavior was extended for a SBO scenario considering the implementation of SA management measures under two different circumstances [9], as follow:

- Case 1, for which the SGs depressurization was considered 1hr after the SBO initiation, followed by the water injection 100s after the depressurization;
- Case 2, for which the SGs depressurization was assumed similar to Case 1, and the water was injected 1hr after the depressurization (considering a significant time after the SGs dryout moment, event that occurs shortly after the depressurization moment).

To be noted that for the both cases a mass flow rate of 26 l/s (6.5 l/s to each SG) was injected in the downcomers. The analyses sowed that the fuel channel integrity was preserved, proving that the implementation of the accident management measures considered are feasible.

3. Delayed SA management measures in the BEPU approach of a SBO

The uncertainty analysis methodology used was developed jointly by the Spanish Regulatory Commission (CSN) and the Technical University of Catalonia (UPC). Basically, it is a statistical methodology in which the uncertainty is represented by probability distributions and propagated through the code runs by the variation of a set of input parameters. The essential feature of this methodology is the use of statistical methods. The CSN-UPC basically follows the scheme proposed by the Code Scaling, Applicability and Uncertainty (CSAU) methodology in combination with the Wilks` formula, according to [10]. The CSAU methodology adresses three main issues: (1) requirements and code capabilities, (2) assessment and ranging of parameters, and (3) sensitivity and uncertainty analysis.

RELAP/SCDAPSIM code was selected to perform the present analysis, which is a best-estimate tool that was initially designed to predict the NPPs behavior during normal operation and transient conditions, developed as part of the international SCDAP Development and Training Program (SDTP) at Innovative Systems Software [11]. 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, and it benefits as a computational package for integrated uncertainty analyses.

For the SBO accident scenario, considering the implementation of delayed SA management measures, the very late depressurization of the secondary side of all four SGs was considered. The operator is assumed to manually open the MSSVs in order to depressurize the SGs. The depressurization moment was selected according to the earliest fuel channel failure estimated in the SBO scenario without any heat sink available as reported in [8]. The depressurization of SGs was assumed at 3.5 hrs. from the initiating event and a total of 26 l/s water flow in the secondary side of the SGs (meaning 6.5 l/s per SG) was injected 100s right after the depressurization moment. This method was not considered in the previous studies on SA since it requires additional thermo-mechanical analysis for the SGs tubes, where failures may occur as a result of the mechanical / thermal stress they are subjected to, according to CNCAN [5].

4. Uncertainty analysis and results

The early phase of the SBO analysis was limited to 4.7 hours, considering the first 400s as the steady state analysis for the plant balance. The actual limitation in time of the accident analysis comes from the uncertainties introduced by the actual nodalization model during the late phase of the accident progression.

As the first phase of the uncertainty analysis with RELAP/SCDAPSIM a complete set of key parameters were selected and divided into “input treatable parameters” and “source correlation parameters”. The parameters described in the input file for the SBO analysis refer to core power (including peaking factor), fuel channel behavior (thermal conductivity and specific heat of the pressure tubes, calandria tubes and fuel sheath), pressure loss in the core and secondary side of the SGs, pump velocity, flow rate at Main Steam Safety Valves (MSSVs) and Liquid Relief Valves (LRVs). On the other hand, the source correlation parameters selected are related to the heat transfer (heat transferred from the fuel assemblies to the primary coolant, primary coolant to the secondary coolant through the SGs U tubes, fuel channels to moderator, considering different conditions for the liquid) and the critical heat flux (which is computed in the code from the Groenveld lookup table method). A complete list of the 36 uncertain parameters and specific details of their perturbation (including PDFs, limits of variation) is provided in [12], and were also summarized in [13].

In order to perform the uncertainty analysis, the Wilks` formula 2nd order of application was used with a confidence level of 95/95, establishing 93 code runs independent from the number of the uncertain parameters selected. The results of all uncertainty calculations are used to directly derive the tolerance intervals by using order statistics.

Accident initiation will trip the reactor immediately, with a delay of the turbine trip, at 20s after the SBO initiation. Since the isolation valves of the PHTS loops are not available, both loops will show a similar behavior in terms of pressure, as described in Fig. 4.

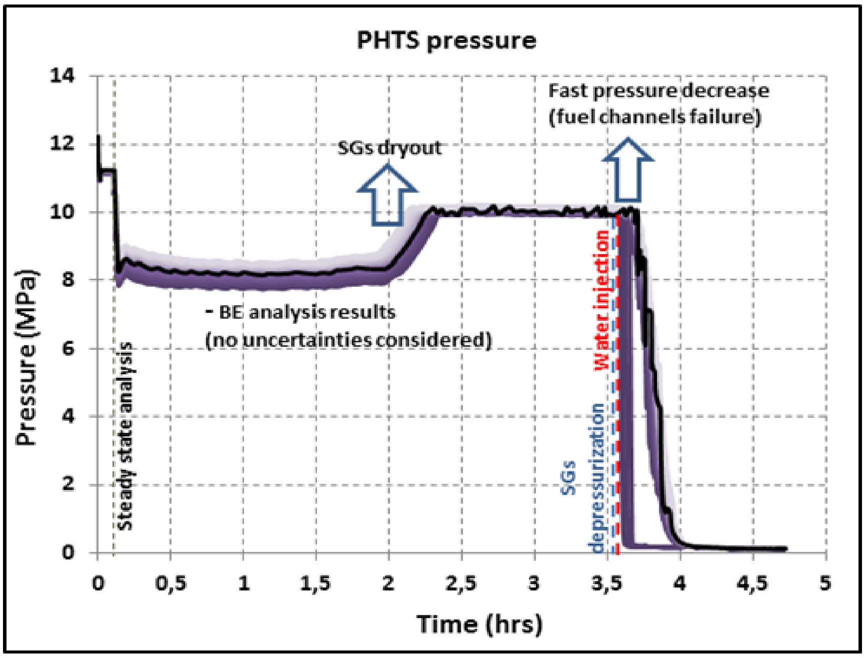

Fig. 4 PHTS pressure

The PHTS pressures decrease slightly with time due to the loss of fission power in the reactor core, the core decay heat is efficiently transferred from the PHTS to the SGs secondary side; during this short period the water from the SGs secondary side boils fast causing the significant decrease to its level. After this point, the pressure of the SGs secondary side increases (Fig. 5) up to the MSSVs opening setpoint, discharging steam outside the containment into the atmosphere, which results in the increase of the uncertainty band for this parameter, but also for the PHTS pressure. The SGs secondary side pressure then oscillates as the MSSVs open and close, leading to the PHTS pressure variation in between the values of 7.8-9 MPa.

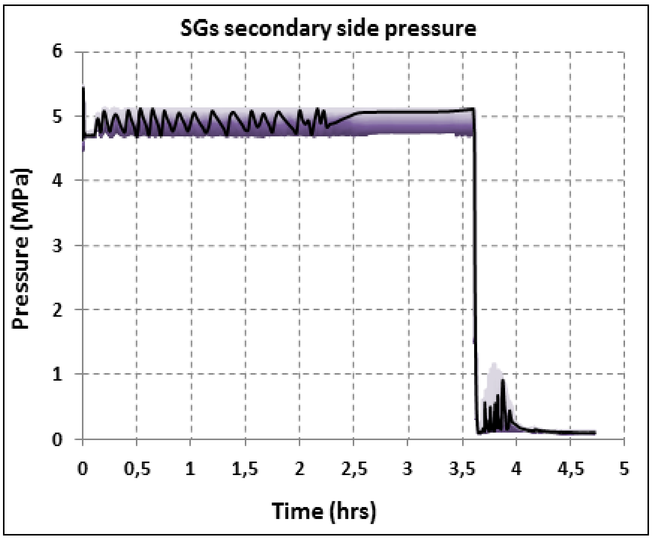

Fig. 5 SGs secondary side pressure (SG1)

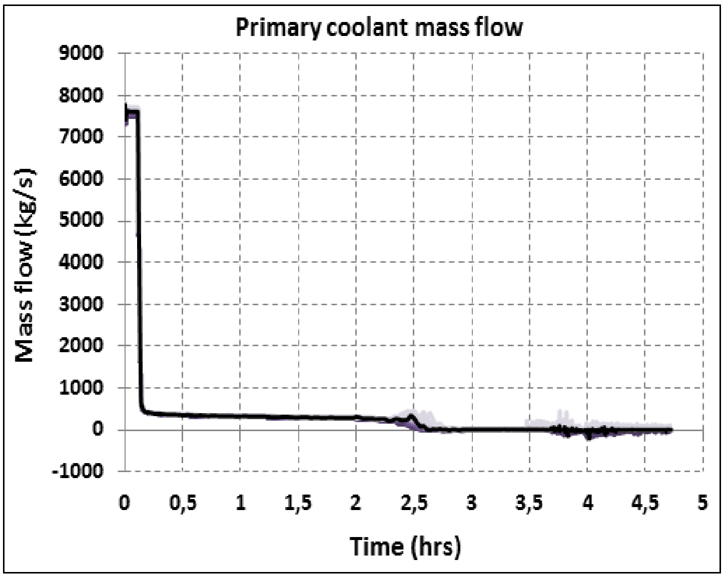

Fig. 6 Coolant total mass flow through the core

The initiating event will cause the primary pumps trip, which will turn into a fast decrease of the primary coolant mass flow through the core, to the level of natural circulation as shown in Fig. 6. This loss of flow of the primary coolant in combination with the MSSVs turn into the eventual loss of SGs secondary side inventory, which leads to the continuous decrease of the water level. The uncertainty analysis shows that the SGs secondary side water is depleted at 1.75-2 hours from the initiating event (Fig. 7), after this interval SGs no longer act as heat sink. The estimated dryout of the SGs secondary side causes the PHTS pressure gradual rise during this time interval, until it reaches the LRVs opening setpoint. Note that after the estimated time interval of the SGs dryout reduces the uncertainty band for the PHTS pressure variation but increases the uncertainty band of the mass flow and inventory variation through the core.

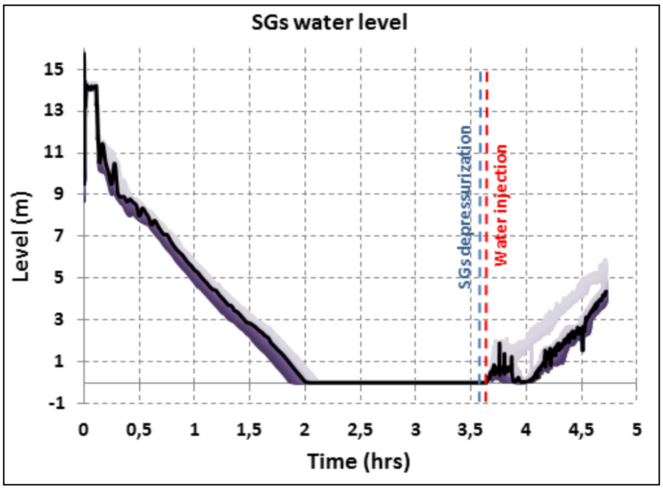

Fig. 7 Water level in the SGs secondary side

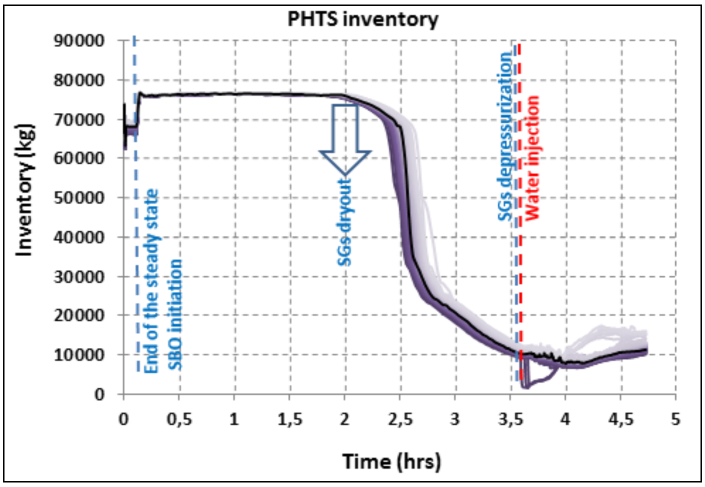

Fig. 8 PHTS D_2O inventory

Continuous loss of PHTS inventory through the LRVs (Fig. 8) results in fuel channel dryout. The boil-off of the coolant inside the fuel channel gives rapid rise to the fuel channel temperatures, and the ballooning of the pressure tube under high pressure conditions in the PHTS (as depicted in Fig. 4, the PHTS pressure before the SGs depressurization is 9.9-10.2 MPa, and the fuel channel failure criterion will be based on the PT inner surface temperature of 1000 K).

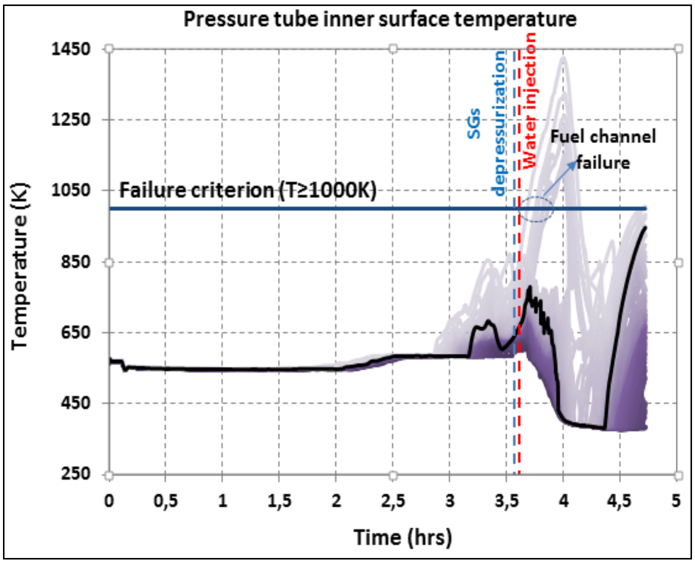

Fig. 9 Pressure tube inner surface temperature

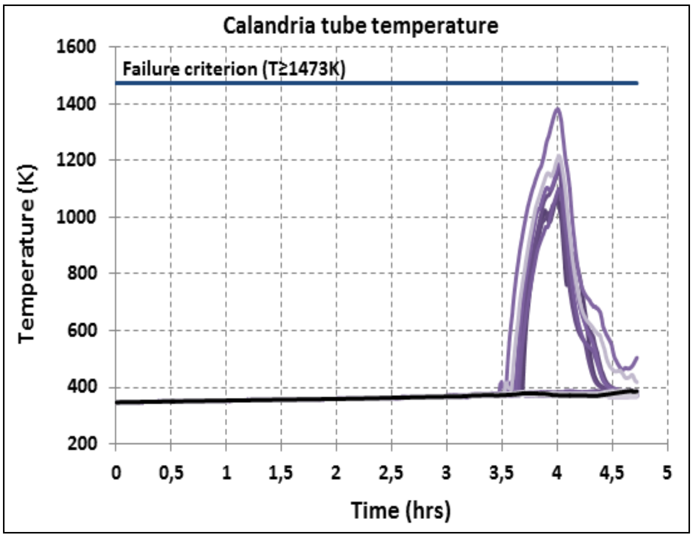

Fig. 10 Calandria tube outer surface temperature

No fuel channels failures have been observed until 3.5 hours after the initiating event (3.61 hours from the beginning of the analysis), the SGs depressurization is initiated by manually opening the MSSVs (operator manually lock the MSSVs in open position from the field) and water flows gravitationally from the dousing tank into SGs downcomers, at a total mass flow rate of 26 kg/s. Even though the injection starts right after the depressurization (with a delay of 100s, in order to allow pressure of the SGs secondary side to decrease so as to water from the dousing tank could be injected), the primary pressure shows a fast decrease for 7 of the uncertainty runs, similar to the case where the fuel channel rupture occurs. The PT inner surface temperature trend proved that fuel channels failure criterion is met for these cases, as it can be observed in Fig. 9.

From the perspective of the base case analysis (where no uncertainties were associated to any parameters), the best-estimate results could lead to the conclusion that there are no fuel channels ruptures during this phase of the accident, and the delayed actions to prevent the accident progression could be assumed efficient. However the uncertainty analysis showed a different response for some of the analyzed cases; for a certain combination of the uncertain parameters, fuel channels may fail due to ballooning of the PT.

5. Conclusions and discussions

An uncertainty analysis of a Station BlackOut accident was performed for a CANDU 6 reactor, using the RELAP/SCDAPSIM computer code considering delayed accident management measures. The proposed case study was initiated to estimate the efficiency of the proposed measures implemented long after the SGs dryout moment, but before the first group of channels failure estimated in the uncertainty analysis of the SBO scenario previously performed. Thus, the present analysis does not consider ageing due to the lack of available data.

The results of the analysis indicated that the SGs can act as a heat sink for about 1.75-2 hours from the initiation of the event.

The proposed SA management measures involve a depressurization of the SGs secondary side followed by water addition from the dousing tank from the reactor building in an integrated uncertainty analysis of a SBO accident for CANDU 6 reactors. The best-estimate analysis of the selected scenario, considering the implementation of the above-mentioned management measures, at 3.5 hours from the initiating event resulted in no possible fuel channels failures, during the analysis, while the uncertainty analysis gives a clear perspective about the challenges on the fuel channels integrity. For a total number of seven random sampling of the uncertainty parameters range, the pressure tube fails short time after the depressurization and water injection in the SGs, so the considered measures are not effective. Two possible problems were identified at the end of the study, as follow: (1) operator action considered was too late, and (2) the water mass flow added from the dousing tank was not sufficient to partially recover the SGs cooling function (note that the maximum flow supplied from the dousing tank is 43 l/s, while in the present study a flow of 26 l/s was considered). To cover the above-mentioned aspects, an earlier moment can be considered for the operator`s action to depressurize the SGs (3 hours after the initiating event), or a mass flow of 28 l/s to be added in the SGs downcomer could be analyzed.

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