

Verification by analytical means of the efficiency of some accident management measures for SBO at a CANDU-6 NPP

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Abstract

This paper presents the analysis of the consequences of a Station Black-Out (SBO) accident at a CANDU-6 NPP, considering both the reference case, without any heat sink credited for nuclear fuel cooling and the situation when an accident management measure is implemented for the prevention of a severe accident. This measure, consisting in the steam generators' depressurization followed by addition of water into the steam generators from the dousing tank, has been considered in the analysis to be implemented according to the actual emergency operating procedure specific for SBO initiating event, at Cernavoda NPP, considering different flowrates for the water added into SGs after their depressurization. The aim of these calculations is the verification by analytical means of the efficiency of these management measures in ensuring the seat sink after a SBO at a CANDU-6 NPP. Selected accident sequences have been analysed using the RELAP/SCDAPSIM.MOD3.6(a) computer code.

Keywords

CANDU, SBO, severe accident, SG depressurization, RELAP/SCDAPSIM, nuclear safety, accident analysis

1.0 Introduction

1.1 Scope and Objectives

One of the most challenging initiating events for nuclear power plants (NPPs), including for CANDU NPPs, is the Station Black-out event that represents the loss of alternating current electrical power to different equipment, which is essential in ensuring the extraction and removing the heat generated by the nuclear fuel, after reactor trip. This paper presents an analysis of the behavior of a CANDU-6 NPP in case of a Station Black-Out (SBO) initiating event occurring with reactor at full power and with the nuclear fuel at equilibrium condition.

In certain accident conditions, when the cooling of the nuclear fuel cannot be ensured, the SBO event can progress to a severe accident with reactor core damage. SBO has been intensively studied for all types of NPP designs, including of CANDU type, due to its potential consequences on the reactor core, release of fission products and hydrogen to the containment. The SBO analyses for CANDU reactors have been performed until now for different accident conditions and using different computer codes, as, for example, MAAP4-CANDU, ISAAC and RELAP/SCDAPSIM. The objectives of some of the analyses included the verification of the efficiency of implementation of specific accident management measures, considered for SBO in Emergency Operating Procedures (EOPs) or in Severe

Accident Management Guides (SAMGs). Such studies have been performed both before and after the Fukushima Daiichi accident occurred in March 2011, including in the frame of so-called “stress tests”, safety re-assessments, performed for all European NPPs, as it is shown in reference [13] for the safety review of Cernavoda NPP. The prevention and mitigation of severe accidents have become in the last period of time a serious and continuous concern of the nuclear industry, regulatory authorities in the nuclear field as well as of different technical support institutes and of international organizations such as the International Atomic Energy Agency, the Organization for Economic Co-operation and Development (OECD) and the European Commission. In line with the international trend in this field, the Romanian regulatory body, the National Commission for Nuclear Activities Control (CNCAN) has intensified in the last period of time its actions related to the verification of the accident management measures, including by analytical calculations, in addition to the general review of the Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) developed by the licensees.

Actions provided in EOPs and SAMGs, including for the SBO event, are validated by the licensees, taking into account the different activities that operators have to perform, considering the available time for different activities, the availability of different tools, water sources, etc. Different calculations have been performed in order to determine the efficiency of the activities performed by operators for SBO accident management, in accordance with the specific EOP actions. One of the most important automatic or operator actions following a SBO event is to ensure the cooling source for the primary coolant, by depressurization of the steam generators (SGs) and addition of water into the SGs. SGs auto-depressurization occurs at about 30-35 minutes after SBO [13].

The main objective of this paper is to demonstrate by calculations, performed with the RELAP/SCDAPSIM/MOD3.6 computer code, the efficiency of the steam generators as a heat sink in case of a Station Black-Out accident at a CANDU-6 NPP. The efficiency of this heat sink and its importance in case of a SBO has been demonstrated until now by different other analyses, performed for CANDU-6 reactors [13, 16]. These studies have been performed using CANDU specific computer codes, such as CATHENA and MAAP4-CANDU. Other accident management measures have also been verified by calculations, as in [14], using the ISAAC code.

The present analysis is performed using an alternative, non-CANDU specific computer code, the best-estimate computer code RELAP/SCDAPSIM/MOD3.6. It also takes into account different values of the water flow that is added into SGs after SGs depressurization. Based on the results obtained, some conclusions are drawn regarding the efficiency of this SBO management measure, as well as with regard to the mass flow required to be supplied to the SGs in order to have them available as a heat sink for nuclear fuel cooling.

1.2 CANDU-6 NPP description

CANDU is a Canadian-designed power reactor of PHWR type (Pressurized Heavy Water Reactor) that uses heavy water (deuterium oxide) for moderator and coolant, and natural uranium for fuel. A CANDU-6 reactor consists of a large tank - the calandria vessel (CV), which is penetrated by 380 horizontally oriented fuel channels, containing 12 natural uranium fuel bundles with 37 fuel rods each of them. The fuel channel consists of a 104 mm diameter, 4.3 mm thick Zirconium - Niobium pressure tube (PT), inserted into a slightly larger calandria tube, CT (thinner, from zirconium alloy), and two end fittings at the ends of the fuel channel, from stainless steel. The pressure and calandria tubes (PT and CT) are 6.3 m long, and

between them there is the annulus system space, filled with carbon dioxide. A schematic of CANDU reactor is shown in Figure 1.

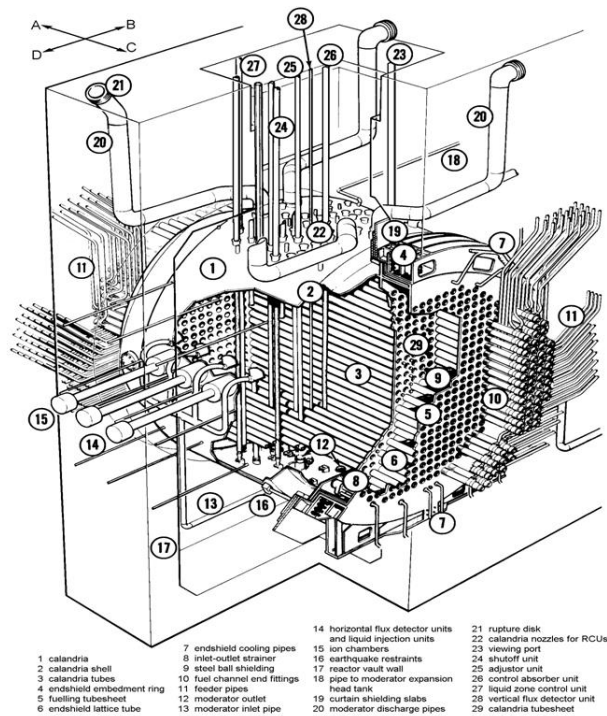


Fig. 1: CANDU-6 reactor – schematic diagram

The core of the CANDU-6 reactor is divided vertically into two halves, each of them representing a separate coolant circuit (or loop). There are two loops, with 2 passes through the core for each loop. Heavy water coolant is supplied to the PTs in each circuit via large reactor headers at each end of the CV, one pair of headers (inlet - RIH/ outlet - ROH) for each pass through the core. Heavy water flows through PTs, removing heat from fuel bundles and transferring it to the 4 steam generators (SGs), 2 on each loop, where the heat is transferred to the secondary circuit light water and converted into steam.

The primary circuit has, during the reactor operation, a nominal value of 10 MPa for RIH pressure and a temperature up to 310°C and heavy water is circulated in the loop with two primary pumps, one after each SG. The two loops are interconnected for balance, but they can be isolated in case of a Loss of Coolant Accident, LOCA. The pressure control is ensured by a pressurizer and the overpressure protection of the Primary Heat Transport System (PHTS) is ensured by Liquid Relief Valves (LRVs) that discharge heavy water from PHTS into degasser-condenser (DGC). DGC overpressure protection is ensured by the discharge of steam and water into containment through the two spring relief valves, DGC-RVs.

2.0 Analysis

2.1 SBO Accident conditions

2.1.1 Unmitigated SBO

In case of SBO, when following of loss of Class IV power the automatic or manual startup of the Standby Diesel Generators (SDGs) does not succeed, as well as the Emergency Power Supply (EPS) Diesel generators, and none of the heat sinks could be put in service (unmitigated SBO accident) at a CANDU-6 NPP, the decay heat will be removed for a while

from the primary coolant by the evaporation of the SGs water inventory, through the Main Steam Safety Valves (MSSVs), that open and close to ensure the SGs overpressure protection. After the SGs dry-out, the PHTS pressure starts to increase and overpressure protection of PHTS will be ensured by the discharge of some heavy water through the LRVs into the DGC, and then through the spring actuated relief valves, DGC-RVs, into reactor containment atmosphere. The PHTS pressure will remain at about the primary nominal value, as the DGC-RVs close once the DGC pressure decrease under RVs setpoint. LRVs/RVs will cycle, discharging into containment a large part of the PHTS inventory.

PHTS high pressure does not allow the injection of water by the high pressure stage of the Emergency Core Cooling System (HP-ECCS) to recover the lost inventory. This will lead to the increasing of the fuel and fuel channels temperature, until one or more fuel channels will break and discharge hot heavy water into the moderator in CV, determining both the PHTS depressurization and also moderator pressure increasing up to the setpoint of CV rupture disks break. At this moment, a large amount of moderator is ejected into the reactor containment atmosphere. The top rows of the fuel channels remain uncovered and start to heat-up, if a water source is not provided for fuel cooling, at a high enough flow to recover the lost inventory of primary coolant and/or moderator. The hot fuel channels will sag and break. The resulted debris can remain for a while suspended over the still well-cooled fuel channels, immersed into the moderator, but the core collapse occurs when the disassembled fuel channels weight can no longer be supported by the channels below. The fragmentation of the debris leads to more radioactive products release. The oxidation of the fuel cladding and core structures that contain zirconium will generate hydrogen. This is released also into containment and in some conditions (hydrogen concentration) can challenge the containment integrity.

2.1.2 SBO with implementation of accident management measures

In case that following the loss of Class IV power the automatic or manual startup of the standby Diesel generators (SDGs) does not succeed, as well as start of the Emergency Power Supply (EPS) Diesel generators, the operator must initiate SG's depressurization in order to bring into service the low pressure water supply systems to SGs: Boiler Make-up Water (BMW) or Emergency Water Supply (EWS) system. The operator will open MSSVs and when the SGs pressure decrease below 345 kPa, the water from the dousing tank will start to be fed by gravity into the SGs. Since the batteries are still available, if the operator does not depressurize the SGs then the SGs auto-depressurization is automatically initiated, when the level in at least two SGs will be below -2.6 m (around 9 m over SG's tubesheet, in this simulation) for 20 consecutive minutes. The automatic action is conditioned by the feedwater header pressure below 4.93 MPa(a), that indicate the total loss of feedwater to SGs.

The water available from the dousing tank will flow gravitationally to the SGs once the BMW pneumatic isolating valves are open and the SGs are depressurized to atmosphere. The BMW valves can be operated manually from the secondary control area (SCA) or manually from the field in order to control the SGs level. The minimum available demineralized water inventory from the dousing tank is about 2000 m³ according to [13]. This reference also confirms that the gravitational water flow to all SGs is about 43 l/s, considering the maximum water level in the dousing tank and the SGs depressurization and water injection into SGs occur at about 30-35 minutes after SBO initiation.

As long as the flow path will provide water to the SGs, at a sufficient flow, the thermosiphoning process will ensure decay power removal. For the long-term, the operator is directed to start EPS and initiate EWS system to provide water supply to the SGs.

2.2 Analysed cases

This paper presents the analysis of:

- Case A, or reference case: Severe accident resulted from the unmitigated station blackout (SBO) event. The study of unmitigated SBO has the aim to understand, based on the behavior of the PHTS and reactor core, which are the accident management measures that can be taken in order to prevent the severe accident potentially developed from a SBO, as well as the time windows available to implement them. The reference case has been analysed considering only the behaviour of the PHTS and reactor core (early phase).
- Case B: SBO with depressurization of SGs, followed by the addition of water into SGs. For this case, the analysis is performed considering that the SGs depressurization is performed according to the design intent, by auto-depressurization of SGs. According to [13], but also according to the results of simulations performed with RELAP/SCDAPSIM code, the specific conditions for auto-depressurizations are fulfilled at about 2150 s (from the SBO initiation). Conservatively, the following cases have been analysed, to simulate the implementation of SBO accident management measures (according to SBO Emergency Operating Procedure):
 - Case B1: SGs depressurization at 2200 s and a constant flow of 30 l/s, total for all SGs (7.5 l/s/SG), is added from 2300 s to 20000s; at 20000s the water flow is reduced at 18 l/s till the end of the simulation (38000 s)
 - Case B2: SGs depressurization at 2200 s and a constant flow of 40 l/s , total for all SGs (10 l/s/SG), added from 2300 s to 35000 s (end of the simulation).

The flow selected considered the low limit of 30 l/s, as the flow that can be provided by EWS, and the maximum flow that can pour gravitationally from the dousing tank into SGs (43 l/s value for maximum water level in dousing tank - through open BMW isolation valves, according to [13]).

The periods of time selected for the simulations of accident sequences were based on obtaining by calculations clear indications about the efficiency of SGs as a heat sink, following the SGs depressurization and water addition into SGs in the case of SBO; these periods are similar with the duration of the simulation for reference case (around 40000 s). The two cases with SGs depressurization at 2200 s after SBO initiation consider different water flows in order to determine the effect of this parameter on the SGs efficiency as a heat sink in case of SBO.

2.3 RELAP/SCDAP computer code

Severe accidents at CANDU reactors have been analyzed up to present using MAAP4-CANDU [2] and the Korean ISAAC computer code [3], and there are also analyses performed by using the RELAP/SCDAPSIM computer code, [4], [5]. MAAP4-CANDU is a modular accident analysis code developed by AECL and ISAAC computer codes is an integrated severe accident analysis code developed by KAERI. Both computer codes were initially developed from MAAP4 code, originally used for severe accident analyses at PWR as support for Probabilistic Safety Assessments.

RELAP/SCDAPSIM/Mod3.6 code version has been used for this analysis; this version represents an extension of Mod3.5 after including new ATUCHA and CANDU specific

features, [17]. Mod3.5 contains also important improvements, comparing with the previous Mod3.4 version of the code [5].

The RELAP/SCDAPSIM computer code has been designed to predict the behavior of PWR reactor systems during normal and accident conditions, and it includes detailed fuel and severe accident models and features. RELAP/SCDAPSIM uses the US NRC thermal hydraulic and severe accident models originally derived from RELAP5/MOD3.3 and SCDAP/RELAP5/MOD3.2, respectively, in combination with advanced coding and numerics and models developed by Innovative Systems Software (ISS) and other members of the SCDAP Development and Training Program (STDP). RELAP/SCDAPSIM includes the RELAP best estimate thermal hydraulic models plus detailed fuel rod, control rod, and other models that allow best estimate fuel behavior as well as severe accident calculations.

Although RELAP/SCDAPSIM computer code was developed mainly for vertical core flow light water reactors, it has been also used for a range of applications to PHWR analyses (as [6], [7], [8], [9], [10], [11], [12], etc). The SCDAP part of the code has not been utilized at the same level as RELAP due to difficulties in the modelling of horizontal fuel channels degradation and failure and core disassembly, as the code has been designed with models for fuel and core components for vertical core reactors. The horizontal geometry of the fuel channels in CANDU reactors makes the application of SCDAP models difficult beyond the heat-up and oxidation phases of an accident. Some modifications in the code subroutines have been proposed by the Romanian participants for RELAP/SCDAP code improvement to increase the applicability of this code to CANDU reactors.

2.4 CANDU Model

The CANDU-6 plant model developed for the analysis of a CANDU plant behavior, during accident conditions (design basis and severe accidents) by using RELAP/SCDAPSIM computer code has been continuously improved by the Romanian specialists (as in [6], [7], [8], [9], [11], etc.). The nodalization scheme contains the representation for the two loops of PHTS, with fuel channels, Reactor Inlet Headers (RIH), Reactor Outlet Headers (ROH), feeders and end fittings, primary coolant pumps, pressurizer, SGs and associated pipes (Fig. 2). A simplified model for balance of plant systems is also used in analysis.

All simulated CANDU-6 plant components are modelled using RELAP5 components, excepting the fuel and fuel channel thermal response that are modelled using SCDAP components. CV is modelled as two parallel pipe components with three vertical sub-volumes. Each of these pipe components simulates half of the CV volume and represents the moderator surrounding fuel channels of one PHTS loop. The analogous volumes of the two parallel pipes are connected through cross flow junctions, as it is shown in Figure 3, representing the CV model used in analysis. The four CV pressure relief ducts are modelled as a single pipe component with three sub-volumes having vertical orientation. The over pressure protection of CV is ensured by the break of the rupture disks that are modelled as a single trip valve connecting CV with reactor containment. A detailed description of RELAP/SCDAP model of CANDU-6 plant, as well as the model of CANDU-6 reactor (modelled both with RELAP5 and SCDAP components), the analysis methodology, assumptions and failure criteria used in analysis of a SBO accident can be found in [1].

In addition to PHTS model used in [1], the model for PHTS overpressure protection has been improved, considering the discharge of primary coolant through LRV's into degasser-condenser and the discharge from degasser-condenser through spring actuated Relief Valves into reactor building atmosphere, for an increased accuracy of calculations performed for PHTS inventory and pressure.

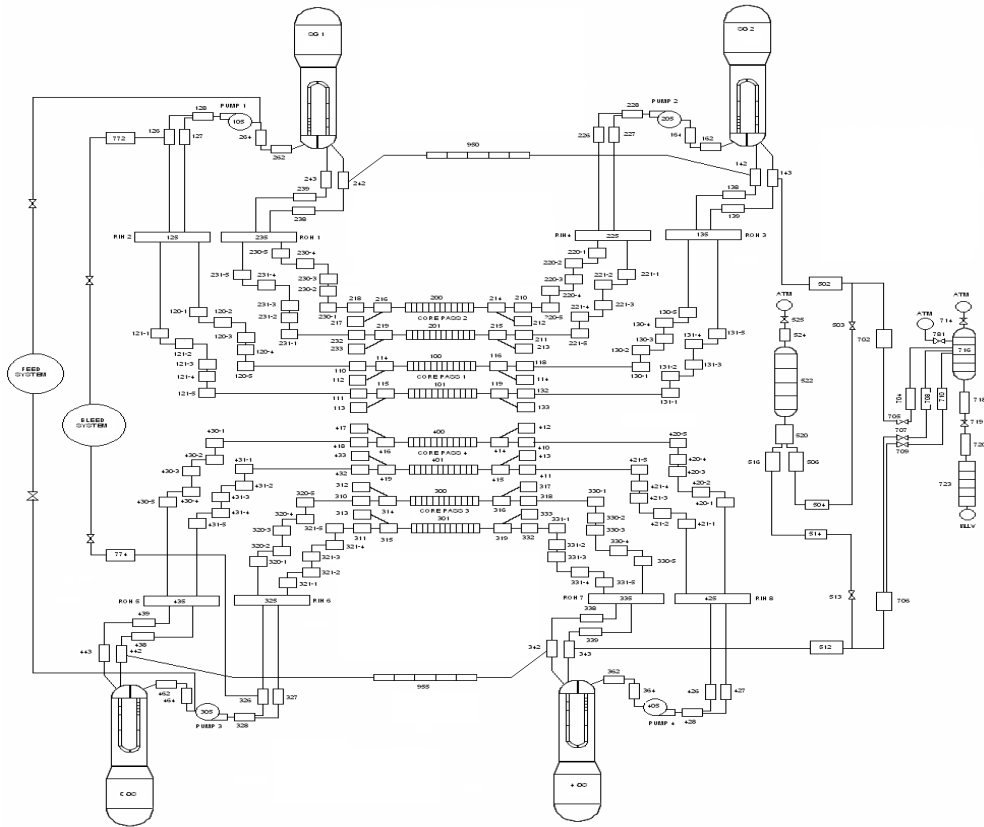


Fig. 2: CANDU-6 – PHTS nodalization scheme

CANDU fuel channel, including the oxidation of the PT and CT that contain zirconium (which can in specific conditions oxidize and generate hydrogen) is simulated by using the specific shroud model of SCDAP from RELAP/SCDAPSIM computer code. 16 fuel channels are modeled for the entire reactor core, and 32 SCDAP components are used (16 for fuel, 16 for PT/CT). The 37 fuel elements of a fuel assembly, pressure tube (PT) and calandria tube (CT) are modelled using SCDAP components for the reactor core, as is shown in Fig. 4. The 12 fuel assemblies of the fuel channels are modelled as axial nodes of the channel.

The discharge of fluids into the containment, in different locations (DGC-RVs, rupture disks, MSSVs), as well as the addition of water into SGs, are modelled using time-dependent volumes, with boundary conditions imposed by the user.

2.5 Input Data and Assumptions

The input data used are for a generic CANDU-6 reactor. Data used in the analysis performed, are the same as those used in [1], and they have been collected during the years by the Romanian specialists that developed CANDU models. In addition, the model for overpressure protection has been improved, considering the discharge of primary coolant through LRV's into degasser-condenser and discharge through spring actuated relief valves into the reactor building containment for an increased accuracy of primary inventory calculations

In this analysis, the same failure criteria as in [1] have also been used. The fuel channel is assumed to fail under high pressure and temperature when the ballooning criterion is satisfied. This criterion is the temperature of 1000°K on the inner wall of the PT, when the fuel channel pressure is higher than 1 MPa. At low PHTS pressure (lower than 1 MPa), the fuel channels may fail due to local melt-through or sagging of the pressure and CT. The temperature criterion in this case is the average temperature of CT to be higher than 1473°K. A value of

25000 kg per one PHTS loop was considered by the users as a criterion for the core collapse in the analysis. The criteria are the same as those used in MAAP4-CANDU calculations, according to [1].

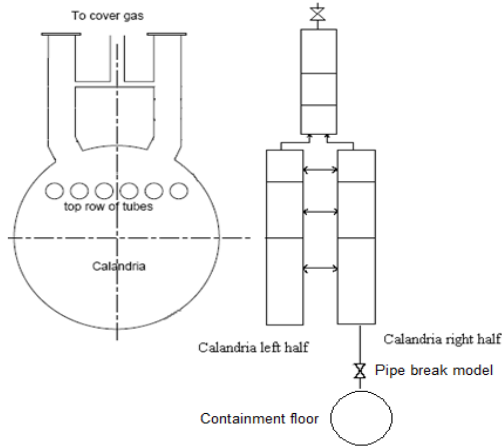


Fig. 3: Calandria Vessel model

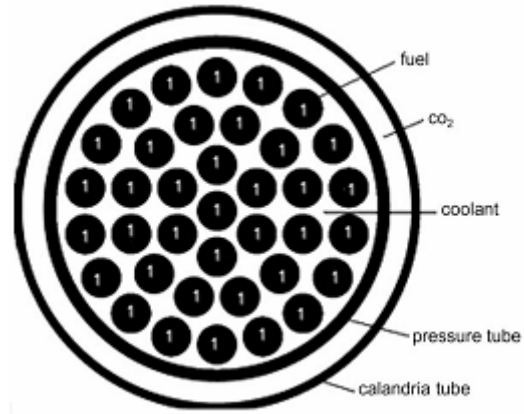


Fig. 4: Fuel channel model

This core collapse criterion cannot be simulated yet in the RELAP/SCDAPSIM code, therefore the user should determine in the output the moment of the reactor core collapse. The beginning of the channel disassembly has been simulated using an input model developed, where the fuel channel break occurs when the temperature criteria are fulfilled.

3.0 Analysis Results

3.1 SBO reference case (Case A)

The main results obtained in the analysis performed for the SBO, reference case, are presented in Table 1.

Table 1: Evolution in time of unmitigated SBO accident at CANDU-6 – reference case

Event	Time (h)	Time (s)
Loss of all power sources (Class IV, Class III, Emergency Power Supply unavailable) – except batteries	0.0	0
Reactor trip	0.0	0
Secondary side of SGs is dry (at all SGs)	2.086	7510
First time LRVs open	2.408	8970
DGC-RVs open first time and discharge D2O into containment	2.561	9220
First fuel channel breaks	3.6	12960
CV opening and moderator expulsion due to CV rupture disks breaking	3.6	12960
Start of CANDU core disassembly ($T_{CT} > 1473^{\circ}K$)	3.971	14301
Core collapse on the calandria vessel bottom	5.861	21100

For this case, the evolution of the SGs water level is shown in Fig. 5 and the evolution of the PHTS pressure is shown in Fig.6. Fig. 7 shows the evolution of the maximum fuel surface

temperature for the unmitigated SBO (this temperature is conservatively used as a value for cladding temperature).

The evolution of the PHTS inventory in case of unmitigated SBO accident can be seen in Fig. 8. From the analysis of this case resulted that, at the moment of fuel channel break, the maximum pressure at the bottom of calandria vessel (CV) is 20.66 bars and the pressure in the calandria ducts is 11.02 bars. The maximum flow of the moderator discharged into reactor containment at the rupture disks break is 13905.4 kg/s.

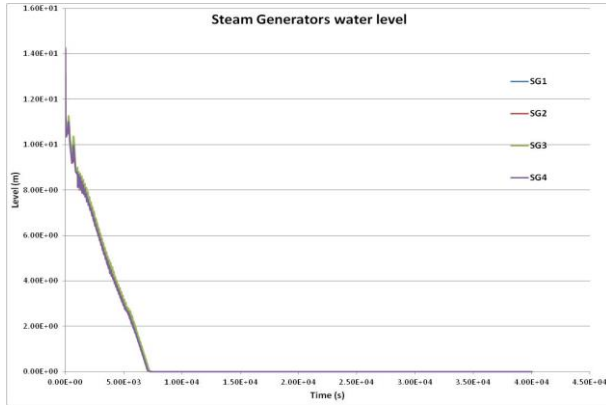


Fig.5: SBO – Case A: SGs water level

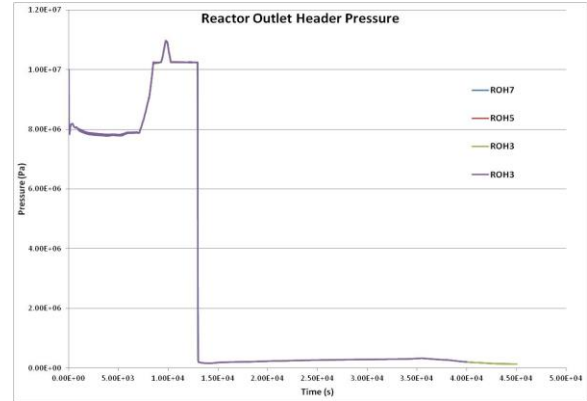


Fig.6: SBO – Case A: PHTS pressure

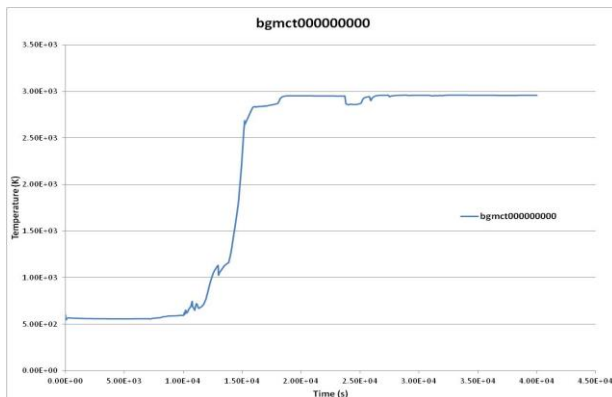


Fig.7: SBO – Case A: Maximum fuel surface temperature ($^{\circ}K$)



Fig.8: SBO – Case A: PHTS inventory

3.2 SBO with SGs depressurization and water addition (Case B)

3.2.1 SBO - Case B1

In this case, the plant behavior will be similar with what was described in Section 2.1.2. The SGs depressurization occurs at 2200s after SBO initiation and 100 s later (a calculation assumption) a total flow of water of 30 l/s (7.5 l/s for each SG) is injected from the dousing tank through the BMW isolation valves into SGs. This flow is maintained constant till 20000s, when it is reduced at 18 l/s till 38000s, the end of the simulation. At the SGs depressurization the SGs water level decreases practically at zero, as it can be seen in Fig. 9.

The evolution of the PHTS pressure is shown in Fig. 10, where it can be observed that after an initial decrease, the PHTS pressure is restored, exceeding the nominal pressure and conducting to the opening of LRVs, which will discharge water from PHTS into DGC. This

PHTS pressure increase is due to the imbalance between the heat transferred from the nuclear fuel to the primary coolant and the heat transferred from the primary coolant to the secondary side of SGs – as a result of insufficient water added into SGs. This water will immediately evaporate and the SGs water level remains zero for a long period of time. The PHTS peak pressure of about 11 MPa is reached at around 13000 s, and then pressure decreases continuously to atmospheric pressure after 30000 s, where it is maintained till the end of the simulation. It can be observed that the water flow reduction at 18 l/s after 20000 s, does not impact much on the PHTS decreasing trend. This can be explained by the decreasing of the decay heat of the nuclear fuel after reactor trip, coming under the possibility of heat removal of SGs. The SGs water level remains very close to zero till 27000s, when it starts to increase slowly but continuously by the end of the simulation, due to reduction of the decay heat.

The heavy water discharged into DGC at the LRVs opening will conduct to the PHTS liquid inventory decrease, as it can be seen in Fig. 12 (total, and for each of the two loops). The big reduction of the PHTS liquid inventory during 10000-20000 s is determined partially by the coolant vaporization (PHTS inventory restart to increase after 15000 s, as a result of steam condensation). Fig. 12 shows the evolution of the maximum fuel surface temperature for this case B1, with SBO mitigated.

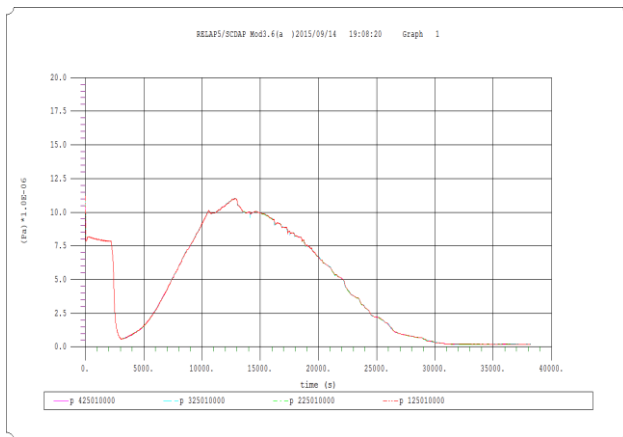


Fig.9: SBO , Case B1: PHTS pressure

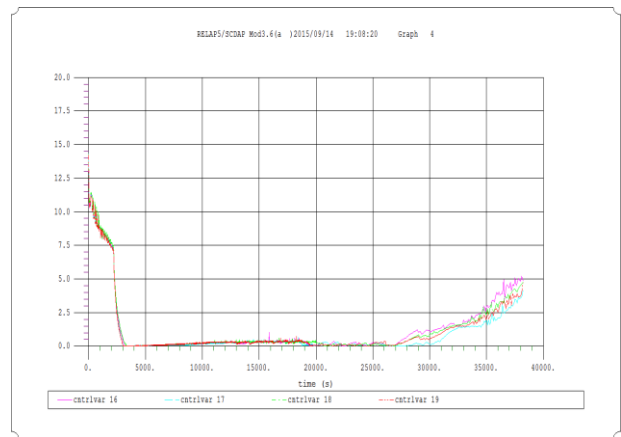


Fig.10: SBO, Case B1: SGs water level

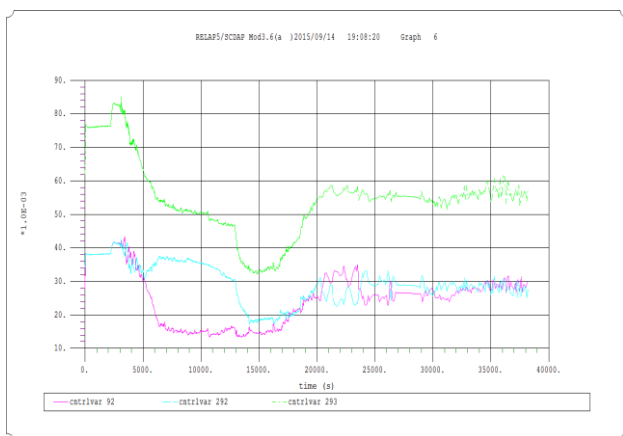


Fig.11: SBO, Case B1: PHTS inventory

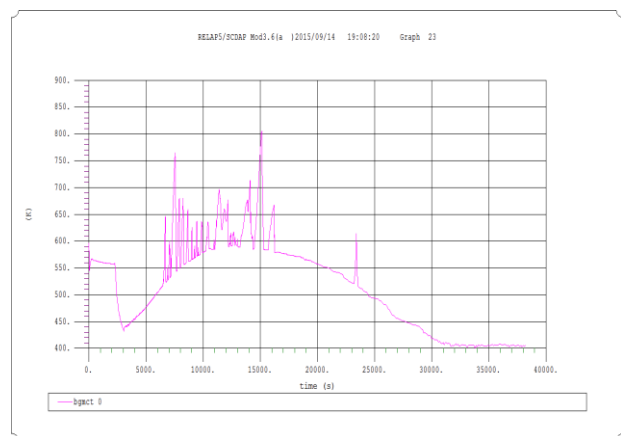


Fig.12: SBO, Case B1: Maximum fuel surface temperature (°K)

The evolution of this temperature (also assimilated conservatively to the cladding temperature) shows that it has a tendency of increasing, on the period of PHTS pressure increase and coolant inventory reduction, but this temperature will start to decrease after

17000 s. It can be observed some spikes in the temperature evolution, the maximum values being under the value for cladding failure (criterion is 800°C, or 1073°K).

From the analysis of this case B1, it resulted that the SGs will become an effective heat sink for the nuclear fuel, as the fuel is well cooled, no cladding or fuel channel failures occur, by comparison with the reference case. However, there are some effects, in the evolution of the PHTS pressure and inventory, that are determined by the insufficient water flow added into SGs after their depressurization. These have in turn an impact on the fuel cooling until 20000 s, when the decay heat is higher. Practically, the reduction of water flow added into SGs after 20000 s has a little impact on the evolution of the most important parameters. The 30 l/s total flow of water added into SGs can be considered as a low limit of this parameter in case of SBO at a CANDU-6 followed by SGs depressurization at 2200 s. An earlier depressurization of SGs, in the same conditions of water flowrate added, will not improve the PHTS and nuclear fuel behavior; on the contrary, it will discharge a large amount of water from steam generators, which could be used for cooling for a period, during SBO.

3.2.2 SBO - Case B2

This case is very similar with the case B1 presented above. The only one difference is that the flow of water added into SGs 100 s after the SGs depressurization is higher, 40 l/s total for all SGs. This flow is almost the maximum flow that can pour gravitationally from the dousing tank into SGs (43 l/s value for maximum water level in dousing tank) through open BMW isolation valves, according to [13]. This value has been mentioned constant in the analysis till the end of the simulation, of 35000 s, to observe the SGs, PHTS and fuel behavior, as well as the differences between this case, case B1 and the reference case. As in the case B1, at the SGs depressurization SGs become empty and the vaporization of the water added into SGs will remove heat transferred from PHTS. The evolution of the SGs water level is shown in Fig. 13. The impact of the implementation of these SBO accident measures on the PHTS pressure can be seen in Fig 12. It can be observed that in case B2 the PHTS pressure decreases initially like in case B1, under 1 MPa, but the higher flow of water added into SGs will not determine the pressure to increase too much; the maximum value of the PHTS pressure (ROH pressure) is 2.4 MPa, much lower than in the case B1, and this pressure does not determine PHTS inventory reduction as LRVs remain close.

The higher flow of water added into SGs in case B2 will lead to a faster recovery of the SGs water inventory. Fig. 13 shows how the SGs water level starts to increase after 13000 s (from the SBO initiation) and at 25000 s the full inventory of the SGs is recovered.

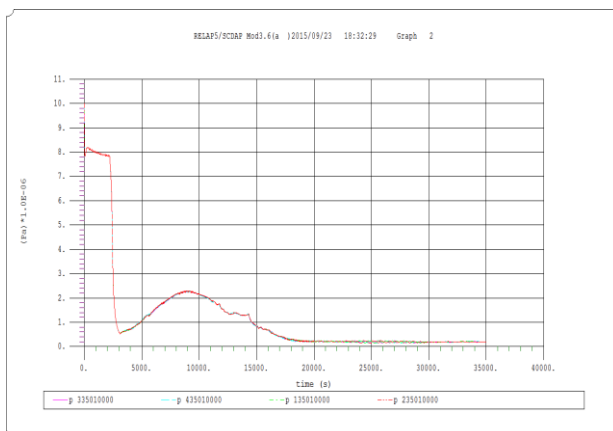


Fig.12: SBO , Case B2: PHTS pressure

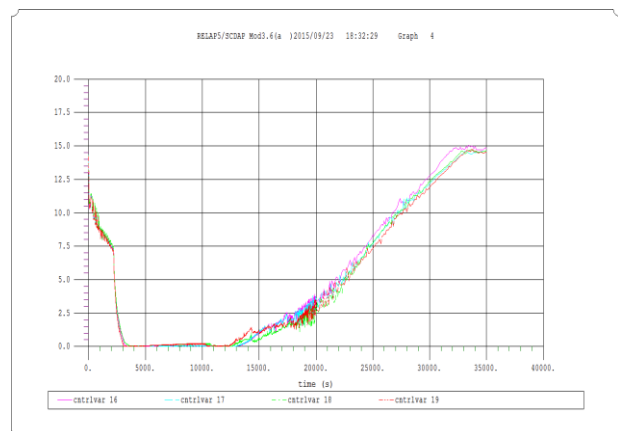


Fig.13: SBO, Case B2: SGs water level

In these conditions and based on the results obtained in case B1 analysis (with reference to the flow reduction after 20000 s), it can be concluded that after 20000 s the flowrate of water added into SGs can be decreased by the operators at less than 18 l/s, at a value necessary to maintain the demanded water level into SGs. This is possible due to the fuel decay heat continuous reduction.

The evolution of the maximum fuel surface temperature can be seen in Fig. 14. This temperature shows that the fuel is much better cooled in case B2 than in case B1, the value of fuel surface temperature going to around 400°K before 20000 s.

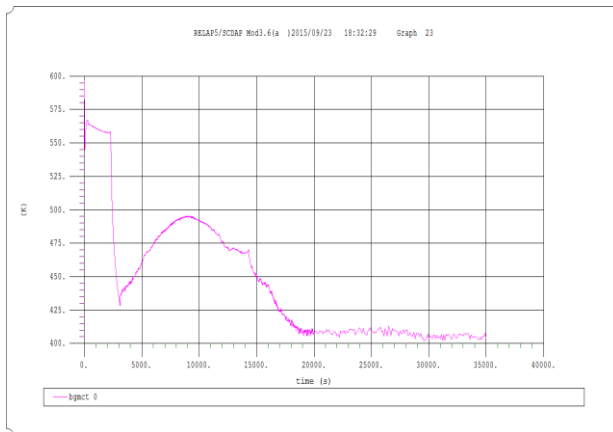


Fig.14: SBO, Case B2: Maximum fuel surface temperature (°K)

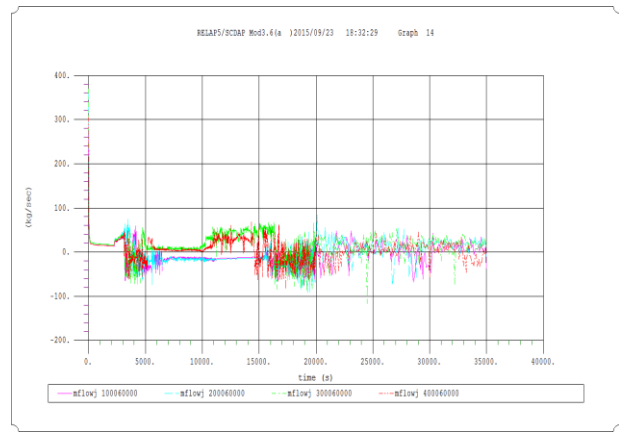


Fig.15: SBO, Case B2: Fuel channels flow

The analysis performed for case B2 shows that the natural circulation removes the decay heat of the nuclear fuel, even if the coolant flow in the fuel channels is unstable in terms of values and direction, as it can be seen in Fig. 15. In this analyzed case B2, the SGs become an efficient heat sink, without any concern regarding PHTS pressure, inventory or the nuclear fuel behavior.

4.0 Conclusions

An investigation has been presented in this paper on the efficiency of the implementation of certain accident management measures in case of a SBO event at a CANDU-6 NPP. The measures considered consist in depressurization of the Steam Generators (SGs) and addition of water into SGs from the dousing tank, in order to ensure the necessary heat sink for the primary coolant that removes heat from the nuclear fuel, in case of SBO.

The analysis has been performed using RELAP/SCDAPSIM/MOD3.6(a) computer code and a CANDU-6 plant model developed during last decades by the Romanian specialists. Two cases have been considered in this analysis for the SBO accident, followed by the SGs depressurization at 2200 s, and with addition of cooling water from the dousing tank 100 s later. The difference between the two cases was the value of the flowrate of water added into SGs after depressurization, as is presented in Section 2.2. For comparison purposes, the unmitigated SBO accident analysis results have also been presented in this paper.

Taking into account the results obtained, presented in Section 3, the following conclusions derive for this study:

- The unmitigated SBO determines the damage of the reactor core. This will trigger serious consequences in terms of fission products and hydrogen release into containment.

- In both cases B1 and B2 analyzed in this paper, with implementation of SBO accident management measures, the steam generators become an efficient heat sink for the nuclear fuel, because the maximum fuel surface temperature remains under the criterion of fuel cladding failure, and the fuel cooling is stable till the end of the simulations performed.
- The flowrate of the water added into SGs after depressurization is an important parameter. His value should be high enough mainly in the first 10000 s, when the decay heat of the nuclear fuel is high, too. The water flow can be reduced progressively once the decay heat decreases.
- According to the results of analysis performed for case B1, a minimum flowrate of 30 l/s of water should be provided to SGs after SGs depressurization.
- An optimum flowrate of 40-43 l/s of water is recommended to be added into SGs after depressurization, better until the SGs inventory is almost recovered. The water flow can be then reduced by the operator at the value necessary to maintain the SGs level (less than 18 l/s, as resulted from the analysis of case B1). This can ensure the necessary water for SGs secondary side for a long period of time. The calculation of this period was not included in the scope of the present analysis.
- There are some limitations related to the models in RELAP/SCDAPSIM computer code (some SCDAP models are not CANDU specific), that could affect the analysis of case A, after the beginning of the fuel channel disassembly.

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