ANALYSIS OF THE CANDU-6 PLANT BEHAVIOR IN CASE OF VERY LATE STEAM GENERATORS DEPRESSURIZATION AND WATER INJECTION FOLLOWING A STATION BLACK-OUT ACCIDENT

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This paper analyses consequences of a CANDU-6 NPP in case of a Station Black-Out (SBO) accident, considering both the reference case, with no any heat sink credited for nuclear fuel cooling and the situation when an accident management measure is implemented for prevention of a severe accident. This measure consisting in the steam generators (SGs) depressurization followed by addition of water into steam generators from the dousing tank, has been considered in the analysis to be implemented at different moments after SBO initiation, but after all SGs dry-out in order to determine the efficiency of steam generators as a heat sink in case of SBO, including for those situations when this management measure is implemented late and very late - much after steam generators dry-out. Selected accident sequences have been analysed using RELAP/SCDAPSIM.MOD3.6(a) computer code.

Keywords: CANDU, Station Black-Out (SBO), steam generator (SG) depressurization, RELAP/SCDAPSIM

1. Introduction

The paper is related to the analysis of the CANDU-6 Nuclear Power Plant (NPP) behavior in case of a Station Black-Out (SBO) initiating event. This event that consists in loss of all alternative electrical power sources in a CANDU-6 NPP (with reactor at full power and equilibrium fuel conditions), can progress in some accident conditions, when the cooling of nuclear fuel from the core can't be ensured anymore, to a severe accident with core damage. This type of accident that can also result from a common cause event, as a seismic event exceeding the design basis of the NPP, has been intensively studied for all types of NPP projects, including CANDU type, due to its potential consequences on the reactor core and release to the containment of fission products and hydrogen, for the

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unmitigated accident (or reference case). Some of the studies performed for CANDU plants for severe accidents, including those resulted from SBO, are shown in the References section ([4], [9], [12], [13], 14]). Such studies have been performed both before but also after Fukushima Daiichi accident in March 2011, including in the frame of so-named “stress tests”, safety re-assessments, performed for all European NPP, and in general for all NPPs in the world, as is shown in [11].

After the loss of all heat sinks that have been designed to ensure the cooling of the nuclear fuel in the core during normal operation and in emergency conditions (for design basis accidents), following loss of power due to SBO initiating event, different alternative cooling means and accident management measures that have been provided both by design but also by safety improvements, can be used at CANDU NPP’s. The main accident mitigation measure in case of a SBO event is the depressurization of Steam Generators (SGs), by opening at least 8 Main Steam Safety Valves (MSSVs), and injection of water into SGs, from Dousing Tank or by Emergency Water System (EWS). SGs depressurization can be done automatically or by the operator action but in both cases MSSVs, that ensure SGs depressurization, have to be blocked in open position in order to maintain a path for continuous steaming to environment for heat removal. Dousing tank can ensure around 2000 m^3 of water to SGs, that is enough for at least 27 hours (according to [11]).

The effectiveness of this heat sink and its importance has been demonstrated until now by different specific analyses, performed for CANDU reactors (as [11], [13], etc.). These studies have been performed with CANDU specific computer codes, as CATHENA and MAAP4-CANDU, and considered that the SGs depressurization occurs at 30-40 minutes after the SBO initiation.

This paper presents the effects of an unmitigated SBO accident at a CANDU NPP, as a reference case, analyzed in order to compare the evolution and the effects of the resulted severe accident with the cases where a management measure is implemented, namely the late and very late SGs depressurization followed by the cooling water addition into SGs, at enough flow (at least 30 l/s). This implementation of this SBO accident management measure is analyzed for more cases, in a sensitivity study, in order to determine the limitations regarding the timing of SGs depressurization and the efficiency of this measure in each case, with the aim to have a validated conclusion on the application of this management measure and to know the expected effects. This study considers late depressurization of SGs with addition of cooling water starting at 2 after SBO initiation (at the first SG dry-out) and very late after 2.5 h, when the Primary Heat Transport System (PHTS) inventory starts to be reduced by discharging heavy water into degasser condenser tank (DGC). This type of study, has not been performed before, or evidence that it was done has not been found.

The main purpose of this paper is to determine if the depressurization of the SGs, performed late and very late, after the SGs dry-out, will bring SGs back as a heat sink for the primary coolant that extract decay heat from the reactor core. A limitation in time for the application of this SBO accident management measure could by the loss of a significant part of the PHTS inventory through the Liquid Relief Valves (LRVs), provided for PHTS overpressure protection.

2. Analysis methodology

The paper presents the analysis of the consequences of the severe accident resulted from the unmitigated station blackout (SBO) event and the case when the depressurization of SGs is performed, immediately (100 s later) followed by the addition of water into SGs. For this second case, the analysis is performed considering that the SGs depressurization is performed after first SG dry-out, in the following conditions:

a) *SGs depressurization at 7200 s and a constant flow of 30 l/s, total for all SGs, added until the end of the simulation (20000s)*
b) *SGs depressurization at 9000 s and a constant flow of 30 l/s , total for all SGs, added until the end of the simulation (40000s)*
c) *SGs depressurization at 10800 s and a constant flow of 30 l/, total for all SGs, added until the end of the simulation (40000s).*

The simulation periods of time have been selected considering the criterion of 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 a SBO.

The RELAP/SCDAPSIM/Mod 3.6 code ([1], [2], [3], [4]) was used for the analysis of the mentioned SBO accident sequences. RELAP/SCDAPSIM has been designed to describe the overall PHTS thermal hydraulic response and core behavior under normal operating conditions or under design basis or severe accident conditions. The RELAP5 models calculate the overall PHTS thermal hydraulic response, control system behavior, reactor kinetics, and the behavior of special reactor system components such as valves and pumps. The SCDAP models calculate the behaviour of the core and vessel structures under normal and accident conditions [1]. These models calculate the progression of damage in the reactor core: heat-up, oxidation and meltdown of fuel rods and control rods, ballooning and rupture of fuel rod cladding, release of fission products from fuel rods and disintegration of fuel rods into porous debris and molten material. 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 [14]. Mod3.5 contains also important improvements, comparing with the previous Mod3.4 version of the code [3].

3. CANDU-6 plant models

CANDU-6 reactor is a Canadian designed Pressurized Heavy Water Reactor (PHWR) type, having a gross capacity of about 700 MW(e), using heavy water both for moderator and for primary coolant, in separate circuits, and natural uranium for fuel. CANDU-6 reactor consists in a large Calandria Vessel (CV), as reactor vessel, penetrated by 380 fuel channels. Each fuel channel, composed by a Zirconium - Niobium pressure tube (PT) and a zirconium alloy Calandria tube (CT), separated by an annulus system space filled with carbon dioxide, has inside 12 fuel bundles of 0.5 m, with 37 fuel elements each of them. The fuel channel has two end fittings at the ends of the fuel channel, from stainless steel. PT and CT are around 6.3 m long. The core of CANDU-6 reactor is divided vertically into two halves, each half representing a separate coolant circuit, or loop. Each loop has two eight shape passes through the core. Each pass has a reactor inlet, a reactor outlet, one primary pump, one SG and 95 feeders connected at end-fittings on each side of reactor vessel, as well as 95 fuel channels. 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 PHTS in 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.

The CANDU-6 plant model developed for the analysis of a CANDU plant behaviour, during accident conditions - design basis and severe accidents, by using RELAP/SCDAPSIM computer code has been continuously improved by the Romanian specialists (as [5], [6], [7], [8], [9], etc.). The nodalization scheme, from Fig. 1, contains the representation for the two loops with fuel channels, inlet headers, feeders and end fittings, outlet headers, feeders and end fittings, primary coolant pumps, pressurizer, steam generators and associated pipes. 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. A detailed description of RELAP/SCDAP model of CANDU-6 plant, the analysis methodology, assumptions, and failure criteria used in analysis of a SBO accident can be found in [4]. CANDU-6 reactor is modelled both with RELAP5 and SCDAP components, in this SBO simulation. 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 2, 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 (CV-OPRD), that are modelled as a single trip valve that connects CV with reactor containment.

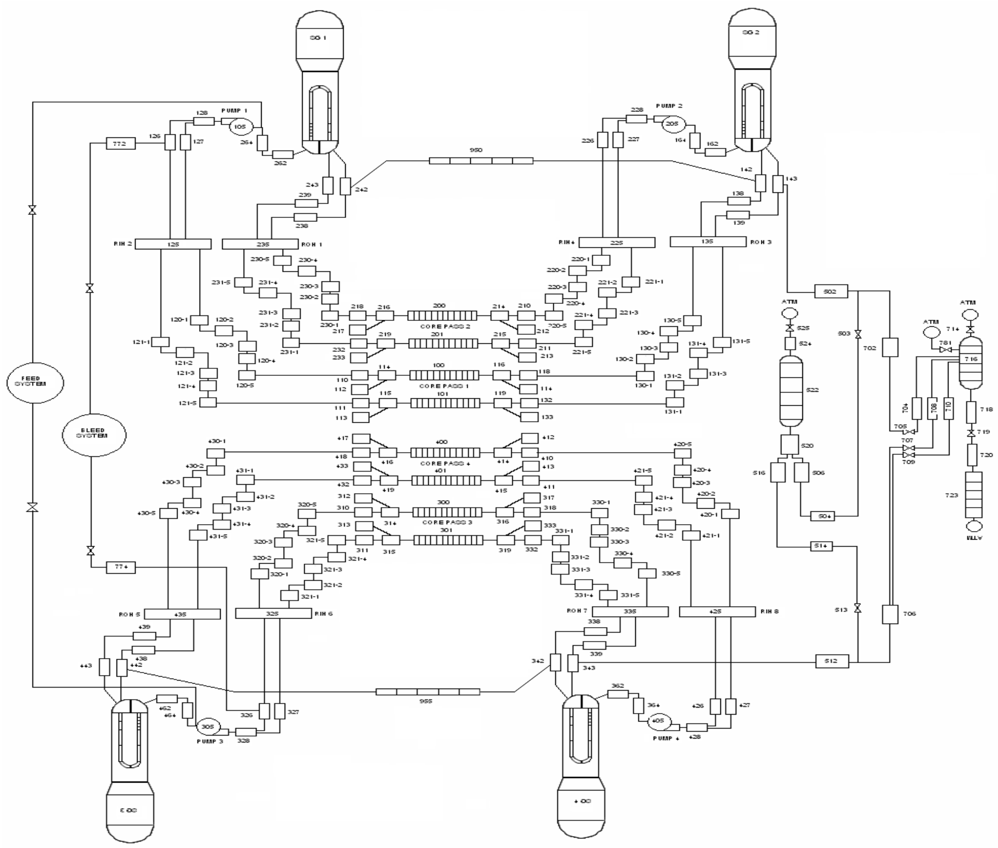

Fig. 1: CANDU-6 – PHTS nodalization scheme

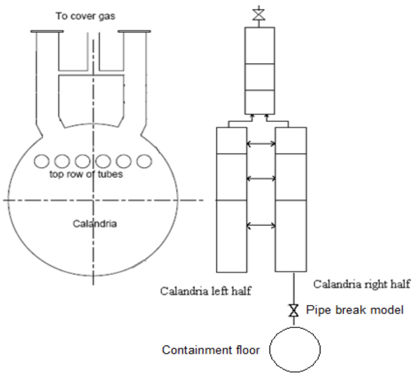

Fig. 2: Calandria Vessel model

4. SBO analysis –results for the reference case

The unmitigated SBO accident (reference case) will lead to the fuel channel overheating and after some fuel channels rupture (10, according to the assumption used in [4]), the primary coolant is discharged into moderator, conducting both to the depressurization of the PHTS and also to the pressurization of the moderator into CV. After the rupture disks of the CV ducts, moderator is expelled from the CV into reactor containment atmosphere, and without any other source of water for fuel cooling, reactor core damage occurs. This will trigger serious consequences in terms of fission products and hydrogen release into containment. For the purpose of this paper, the SBO analysis is carried out until the core collapse occurs, for the case without credit for any heat sink or mitigation measure. The behaviour of the debris on the bottom of CV, the behaviour of the molten pool or CV were not considered, as well as the containment behaviour as a result of steam and water discharge from PHTS, CV or other pressurization sources. The evolution in time of the unmitigated SBO accident (reference case) is 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°K)	3,971	14301
Core colaps on the calandria vessel bottom	5,861	21100

The evolution of the SGs water level is shown in Figure 3 and the evolution of the PHTS pressure is shown in Fig.4. Fig. 5 shows the evolution of the maximum fuel surface temperature for the unmitigated SBO. The evolution of the PHTS inventory in case of unmitigated SBO accident can be seen in Fig. 6. From the analysis of this case resulted that, at the moment of fuel channel break, the maximum pressure is 20.66 bars at the bottom of CV 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.

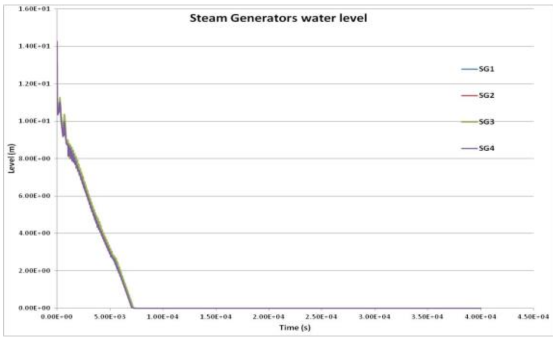

Fig.3: Unmitigated SBO at CANDU-6:
SGs water level

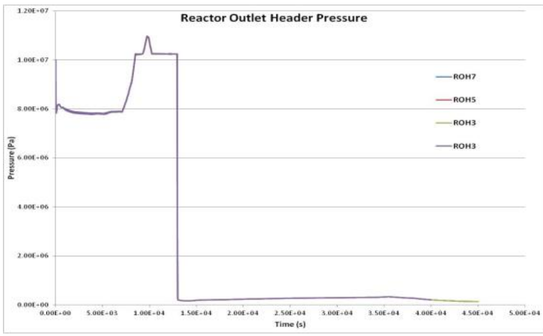

Fig.4: Unmitigated SBO at CANDU-6:
PHTS pressure

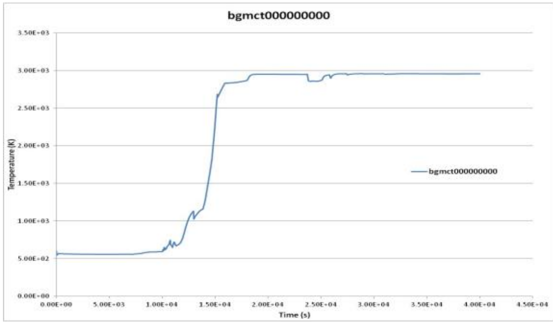

Fig.5: Unmitigated SBO at CANDU-6:
Maximum fuel surface temperature (K)

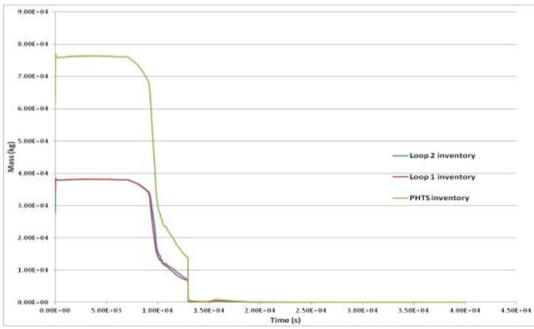

Fig.6: Unmitigated SBO at CANDU-6:
PHTS inventory

5. SBO analysis – cases with SGs depressurization and water injection

The case with implementation of the SBO management measure consisting in SGs depressurization at 9000s (2.5 h after SBO initiation), followed by the water addition into SGs (100 s later - analysis assumption), is considered as representative for the study performed. At 9000 s after SBO initiation, all SGs lost their inventory for a long already period of time (the last SG dry-out at 7510 s), and LRVs have open (at 8970 s, according to Table 1) and started to discharge heavy water into DGC. In this case, following the SGs depressurization and start of water addition 100s after SGs depressurization, the SGs water level will remain almost zero for a while, as the water injected is immediately evaporated; once the heat transferred from the primary coolant to SGs decrease enough, the SGs inventory start to recover, as it can be seen in Fig. 7. The PHTS pressure has a peak of about 11 MPa for a short period of time but the addition of cold water into SGs will reduce the PHTS pressure continuously, to almost atmospheric value, as it is shown in Fig. 8. The efficiency of SGs as a heat sink can be observed in the evolution of the maximum fuel surface temperature, which is also assimilated to the cladding temperature (conservative assumption). Fig. 9 shows that after the SGs depressurization and start of water addition, this temperature decreases continuously and remains low and stable by the end of the simulation. The maximum value of the cladding temperature is in this case lower than the criterion used for the cladding failure (1073°K, or 800°C). Fig. 10 shows the evolution of the fuel channels coolant flow. It can be seen that at the moment of SGs depressurization the channels flow was almost zero and after SGs depressurization and water injection into SGs, the condensation of heavy water inside PHTS leads to increasing of channel flow, without a stable direction but ensuring an efficient nuclear fuel cooling.

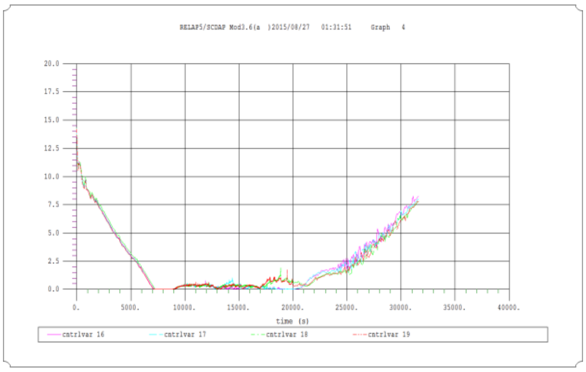

Fig.7: SBO with SGs depr. at 9000s: SGs water level

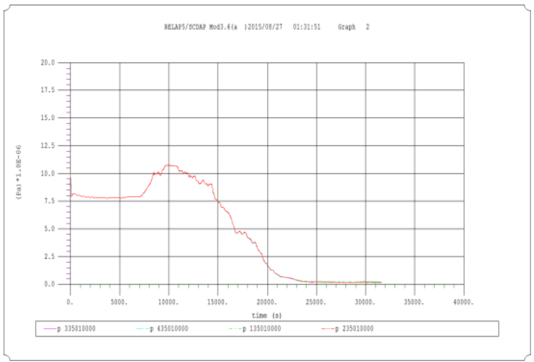

Fig.8: SBO with SGs depr. at 9000s: PHTS pressure

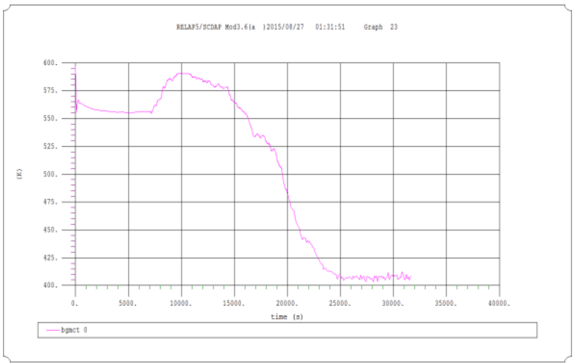

Fig.9: SBO with SGs depr. at 9000s: Maximum fuel surface temperature (K)

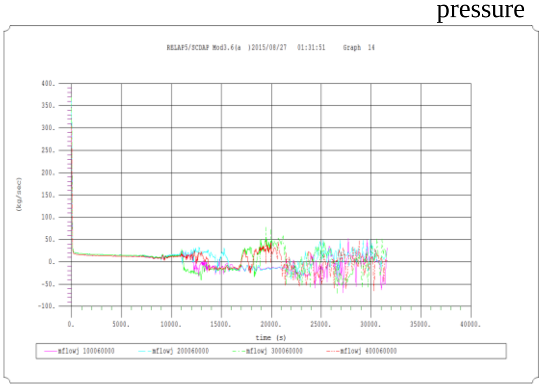

Fig.10: SBO with SGs depr. at 9000s: Fuel channels flow

The most important parameter in demonstrating the effectiveness of SGs as a heat sink, after implementation of the measure of SGs late and very late depressurization followed by water addition into SGs, is the nuclear fuel temperature. Therefore, the evolutions of the maximum fuel surface temperature for the other analysed cases, with SGs depressurization at 7200s and at 10800s, respectively (followed in both cases by the addition of water into SGs) are shown in Figure 11 and Figure 12. These figures show that the maximum fuel surface temperature (and cladding temperature) is lower than the cladding failure criterion of 1073°K, until the end of the simulation periods, having in the same time a decreasing trend. In the case of SGs depressurization at 10800 s (3 h after SBO initiation), there are some spikes in fuel temperature evolution, due to significant reduction of the PHTS inventory, as a result of LRVs discharge. Therefore, the SGs depressurization for SBO accident management at around and mainly after 3 h, when PHTS inventory is significantly reduced is not recommended.

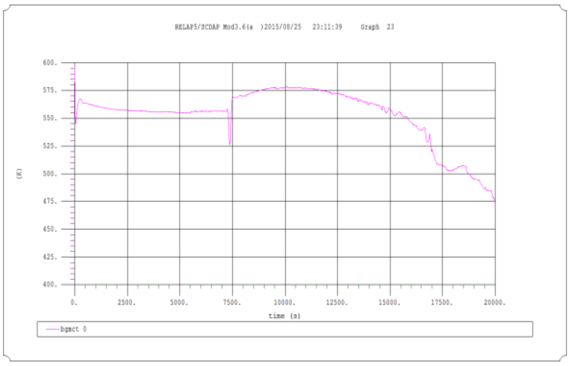

Fig.11: SBO with SGs depr. at 7200s:
Maximum fuel surface temperature (K)

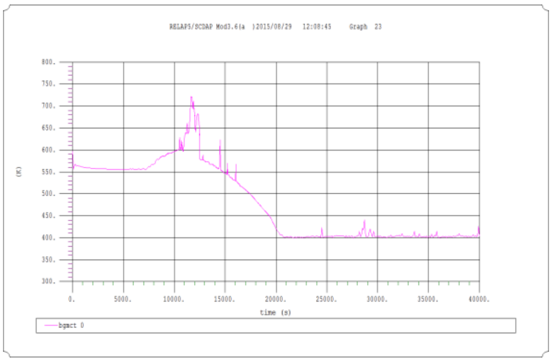

Fig.12: SBO with SGs depr. at 7200s:
Maximum fuel surface temperature (K)

6. Conclusions

This paper adds an important contribution to the analysis, and validation through calculations, of one of the most important management measures to prevent a severe accident following a SBO - an accident beyond the originally design basis of the CANDU NPP, by ensuring the nuclear fuel cooling. This analysed management measure is represented by the late and very late depressurization of SGs (automatically or manual), followed immediately by the addition of water into SGs from the dousing tank or EWS. This measure is expected to be done normally at 30-34 minutes after SBO initiation but this study considered that it is applied only after 7200s, 9000s and 10800s after SBO.

The RELAP/SCDAPSIM/Mod 3.6 code ([1], [2], [3], [13]) was used to perform the analysis of the consequences of a SBO event at a CANDU-6 plant, in the case without credit for any heat sink for the primary coolant and also for the case when SGs depressurization is credited, late and very late, followed by water addition in SGs, as a measure of SBO accident management.

The sensitivity study performed indicated that SGs become an efficient heat sink, in case of SBO with SGs depressurization and injection of cooling water into SGs from the dousing tank or EWS, in all analysed cases, implementation of this management measure late and very late succeeding to bring back the SGs as a heat sink, removing the decay heat produced by the nuclear fuel and maintaining the core integrity and cooling. It is not recommended to apply this SGs depressurization measure with addition of water into SGs after 3 hours from the SBO initiation, without a specific thermo-mechanical analysis of the SG tubing area due to the risk of a potential SGs tubing failure. This could create a direct path for fission products release into the environment outside the reactor building, through the open MSSVs.

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