Analysis of severe core damage accident progression for the heavy water reactor

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Abstract In this study, the severe accident progression analysis of generic Canadian deuterium uranium reactor 6 was preliminarily provided using an integrated severe accident analysis code. The selected accident sequences were multiple steam generator tube rupture and large break loss-of-coolant accidents because these led to severe core damage with an assumed unavailability for several critical safety systems. The progressions of severe accident included a set of failed safety systems normally operated at full power, and initiative events led to primary heat transport system inventory blow-down or boil off. The core heat-up and melting, steam generator response, fuel channel and calandria vessel failure were analyzed. The results showed that the progression of a severe core damage accident induced by steam generator tube rupture or large break loss-of-coolant accidents in a CANDU reactor was slow due to heat sinks in the calandria vessel and vault.

Key words Severe Accident, steam generator tube rupture, large break loss-of-coolant accidents

1 Introduction

The CANDU6 (Canadian deuterium uranium reactor) is PHWR (pressurized heavy water reactor). Its main feature is the horizontal core consisting of fuel bundles surrounded by calandria tubes inside the pressurized heavy water. The 380 fuel channels are arranged in a 22×22 circular array, and each channel contains 12 short fuel bundles. Two symmetric loops are connected to a pressurizer by an “8”-shaped configuration. The moderator contained in the moderator tank is surrounding the calandria tubes to provide a potential heat sink if a loss of coolant accident coincident (LOCA) occurs with the failure of emergency core cooling system[1].

Atomic Energy of Canada, Ltd. (AECL) has reported the mechanism for CANDU core melting, and developed the CANDU-specific computer codes to predict the progression of severe core damage[2]. The experimental facilities, built in Korea, India and Romania, have been used to investigate the typical phenomena[3–5]. In China, we mainly focused on sequence analysis and mitigation measures for pressurized water reactors. Up to now, the severe accident phenomena of the heavy water reactor have not been analyzed yet in detail. Therefore, it is necessary to analyze the severe scenario for a CANDU station.

In this study, the preliminary severe accident progression analysis for a generic pressurized heavy water reactor was performed using the code for integrated severe accident analysis. The selected accident sequences were steam generator tube rupture and large LOCA with an assumed unavailability for several critical safety systems.

2 Analysis model

2.1 Description of plant model

AECL uses the code for integrated severe accident analysis to simulate response of the CANDU station to a severe core accident, and develops the fuel bundles model, fuel heat-up, pressure-tube and calandria-tube failures, core collapse model in experiments, and core disassembly facility[6–7].

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Fig.1 shows the primary heat transport system loops. Each loop has two cores with coolant flowing in opposite directions, and is divided into 14 nodes, representing pump discharge lines, reactor inlet headers, reactor outlet headers, inlet piping of steam generators, hot leg tubes of steam generators, cold leg tubes of steam generators and pump suction lines after the cold leg tubes of steam generators .

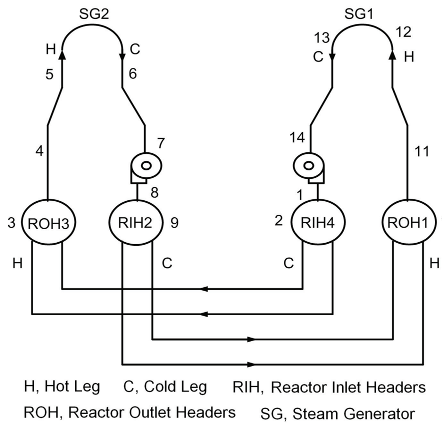

Fig.1 Nodalization of the primary heat transport system.

The primary heat transport system loops, which close the isolation valves of the pressurizer loop, are isolated with the LOCA signal. The steam generators are rapidly cooled to depressurize the heat transport system via main steam safety valves, that is, crash cool-down system.

A simplified nodalization represents the total 380 fuel channels. The 22×22 fuel channels are divided into two symmetric loops from the vertical axis, and 6 vertical nodes. The 12 bundles in a fuel channel and a fuel bundle of 37 fuel elements are modeled as 12 axial nodes and 7 concentric rings. The pressure tubes and calandria tubes are modeled as concentric rings. Fig.2 shows a fuel channel, which is divided into nine rings. Key parameters for CANDU6 plant operation is given in Table 1.

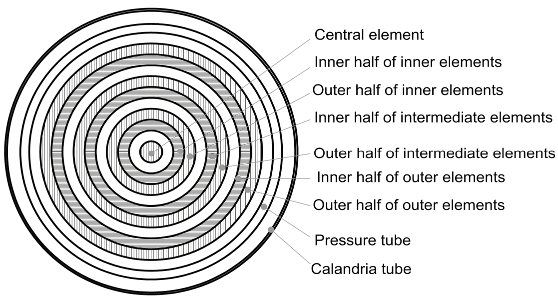

Fig.2 Section plane of reactor fuel bundle nodalization.

2.2 Analysis assumption

The severe accident sequences causing core damage were first selected. This is appropriate for identifying accident scenarios, and investigating preventive and mitigatory actions[8]. The accident sequences were steam generator tube rupture and large LOCA due to the CANDU probabilistic safety assessment and typical phenomena.

Table 1 Key parameters for the CANDU 6 station.

Thermal reactor power, 2064 MW
Reactor outlet header pressure, 9.99 MPa(a)
Reactor inlet header coolant temperature, 539 K
ROH coolant temperature, 583 K
Pressure of a secondary side steam generator, 4.7 MPa(a)
H_2O inventory in each secondary side steam generator, 38 t
H_2O inventory in calandria vessel, 230 t

We assumed that the shutdown cooling system, moderator cooling and shield cooling systems, and main and auxiliary water-feed systems were unavailable; the primary heat transport system loop was creditably isolated; and the main steam isolation valves were closed after accident. The available main steam safety valves were opened and closed at 5.24 MPa(a) to relieve the pressure, and operator interventions were not considered.

The initiating event of steam generator tube rupture was a guillotine rupture for 10 steam generator tubes, which was equivalent to the total break area of 0.0029 m^2, and located in the primary heat transport system loop 1 at the top of the steam generator tube sheet. The severe steam generator tube rupture accident scenario was simulated without the emergency core cooling system, and crash cool-down system was unavailable according to the CANDU probabilistic safety assessment.

The event of large LOCA was a guillotine rupture for reactor outlet headers in Loop 1 of the primary heat transport system, followed by a double-sided blow-down of the primary heat transport system coolant. A reactor shutdown was immediately triggered upon accident. The high pressure and medium pressure injection of the emergency core cooling system, and crash cool-down system were available, but the low pressure injection of the emergency core cooling system was unavailable.

3 Steam generator tube rupture accident

3.1 PHTS and fuel bundle response

When the postulated initiating event of steam generator tube rupture accident scenario is imposed, the pressures decrease rapidly in the primary heat transport system loops, and in the pressurizer, because of guillotine break, the loss of fission power from the reactor core, and the heat sink feature of the steam generators, are shown in Fig.3. At 183 s when the pressure of the primary heat transport system drops to 8.6 MPa, the reactor shutdown occurs. The LOCA signal is initiated at about 284 s. The loop valves are used to isolate the pressurizer from Loops 1 and 2 resulting in their different behaviors. At the main steam safety valves failure point, the pressure of Loop 1 reaches a relative constant at about 5.2 MPa(a) due to the continuous loss of inventory from the steam generator U-tube rupture. The pressure decreases in Loop 1 after the rupture of pressure- and calandria-tube rupture. The main sequence progression is given in Table 2.

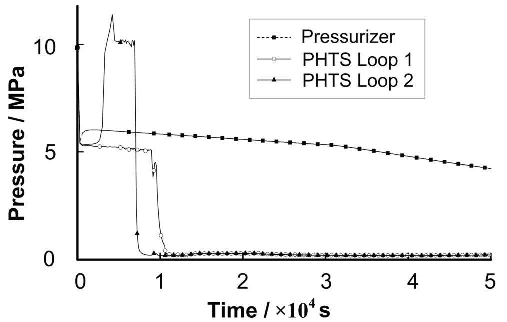

Fig.3 Pressure in the primary heat transport system (PHTS) loops and pressurizer.

Table 2 Sequence of steam generator tube rupture.

Events	Time / s
Reactor shutdown	183
Generation of LOCA signal	284
Pressureizer and loop isolation	304
Dried secondary water of steam generator in Loop 2	2986
Dried secondary water of steam generator in Loop 1	10202
Pressure-tube and calandria-tube rupture in Loop 2	7231
Pressure-tube and calandria-tube rupture in Loop 1	9043
Calandria vessel rupture-disk burst	9244
Beginning of the core disassembly	10202
Calandria vessel fail due to creep	154039

In Loop2, the pressure drops initially and increases rapidly because the secondary water of steam generators is drying out until the liquid relief valves are opened. The coolant releases to the containment. At about 7231 s, a pressure-tube and calandria-tube rupture occurs in Loop 2, and the pressure drops rapidly to atmosphere.

As shown in Fig.4, the temperatures for pressure- and calandria-tubes, and the central fuel ring in Loop 1 with the Channel 4 and Bundle 7, remain constant up to 9100 s, and start to increase at the dried fuel channel, because the coolant is boiled off, and lost by the steam generator U-tube rupture. When the pressure tube is at about 900 K, the fuel channel is ruptured at 9043 s. The segments of fuel channel are disassembled into the suspended debris at 10202 s, when the segment fuel, and temperatures of the pressure- and calandria-tubes, are no longer tracked.

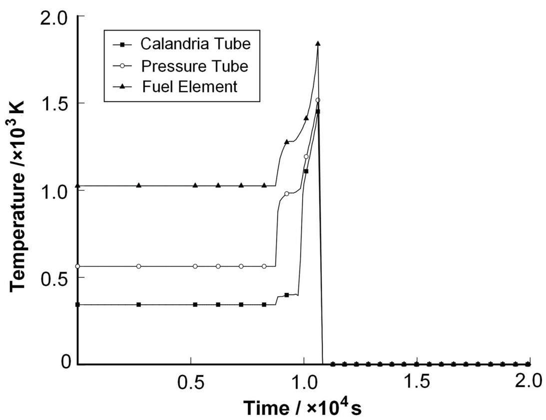

Fig.4 Temperature of the pressure tubes, calandria tubes, and fuel elements of Channel 4, Bundle 7 in Loop 1.

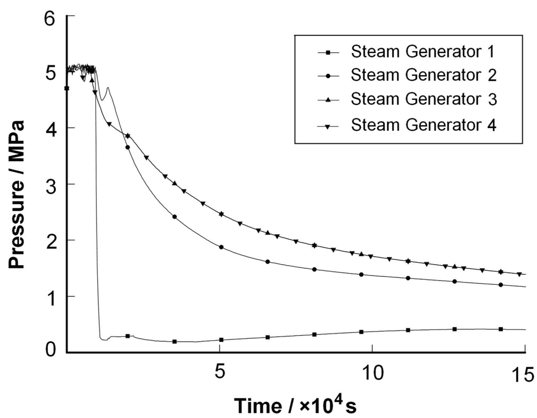

Fig.5 Pressure of the four steam generators.

3.2 Response for steam generator

Fig.5 shows that pressure of the broken steam generator 1 in Loop 1 increases initially due to the flow of the primary side via the ruptured tubes, and causes the main steam safety valves to open for steam discharge. The secondary side pressure oscillates at the main steam safety valve failure point. The Loop 1 pressure tube fails at 9043 s, and pressure of Steam Generator 1 drops rapidly to atmosphere. The pressure of the intact Steam Generator 2 in Loop 1 falls gradually because of its availability of heat sink. When the heat transfer from the primary heat transport system to Steam Generators 3 and 4 in Loop 2 makes the secondary water boil off, the increased pressure oscillates on opening and closing points of the main steam safety valves. At the pressure-tube and calandria -tube failure at 7231 s in Loop 2, the pressures of Steam Generators 3 and 4 decrease gradually.

Fig.6 shows secondary water level of the steam generators. The water level of Steam Generator 1 increases initially, because the primary coolant flows into the secondary side from the broken tubes, followed by a slow decrease and a quick drop due to boil-off, and discharges to the calandria vessel induced by pressure tube and calandria tube rupture. The water level of Steam Generator 2 has a rapid decrease at first and remains constant relatively due to the heat-transfer from the primary heat transport system to the generator's secondary side, and the primary hot coolant discharges into the calandria vessel induced by pressure tube and calandria tube rupture. The water level of Steam Generators 3 and 4 drops rapidly due to the opening of the main steam safety valves, and can not be used as heat sinks to remove the heat at about 3029 s.

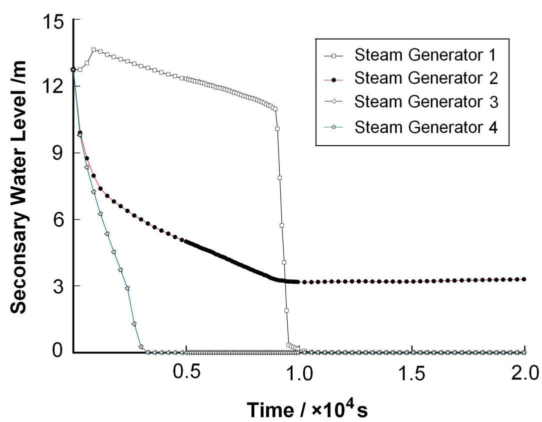

Fig.6 The SG secondary water level.

3.3 Response of calandria vessel

The water level of initial moderator is 7.9 m at 342 K. At beginning of the accident, the temperature and pressure increase, the coolant of primary heat transport system discharges into the calandria vessel at both ends of the ruptured fuel channel, because the cooling failure and the core heat-transfer cause the pressure- and calandria-tube rupture (Fig.7). At 9244 s, pressure of the calandria vessel reaches the rupture-disk burst point (Table 2). The moderator level decreases rapidly due to the expulsion, and discharges gradually into the containment due to boil-off via heat transfer of the core.

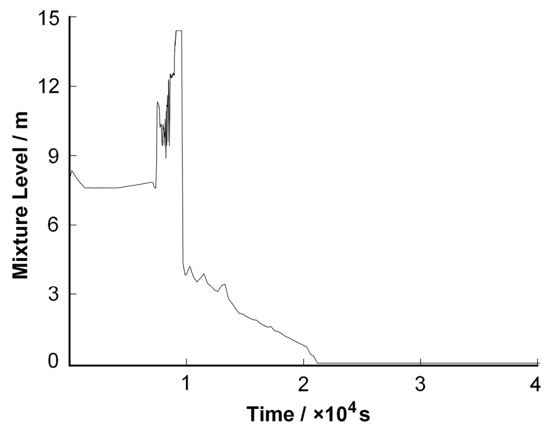

Fig.7 Mixture level of the calandria vessel.

The water in the reactor vault acts as a heat sink with good heat transfer, and cools external wall of the calandria vessel due to moderator depleting at 20700 s. Fig.8 shows that wall temperature of the calandria vessel is relatively low with saturation and boil-off of the water in the calandria vault, the vessel wall begins to boil off, and fails at 154039 s.

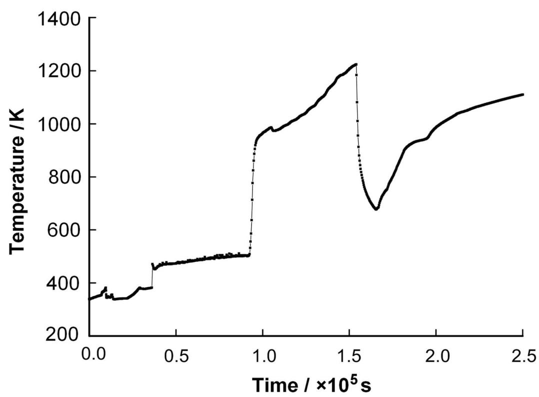

Fig.8 Temperature of calandria vessel wall.

4 The large LOCA

Assuming that a large LOCA scenario is initiated by a guillotine rupture of reactor outlet headers in Loop 1 of the primary heat transport system, the major event and processes are shown in Table 3 and Figs. 9–13.

Fig.9 shows the pressures in the primary heat transport system loops and pressurizer. They decrease faster in Loop 1 than in Loop 2 due to break of the reactor outlet headers. At Loop 1 pressure of 5.52 MPa(a), the isolation valves are closed, and the high pressure injection of the emergency core cooling system acts mainly to Loop 1, because of the lower pressure in Loop 1 than in Loop 2. At the action point of the emergency core cooling system, the medium pressure injection starts at about 98 s, and water is injected mainly into Loop 1 of the primary heat transport system. When the medium pressure injection terminates, the pressure in Loop 2 increases up to the point of liquid relief valves action and oscillates at about 10 MPa(a) because of the periodical opening and closing. Then Loop 2 pressure drops rapidly due to fuel channel rupture at about 30346 s.

Table 3 Sequence of the large LOCA.

Events	Time / s
Reactor shutdown	0
High-pressure injection of emergency core cooling system	6
Pressureizer and loop isolation	24
Crash cool-down system is on	34
medium-pressure injection of emergency core cooling system	99
Dried secondary water of steam generator in Loop 2)	2808
Dried secondary water of steam generator in Loop 1	18809
Pressure and calandria tube rupture in Loop 2	31320
Beginning of the core disassembly	32183
Calandria vessel failed due to creep	204649

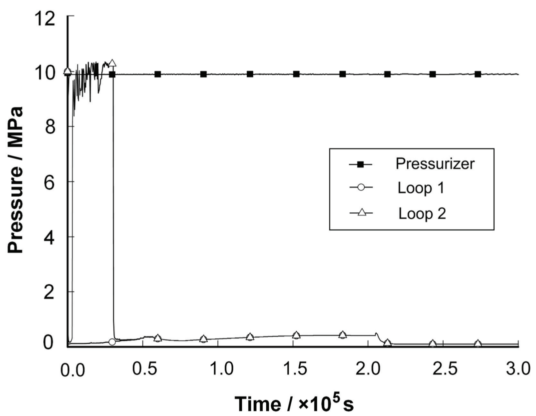

Fig.9 Pressure in the PHTS Loops and Pressurizer.

Fig.10 shows the temperature of pressure tubes, calandria tubes and fuel Ring 7 in Loop 2. At high pressure, the pressure tube temperature reaches about 900 K, and fuel channel ruptures at 31320 s. The fuel channel disassembles at about 32183 s.

Fig.11 shows the water level in the four steam generators. On receiving the LOCA signal, the main steam safety valves of the steam generators open to initiate crash cool-down at 34 s, resulting in the blow down through the open valves, and the water boils off. The water levels are higher in Steam Generators 1 and 2 than that in Steam Generators 3 and 4 because the former loops are broken, and are dried out at 18809 and 2808 s. When a little of emergency core cooling water is injected into Loop 2, the coolant is hotter in Loop 2 than Loop 1, and its water level boils off faster.

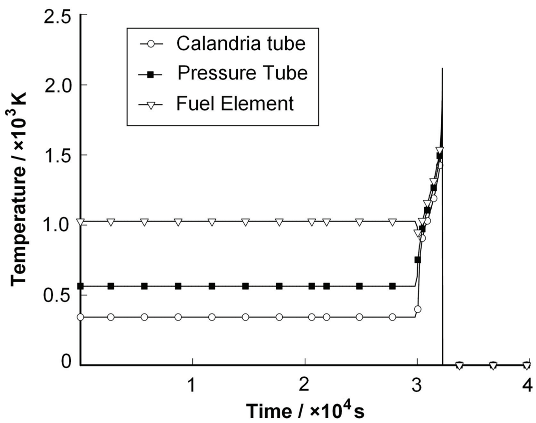

Fig.10 Temperature of pressure tubes, calandria tubes and fuel Ring 7 in Loop2.

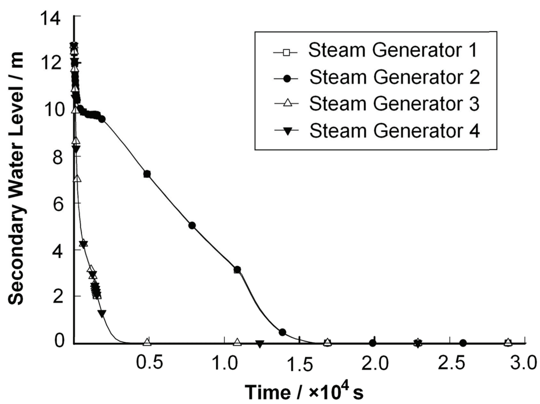

Fig.11 Secondary water level of the steam generators.

When a pressure- and calandria-tube rupture occurs, the coolant discharges into the calandria vessel. The burst of rupture disks results in moderator expulsion, lowering the mixture level in the calandria vessel (Fig.12). After the initial rapid expulsion, the moderator continues to discharge into the containment, resulting in moderator boiling off and depleting. Then the water in the reactor vault acts as a heat sink and cools the external wall of the calandria vessel. As shown in Fig.13, the wall temperature is relatively low because of good heat transfer over a long time period. When water temperature in the reactor vessel reaches saturation and boils off, the calandria vessel wall heats up and fails eventually at about 204649 s due to creep.

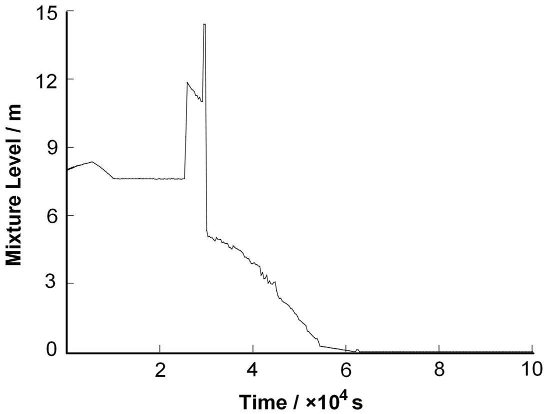

Fig.12 Level of water-steam mixture in the calandria vessel.

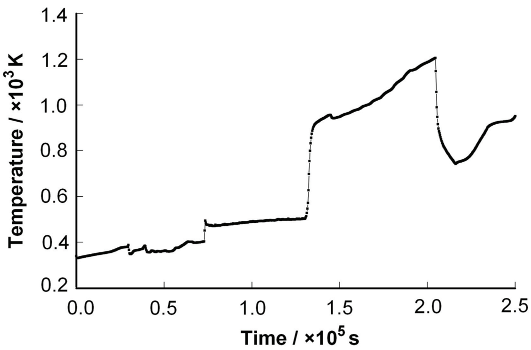

Fig.13 Temperature of calandria vessel wall.

5 Conclusions

The severe accident analysis for CANDU 6 station is simulated using the multiple tube rupture accident scenario of the steam generators and the large loss of coolant accident scenario. For the steam generator tube rupture scenario, the core disassembly begins at 10202 s and the calandria vessel fails at about 154039 s. For the large loss of coolant accident scenario, the core disassembly begins at 32183 s and the calandria vessel fails at about 204649 s. The results show that the progression of a severe core damage accident induced by the steam generator tube rupture or the large loss of coolant accident is relatively slow. The water surrounding the fuel in the calandria vessel and calandria vault plays an important role in a heat sink. The operators would have enough time to initiate mitigation measures and arrest the progression before the calandria vessel fails.

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