MAAP4-CANDU Analysis of a Generic CANDU 6 Station: Preliminary Results for a Large LOCA Scenario

by

S.M. Petoukhov and P.M. Mathew
Atomic Energy of Canada Limited
Containment Analysis Branch
Chalk River Laboratories
Chalk River, ON, Canada K0J 1J0

ABSTRACT

To support the generic probabilistic safety analysis (PSA) program at AECL, in particular to perform Level 2 PSA of a CANDU® 6 plant undergoing a postulated severe accident, the capability to conduct severe accident consequence analysis for a CANDU plant is required. For this purpose, AECL selected MAAP4-CANDU from a number of other severe accident codes. The necessary models for a generic CANDU 6 station have been implemented in the code, and the code version 4.0.4A was tested using station data, which were assembled for a generic CANDU 6 station. This paper describes the preliminary results of the consequence analysis using MAAP4-CANDU for a generic CANDU 6 station, when it undergoes a postulated, low probability large loss-of-coolant accident with assumed unavailability of several critical safety systems leading to a severe core damage accident. The analysis results show that the plant response is consistent with the physical phenomena modeled and the failure criteria employed. The results also confirm that the CANDU design is robust with respect to severe accidents, which is reflected in the calculated long times that are available for administering accident management measures to arrest the accident progression before the calandria vessel or containment become at risk.

INTRODUCTION

This paper describes test results of the consequence analysis for a generic CANDU 6 station when it undergoes a postulated, low probability large break loss-of-coolant accident (LOCA) with assumed unavailability of several critical safety systems, which can lead to severe core damage. The Modular Accident Analysis Program MAAP4-CANDU was selected as the severe accident consequence analysis code for Level 2 PSA studies of CANDU products [1]. The consequence analysis was performed using MAAP4-CANDU (M4C) version 4.0.4A. The selected accident sequence used in this study, however, does not necessarily represent the actual sequence identified for Level 2 PSA; the purpose of this study was not to produce final results for Level 2 PSA, but to demonstrate the code capability for Level 2 PSA applications.

MAIN FEATURES OF THE MAAP4-CANDU CODE

The M4C code can simulate the progression of severe accidents for AECL CANDU and Ontario Power Generation Inc. (OPG) stations, including many of the actions undertaken as part of

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accident management [2]. The code was developed on the basis of the MAAP4 code, which is used for pressurized and boiling water reactors.

M4C calculates the progression of severe accidents, starting from normal operating full power conditions for a set of plant system faults and initiating events that lead to primary heat transport system (PHTS) inventory blowdown or boil off, core heatup and melting, fuel channel failure, calandria vessel (CV) failure, reactor vault (RV) failure and containment failure. Furthermore, some models are included in the code to analyze accident mitigation measures, such as debris cooling in the CV or containment. For this purpose, M4C contains the principal models of the CANDU systems that are required for severe accident analysis. The most important distinguishing feature of M4C compared with MAAP4 is the model of the CANDU horizontal reactor core with fuel bundles situated inside pressure and calandria tubes using the "Channel Module". M4C treats the spectrum of physical processes that might occur during an accident, such as steam formation in the PHTS, core heatup, hydrogen generation, core debris-concrete interaction, ignition of combustible gases, energetic steam interactions, fluid entrainment by high velocity gas, and fission product release, transport and deposition. In the current code version, the channel module calculates the thermal hydraulic processes in the fuel channels after the channel is dry inside. Before the fuel channel is dry inside, the decay heat from the fuel is transferred to the PHTS coolant.

NODALIZATION OF CANDU 6 STATION

M4C simulates only the most significant systems, components and processes that are deemed necessary to demonstrate the overall response of the plant to a severe accident. For reference, some details of the nodalization scheme used in the present work to simulate some of the systems and components of CANDU 6 are given below.

M4C has a generalized containment model, which was used to model the CANDU 6 containment. The containment was represented by 13 nodes, 31 flow junctions and 90 wall heat sinks representing horizontal and vertical containment walls.

The PHTS is represented by two symmetric loops, with the flow through each loop following a "figure of eight" configuration, and with some channels carrying the flow inward and others outward from each reactor face. Fourteen nodes in each PHTS loop represent the following components: pump discharge lines, reactor inlet headers, reactor outlet headers (ROH), inlet piping of steam generators, hot leg tubes of steam generators, cold leg tubes of steam generators, and pump suction lines after cold leg tubes of steam generators. M4C has a simple PHTS thermal hydraulics model; the coolant pressure is the same in all nodes within the same PHTS loop.

The CANDU 6 core has 380 fuel channels arranged in 22 rows and 22 columns. A simplified core nodalization was used to represent the total number of fuel channels. The 22 rows of the core were divided into 6 vertical nodes, with 4, 4, 3, 3, 4, 4 rows in each node. Within each vertical node, the fuel channels were divided into 3 groups (low, medium and high power) of characteristic channels according to their fuel powers. The 22 columns of fuel channels were divided into 2 loops, symmetric about the vertical axis. The 12 bundles in a CANDU channel were modeled as 12 axial nodes. In a fuel channel, the calandria tube (CT) and the pressure tube

(PT) are modeled as two concentric rings. The 37 fuel elements of the fuel bundle are modeled as 7 concentric rings. Thus, 9 rings represent a CANDU 6 fuel channel. Figure 1 is a cross-section of the CANDU 6 fuel channel and the nodalization used in M4C. The secondary side of the steam generator is represented as one node and the primary side of the steam generator contains two nodes.

The pressure and inventory control system is represented by a pressurizer joined with two PHTS loops. Each line connecting the pressurizer with the PHTS contains a motor-operated pressurizer loop isolation valve, which can be closed in case of a LOCA. All basic thermal hydraulic processes, such as boiling and condensation, flashing and rain out, and the behavior of fission products are modeled in pressurizer. Pressurizer heaters are also modeled. The degasser condenser is not currently modeled.

MAJOR MODELING ASSUMPTIONS AND FAILURE CRITERIA

Several assumptions were made in the current analysis. These assumptions are either embedded in the code as models with input control parameters, or they are assumed in the present analysis.

Analysis Assumptions

- Reactor shutdown is initiated immediately after accident initiation.
- Moderator cooling and shield cooling are unavailable.
- Shutdown cooling system is unavailable.
- Main and auxiliary feed water are unavailable.
- No containment leakage or ventilation is modeled.
- Governor and main steam isolation valves are closed after accident initiation.
- Liquid relief valves (LRVs) and pressurizer relief valves discharge the PHTS inventory into containment. In reality, these valves should discharge into the degasser condenser; but the degasser condenser is not currently modeled.
- Steam generator main steam safety valves (MSSV) are available; they open and close at the set point to relieve pressure.
- Crash cool down system is available.
- Emergency Core Cooling System (ECCS): high pressure injection (HPI) and medium pressure injection (MPI) are available.
- ECCS low pressure injection (LPI) is unavailable.
- Containment dousing spray system is available.
- All operator interventions are not credited.

Failure criteria

In addition to the models employed, the results of the analysis will depend on the failure criteria used to fail the components (such as containment, CV, fuel channel, RV, etc.). Some of the failure criteria are user-input, whereas others are calculated and applied by the code. Details regarding some of the failure criteria available in M4C are given in Reference [2]. For reference, a brief description of the failure criteria used in the present analysis is given below.

Containment Failure Criteria

The present understanding of a CANDU 6 containment failure under pressure loads is largely based on results of assessments [3] of experiments, which were conducted to study the effect of overpressure on a model 1/14 scale containment at room temperature. Analysis of the results show for a CANDU 6 containment [3] that the first through-wall cracks appeared at a containment pressure of 430 kPa (a). At that pressure the reinforcing horizontal tendons were still intact. The yielding of the horizontal tendons began at 555 kPa (a) and at 630 kPa (a) the reactor building failed as a result of the rupture of the horizontal tendons.

In the current analysis we used a simple methodology to determine whether the containment failed. If the pressure in a given containment node is higher than a user-input value, then an artificial junction (flow path) is formed through the containment concrete wall. The diameter of this flow passage is also a user-input value. In the present work, we used 500 kPa (a) for the containment failure pressure and 0.4 m for the containment leak diameter. As mentioned earlier, no containment leakage is modeled prior to failure. If containment leakage is modeled, some gas can be released through the through-wall cracks at a lower pressure, which would result in a lower containment end-pressure. Thus, the containment failure time could be longer.

Calandria Vessel Failure Criteria

Several criteria are used to determine the point at which the CV fails:

- failure by creep based on Larson-Miller parameter [2]. The creep rupture is caused by high temperature and pressure combined with the stresses from imposed loads. No user-input is required for this criterion.
- failure by high pressure in the CV. This catastrophic rupture pressure is a user-input value; a value of 2.25 MPa (a) is used in the present work.
- failure due to a coherent jet of debris impinging directly onto the CV wall, causing localized ablation of the CV wall.
- failure due to molten metal layer attack on the CV wall.

Fuel Channel Failure Criteria

Fuel channel failure is defined as a perforation in its pressure boundaries followed by mass transfer between the environment inside the PT and the CV, which means that both the PT and CT are perforated. The following mechanisms for fuel channel failure are considered in the M4C code, depending on the level of pressure in the PHTS:

High PHTS pressure

High temperature fuel channel experiments conducted at AECL show that non-uniform circumferential temperature distributions could lead to pressure tube rupture at high pressures [4]. The M4C code does not calculate circumferential temperature distribution in the channel; therefore, a simplified channel failure criterion is used in the current analysis. The channel is assumed to have failed under high pressures and temperatures when the ballooning criterion is satisfied, based on the experimental results for an isothermal PT [5].

Low PHTS pressure

At low PHTS pressures, the fuel channels may fail due to local melt-through or sagging of the pressure and calandria tubes.

Fuel Channel Disassembly Criteria

Disassembly is a process during which fuel and channel structural materials are separated from the original channel and relocate into "holding bins", i.e., artificial bins that are constructed to keep track of disassembled core parts until they are released into the CV. An axial segment of the fuel channel is deemed to be disassembled if and when the average temperature of the PT and CT walls reaches a value calculated by the code, which is the melting temperature of oxygenated Zr.

Core Collapse Criteria

When a large amount of debris becomes lodged on top of the supporting channels and thus exceeds the load bearing capacity, the supporting channels collapse. Under those conditions the supporting channels are expected to pull out from the rolled joints. When the suspended debris bed mass located in the holding bins inside the CV exceeds a user-specified value, the core material in the suspended bed, plus some of the intact channels covered by the moderator relocate into the bottom of the CV. The suspended debris bed mass per PHTS loop, which will trigger channel pullout from the rolled joints, was estimated from the accumulated channels mass required to exceed the peak stress at the CT rolled joint. A value of 25,000 kg/loop was used in the current analysis.

Reactor Vault Failure Criteria

When the core debris/corium are relocated into the RV after CV failure, and in the absence of cooling, the reactor vault floor erodes as a result of core-concrete interaction. When the eroded concrete thickness reaches a certain "critical" depth, the floor can no longer support the weight of concrete and corium, and the RV is considered failed. This eroded concrete depth is a user-input value, and is set to 2 m in the present work.

Fission Product Release Criteria

M4C models the fuel pin as a mixture of UO_2 and the fuel sheath material. The following criterion is used in M4C for the release of fission products: if the combined fuel sheath/UO_2 temperature of the given core node reaches a user-defined value, then noble gases in the gap are released from the fuel bundle. In this analysis, we used 1000 K based on Phebus data [6]. Fission products from the fuel matrix are released depending on the fuel temperature, based on fractional release models described in [7] and [8]. When the fuel bundles leave the original core boundary, fission products are released from the suspended debris bed into the CV and into the containment during corium-concrete interaction. It is also assumed that no fission products are released from the terminal debris bed that accumulates on the CV bottom, because it is assumed that the top crust will prevent the release of fission products.

INPUT PARAMETERS FOR A GENERIC CANDU 6 STATION

The M4C parameter file for a generic CANDU 6 station consists of about 5,000 parameters, which describe the important systems and components considered in the analysis. Table 1 presents only some of the key input parameters for the station. The generic CANDU 6 station data used here are not station-specific; they were assembled from documents from various CANDU 6 stations.

The M4C version 4.0.4A was run on a PC platform (Pentium-4, 1700 MHz processor) to generate the results reported below. The code runs smoothly and the computation speed is about one hundred times faster than real time.

RESULTS AND DISCUSSION

In the present analysis, we consider a large LOCA scenario initiated by a guillotine rupture of the ROH in loop 1, followed by a double-sided blowdown of the PHTS coolant. The break area of the ROH considered is 0.2594 m^2. Table 2 lists the sequence of significant events observed during this simulation. The event times are reproduced in Table 2 as calculated by the code, rounded up to the nearest second. The times in the body of the text are rounded up to the nearest 100 s. For easy reference Table 2 also shows the event times in hours.

Primary heat transport system and ECCS response

Figure 2 shows the pressure in the two PHTS loops and in the pressurizer. As a result of the break in the ROH in loop 1, the PHTS pressure in loop 1 decreases faster than in loop 2. When the pressure in loop 1 reaches 5.5 MPa (a), the loop isolation valves are closed at about 24 s to isolate the pressurizer from the two loops, and loop 2 from loop 1. HPI into loop 1 starts when its pressure reaches 4.14 MPa (a), and terminates at about 98 s. Since the loop 2 pressure is greater than loop 1, water from HPI ECCS goes mainly into loop 1. After ECCS injection terminates, the pressure in the intact loop (loop 2) increases as a result of core decay heat, and because of the unavailability of the auxiliary feed water and shutdown cooling systems. The pressure in loop 2 reaches a constant value of about 10 MPa (a) and oscillates as a result of the periodical opening and closing of the LRVs. The pressure in loop 2 then drops rapidly at about 35,100 s, due to fuel channel failure in loop 2.

When the pressure difference between the MPI water source (dousing tank) and the PHTS reaches the set point of 114 kPa, MPI starts at about 98 s and water from the dousing tank is pumped into the PHTS. MPI continues until 1100 s, when water in the dousing tank is no longer available.

Because M4C uses a simple PHTS thermal hydraulics model, as mentioned earlier, very good agreement between results obtained from a detailed thermal hydraulics code and M4C cannot be expected during the short-term accident progression events. The scope of the M4C analysis is to provide results on the long-term behavior of a CANDU plant to severe accidents.

Steam generator response

The steam generator MSSVs are opened after receiving the LOCA signal to initiate crash cooldown at about 34 s, which decreases the pressure in the primary side of the steam generators. As a result of the blowdown through the open MSSVs and the boil-off of water from the secondary side of the steam generators, the water level in all four-steam generators decreases. The steam generators dry out by about 2,600 s in loop 2, and by about 14,400 s in loop 1 (see Table 2). The water level in the steam generators in the broken loop (loop 1) is higher than in the unbroken loop (loop 2). Because very little HPI water is injected into loop 2, the coolant in loop 2 is hotter than in loop 1, resulting in faster boil-off of water in the secondary side of the steam generators in loop 2. As a result, the water level in the secondary side of the steam generators in loop 2 decreases faster than in loop 1.

Fuel channel response

Table 2 shows that the fuel bundles are uncovered inside the fuel channels at about 2,900 s in loop 1 and at about 22,100 s in loop 2. The uncovering of the fuel bundles is the result of a combination of the following phenomena: (1) coolant boil-off due to decay heat from the core, (2) loss of coolant through the break, (3) loss of coolant through PHTS LRVs, and (4) loss of heat sink in the steam generators due to the loss of the secondary side steam generator inventory.

As steam generators dry out by about 2,600 s in loop 2 (see Table 2), the PHTS pressure increases to the LRV set points and the PHTS coolant inventory is discharged into the containment. When the temperature of the PT reaches about 900 K at the high PHTS system pressure in loop 2, one fuel channel in loop 2 ruptures at about 35,100 s.

The temperature of the PT and CT, as well as that of the central fuel ring for a selected fuel channel are presented in Figure 3 for loop 1 and in Figure 4 for loop 2. The PT, CT and fuel temperatures remain constant up to about 20,000 s for the channel in loop 1 (Figure 3) and up to about 27,000 s for loop 2 (Figure 4), because the current code version `assumes that the core decay heat goes directly into the PHTS coolant during that period. After the given fuel channel is dry, the channel module of the code is initialized and the fuel channel conditions and PT, CT and fuel temperatures are analyzed at every time step.

When the disassembly criteria are satisfied, channel sections relocate into the "holding bins" and stay there temporarily as a suspended debris bed. When the suspended debris bed mass exceeds the user-input value of 25,000 kg/per loop, the core material in the suspended debris bed, and most of the intact channels relocate into the CV bottom by core collapse at about 50,800 s.

Calandria vessel response

Following the initiating event, the moderator temperature and pressure in the CV increase as a result of the loss of moderator cooling and heat transfer from the core. The moderator in the CV reaches the saturation temperature at about 19,800 s. At about 26,200 s, the pressure inside the CV reaches the set point of the rupture disks, and the rupture disks fail, resulting in moderator expulsion through the relief ducts. The moderator continues to discharge into the containment, resulting in a further gradual decrease of the CV water level.

Following the core collapse at about 50,800 s, the water inside the CV is depleted. Water in the RV acts as a heat sink and cools the CV. Eventually, water in the RV reaches the saturation temperature and boils off. Crusts are formed on the CV walls very soon after core collapse; the crust thickness on the CV walls is in the range of 5 to 10 cm. After water in the CV is depleted, the core debris in the CV begins to heat up.

When the water level in the RV falls to the CV bottom level, which occurs at about 199,000 s, the CV bottom heats up rapidly and fails due to creep, at about 199,300 s. When the CV fails, the debris relocate into the reactor vault.

Reactor vault and end-shield response

The pressure and water level in the RV and end-shields increase gradually after the initiating event, due to the unavailability of the shield and moderator cooling systems and the resulting thermal expansion of water in the RV. The RV and end-shields are connected to combined vent lines to relieve over-pressure through rupture disks. At about 19,100 s, these rupture disks burst. Steam is discharged from the end shields to the containment, resulting in a decrease in the end-shield water level. The water in the RV begins to boil off at about 81,500 s, which results in a gradual water level decrease.

At about 199,300 s, the CV fails and the corium in the CV relocates to the RV floor. Energetic corium/steam interaction was predicted by the code at about 199,300 s in the RV, following the corium relocation. Eventually, all water in the RV dries out and corium reacts with the concrete floor. When the eroded depth of the concrete reaches 2 m, the RV fails at about 439,000 s.

Containment response

Figure 5 shows the pressure in the lower half of the steam generator enclosure. After accident initiation, the containment pressure increases, because the PHTS coolant is discharged into the containment through the outlet header break and the PHTS LRVs. When the containment pressure reaches 114 kPa (a), the dousing sprays are turned on at about 6 s; the containment pressure is thus reduced. The sprays are turned off when the containment pressure decreases to 107 kPa (a). The rapid increase (or decrease) of containment pressure, as shown in Figure 5 at the approximate times of 26,000 s, 50,800 s, 199,300 s and 440,000 s, are due to the following events, respectively: (1) the opening of CV rupture disk, (2) core collapse, (3) corium relocation from the CV, corium/steam energetic interaction and the subsequent containment failure, and (4) corium relocation into the basement after RV failure and subsequent steaming. The containment failure pressure of 500 kPa (a) is reached at about 199,400 s.

Fission product and hydrogen release

The original inventory of the noble gases in the core is 57.7 kg, based on calculations using the coupled multi-region WIMS-AECL/ORIGEN-S code. Major portion of the noble gases is released into the CV from the fuel and the suspended debris bed during core disassembly and core collapse from about 36,000 s to about 51,000 s. Eventually, all noble gases are released into the environment, when the containment fails at about 199,400 s. No containment leakage or ventilation is modeled in the present analysis.

Figure 6 shows the mass of CsI released in-vessel, ex-vessel (outside the CV), in the PHTS, in the CV, in containment and into the environment. The initial inventory of CsI is 27.96 kg. At about 22,000 s, fuel temperature for loop 1 is higher than 1000 K, when the fission product release from the fuel matrix begins. At about 35,100 s, the PT and CT rupture in loop 2, and the fuel element temperatures are greater than 1000 K; therefore, fission products are released, as shown in Figure 6. Because the CV rupture disks are already opened at about 26,000 s, the fission products are released through the CV rupture disks into the containment. The mass of CsI in the containment (including airborne and deposited) remains at about 1.4 kg until about 220,000 s. Because almost all of the CsI is retained in the containment by various fission product retention mechanisms, only a very small amount of CsI and CsOH totaling about 0.0068% is released into the environment, when the containment fails. When water in the RV is depleted at about 209,000 s, the corium reacts with the concrete, and fission products are released ex-vessel. At about 475,000s, 0.196 kg of CsI and 0.904 kg of CsOH are released to the failed containment and subsequently to the environment. The total amount of Cs and I released to the environment in the form of CsI and CsOH is about 0.996 kg or 3.6% of the initial Cs and I inventory.

Hydrogen is generated during the accident as a result of the following reactions: (1) Zr-steam reaction in fuel channels and during core debris oxidation in the suspended debris beds, (2) jet breakup of molten debris in the water pool of the RV, and (3) molten core-concrete interaction. Analyses show that the mass of hydrogen generated in the PHTS and CV prior to CV failure is 261 kg, and the mass generated in the RV is about 2,202 kg as a result of jet breakup and molten corium-concrete interaction.

CONCLUSIONS

The following general conclusions are drawn from the study reported here:

- MAAP4-CANDU Version 4.0.4A contains the most significant CANDU 6 systems required for severe accident analysis.
- The MAAP4-CANDU code runs smoothly on a PC platform for CANDU 6 analysis, and the computation speed is about one hundred times faster than real time.
- The results obtained in this study demonstrate the capability of the code for Level 2 PSA application.
- The obtained results and observed trends in this study are consistent with prior engineering judgment for the Large LOCA scenario analyzed.
- The total mass of Cs and I released to the environment in the form of CsI and CsOH is less than 4% of the initial Cs and I inventory.
- The analysis results confirm that the CANDU design is robust with respect to severe accidents, which is reflected in the calculated long times that are available for administering accident management measures to arrest the accident progression before the calandria vessel or containment become at risk.

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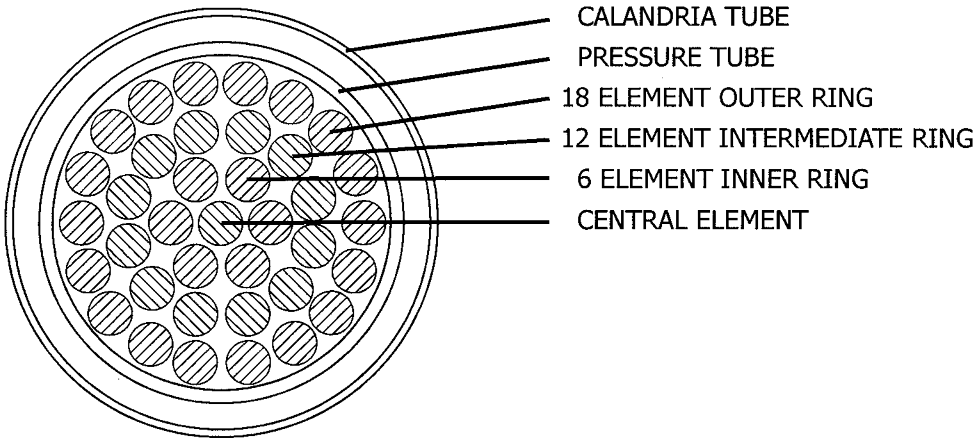
TABLE 1. Key Input Parameters for CANDU 6 Station

Thermal reactor power (MW)	2,064
ROH pressure (MPa (a))	10.0
SG secondary side pressure (MPa (a))	4.7
ROH coolant temperature (K)	583
Uranium dioxide mass in the core (kg)	98,815
Zr mass in the core (including pressure and calandria tubes) (kg)	38,647
D2O inventory in PHTS (without pressurizer) (kg)	130,000
D2O inventory in calandria vessel (kg)	227,000
H2O inventory in RV (kg)	465,324
H2O inventory in each SG secondary side (kg)	38,000
RV Rupture disk flow area (m^2)	0.308
Free volume inside containment (m^3)	48,000

TABLE 2. Sequence of Significant Events for Large LOCA

Time (hr)	Time (s)	Event
0	0.	ROH guillotine rupture on loop 1
0.0015	6	ECCS HPI is on
0.0016	6	Dousing system is on
0.0068	24	Pressurizer and PHTS loops are isolated
0.0094	34	Steam generator MSSVs are open, crash cooldown system is on
0.027	98	ECCS HPI is terminated
0.027	98	ECCS MPI is on
0.08	298	Dousing tank water is depleted for containment sprays
0.31	1,100	ECCS MPI is off
0.60	2,614	Steam generator is dry, loop 2
0.80	2,860	Fuel bundles are uncovered inside fuel channels in loop 1
4.0	14,386	Steam generator is dry, loop 1
5.0	17,940	At least one channel is dry in loop 1
5.5	19,826	CV water pool is saturated
6.1	22,060	Fuel bundles are uncovered inside fuel channels in loop 2
7.0	25,180	At least one channel is dry in loop 2
7.3	26,236	CV rupture disk #1 is open
9.7	35,066	PT and CT rupture, loop 2
10.1	36,381	Beginning of the core disassembly
16.8	50,781	Core collapse onto the CV bottom
18.9	68,068	Water is depleted inside CV
55.4	199,308	CV bottom wall failed due to creep
55.4	199,328	Energetic core debris-steam interaction occurred in RV
55.4	199,351	Containment failed
55.4	199,480	Corium is discharged into RV
58.1	209,109	Water is depleted in RV
122.0	439,019	RV floor failed because of concrete erosion



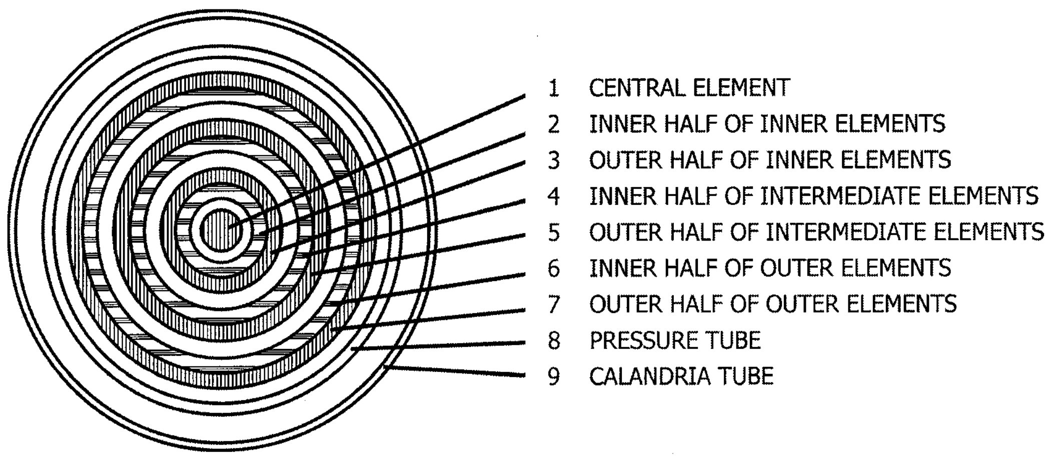


FIGURE 1. CANDU 6 Fuel Channel cross-section (top) and
Fuel Channel Nodalization Scheme used in M4C (bottom)

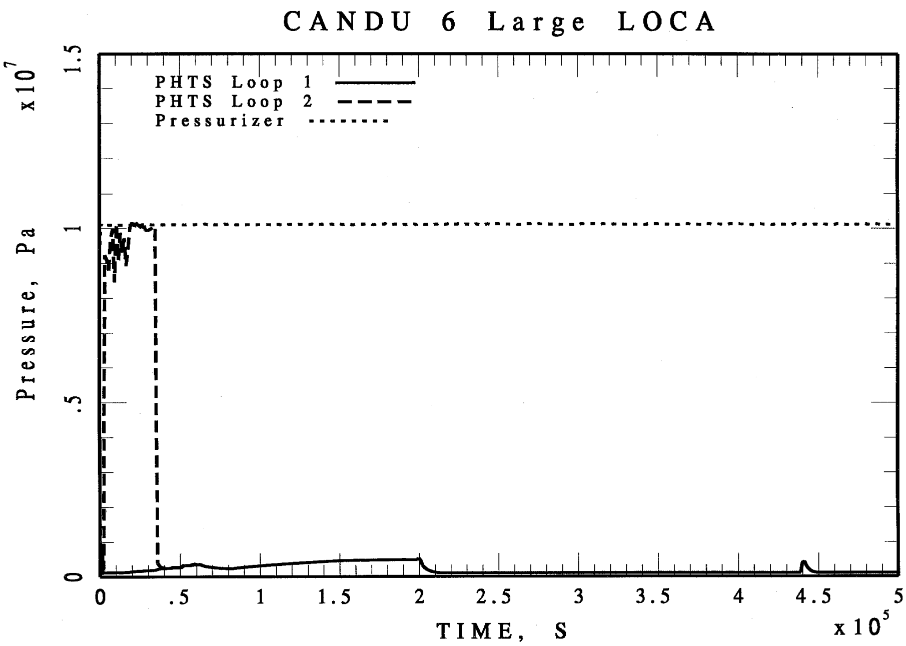


FIGURE 2. Pressure in PHTS Loops and Pressurizer (Large LOCA)

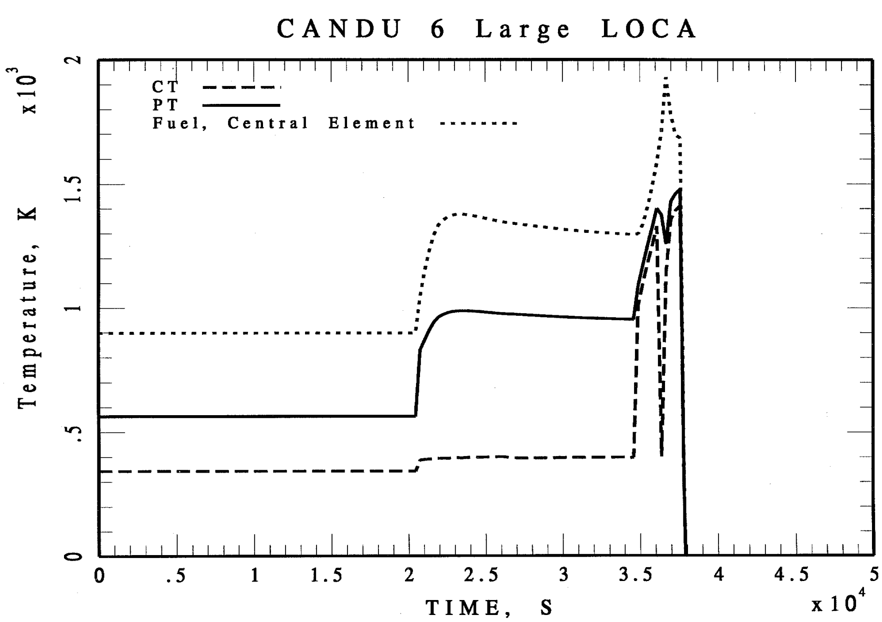


FIGURE 3. PT, CT and Fuel Temperature for Channel 2, Bundle 7, Loop 1 (Large LOCA)

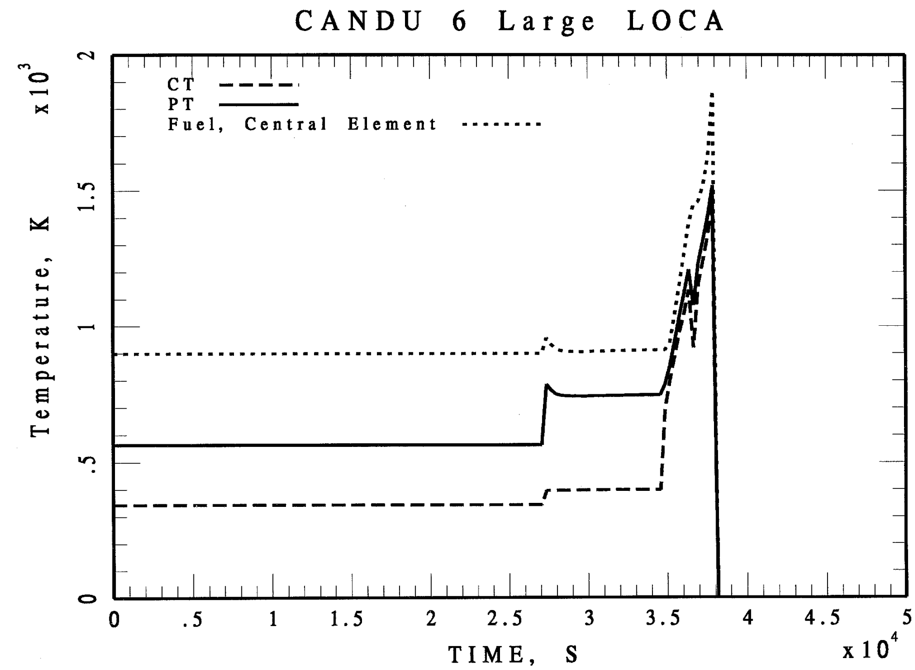


FIGURE 4. PT, CT and Fuel Temperature for Channel 2, Bundle 7, Loop 2 (Large LOCA)

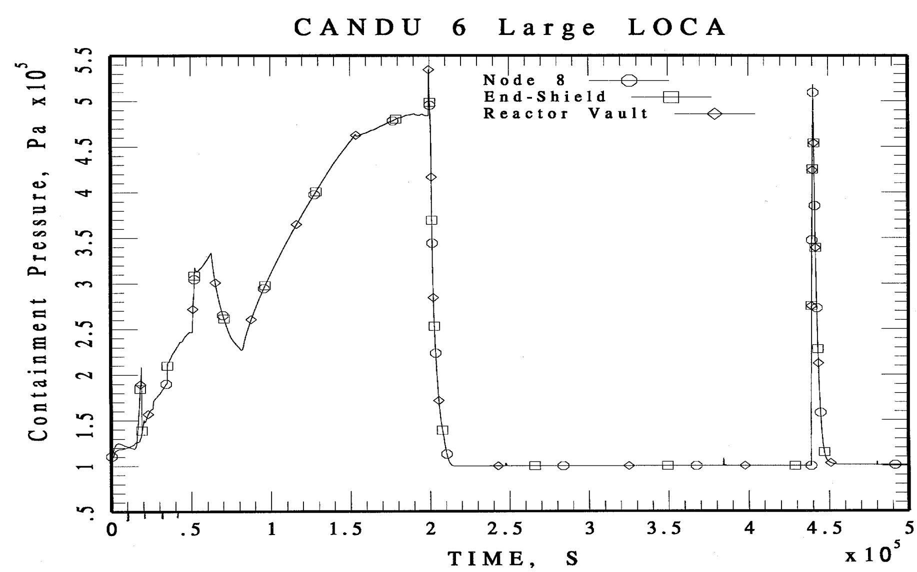


FIGURE 5. Pressure in Containment, Reactor Vault and End Shield (Large LOCA)

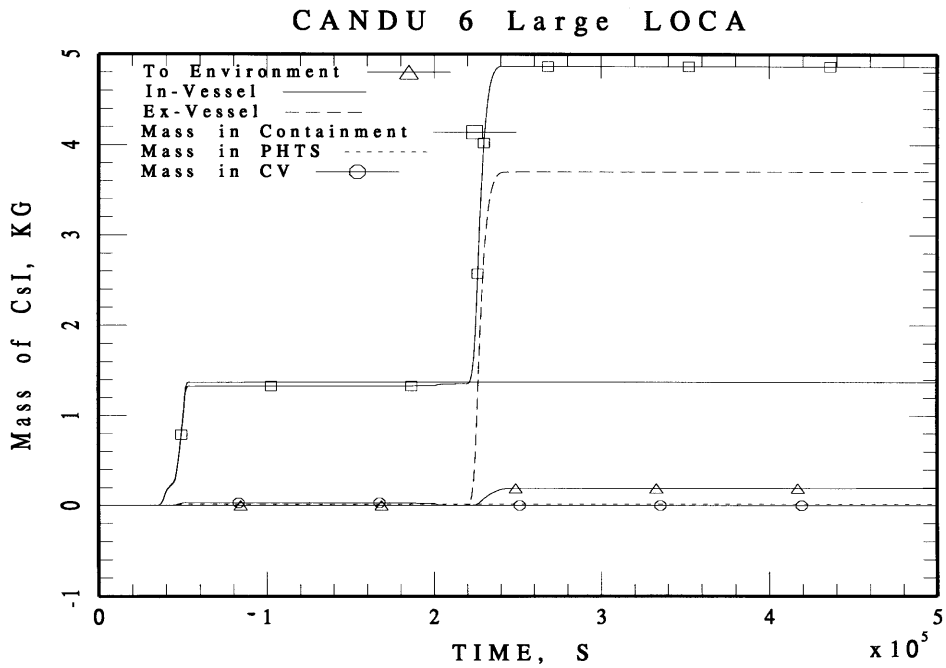


FIGURE 6. Mass of CsI Released (Large LOCA)