MAAP4-CANDU APPLICATION TO THE PSA LEVEL 2 FOR THE POINT LEPREAU NUCLEAR GENERATING STATION REFURBISHMENT PROJECT

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Abstract

A Level 2 Probabilistic Safety Assessment was performed for the Point Lepreau Generating Station. It focused on determining: i) the frequency and timing of containment failure, and ii) the quantities and species of radioactive fission products that might be released to the environment, during a severe core damage accident. The MAAP4-CANDU code was used to calculate the progression of postulated severe core damage accidents and fission product releases. Five representative severe core damage accidents were selected: Station Blackout, Small Loss-of-Coolant Accident, Stagnation Feeder Break, Steam Generator Tube Rupture, and Shutdown State Accident. Analysis results for these scenarios (including some sensitivity cases) are discussed in this overview paper.

1. Introduction

A Level 2 Probabilistic Safety Assessment (PSA) was performed by Atomic Energy of Canada Limited (AECL) for the Point Lepreau Generating Station (PLGS, a CANDU® 6 reactor) by the Point Lepreau Refurbishment Project. An overview of Point Lepreau operations and refurbishment activities was given in [1]. Results from the Level 1 PSA provided accident scenarios as input to the Level 2 PSA, which determined the frequency and timing of containment failure. A Level 2 PSA also includes a determination of the quantities and species of radioactive fission products that might be released to the environment during a severe core damage accident.

The determination of the accident progression, for application to a Level 2 PSA, uses computer codes to model severe accident phenomena for scenarios identified in the Level 1 PSA. A Level 2 PSA quantifies challenges to containment integrity, plus the location and species of fission product releases to the environment. For the PLGS Level 2 PSA, MAAP4-CANDU [2] v4.0.5A+ was used to estimate:

- The timing of the event progression and accompanying thermo-physical and thermo-chemical phenomena,
- The effect of safety and normal operational system availabilities,
- Source terms for combustible gases, the resulting hydrogen and carbon monoxide concentrations in containment, and whether burning occurs (also dependent upon oxygen and steam concentrations),
- Fission product transport and retention within containment,
- The magnitude and nature of fission product releases to the environment, from containment or from the secondary side of a steam generator,
- The timing and duration of challenges to containment integrity, and
- The effect of operator actions in mitigating severe accident consequences (reducing challenges to containment integrity and reducing fission product releases from the reactor building).

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Five representative severe core damage accidents were selected for the Level 2 PSA performed for the Point Lepreau Refurbishment Project:

1. Station blackout (SBO), with the loss of moderator cooling and shield cooling systems due to the loss of electrical power, and with a partial or full loss of emergency core cooling (LOECC);
2. Small loss-of-coolant accident (SLOCA, 2.5% reactor inlet header break), with a full or partial loss of emergency core cooling, loss of moderator cooling, and loss of other safety-related systems;
3. Stagnation feeder break (SFB) LOCA, with a full or partial loss of emergency core cooling, loss of moderator cooling, and with the moderator draining through a failed fuel channel bellows;
4. Steam generator (i.e., boiler) tube rupture (SGTR, a potential containment bypass) with a full or partial loss of emergency core cooling, and the loss of the moderator cooling system; and
5. Shutdown state accident (SSA). The initiating event was a leak from the bearing seal of the shutdown cooling system (SDCS) pumps, with a simultaneous loss of the shutdown cooling system. The primary heat transport system (PHTS) eventually drained almost down to the reactor header level, and this was combined with a partial or full LOECC and the loss of the moderator cooling system, shield cooling system and other safety-related systems.

In each of the above accident scenarios, the reference case assumed no operator interventions and credited only a limited number of safety-related systems (the others were assumed unavailable). For each accident, the reference case was followed by a series of sensitivity cases assuming certain system availabilities, to assess their effects on the accident sequence. A total of about 50 cases were analyzed. The timing of major events and fission product releases to the environment, for the five major scenarios, including some sensitivity cases, are discussed in this paper.

2. Brief description of the MAAP4-CANDU code

The MAAP4-CANDU [2] code can simulate the progression of severe accidents in CANDU stations, including many of the actions undertaken as part of accident management. The code was developed on the basis of the MAAP4 code, developed by Fauske and Associates Inc. MAAP4 is owned by the Electric Power Research Institute, and is used for severe accident analysis in pressurized and boiling water reactors (PWRs and BWRs). OPG is the MAAP-CANDU code licensee (code holder), and AECL holds a sub-license from OPG.

MAAP4-CANDU calculates the progression of severe accidents, typically starting from normal operating conditions, for a set of plant system failures and initiating events. The accident eventually progresses to: PHTS inventory blow-down or boil off; core heat-up and melting; fuel and fuel channel disassembly; calandria vessel failure; reactor vault failure; and containment failure. In addition, some models are included in the code to analyze accident mitigation measures, such as debris cooling in the calandria vessel (CV) or containment.

The most distinguishing feature of MAAP4-CANDU, compared with the MAAP4 PWR/BWR code, is the model of the CANDU reactor core: fuel bundles situated inside horizontal pressure and calandria tubes. MAAP4-CANDU models a large spectrum of physical processes that might occur during an accident, such as: steam formation in the PHTS; core heat-up, disassembly and collapse; hydrogen generation; core debris-concrete interaction; ignition of combustible gases; energetic molten fuel-coolant interactions; fluid entrainment by high velocity gas; and fission product release, transport and deposition.

3. Nodalization of the Point Lepreau Station

MAAP4-CANDU simulates the most significant systems and components that are deemed necessary to demonstrate the overall response of the plant to a severe accident. Some details of the nodalization scheme used in the present work, to simulate the significant systems and components of Point Lepreau Station, are described in this section. A more detailed description of the CANDU 6 station nodalization can be found in [2].

MAAP4-CANDU has a generalized containment model, which was used to model the Point Lepreau Station containment. The containment building was represented by 13 nodes, and connected by 31 flow junctions. Concrete containment walls, floors, ceilings and structural steel were represented in the MAAP4-CANDU containment model with 94 heat sinks.

The PHTS was represented by two symmetric loops, each following a "figure of eight" configuration. Fourteen nodes in each PHTS loop represented the two pump discharge lines; two reactor inlet headers; two reactor outlet headers (ROH); the inlet piping of two steam generators (SGs); the hot leg tubes of two steam generators; the cold leg tubes of two steam generators; and the pump suction lines connected to the cold legs of each steam generator. These nodes tracked water and gas volumes and temperatures, and the corresponding heat sink capability of the water, gas and component material.

The MAAP4-CANDU loop thermalhydraulics is simplified; only mass and energy conservation are modelled, but not momentum. Each PHTS loop is modelled with only one pressure. The primary coolant is initially a homogeneous water + steam mixture but, if the global PHTS loop void fraction exceeds a user-input value, the water and steam phases separate. A PHTS break or valve opening discharges coolant based on the local coolant type (i.e., water, steam, mixture of separated water and steam, or homogeneous mixture of water + steam), and the coolant conditions (i.e., temperature and pressure). The MAAP4-CANDU thermalhydraulics model, while simplified, can be tuned to approximate simulation results from using a dedicated thermalhydraulics code (e.g., CATHENA).

A pressurizer, joined to both PHTS loops via the reactor outlet headers, represented the pressure and inventory control system. Each line, connecting the pressurizer with a PHTS loop, contained a motor-operated loop isolation valve that could be closed in case of a LOCA. In the Point Lepreau Station the PHTS liquid relief valves (LRVs) discharge into a degasser condenser that, in turn, discharges to containment if the pressure exceeds the degasser condenser relief valve set point. The degasser condenser volume was not modelled in the current analysis, and the LRVs were assumed to discharge directly to containment. The MAAP4-CANDU LRV model is a passive spring-loaded safety valve, and it was sized so that it could accommodate the same flow, and operate at the same pressure, as the degasser condenser safety valves.

The Point Lepreau station core has 380 fuel channels arranged in 22 rows and 22 columns. The 22 rows of the core were divided into 6 vertical core nodes. The 22 columns of fuel channels were divided, according to the two PHTS loops, symmetrically about a vertical plane in the centre of the core. The 380 fuel channels were modelled with 36 characteristic fuel channels (2 loops × 6 vertical core nodes × 3 characteristic channels per vertical core node per loop).

The 12 fuel bundles in each fuel channel were modeled as 12 axial nodes. In a fuel channel, the calandria tube (CT) and the pressure tube (PT) were modeled as two concentric rings. The 37 fuel elements of the fuel bundle were modeled as 7 concentric rings. Thus 12 axial nodes, each with 9 concentric model rings, represented a fuel channel, including the fuel, pressure tube and calandria tube.

4. Failure criteria

In addition to the analysis assumptions and models employed, the results of the analysis are affected by the failure criteria for systems and components such as containment, CV, fuel channel, reactor vault (RV, also known as the calandria vault), etc. A brief description of the failure criteria used in the present analysis is given in the following sections.

4.1 Containment failure criterion

MAAP4-CANDU uses a simple failure junction model to link a containment node with the environment. If the pressure difference, between the containment node and the environment, exceeds the user-input value for that junction, a flow path is formed. The opening area is a user-input value.

In the Point Lepreau refurbishment project, two failure junctions were used to represent airlock seal blowouts at a differential pressure (containment to environment) of 234.4 kPa (g). The first junction opening area was 0.014624 m^2, and the second junction opening area was 0.00394 m^2. These two junctions represented the cross-sectional areas of the openings formed during the blowouts of the equipment airlock and personnel air lock seals, respectively. In reality, the airlocks connect the reactor building (containment) with the service building. In this paper, the term environment represents anything outside the containment boundary.

In MAAP4-CANDU it was conservatively assumed that the seals on both the inner and outer airlock doors failed together, with no time delay due to the airlock volume between the doors. Therefore, an airlock seal blowout, connecting the containment to the environment, was assumed to occur when the containment pressure reached 234.4 kPa (g).

4.2 Calandria Vessel failure criteria

Several criteria are used in MAAP4-CANDU to determine the conditions at which the CV fails: a) failure by creep based on Larson-Miller parameter; b) failure by high pressure in the CV; c) failure due to a coherent jet of debris impinging directly onto the CV wall, causing localized ablation of the CV wall; and d) failure due to molten metal layer attack on the CV wall. One more (user input) failure criterion, e) was based on the water level in the calandria vault surrounding the calandria vessel. Normally the calandria vault is water-filled, which would cool the calandria vessel and any core debris in the calandria vessel terminal debris bed. As the accident progressed, however, the water level in the calandria vault decreased by boil off. It was assumed that when the calandria vault water level decreased to the level of the top of the terminal debris bed, the calandria vessel failed because there was insufficient external cooling in that region of the cylindrical calandria shell.

MAAP4-CANDU checks the predicted calandria vessel conditions at every time step to identify if any of the above calandria vessel failure criteria is met. During each simulation performed for this study, the calandria vessel was predicted to fail due to the user-input criterion (point e above).

4.3 Fuel channel failure criteria

Fuel channel failure is defined as a perforation of its pressure boundaries followed by mass transfer between the inside of the PT and the CV (i.e., both the PT and CT fail). The following fuel channel failure mechanisms are considered in the MAAP4-CANDU code, depending on the PHTS pressure.

High temperature fuel channel experiments conducted at AECL showed that non-uniform circumferential temperature distributions could lead to pressure tube rupture at high pressures [3]. The MAAP4-CANDU code does not calculate circumferential temperature distribution in a channel; instead, a channel is assumed to fail at high PHTS pressures when a ballooning criterion is satisfied. The criterion is based on the experimental results for pressure tubes at high temperature isothermal conditions [4]. The PT material used in [4] was identical to that used for CANDU 6 pressure tubes.

At low PHTS pressures, the fuel channels may fail due to local melt-through. The calandria tubes may also become perforated due to channel sagging, which allows steam to penetrate into the fuel channel annulus and increases the channel heat up rate due to the exothermic Zr-steam reaction.

4.4 Fuel channel disassembly criteria

Disassembly is a process during which fuel and channel structural materials separate from the original channel and relocate into “holding bins” - artificial constructs used to track disassembled core material until it moves down to the floor of the CV. These artificial bins allow some fuel channel debris (suspended debris) to be supported by colder (hence stronger) underlying fuel channels immersed in the moderator. Suspended debris will melt and relocate to the bottom of the CV, unless the core collapse criterion (Section 4.5) is satisfied and the suspended debris drops rapidly to the floor of the CV. An axial fuel channel segment is deemed to be disassembled if and when the average temperature of the PT and CT walls reaches the melting temperature of oxygenated Zr.

4.5 Core collapse criterion

If a sufficiently large amount of core debris becomes lodged on top of the supporting channels, the underlying supporting calandria tubes would pull out from the rolled joints in the calandria vessel tube-sheets. The debris and supporting channels would then collapse to the bottom of the CV. In MAAP4-CANDU, when the total suspended debris bed mass inside the CV exceeds a user-specified value, the core material in the suspended debris bed relocates to the bottom of the CV. The relocation (collapse) also causes many of the underlying intact channels, submerged in the moderator, to relocate to the bottom of the CV. The suspended debris bed mass per PHTS loop, necessary to trigger channel pullout from the rolled joints, was estimated from the accumulated mass of the channels required to exceed the peak stress at the CT rolled joint. A value of 25,000 kg of suspended debris, per PHTS loop, was used in the current analysis.

4.6 Fission product release criteria

MAAP4-CANDU models the fuel elements as a mixture of UO_2 and the fuel sheath material. The fuel sheath is modelled to fail if the combined fuel sheath/UO_2 temperature of a radial channel node (i.e., fuel ring) reaches a user-defined value. In this analysis, 1,000 K was used, based on PHEBUS test data [5].

Fission products are modelled to be released from the fuel only after the fuel sheath has failed, and the fractional release models [6] are based solely on the fuel temperature. There is no model of a fuel-to-sheath gap inventory of fission products. MAAP4-CANDU v4.0.5A+ assumes that no fission products are released from the terminal debris bed that accumulates on the bottom of the CV, because of the barrier provided by the top debris crust.

5. Key input parameters for Point Lepreau Station

The key Point Lepreau Station input parameters used in this study are presented in Table 1. Note that, in these analyses, only the inventories of radioactive fission products were input to MAAP4-CANDU.

6. Description of the cases and analysis assumptions

6.1 Common (for all cases) analysis assumptions

The following analysis assumptions are common for all the cases assessed in this study:

- Class III and IV power: credited (except SBO);
- Initial reactor power: full (except SSA);
- Reactor shutdown system: credited;
- Moderator and shutdown cooling: unavailable;
- Main feed water: unavailable;
- Main Turbine Stop Valves: closed after accident initiation;
- Emergency water supply system: not credited;
- Containment isolation: credited, but containment leakage was modeled by an area that allowed 2.5% of the containment volume to leak per day at a pressure difference of 124 kPa;
- Containment ventilation system: not modeled;
- Containment dousing spray system: available;
- PARs (Passive Autocatalytic hydrogen Recombiners): credited;
- Operator interventions: none were credited.

6.2 Specific analysis assumptions

6.2.1 Station Blackout specific analysis assumptions

The SBO accident was initiated by a loss of Class IV power. For the SBO reference case (Case A) the following assumptions were also used:

- Class III power: not credited;
- PHTS loop isolation: not credited (no power);
- Shield cooling: unavailable;
- ECC: high pressure (HP), medium pressure (MP), and low pressure (LP) stages unavailable;
- Auxiliary feed water (AFW) to steam generators: unavailable;

- SG crash cool-down system: not credited.

More than ten additional SBO cases were analyzed, mostly with variations in the available safety systems/features: HP, MP and LP ECC, ECC heat exchanger, emergency power system (EPS), PARs, SG crash cool-down, and moderator drain.

SBO Case D1 was identified as the most representative case (in terms of accident frequency). For this case, the same assumptions were used as in SBO Case A, with the following exceptions:

- ECC: HP, MP and LP credited (EPS credited for 72 h);
- ECC heat exchanger: not credited;
- SG crash cool-down system: credited;
- Moderator: drains via the bellows of a ruptured fuel channel, with a flow rate of 4.2 kg/s.

6.2.2 <u>Small LOCA specific analysis assumptions</u>

The small LOCA was initiated by a 2.5% break in the reactor inlet header (RIH), with a total break area of 0.00537 m^2. For the SLOCA reference case (Case A) the following assumptions were used:

- ECC: HP, MP, LP unavailable;
- SG AFW: unavailable;
- SG crash cool-down system: not credited;
- Shield cooling: unavailable;
- Loop isolation: credited.

More than ten additional SLOCA cases were analyzed, mostly with variations in the availability of safety systems: HP and MP ECC, LACs, steam generator auxiliary feed water (AFW), and SG crash cool-down. SLOCA Case E was identified as the most representative case (in terms of accident frequency). For this case, the same assumptions as in SLOCA Case A were used, with the following exceptions:

- SG AFW: credited;
- HP and MP ECC: credited;
- SG crash cool-down: credited.

6.2.3 <u>Stagnation Feeder Break specific analysis assumptions</u>

The initiating event of the stagnation feeder break scenario is an inlet feeder pipe break in PHTS Loop 1. The size of the break is such that the coolant flow in the fuel channel slows down and rapidly stagnates. The opening in the inlet feeder pipe, necessary to cause the channel flow to stagnate, is fairly small (normally from 2 to 17 cm^2). Because the reactor is still at full power, and the stagnant coolant has a limited cooling ability, the channel voids and the fuel elements and the pressure tube heat up. The hot pressure tube balloons and is predicted (by the CATHENA thermalhydraulics code) to rupture within 10 – 30 s after the initiating event, depending on the location of the break in the feeder pipe and which channel is affected. Consequently, the calandria tube sees a rapid pressure pulse and experiences some localized hot spots from the superheated steam discharged out of the failed pressure tube.

When the pressure tube ruptures, the fuel channel annulus is pressurized and the bellows at both ends of the fuel channel are assumed to rupture. The calandria tube is expected to rupture very soon after the pressure tube and bellows rupture. This allows PHTS coolant and moderator to leak through the annulus between the PT and CT, and out the opening in the bellows. Thus moderator can drain from the calandria vessel.

In the current study, the break in the feeder pipe and the flow stagnation were not modeled because of the simplified MAAP4-CANDU channel thermalhydraulics model. Instead, it was assumed, based on previous CATHENA analyses, that the PT/CT/bellows ruptured 10 s after the initiating event.

For the SFB reference case (Case A1) the following assumptions were used:

- Moderator: drains with a flow rate of 30 kg/s;
- Shield cooling: available;
- ECC: HP, MP, LP unavailable;
- SG crash cool-down system: credited;
- SG AFW: unavailable;
- Loop isolation: credited.

More than 10 additional SFB cases were analyzed, mostly with variations in the availability of safety systems: HP and MP ECC; LACs; AFW; SG crash cool-down; and shield cooling. SFB Case C was identified as the most representative case (in terms of accident frequency). For this case, the same assumptions were the same as in reference SFB Case A1, with the following exceptions:

- SG auxiliary feed-water: credited;
- Shield cooling: not available.

6.2.4 Steam Generator Tube Rupture specific analysis assumptions

The initiating event in the SGTR case was a single SG tube rupture (one-sided blow-down with a total break area of 0.0001375 m2) at t=0 s. The break was located in the “cold” SG leg, just above the top of the SG tube sheet.

For the SGTR reference case (Case A) the following assumptions were used:

- Shield cooling: unavailable;
- ECC: HP, MP, LP unavailable;
- Crash cool-down system: not credited;
- Loop isolation: credited.

More than 10 additional SGTR cases were analyzed, mostly with the variation of the safety system availabilities: HP, MP and LP ECC; ECC heat exchanger; LACs; AFW; SG crash cool-down; number of SG tubes ruptured, etc. SGTR Case B was identified as the most representative case (in terms of frequency). For this case, the same assumptions as in reference SGTR Case A were used, with the following exceptions:

- ECC: HP and MP unavailable;

- ECC: LP injection credited (but ECC heat exchanger not credited);
- SG auxiliary feed-water: credited.

6.2.5 Shutdown State Accident specific analysis assumptions

The initiating event of the shutdown state accident (SSA) was a leak from the seal of the bearing of one of the shutdown cooling system pumps, starting six hours after the reactor shutdown, which caused a coolant loss from both PHTS loops.

At the beginning of the accident sequence, the PHTS was fully cooled down, full of coolant and depressurized. The primary heat transport system was eventually drained almost down to the reactor header level because of the coolant leak. The coolant heated up in a partially-voided PHTS and eventually boiled because of: i) the decay heat produced in the core, ii) the loss of moderator, shield and shutdown cooling systems, and iii) the partial or full loss of ECC.

The moderator heated up and gradually boiled off. The fuel channels voided and the fuel heated up; fuel melted inside the fuel channels and the fuel channels began to disassemble. Severe core damage progressed in both loops, and eventually fuel channels collapsed to the bottom of the calandria vessel. When the calandria vault water boiled off and the level decreased to the elevation of the CV terminal debris bed, the calandria vessel cylindrical wall heated up and failed. Debris from the calandria vessel relocated to the calandria vault, where it was quenched in the remaining vault water. Once the water in the calandria vault evaporated, the corium heated up and molten corium-concrete interaction began.

For the SSA reference case (Case A) the following assumptions were used:

- Initial core power: shutdown conditions;
- PHTS loop isolation: not credited;
- Shield cooling: unavailable;
- ECC: HP, MP and LP unavailable;
- Main Steam Safety Valves: locked open.

More than five additional SSA cases were analyzed, mostly with variations in the availability of safety systems: MP and LP ECC; ECC heat exchanger; LACs; AFW; D_2O recovery system, etc. SSA Case A was identified as the most representative case (in terms of accident frequency).

7. Major analysis results

As mentioned above, a total of about 50 cases were analyzed for the PSA Level 2 activities for Point Lepreau Refurbishment Project, using MAAP4-CANDU v4.0.5A+. Analysis results (timing of major events and fission product releases to the environment) for the five major scenarios SBO, SLOCA, SFB, SGTR, and SSA were obtained, including some sensitivity cases. A summary of the sequence of significant events for the most representative cases (in terms of accident frequency) can be found in Table 2. The last row of Table 2 also lists the fraction of the initial core inventory of the Cs+Rb+I (active isotopes only) released to the environment.

From the most representative cases analyzed, the earliest core disassembly (1.4 h) was predicted in SFB Case C. This occurred because the moderator drained from the CV at a flow rate of 30 kg/s, so that fuel channels were uncovered from the outside; no ECC was credited to cool the fuel inside the fuel channels.

The early core disassembly in SFB Case C, as well as the unavailability of the shield cooling system, resulted in the earliest CV failure (~55 h) among the most representative cases assessed in this study.

As indicated in Table 2, the earliest containment failure was predicted in SBO Case D1 (~23 h). Containment failure in this case was caused by over-pressurization as a result of steaming in the core and calandria vessel during the PHTS coolant discharge from the break in the ruptured fuel channel, and during the continuous ECC injection.

The greatest release of the Cs+Rb+I fission products to the environment (~13% of the radioactive isotopes) was predicted in SGTR Case B. This was primarily due to the core disassembly starting relatively late (~52 h), when the containment had already failed (at ~37 h). The volatile fission products were thus not deposited on the containment walls to the same degree as in other cases, but instead were released to the environment through the failed containment airlock seal.

The analysis of all the cases performed for this project also showed that the most effective systems to delay core disassembly are the LP ECC and SG AFW. The most effective systems to delay containment failure are the LACs and LP ECC. The most effective system to delay calandria vessel failure is shield cooling.

8. Summary

- MAAP4-CANDU v4.0.5A+ was used for PSA Level 2 activities for PL Refurbishment Project;
- More than 50 severe core damage accident sequences were analyzed;
- The timing of significant events for the scenarios, as well as the estimated fission product and hydrogen releases, were obtained for the PLGS;
- From the most representative cases analyzed, the earliest core disassembly (1.4 h) was predicted in SFB Case C, because of moderator drain through the bellows of the ruptured channel;
- The earliest containment failure was predicted in SBO Case D1 (23 h), due to steaming from the calandria vessel as the core disassembly progressed;
- The earliest calandria vessel failure (55 h) was predicted in SFB Case C;
- The largest fission product release to the environment (~13% of the initial Cs+Rb+I inventory in the core) was predicted in SGTR Case B;
- The most effective systems to delay core disassembly are LP ECC and SG AFW;
- The most effective systems to delay containment failure are LACs and LP ECC;
- The most effective system to delay calandria vessel failure is shield cooling.

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10. Nomenclature

AECL	**A**tomic **E**nergy of **C**anada **L**imited
AFW	**a**uxiliary **f**eed**w**ater
BWR	**b**oiling **w**ater **r**eactor
CANDU	**Can**ada **d**euterium **u**ranium, pressurized heavy water reactor
CATHENA	**C**anadian **A**lgorithm for **The**rmal-hydraulic **N**etwork **A**nalysis (AECL-developed thermal-hydraulics code)
CT	**c**alandria **t**ube
CV	**c**alandria **v**essel
ECC	**e**mergency **c**ore **c**ooling (3 stages: high, medium and low pressure)
EPS	**e**mergency **p**ower **s**ystem
FAI	**F**auske and **A**ssociates **I**ncorporated, developer of the MAAP codes
HP	**h**igh **p**ressure stage of the ECC (accumulators)
LAC	**l**ocal **a**ir **c**ooler
LOCA	**l**oss **o**f **c**oolant **a**ccident
LOECC	**l**oss **o**f **e**mergency **c**ore **c**ooling
LRV	**l**iquid **r**elief **v**alve on PHTS
LP	**l**ow **p**ressure stage of the ECC (pumped, fed by water from the containment sump)
MAAP4-CANDU	**M**odular **A**ccident **A**nalysis **P**rogram for the **CANDU** reactor
MCCI	**m**olten **c**ore – **c**oncrete **i**nteraction
MP	**m**edium **p**ressure phase of the ECC (pumped, fed by water from the dousing tank)
MSSV	**m**ain **s**team **s**afety **v**alve, a spring-loaded valve on the secondary side of an SG
OPG	**O**ntario **P**ower **G**eneration Inc.
PHTS	**p**rimary **h**eat **t**ransport **s**ystem
PLGS	**P**oint **L**epreau **G**enerating **S**tation
PSA	**p**robabilistic **s**afety **a**ssessment
PT	**p**ressure **t**ube
P&IC	**p**ressure and **i**nventory **c**ontrol
PWR	**p**ressurized **w**ater **r**eactor
RIH / ROH	**r**eactor **i**nlet **h**eader / **r**eactor **o**utlet **h**eader
RV	**r**eactor **v**ault, also known as the calandria vault
SBO	**s**tation **b**lack**o**ut accident
SDCS	**s**hut**d**own **c**ooling **s**ystem
SFB	**s**tagnation **f**eeder **b**reak accident
SG	**s**team **g**enerator
SGTR	**s**team **g**enerator **t**ube **r**upture accident
SLOCA	**s**mall **l**oss **o**f **c**oolant **a**ccident
SSA	**s**hutdown **s**tate **a**ccident

11. References

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Table 1 Key input parameters for a Point Lepreau CANDU 6 Station

Total thermal power including gamma heating of moderator, required as input to MAAP4-CANDU, MW (th)	2155.9
ROH pressure, MPa (a)	10.0
SG secondary side pressure, MPa (a)	4.7
ROH coolant temperature, K	583
Moderator temperature in calandria vessel, K	339
Water temperature in calandria vault, K	327.15
Uranium dioxide mass in the core, kg	97,251
Zircaloy mass in the core (including pressure and calandria tubes), kg	41,944
D_2O inventory in PHTS (without pressurizer), kg	89,500
D_2O inventory in pressurizer, kg	23,400
D_2O inventory in calandria vessel, kg	229,810
H_2O inventory in calandria vault, kg	517,890
H_2O inventory in each SG secondary side, kg	32,733
Dousing system inventory available for containment sprays, m^3	1,559*
Pressure to open calandria vault rupture disk, kPa (g)	69
Pressure to open calandria vessel rupture disk, kPa (g)	138
Free volume space inside containment (assuming the dousing tank is empty), m^3	50,300
Thickness of the calandria vessel circumferential wall, m	0.0286
Flow area of calandria vault relief pipes (one pipe of 24 inch diameter and two pipes of 4 inch diameter), m^2	0.3081**
AECL passive autocatalytic recombiners, number	19

* Design value. The value used in each simulation varies depending upon analysis assumptions.
** It was assumed that the 24-inch rupture disk would be installed (an approved design change)

Table 2 Summary of simulated significant event timing (hours) for the most representative cases

Event/Case	SBO case D1	SLOCA Case E	SFB Case C	SGTR Case B	SSA Case A
SG dry	0.8	2	33	10.7	138
PT/CT rupture	3.8	41	38	13.1	N/A
Core disassembly starts	76	17	1.4	52	13.2
Containment fails	23	47	38	37	37.6
CV fails	N/A	81	54.5	120	66
MCCI begins	N/A	92	63	N/A	78
Calandria Vault floor failure	N/A	N/A	137	N/A	N/A
Fraction of initial (Cs+Rb+I) inventory released to environment at 500,000 s	3.2%	2.7%	6.8%	12.8%	0.55%