SEVERE ACCIDENT ANALYSIS OF A STEAM GENERATOR TUBE RUPTURE ACCIDENT USING MAAP-CANDU TO SUPPORT LEVEL 2 PSA FOR THE POINT LEPREAU GENERATING STATION REFURBISHMENT PROJECT

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Abstract

A Level 2 Probabilistic Safety Assessment was performed for the Point Lepreau Generating Station. The MAAP-CANDU code was used to simulate the progression of postulated severe core damage accidents and fission product releases.

This paper discusses the results for the reference case of the Steam Generator Tube Rupture initiating event. The reference case, dictated by the Level 2 Probabilistic Safety Assessment, was extreme and assumed most safety-related plant systems were not available: all steam generator feedwater; the emergency water supply; the moderator, shield and shutdown cooling systems; and all stages of emergency core cooling. The reference case also did not credit any post-Fukushima lessons or any emergency mitigating equipment.

The reference simulation predicted severe core damage beginning at 3.7 h, containment failure at 6.4 h, moderator boil off by 8.2 h, and calandria vessel failure at 42 h. A total release of 5.3% of the initial inventory of radioactive isotopes of Cs, Rb and I was predicted by the end of the simulation (139 h). Almost all noble gas fission products were released to the environment, primarily after the containment failure. No hydrogen/carbon monoxide burning was predicted.

1. Introduction

A Level 2 Probabilistic Safety Assessment (PSA) was performed by Atomic Energy of Canada Limited (AECL, now Canadian Nuclear Laboratories) for the Point Lepreau Generating Station (PLGS, a CANDU®6 reactor), as part of the Point Lepreau Refurbishment (PLR) Project [1]. An overview of PLGS refurbishment activities was given in Reference [2]. The Level 2 PSA quantified challenges to containment integrity, and the releases of fission products (FP) to the environment. Five representative initiating events, leading to severe core damage, were selected for the Level 2 PSA performed for the PLR Project:

1. Station blackout (SBO) [3],
2. Small loss-of-coolant accident (SLOCA) [4],
3. Stagnation feeder break (SFB) LOCA [5],
4. Shutdown state accident (SSA) [6], and
5. Steam generator (i.e., boiler) tube rupture (SGTR).

Each initiating event was analyzed with an extreme reference case, plus several sensitivity cases with different availabilities of accident mitigation systems, e.g., emergency core cooling (ECC), crash cool-down, and steam generator (SG) feedwater. This paper discusses only results for the reference SGTR scenario (SGTR Case A), a very low probability event.

MAAP-CANDU (Modular Accident Analysis Program) v4.0.5A+ [7] was used to estimate:

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- The timing of the accident progression and accompanying phenomena,
- The effect of the availability of safety and normal operational systems,
- Hydrogen and carbon monoxide concentrations in containment, and whether burning occurs (dependent upon gas, oxygen and steam concentrations),
- FP transport and retention within reactor systems and containment,
- The timing of challenges to containment integrity,
- The magnitude of FP releases from containment to the environment, and
- The effect of operator actions in mitigating severe accident consequences (challenges to containment integrity and FP releases from the reactor building).

SGTR Case A was the extreme reference case dictated by the Level 2 PSA; it assumed no operator interventions, credited few safety-related systems, and did not credit any post-Fukushima lessons nor any emergency mitigating equipment. The Level 2 PSA included a set of seven sensitivity cases that assumed various system availabilities, to assess their effects on the accident sequence, but the results of only reference Case A are discussed in this paper due to space limitations. SGTR Case A assumptions are listed in Section 6.2, but the major ones are:

- All emergency core cooling stages (high, medium and low pressure) were unavailable,
- All SG feedwater sources (main, auxiliary, dousing tank and deaerator tank) were unavailable,
- Containment local air coolers were unavailable,
- Shield cooling, moderator cooling and SG crash cooldown systems were unavailable,
- Available: containment dousing spray, grid (Class IV) and backup (Class III) power supplies, loop isolation, hydrogen recombiners, and main steam safety valves.

The analysis was performed in 2008 as part of the PLGS refurbishment process, so no post-Fukushima lessons or Emergency Mitigating Equipment was considered.

2. Brief Description of the MAAP-CANDU Code

The MAAP-CANDU code [7] can simulate severe accidents in CANDU stations, including many accident management actions. The code was based on the MAAP4 code developed by Fauske and Associates Inc. (FAI) [8]. MAAP4, owned by the Electric Power Research Institute, is used for severe accident analysis of light water reactors.

MAAP-CANDU simulates the most significant systems and components to demonstrate the overall response of a CANDU station to a severe accident. MAAP-CANDU tracks the mass and energy content of various heat sinks (liquid, solid and gas) within the reactor and containment, and the FPs within the fuel, Primary Heat Transport System (PHTS), containment, or released to the environment. Primary system thermalhydraulics is modeled simply, but some parameters can be adjusted so the PHTS better emulates results from dedicated thermalhydraulic codes. However, this has only a small effect upon the results; the primary purpose of MAAP-CANDU - to model the availability and use of heat sinks to absorb decay and chemical heat over an entire power plant, for extended periods well after the PHTS is dry.

3. Nodalization of the Point Lepreau Generating Station

Some details of the PLGS nodalization scheme are described in this section; more details are found in Reference [1].

The PLGS containment building was represented by 13 volumetric nodes, connected by 31 flow junctions. Concrete containment walls, floors and ceilings, and structural steel, were represented by 94 heat sinks. Adjacent rooms with large connecting openings were lumped together, as were similar heat sinks located within a containment node.

The PHTS was modelled as fourteen nodes in each of two PHTS loops, representing the pump discharge lines; reactor inlet headers (RIH); reactor outlet headers (ROH); inlet piping of both SGs; hot and cold leg tubes of both SGs; and the pump suction lines.

The 380 fuel channels were modeled with 36 characteristic fuel channels (2 loops, 6 vertical core nodes, 3 characteristic channels per vertical core node per loop). The fuel bundles in each characteristic channel were modelled as 12 axial channel nodes, and the calandria tube (CT), pressure tube (PT) and fuel bundle were modelled as nine concentric rings.

4. Failure Criteria

A brief description of the failure criteria used is given in the following sections (Reference [1] and [3] to [7] have more details). The assumptions are based on best estimates and engineering judgment, and do not reflect the only possible severe accident progression. Sensitivity studies of key assumptions are beyond the scope of this paper.

4.1 Containment failure criteria

MAAP-CANDU used a simple failure junction to model each airlock (equipment and personnel) linking the reactor building with the environment (i.e., the service building) [1]. The "environment" represented anything outside the containment boundary.

If the pressure difference exceeded 234 kPa between the containment and the environment, the failure junctions opened, based on small-scale experiments with airlock seals. The junction areas (0.0146 m^2 and 0.0039 m^2) represented the openings anticipated for airlock seal blowouts. Both the inner and outer airlock door seals were assumed to fail simultaneously, due to the small airlock volume between the doors. Other containment failure modes (e.g., containment cracking) were not modelled because the pressures required (>400 kPa) significantly exceeded those of the airlock seal failure, and the seal failures were sufficient to depressurize the containment.

4.2 Calandria vessel failure criteria

The water-filled calandria vault was able to remove the corium heat, due to the large surface area of the CV beneath the core debris, and the resulting relatively low heat flux. The insulating oxidic crust, a gap between the crust and CV, and adjacent heat sinks (steel ball-filled end shields and the CV located above the core debris) reduced the heat transfer to the calandria vault water.

MAAP-CANDU checks the calandria vessel (CV) conditions at every time step, to determine if any CV failure criteria are met. During the PLGS Level 2 PSA analyses, the CV always failed when the calandria vault water level dropped to the level of the top of the terminal debris bed within the CV. This user-input criterion assumes the CV fails because of insufficient external cooling on the cylindrical CV shell. The CV was assumed to fail at the bottom, so all the terminal debris poured into the calandria vault (no debris was assumed retained within the CV).

4.3 Fuel channel failure criteria

Fuel channel failure is a perforation of its pressure boundaries followed by mass and energy transfer from the inside of the PT to the CV (i.e., both the PT and CT fail).

At high PHTS pressures and fuel channel temperatures, MAAP-CANDU models channel failure when a PT and CT uniform balloons to the rupture criterion. When this happens, no fuel channel debris is formed. The user can force a channel to fail earlier, emulating hot spots or steep PT thermal gradients, to determine the effect upon the overall severe accident progression; only the built-in channel failure model was used in the current simulations.

At lower PHTS pressures, fuel channels may disassemble and form debris, due to local melt-through (Section 4.4). Channel sagging can perforate the CT, allowing CV steam into the fuel channel annulus, increasing the channel heat up rate due to the exothermic Zr-steam reaction.

4.4 Fuel channel disassembly criteria

Fuel channel disassembly occurs when sections of fuel and channel materials separate from intact channels or channel stubs. An axial fuel channel node disassembles when the average PT and CT temperature reaches the melting temperature of oxygenated Zr. A disassembled channel node becomes suspended debris, supported by underlying intact channels. The suspended core debris is tracked as separate components until it moves to the CV terminal debris bed

4.5 Core collapse criterion

The underlying channels, supporting channel debris, are immersed in the remaining moderator; thus they are cool and strong enough to support debris until a core collapse criterion is met.

The suspended core debris is modeled to relocate to the bottom of the CV, over a few hundred seconds, when the total suspended debris mass exceeds a user-specified value. A criterion of 25,000 kg, per PHTS loop, was used, based on the strength of cooled calandria tubes.

4.6 Fission product release criteria

The fuel is modeled as fuel rings comprised of UO_2 and Zr fuel sheath. A fuel sheath fails if its smeared fuel ring temperature exceeds 1,000 K, whereupon the fuel-sheath gap inventory of noble gas FPs in the affected fuel ring is released to the PHTS. This model is simple, but the resulting FP releases are small compared to later releases from hot fuel.

The release of FPs from the fuel matrix is based on fractional release models driven by the fuel temperature [9]. The initial FP distribution was based upon the local heat generation at normal operating conditions. MAAP-CANDU has no model of radioactive decay, but uses a total core decay heat curve. The decay heat is distributed amongst the FP groups based on a single time (the decay heat distribution does not change significantly past ~6 hours after reactor trip). MAAP-CANDU v4.0.5A+ assumed no FPs are released from the terminal debris bed that accumulates on the bottom of the CV, because the top debris crust acts as a barrier, although this model was revised in later code versions. The CV is dry by the time the terminal bed heats up enough to release FPs, so there is no steam flow through the debris to transport the FPs.

5. Key Input Parameters for the Point Lepreau Generating Station

The key PLGS input parameters were presented in Table 1 of Reference [1], including the initial normal operating conditions (e.g., total reactor heat: 2156 MW, PHTS outlet header pressure:

10.0 MPa, SG pressure: 4.7 MPa), UO_2 core mass: 97.25 tonnes; Zr core mass: 41.9 tonnes; and water inventories in the PHTS 112.9 tonnes, CV 229.8 tonnes, calandria vault 517.9 tonnes, SGs 130.9 tonnes, and dousing tank 1559 tonnes. In these analyses, the inventories of only the radioactive FPs were input to MAAP-CANDU.

6. Accident Description and Analysis Assumptions

MAAP-CANDU was used to simulate a variety of different initiating events that progressed to severe accidents (Section 1). For this paper, the initiating event was the rupture of a single steam generator tube.

6.1 Description of the SGTR accident

For an SGTR initiating event, the postulated severe accident starts with a relatively low loss rate of primary coolant via the ruptured SG tubes. It progresses to core damage and disassembly (i.e., a severe accident) because of the losses of: i) long-term ECC, ii) moderator cooling, and iii) end shield and calandria vault cooling. Other simulations performed for the PLGS project credited safety-related system functions (e.g., SG feedwater, ECC injection, crash cool-down), but these tended only to delay the event timing and are beyond the scope of this paper.

The expected sequence of events during an SGTR-initiated severe accident is as follows:

a) One or more SG tubes ruptures, allowing PHTS coolant to discharge into the secondary side of the SG, bypassing containment.

b) Secondary side SG water boils, removing PHTS heat. Increased SG pressure opens main steam safety valves (MSSVs) to environment. The initial inventory and availability (or not, as in Case A) of SG feed water determines duration SGs act as heat sinks, if enough PHTS coolant is available to transfer core decay heat to SGs. Ruptured SG tubes provide a source of SG makeup to the damaged SG.

c) Secondary side SG water is depleted, ending availability of SGs as heat sinks. Subsequently, PHTS heats up and pressurizes (ruptured SG tubes provide only a small opening to SGs, limiting their ability to relieve PHTS pressure).

d) PHTS liquid relief valves (LRVs) open and close, venting PHTS coolant to degasser condenser. Relief valves on degasser condenser vent to containment, when pressure reaches 10.1 MPa(g). When containment pressure exceeds 114 kPa(a), dousing is triggered.

e) Fuel is uncovered and heats up as the PHTS coolant level decreases, due to coolant losses through LRVs and ruptured SG tubes. Coolant boils off in channels.

f) Dry fuel channel(s) overheat and rupture, because PHTS is still pressurized; failed SG tubes cannot sufficiently relieve PHTS pressure, because ruptured tube is small and vents to pressurized SG secondary side.

g) Channel rupture and pressurization of CV bursts rupture disks and expels some moderator into containment. Remaining moderator boils off as fuel channels dry out on inside and transfer core decay and Zr oxidation heat to moderator.

h) When dry fuel channels are uncovered by decreasing moderator level, fuel channel sections overheat and disassemble. Lower intact channels, cooled by moderator, support debris but some (molten and solid) relocates to bottom of CV, where remaining moderator quenches it.

i) Majority of core drops to the bottom of the CV (core collapse) if load of debris exceeds ability of underlying channels to support it. Otherwise, core disassembles and relocates more slowly to CV bottom. Remaining moderator boiled off by core debris quench and decay heat.

j) Calandria vault water, outside CV, cools CV and thus core debris inside CV.

k) Calandria vault water boils off, due to decay heat from debris inside CV. Portions of CV walls heat up as vault water level decreases.

l) CV wall overheats and fails, when calandria vault water level decreases below level of top of debris in CV. Core debris relocates from CV to calandria vault floor. Remaining vault water boils off by debris quenching and decay heat.

m) Molten core-concrete interaction begins after last calandria vault water boils off. Corium ablates through calandria vault floor.

n) Molten corium pours into reactor building basement, where it is quenched by water.

o) Containment fails (airlock seals blow out) by over pressure, most likely during events *i)* core collapse, *k)* vault boil off, *l)* CV failure, or *n)* calandria vault failure, that produce large amounts of steam more rapidly than can be condensed by containment walls.

During many of the accident stages, core debris can release FPs to the containment. The FP release rate can be significant during the initial fuel heat up (via sheath failures), the suspension of hot core debris in the CV, and the core-concrete interaction. It was assumed there were no releases from the corium in the CV terminal debris bed.

6.2 Analysis assumptions

Severe accident processes are modeled by MAAP-CANDU, some in greater detail or sophistication than others. Several complex severe-core-damage processes (core debris formation and movement, debris-CV wall interaction, etc.) are based on engineering judgment due to a lack of experimental data. Some modeling assumptions are briefly described as they have a significant effect upon the accident progression or consequences.

The initiating event (time = 0 s) of the SGTR transient was a single SG tube rupture with a total break area of 1.375×10^{-4} m^2, equivalent to a one-sided blow-down of one SG tube [1]. The location of the break was in the “cold” SG leg (PHTS Node 6 in Figure 4 of Reference [7]), at the top of the SG tube-sheet. The simulation was run to 500,000 s (~139 h).

The SGTR-initiating event scenario was simulated with the following assumptions, based on system capabilities, MAAP-CANDU models, and PSA Level 2 requirements:

a) Total initial reactor heat: 2156 MW(th), including the heat deposited in the moderator by gamma heating, as required as an input to MAAP-CANDU.

b) The power grid (Class IV) and backup (Class III) power supplies were available.

c) Reactor shutdown was initiated at 0 s. MAAP-CANDU PHTS models are simple, and the PHTS thermalhydraulics results are not as accurate as from a dedicated thermalhydraulics code. Therefore, PHTS conditions that would automatically trip the reactor may not be modelled with sufficient accuracy by MAAP-CANDU, particularly for a small LOCA.

   The reactor was assumed to trip at 0 s. In an actual SGTR accident the trip might be delayed, meaning a larger loss of PHTS coolant via the broken SG1 U-tube, but this would have had a relatively small effect compared with the overall severe accident progression.

In addition, the SGTR simulations assumed other initial conditions / events (e.g., loss of cooling systems, SG feed water trip) starting at the same time as the SG U-tube break. Some of these conditions may have triggered reactor shutdown much sooner than the loss of PHTS coolant via the SG U-tube break.

d) Moderator cooling, shield cooling and shutdown cooling systems were unavailable after the reactor tripped.

e) The main SG, auxiliary feed water systems (plus deaerator and dousing tank supplies) were unavailable after the reactor tripped.

f) The main turbine stop valve was closed after the accident initiation. The code models the common steam header linking all the steam generators (there are no MSIVs at PLGS).

g) Containment leakage was modelled by an opening to the environment that allowed 2.5% containment volume to leak out per 24 hours, given a pressure difference of 124 kPa.

h) PHTS loop isolation was credited, occurring at a PHTS pressure of 5.52 MPa(g).

i) A total of 19 CNL passive autocatalytic recombiners (PARs) were credited.

j) Upon opening, the model LRVs were assumed to discharge PHTS coolant directly into containment. In reality, LRVs would discharge PHTS coolant into the degasser condenser, which may later discharge into containment via relief valves, but the degasser condenser was not modelled in these analyses.

k) Moderator cover gas system bleed valves, connecting the CV and vapour recovery system, opened and closed at a differential of 29 kPa between the CV and the containment, allowing some moderator to leave the CV. The valves were assumed operable for the first 7,200 s, then failed open after depleting their air reservoirs.

l) The SG MSSVs were available, opening and closing at their spring-operated set points (~5.1 MPa) to relieve the SG secondary side pressure. Crash cooling was unavailable.

m) Local air coolers were unavailable.

n) All ECC stages were unavailable.

o) Containment dousing spray was available.

p) D_2O supply, D_2O recovery and D_2O makeup to PHTS were unavailable.

7. Major Analysis Results for Case A

Significant Case A results are plotted in Figure 1 to Figure 13. Table 1 lists significant event timings, reported to the nearest second for consistency with the code output. Specifying times in seconds does not imply accuracy, which is expected to be in the order of hours.

7.1 Case A – Primary heat transport system and steam generator response

The initiating event for the reference SGTR accident scenario was the rupture of a single steam generator tube in PHTS Loop 1, at 0 s. The pressure in the PHTS loops and pressurizer rapidly decreased from an initial 10 MPa(a) to ~9 MPa (Figure 1), because:

1. Fission ceased in the reactor core, so the heat source decreased to decay heat,
2. Core decay heat was transferred via the PHTS to the SGs, which had sufficient heat sink capacity (secondary side water) to absorb much of the decay heat until SG dry out, and

3. Primary coolant was lost through the Loop 1 SG1 U-tube break (Figure 2). The break was small (1.375×10^{-4} m^2), and had little effect on the PHTS pressure compared to the previous two reasons. The break flow was, however, very important for the PHTS loop coolant masses and SG1 refill.

The PHTS pressure started to increase again at ~200 s (Figure 1), because the pressurizer heaters operated when the PHTS pressure fell below 10.1 MPa(a). The pressurizer heaters were turned off at 1,324 s (0.4 h, Table 1) due to low water level in the pressurizer, causing the PHTS pressure to decrease rapidly to 5.7 MPa(a) (Figure 1).

Due to the PHTS heat and the closure of the turbine stop valve, the SG secondary side pressure (Figure 3) quickly increased from the initial 4.7 MPa(a) to 5.1 MPa(a). The MSSVs opened when the pressure reached 5.11 MPa(a), and discharged steam to the environment. The pressure stayed at 5.1 MPa(a) as the MSSVs opened and closed at the MSSV set point.

The SG water level decreased (Figure 4) as the water boiled and steam discharged via the MSSVs. Figure 2 shows the PHTS coolant loss rate via the broken SG1 U-tube. The Loop 2 coolant mass (Figure 5) was almost constant up to ~2,000 s (0.5 h), when it began to decrease.

The Loop 2 coolant began to boil at 1,821 s (0.5 h), increasing the pressure so coolant flowed to the pressurizer and Loop 1, and the Loop 2 void increased. Loop 1 lost coolant via the SG tube rupture, but its inventory was largely replenished from Loop 2 during this period; therefore, the Loop 2 inventory decreased sooner than in Loop 1.

Loop 2 SGs 3 and 4 dried out by 2,863 s (0.8 h, Table 1), so they no longer provided a PHTS heat sink. Subsequently almost all Loop 2 decay heat went into boiling its primary coolant, which further pressurized Loop 2 (Figure 1) and increased the coolant loss to Loop 1 (Figure 5).

Since the PHTS loops and pressurizer were not isolated, the common PHTS pressure increased until the LRVs in both loops opened at 3,327 s (0.9 h). The LRVs opened and closed to relieve the pressure from the continuing steam production. Meanwhile, the intact Loop 1 SG2 dried out at 3,468 s (1.0 h), and the broken SG1 dried out at 4,219 s (1.2 h), so the Loop 1 SG heat sink was lost (Figure 4) and Loop 1 coolant began to boil. The resulting steam generation from both PHTS loops could no longer be accommodated by the LRVs, so the PHTS pressure increased again. The LRVs opened continuously to relieve the PHTS pressure, and did not close until after the Loop 1 coolant separated into two phases at 4,920 s (1.4 h), when the Loop 1 LRV flow switched from homogeneous two-phase to steam (Figure 2).

Loop 2 Channel 1 (high power, top core node) was the first characteristic fuel channel to dry out (7,229 s, 2.0 h). Both loops were still at 10.1 MPa(a), the LRV set point, so Channel 1 heated up to rupture (7,390 s, 2.0 h). The Loop 2 pressure rapidly decreased to the CV pressure (~1 atmosphere, Figure 1), which triggered the loop isolation valves to close (7,510 s, 2.1 h). Thus Loop 2 depressurized via its ruptured channel, but Loop 1 remained intact, except for the ruptured SG1 tube connecting it to the SG secondary side at 5.1 MPa(a).

The PHTS coolant continued to boil so the Loop 1 pressure increased. The Loop 1 coolant mass decreased further after loop isolation, due to the flow through the ruptured SG tube (Figure 2), but the pressure remained well below that needed to open the Loop 1 LRVs (Figure 1).

Loop 1 characteristic Channel 1 (high power, top core node) dried out at 9,076 s (2.6 h), followed soon after by Channel 4 (high power, second from top core node). Channel 4 heated to rupture at 9,337 s (2.6 h, Table 1), when the Loop 1 pressure decreased rapidly to the CV pressure.

The SG secondary side pressure was 5.1 MPa(a) when the Loop 1 channel ruptured; the SG pressure then decreased through a reverse flow to the PHTS and CV (Figure 2), reaching the CV pressure at ~80,000 s (22.2 h), Figure 3. The SG secondary side pressure changed slightly over the remainder of the accident, because it remained connected to containment via the ruptured SG tube, the Loop 1 PHTS, the ruptured Loop 1 channel and the burst CV rupture disks.

7.2 Case A – Fuel and fuel channel response

The first characteristic fuel channel dry out was Channel 1 in Loop 2 (high power channel, top core node) at 7,229 s (2.0 h, Table 1). The coolant from intact Loop 2 flowed to Loop 1 because the loops were not isolated until 2.1 h, so Loop 1 channels were kept filled for longer. Loop 2 fuel elements were uncovered as a result of: a) the loss of PHTS coolant to Loop 1 through the still-open isolation valves, and b) coolant boil-off due to the core decay heat.

Channel 1 was the first to dry out in Loop 1 at 9,076 s (2.6 h), about 0.5 h after those of Loop 2. Figure 6 shows example pressure and calandria tube temperatures, and fuel ring 1, for Loop 1 Channel 2 (top core node) axial Node 7 (typically the hottest channel node).

The moderator reached saturation at 7,691 s (2.1 h); it still covered all the calandria tubes and kept them cool by convection and nucleate boiling heat transfer. The moderator acted as a heat sink to prevent fuel from melting and channels from disassembling. At 10,943 s (3.0 h), a CV rupture disk burst and ~70 tonnes of moderator was expelled. The moderator level continued to boil off. Channel 2 (both loops) was uncovered at ~12,700 s (3.5 h), so the moderator no longer cooled Channel 2 and the CT, PT and fuel temperatures increased rapidly (Figure 6).

Fuel and fuel channel temperatures increased in dry and uncovered channels, due to the decay heat and the exothermic zirconium-steam reaction. The first core disassemblies occurred at 13,197 s (3.7 h) in Loop 1, and at 13,810 s (3.8 h) in Loop 2. Channel 2 Bundle 7 in Loop 1 disassembled at 13,440 s (3.7 h), and the same bundle in Loop 2 disassembled at 14,426 s (4.0 h).

The dynamics of channel disassembly and relocation phenomena are shown in Figure 7. The mass of UO_2 in the intact core began to decrease at 13,197 s (3.7 h) for Loop 1, and decreased steadily thereafter. In Loop 2, the decrease in the intact UO_2 mass followed a similar path, although it began at 13,810 s (3.8 h) and continued to be delayed relative to Loop 1. At 20,963 s (5.8 h), the total mass of suspended debris in Loop 1 exceeded the 25,000 kg value to trigger core collapse, and all the Loop 1 suspended debris dropped to the terminal debris bed.

The Loop 2 debris did not accumulate enough to trigger a core collapse, so the suspended debris continued to relocate more slowly to the CV terminal debris bed. By 30,107 s (8.4 h) 93.7% of the entire core (both loops) had disassembled (Figure 7), but it was not until 35,508 s (9.9 h) that the Loop 2 suspended debris finished relocating to the terminal debris bed. About 6% of the Loop 2 channel nodes did not disassemble during the simulation, because they represented low power peripheral channel stubs that did not attain the disassembly criteria.

7.3 Case A – Calandria vessel response

The moderator level (Figure 8) increased gradually from 8.05 m (measured from the CV bottom) at the beginning of the accident. The moderator cooling system was assumed inoperable for this case so the moderator swelled from thermal expansion. The moderator cover gas system bleed valves opened at 1,639 s (~0.5 h, Table 1). This slowed the rise in the two-phase moderator level by venting water (Figure 8). At 7,200 s the valves were assumed to fail open, having run out of instrument air (Section 6.2, item k).

When the Loop 2 fuel channel ruptured at 7,390 s (2.0 h, Table 1), the steam blown into the moderator increased the two-phase level (Figure 8) and pressurized the CV (Figure 9). The moderator reached saturation at 7,691 s (2.1 h); however, it took until 10,943 s (3 h) for the CV rupture disk to burst, because the containment pressure was also increasing from the continued discharge of PHTS coolant via the LRVs. After the CV rupture disk opened, the CV pressure remained a little higher than the containment pressure, due to the steam flow out the relief duct.

After the CV rupture disk burst, the moderator boiled extensively because it had been supersaturated relative to containment pressure, and the CV pressure dropped to approximately containment pressure. The net result was an expulsion of ~35% of the initial moderator mass into containment. By the time the expulsion ended (~11,500 s, ~3.2 h), the two-phase moderator level had dropped to ~6.8 m above the CV bottom (Figure 8). This was still above the topmost characteristic fuel channel, but heat from the fuel channels continued to boil the moderator and so the level dropped. By 29,646 s (8.2 h), the CV was dry on the inside but cooled on the outside by the water in the surrounding calandria vault.

Core debris built up on the CV floor due to the channel disassembly (Figure 7). Figure 10 shows the total corium mass (particulates + crust + molten), the mass of the corium crusts (bottom, side and top surfaces) and the mass of particulates in the CV. Particulates were solid core debris particles formed during fragmentation of the molten core material, as it poured down from the suspended debris into the remaining moderator. A crust, ~5 to 10 cm thick, formed on the CV walls as the core material finished relocating to the CV bottom. After the CV water was depleted, the core debris in the CV started to heat up; eventually the particulates melted to form the molten debris pool at ~42,000 s (~11.7 h, Figure 10).

The calandria vault water level decreased to the elevation of the top of the CV debris bed at 151,436 s (~42 h), triggering the CV wall failure (assumed to occur at the bottom of the CV, Section 4.2). All the core debris then relocated out of the CV and onto the calandria vault floor; it was assumed that no crust remained in the CV. The corium was predicted to rapidly quench in the calandria vault water, when the CV cylindrical shell failed.

7.4 Case A – Calandria vault and end shields response

The pressure (Figure 9) and water level (Figure 11) in the calandria vault and end shields increased gradually after the initiating event, due to the unavailability of the shield and moderator cooling systems and the resulting thermal expansion of the calandria vault water. The calandria vault and end shields are connected via combined vent lines, and a rupture disk can relieve over-pressure of the combined system. At 10,538 s (2.9 h, Table 1), this rupture disk burst. Steam was discharged from the end shields to the containment, resulting in a decrease of the calandria vault and end shield water levels. The water in the calandria vault began to boil off at about 42,163 s (11.7 h), which resulted in a gradual water level decrease.

The CV failed at 151,436 s (42 h). As a result, all the core debris poured out onto the calandria vault floor (Figure 10); the core quench caused a pressure spike in the vault and containment (Figure 9) and by 171,651 s (48 h) the last of the calandria vault water had boiled off.

After the loss of the calandria vault water the corium temperature increased due to the decay heat, and molten corium - concrete interaction began at 180,616 s (50 h). When the eroded depth of the concrete floor reached 2 m at 433,437 s (120 h), the calandria vault was assumed failed. At that time, the debris from the reactor vault relocated to the reactor building basement, where it was quenched in the water pool, resulting in a final containment pressure spike (Figure 9).

7.5 Case A – Containment response

Figure 9 shows the pressure in the lower half of the SG enclosure, one of the largest containment compartments. The rapid containment pressure increase, during the initial period of the accident, was due to the hot PHTS coolant discharging through the PHTS LRVs and flashing to steam. Further events added to the containment pressurization: at 7,200 s (2.0 h) the CV bleed/relief valves failed open, and a CV rupture disk burst at 10,943 s (3 h, Table 1). The moderator began to boil at 7,691 s (2.1 h) and a Loop 2 fuel channel ruptured at 7,390 s (2.0 h); when the CV rupture disk burst, the Loop 2 PHTS coolant plus steam from the boiling moderator discharged into containment via the open CV relief duct, rapidly increasing the containment pressure.

The rapid increase in containment pressure at ~21,000 s (5.8 h) was caused by the core (Loop 1) collapsing to the CV bottom, followed by steam generation as the remaining moderator quenched the debris. This pressurization blew out an airlock seal (234 kPa(g), Section 4.1) at 22,927 s (6.4 h). The containment pressure continued to increase until 23,532 s (6.5 h), when the second airlock blew out, because the steam generation rate exceeded the steam/gas mixture discharge rate to the environment plus the condensation rate within containment.

The containment pressure continued to decrease until, at 42,163 s (11.7 h), the calandria vault water reached saturation conditions (the calandria vault rupture disk burst at 10,538 s, 2.9 h). The increased steam flow into containment exceeded the discharge through the blown out airlock seals plus the condensation in containment, so the containment pressure began to increase again.

The containment pressure increased slowly (Figure 9) after the calandria vault water began to boil; it reached a maximum of 142.1 kPa(a) at 93,240 s (25.9 h), when the steam generation matched the discharge plus condensation in containment. The containment pressure rapidly increased, due to corium relocating into the calandria vault after the calandria vessel failed at 151,436 s (42 h, Table 1), followed by steam generation as the hot core debris quenched in the calandria vault water. The containment pressure decreased soon after, mainly from the steam discharge through the airlock seals. In addition, the corium was quenched by the calandria vault water so the steam generation was reduced, and steam condensed on the containment walls. The containment pressure decreased to about atmospheric pressure by ~190,000 s (~52.8 h), because the calandria vault water was depleted (171,651 s, 48 h) and thus the steam generation ended.

The containment pressure rapidly increased at 433,437 s (120 h), due to corium relocating into the basement after the calandria vault failed, followed by steam generation as the hot core debris was quenched in the basement water. The containment pressure decreased soon after, mainly because of the steam discharge through the airlock seals. Also, the corium was quenched in the basement water (steam generation ceased temporarily) and steam condensed on the containment walls and internal structures. The simulations did not use a criterion for a large-scale failure of containment, only one for the airlock seals blowout.

7.6 Case A – Fission product release and distribution

After 7,229 s (~2 h, Table 1), the first fuel channels became dry and fuel element temperatures exceeded 1000 K, so FPs began to be released from fuel elements (Section 4.6). The initial inventory of noble gases (Kr and Xe, radioactive isotopes only) was 1.04 kg. Most of the noble gases were released into the CV from the fuel and the suspended debris bed during core disassembly (13,197 to ~29,000 s, ~3.7 to ~8 h), Figure 12. The noble gases vented to containment through the burst CV rupture disk, and some escaped to the environment (Figure 12) via containment leakage and through the blown out airlock seals. Eventually all noble gases escaped to the environment, following containment failure at 22,927 s (6.4 h).

Figure 13 shows the masses of radioactive CsI released from fuel and debris a) in-vessel (inside the CV), b) ex-vessel (outside the CV), and c) from containment to the environment. The initial inventory of CsI in the core (radioactive isotopes) was 17.1 kg. Fission products were released to the CV from fuel in the core and from the suspended debris bed. When the CV rupture disk burst open at 10,538 s (2.9 h), FPs were transported from the CV into containment. It was assumed that the corium crust, formed in the CV, prevented further in-vessel FP releases. When the CV failed, the crust failed but the debris was quenched in the remaining calandria vault water, so any further FP releases would have been very low. After the calandria vault water was depleted (171,651 s, 48 h), the corium heated up and reacted with the concrete starting at 180,616 s (50 h); the corium-concrete interaction allowed more ex-vessel FPs releases (Figure 13).

The release of FPs to the environment was almost entirely due to containment leakage and to failed air lock seals. In Case A, the MSSVs were not locked open (crash cool-down was not credited), so these valves opened and closed based on the secondary side pressure. Therefore, this scenario had a limited “containment bypass” lasting from 2 s until the PHTS depressurized when Loop 1 Channel 4 ruptured at 9,337 s (2.6 h). In addition, the fission product model in MAAP-CANDU was not activated until 3,407 s (0.9 h), preventing any FP releases until after that time (note that all fuel was covered until 3,925 s, and the first channel was not dry until 7,229 s). Any FPs released from the fuel, and transported through the broken SG1 U-tube to the SG secondary sides, could have been released to the environment via the MSSVs. Only about 0.0001% of the noble gas inventory was released via this route.

At the end of the simulation (500,000 s, 138.9 h), the total mass of Cs, Rb and I (radioactive isotopes only), released to the environment, was 5.3% of the initial inventory of these isotopes.

7.7 Case A – Hydrogen release

Hydrogen was generated by steam reacting with Zr in hot and dry fuel channels, and in the suspended debris beds of both PHTS loops. The first fuel channels dried out in Loop 2 at 7,229 s (2.0 h, Table 1), and in Loop 1 at 9,076 s (2.6 h). Steam inside the channels reacted with the Zr fuel cladding and inner surface of the Zr PT. This exothermic reaction produces hydrogen, and the reaction rate increases at higher temperatures. The hydrogen production increased after core disassembly started at 13,197 s (3.7 h) in Loop 1, and at 13,810 s (3.8 h) in Loop 2. At 29,646 s

(8.2 h) the CV had dried out, significantly decreasing hydrogen production due to the lack of fresh steam. A small amount of hydrogen was produced from the remaining intact channel stubs still located in the CV. The solid crust, formed on top of the CV debris bed, was assumed to prevent steam access to the debris.

The total amount of hydrogen produced during the core heat up and disassembly was 372 kg, which was produced inside of the PHTS loops and from the debris suspended inside the CV. This corresponds to ~20% of the initial Zr core inventory being consumed.

Hydrogen was generated in the calandria vault due to the molten core-concrete interaction (MCCI), which began at 180,616 s (50 h). In addition to zirconium oxidation, iron from rebar in the concrete was also oxidized to produce hydrogen, during the MCCI period. After the calandria vault floor failed (433,437 s, 120 h), hydrogen was no longer generated because the molten corium had relocated to the basement where it was quenched in the basement water. The total amount of hydrogen produced outside calandria vessel was 2,452 kg, almost all from the MCCI.
No hydrogen or carbon monoxide burning in containment was predicted in Case A, due largely to the airlock seals failure and the subsequent steam generation that purged containment.

8. Summary

A series of eight simulations (reference Case A plus seven sensitivity cases) were performed to assess the consequences of a severe accident, initiated by a steam generator (i.e., boiler) tube rupture, as part of the Level 2 PSA performed for the Point Lepreau Refurbishment project. These simulations were run using the severe accident analysis code MAAP-CANDU v4.0.5A+ and a PLGS specific parameter file.

Only reference Case A was discussed in this paper; it assumed the most extreme conditions where most safety-related plant systems were not available, and no post-Fukushima emergency mitigating equipment was credited:

Unavailable: Main and auxiliary SG feedwater, emergency water supply, moderator cooling, shield cooling, shutdown cooling, and emergency core cooling (all stages).

Available: Reactor shutdown, dousing spray, class III and class IV power, PARs, SG MSSVs.

In reference Case A, severe core damage began at 3.7 hours, the moderator was depleted at 8.2 h, containment failed at 6.4 h, and the CV failed at 42 h. The total release of radioactive isotopes of Cs, Rb and I was 5.3% of their initial inventory. Almost the entire initial noble gas inventory in the core was transported with the containment atmosphere to the environment, primarily after the containment failure. No hydrogen/carbon monoxide burning was predicted in the containment.

This reference case was extreme, as dictated by the PLGS Level 2 PSA, and does not reflect lessons learned neither post-Fukushima nor any emergency mitigating equipment.

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11. Acronyms

AECL	Atomic Energy of Canada Limited
BTF	Blowdown Test Facility
CANDU	CANada Deuterium Uranium reactor
CNL	Canadian Nuclear Laboratories (formerly AECL)
CT	Calandria Tube
CV	Calandria Vessel

ECC	Emergency Core Cooling
FAI	Fauske and Associates Inc.
FP	Fission Products
LOCA	Loss Of Coolant Accident
LRV	Liquid Relief Valve
MAAP	Modular Accident Analysis Program
MCCI	Molten Core-Concrete Interaction
MSIV	Main Steam Isolation Valve (between SG and turbine)
MSSV	Main Steam Safety Valve (on SGs)
OPG	Ontario Power Generation Inc.
PAR	Passive Autocatalytic Recombiner
PLGS	Point Lepreau Generating Station
PHTS	Primary Heat Transport System
PLR	Point Lepreau Refurbishment project
PSA	Probabilistic Safety Assessment
PT	Pressure Tube
RIH/ROH	Reactor Inlet Header / Reactor Outlet Header
SBO	Station BlackOut accident
SFB	Stagnation Feeder Break accident
SG	Steam Generator
SGTR	Steam Generator Tube Rupture accident
SLOCA	Small Loss Of Coolant Accident
SSA	Shutdown State Accident

Table 1
Sequence of Significant Events for SGTR Reference Case A

		Case A: SGTR-PLR18BN1
Time (h)	**Time (s)***	**Event**
0.0	0	SG1 in Loop 1 has U-tube rupture (one-sided blow-down)
0.0	0	Moderator cooling and shield cooling system off, Turbine main stop valves closed
0.0	0	Reactor trip, SG main and auxiliary feed water assumed unavailable,
0.0	0	Pressurizer heater on
0.4	1,324	Pressurizer heater off
0.4	1,457	Pressurizer empty (first time)
0.5	1,639	Calandria vessel bleed valve first opens
0.5	1,821	PHTS in Loop 2 reaches saturation conditions
0.8	2,863	Secondary side dry in both SGs, Loop 2
0.9	3,327	LRVs first opening, Loops 1 and 2
0.9	3,398	Dousing system starts
1.0	3,468	Secondary side of unbroken SG is dry, Loop 1
1.0	3,489	Primary coolant phases are separated in Loop 2
1.0	3,923	Dousing system exhausted
1.2	4,219	Secondary side of broken SG is dry, Loop 1
1.2	4,452	PHTS in Loop 1 reaches saturation conditions
1.4	4,920	Primary coolant phases are separated in Loop 1
2.0	7,200	Calandria vessel bleed valves fail open (stay open up to the end of the simulation)
2.0	7,229	At least one channel is dry in Loop 2 (complete boil-off) – Channel 1
2.0	7,390	Pressure and calandria tubes rupture in Loop 2 – Channel 1
2.1	7,510	PHTS loops isolated when pressurizer isolated
2.1	7,691	Moderator in calandria vessel reaches saturation temperature
2.6	9,076	At least one channel is dry in Loop 1 (complete boil-off) – Channel 1
2.6	9,337	Pressure and calandria tubes rupture in Loop 1 – Channel 4
2.9	10,538	Calandria vault rupture disk bursts open, connecting to SG room
3.0	10,943	Calandria vessel rupture disk 1 bursts open
3.7	13,197	Beginning of core disassembly, Loop 1
3.8	13,810	Beginning of core disassembly, Loop 2
5.8 N/A	20,963 N/A	Core collapse, Loop 1 Core collapse, Loop 2
6.4	22,927	Containment fails (small airlock seals failed)
8.2	29,646	Water is depleted inside calandria vessel
11.7	42,163	Water in calandria vault reaches saturation temperature
42	151,436	Calandria vessel fails
42	151,441	Energetic core debris-steam interaction occurred in calandria vault
48	171,651	Water is depleted inside calandria vault
50	180,616	Molten corium-concrete interaction begins in calandria vault
120	433,437	Calandria vault floor fails

* The number of significant digits does not reflect code accuracy.

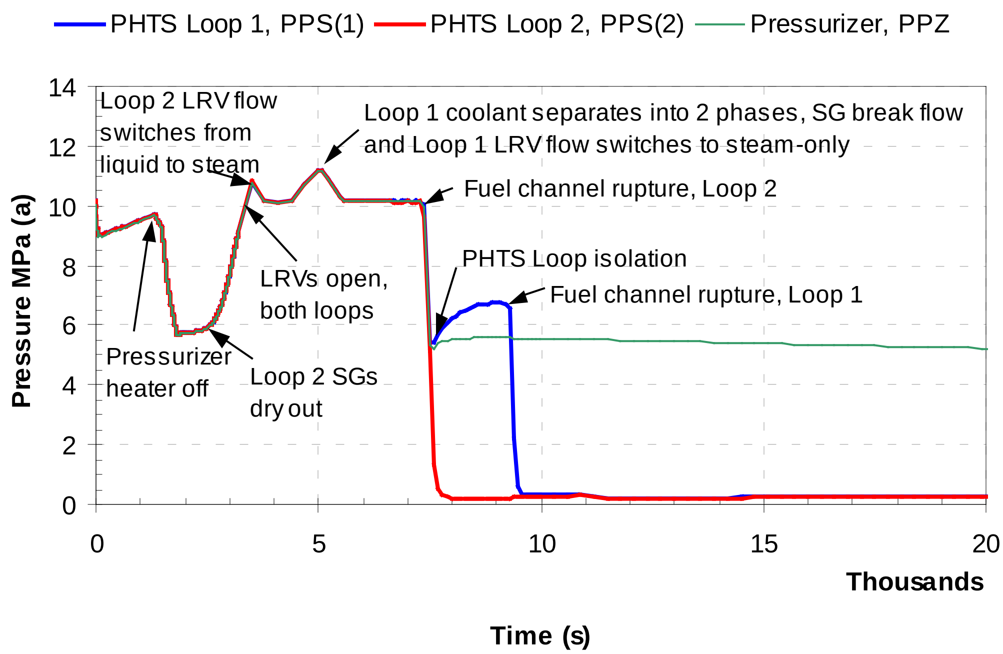

Figure 1 Pressures in the primary heat transport system and pressurizer, Case A (0 – 20,000 s)

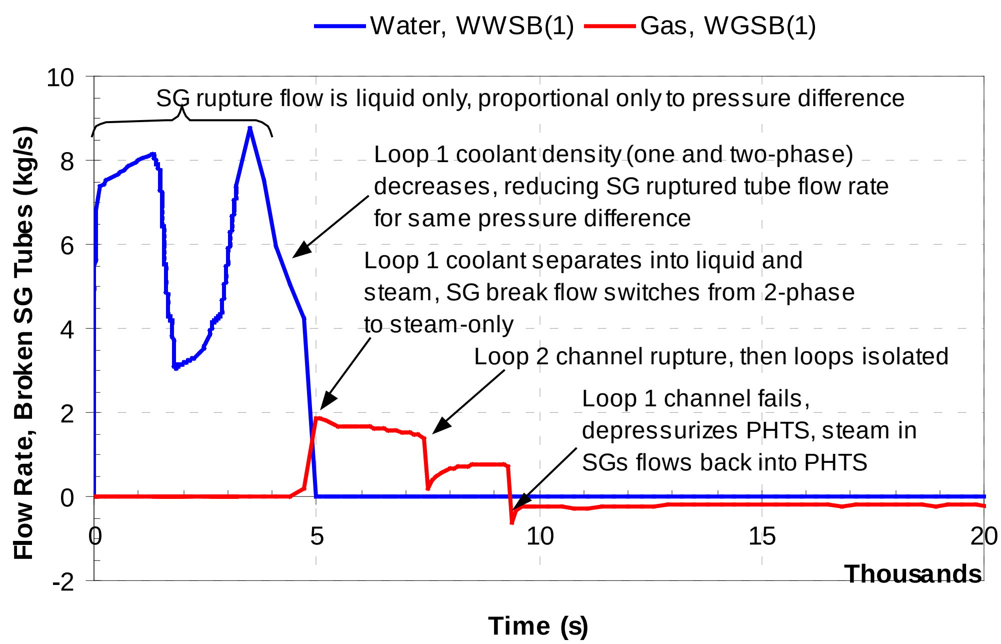

Figure 2 Break flow through the ruptured Loop 1 SG1 U-tube, Case A (0 – 20,000 s)

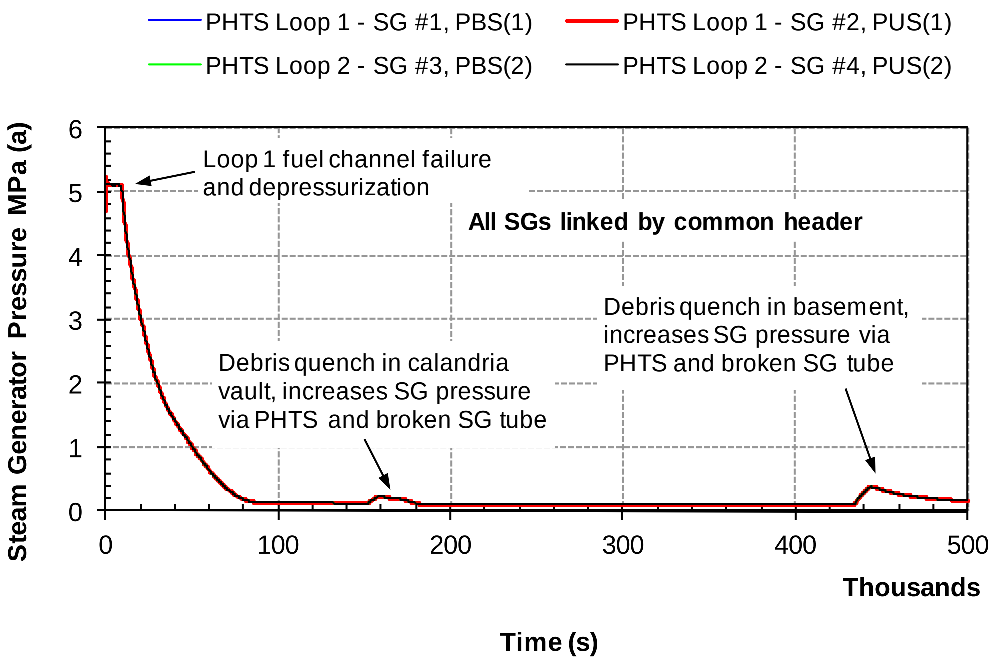

Figure 3 Steam generator secondary side pressure, Case A (0 – 500,000 s)

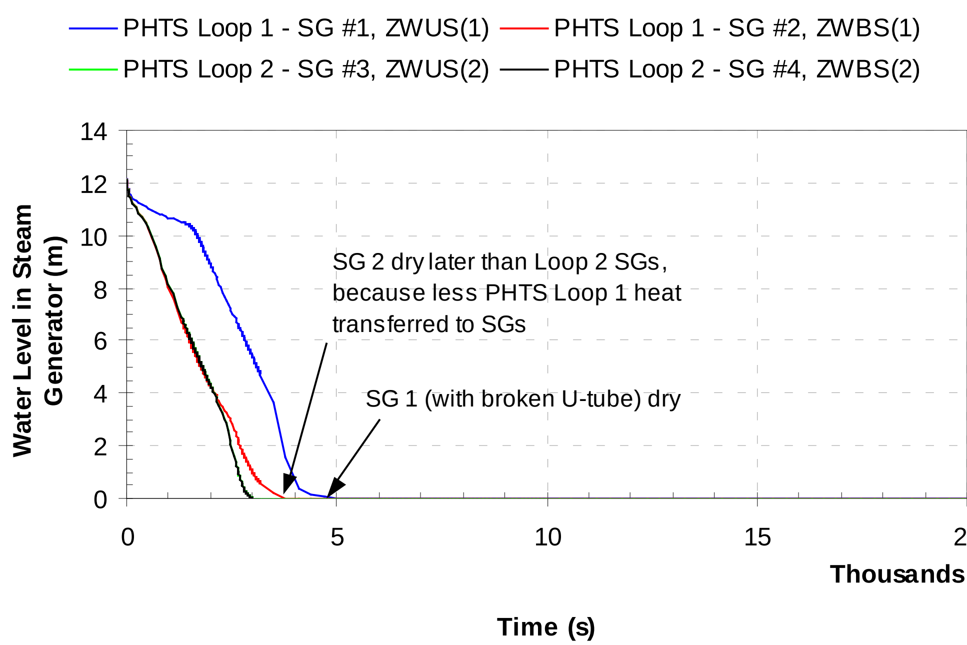

Figure 4 Steam generator water level, Case A (0 – 20,000 s)

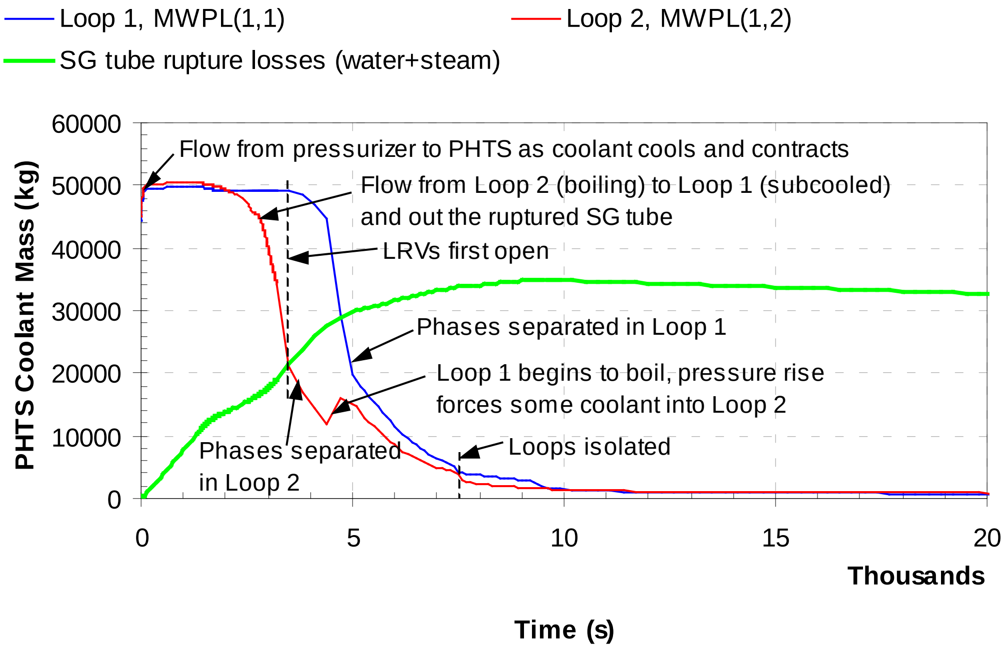

Figure 5 Coolant masses in both PHTS loops and integrated coolant loss via the SG tube rupture, Case A (0 - 20,000 s)

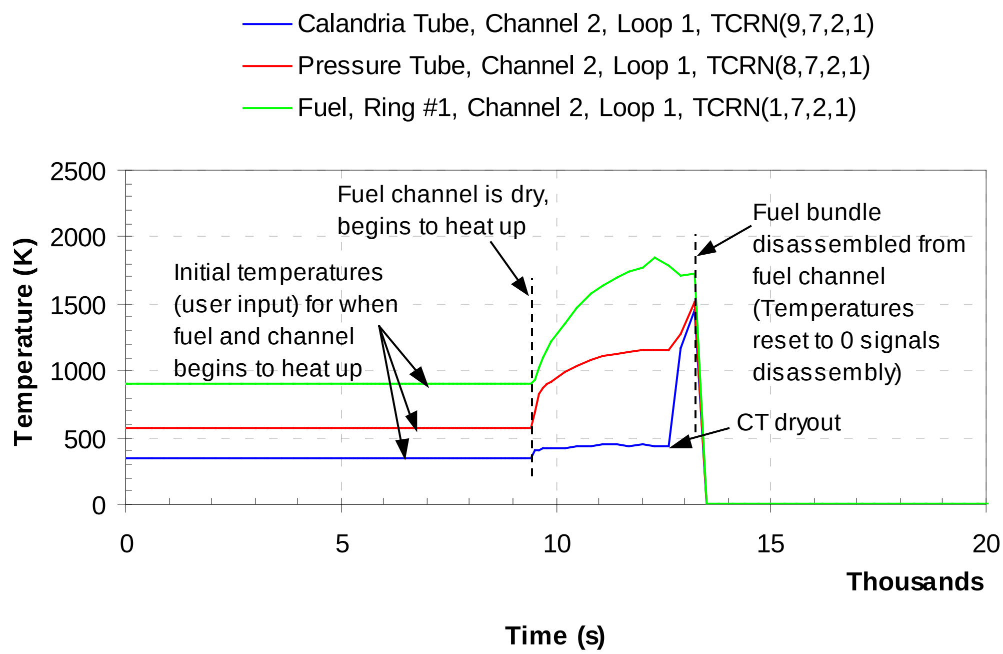

Figure 6 Fuel and fuel channel temperatures of Channel 2 Node 7 in Loop 1, Case A (0 - 20,000 s)

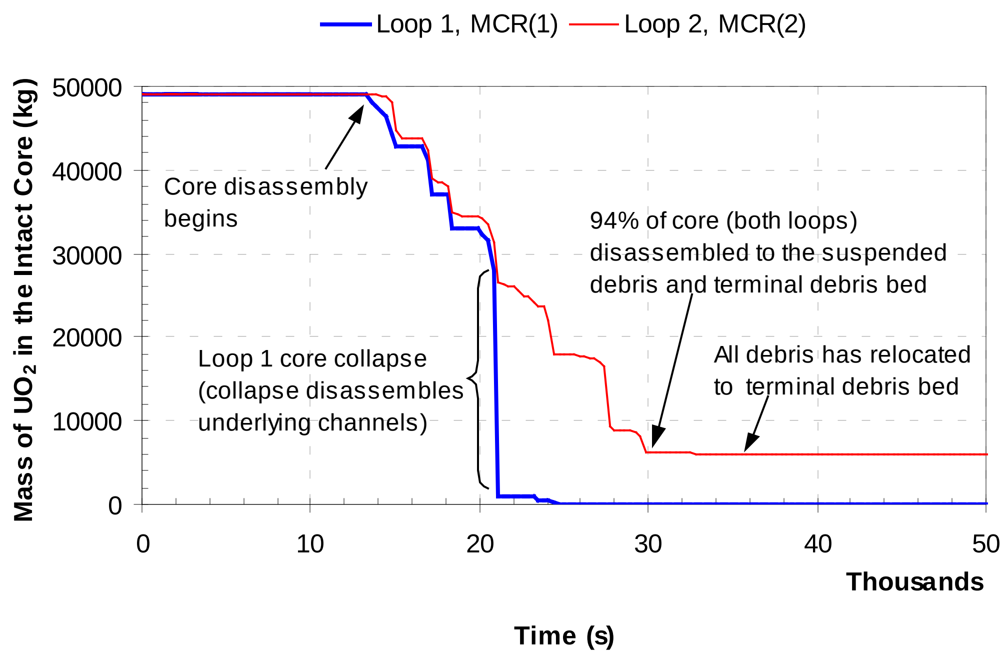

Figure 7 UO_2 mass remaining in the intact fuel channels, Case A (0 - 50,000 s)

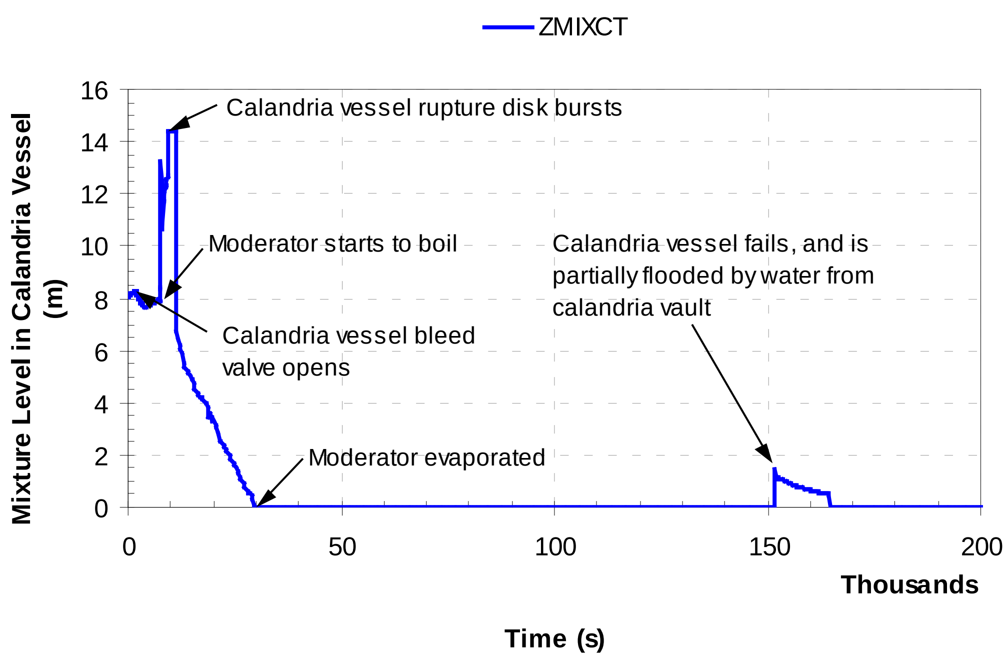

Figure 8 Calandria vessel two-phase water level, Case A (0 – 200,000 s)

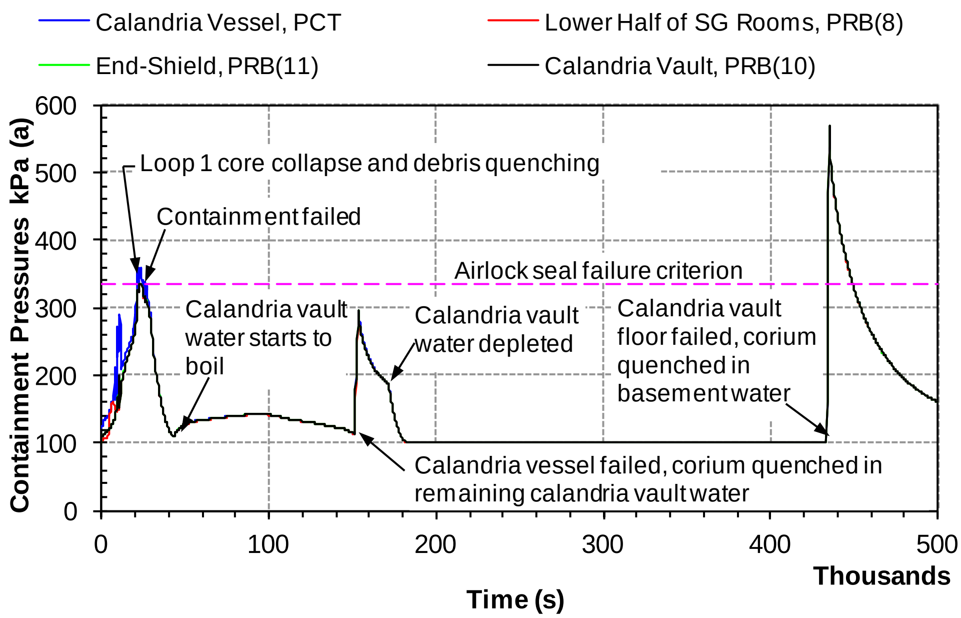

Figure 9 Containment, calandria vessel and calandria vault pressures, Case A (0 - 500,000 s)

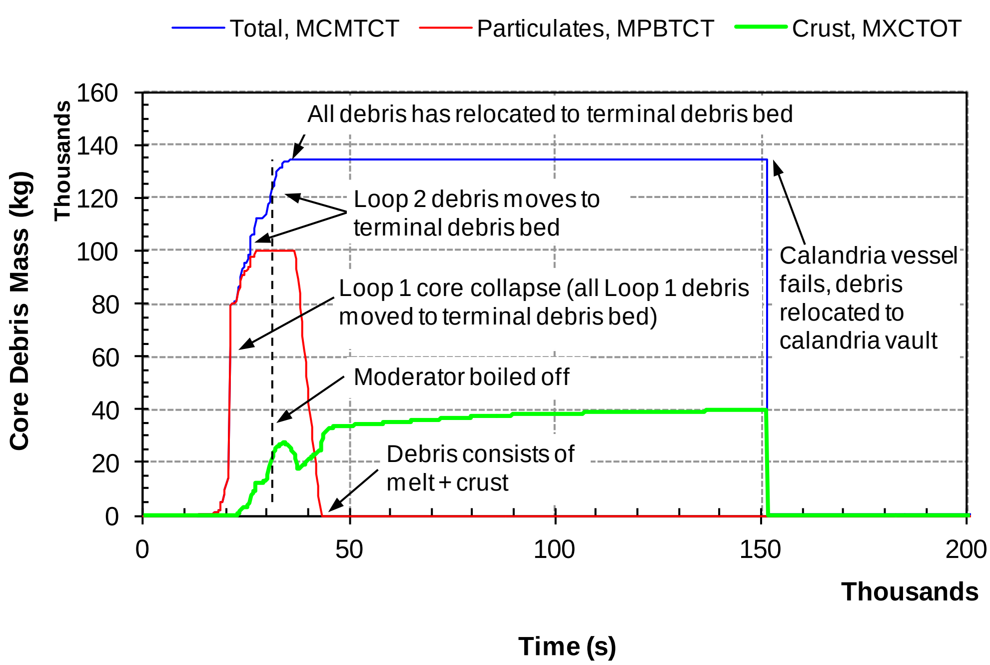

Figure 10 Mass of corium (crust, particulates and total) in the calandria vessel, Case A (0 - 200,000 s)

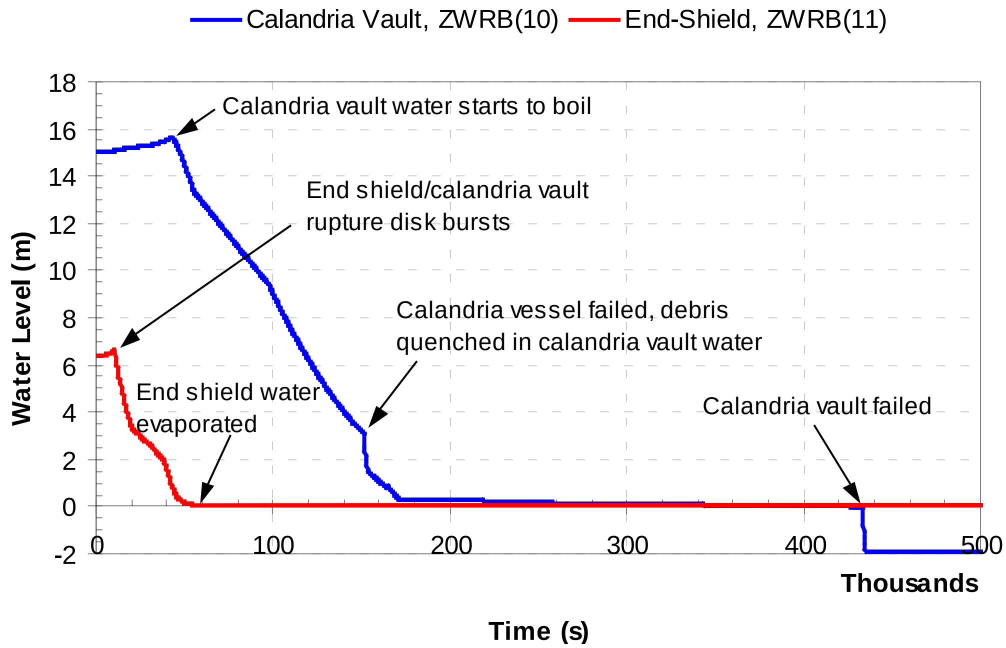

Figure 11 Calandria vault and end shield water level, Case A (0 – 500,000 s)

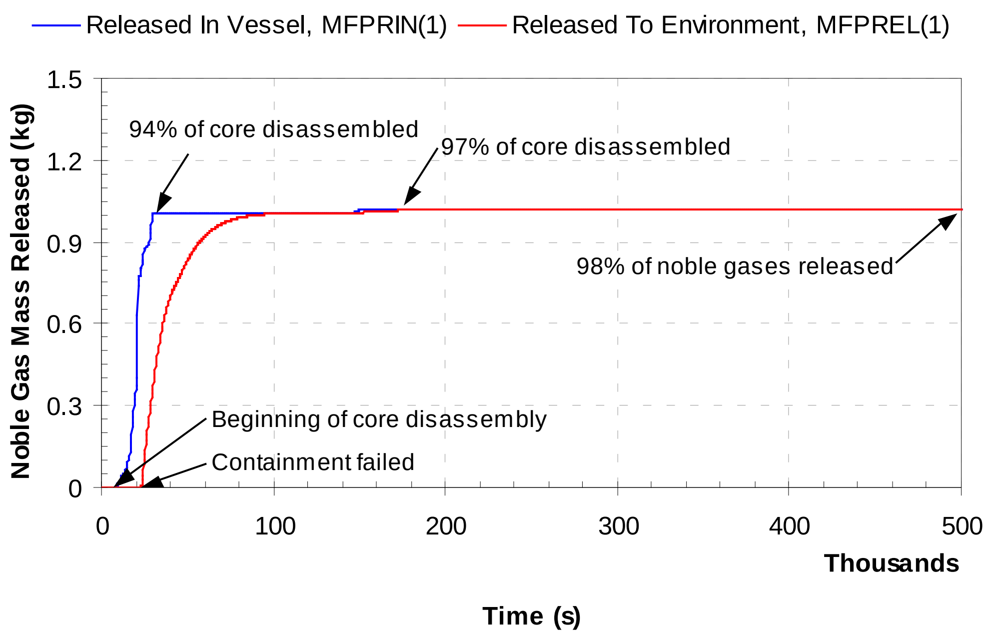

Figure 12 Mass of noble gases released (active components only), Case A (0 - 500,000 s)

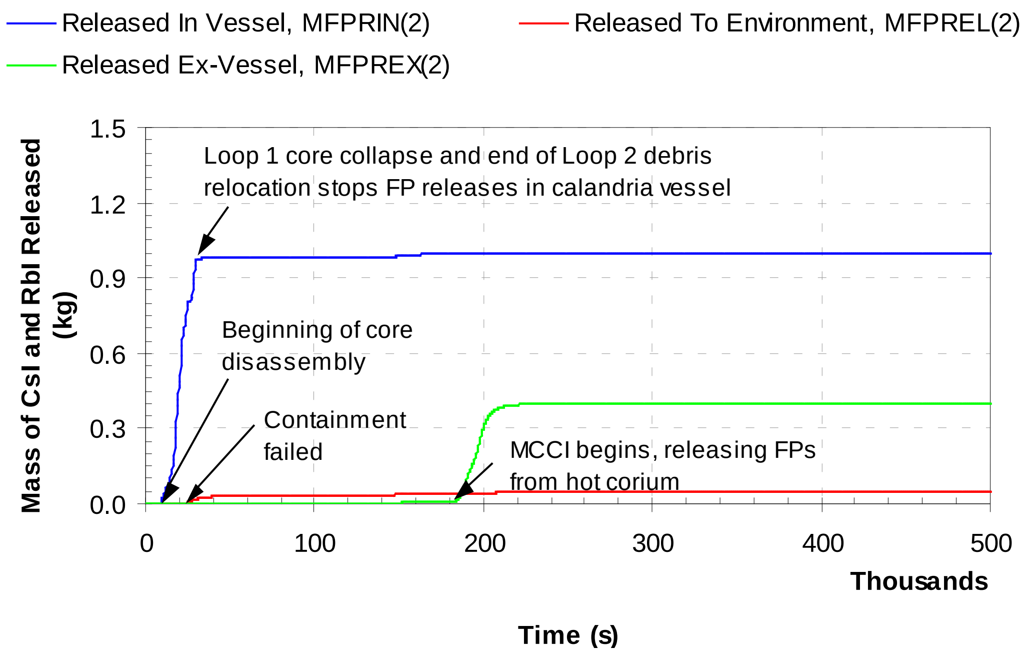

Figure 13 Mass and location of released CsI and RbI, Case A (0 - 500,000 s)