SEVERE ACCIDENT ANALYSIS OF SHUTDOWN STATE ACCIDENT USING MAAP4-CANDU TO SUPPORT LEVEL 2 PSA FOR THE POINT LEPREAU STATION REFURBISHMENT PROJECT

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Abstract

A Level 2 Probabilistic Safety Assessment was performed for the Point Lepreau Generating Station. 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 the reference Shutdown State Accident scenario are discussed in this 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 [1]. An overview of PLGS operations and refurbishment activities was given in [2]. A Level 2 PSA quantified challenges to containment integrity and the location, as well as species, of fission product (FP) releases to the environment. Five representative severe core damage accidents were selected for the Level 2 PSA performed for the Point Lepreau Refurbishment Project [1]:

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

Analysis results for the reference SSA scenario are discussed in this paper. The MAAP4-CANDU (Modular Accident Analysis Program for CANDU) code v4.0.5A+ was used to estimate:

- The timing of the accident 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),
- FP transport and retention within containment,
- The timing and duration of challenges to containment integrity,
- The magnitude and nature of FP releases from containment to the environment, and
- The effect of operator actions in mitigating severe accident consequences (reducing challenges to containment integrity and reducing FP releases from the reactor building).

In the analysis of the SSA sequences, reference Case A assumed no operator interventions and credited only a limited number of safety-related systems; other systems were assumed to be unavailable. Case A was followed by a series of sensitivity cases assuming certain system availabilities, to assess their effects on the accident sequence. A total of seven cases were analyzed, but the analysis results (major event timing and FP releases to the environment) of only reference Case A are discussed in this paper.

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2. Brief Description of the MAAP4-CANDU Code

The MAAP4-CANDU [3] 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 by Fauske and Associates Inc (FAI). MAAP4 is owned by the Electric Power Research Institute, and is used for severe accident analysis in light water reactors. Ontario Power Generation Inc. (OPG) is the MAAP-CANDU code licensee (code holder), and AECL holds a sub-license from OPG. A brief description of the MAAP4-CANDU code, including major features, capabilities and limitations, can be found in [3].

3. Nodalization of the Point Lepreau Generating Station

MAAP4-CANDU simulates the most significant systems and components 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 PLGS, are described in this section. A more detailed description of the CANDU 6 station nodalization is reported in Reference [1].

The PLGS containment building was represented by 13 volumetric nodes, connected by 31 flow junctions. Containment concrete walls, floors, ceilings and structural steel were represented in the MAAP4-CANDU containment model by 94 heat sinks.

The Primary Heat Transport System (PHTS) was modelled as two symmetric loops. Fourteen nodes in each PHTS loop represented the pump discharge lines; reactor inlet headers (RIH); reactor outlet headers (ROH); the inlet piping of two steam generators (SGs); the hot leg tubes of two SGs; the cold leg tubes of two SGs; and the pump suction lines connected to the cold legs of each SG.

The PLGS reactor core has 380 fuel channels. They 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 characteristic channel were modelled as 12 axial channel nodes, and the calandria tube (CT), pressure tube (PT) and fuel bundle as nine concentric rings.

4. Failure Criteria

A brief description of the failure criteria used in the present analysis is given in the following sections. A more detailed description of the channel failure criteria is presented in [1] and [3].

4.1 Containment failure criteria

In this study, MAAP4-CANDU used a simple failure junction to model each of two airlocks (equipment and personnel) linking two different nodes in containment (the reactor building) with the environment (service building) [1]. The "environment" represents anything outside the containment boundary.

If the pressure difference, between containment node containing an airlock and the environment, exceeded 234.4 kPa (d), a flow path formed. The opening areas (0.0146 m^2 and 0.0039 m^2) represented the openings anticipated for airlock seal blowouts. The areas and pressure difference were based on small-scale experiments of airlock seals. The seals on both the inner and outer airlock doors were assumed to fail simultaneously, with no time delay due to the airlock volume between the doors.

4.2 Calandria vessel failure criteria

Several criteria are used in MAAP4-CANDU to determine when the calandria vessel (CV) fails. MAAP4-CANDU checks the predicted CV conditions at every time step, to identify if any of the CV failure criteria is met. During each simulation performed for this study, the CV was predicted to fail based on a user input criterion. This criterion assumed CV failure occurred when the calandria vault water level decreased to the level of the top of the terminal debris bed located within the CV, because of insufficient external cooling in that region of the cylindrical CV shell. The CV was conservatively assumed to fail at the bottom, rapidly pouring all the terminal debris into the calandria vault (no debris was assumed to remain within the CV).

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 criteria are implemented in the MAAP4-CANDU code, depending on the PHTS pressure.

At high PHTS pressures, the channel is assumed to fail when a PT and CT creep (i.e., ballooning) criterion is satisfied. When this happens, no fuel channel debris is formed, but the PHTS depressurizes into the CV. At low PHTS pressures, the fuel channels may fail and form debris, due to local melt-through (Section 4.4). Channel sagging may perforate the CT, allowing CV steam to penetrate into the fuel channel annulus and increasing the channel heat up rate due to the exothermic Zr-steam reaction.

4.4 Fuel channel disassembly criteria

Disassembly is the separation of fuel and channel materials from the original channels. 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. These underlying channels are immersed in the remaining moderator, and are cool and strong enough to support the suspended debris until a core collapse criterion is met (Section 4.5). The suspended core debris is tracked by separate components until it moves to the CV terminal debris bed.

4.5 Core collapse criterion

In MAAP4-CANDU, the suspended core debris rapidly relocates (collapses) to the bottom of the CV when the total suspended debris mass exceeds a user-specified value. A mass 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 (1000 K in this analysis).

Fission products from the fuel matrix are modelled to be released based on fractional release models driven by the fuel temperature. MAAP4-CANDU v4.0.5A+ assumes that no FPs 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. Also, 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 used in this study are presented in Table 1 of Reference [1]. In these analyses, the inventories of only the radioactive FPs were input to MAAP4-CANDU.

6. Accident Description and Analysis Assumptions

6.1 Brief description of the accident

The reactor was assumed to have been shut down for six hours prior to the beginning of the accident sequence. The PHTS was cool, full of primary coolant and depressurized. The SSA initiating event was a leak from the bearing seal of one of the shutdown cooling system (SDCS) pumps, which caused a coolant loss from both PHTS loops. A simplified flow sheet of the PHTS and SCDS is shown in Figure 1, showing the SDCS connections to both PHTS loops.

The SCDS pump connections to the PHTS, and their elevations, are shown in Figure 2. The pump leak was assumed to occur at time = 0 s at the SDCS pump elevation of 21.57 m. MAAP4-CANDU is not able to directly model the SDCS piping and the pump, nor air being drawn into the PHTS counter-current to the water leaking out. Instead, the leak was modelled to occur at the junction between: i) the PHTS pump discharge line to the reactor inlet header, and ii) the SDCS piping. The junction is at an elevation of 27.01 m (Figure 2).

One minor problem encountered with this simple model was that, during the initial period; the PHTS was predicted by MAAP4-CANDU to drain down and depressurize to sub-atmospheric conditions, which is unlikely to happen in the actual accident progression. This occurred before the decay heat (no longer removed by the SDCS, which was assumed unavailable at time = 0 s) heated and pressurized the PHTS. In reality, a leak in the pump would have allowed air back into the PHTS as the accident progressed, and the PHTS pressure would not have been sub-atmospheric. The pressure at the leak location was initially higher than atmospheric, and gradually decreased as the head of water above the leak and the pressure of the gas above the water both decreased. However, the MAAP4-CANDU PHTS thermal-hydraulics models do not model the air penetration into the PHTS via the break. The impact of this MAAP4-CANDU deficiency on the overall analysis result is expected to be small.

The PHTS eventually drained almost down to the reactor header level because of the coolant leakage. 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 loss of emergency core cooling (ECC).

The moderator heated up and gradually boiled off, due to heat transferred from the fuel channels and later from debris. The fuel channels voided and the fuel heated up, and eventually the fuel channels disassembled. Severe core damage progressed in both loops, because the failed SDCS was connected to both loops, and eventually fuel channel debris collapsed to the bottom of the CV. The moderator boiled off, so the decay heat was then transferred to the calandria vault water, which also boiled off.

As the vault water level decreased, the CV cylindrical wall heated up and eventually failed. Debris from the CV relocated to the calandria vault, where it was quenched in the remaining vault water. Once the calandria vault water evaporated, the corium heated up and molten corium-concrete interaction began on the vault floor.

The simulations were run up to 500,000 s (~139 h or 5.8 days). The timing and duration of various events/processes varied depending on the analysis assumptions in a particular simulation. For example, the availability of low and medium pressure ECC, in Cases D and D1, prevented core damage from occurring. Analysis assumptions are given in Section 6.2.

6.2 Analysis Assumptions

Other analysis assumptions were:

a) The initiating event of the shutdown state accident was a leak (at t = 0 s, six hours after reactor trip) from the bearing of one SDCS pump (Figure 2), elevation 21.57 m. The leak resulted in the PHTS coolant in both loops eventually draining down to about the reactor header level.

b) The AC (Class III and IV) and DC power supplies: available.

Currently MAAP4-CANDU does not have an explicit model of the SDCS pump. Therefore, a PHTS break was modeled at the primary pump discharge line in each PHTS loop, because both loops feed one SDCS pump. The size of the opening in each of the PHTS loops was assumed to be 1.48×10^{-3} m^2, defined so that the total of the initial break flows would match the flow rate identified by PSA analysis (16 kg/s). The elevation of the opening was assumed to be at the junction between the primary pump discharge line and the line from the shutdown cooling system heat exchanger, at an elevation of 27.01 m (Figure 2). This is 11.09 m above the inside bottom of the calandria vessel.

c) When the accident began, the PHTS pressure was assumed to be 307 kPa (a), and the PHTS coolant temperature was 42°C (315 K). The core decay heat at t = 0 s was 18.71 MW (th).

d) Moderator, shield and shutdown cooling became unavailable immediately after the accident initiation (i.e., the pumps and heat exchangers were not credited). These systems were available prior to the accident initiation, i.e., during the reactor cool-down period in the shutdown state.

e) The main steam generator (SG) feed water was unavailable after the accident initiation.

f) Containment leakage was modelled by a special junction, which remained open to the environment. The junction was sized to allow 2.5% of the containment volume to leak out per 24 hours, given a pressure difference of 124 kPa.

g) The pressurizer: isolated from the PHTS at the beginning of the accident.

h) 19 AECL passive autocatalytic recombiners (PARs): credited.

i) The moderator cover gas system bleed valves 3231-PV1 and PV-2 (connecting the CV and vapour recovery system) were operable. These valves allowed some of the moderator to leave the CV.

j) The SG main steam safety valves: assumed open at the beginning of the accident and stayed locked open for the whole accident simulation (to 500,000 s). The SGs were cool at t = 0 s, with 315 K secondary side water.

k) The containment dousing system: available.

l) The high-pressure ECC phase: unavailable.

m) The D_2O supply system make-up to the PHTS: unavailable.

n) Operator interventions: not credited.

In addition to the above assumptions, which were applied to all the simulation cases, there were several system availability variations for different cases:

i. Auxiliary SG feed water: available (Case B) or unavailable (all other cases).

ii. Local air coolers: available (Case A2) or unavailable (all other cases).

iii. Containment failure: containment failure pressure was set very high to avoid containment failure in Case A2-No LACs.

iv. D_2O recovery system: available (Case C) or unavailable (all other cases).

v. Low-pressure ECC phase: available (Cases D and D1) or unavailable (all other cases).

The ECC heat exchanger was not credited, except in Case D1. An operator initiated all ECC injections.

vi. Medium pressure ECC phase: available (Cases D, D1) or unavailable (all other cases).

7. Major Analysis Results

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

The reactor was shut down six hours (t = -21,600 s) prior to the accident initiating event, and the PHTS was cool, full, and depressurized. The initiating event (t = 0 s) was a leak from the bearing seal of one shutdown cooling system pump, resulting in leak of the coolant from both PHTS loops. As a result, the PHTS eventually drained almost to the reactor header level. The mass of PHTS coolant in Loops 1 and 2 is shown in Figure 3; it decreased to ~5,000 kg (~8% of initial inventory) in each loop by ~30,000 s. Initially the PHTS pressure was 307 kPa (a) (Figure 4). Because the coolant loss from the break was slow and at a relatively low elevation, the PHTS pressure initially decreased. As mentioned earlier, the

MAAP4-CANDU PHTS thermal-hydraulic model does not allow counter-current air-water flow in the PHTS break. Therefore, after some coolant was lost, a void developed in the PHTS, which resulted in sub-atmospheric pressure for a short initial period of the accident (Figure 2). Because of: i) the decay heat from the core and, and ii) the unavailability of the shield and shutdown cooling systems, the PHTS coolant heated up and boiled. Due to the steam generation, the PHTS pressure increased and exceeded 100 kPa (a) at 4,850 s (~1.3 h, Figure 3).

Until 4,943 s (1.4 h), the PHTS coolant was homogeneous; the water and steam phases were not separated and natural circulation was modelled in the PHTS. When the global PHTS void reached 50%, at 4,943 s (~1.4 h) in Loop 1 and 4,988 s in Loop 2, the water and steam phases were assumed to have separated. Following phase separation, heat was transferred from the primary coolant to the secondary side SG water by: i) condensation on the inside of the SG tubes, ii) conduction through the SG tube walls, and iii) heatup and boiling of the secondary side water on the outside of the SG tubes. The SG secondary side steam vented through the MSSVs, which were locked open to atmosphere

MAAP4-CANDU reflux condensation model was active only while the PHTS coolant level remained above the RIH/ROH centerline (a user-defined condition). Reflux steam condensation is a very efficient heat transfer mode, so by switching off this process the heat transferred from the primary to the SG secondary side was significantly reduced (i.e., the SG secondary sides boiled dry only at the very end of this simulation, Figure 5). After the coolant level dropped below the RIH/ROH centerline at 7,340 s, the heat transfer was by natural convection of steam on the primary side and water boiling on the SG secondary side. This was essentially a loss of the SG heat sinks; the PHTS coolant thus boiled and steam production in the PHTS exceeded the break (leak) flow rate, on a volume basis.

This high steaming rate caused the PHTS pressure to increase rapidly in both loops, reaching ~0.9 MPa (a) at 19,300 s (~5.4 h, Table 1 and Figure 4). The pressure then began to decline, because the steaming rate decreased below the break flow rate, as the water levels in the channels decreased (i.e., the steam generation rate in a channel is modelled to be proportional to the depth of the water).

The core decay heat continued to boil off the PHTS coolant. Eventually, some fuel channels became dry on the inside and heated up (Figure 6). Heat was transferred to the moderator, through the pressure and calandria tubes, which heated up and started to boil at 33,780 s (~9.4 h, Table 1). When the moderator level dropped below the topmost fuel channels, those channels lost their final heat sink. Thus the fuel, PT and CT, of dry (inside) and uncovered (outside) fuel channels, heated up further (Figure 6) and disassembled as the pressure and calandria tubes melted through. Fuel channel disassembly started at 47,450 s (Loop 2) and 47,496 s (Loop 1) (~13.2 h, Figure 7).

7.2 Case A – Fuel and fuel channel response

The first fuel channel to be declared dry (i.e., less than 1 kg of water left in the channel) was characteristic Channel 4 in Loop 1 at 29,281 s (8.1 h); Channel 4 was also the first fuel channel to dry out in Loop 2, at 29,700 s (8.3 h, Table 1). Channel dry-out was the result of: i) the loss of PHTS coolant through the leak in the shutdown cooling pump seal, and ii) coolant boil-off from the core decay heat. In MAAP4-CANDU, only dry channels are modelled to heat up. The temperatures of the pressure and calandria tubes, as well as fuel ring #1 (i.e., the centre element) for bundle #7 of Channel 2, located in the top core node in PHTS Loop 1, are shown in Figure 6.

When the pressure and calandria tubes of an axial channel node reached disassembly conditions, the fuel and fuel channel section relocated into “holding bins” as described in Section 4.4, and stayed there temporarily as suspended debris. The core material in the suspended debris bed heated up further, due to decay heat and the exothermic Zr-steam oxidation reaction.

Core collapse (Section 4.5) was not predicted in Case A. Instead, a slower process of molten debris and solid debris relocation started at 47,450 s (13.2 h). The dynamics of channel disassembly and relocation phenomena are represented in Figure 7. The mass of UO_2, in the intact core material in

PHTS Loop 2, started to decrease at 47,450 s (13.2 h), with the disassembly of the first axial channel node. This was followed by the first disassembly in Loop 1 at 47,496 s. The core disassembly process continued until ~83,800 s (~23.3 h) as shown in Figure 7. Some remnants (peripheral stubs) of low-power channels did not reach disassembly conditions, and stayed attached to the tube sheets until after the calandria vessel failed.

7.3 Case A – Calandria vessel response

Figure 8 shows that the moderator mass was initially nearly constant, and then decreased slowly after the CV bleed valve opened at 19,618 s (5.4 h, Table 1). At ~35,000 s (~9.8 h) the moderator mass started to decrease more rapidly as the moderator reached the saturation temperature and began to boil-off. The CV relief duct rupture disk burst at 43,585 s (12.1 h), and approximately 40,000 kg of moderator was lost due to two-phase swell and venting. The moderator mass decreased steadily as submerged channels continued to transfer heat, and as core debris formed and then relocated to the terminal debris bed, where it was quenched in the remaining moderator.

Figure 9 shows the total corium mass (particulates, corium crust, and molten corium), the mass of the corium crusts (bottom, side and top surfaces) and the mass of particulates in the CV. Particulates were solid core debris formed during fragmentation of the molten core material (pouring down from the suspended debris) in the remaining moderator. The typical calculated size of particulates was 5-10 mm. A crust began to form on the CV walls as the core material finished relocating to the CV bottom. The crust thickness on the CV walls was in the range 5 to 10 cm. 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 ~130,000 s (~35 h, Figure 9).

The calandria vault water level decreased to the elevation of the top of the CV debris bed at ~238,793 s (66.3 h), so the calandria vessel wall failed at the bottom of the vessel (See 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. An energetic interaction of corium and calandria vault water was predicted, when the CV cylindrical wall failed.

7.4 Case A – Calandria vault and end shields response

The pressure and water level 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, which can relieve over-pressure via rupture disks. At 35,240 s (9.8 h), these rupture disks 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 90,760 s (25.2 h), which resulted in a gradual water level decrease.

As described in Section 7.3, the calandria vessel failed at 238,800 s (66.3 h). As a result, all the core debris was modelled to pour out onto the calandria vault floor (Figure 9). The calandria vault water continued to boil off, at a faster rate than previously due to the water contacting a larger area of core debris, and at 265,721 s (73.8 h) the last of the calandria vault water boiled off.

After the loss of the calandria vault water heat sink the corium temperature increased due to the decay heat, and corium-concrete interaction began. The eroded depth of the calandria vault concrete floor was ~1.56 m at the end of the simulation at 500,000 s (~139 hours or ~5.8 days). When the eroded depth of the concrete reached 2 m, the calandria vault would have been considered failed. Under this assumption the calandria vault had not failed by the end of the simulation, although without the addition of water to the calandria vault the floor would have failed within a few more hours.

7.5 Case A – Containment response

Figure 10 shows the pressure in the lower half of the steam generator enclosure. Initially, the containment pressure was about 100 kPa (a); it started to increase during the initial period of the accident, due to steam generation from PHTS coolant discharge through the SDCS pump seal leak. At ~25,000 s (~7 h), containment pressure started to decrease because of: i) diminished steaming from the PHTS, ii) condensation on the containment walls, and iii) the assumed leakage from containment.

At 35,300 s (9.8 h), the containment pressure began to increase again as the moderator started to boil (Table 1). The containment pressure continued to increase as fuel channels started to disassemble at 47,450 s (13.2 h), and the resulting core debris was quenched in the moderator. The subsequent coolant/steam discharge, into containment through the open CV relief duct, caused an increase in containment pressure until ~77,800 s (~21.6 h), by which time most of the core debris had moved to the CV terminal bed and the CV was almost dry. The pressure then started to decrease because of the diminished steaming from the CV and the continued condensation on the containment walls. The containment pressure had almost reached the airlock seal failure pressure of 334.4 kPa (a) (Section 4.1), when the CV steaming ended.

The renewed increase in containment pressure at ~91,000 s (~25.3 h) was caused by the increased steaming from the calandria vault, as its water reached saturation and started to boil at 90,758 s (25.2 h). This pressurization blew out the door seals on both airlocks at 135,203 s (37.6 h); the containment pressure then decreased, as the steam/gas mixture vented to the environment.

The containment pressure continued to decrease until 238,793 s (66.3 h), when it began to increase sharply despite the blown out airlock seals. At that time the CV failed, and the hot core debris was discharged into the calandria vault. The debris was quenched by the remaining calandria vault water, and the resulting steam flow into containment exceeded the combination of the loss of steam through the blown out airlock seals and containment leakage, and the condensation within containment. After the core debris quenched, the steaming rate decreased and the containment pressure decreased (Figure 10). At ~265,721 s (73.8 h), the calandria vault water was depleted so the steaming stopped. The blown out airlock seals then permitted the complete depressurization of the containment.

7.6 Case A – Fission product release and distribution

After 29,300 s (~8.1 h), when the fuel channels became dry and the fuel element temperatures exceeded 1000 K, fission products (FPs) were released from fuel elements according to the criterion described in Section 4.6. The initial inventory of the noble gases (Kr and Xe) in the core was 1.04 kg (radioactive isotopes only). The major portion of the noble gases was released into the CV from the fuel and the suspended debris bed during core disassembly, from ~13 h until ~25 h. Noble gases released in the CV escaped to the containment through the burst rupture disks. From containment, some noble gases escaped to environment (Figure 11) via the containment leakage (see Table 1 of [1] for the containment leakage details). Eventually all noble gases were released to the environment, after the containment failed at 135,203 s (37.6 h).

Figure 12 shows the mass of radioactive CsI and RbI released in-vessel (inside the CV) and to the environment. The initial inventory of CsI and RbI in the core (radioactive isotopes) was 17.1 kg. Fission products were released to the CV from the PHTS and from the suspended debris bed. When the CV rupture disks burst open at ~43,600 s (~12.1 h), a flow path was established for the FPs to be transported from the CV into containment. After the calandria vault water was depleted at ~265,721 s (~73.8 h), corium reacted with the concrete starting at 280,667 s (78 h) and FPs were released ex-vessel (outside the CV) (Figure 12).

At the end of the simulation at 500,000 s (5.8 days), the total mass of Cs, Rb and I (radioactive isotopes) released to the environment, in the form of CsI, RbI and CsOH and RbOH, was 0.093 kg or 0.55% of the initial inventory of these isotopes in the core. This release was relatively small because containment failure was predicted to occur late (at 37.6 h). This resulted in significant deposition of the volatile FPs on containment walls and floors, and dissolution in water pools.

7.7 Case A – Hydrogen release

Hydrogen was released in hot and dry fuel channels, and in the suspended debris beds of both PHTS loops. At ~29,280 s (~8.1 h), some fuel channels dried out in Loop 1 and at ~29,700 s (~8.3 h) fuel channels dried out in Loop 2. Steam inside the channel reacted with the fuel cladding and inner surface of the pressure tube. This exothermic reaction produces hydrogen, and the reaction rate increases at higher temperatures. The hydrogen production increased after core disassembly started at ~47,500 s (~13.2 h) in PHTS Loops 1 and 2. At ~77,570 s (21.5 h) the CV was dry, so hydrogen production was significantly reduced. A very small amount of hydrogen was produced from the debris located at the CV bottom, after the core finished relocating there at ~23 h. The solid crust formation, on top of the debris bed, severely limited steam access to the Zr debris.

The total amount of hydrogen produced during the core heat up and disassembly was 347 kg, which was produced inside of the PHTS loops and from the debris suspended inside the CV. This corresponds to ~19.4% of the initial Zr inventory in the core.

No hydrogen or carbon monoxide burning in containment was predicted in Case A.

7.8 Sensitivity cases

As mentioned, seven SSA cases (including reference Case A) were analyzed for the PSA Level 2 activities for PLGS Refurbishment Project. Sensitivity cases involved crediting various plant safety-related features/operator actions. The results of the sensitivity cases are beyond the scope of this paper and will be published separately.

8. Summary

A series of seven simulations were run to assess the consequences of a severe accident, due to a leak in the shutdown cooling pump seal. This was done for a Level 2 probabilistic safety assessment, for the PLGS refurbishment. The accident began with a cool and depressurized PHTS, six hours after the reactor was shut down. The simulations were run with the severe accident analysis code MAAP4-CANDU v4.0.5A+, using the PLGS specific parameter file. This paper discusses major results of SSA reference Case A, which assumed most safety-related plant systems were not available:

Unavailable: Main and auxiliary SG feedwater, emergency water supply, moderator cooling, shield cooling, shutdown cooling, and high pressure emergency core cooling;

Available: Reactor shutdown, dousing spray, class III and class IV power, AECL PARs, SG main steam safety valves (locked open).

In reference Case A, severe core damage began at 13 hours, the moderator was depleted (boiled off) in the CV at 21.5 h, containment failed at 38 h, and the CV failed at 66 h. The total release of radioactive isotopes of Cs, Rb and I was 0.55% 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 predicted in the containment.

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[3] P.M. Mathew, S.M. Petoukhov and M.J. Brown, "An Overview of MAAP4-CANDU Code", Proceedings of the 28th Annual Conference of the Canadian Nuclear Society, Saint John, New Brunswick, Canada, 2007 June 3-6.

Table 1
Sequence of significant events for shutdown state accident scenario reference Case A

Time (h)	Time (s)	Event in Reference Case A (SSA-PLR4H1N)
-6.0	-21,600	Reactor shutdown
0.0	0	Seals in one shutdown cooling system pump, which is connected to both PHTS loops, begin to leak
0.0	0	Turbine main stop valves (MSIVs) closed because MAAP4-CANDU does not model them
0.0	0	Moderator and shield cooling systems off
0.0	0	Main and auxiliary feed water off
0.0	0.1	Main steam safety valves in the SGs are locked open to the atmosphere
0.0	1	Pressurizer and PHTS loops are isolated
1.4	4,943	Water and steam phases are separated in Loop 1 (PHTS coolant stagnant)
1.4	4,988	Water and steam phases are separated in Loop 2 (PHTS coolant stagnant)
2.3	8,119	Containment dousing system on
2.4	8,645	Dousing tank water is depleted for containment sprays
4.3	15,304	Fuel bundles are uncovered inside fuel channels in Loop 1
4.4	15,679	Fuel bundles are uncovered inside fuel channels in Loop 2
5.4	19,618	Calandria vessel bleed/relief valves open
8.1	29,281	At least one channel is dry Loop 1 – Channel #4
8.3	29,700	At least one channel is dry Loop 2 – Channel #4
9.4	33,780	Moderator in calandria vessel reaches saturation temperature
9.8	35,240	Calandria vault rupture disk opens to Node 8 (lower SG room); end shield water starts to boil
12.1	43,585	Calandria vessel rupture disk 1 is open to Node 8 (lower SG room)
13.2	47,450	Beginning of the core disassembly in Loop 2
13.2	47,496	Beginning of the core disassembly in Loop 1
13.2	47,491	Beginning of core relocation onto calandria vessel bottom
21.5	77,567	Water is depleted in calandria vessel

Time (h)	Time (s)	Event in Reference Case A (SSA-PLR4H1N)
23.3	83,800	End of main core relocation to the calandria vessel bottom
25.2	90,758	Calandria vault water reaches saturation temperature
37.6	135,203	Containment failed (seals in both airlocks)
66.3	238,793	Calandria vessel bottom wall failed, core debris relocates to calandria vault
73.8	265,721	Water is depleted in calandria vault
78.0	280,667	Molten corium-concrete interaction begins in calandria vault
N/A*	N/A*	Calandria vault floor failed because of the molten core-concrete interaction

* Event did not occur by the end of the simulation (500,000 s)

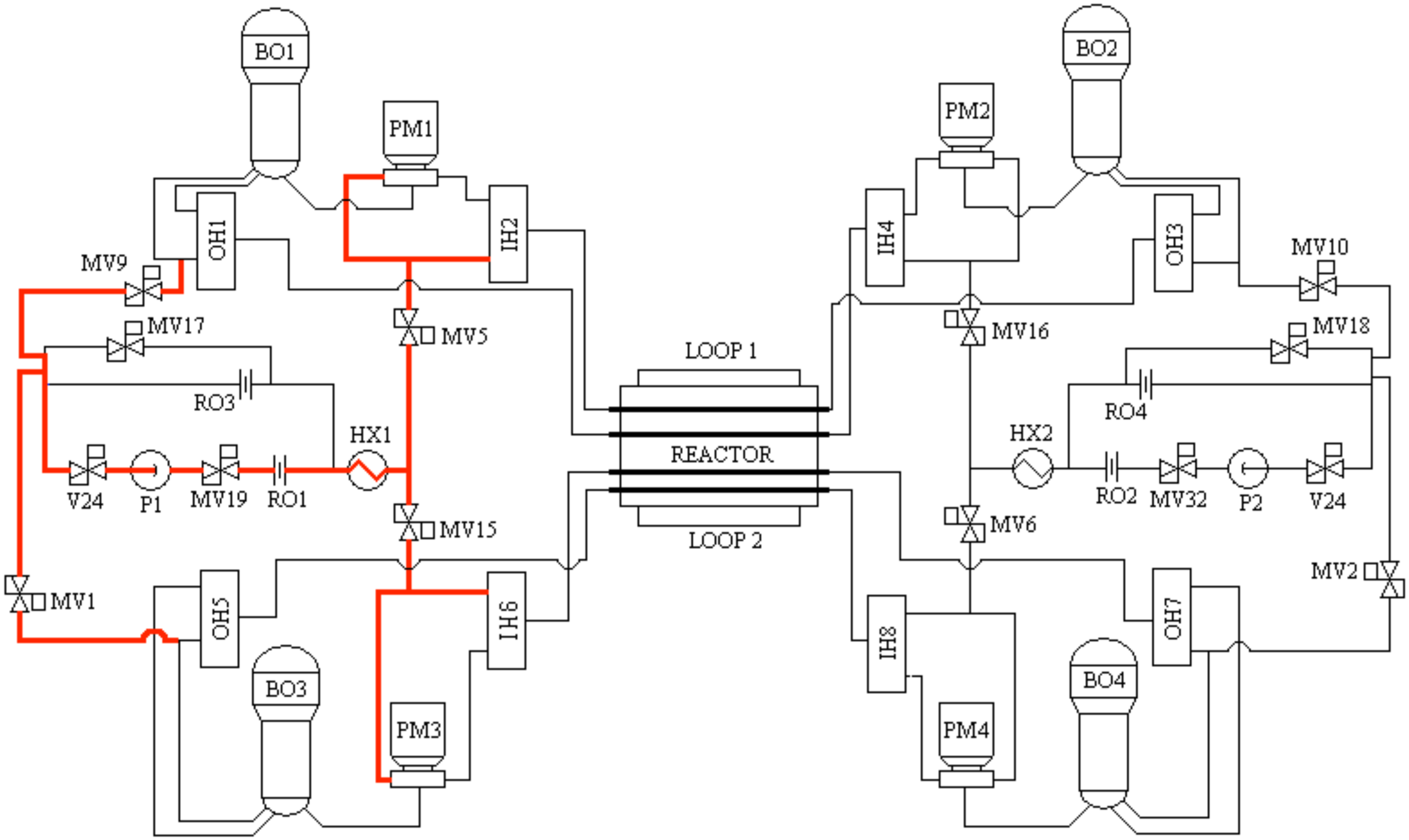

Figure 1 CANDU Shutdown Cooling System Connections (in Red)

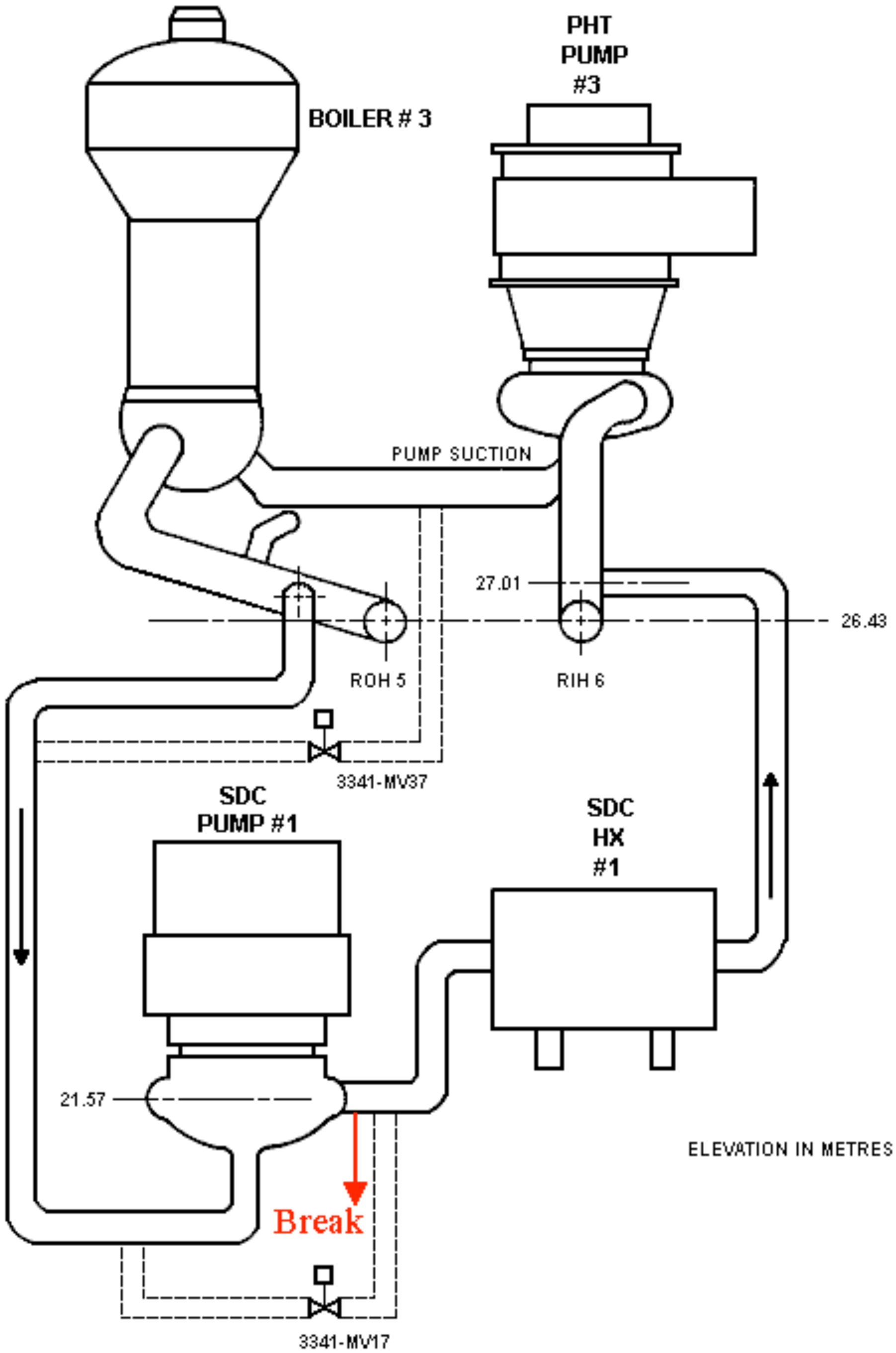

Figure 2 CANDU Shutdown Cooling System Elevations

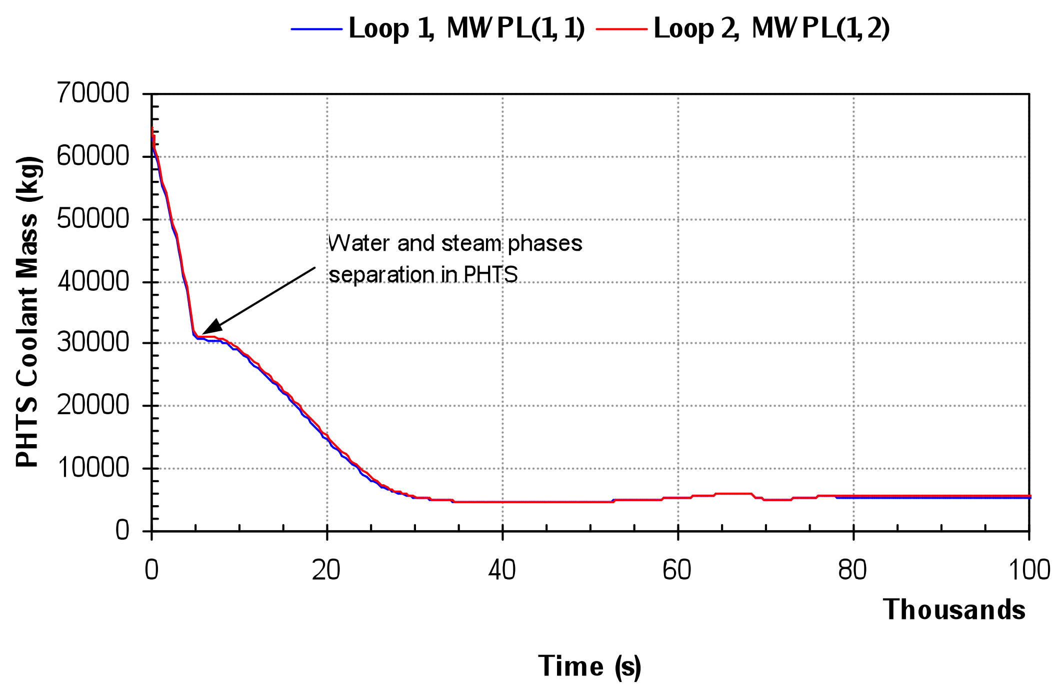

Figure 3 Primary System Water Masses in both Loops (0 - 100,000 s)

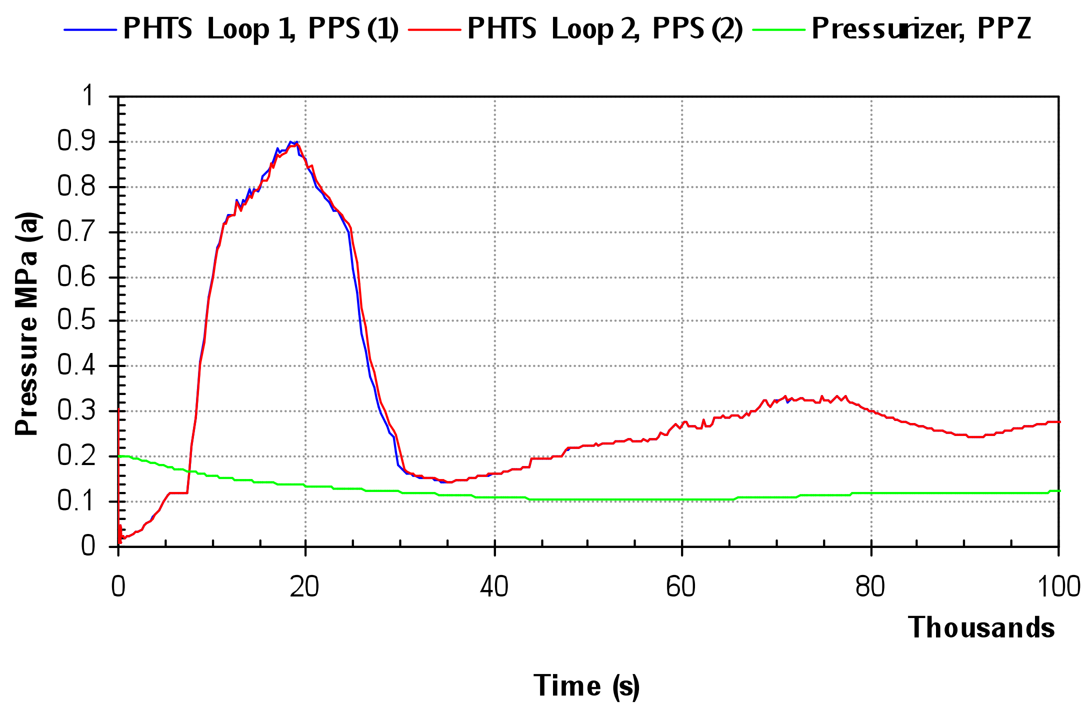

Figure 4 Primary System Pressure (0 - 100,000 s)

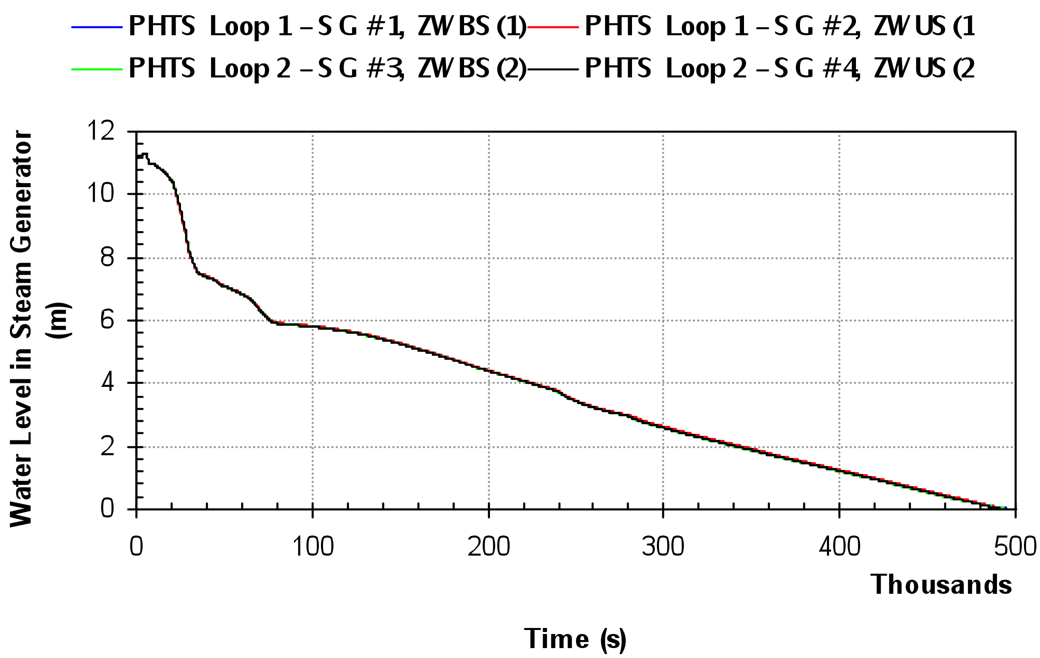

Figure 5 Steam Generator Water Level (0 - 500,000 s)

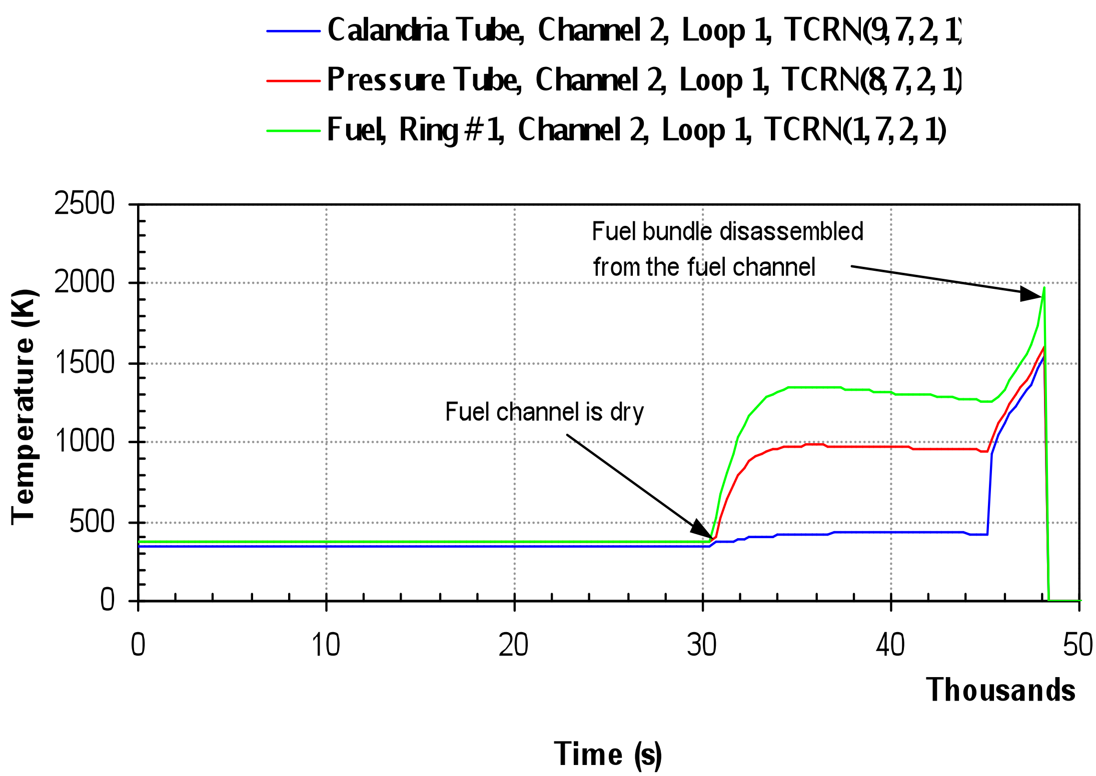

Figure 6 Fuel and Fuel Channel Temperatures: Bundle 7, Channel 2, Loop 1 (0 - 50,000 s)

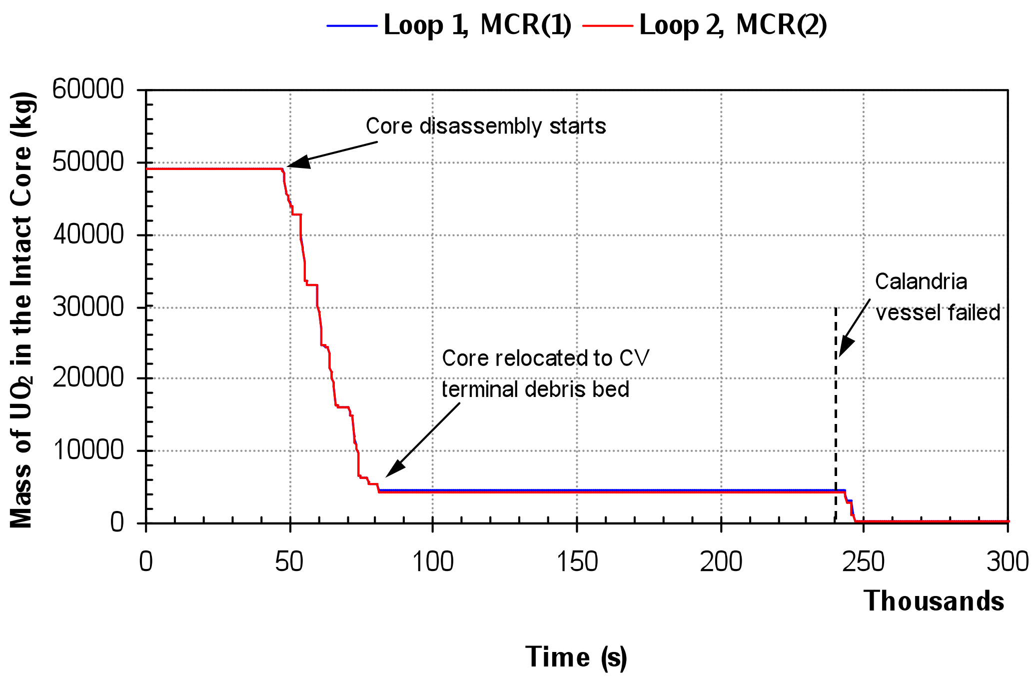

Figure 7 UO_2 Mass remaining in the Intact Fuel Channels (0 - 300,000 s)

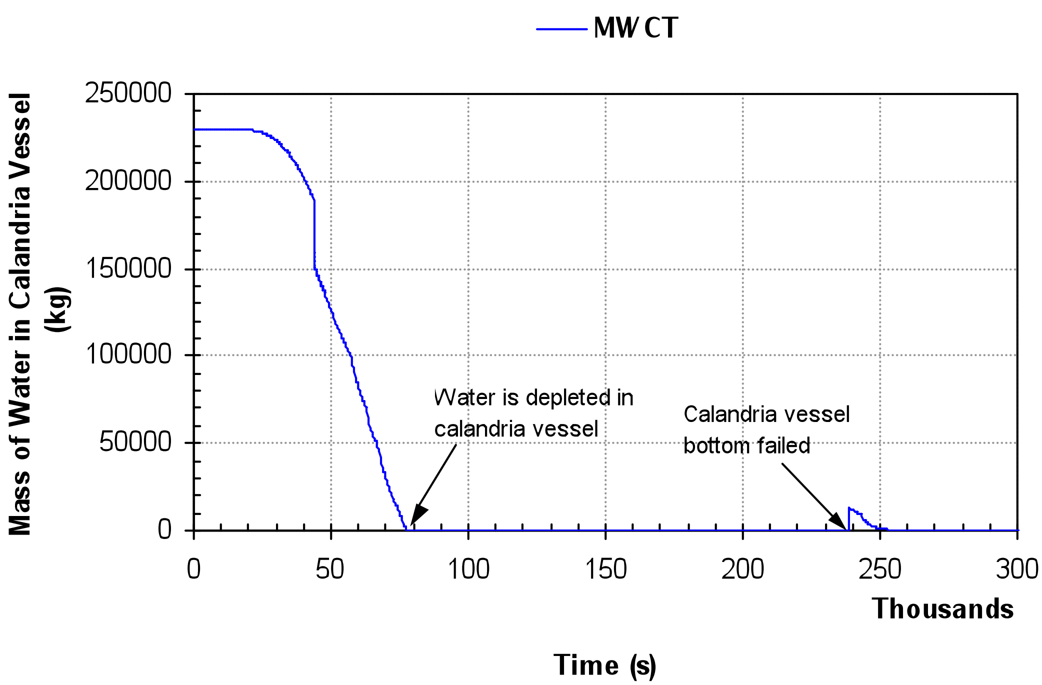

Figure 8 Calandria Vessel Water Mass (0 - 300,000 s)

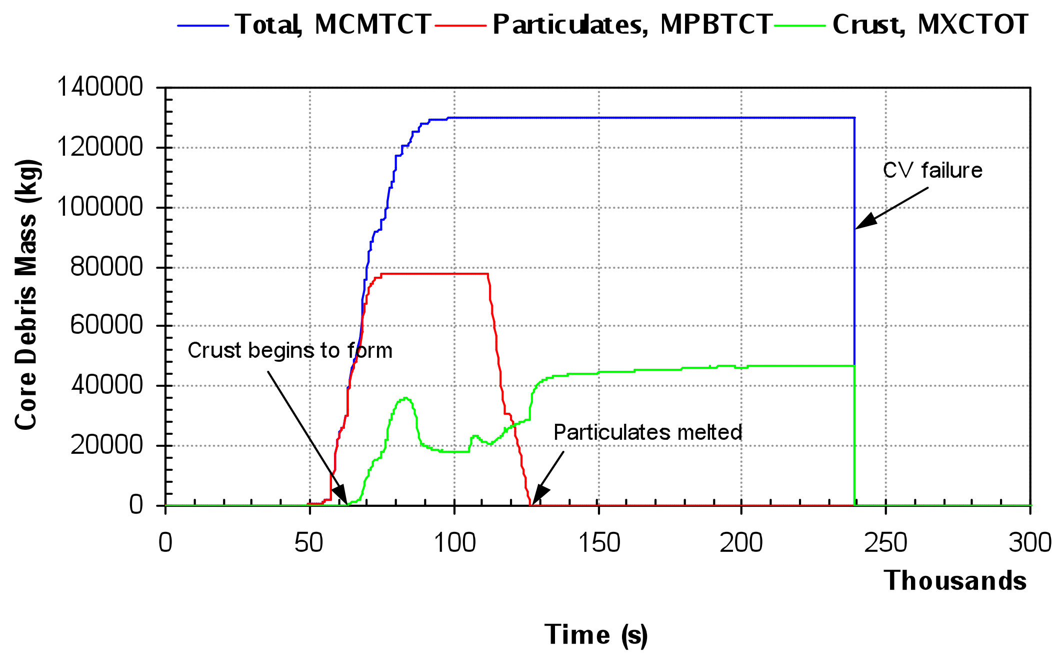

Figure 9 Mass of Corium (Crust, Particulates and Total) in the Calandria Vessel (0 - 300,000 s)

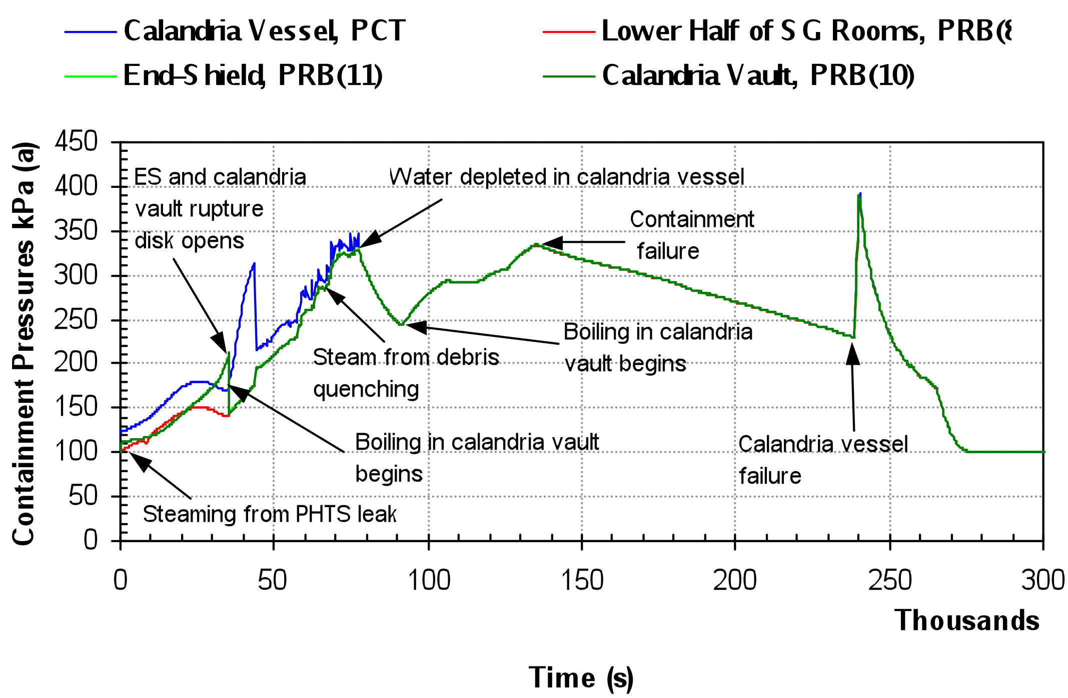

Figure 10 Containment Pressure (0 - 300,000 s)

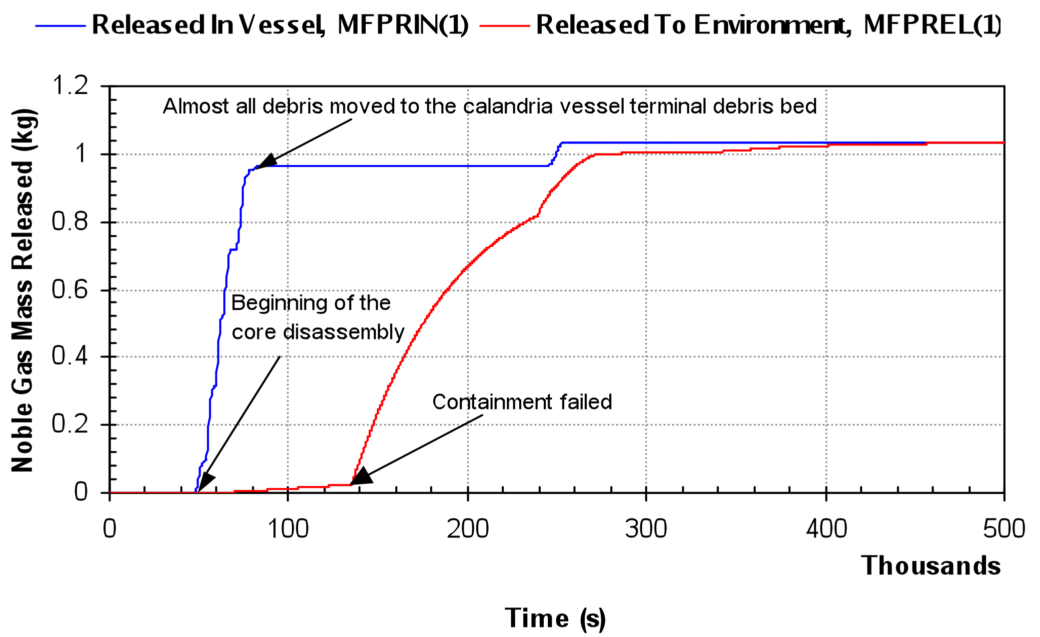

Figure 11 Mass of Noble Gases Released (0 - 500,000 s)

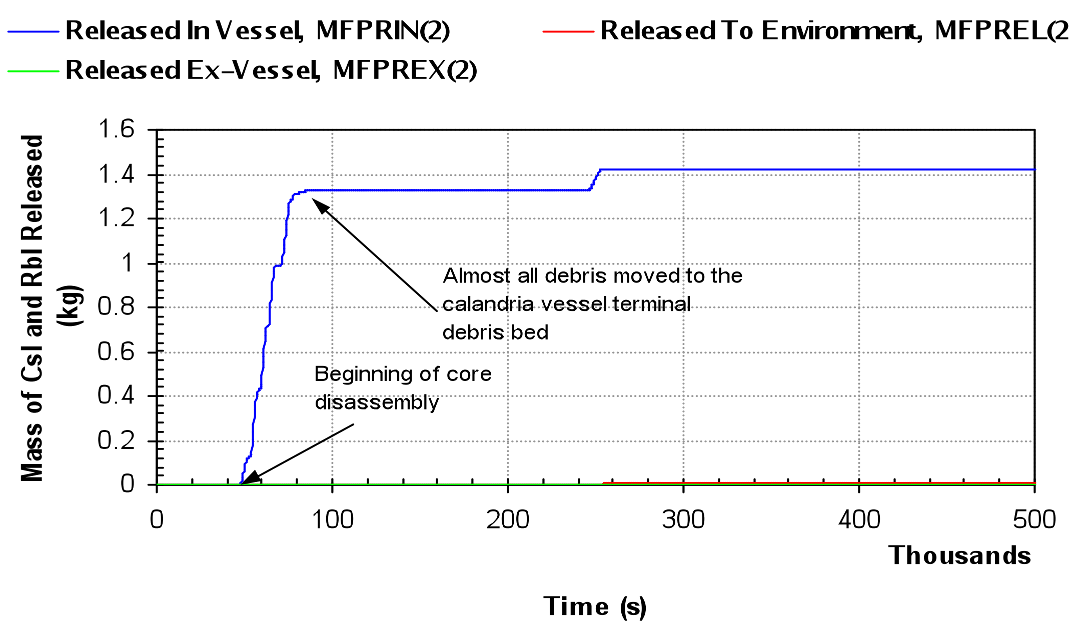

Figure 12 Mass and Location of Iodides Release (0 - 500,000 s)