SEVERE ACCIDENT ANALYSIS OF STAGNATION FEEDER BREAK SCENARIOS USING MAAP4-CANDU FOR APPLICATION TO THE 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 in this study to calculate the progression of postulated severe core damage accidents and fission product releases. Five representative severe core damage accidents were selected for this assessment: Station Blackout, Small Loss-of-Coolant Accident, Stagnation Feeder Break, Steam Generator Tube Rupture, and Shutdown State Accident. Analysis results for the reference Stagnation Feeder Break scenario are discussed in this paper. The major findings of this study are that the calandria vessel did not fail, containment did not fail, and therefore the release of fission products to the environment during this scenario is predicted to be very low.

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 Point Lepreau operations and refurbishment activities was given in [2]. The Level 2 PSA quantifies challenges to containment integrity and the location, as well as species, of fission product 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).

Each of the above initiating accidents was analyzed with a reference case and several sub-cases with different availabilities of accident mitigation systems (e.g., ECC, crash cool-down, SG feedwater). Due to space limitations, analysis results for only the reference SFB scenario are discussed in this paper.

MAAP4-CANDU (Modular Accident Analysis Program for CANDU) v4.0.5A+ was used to estimate:

- The timing of 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,

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- The timing and duration of challenges to containment integrity,
- The magnitude and nature of fission product releases from containment to the environment, 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).

In the analysis of the SFB accident sequences, the reference case assumed no operator interventions and credited only a limited number of safety-related systems (other safety-related systems were assumed to be unavailable). The reference case was followed by a series of sensitivity cases assuming certain system availabilities, to assess their effects on the accident progression. A total of about 12 cases were analyzed. Analysis results (the timing of major events and fission product releases to the environment) for the reference SFB scenario are discussed in this paper.

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, developed 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 the code limitations can be found in [3].

3. Nodalization of a 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 [1].

The Point Lepreau Station containment building was represented by 13 nodes, and 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. Adjacent rooms with large connecting openings were lumped together, as were similar heat sinks located in a room or node.

The Primary Heat Transport System (PHTS) was represented in the code by two symmetric loops. Fourteen nodes in each PHTS loop represented the PHTS components: 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 steam generators; the cold leg tubes of two steam generators; and the pump suction lines connected to the cold legs of each steam generator.

The Point Lepreau station core has 380 fuel channels. They were modelled with 36 characteristic fuel channels (2 loops × 6 vertical core nodes × 3 characteristic channels per node per loop).

The 12 fuel bundles in each fuel channel were modelled as 12 axial nodes. In a fuel channel, the calandria tube (CT), the pressure tube (PT) and the fuel were modelled 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], [3].

4.1 Containment failure criteria

In this analysis, MAAP4-CANDU uses a simple failure junction model to link a containment node with the environment [1]. If the pressure difference between the containment node and the environment exceeds the user-input value (234.4 kPa (d)), a flow path is formed. The opening area represented the cross-sectional areas of the openings formed during the blowouts of the equipment airlock and personnel air lock seals. The airlocks connect the reactor building (containment) with the service building. In this paper, the term environment represents anything outside the containment boundary.

Two failure junctions were used to represent airlock seal blowouts. The first junction opening area was 0.014624 m^2, and the second junction opening area was 0.00394 m^2. 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.

4.2 Calandria vessel failure criteria

Several criteria are used in MAAP4-CANDU to determine the conditions at which 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 considered to fail at the bottom due to the following user-input criterion: it was assumed that when the calandria vault water level decreased to the level of the top of the terminal debris bed, the CV failed because there was insufficient external cooling in that region of the cylindrical calandria shell.

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 ballooning criterion is satisfied.

At low PHTS pressures, the fuel channels may fail due to local melt-through. Channel sagging may also cause the calandria tube to perforate, allowing 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 a process where fuel and channel structural materials separate from the original channels and relocate into “holding bins” - artificial compartments used to track disassembled core material until it moves down 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

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 rapidly relocates to the bottom of the CV. A value of 25,000 kg of suspended debris per PHTS loop was used in the current analysis, based on an assumption of the pull-out strength of cooled calandria tubes.

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, a conservative criterion was used (1000 K, based on PHEBUS and BTF test results).

Fission products from the fuel matrix are modelled to be released depending on the fuel temperature, based on fractional release models. 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, following the formation of a 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 of reference [1]. Note that in these analyses, only the inventories of radioactive fission products were input to MAAP4-CANDU.

6. Brief accident description and analysis assumptions

6.1 Brief description of the initial period of the accident

The initiating event of the stagnation feeder break scenario is an inlet feeder pipe break in PHTS Loop 1 (Figure 1). The size of the break is such that the coolant flow in the fuel channel slows down and soon becomes stagnant. The opening in the inlet feeder pipe, necessary to cause the channel flow to stagnate, is fairly small (from 2 to 17 cm^2 according to CATHENA analysis). Because of the limited cooling ability of the stagnant coolant, the fuel elements and pressure tube heat up (reactor is still at full power). CATHENA predicts that the hot pressure tube balloons and ruptures within 10 – 30 s after the initiating event, depending on the location of the break on the feeder pipe and the channel location. Consequently, the calandria tube sees a rapid pressure pulse and experiences some localized hot spots from the superheated steam ejected out of the failed pressure tube. The calandria tube is expected to rupture very soon after the pressure tube ruptures.

When the pressure tube ruptures, the fuel channel bellows at both ends of the fuel channel are assumed to rupture as well. This allows PHTS coolant and moderator water to leak through the annulus between the PT and CT, and finally out the opening in the bellows. Thus the moderator can drain from the CV to the fuelling machine vault.

6.2 Analysis assumptions

MAAP4-CANDU has a simple PHTS and fuel channel thermal-hydraulics model [1], [3]. MAAP4-CANDU has no model for flow stagnation, voiding and transient critical heat flux in an individual fuel channel; therefore, MAAP4-CANDU cannot model flow stagnation in the fuel channel and fuel heating during the first 10 s after the accident initiation. Information on the thermal-hydraulic behaviour in the fuel channel during the initial period of the accident was taken from the CATHENA code thermal-hydraulic analysis to develop an appropriate MAAP4-CANDU model.

The analysis of the accident sequences uses a reference case (A1) performed without operator interventions and with a limited number of credited safety-related systems. The reference case analysis is followed by a series of sensitivity cases that assume certain system availabilities in order to assess their mitigating effects. Analysis assumptions for the reference Case A1 are:

a) No feeder pipe break opening and no discharge flow were directly modelled in this MAAP4-CANDU analysis. It was simply assumed (based on results obtained in CATHENA analysis) that the PT ruptures at time ~10 s after the accident initiation (feeder pipe break occurrence at time = 0 s). When the PT ruptures, the CT is assumed to rupture at the same time. It is also assumed that the bellows rupture at both ends of the fuel channel at the same time as the PT/CT rupture. PT/CT rupture was modeled in this study as a user-defined event. It was assumed that PT/CT rupture occurred in channel T9.

b) The PT rupture mode is a “guillotine” type of rupture; the blow-down area is assumed to be equal to the sum of cross-section areas of inlet and outlet feeder pipes of the channel T9. The total coolant discharge area was assumed to be 0.0046394 m^2.

c) It was conservatively assumed in this study that when the PT/CT ruptures, the moderator starts draining. The drain rate was assumed to be constant at 30 kg/s, which includes 25 kg/s from the feeder pipe opening and 5 kg/s from the ruptured bellows. It was assumed in the current study that the break in the feeder pipe is located at a low elevation so that moderator drains simultaneously from the bellows and from the break in the feeder pipe. Note that the moderator drain would not occur through the feeder pipe break and bellows until the broken loop of the PHTS is nearly empty and depressurized – and then only if the break is near the end fitting. Before the PHTS empties, some PHTS coolant goes directly into containment through the feeder break, as well as into the CV. This could take about 15 to 30 minutes, depending on the feeder break size required for stagnation.

d) It was assumed that the inlet feeder pipe is leaking in the high power channel #16 at the bottom core node (please see Section 3.2.2 of [3] for MAAP4-CANDU core nodalization details and Figure 3 of [3], 6th core node for channel #16 location). Flow stagnation and heating of the fuel will cause failure (pressure and calandria tube rupture) of the 16th representative fuel channel, which occurs 10 s after the initiating event.

e) The reactor was shut down at the start of the run (time = 0 s). Note that MAAP4-CANDU code has a simple thermal-hydraulics model of PHTS; therefore, the PHTS response before reactor shutdown and reactor shutdown system initiation is not expected to be as accurate as the results from more detailed thermal-hydraulic codes such as CATHENA. This assumption is acceptable since the time scale considered for this type of accident sequence is very large up to ~500,000 s (5.8 days). The exact timing of shutdown system initiation, which would range from 10 seconds to several minutes following the event, is not significant for the objectives of this work.

f) In the simulations reported using CATHENA, the stagnant fuel channel was assumed to be operating at full power, whereas in the present MAAP4-CANDU analysis no flow stagnation and no fuel heating was modeled in the channel during the first 10 s after the accident initiation. The

MAAP4-CANDU code has no model for flow stagnation in the fuel channel. Some details of the fuel channel modeling are given in the Section 3.2.1 of [3].

g) PHTS loop isolation system was credited. Isolation was complete 30 s after the loop isolation signal was generated, when the PHTS loop pressure decreased to 5.62 MPa (a).

h) In this analysis, the liquid relief valves (LRVs) were modelled to discharge PHTS coolant directly into containment. More details on LRVs and degasser condenser model can be found in [1].

i) The Moderator Cover Gas System bleed valves (connecting the CV and vapour recovery system) were operable for the first 7,200 s, then failed open at 7,200 s after depleting their air reservoirs. These valves allowed some of the moderator to leave the CV.

j) Main SG feed water was unavailable.

k) The Emergency Water Supply system was not available.

l) AECL passive autocatalytic recombiners (a total of 19 PARs) were credited.

m) The PHTS coolant was initially modelled as a single-phase liquid (i.e., the user-input initial void fraction = 0). As the void fraction increased, due to a break or boil off, the coolant was initially modelled as a homogeneous two-phase fluid. When a PHTS loop void fraction increased to a user-input value of 0.5, the PHTS coolant in that loop separated into two phases – water below and steam above – and the water was tracked in separate PHTS pools (corresponding to PHTS nodes and the fuel channels).

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

o) Emergency Core Cooling System (ECCS) low-pressure injection was not credited.

p) The steam generator main steam safety valves (MSSV) opened and closed at the set point to relieve the SG secondary side pressure.

q) No operator interventions were credited.

In addition to the above assumptions, which applied to all the SFB simulation cases (i.e., reference Case A1 and sensitivity cases), the availability of several safety-related systems depended on the analysis case:

a) SG crash cool-down: available or unavailable (available in Case A1).

b) Shield cooling system: available or unavailable (available in Case A1).

c) ECC high and medium pressure injections: available or unavailable (unavailable in Case A1).

d) Auxiliary SG feed water: available or unavailable (unavailable in Case A1).

e) Local air coolers: unavailable or available (unavailable in Case A1).

f) Containment dousing system: available, but the total capacity and maximum flow rate were different in some cases.

7. Major analysis results

7.1 Reference Case A1 results

7.1.1 Case A1 – primary heat transport system and steam generator response

The postulated initiating event for the Stagnation Feeder Break (in PHTS Loop 1) is imposed at the start of the run (time = 0 s). It was assumed that fuel in the damaged channel heats up and the pressure and calandria tubes rupture at 10 s. PHTS coolant inventory is discharged through the break in PT/CT into the CV, which causes PHTS pressure to decrease (Figure 2).

The pressure in Loop 1 (and initially in Loop 2) decreases because of: (1) the loss of fission power from the reactor core, (2) the core decay heat is transferred through the PHTS coolant to the steam generators, and (3) the secondary side of the steam generators has sufficient capacity to absorb the core decay heat as the crash cool-down system was credited. When the pressure in Loop 1 decreases to 5.62 MPa, the pressurizer and PHTS loops are isolated at ~169 s. The PHTS inventory decreases as well (Figure 3).

At ~134 s steam generator crash cool-down is initiated, and the MSSVs then remain open to depressurize the SG secondary side. The water level in the steam generators continues to decrease as a result of the secondary side water boil off (Figure 4). When the steam generator secondary side inventory is depleted for PHTS Loop 2 at ~1,609 s (~0.45 h), the steam generators are no longer a heat sink. The steam generator secondary side inventory in Loop 1 is depleted at 1,714 s (~0.5 h). As a result of the SG emptying, the pressure in the PHTS Loop 2 increases (see Figure 2) until it reaches the assumed PHTS liquid relief valve opening set point of 10.16 MPa (a) at ~4,723 s (~1.3 h).

In PHTS Loop 2, continuous loss of inventory through the LRVs and heating from the decay heat results in separation of water/steam phases at ~6,622 s (~1.8 h) and fuel bundles start to uncover inside the fuel channels at ~7,155 s (~2 h). At the same time, the moderator level in the CV decreased as a result of leakage through the ruptured inlet feeder pipe and fuel channel bellows. Fuel channels become dry in Loop 2 at ~7,920 s (~2.2 h) (Table 1), and fuel elements heat up. Subsequent heat-up of the pressure tube and calandria tube at PHTS pressure greater than 10 MPa, results in the PT ballooning and the rupture of both tubes. This fuel channel rupture causes a rapid coolant blow-down from the PHTS into the CV and Loop 2 pressure drop at ~8,126 s (~2.3 h).

7.1.2 Case A1 – Fuel and Fuel Channel Response

The first fuel channel (channel #16) is dry in Loop 1 at ~1,480 s (~0.41 h), while in Loop 2 the first fuel channel is dry at ~7,920 s (~2.2 h). Uncovering of fuel elements inside the fuel channel is the result of the following phenomena: (1) loss of PHTS coolant through the openings in the broken channel, and (2) coolant boil-off because of the decay heat from core. The temperature of the pressure and calandria tubes, as well as fuel ring #1 for bundle #7 in channel #2, which is empty of water at ~4,390 s (~1.2 h) located in the top core node for PHTS Loop 1 and Loop 2 is presented in Figure 5.

When the pressure and calandria tubes reach disassembly conditions (Section 4.4), fragments of the fuel channel relocate into “holding bins” and stay temporarily there in the form of a suspended debris bed. The core material in the suspended debris bed heats up further due to core decay heat.

When the suspended debris bed mass in the core exceeds a user-specified value (25,000 kg per PHTS loop is used in the present work, as mentioned in Section 4.5), the core material in the suspended bed and most of the intact channels of the damaged Loop 1 relocate to the bottom of the CV. This process of massive relocation of core material to the CV bottom happens at ~6,305 s (~1.8 h) for Loop 1 and at ~9,620 s (~2.7 h) for Loop 2. The dynamics of channel disassembly and relocation phenomena are represented in Figure 6. The mass of UO_2 in the intact core-material in Loop 1 decreases rapidly at

~6,000 s (~1.7 h) and at ~9,000 s (~2.5 h) for Loop 1 and Loop 2, respectively. It is the result of material transfer to the holding bins and relocation of molten material to the CV bottom.

7.1.3 Case A1 – Calandria Vessel Response

Figure 7 shows the water mass in the CV. At 10 s, the PT and CT in Loop 1 rupture, and the PHTS inventory is discharged into the CV. This discharge causes the notable increase in the CV water mass. The coolant/steam discharge into the CV from PHTS caused CV pressure to increase until the CV rupture disk #1 failed at ~2,428 s (0.67 h) (Table 1). Discharge of the coolant/steam from the CV through the rupture disk opening is associated with a rapid decrease in the CV water mass at ~2,430 s (~0.67 h), as seen in Figure 7.

The moderator inventory in CV decreases for several reasons. First, moderator drains through the broken pressure and calandria tubes and then through the break in the inlet feeder and through the openings in the fuel channel bellows. Secondly, the moderator boils off due to the heat transfer from the core. As a result of relocation of core material from the suspended debris bed into the moderator inside the CV, at ~6,585 s (~1.8 h) the moderator inside the CV is depleted as seen in Figure 7.

After the entire core collapsed at ~2.7 h, the corium remains in the CV terminal debris bed (Figure 8). A crust is formed on the CV walls soon after the core collapses onto the CV bottom; the crust thickness is in the range 5 to 10 cm. Since the shield cooling system is available, water in calandria vault cools down the CV cylindrical wall. Calculations showed that the CV did not fail up to 500,000 s (~5.8 days), when the simulation was terminated.

7.1.4 Case A1 – Calandria Vault and End Shields Response

As mentioned in Section 6.2, the reactor was shut down at the beginning of the sequence. As a result, the reactor core power decreased to decay power level.

As it was noted, the shield cooling system was credited and therefore, the CV was predicted not to fail in this sequence (Section 7.1.3). The shield cooling system has removed almost all of the decay heat generated in the terminal debris bed in the long term. The heat removal capacity of the shield cooling system is sufficient to remove the decay heat generated in the terminal debris bed. The calandria vault rupture disk was predicted to fail at ~10.6 h (~38,260 s) as a result of water thermal expansion.

7.1.5 Case A1 – Containment Response

Figure 9 shows the pressure in the lower half of steam generator enclosure. Initially, the containment pressure is ~100 kPa (a). Shortly after the accident initiation, the containment pressure increases because steam and water are discharged from the broken fuel channel into the CV and then from the CV into the containment through the open CV bleed valves and subsequently through the rupture disk, which opened at ~2,430 s (0.7 h). The containment pressure continues to increase after fuel channels in the PHTS Loop 1 begin to disassemble and finally, at ~6,300 s (~1.8 h), the entire reactor core is predicted to have collapsed to the bottom of the CV. As the hot core debris falls into the moderator, it is quenched. This quenching process provides an additional steaming source and causes containment pressure to increase during the core disassembly process, which is completed at 6,300 s (~1.8 h).

At ~8,126 s (~2.3 h) rupture of the PT and CT of the PHTS Loop 2 channel is predicted (one channel ruptures per loop as noted in Section 7.1.1), and PHTS coolant is discharged in the CV from the broken fuel channel. Steam generated during this blow-down process is released to containment through the open CV rupture disk, which causes the pressure in the containment to increase to ~225 kPa (a). Steam in the containment compartments is removed by condensation on the containment walls and metal structures such as stairs, columns, etc.; therefore, the containment pressure decreases after ~8,200 s (~2.3 h).

A slow leakage from the containment to environment was modeled in this scenario. The leakage rate was assumed to be ~2.5% of the containment volume for 24 hours.

Containment did not fail in this case because its pressure did not exceed the failure pressure limit of 234.4 kPa (d) as specified in Section 4.1.

7.1.6 Case A1 – Fission Product Release and Distribution

The initial inventory of the noble gases (Kr and Xe) in the core is 1.04 kg. This mass of noble gases includes radioactive isotopes only. Analysis shows that the major portion of the noble gases is released into the CV from the fuel and the suspended debris bed during core disassembly and core collapse, starting from ~5,300 s (~1.5 h) until ~10,000 s (~2.8 h). Noble gases released in-vessel (in CV) escape from the CV to the containment through the open rupture disks. From the containment, noble gases escape to the environment by leakage (Table 1 of [1]). The total amount of noble gases released to the environment is ~0.1 kg at the end of the simulation (500,000 s or~5.8 days), which is ~9.6% of the total noble gases inventory in the core.

Figure 10 shows the mass of CsI and RbI (only radioactive components are considered) released in-vessel (inside the CV) and to the environment. The initial inventory of CsI and RbI in the core (radioactive isotopes) is 17.1 kg. After 4,400 s (~1.2 h), when fuel channels in Loop 1 are dry and the fuel element temperatures exceed 1000 K, the fission products are released from fuel elements according to the criterion described in Section 4.6. Fission products are released to the CV from the PHTS through the broken fuel channel. Because the CV rupture disks opened much earlier at ~2,430 s (~0.7 h), a flow path was already established for the fission product release from the CV into the containment.

The fission products escape from the containment to the environment by leakage. The total mass of Cs, Rb, and I released to the environment, in the form of CsI, RbI, CsOH and RbOH, was ~0.00120 kg or 0.0071% of the initial inventory of these isotopes in the core.

7.1.7 Case A1 – Hydrogen Release

Hydrogen is released in fuel channels and in the suspended debris beds for both Loop 1 and Loop 2. Some fuel channels dry out in Loop 1 at ~4,390 s (~1.2 h), and at ~7,900 s (~2.2 h) some fuel channels dry out in Loop 2. Steam inside the channel reacts at high temperature with the fuel cladding and inner surface of the pressure tube to produce hydrogen. The hydrogen production increases after core disassembly started at ~5,306 s (~1.5 h) in PHTS Loop 1 and at ~8,870 s (~2.5 h) in PHTS Loop 2. At ~6,585 s (1.8 h) the CV is dry and hydrogen production is reduced. Only a small amount of hydrogen is produced from the debris located at the CV bottom after core collapse (which occurred at ~2.7 h). This is mainly because of the solid crust formation on top of the debris bed, which limits steam access to un-oxidized zirconium alloys.

The total amount of hydrogen produced during the accident is ~89.6 kg at the end of the simulation ~500,000 s (~5.8 days), which corresponds to ~5.0% of the initial Zircaloy inventory in the core.

No hydrogen/carbon monoxide burning was observed in containment in Case A1.

7.2 Sensitivity Cases

As mentioned, a total of 12 SFB cases (including reference Case A1) were analyzed for the PSA Level 2 activities for Point Lepreau Refurbishment Project. Sensitivity cases involve crediting of various plant safety-related features/operator actions (see Section 6.2). The results of the sensitivity cases are beyond the scope of this paper and will be published in a separate paper.

8. Summary

An analysis was performed to assess the consequences of a Stagnation Feeder Break severe core damage accident, in the Point Lepreau CANDU 6 generating station. This paper focuses on the SFB reference Case A1, which assumes that most of the safety-related plant systems were not available. This work was performed as part of the Level 2 probabilistic safety assessment, in preparation for the PLGS refurbishment project. The simulations were run with the severe accident analysis code MAAP4-CANDU, v4.0.5A+.

The analysis assumed that the pressure and calandria tubes of the affected channel ruptured in PHTS Loop 1 at 10 s after the accident initiation, in accordance with previous analysis performed by CATHENA. The following timing of events was obtained during MAAP4-CANDU simulation: the first fuel channel in PHTS Loop 1 is dry at ~0.4 h, core disassembly starts at ~1.5 h, one vertical half of the core collapses at 1.8 h, moderator is depleted (boiled off) in CV at ~1.83 h. The PHTS Loop 2 core disassembly starts at ~2.5 h, the 2nd vertical half of the core collapses at 2.7 h.

The CV remains intact mainly because the water in the reactor vault cools it externally and also since the shield cooling system was credited. Containment does not fail up to 5.9 days (or 500,000 s, when the run was terminated) after the accident initiation, mainly because the shield cooling system was credited, which prevents possible containment over-pressurization.

Fission products were released from the damaged core inside the CV and from the CV through the open relief ducts into containment. The total amount of the noble gases released to the environment is ~9.6% of the total noble gases inventory in the core. The total amount of Cs and I released to the environment is ~0.0071% of the initial inventory. The fission products escape to environment during the accident since a small containment leakage was modeled.

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10. References

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Table 1
Timing of Significant Events for Stagnation Feeder Break Scenario Case A1

Time (hr)	Time (s)	Event
0.00	0	Reactor shutdown, Moderator cooling system off, Main and Auxiliary Feed water off
0.00	10	PT and CT ruptured, PHTS Loop 1 - Channel #16
0.00	20	CV bleed valve open
0.04	134	Crash cool-down system is on
0.05	169	Pressurizer and PHTS loops are isolated
0.27	981	CV Water reaches saturation temperature
0.41	1,480	At least one channel is dry in Loop 1 – Channel #16
0.45	1,609	Steam generator secondary side is dry, Loop 2
0.48	1,714	Steam generator secondary side is dry, Loop 1
0.67	2,428	CV rupture disk #1 is open
0.68	2,445	Containment dousing system is on
0.76	2,750	Dousing tank water is depleted for containment sprays
1.31	4,723	LRVs open for Loop 2
1.47	5,306	Beginning of the core disassembly in Loop 1
1.75	6,305	Core collapse onto CV bottom Loop 1
1.83	6,585	Water is depleted in CV
1.84	6,622	Steam and water phases separated in PHTS Loop 2
2.20	7,922	At least one channel is dry in Loop 2 – Channel #2
2.26	8,126	PT and CT ruptured, PHTS Loop 2 - Channel #1
2.46	8,867	Beginning of the core disassembly in Loop 2
2.67	9,622	Core collapse onto CV bottom Loop 2
-	-	CV did not fail & Containment did not fail

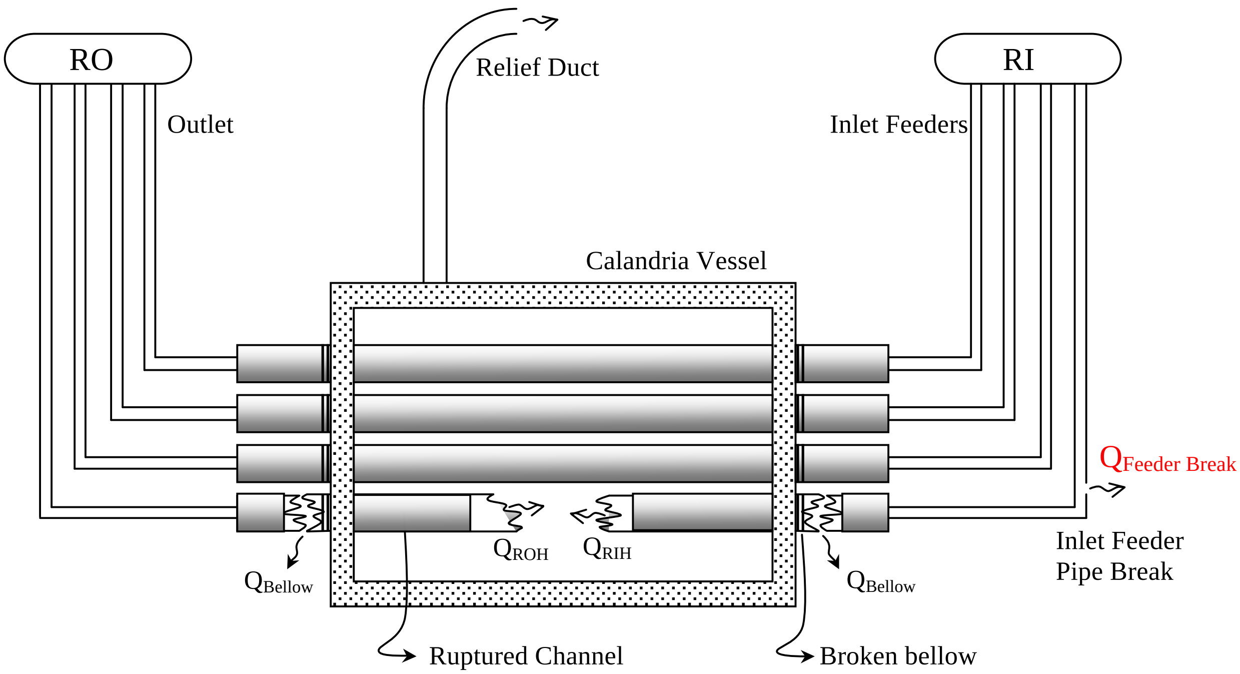

Figure 1 Simplified Fluid Flow Schematic for Stagnation Feeder Break Scenario

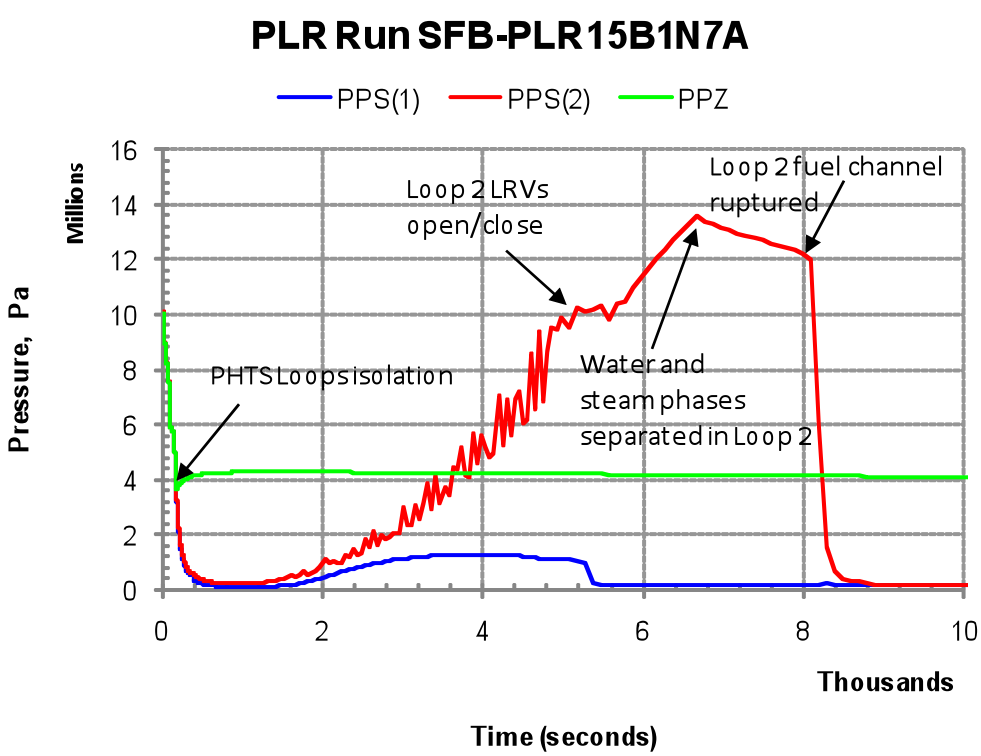

Figure 2 Primary System Pressure for Case A1 (0-10,000 s)

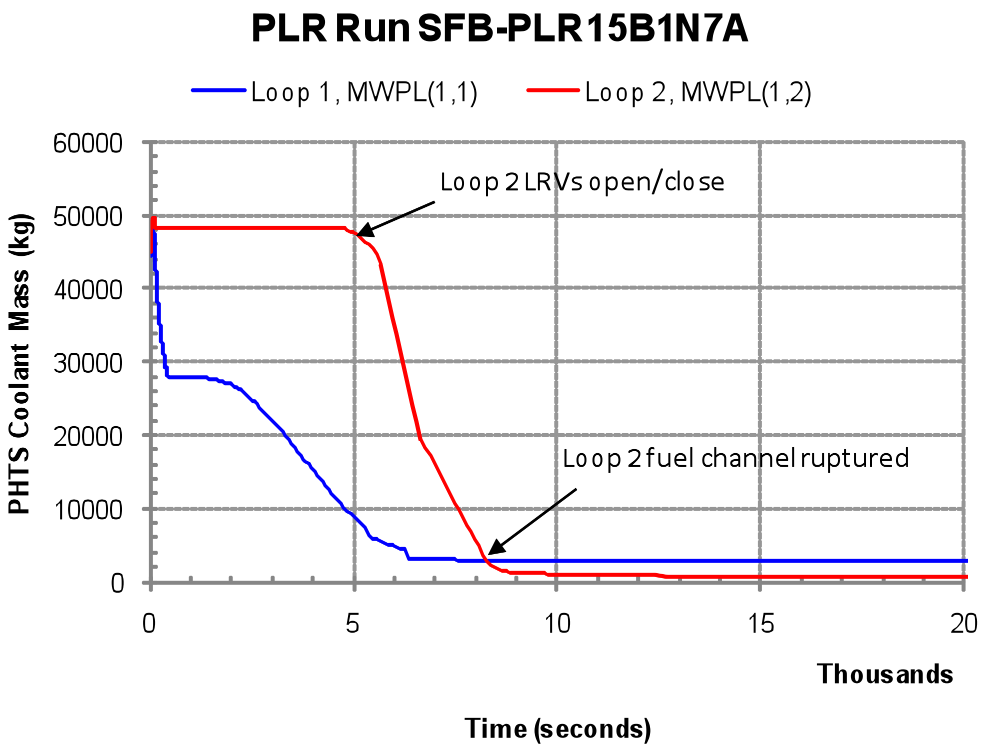

Figure 3 Primary System Water Mass for Case A1 (0-20,000 s)

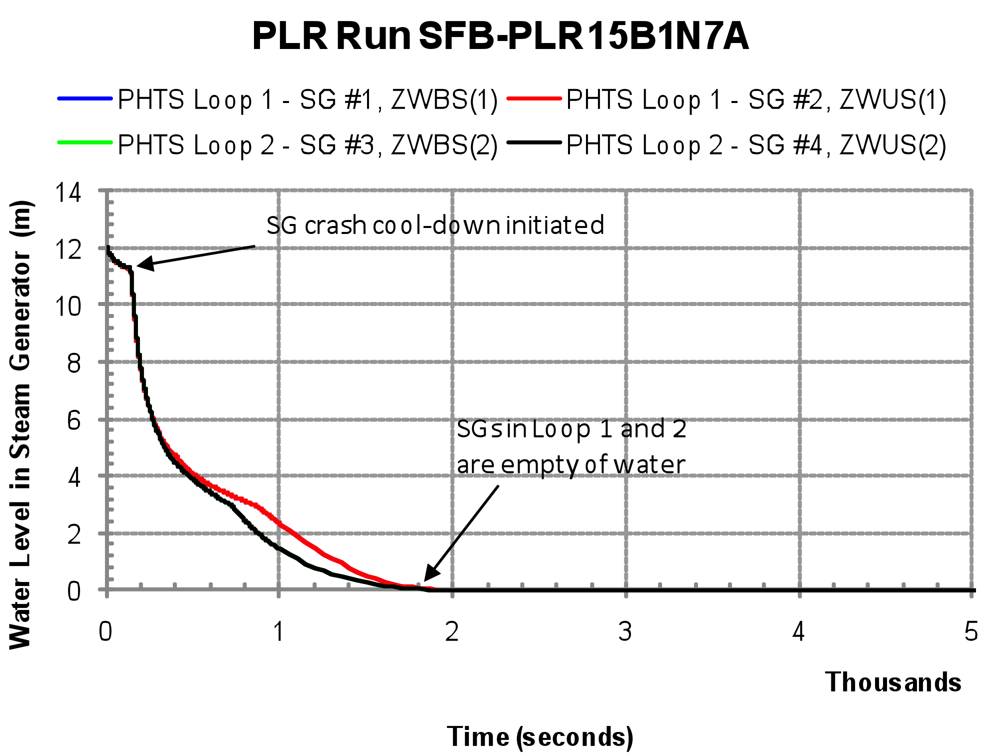

Figure 4 Steam Generator Water Level for Case A1 (0-5,000 s)

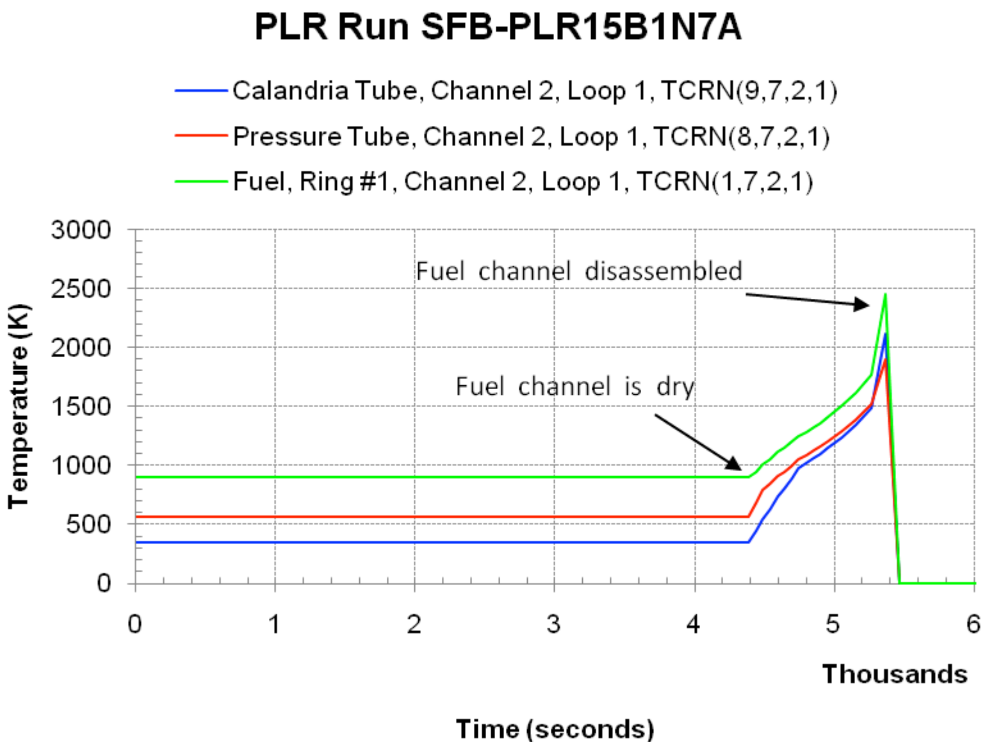

Figure 5 Fuel and Fuel Channel Temperatures, Loop 1 for Case A1 (0-6,000 s)

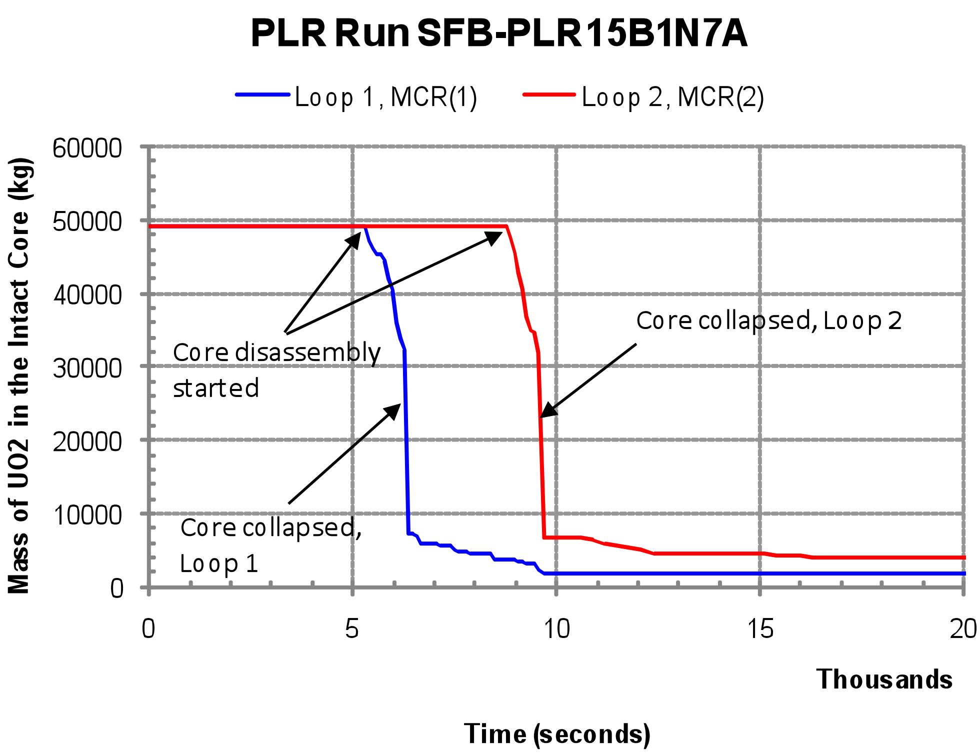

Figure 6 UO_2 Mass Remaining in the Intact Fuel Channels for Case A1 (0-20,000 s)

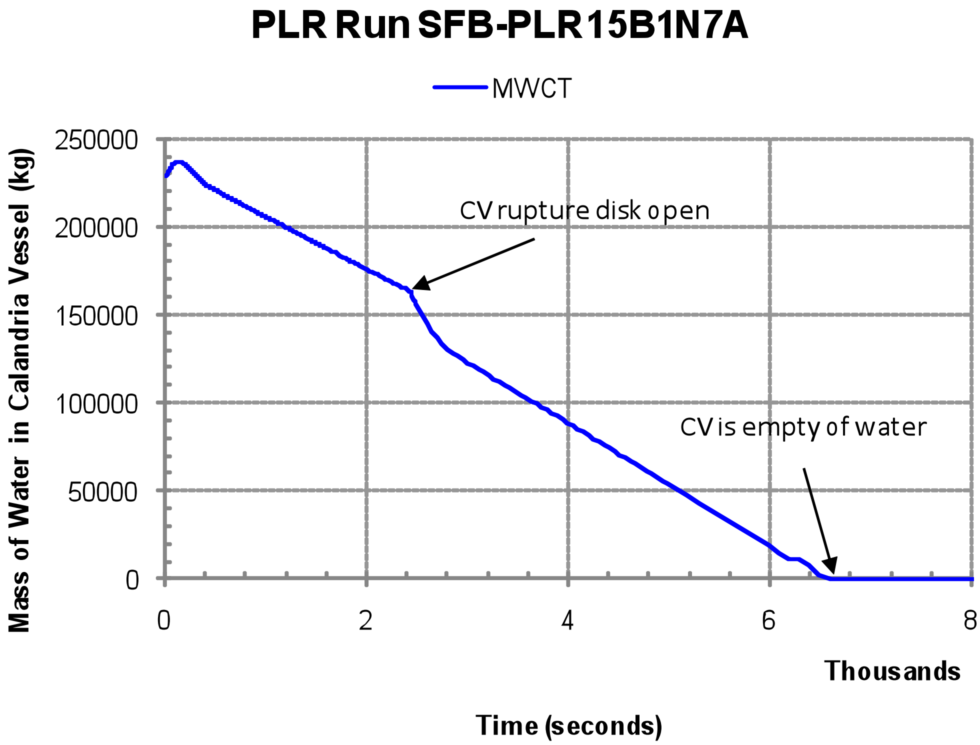

Figure 7 CV Water Mass for Case A1 (0-8,000 s)

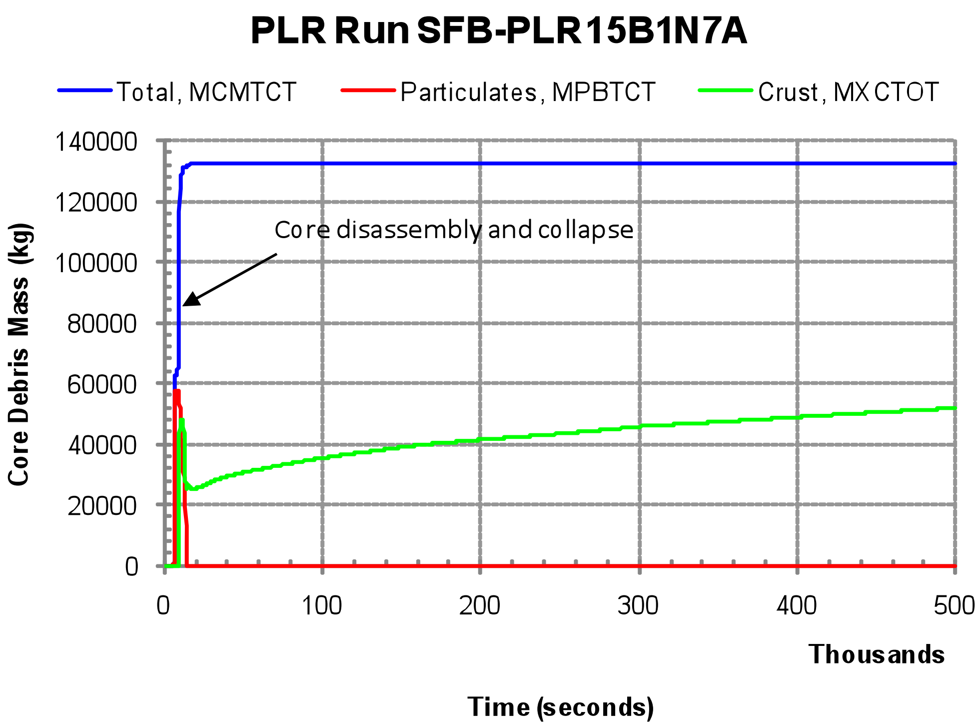

Figure 8 Mass of Corium Crust, Particulates and Total Core Debris in CV for Case A1 (0-500,000 s)

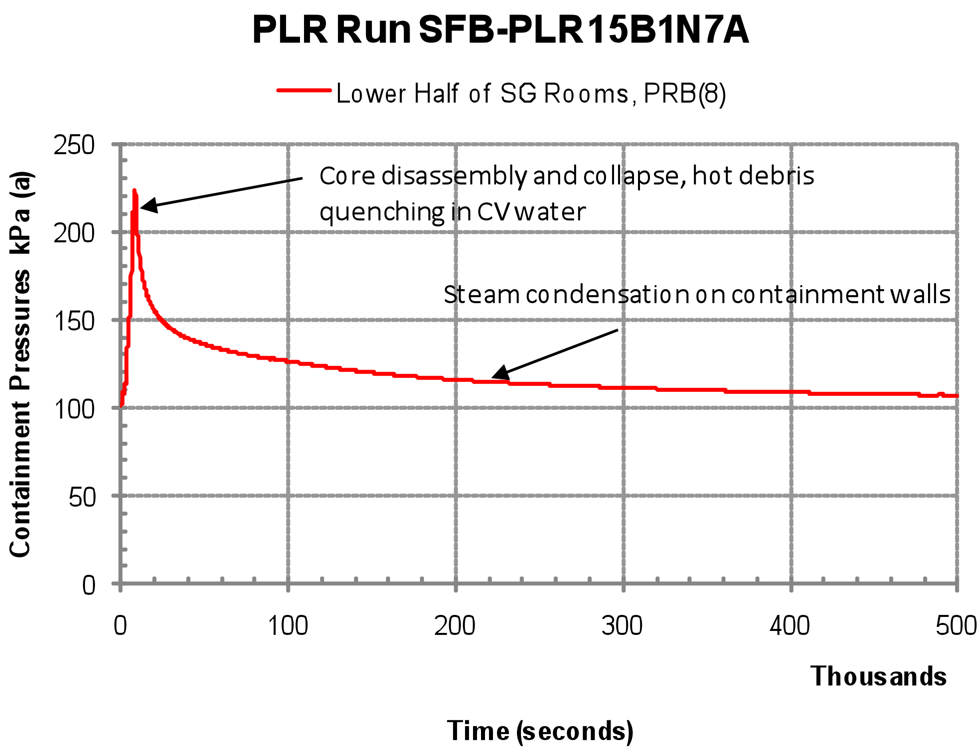

Figure 9 Containment Pressure for Case A1 (0-500,000 s)

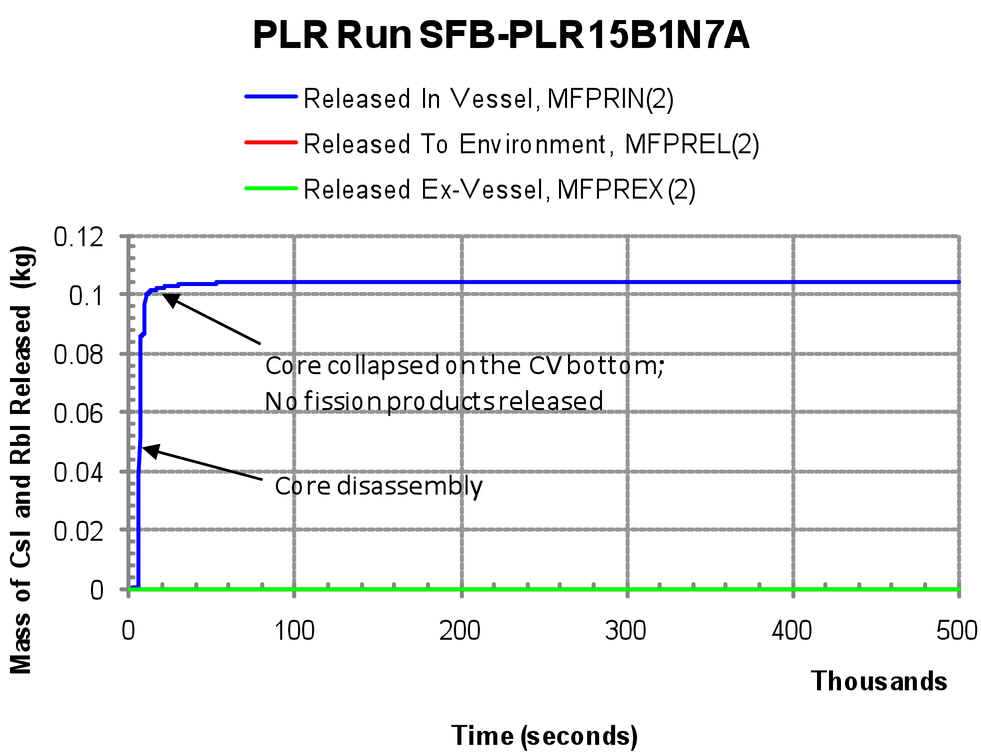

Figure 10 Mass and Location of Iodide-Release for Case A1 (0-500,000 s)