Application of PSA to CANDU Design and Licensing

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Abstract

AECL began a Generic CANDU Probabilistic Safety Assessment (GPSA) Program in 1998 to provide the basis for Level 1 and Level 2 PSA studies of AECL projects. Methodologies for a number of aspects of PSA were developed as part of that program and they were applied to critical accident sequences.

This paper focuses in a specific area of Level 2 PSA work, in particular, the development of MAAP CANDU consequence analysis code as part of the GPSA program and subsequent activities in this area. A beta version of the MAAP CANDU code was developed and tested with generic models and data from CANDU 6 and CANDU 9 designs as part of the GPSA program, and subsequently a new version of the code, version 4.04A+, was developed. This paper presents the preliminary results obtained with MAAP CANDU version 4.04A+ for a generic CANDU 6 station undergoing a postulated, unlikely station blackout accident sequence, where all alternate current (AC) normal, standby/emergency electrical power supplies and critical plant safety systems are assumed unavailable. Preliminary results show that the containment would not fail for at least a day and would thus provide adequate time for the operator to arrest the accident progression by administering accident management measures. These findings confirm the strong design resistance of CANDU reactors to severe accidents and show how the PSA can provide an effective support to CANDU licensing.

KEY WORDS

Severe Accidents, CANDU, MAAP CANDU, MAAP4 CANDU, Level 2 PSA, Station Blackout, CANDU 6

1. Introduction

A Generic CANDU Probabilistic Safety Assessment (GPSA) Program was initiated by Atomic Energy of Canada Limited (AECL) in 1998 to provide the basis for Level 1 and Level 2 Probabilistic Safety Assessment (PSA) studies of AECL products [1]. The GPSA Program had the following objectives:

- Develop a methodology to cover the full scope of Level 1 and Level 2 PSA,
- Acquire the tools and develop models for application of the methodology,
- Generate a reference analysis for use as a framework by future AECL projects, and

- Gain insights into the design of the fully developed AECL's reactor products, CANDU 6 and CANDU 9.

Figure 1 illustrates the GPSA program plan [2]. The final product of the program, the Generic CANDU PSA Report, can serve as a reference document for CANDU PSA practitioners, so they may compare the assumptions, methods and results of their PSAs with those performed at AECL. The GPSA methodology development was based on internationally accepted practices and procedures. While the methods for the analysis of internal events were generally well established and had been applied already to various projects at AECL, other PSA areas were tackled for the first time. Those areas included some Level 1 enhancements (improved human reliability analysis and common cause failure analysis), the probabilistic analysis of external events, and the Level 2 aspects of the PSA, i.e. the analysis of severe core damage progression and containment response.

During the execution of the GPSA Program, various activities shown in Figure 1 were conducted in parallel. For example, tools were acquired to perform new analyses in parallel to the development of the methodology. Such tools included computer codes for the analysis of the integrated plant response to severe core damage accidents. For the severe core damage accidents analyses, a computer code called Modular Accident Analysis Program for CANDU (MAAP4 CANDU or M4C) was acquired from EPRI through a sublicense with Ontario Power Generation Inc. (OPG, formerly Ontario Hydro).

This paper focuses on the development of the M4C severe accident consequence analysis code as part of the GPSA program and the subsequent activities. A beta version of the code was developed initially and tested with generic models and data from generic CANDU 6 and CANDU 9 designs as part of the GPSA program. Subsequently, a new version of the code, version 4.04A+, was developed. This paper presents the preliminary analysis results obtained with the new code version for a generic CANDU 6 station, when it undergoes a postulated, low probability Station Blackout accident sequence (SBO). The analysis also used additional assumptions, so that the SBO sequence would result in severe core damage.

2. Brief Description of MAAP4-CANDU Code and Capabilities

M4C is a computer code that can simulate the response of the AECL CANDU stations and the OPG stations during a severe accident, including actions undertaken as part of the accident management. The M4C code was developed on the basis of the MAAP4 code, which is used for pressurized and boiling water reactors. The most important distinguishing feature of M4C compared with MAAP4 is the model of the CANDU horizontal reactor core with fuel bundles situated inside pressure and calandria tubes.

M4C models the severe accident progression in a CANDU station starting from normal operating full power conditions for a set of plant system faults and initiating events followed by PHTS inventory blowdown, core heatup and melting, fuel channel failure, calandria vessel (CV) failure, reactor vault (RV) failure and containment failure. Fission product release, transport and deposition is modeled in M4C. Furthermore, some models are included in the code to analyze accident mitigation measures, such as core debris cooling in the CV and containment.

3. Nodalization of a CANDU 6 Station

Since a CANDU station consists of several complex systems and components, M4C simulates only the most significant systems, components and processes that are necessary to demonstrate the overall response of the plant to a severe accident.

The PHTS nodalization is comprised of two symmetric loops, with the flow through each loop following a "figure of eight" configuration. Each PHTS loop primary side is represented by 14 nodes. Steam generator (SG) secondary side is modeled as one node.

Although the CANDU 6 core has 380 fuel channels arranged in 22 rows and 22 columns, the M4C core nodalization uses a simplified core. The 22 rows of the core are divided into 6 vertical nodes, with each vertical node having 3 groups of characteristic channels representing 3 fuel channel power levels. The 22 columns of fuel channels are divided into 2 symmetrical halves, each representing one PHTS loop. The twelve bundles in a fuel channel are represented by 12 axial nodes. Thus, the total 380 channels are modeled by a total of 432 nodes. The 37 fuel pins of a CANDU fuel bundle are represented by 7 concentric rings and the pressure tube (PT) and the calandria tube (CT) each by a ring.

4. Major Modeling/Analysis Assumptions

Several simplified assumptions were made in the current analysis. These assumptions are either assumed in the present analysis or either embedded in the code.

Analysis Assumptions

- AC power and all onsite standby/emergency electric power are unavailable.
- Reactor shutdown is initiated immediately after accident initiation.
- Moderator cooling and shield cooling are assumed unavailable.
- Shutdown cooling system is unavailable.
- Main and auxiliary feed water are assumed unavailable.
- Emergency Core Coolant System (ECCS), including high (HPI), medium (MPI) and low pressure (LPI) injection, is unavailable.
- Crash cool-down system is not credited.
- Steam generator safety valves are available; they open and close at the set point.
- Containment dousing system is unavailable.
- Governor and main steam isolation valves are closed after accident initiation.
- Liquid relief valves (LRV) and pressurizer relief valves are assumed to discharge PHTS inventory into containment.
- All operator interventions are not credited.

In addition to the assumptions and models employed, results of the analysis will depend on the failure criteria used to fail components and systems, such as containment, CV, fuel channel, RV and so on. Failure criteria based on experimental results, if available, or analytical solutions and engineering judgments are used in the analysis.

The key input parameters for the analysis are in a parameter file. The parameter file for a generic CANDU 6 station consists of ~5000 parameters, which describe the important systems and components considered in the analysis.

5. Results and Discussion

For the present analysis, a SBO scenario is considered as a transient initiated by a loss of off-site AC (Class IV) power with subsequent loss of all on-site standby and emergency electric power supplies. Timings of the significant events calculated by the code are summarized in Table 1. The responses of the key systems are as follows:

Response of Primary Heat Transport System and Steam Generators

The postulated initiating events for the SBO accident scenario are imposed at the start of the simulation. Figure 2 shows the pressure in the PHTS loops and in the pressurizer. As expected, both loops show similar behaviour. Initially the pressure in both loops decreases with time because the core decay heat is transferred to the SGs by natural convection. The heat transfer from PHTS to SGs causes the water in SG secondary side to boil off resulting in opening of the main steam safety valves (MSSVs) and to discharge steam from the secondary side to outside the containment. The secondary side SG pressure then oscillates at the MSSV set point as the safety valves open and close. The secondary side water level in SGs decreases as boil-off proceeds. The SGs dry off at ~2.5 h, and are no longer a heat sink to remove heat from the PHTS. Thus, PHTS pressure starts to increase until it reaches the PHTS liquid relief valve (LRV) set point 10.16 MPa (a) and then oscillates at the relief valve set point as shown in Figure 2. Continuous PHTS inventory loss through the LRVs results in fuel channel dryout. In parallel, the moderator level in the CV decreases as a result of moderator boiloff and causes eventually the uncovery of some of the top fuel channels. Subsequent heatup of the CT and the PT at ~10 MPa PHTS pressure results in ballooning and rupture of the both tubes causing a rapid blowdown of the PHTS coolant into the CV. Therefore, the PHTS pressure drops drastically at ~4.4 h (see Figure 2).

Fuel Channel Response

Table 1 shows that the first fuel bundles are uncovered inside fuel channels at ~3.5 h in both loops following the loss of PHTS coolant. When the pressure tube temperature rises the PT and CT rupture at ~4.4 h. As the PT and CT axial segments heat up they sag, break up and form debris. The fuel channel fragments relocate to “holding bins” and are held there temporarily as a “suspended debris bed”. The suspended debris bed heats up further from the decay heat and from the Zr/steam exothermic reaction resulting in partial melting. Some of the molten material relocates from the suspended debris bed to the bottom of the CV and is quenched in the water. When the suspended debris bed cannot be supported by the intact channels covered by water, the suspended bed and most of the intact channels relocate to the bottom of the CV by core collapse at ~8.3 h.

Calandria Vessel Response

The moderator temperature and pressure in the CV increase as a result of the loss of moderator cooling and heat transfer from the core. As a result of fuel channel rupture at ~4.4 h, the pressure inside the CV reaches 238 kPa (a), the set point of the rupture disks. The rupture disks fail and the moderator is lost through the relief ducts.

Following core collapse the moderator in the CV is depleted at ~8.9 h; therefore, no steaming source is available to cause pressure increase. The water in the RV cools the external CV wall. Steam generated in the RV is released into the containment. When water level in the RV reaches the CV bottom, the CV bottom heats up rapidly by the heat from the core debris and the CV fails due to creep at ~42.4 h. The debris relocate into the RV, where they are cooled by water.

Reactor Vault and End-Shield Response

The pressure and water levels in the RV and end-shields increase gradually after the initiating event because of the loss of shield and moderator cooling. At ~4 h the rupture discs connected to the combined vent lines of the RV and end-shields burst and steam is discharged from the end-shields to the containment. The RV water boils off beginning at ~14.5 h resulting in a gradual decrease of the RV water level.

At ~42.4 h, the CV bottom fails and corium from the CV relocates into the reactor vault floor. Eventually all water in the reactor vault dries out at ~46 h and the corium then reacts with the concrete floor. At ~104.3 h the RV floor fails and corium/concrete mixture relocates into the basement, where it interacts with the water.

Containment Response

Figure 3 shows the pressure in the containment node representing the lower half of steam generator enclosure along with the pressure in the end-shield and RV. After accident initiation, the containment pressure increases gradually because water is discharged into the containment through the PHTS LRVs. The rapid increase or (decrease) of containment pressure as shown in Figure 3 at the various times ~2.5 h, ~4.4 h, ~6.3 h, ~27.1 h, ~42.4 h, ~104.3 h can be explained by the following processes, which occur at those respective times: (1) PHTS coolant release via LRVs and CV bleed valves, (2) PT and CT rupture, (3) core collapse, (4) containment failure, (5) CV failure and corium relocation into the reactor vault and (6) corium relocation into the basement after reactor vault failure. At ~27.1 h, the containment pressure reaches the failure set point of 500kPa (a) resulting failure of the containment.

Fission Product and Hydrogen Release

The major portion of the initial inventory of the noble gases (57.7 kg) is released into the CV from the fuel and the suspended debris bed during core disassembly and core collapse starting from ~4.8 h up to ~8.3 h. Since no containment leakage or ventilation is modeled, eventually all noble gases are released to the environment when the containment fails at ~27.1 h.

The initial inventory of CsI and RbI in the core is ~32 kg. Figure 4 shows the mass of CsI and RbI released in-vessel, ex-vessel (outside CV), in the PHTS, in CV, in containment, and to the environment. The fission product release models used in M4C are based on references [3,4]. Since the PT and CT rupture at ~4.4 h fission products in the form of aerosol are released. Because the CV rupture disks are already opened at ~4.4 h, the fission products are released through the CV rupture disks into the containment. The mass of CsI and RbI in containment (including airborne and deposited) remains at ~1.2 kg until ~46 h, when the corium/concrete interaction releases more fission products. Because almost all of CsI and RbI are retained in the containment by various retention mechanisms, the total mass of Cs and I released to the environment in the form of CsI, RbI and CsOH is only ~1.82% of the initial inventory.

During the SBO sequence hydrogen is generated as a result of the following processes: (1) Zr-steam reaction in the fuel channels and in the suspended debris bed during core debris oxidation, (2) Zr-steam reaction in the RV due to molten debris jet breakup in the water pool and (3) chemical reactions during molten core-concrete interaction. The analysis results show that the mass of hydrogen generated in PHTS and CV is ~200 kg prior to CV failure and it is ~ 2200 kg in the RV as a result of molten debris jet breakup in RV and molten corium-concrete interaction.

6. Conclusions

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

- The analysis performed in this study has demonstrated the capability of the MAAP4 CANDU V4.04A+ code for application to Level 2 PSA of CANDU 6 stations.
- The results and trends obtained in this study are consistent with prior engineering judgement for the Station Blackout accident scenario analyzed.
- The total mass of Cs and I released to the environment in the form of CsI, RbI and CsOH is less than 2% of the initial Cs and I core inventory.
- The results show that significant time is available for operator action during the Station Blackout accident to arrest the accident progression.

7. References

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TABLE 1. Sequence of Significant Events for Station Blackout Scenario

Time (h)	Comments
0	Loss of AC and all Backup and Emergency Power supplies
2.5	SG secondary side is dry
2.5	LRVs first opening
3.5	Fuel bundles uncovered within fuel channels (at least one channel)
4.0	Reactor vault rupture disk open
4.3	At least one channel is dry (complete boil-off)
4.4	Pressure and calandria tubes are ruptured
4.4	Moderator in calandria vessel reaches saturation temperature
4.8	Beginning of the core disassembly
6.3	Beginning of core debris relocation onto the CV bottom
8.3	Collapse of entire core onto the CV bottom
8.9	Calandria vessel water is depleted
14.5	Water in reactor vault reaches saturation temperature
27.1	Containment failed
42.4	Calandria vessel failed due to creep
46.0	RV Water depleted
104.3	RV failed as a result of concrete erosion

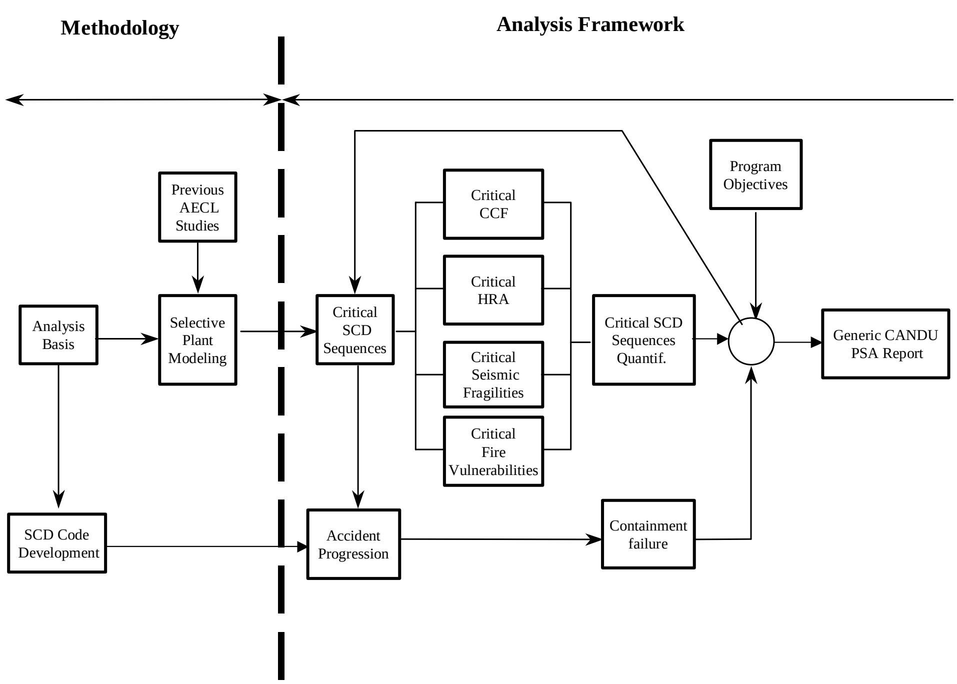

Figure 1: Program Structure of the Generic CANDU PSA Program

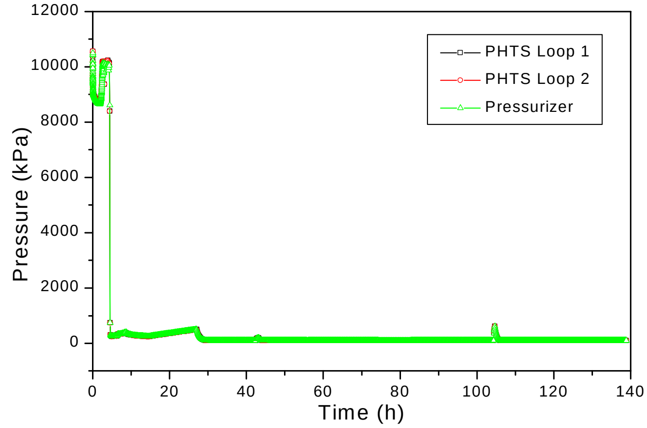

Figure 2: Pressure in PHTS loops and in Pressurizer (SBO)

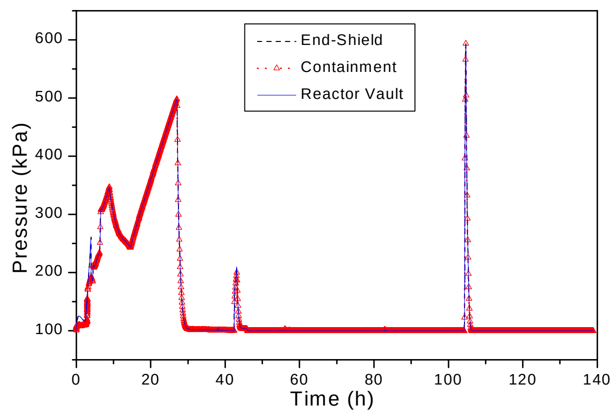

Figure 3: Pressure in Containment, Reactor Vault and End-Shield (SBO)

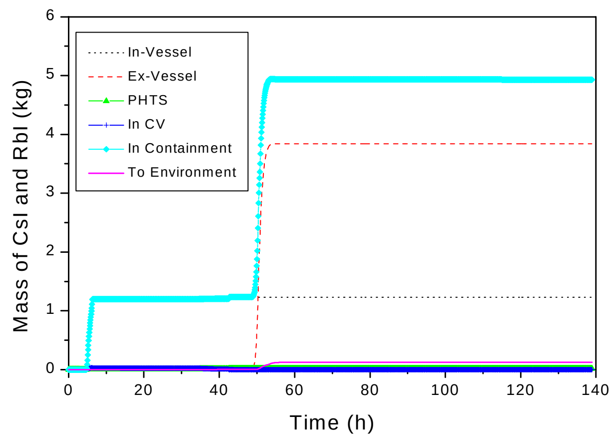

Figure 4: Mass of CsI and RbI released and in various parts of Reactor Building (SBO)