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LEVEL-2 PSA SENSITIVITY STUDY FOR AIRLOCK SEAL FAILURE PRESSURE UNDER SEVERE ACCIDENTS IN CANDU-6 PLANTS

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1. Introduction

The major function of the airlocks (see Fig.1) in CANDU-6 (C6) plants is to provide access for the personnel and equipment to and from the interior of the Reactor Building (RB), without breach of containment and with maintaining the pressure tight barrier. As such, the airlocks are systems important to safety as a part of the RB (corresponding to Containment in PWRs) system [1]. It was reported that there were maintainability issues with this airlock in the past [2]. Especially, inflatable seals used in C6 large and small size airlock doors had issues in maintaining the leak tightness related to the components that maintain the seal inflation and operation. Specifically, recent study of the pressure capacity of RB penetrations and airlocks for C6 stations identified that the RB airlock seals blow-by may occur at lower pressure than the concrete through wall cracking pressure [3].

This paper is about an impact analysis on the overpressure protection capability of the containment airlock seal related with a leak failure resulting from the decrease in the pressure at which the C6 RB breaks down. Specifically, the pneumatic seal area for mechanical door seals is highlighted as the maximum weakness of the RB pressure boundary at Wolsong plants. This study is motivated by the insufficient UPC (Ultimate Pressure Capability) analysis used in the previous full power PSA, where a value of 48 psig was used for the median failure pressure. In this sensitivity study, the median failure pressure is lowered from 48 psig (by concrete through wall cracking failure) to 38 psig (by seal leak failure). In the previous analysis, the RB damage probability (which means containment failure frequency) was only 5% at 38 psig, but the failure probability based on the new UPC-based fragility curve increased to 50%, which should have a negative impact on existing PSA results. This study requires not only probabilistic safety analysis (PSA) but also deterministic safety analysis (DSA). The DSA analyzes the possibility for new breakages of RB using MAAP(1)-ISAAC severe accident analysis code [4] developed by KAERI. The Level-2 PSA then uses the DSA results to create an impact analysis model (for example, Containment Event Trees) for the UPC decrease.

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(1) MAAP is an Electric Power Institute (EPRI) software program that performs severe accident analysis for nuclear power plants including assessments of core damage and radiological transport. A valid license to MAAP4 and/or MAAP5 from EPRI is required.

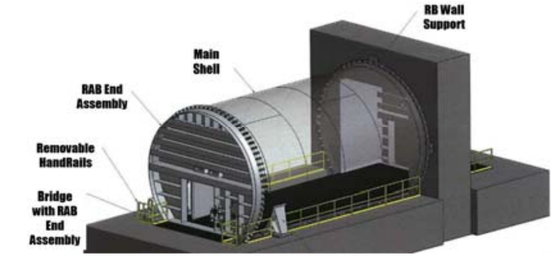

Fig. 1 Typical PHWR Airlock configuration (CANDU-6 Plant)

2. Review of Failure Mode and Timing of CANDU-6 Reactor Building under Severe Accidents

When the RB failure is considered, typically early and late failure modes are taken into account. Early failure mode consists of containment bypass and isolation failure, which is the same with PWR case and is not described here. In late failure modes, the reactor building can fail due to an energetic hydrogen burn and steam explosion. MAAP-ISAAC code models the hydrogen generation both from the core and from the concrete along with the concrete ablation. As Wolsong plants installed 27 passive autocatalytic recombiners (PARs), they are considered in the code. The MAAP-ISAAC calculates the amount of hydrogen in the RB in each scenario. When hydrogen mass is enough to burn, then the code calculates the pressure rise based on adiabatic isochoric complete combustion. No dynamic pressure load is taken into account. Regarding the steam explosion, the pressure rise from the steam spike is modeled. Except the hydrogen burn and steam explosion, the most dominant RB failure mode is caused by the RB steam over-pressurization according to general results of DSA. Once the PHTS has voided (via break or boil-off), steam in the calandria vessel (as the first stage) and the reactor vault (as the second stage) gradually pressurizes the RB. Also non-condensable gases from the interaction of molten core material with the concrete basemat of the reactor vault (as the third stage) increases RB pressure. This pressurization process could last from hours to several days, depending upon stages and the effectiveness of engineered safety features. The local air coolers (LACs) are effective in condensing steam produced by decay heat emanating from the debris. In addition, the availability of the shield cooling system removes decay heat from the debris to avoid the RB pressurization (in the second stage).

3. Review of CANDU-6 Containment Performance with Seal Failure under Severe Accidents

According to the Wolsong DSA model, the RB pressure under extreme severe accidents (with failure of the engineering safety equipment) has typically showed peaks at two time points such as a coolant depletion time (= about 12 hours into the accident) and a CT failure time (= about 36 hours into the accident) in Fig.2.

In the meantime, as Wolsong has a large amount of cooling water, it takes a long time to proceed with the severe accident. Therefore, except containment bypass and isolation failure, there is almost no early failure of RB while late and very late damages, which correspond to the steam overpressure before CT failure and the hydrogen explosion after CT failure, are the respective predominant failure modes. Especially, the steam overpressure failure typically occurs before three days into the severe accidents except for three conditions such as (1) successful LACs, (2) successful CFVS, and (3) the LOCA, where cooling of the steam generator secondary side and the loop isolation are successful) regardless of the initiating events. Together with the above three cases, medium-sized overpressure (not being certain of steam overpressure failure) occurs at the (4) successful ESC (End Shield Cooling) condition. Here, if the median failure pressure of the UPC drops from 48 psig to 38 psig, overpressure failure probability increases because the overpressure frequency in case-(3) and case-(4) conditions is newly created and/or becomes higher, respectively.

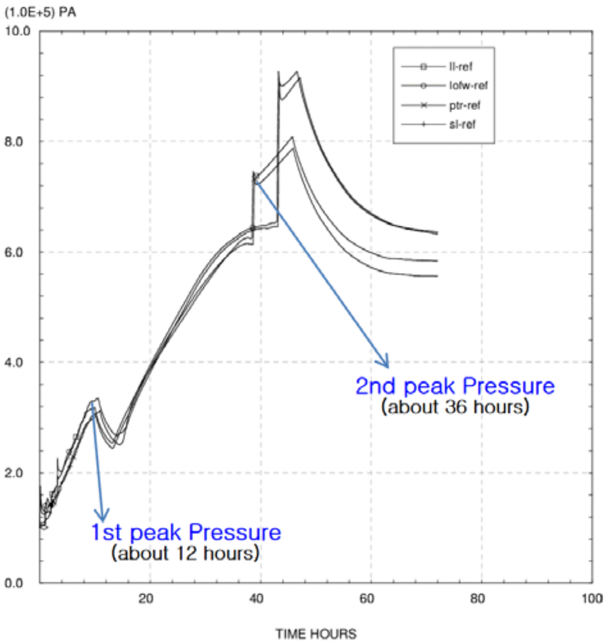

Fig. 2 Typical Wolsong RB Pressure Trend under Extreme Severe Accidents

4. Level-2 PSA Sensitivity Analysis Results

According to full power PSA results developed for Wolsong by KHNP (2011) [5], the median UPC failure pressure used for containment damage from wall cracking (rupture sized failure) was 48 psig. At that time, CFVS not installed yet is not considered, whereas the effect of PARs is included. As results of 2011 PSA analyses, the value of core damage frequency (CDF) or total plant damage state (PDS) frequency was 1.85E-6/yr in which the probabilities of all four modes (no failure / early failure / late failure / very late failure) of PDS are shown in Table 1. As can be seen in the table, the failure rate due to the hydrogen explosion is enhanced by 3.8% of CDF (from 1.395E-7/yr to 6.973E-8/yr). If the PAR operation is considered, the containment failure frequency due to the hydrogen explosion, which is responsible for the most part of very late failure mode, is reduced to 50%.

Failure Mode	Failure Mode	2011 Internal Event PSA (CFVS uninstalled, 48 psig)	
		No PAR	PAR
No Failure (=Intact)	Calandria Arrest	26.94	26.94
	No RB Failure	8.21	11.98
	Sum	35.15	38.92
Early Failure	Isolation Failure	0.33	0.33
	SGTR w/CC	0.29	0.29
	SGTR w/o CC	0.02	0.02
	V-Sequence	0.03	0.03
	Sum	0.68	0.68
Late Failure	Steam Overpressure	56.46	56.47
	Steam Explosion	0.02	0.02
	Sum	56.48	56.48
Very Late Failure	Steam Overpressure	0.08	0.04
	Steam Explosion	0.08	0.11
	Energetic H_2 Burn	7.54	3.77
	Sum	7.69	3.92
Total Sum		100%(=1.851E-06)	100%(=1.851E-06)

Tab. 1 Containment Failure Probability for Wolsong

Since the 2011 PSA model for full power, KAERI developed the 'low power & shutdown' PSA model [6] in 2016. In this model, the CDF value increased by about 12 times (from 1.85E-06/yr to 2.156E-05/yr) compared to the 2011 PSA model, where most of the increased value was added to the mode of no containment failure. This is because the CDF value in 2016 included limited core damage (LCD) without RB failure. In other words, both the core damage frequency and the containment failure frequency increased when compared to the 2011 model, but the probability of containment failure in the late and very late periods decreased. Specifically, the probability of late and/or very late failures among PDS become less than a quarter of 2011 values. The intercomparison results between 2011 and 2016 PSAs where no operation of CFVS is presumed are shown in the table below.

RB 상태 (CDF [1/yr])	No Failure	Early	Late	Very Late
2011 (1.85E-6) No CFVS 48psig	**7.204E-7 (38.9%)**	**1.252E-8 (0.7%)**	**1.045E-6 (56.5%)**	**7.255E-8 (3.9%)**
2016 (2.156E-5) No CFVS 48psig	**1.848E-5 (85.7%)**	**1.544E-7 (0.7%)**	**2.753E-6 (12.8%)**	**1.756E-7 (0.8%)**

Next, using the 2016 model, a UPC reduction (48→38 psig) model was developed and its effect was analyzed. In addition to the base analysis, the '2HR' sensitivity analysis mentioned in the previous DSA model as '(3)' (new case for overpressure failure in the LOCA scenario where the steam generator secondary cooling succeeded), and the '2HR+ESC' sensitivity analysis as the composite of '(3)' and '(4)' (increased overpressure failure at the success of ESC) are analyzed. If classified by three (base/2HR/(2HR+ESC)) cases with CFVS operation (or success) / non-operation (or failure) cases, total six sensitivity cases were analyzed for the UPC value of 38 psig. Summary of sensitivity results is shown in the Table 2. In the summary results, UPC reduction analysis showed that only the late containment failure, which is dependent on the steam overpressure, increased by about 33% (see Table 2 in the third line) while the very late failure, which is dependent on the hydrogen explosion, was not affected. On the other hand, if the operation of the CFVS installed after 2016 is successful, the late containment failure due to steam overpressure has prevented by a half or more (see Table 2 in the fourth line), though a limited source term (mainly due to inert gas nuclides that is not filtered) is inevitable.

RB Failure mode / Frequency [/ry]	No Failure	Early	Late	Very Late
Total: 2.156E-5 **(2016) CFVS Unavailable, 48psig**	1.848E-5 (85.7%)	1.544E-7 (0.7%)	2.753E-6 (12.8%)	1.756E-7 (0.8%)
Total: 2.156E-5 **(2016) CFVS Unavailable, 38psig**	1.757E-5 (81.5%)	1.544E-7 (0.7%)	3.662E-6 (17.0%)	1.756E-7 (0.8%)
Total: 2.156E-5 **(2016) CFVS Available, 38psig**	1.921E-5 (89.1%)	1.544E-7 (0.7%)	1.757E-6 (8.2%)	4.391E-7 (2.0%)

Tab.2 Sensitivity Results Using 2016 PSA Model

5. Summary

According to the Wolsong PSA history, the probabilities of late and very late breakdown of reactor building have changed owing to the adoption of new software (methodology update) or hardware (system upgrade). For example, the frequency of core damage has increased in recent year (2016) compared to 2011 because it included limited core damage without RB damage in 2016 due to the safety system upgrade. Nevertheless, the fraction of late and very late failures becomes smaller in 2011 due to bigger increase in the intact probability of RB. If UPC decrease (from 48 psig to 38 psig) is applied, the late breakdown, which is dependent on the steam overpressure, has increased by about 33% whereas the very late breakdown mainly due to the hydrogen explosion was not affected. Furthermore, should the operation of the CFVS be successful, the late RB damage due to steam overpressurization is prohibited by more than a half though a limited source term from unfiltered noble gas have to occur.

In conclusion, if the RB failure (leakage) occurs at lower pressure due to the weakness of the airlock seal, source term release into the environment should be larger. There are two reasons. One is additional failures of the RB (with new damage within 3 days after LLOCA) or increased failure probabilities of existing failure scenarios (with new damage at the successful ESC condition). The other is possible earlier timing of RB failure (occurring from 20~24 hours (existing) to within 12 hours (new)) in which RB opening keeps longer even if the same event progress occurs inside the RB. In order to prevent this, CFVS operation should be considered but it could have a disadvantage that is likely to act as an obstacle to meet the 100 TBq standard of AMP (Accident Management Program) source term required by recent Korean regulation.

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