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OVERVIEW OF CONTAINMENT FILTERED VENT UNDER SEVERE ACCIDENT CONDITIONS AT WOLSONG NPP UNIT 1

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Containment Filtered Vent Systems (CFVSs) have been mainly equipped in nuclear power plants in Europe and Canada for the controlled depressurization of the containment atmosphere under severe accident conditions. This is to keep the containment integrity against overpressure during the course of a severe accident, in which the radioactive gas-steam mixture from the containment is discharged into a system designed to remove the radionuclides. In Korea, a CFVS was first introduced in the Wolsong unit-1 nuclear power plant as a mitigation measure to deal with the threat of over pressurization, following post-Fukushima action items. In this paper, the overall features of a CFVS installation such as risk assessments, an evaluation of the performance requirements, and a determination of the optimal operating strategies are analyzed for the Wolsong unit 1 nuclear power plant using a severe accident analysis computer code, ISAAC.

KEYWORDS : Containment Filtered Vent, CFVS, Wolsong NPP, Steam Overpressurization, ISAAC

1. INTRODUCTION

A CFVS is intended to protect against a loss of integrity of the containment in the event of an internal over pressurization after a severe accident, and to thereby reduce the land contamination and health risk issues. This technique has been studied for more than 25 years [1], and has been applied to nuclear power plants mainly in Europe [2] and Canada [3].

Unlike PWRs, CANDU plants have about twice the coolant inventory that can be released from the reactor vessel (=calandria vessel). If ex-vessel coolant inventory is included, more than triple the mass is possible. Most of it will become steam under severe accident conditions such as an SBO (Station BlackOut) with a simultaneous loss of cooling caused by the secondary side of the steam generator, ECC (Emergency Core Cooling) heat exchangers, and moderator heat exchangers. Furthermore, if local air coolers (LACs) do not operate as designed, the pressure of the containment (i.e., the reactor building in the Wolsong plant) will increase to over the ultimate failure pressure in the long run. In contrast, the design and failure pressures of the CANDU reactor building (RB) are less than one-half that of PWRs. Hence, steam over pressurization is a big threat to the integrity of the CANDU RB if no recovery action can be taken for a long period of time, as in the case of the Fukushima incident.

Wolsong unit 1 (WS-1) was recently chosen to be backfitted with a wet-style CFVS. If the operational strategy of the CFVS is well established, it will greatly reduce the threat from a steam over pressurization failure mode of the WS-1 RB. In addition, it will minimize the uncontrolled releases of airborne particulate radionuclides and radioiodine isotopes into the environment by means of controlled filtering. In this study, the anticipated effects of a CFVS are analyzed using the ISAAC (Integrated Severe Accident Analysis code for CANDU plants) computer code [4], which was developed in Korea. As both the design specification and mechanical models for a CFVS are not available at this time, an RB leak and/or rupture are assumed, and a change in the inner RB conditions reflecting the thermal hydraulic requirements for the CFVS performance is analyzed. From this study, it is expected that the positive and possible negative effects of a CFVS with different vent sizes and operational strategies under a hypothetically severe SBO accident will be elucidated.

2. ISAAC CALCULATION TOOL

The current study uses version 4.0.3 of the ISAAC computer code [5]. The ISAAC computer code is a flexible, efficient, and integrated tool for evaluating the in-plant effects of a wide range of postulated accidents and for examining the impact of operator actions on accident progressions. The code can predict the progression of hypothetical accident sequences from a set of initiating events to either a safe, stable, and coolable state, or to an impaired RB and depressurization.

The ISAAC code is constructed into modules covering individual regions of the plant: the primary heat transport system (PHTS), pressurizer, steam generators, calandria vessel (CV), reactor vault (RV), end-shields, degasser condenser tank, and the RB. All major engineered safety features are represented in the code: the shutdown cooling system, emergency core cooling system, moderator and shield cooling system, local air coolers, igniters, a passive autocatalytic recombiner (PAR), and a dousing spray system.

The code evaluates a wide spectrum of phenomena including steam formation; core heat-up; cladding oxidation and hydrogen evolution; vessel failure; corium-concrete interactions; ignition of combustible gases; fluid entrainment by high-velocity gases; and fission-product release, transport, and deposition. The code also addresses important engineered safety systems and allows a user to model the operator interventions. Furthermore, models are added to characterize actions that could stop an accident, i.e., in-vessel cooling, external cooling of the calandria vessel, and ex-vessel cooling. Moreover, mathematical techniques are implemented to maintain a quick-running code suitable for extensive accident screening and parameter sensitivity analysis applications.

The ISAAC code was developed on the basis of the MAAP4 PWR (pressurized water reactor) code and includes three main groups of models: (1) generic models developed for light water reactors (LWRs), (2) CANDU6-specific engineered safety system models, and (3) CANDU6-specific models for a horizontal core, figure-of-eight primary heat transport system, and the calandria. The generic models of ISAAC evolved from the MAAP4 code, which was developed by Fauske and Associates. LLC (FAI), for PWRs. Some of these models required minor modifications to adapt them to the CANDU6 design features for integration with the rest of the code, but these models were fundamentally unchanged from the generic MAAP4 versions. As the CANDU6 type reactor differs from typical PWRs, the CANDU6-specific features are newly modeled and added to the ISAAC code. The CANDU6-specific models include the calandria vessel, end shields, reactor vault, pressure and inventory control, and engineered safety systems (e.g., dousing, ECC). One of the most important distinguishing features between ISAAC and other MAAP4 versions is the CANDU reactor core with fuel bundles situated inside horizontal pressure and calandria tubes. In addition, the large quantities of relatively cool water (moderator and reactor vault water) provide significant heat sinks, distinguishing the CANDU models from LWR models.

ISAAC models a broad spectrum of physical processes that might occur in the core during an accident, such as the followings:

- Fuel and fuel channel temperature excursions, deformation of the fuel and fuel channels, and interactions with the moderator system;
- Zirconium-steam exothermic reaction;
- Thermal mechanical failures of the fuel channels;
- Disassembly of the fuel channels;
- Formation of suspended debris beds;
- Motion of solid and molten debris; and
- Interaction of the core debris with water

In particular, ISAAC models the CANDU feeders, end-fittings, fuel channels, and fuel. The ISAAC models concentrate on the behavior of these core components within the CV as the fuel channels disassemble, form suspended debris supported by intact channels, and relocate the suspended debris to the debris bed within the CV. Each characteristic channel represents a larger number of channels (known as associated channels) with similar powers, elevations, and feeder geometries.

The ISAAC PHTS thermal hydraulic models are simplified using such assumptions as coarse nodalization, equilibrium within the fluid phase, a uniform loop pressure, and a single global void fraction at which a phase separation occurs. The ISAAC PHTS thermal hydraulics results cannot be expected to be as accurate as those from more detailed PHTS models associated with a thermal hydraulic code such as CATHENA. Most importantly, however, ISAAC is an integrated code that models the interactions amongst many systems modeled in an integrated fashion. Thus, ISAAC calculates the effects of the interplay between the RB, CV, PHTS, reactor vault, core, etc. In addition, ISAAC has the ability to input operator actions by enabling, disabling, or modifying a system at the user's request.

3. BASE SCENARIO ANALYSIS FOR HYPOTHETICAL SEVERE ACCIDENT

An SBO scenario, where all off-site power is lost and the diesel generators fail, is simulated as an initiating event of a severe accident sequence. The scenario has been taken as a very low-frequency, but high-risk, accident event. All current generation reactors are only partially designed to cope with a station blackout. During an SBO event, the initiating event (time = 0 s) is a loss of Class IV and Class III power, causing a loss of pumps used in systems such as the PHTS, moderator cooling, shield cooling, steam generator feed water, and recirculating cooling water. The SBO scenario does not credit any of these active heat sinks, but relies only on the passive heat sinks, particularly the initial water inventories of the PHTS, moderator, the secondary side of the steam generator, end shields, and the reactor vault. This scenario succeeds in the operation of a reactor shutdown, reactor building isolation, dousing spray, and PARs, but the LACs are unavailable. The following several assumptions have been made for the analyses in this paper (normally, the best-estimate assumptions were used for this study, but when the availability of the data related to modeling a certain process was limited, conservative assumptions were applied):

- The reactor is at full power (2157.5 MW(th)) initially and a reactor shutdown occurs upon an accident initiation through a complete loss of Class IV power.
- The RB dousing spray system was credited and an RB pressurization up to 114 kPa(a) triggers dousing, the total capacity of which is 1,559 m^3 of water inventory and a maximum water flow rate of 6,800 kg/sec [6].
- A total of 27 PARs are credited in the WS-1 plant [7]. The PARs are intended to be activated with an assumed time delay of 1,800 s after the hydrogen concentration first reaches 2%. After starting, the PARs run until the local hydrogen concentration decreases to 0.2%. Subsequently, the PARs are intentionally started with no delay; whenever the hydrogen concentration is greater than 2%, provided there is sufficient oxygen available.
- A simple model for RB leakage was used, the maximum leak rate of which was 0.5% of the RB volume at an RB design pressure of 224 kPa(a), during 24 hours [6]. The RB failure is assumed to occur with a size of 0.1m^2 at a pressure of 426 kPa(a), which is equivalent to a 50% cumulative probability of the 50% confidence interval (i.e., the median pressure) of the WS-1 RB fragility curve [8].

The simulations were performed using ISAAC 4.0.3, and the simulations were run up to 500,000 s (139 h), including the corium behavior even after an RV bottom concrete melt-through (when compared with a PWR, severe accident progression tends to be delayed in a PHWR because of an abundant initial inventory of the coolant).

The analysis results for the base SBO scenario show that the PHTS inventory is gradually lost through the LRVs (liquid relief valves) resulting in a fuel channel dryout, and the uppermost channel ruptures at about 14,300 s. A significant quantity of the moderator inventory is discharged into the RB when the fuel channel rupture pressurizes the CV. Several top fuel channel rows are uncovered during this process. After the initial rapid moderator expulsion, the moderator continues to discharge gradually into the RB as a result of the continued moderator boil off owing to the heat transfer from the core. Following core material relocation, the remaining water in the CV eventually boils off and the moderator inside the CV is depleted at about 43,800 s.

The water in the RV acts as a heat sink to cool the external CV wall [9], and water in the RV begins to boil off at about 60,000 s resulting in a gradual decrease of the RV water level. The water level in the RV reaches the CV bottom at about 156,800 s when the CV bottom heats up rapidly and fails from a creep. Eventually, all water in the RV dries out (at about 175,000 s) and the corium then reacts

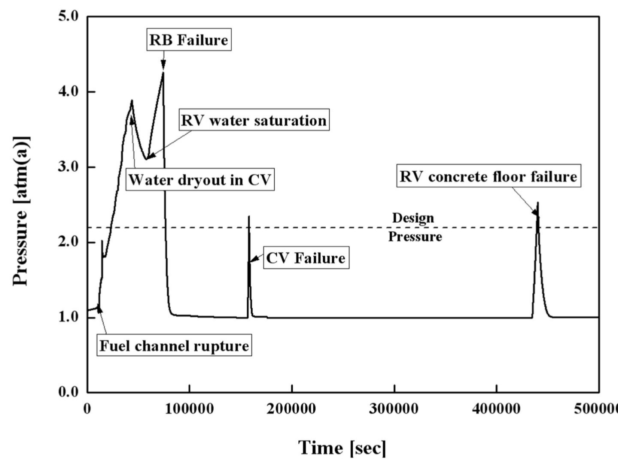

Fig. 1. Reactor Building Pressure Response for Base SBO Sequence

with the concrete floor. When the eroded depth of the concrete reaches 2 m, the concrete floor of the RV is considered to fail at about 434,000 s. Following the RV failure, the corium interacts with the water in the RB basement.

Fig. 1 shows the pressure in the RB node representing the steam generator room. After accident initiation, the RB pressure increases gradually because water is discharged into the RB through the PHTS LRVs. The rapid increase (or decrease) of RB pressure, at approximate times of 14,300 s, 43,800 s, 60,000 s, 156,800 s, and 434,000 s, can be explained through the following processes occurring at these respective times: (1) rupture of the fuel channel, (2) water dryout in the CV, (3) RV water saturation, (4) CV failure and corium relocation into the RV, and (5) corium relocation into the RB basement after RV failure. As mentioned above, at about 60,000 s, water in the RV reaches the saturation temperature and begins to boil off, thus gradually increasing the RB pressure. At about 74,300 s (which is between the time of RV water saturation and water dryout in the RV), the RB pressure reaches the failure set point of 426 kPa(a), resulting in an RB failure.

4. RISK ASSESSMENT (OF VENTING STRATEGY)

A severe core damage accident in PHWR [10] is an accident in which substantial damage is done to the reactor core, whether or not serious off-site consequences occur. These accidents have a very low frequency and result in a loss of the geometry of the core, for which a risk assessment is made for the Wolsong-specific scenarios selected based on the Wolsong PSA results [11].

Once the PHTS has been voided (through a break or boil-off), further gradual pressurization of the RB will result from the generation of steam in the CV (as the first stage) and RV (as the second stage), or from the generation of non-condensable gases from the interaction of the molten core material with the concrete basemat of the RV (as the third stage). This pressurization process can last from a few hours to several days, depending upon the stages (for example, the first stage is shorter than the latter two stages) and the effectiveness of the engineered safety features. The LACs are effective in condensing steam produced by decay heat emanating from the debris, and therefore their failure must be assumed to consider an RB failure from steam pressurization. With debris contained within the CV, a functioning ESC (end shield cooling) removes the decay heat from the debris to avoid the RB pressurization (second stage). These gradual RB pressurizations from steam production are implicitly considered in the Wolsong PSA.

Fig. 2 shows the RB failure frequencies according to the failure timings in WS-1 for event sequences in which severe core damage has occurred. Among the severe core damage sequences (=100%), 35% of the frequencies result in no RB failure, but the rest lead to an RB failure, which can then be grouped into three different failure timings (i.e., early/late/very late). The most dominant mode of RB failure is a late RB failure (56.15%). A late RB failure (the term "late" is used here to represent a relatively late timing compared with the first containment failure timing of the PWR) is defined as a failure of the RB before CV failure. Therefore, the time span of "late" in this context is from about 20 hours into the accident. A late RB failure can result from a slow over pressurization process owing to water vaporization in the RV, energetic late hydrogen combustion before CV failure, or an in-CV steam explosion (an alpha mode failure). As shown in Table 1, a late RB failure occurs mainly owing to the slow steam pressurization after RV water saturates until the CV fails. The chance of an alpha mode failure or energetic late hydrogen combustion is estimated to be negligible. If the long-term RB heat removal (i.e., LACs) is lost and not recovered, which is the case for most contributing sequences (more than 99.7%), the RB should fail before the CV failure. In the mean time, the remaining failure timing such as an early failure (which is defined as a failure from an isolation failure or RB bypass event such as an SGTR (Steam Generator Tube Rupture) [12] or V-sequence that occurs typically before one day into the accident) or very late RB failure (which is defined as the failure of the RB at or after CV failure) is not important in terms of steam over pressurization mode, as shown in Table 1.

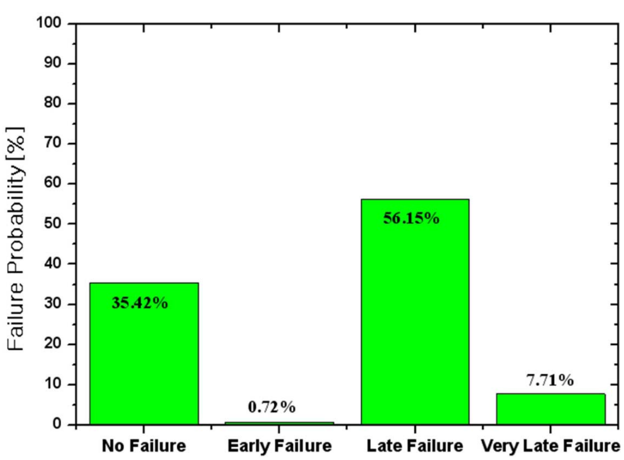

Fig. 2. RB Failure Timing and Frequency in WS-1 NPP

As the CFVS is intended to protect the containment against a loss of integrity in the event of an internal over pressurization owing to a slow steam pressurization, about a 56% failure frequency among the total RB failure frequency (65%) is expected to be removed. This means that the probability of no RB failure will increase from 35% to about 90% even after severe core damage occurs in WS-1, if the venting strategy is successful.

5. EVALUATION OF CFVS PERFORMANCE

The peak RB pressure in the base sequence with no RB failure assumed (‘no-RB-Fail’) is calculated to be about 910 kPa(a), which is caused by rapid steam generation from the molten corium interaction with the coolant in the RV just after a CV rupture. As this value is more than double the failure pressure (i.e., 426 kPa(a)) and is about quadruple the design pressure (i.e., 224 kPa(a)), CFVS operation is highly needed.

Table 1. RB Failure Mode and Frequency According to Failure Timings in WS-1

Failure Timing	Failure mode	Probability [%]
No Failure	Corium arrested in CV	27.15
	No RB failure	8.27
	Sub Total	35.42
Early Failure	Isolation Failure	0.35
	SGTR	0.33
	V-sequence	0.04
	Sub Total	0.72
Late Failure	Steam overpressurization	56.13
	Alpha mode failure	0.02
	Sub Total	56.15
Very Late Failure	Steam overpressurization	0.08
	Alpha mode failure	0.07
	Hydrogen explosion	7.56
	Sub Total	7.71
Total		100

To find a CFVS sizing criteria, a sensitivity study was made for different venting areas ranging from 0.01 m^2 to 0.2 m^2, and the characteristics of the RB pressure decrease were analyzed. As shown in Fig. 3, it takes two hours for the RB pressure to decrease from the failure pressure to atmospheric pressure in the case of 0.1 m^2, and from the design pressure to atmospheric pressure in the case of 0.075 m^2. Using the definition of failure size in the Wolsong PSA [11] in which a rupture takes two hours (at the utmost) for the RB atmosphere (corresponding to a net free volume) to be fully released, three different sizes, i.e., a rupture (0.1 m^2), boundary (0.075 m^2), and leak (0.01 m^2), are defined in this study.

Next, the CFVS operation is assumed to be made between the actuation and closure pressures to see whether the RB integrity can be maintained. To evaluate the effects of the pressure decrease through the CFVS operation, the RB pressure is calculated with the three different CFVS sizes defined above. The results are shown in Table 2 and Fig. 4, and the characteristics of RB pressure are compared with the case of no CFVS operation (SBO-noCFV) in which

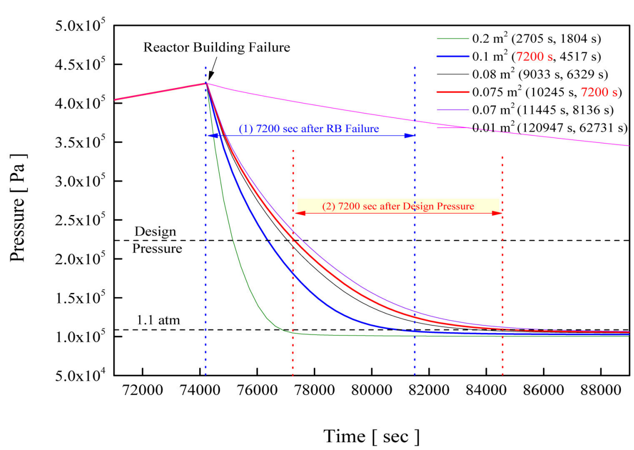

Fig. 3. RB Pressure Response to Various Venting Sizes in WS-1

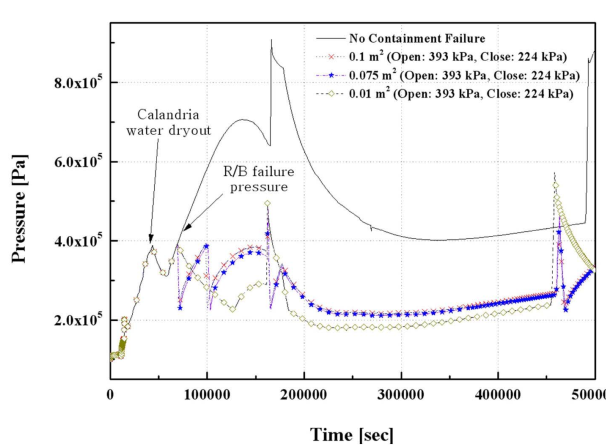

Fig. 4. RB Pressure Response for CFVS Operation

no RB failure ('no-RB-fail') is additionally assumed to show the pressure difference more clearly. In the CFVS operating scenarios, the CFVS should have enough capacity such that the RB pressure can be kept below the failure pressure. The actuation pressure is generally chosen between the design and failure pressures while the closure pressure is chosen between the atmospheric and actuation pressures. The pressures of 224 kPa(a) and 150 kPa(a) are chosen as the actuation and closure pressures in the WS-1, for which optimum operation is provided through an operating manual [13]. In this study, a higher actuation pressure is chosen for bounding calculations of CFVS performance and is set to a 90% value (393 kPa(a)) of the failure pressure using a 10% operational margin. As this actuation pressure is higher than the peak pressure (389 kPa(a)) that appeared at the timing of the water dryout in the CV (12 hours), the first cycle of CFVS operation is delayed until about 20 hours into the accident (refer to Fig. 1). In the mean time, two closure pressures, i.e., 110% of the atmospheric pressure (110 kPa(a)) and design pressure (224 kPa(a)), were selected to see the difference in mitigative control of the RB pressure.

As a result, the RB pressure is controlled between the CFVS actuation and closure pressures during most accident periods, but two peak spikes appeared to be uncontrolled. The first peak occurs at the time of CV failure when corium is relocated into the RV water. The second peak occurs at the time of RV (concrete floor) penetration failure when corium is relocated into the basement water. Both peaks are from rapid steam pressurization resulting from molten corium-water interaction. As noted from the second peak data in Table 2, if the vent area is smaller or the closure pressure is higher, the peak pressure becomes higher, which makes the CFVS less successful. The value of the first peak is inconsistent from the point of the vent area and closure pressure because the operation history before CV failure is inconsistent. For example, the CFVS operation before the timing of the first peak is made once (i.e., single venting cycle) in a '110-rupture' case, and twice (i.e., double venting cycle) in a '224-rupture' case. Furthermore, the value of the first peak in a '110-leak' case is the smallest because the CFVS is not closed and is still operating (owing to the small vent size and low closure pressure) at the start of the peak, which is contrary to the other cases. The highest peak pressure can appear in any of two peaks depending on the cases, as shown in the bold letters in Table 2. Throughout this sensitivity study, RB pressure is successfully controlled below the failure pressure in only a '224-rupture' case, which is judged to be the optimal case among the six calculated cases.

The CFVS shall be implemented as a severe accident mitigation system used for the purpose of the prevention of a containment failure from over pressurization through the controlled filtered venting of the containment following an accident with severe core damage. For this purpose, the CFVS should be designed to cope with harsh environmental conditions including high temperatures and high residual heat of the filtered aerosols. The CFVS also operates by passing the vented non-condensable gases and vapors from the containment atmosphere to remove high activity isotopes and aerosols to contain or control the radioactive releases. In Table 3, the requirements for CFVS performance are briefly analyzed from the viewpoint of thermal hydraulics where special notice is taken for parameters such as the incoming gas temperature, incoming steam and hydrogen masses, and the amount and decay heat of the incoming fission products.

- The maximum temperature of incoming gas into the CFVS is about 450–480K. The maximum temperature appears from the two peak spikes mentioned previously, and the first peak temperature is higher than the second peak temperature, as shown in Fig. 5 (for ‘SBO-noCFV’ case with ‘no RB Fail’). If the PAR does not operate (‘No-PAR’), the temperature can be decreased by a maximum of about 10ºC.

- In Fig. 6, the accumulated mass of hydrogen generated until the end of the calculation (500,000 s) is compared

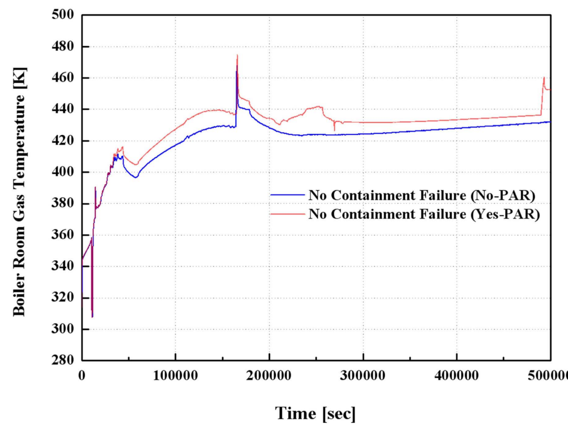

Fig. 5. Temperature Behavior of Gas Inside RB (‘SBO-noCFV’ Case with ‘no RB Fail’)

Table 2. RB Pressure Response to Various CFVS Operation Conditions

Event Sequence	CFVS Vent Area [m^2]	CFV Operation Pressure [KPa(a)]		RB Peak Pressure (>43,800 s) [KPa(a)]	
		Actuation	Closure	1st Peak	2nd Peak
SBO-noCFV with ‘no-RB-Fail’	0	N/A	N/A	**190**	880
110-rupture	0.1	393	110	**440**	415
110-boundary	0.075	393	110	**470**	440
110-leak	0.01	393	110	390	**510**
224-rupture	0.1	393	224	425	**425**
224-boundary	0.075	393	224	**480**	475
224-leak	0.01	393	224	495	**575**

Table 3. Thermal Hydraulic Requirements for CFVS Performance in SBO Sequences

Event Sequence	CFVS Performance Requirement				
	Max. temperature of incoming gas into CFVS [K]	Accumulated mass until 500K sec [Ton]		Total amount of CsI vented until 500Ksec [%]	Max. decay heat of FPs into CFVS [kW]
		H_2	Steam		
SBO-noCFV with ‘no RB Fail’	(475-480)	N/A			
Base-rupture	461	3.5-2.85	800	2.65	381
Base-boundary	462		795	2.65	379
110-rupture	451		555	1.66	370
110-boundary	457		575	1.58	378
224-rupture	452		415	1.26	303
224-boundary	455		425	1.24	306

with the released mass from the RB failure, the size of which is 0.1 m^2 (base-rupture), 0.075 m^2 (base-boundary), and 0.01 m^2 (base-leak), for the base sequence. As most hydrogen (=noncondensible gas) generated is expected to be released until the end of the calculation regardless of the CFVS operation strategy, the vent characteristics of hydrogen are not analyzed. Instead, the release is initiated at an RB failure pressure of 426 KPa(a) and is not closed thereafter. For a conservative analysis, no automatic ignition of hydrogen inside the RB is assumed. All (about 3.5 tons) of the hydrogen generated inside the RB is released until 500,000 seconds. The release response, which depends on the hydrogen amount accumulated inside the RB and the pressure release characteristics, is almost the same for the rupture and boundary sizes, but is somewhat delayed for the leak size. About 0.5 tons less hydrogen is released when the PAR is operated (Yes-PAR) while removing the hydrogen, compared with the case of no PAR operation (No-PAR). However, no more hydrogen can be removed by the PARs after about 100,000 s when the oxygen is depleted inside the RB. This shows that the CFVS should have the capability to treat that amount of hydrogen in the SBO base sequence.

- In Fig. 7, the accumulated mass of steam released after an RB failure at 426 KPa(a) for the base sequence until the end of calculation is compared with the vented mass from the CFVS operation. The CFVS operation is made for the cases ('110-rupture', '110-boundary', '110-leak', '224-rupture', '224-boundary', and '224-leak') defined in Table 2. About 800 tons of maximum steam mass is released if RB failed with the rupture (‘Base-rupture’) or boundary (‘Base- boundary’) sizes when the CFVS is not operated. If the CFVS operates under the presumed conditions, about 415-575 tons of maximum steam mass is vented depending on both the interval time of the vent operation and the vent size. This shows that the CFVS should have the capability to treat that amount of steam in the vent cases presumed.
- In Fig. 8, the decay heat of the fission products (FPs) after RB failure at 426 KPa(a) for the base sequence is compared with that of the FPs vented from the CFVS operation. The CFVS operation is made for the same six cases shown in Fig. 7. The total amount of CsI (as a representative FP) vented until 500,000 seconds is shown in Table 3, which is approximately several percent of the initial inventory. Approximately 380 kW of maximum decay heat is generated by the FPs released if RB failed with the rupture (‘Base-rupture’) or boundary (‘Base- boundary’) sizes when the CFVS is not operated. If the CFVS operates under the presumed conditions, about 300-380 kW of maximum decay heat is generated by the FPs vented depending on both the interval time of the vent operation and the vent size. This shows that the CFVS should have the capability to treat that amount of decay heat in the vent cases presumed.

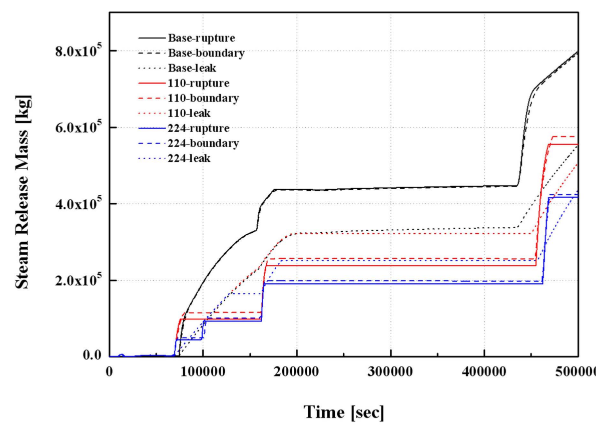

Fig. 7. Accumulated Mass of Steam Vented from CFVS Operation

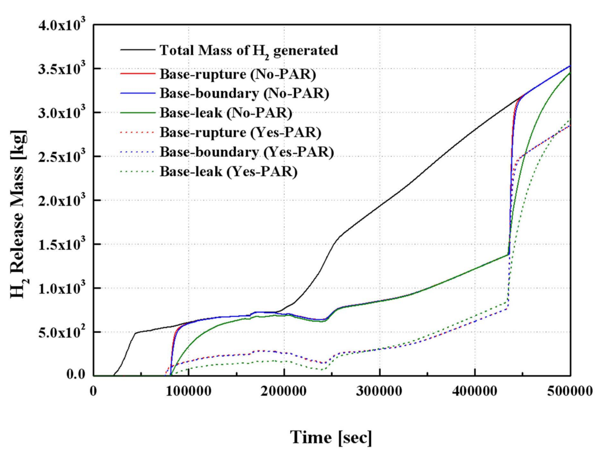

Fig. 6. Accumulated Mass of Hydrogen Released from RB Failure

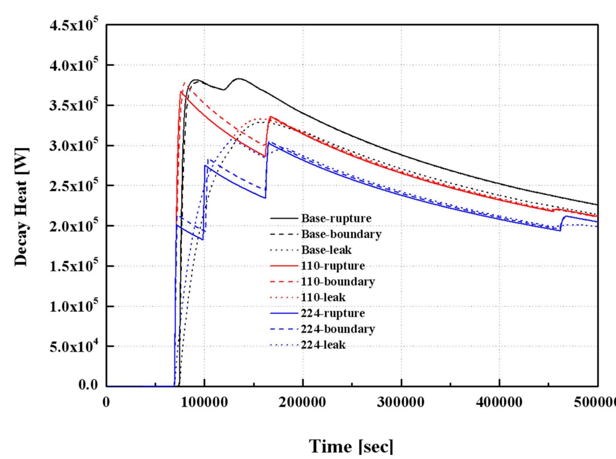

Fig. 8. Decay Heat of Fission Products Vented from CFVS Operation

6. CONCLUSIONS

Accidents involving severe core damage might result in higher pressures than those of the DBA and challenge the integrity of the containment. As one of the post-Fukushima action considerations, setting up a CFVS for an existing nuclear power plant has been proposed for mitigating the critical results of severe accidents. In particular, a PHWR has been evaluated as vulnerable to steam over pressurization, and the first CFVS was addressed for a Korean PHWR plant (Wolsong unit-1). Considering the risk assessments, an overpressure failure of the containment from steam and noncondensible gases is the principal contributor to the risk, and establishing both a venting of the containment and a removal of the aerosol can greatly reduce the risk.

The main conclusions can be summarized as follows:

- The CFVS shall have the capacity to keep the containment pressure below the design pressure during the mission time (which is at least three months after the scram in Wolsong-1). In addition, the CFVS shall have the capacity to treat or remove the hydrogen, steam, and FP decay heat vented from the RB, such that no performance degradation should occur during an autonomous operation time (This autarky time for autonomous operation in Wolsong-1 is at least three days (for intermittent cycling operation) or one and a half days (for continuous operation) into the accident). This study evaluated the CFVS capability requirements that the containment integrity be maintained in a severe core damage scenario of an SBO. This study also suggests how the system actuation and closure pressures should be determined to ensure adequate protection from an over pressure of the containment.
- Steam over pressurization is a significant threat to the CANDU RB integrity if no recovery action can be taken for a long period of time, as in the Fukushima incident. As the CFVS is introduced to Wolsong-1 as a mitigation strategy to deal with the threat of over pressurization (particularly slow steam pressurization), the probability of no RB failure is predicted to increase from 35% to about 90% even after severe core damage occurs in Wolsong-1, if the venting strategy is successful.

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