PHTS Overpressure Protection Capability Weakness under Severe Accident in Wolsong

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1. Introduction

In a light water reactor, once thermosyphoning breaks down upon the depletion of water inventory in the steam generators, the reactor coolant system re-pressurizes. Over-pressure protection on a pressurized water reactor is composed of a number of complementing systems including the reactor shutdown systems and pressure and inventory control systems. Among these, passive, spring loaded, self-closing safety relief valves (SRVs) are the most important components of the pressure control systems, especially upon a reactor trip. In Wolsong, an uncontrolled over-pressurization of the Primary Heat Transport System (PHTS) will likely occur during high pressure severe accident conditions [1][2], which can increase the probability for unplanned and uncontrolled pressure boundary ruptures in fuel channels or steam generator tubes. This results from an insufficient capacity to relieve decay heat equivalent to steam through the engineered safety relief valves called degasser condenser tank relief valves (DCRVs), as shown in Fig.1, which are separated from the PHTS by another series of SRVs called liquid relief valves (LRVs) in their discharge path. These PHTS over-pressure behavior is analyzed using MAAP(1)-ISAAC code [3][4].

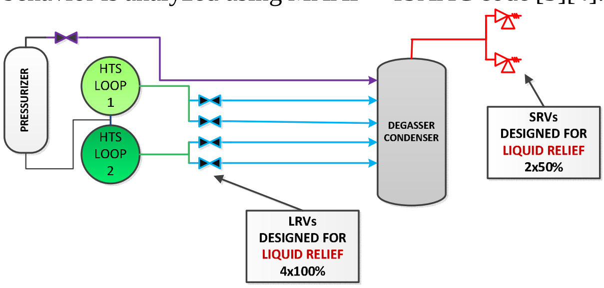

Fig. 1 Typical PHWR SRV configurations

2. CANDU Safety Relief Valve Design Review

In CANDU-6, the HTS over-pressure protection is currently provided through two pairs of LRVs that join the outlet headers from the two loops at one end of the reactor to the degasser condenser tank. These valves form the pressure boundary under normal operating conditions but are not the ultimate pressure relief valves. With outlet headers at 10 MPa, these LRVs (Fig.1) are designed to open at 10.24 MPa and allow a discharge of about 25.2 kg/s of liquid D_2O. Their relief set point is properly kept lower than the 10% over-pressure protection required by the ASME and CSA [5] standards for operational transients of relatively high frequency. These LRVs discharge into a tank called the Degasser or Bleed condenser which normally operates at pressures that are close to atmospheric pressure. The two DCRVs are then the ultimate over-pressure protection devices for the PHTS and are effective only when the upstream LRVs open and pressurize the degasser condenser tank. These DCRVs are also sized, tested and qualified for liquid relief with a discharge capacity of about 26.7 kg/s of liquid D_2O at 10.057 MPa and 268°C. The set point is only slightly higher than the normal outlet header pressure to avoid a loss of HTS inventory through a spurious opening of the upstream LRVs. A schematic of the proportional spring loaded LRVs is shown in Fig.1.

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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.

3. Estimates of steam relief capacity of DCRVs

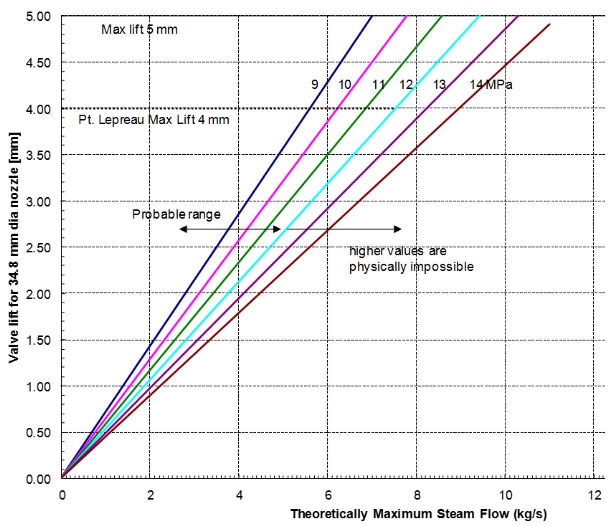

Fig. 2 Theoretical maximum flows through DCRVs

The Bopp & Reuther valves installed in CANDU-6 reactors have a throat diameter of 34.8 mm and a maximum lift of 4 or 5 mm, depending upon the reactor unit. Valves lifted fully for water tests as designed but the steam tests resulted in a lower lift as expected for a maximum lift of about 0.8 to 1.5 mm. Fig.2 shows the maximum theoretical steam flow for Bopp and Reuther valves [6]. It is seen that for up to 2 mm lift, the theoretically maximum steam discharge for one valve is about 3.5 kg/s at 11 MPa.

4. Review of ASME Acceptance Criteria

A review of ASME acceptance criteria (Section III, NCA 2142) and its application to reactors designed and licensed in the US and Europe reveals that the SRVs are sized such that the over-pressurization imposed on the pressure boundary does not exceed the limits imposed by service Level B (meaning no deformations, no damage requiring repair, which is typically an over-pressure of 10%) for almost all foreseeable events, including those later potentially leading to core damage. If stress levels beyond service Level C (meaning deformations in areas of structural discontinuity requiring removal for inspection or repair, which is typically an over-pressure of 20% to 30% but depends upon each component and on seismic loading) is imposed, it could potentially result in an un-repairable nuclear reactor from the gross deformations with loss of dimensional stability. Given the frequency at which off-site power has been lost to operating nuclear reactors and the past occurrences of simultaneous loss or effectiveness of on-site back up power (at TMI and Fukushima) leading to a loss of backup feed water to the boilers, the event of total loss of feed water is expected to have a quite probability of occurrence [7].

5. Sensitivity Study for SBO Using MAAP-ISAAC

In a high pressure sequence such as station black out (SBO) following a failure of boilers as heat sinks [8], ECCS injection is prohibited from PHTS high pressure. Under these circumstances, extensive fuel damage is resulted in but channel integrity is assumed to be assured by the moderator acting as a heat sink. This accident has been classified by some in the industry as a ‘limited’ core damage (LCD) accident. In a Wolsong LCD accident, the PHTS over-pressure is protected if the relief capacity of SRVs exceed that required from surge or steam generated by decay heat. On the contrary, smaller SRV capacity creates PHTS over-pressure while the PHTS coolant inventory decrease rate is slowed down and then results in a delay in core damage such as first timings of a fuel uncover or a fuel channel failure. In the meantime, if an overpressure does continue, the failure by rupture of a component of the HTS is inevitable. If the rupture is in the boiler tubes, the consequences are not limited to within the reactor containment, which means larger direct release of fission products into the environment.

In this paper, a sensitivity study for SBO LCD accident using MAAP-ISAAC (4.03) code is made for two cases (Cases SBO-A and SBO-AA) corresponding to DCRV areas of 43 and 4.75 cm^2 as shown in the following table. (The DCRV area of Case SBO-A is larger by 9 times compared with Case SBO-AA.)

Sensitivity Variable	Case	
	SBO-A	SBO-AA
DCT-RV valve area [m^2]	4.3E-3	4.75E-4

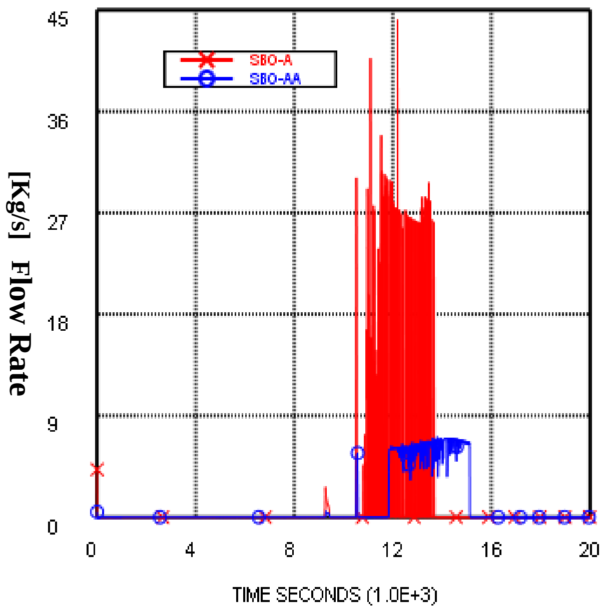

Fig. 3 Total steam flow rate through 2 DCRVs [kg/s]

The total steam flow rate through 2 DCRVs is about 27 and 7 kg/s, respectively for Cases SBO-A and SBO-AA (see Fig.3). After steam generators dry out, the PHTS starts to release steam through SRVs. The PHTS pressure becomes beyond normal one (10 MPa) in Case AA due to a smaller steam relief rate compared with a steam generation rate while no over-pressurization occurs in Case A due to enough steam relief capacity (see Fig.4).

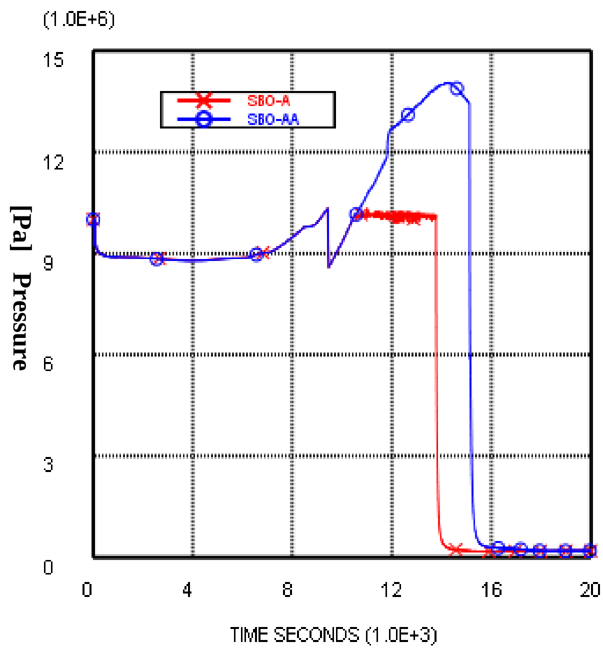

Fig. 4 PHTS Pressure [Pa]

The accident progression timing is shown in Table 1. In Case AA, the over-pressure of 36% (△P=3.6 MPa) is shown at a time of the core uncover start and reaches the peak value of 40% after several minutes at 14,250 sec. The first fuel channel failure occurs by creep rupture about 30 minutes after core uncover start. The defined

SAMG (Severe Accident Management Guidance) entry point in Wolsong is the timing of core exit gas temperature of 650°C in any fuel channel, which exists between a core uncovery start (=13,345 sec or 3.7 hours) and a first fuel channel failure (=15,915 sec or 4.4 hours).

Event Progression [sec]	SBO-A	SBO-AA
Rx trip (loop isolation failed)	0	0
Steam Generator dry out (10% of initial inventory)	7,040	7,040
first LRV open (fail open after first opening)	9430	9430
Core uncovery start (water level in any channel < normal level)	12,420	13,345
first fuel channel failure (by creep rupture)	13,777	15,120
first core empty (loop stagnant + core (L1 or L2) water mass < 100kg)	14,689	15,915

Tab. 1 Accident Progression Timing

At the time of core uncovery (3.7 hours) in Case AA, the over-pressure of 36% appeared in PHTS corresponds to a service Level D load that potentially results in an un-repairable nuclear reactor. This means that plant operators or emergency staffs have less time ($\triangle t \sim 0.7$ hours) to do some recovery actions for PHTS boundary protection. If a service Level B load corresponding to 10% over-pressure is used for recovery due time (11,248 sec or 3.1 hours), another 16% lesser time ($\triangle t \sim 0.6$ hours) is available. On the contrary, if 9 times larger DCRV is used like in Case A, the core damage occurs slightly earlier ($\triangle t \sim 0.3$ hours) due to easier coolant release, but no over-pressurization is appeared which means less concern for the boiler tube rupture. In the analysis of Case A (see Fig.5), PHTS starts to release the water and steam after LRVs open, and the steam total flow rate becomes about 27 kg/s through 4 LRVs (2 LRVs in each loop of L1 and L2) between the times of a core uncovery start (3.7 hours) and a first fuel channel failure (4.4 hours). This means that steam flow rate of max. 27 kg/s is required to remove the decay heat of about 20 MW_{th} (that corresponds to decay power around 4 hours after a reactor trip) without PHTS over-pressurization in Wolsong. But the current 2 DCRVs have smaller steam release capacity which is estimated at ~7 kg/s (corresponding to energy equivalence of about 5 MW). This is significantly lower than decay heat of 20 MW to be removed.

As a result, one stated design basis for the subject valves is ‘loss of inventory control and loss of heat sink’ and the stated design objective for a sustained loss of heat sinks should be met. This can be accomplished by adjusting the steam release capability difference to the same value for two series of valves (4 LRVs and 2 DCRVs). Simple estimation is to increase the capability of DCRVs 4 times larger than current one.

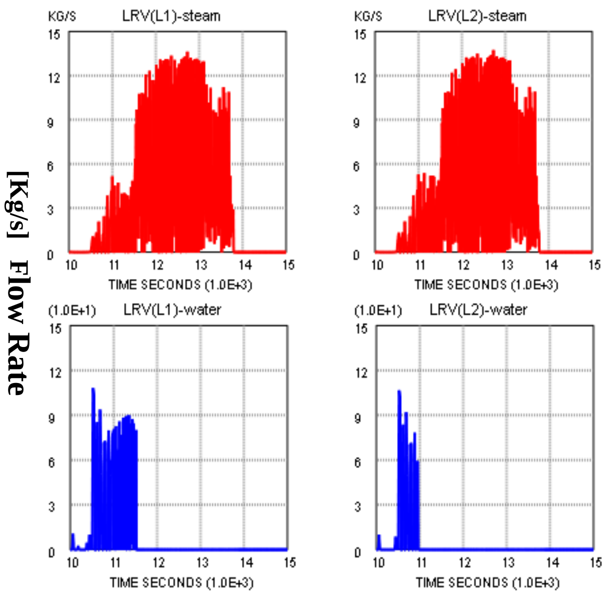

Fig. 5 Loop1(L1)/2(L2) steam/water flow rate through 4 LRVs in Case SBO-A [kg/s]

6. Summary

The over-pressure protection system in the PHTS for Wolsong CANDU-6 reactors needs to be upgraded to include an ability to act as an effective heat sink upon a sustained loss of engineered heat sinks. The relief valves are effective heat sinks when their discharge can remove the liquid swell or the steam produced by decay heat transferred to the stagnant liquid from intact fuel channels. The relief capacity must also exceed that required for design basis transients and accidents enumerated in the station over-pressure protection document.

The over-pressure protection in a typical CANDU reactor is provided through 2 separate series of twin valves (LRVs and DCRVs) and is not a direct and unobstructed pathway for energy release from the main heat transport system as stipulated for Class 1 systems in ASME section III NB 7141(b). Therefore, under certain accident conditions when energy relief must become the sole or dominant heat sink, the steam and surge relief must occur through two series of valves (4 LRVs and 2 DCRVs), separated by long piping and the degasser or bleed condenser tank. The upstream LRVs are larger, less resistive valves with typically double the flow capacity of that by the DCRVs which is needed to increase their steam relief capacity in order not to limit the discharge flow by themselves.

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