THERMAL HYDRAULIC MODELING FOR TRANSIENTS FOLLOWING A BREAK IN STEAM OR FEED LINE OF THE 700 MWE PHWR

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ABSTRACT

A computer simulation model has been developed for the upcoming 700MWe Pressurised Heavy Water Reactor (PHWR) using internationally renowned, best estimate RELAP5/MOD3.2 code for the thermal hydraulic behavior and the corresponding impact on the structural design assessment of Steam Generators (SGs). The output from these simulation studies will be utilized for verification of the structural design of the SGs. The 700 MWe PHWR SGs differs from the earlier SGs in some of the design modifications. The postulated scenarios considered for the steam line/feed line breaks and the corresponding inputs may affect the outcome of the thermal hydraulics & the structural analysis. The most sensitive parameters include the time and signal for the reactor trip, Main Steam Isolation Valve (MSIV) closure, break size and crash cool down using Atmospheric Steam Discharge Valve (ASDVs). These parameters have been judicially selected by the designers. The expert opinion on this matter may differ among the designer, regulators & the safety analysis groups. An attempt is made to the implications of the possible options. Failure of steam or feed water piping has no direct radiological consequences in the PHWR plants. However, the SG internals have to be been analyzed for postulated double ended guillotine break in steam or feed water piping to preclude dual failures. During a hypothetical break in feed line header, the feed water distributor is subjected to a high-pressure load for a very short duration. The nodalisation strategy for this component was also established. To ensure the structure integrity during such scenario, a transient analysis to estimate the various dynamic loads has been carried out. This analysis is also an important part of licensing requirement. The SG model can capture transient details such as, SG internals Differential Pressures (DP), this included the, steam separators, dryers, distribution plates, feed sparger, and DP across the shroud for all the riser and down comer volumes. Provisions were also made to extract frictional drag coefficients for all the SG control volumes. The output is being generated for use as input for structure analysis to verify the mechanical design. All the baffles in the riser region were also simulated. The modeling of the flow distribution plate at the bottom of the riser was also incorporated. The nodalisation also accounts for the variation in DPs across all the important internal components. The paper describes in detail the methodology adopted and the results obtained for steam-line and feed-line breaks.

INTRODUCTION TO 700MWe PRESSURISED HEAVYWATER REACTOR (PHWR)

In the 700MWe PHWR, the primary coolant heavy water under pressure removes (with partial boiling at channel exit) the fission heat generated in the reactor core and transfers it to the secondary coolant (light water) in the steam generators (SGs). The primary heat transport (PHT) system consists of 392 fuel channels. The PHT system is divided into two identical loops. Each loop consists of two primary circulating pumps (PCPs) and two SGs in a figure of eight loop configuration as shown in Figure 1. There are two passes through the core for each loop. As the primary coolant flows over the fuel bundles placed inside the channels, it picks up the fission heat in four passes through the reactor core. In each pass, 98 channels are connected to a common header at each end of the reactor. After picking up heat from the reactor core, the coolant flows through the reactor outlet header (ROH) into the tube side of the SGs. After transferring heat in the SGs, the primary coolant is pumped (by the primary circulating pumps (PCPs)). back to the reactor core through reactor inlet header (RIH). The SG provides the thermal linkage between the PHT system and the secondary coolant system. The SGs deployed in the 700MWe reactors are of the inverted U-tube type with integral drum. The secondary fluid flows in the shell side, and the hot primary coolant from the reactor core (ROH) flows inside the U tubes. The total boiler feed is given at the top of the downcomer. The recirculation flow from the steam drum, aftermixing with the feed water, flows down the annulus (downcomer). Then, it rises up through the main boiling zone, as it picks up the heat. After extracting the heat, the secondary fluid (steam-water mixture) rises through the riser and then passes through the steam separator and dryers. Here, the two-phase mixture gets separated into saturated water and steam. The former is led downwards, after mixing with the feed water, to the annular downcomer of the SG, while the latter goes to the steam outlet and then to the turbine. The average of the two ROH pressures is controlled around a set point of 101.0 kg/cm2, to avoid excessive boiling and over-pressurisation in the PHT system. In the 700MWe PHWR, for controlling the PHT system pressure, a pressuriser (surge tank) is also provided along with the Feed/Bleed system for maintaining the coolant inventory. The feed/bleed system is provided for controlling the water level in the pressuriser. Steam bleed valves (SBVs) are provided on top of the pressuriser vapor volume to control the increase in the PHT system pressure, by relieving the heavy water steam into the bleed condenser (BC) through the PHT system pressure controller. Electrical heaters are provided to take care of the low-pressure transient, by switching on the heater banks to increase the pressuriser pressure.

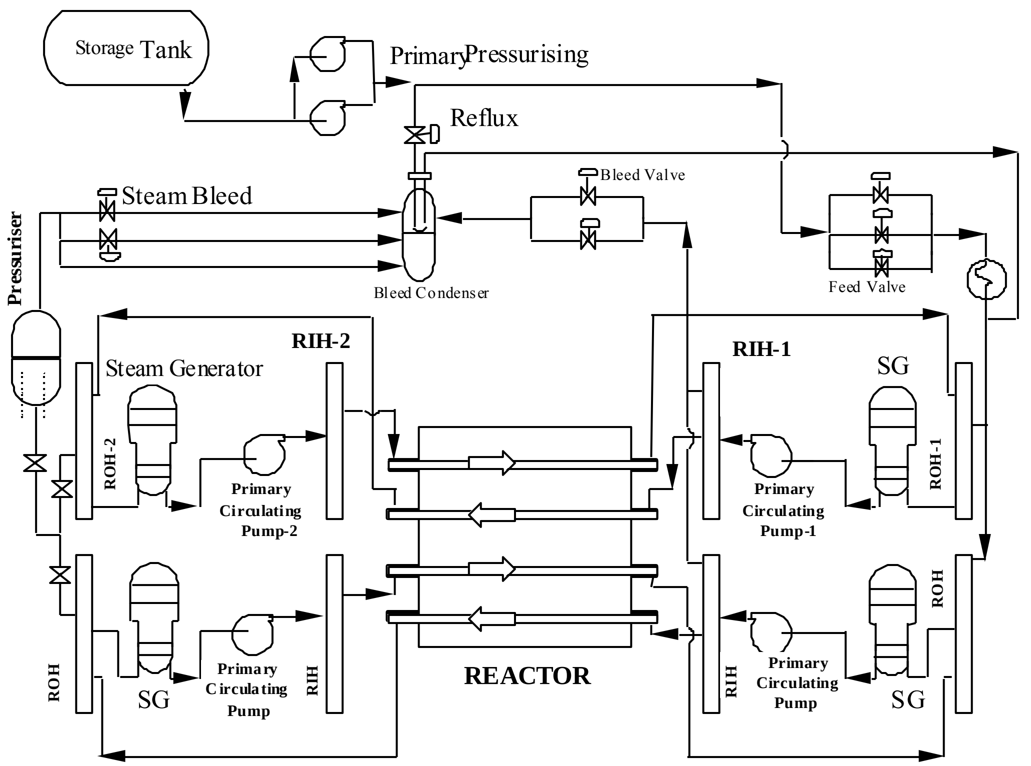

Fig. 1: 700 MWe PHWR primary heat transport system

The hot bleed from the RIH, and the relief from the south ROH flashes into a two-phase mixture inside the BC. The BC pressure is controlled at 34 Kg/cm2. In the event of an increase in the PHT systempressure, the SBVs also start relieving heavy water steam into the BC. At 100% full power steady state, the reactor core inlet temperature is 266 C, the core outlet temperature at the ROH is 310 C, and the reactor core exit quality is around 3% only.

IMPERATIVE ISSUES RELATED STEAM LINE/FEED LINE BREAK ANALYSIS:

The literature review shows scanty reported work on the steam line break and feed line break for the PHWRs. This paper emphasizes more on the efficiency of the separator on the transient scenarios [1]. The current studies show that this issue does not affect the results. The full model of the nuclear power plant along with the Primary Heat Transport system and the all the associated control systems including SG level control has been used for the current studies. Steam line break is one of the important PIEs to be analyzed for all the nuclear power plants, for the containment design and load evaluation in SG internals. Feed line break analysis also partly fulfils the objective mentioned above. The process dynamics analysis and the structural issues are closely related to each other and appropriate PIE scenarios for the safety analysis are not standardized but are rather evolving. The contagious issues are related to minimizing the load on the containment. The loads on the internals of the SG can be tackled by modifying the design after locating the critical areas if any discrepancies are observed. To reduce the energy discharge during such PIEs nozzles sizes are manipulated, which acts as a critical flow discharge limiter. The nozzle sized for the steam out let has a net flow area of 0.04 m2. The nozzle flow area for the feed line has been limited to two nozzles of area 0.04 m2. The values correspond to the optimized design of the steam and feed line which does not hamper the required maximum steady sate flow and erosion limits requirements.

A reactor trip signal is obtained for the containment high pressure for the SDS-1 and SDS-2. Based on the, two group theory, only one of the signals can be ignored; both the signal cannot be ignored. For the present studies reactor trip was considered at 1.2 seconds, which includes all the sensing delays. The other signals which may appear over the time are low steam line pressure, low PHT system pressure, low SG level, high steam flow, MSIV closure. If the reactor trip is taken with delay then the energy discharge will go up to enhance the containment pressure, which is not desired.

Feed flow during the postulated scenarios may also have variation based on the scenarios envisaged. A simple analysis may assume that the feed flow comes down quickly proportional to reactor power following the reactor trip which may not simulate the actual scenario. The feed flow is governed by the individual SG level controller and the embedded logics. As the broken SG pressure reduces the SG pressure reduces this enhances the pressure difference for feed injection. This enhanced feed is only restricted by the flow limiter based on the power drawn by the feed pump. This will restrict the feed flow to a value between 1.5 to 2.0 times the normal flows depending on the design. This aspect was also included in the modeling by appropriate modeling.

The MSIV on the individual SG steam lines are closed on sensing high containment pressure. It takes 120 seconds to close fully. This action blocks the flow of steam from the unbroken SGs to go out from the break, which reduces the energy discharge and load on the containment. The results obtained shows that the higher closing time of the MSIV does not affect the initial phase of the transient though it eventually reduces the discharge energy. The design MSIV closure value has been selected to avoid steam hammer phenomenon in the steam lines. ASDVs and the SG steam relief valves as located on the steam lines running outside the containment building. To reduce the discharge inside the containment these valves are thrown open on sensing high containment pressure signal.

RESULTS AND DISCUSSION

To study the effect of various important parameters discussed in the above section six cases have been analyzed:

1. Steam line break with reactor trip on containment signal.
2. Steam line break with reactor trip on containment signal with MSIV closure.
3. Steam line break with reactor trip on containment signal with MSIV closure and crash cool down.
4. Feed line break with reactor trip on containment signal.
5. Feed line break with reactor trip on containment signal with MSIV closure.
6. Feed line break with reactor trip on containment signal with MSIV closure and crash cool down.

Double ended guillotine rupture has been modeled to represent break flow from both side of the broken pipe. Fig 3 shows all the four SG pressure response for the first case where steam line break with reactor trip t = 1.2 s) on containment signal. For this case no MSIV closure was considered. The opening of ASDVs to minimize the energy discharge was also not considered for this case.

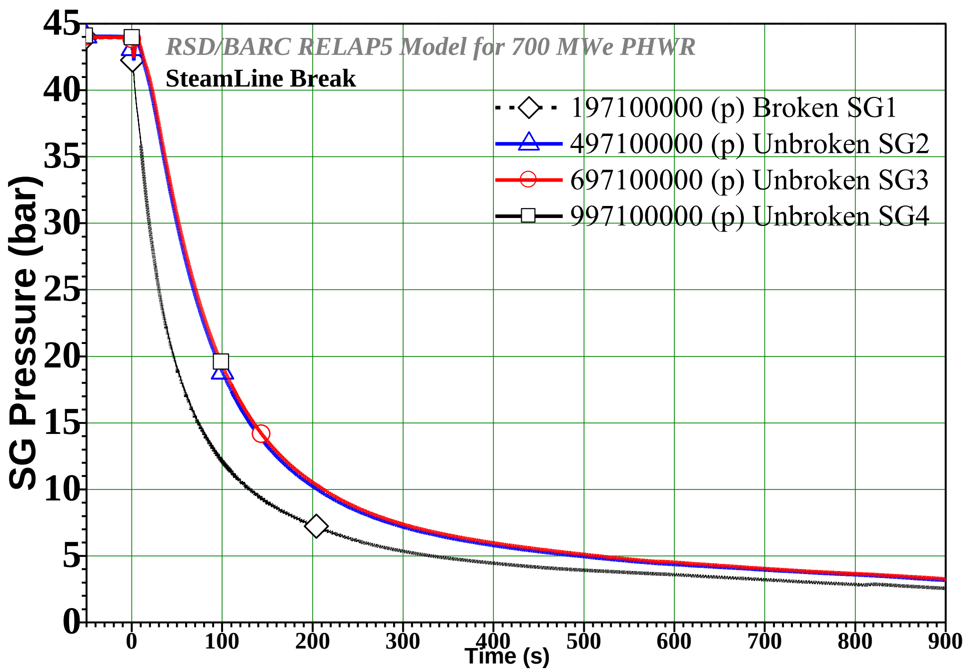

Fig. 3 SG pressure for steam line break case-1

At about 100 s, it is observed that the pressure in the broken SG pressure comes down to 12.22bar, where as for all other unbroken SGs it comes down to approximately to 18.85bar from the initial value of 44bar.

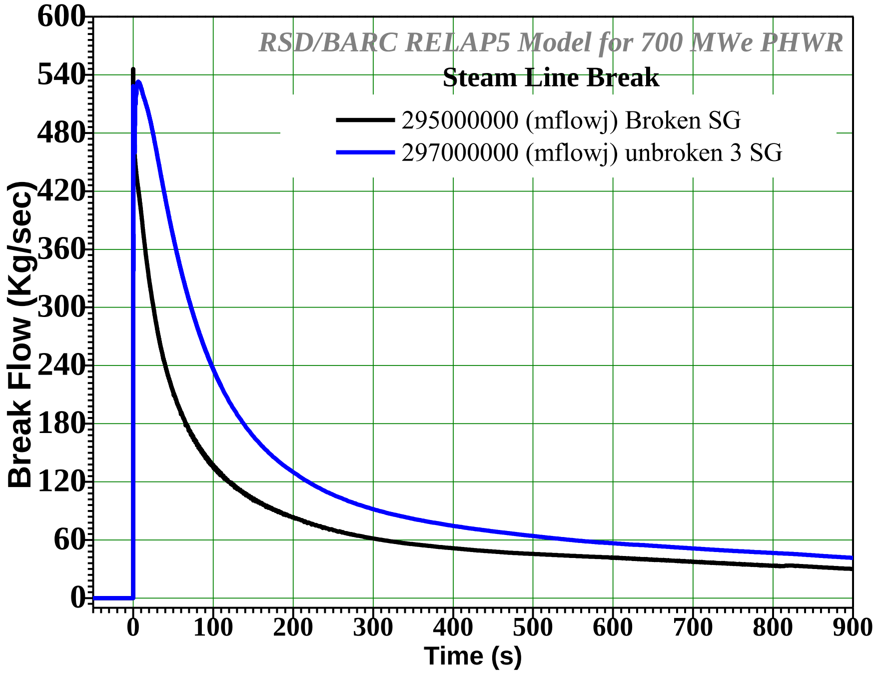

Fig. 4 Break flow for steam line break case-1

The maximum break flow is restricted to about 550kg/sec from both the broken ends. As the unbroken SGs depressurize at a slower rate the break flow from the unbroken 3 SGs shows a higher value throughout. Fig 5 shows the variation of the steam drum levels. It is observed that broken SG level depletes faster as compared to others. Fig 6 shows the PHT system pressure for the case-1. Initially the PHT pressure shows a declining trend, and then it shows a temporary recovery, due to PHT pressure control action. It is seen that in the later part of the transient the PHT system goes down due to over cooling.

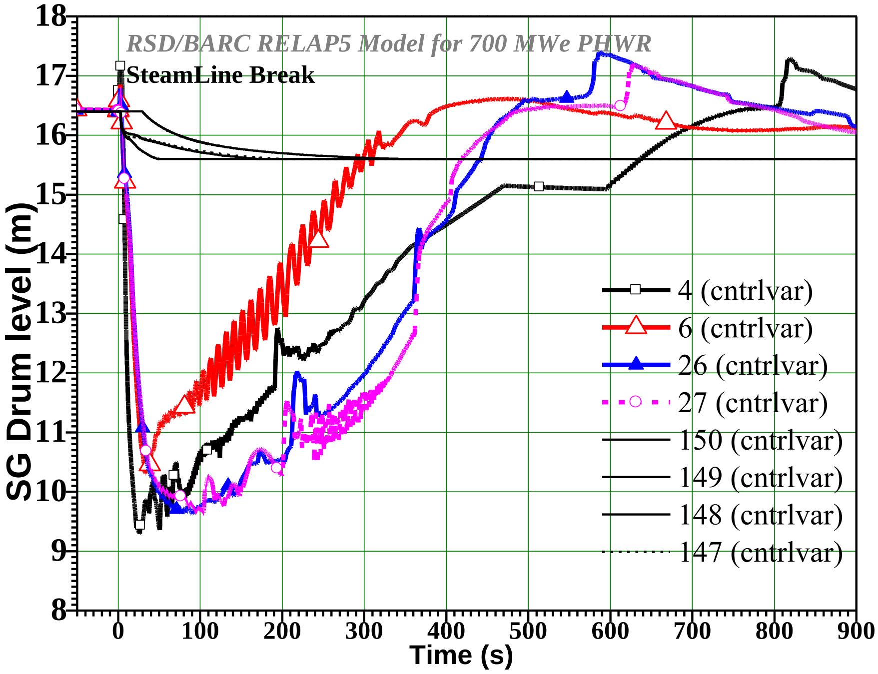

Fig. 5 Steam Drum Levels steam line break case-1

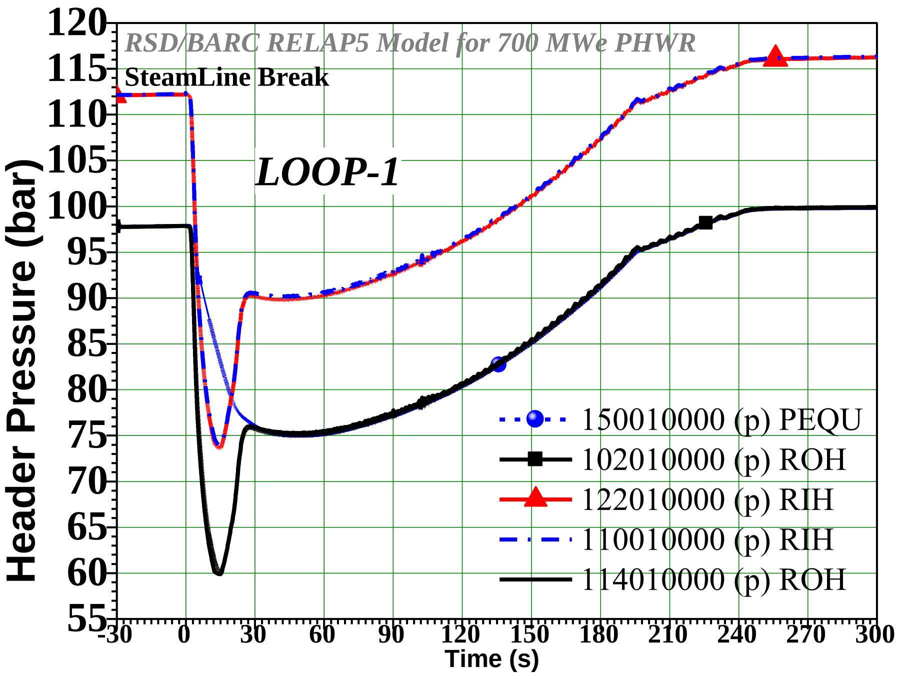

Fig 6 PHT system pressure for case-1

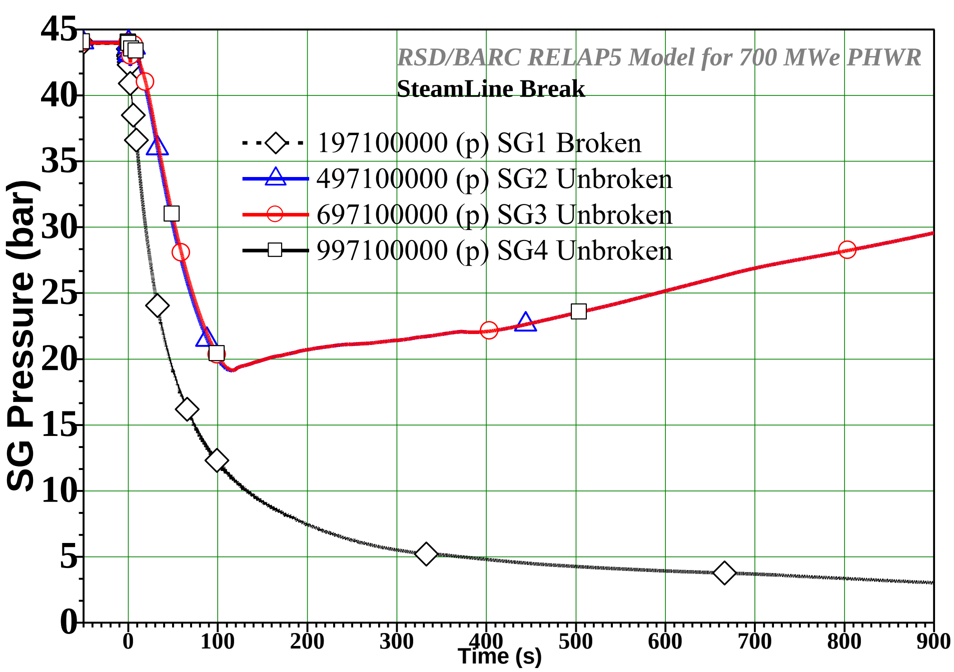

Fig. 7 SG pressure steam line break case-2 with MSIV closure

For case-2 MSIV closure in 120 seconds was simulated on sensing high containment pressure. After t= 121 seconds the break flow from the unbroken 3 SG comes down to zero and the pressure in these SG starts building up (fig 7). It is observed that the break flow from the unbroken 3 SGs reduces due to closure of the MSIV and SG pressure starts increasing beyond 121 seconds. This does not result in any change in SG pressure and break flow (fig 7, 8) for the affected SG. The unbroken 3 SG pressure remains at 20.31bar at t=100 seconds (as compared to 18.81bar for case-1) and the broken SG pressure at this point is same as case-1 i.e. 12.22bar.

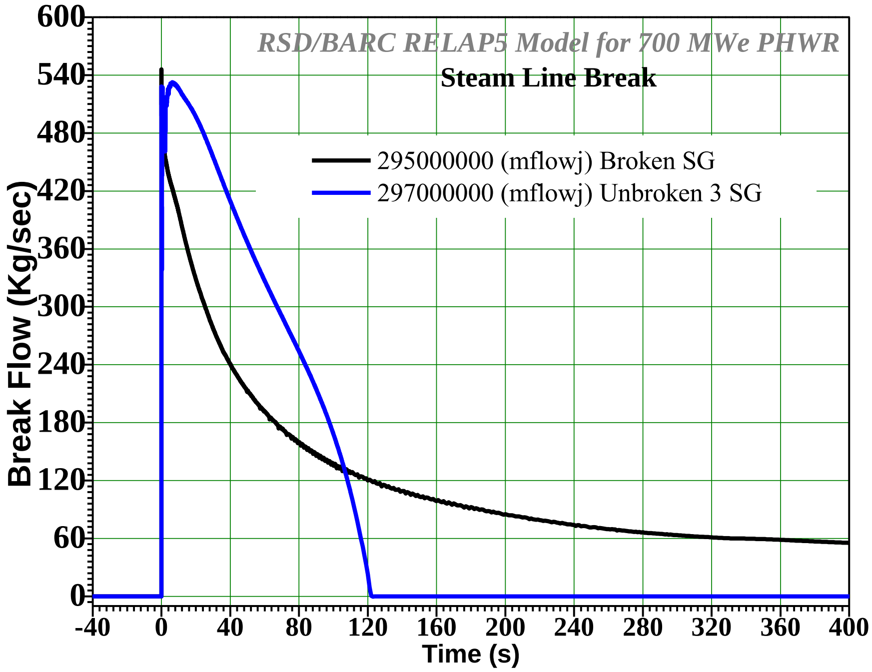

Fig. 8 Break flow steam line break case-2 with MSIV closure

Case-3 simulates MSIV closure and crash cool down following a steam line break. It is observed that the SG depressurization is highest for all the SGs. At t=100 second the broken SG pressure comes down to 10.02bar (fig 9), whereas for case-1 & 2, this value stands at 12.2bar.

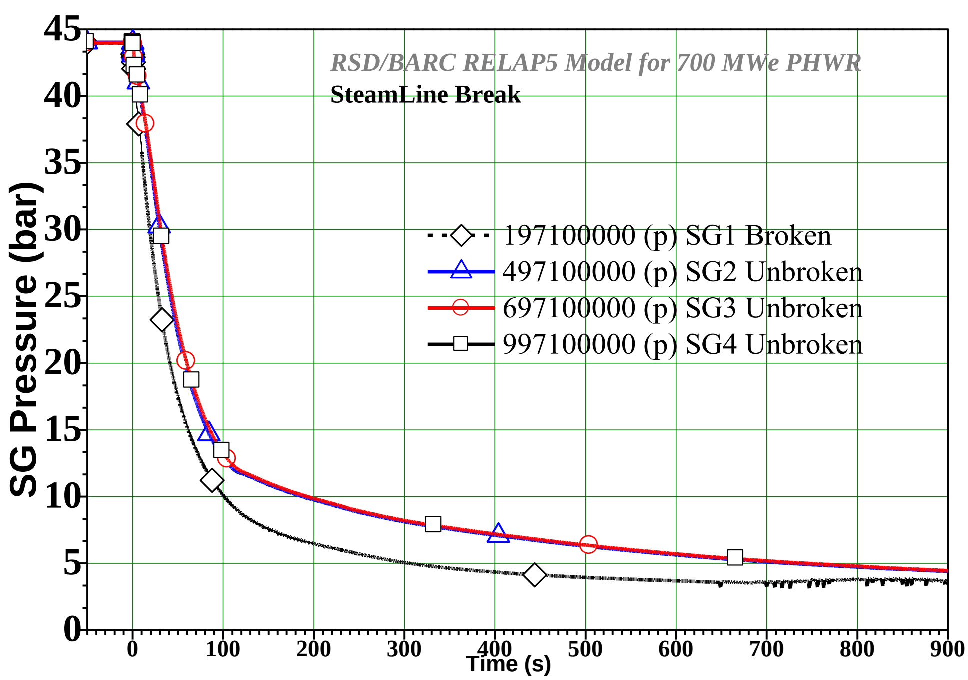

Fig. 9 SG pressure steam line break case-3 with MSIV closure & crash cool down.

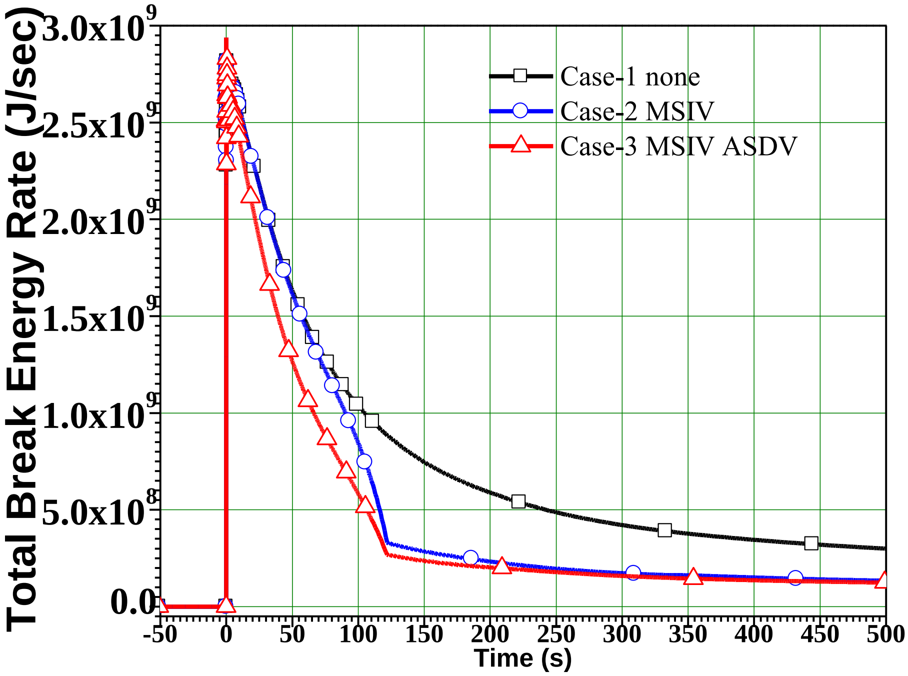

Fig 10 Total Energy Discharge Rate to containment for Steam Line Break all Cases

Fig 10 shows that the initial peak energy discharge is unaffected by the MSIV and ASDV actions. Also it can be seen that the total energy discharge rate for the third case is minimum which fulfils the objective minimizing the containment energy discharge.

FEED LINE BREAK

The feed line has been modeled with two nozzle penetrations in the SG. The break has been simulated as a guillotine rupture in one of the two parallel feed line paths. Fig.11 depicts the transient pressure response following the break. The subcooled blowdown from the SG results in higher depressurization as compared to steam line break. Fig. 12 shows the break flow from both ends of the broken line.

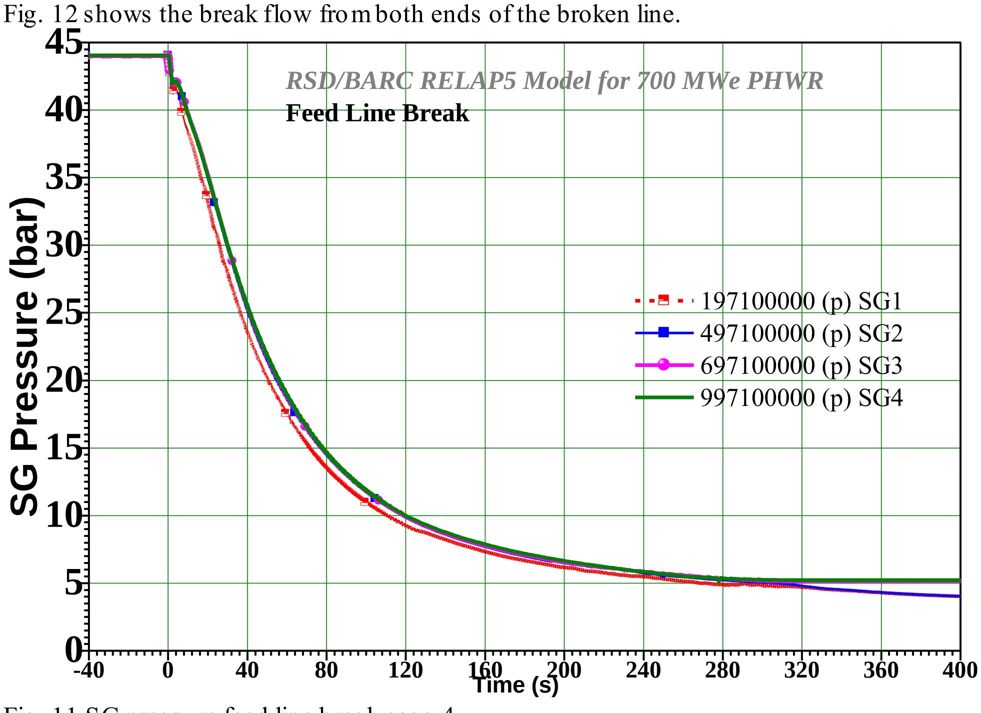

Fig. 11 SG pressure feed line break case-4

As seen from the fig.12, the break flow from the feed line header side shows a large number of oscillations till t=113 s due to mixing of the reverse flow from the SG with the intact feed nozzle line and the forward flow from the BFP side which join together before going through the break from the BFP side.

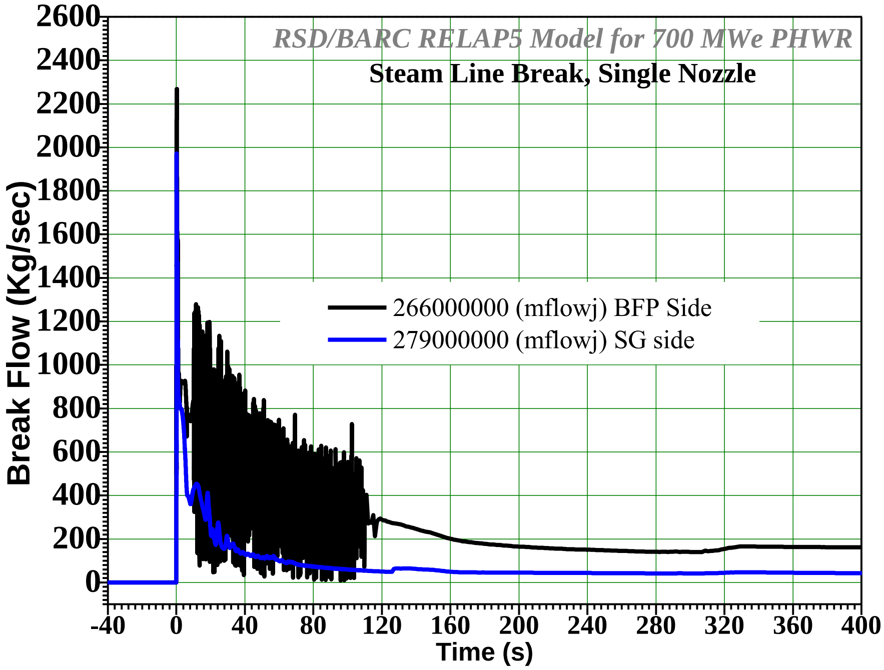

Fig. 12 Break flow feed line break case-4

Fig.13 shows the water level variation in all the 4 SGs. The level in the SG-1 with broken feed line dips to a minimum value of about 4 m from the initial value of 16.4 m whereas for other 3 SG level recovers back to their original value with the level controller action.

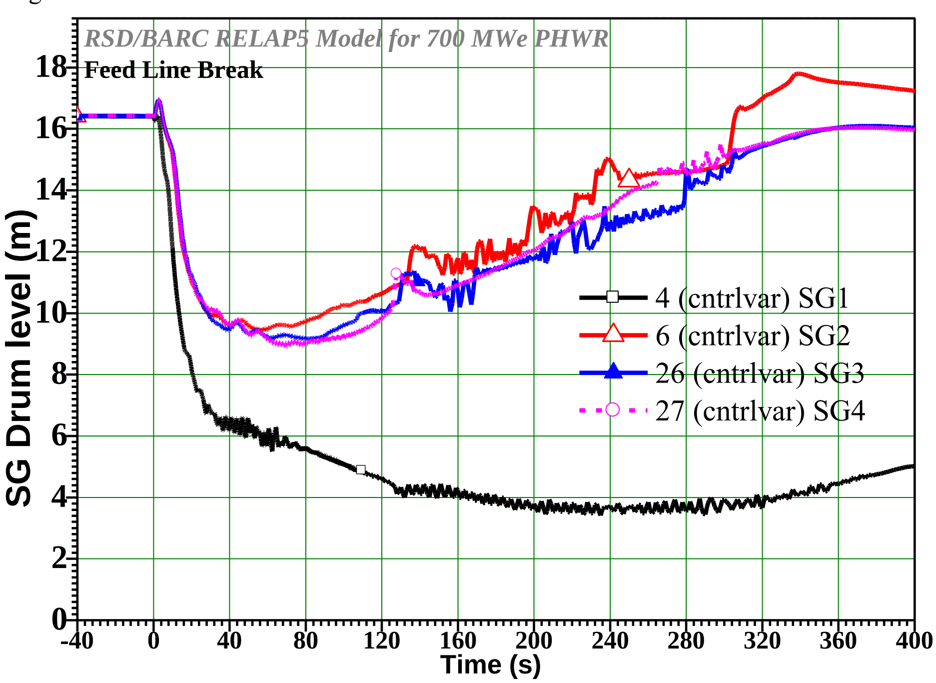

Fig. 13 SG water level feed line break case-4

For case-5, the MSIV closure on high containment signal has been considered. In this case as the steam out flow from broken SG is blocked due to MSIV closure, its pressure remains at a higher value as compared to case-1. This results in higher and longer duration break flow.

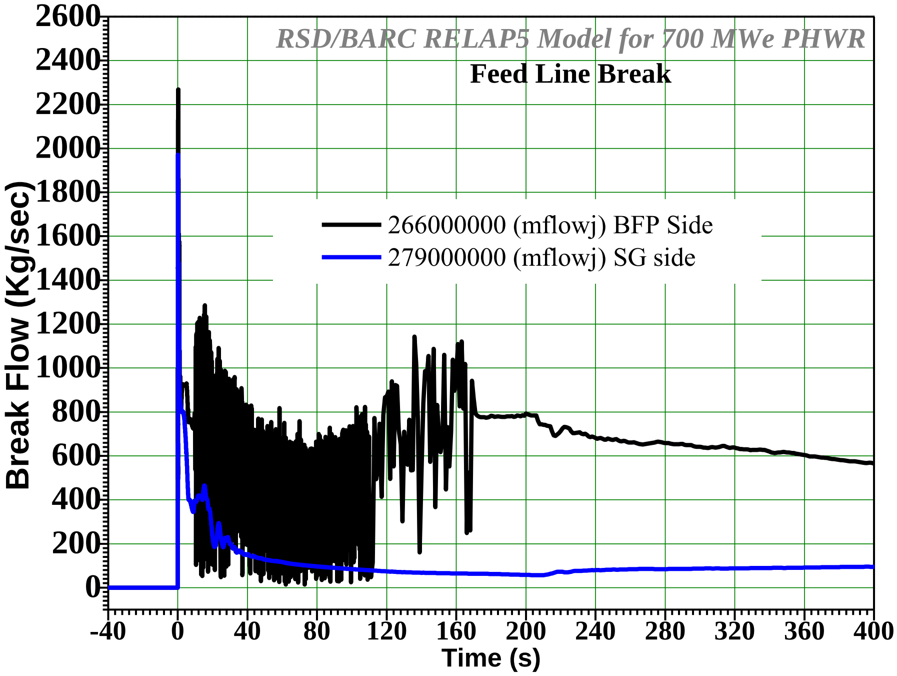

Fig. 14 Break flow feed line break case-5

The steam outlet flows from all 4 SGs are shown in the Fig.15 for case-6 with crash cool down case. The outlet flow from the SG-1 reduces to zero value with the closure of MSIV.

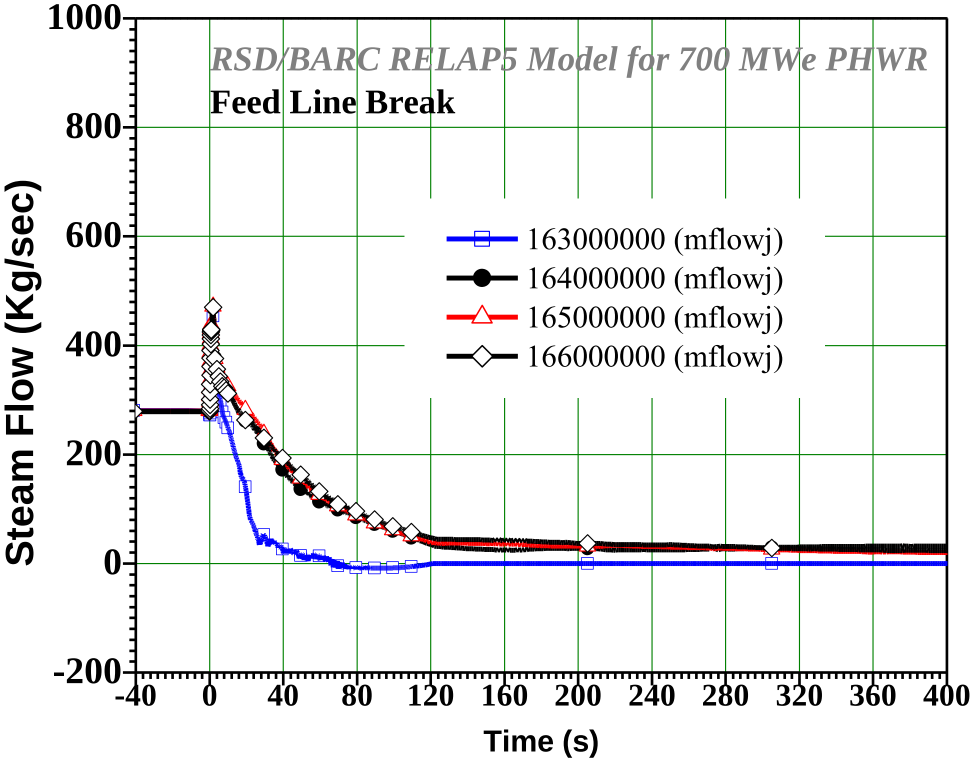

Fig. 15 Steam outlet flow feed line break case-6

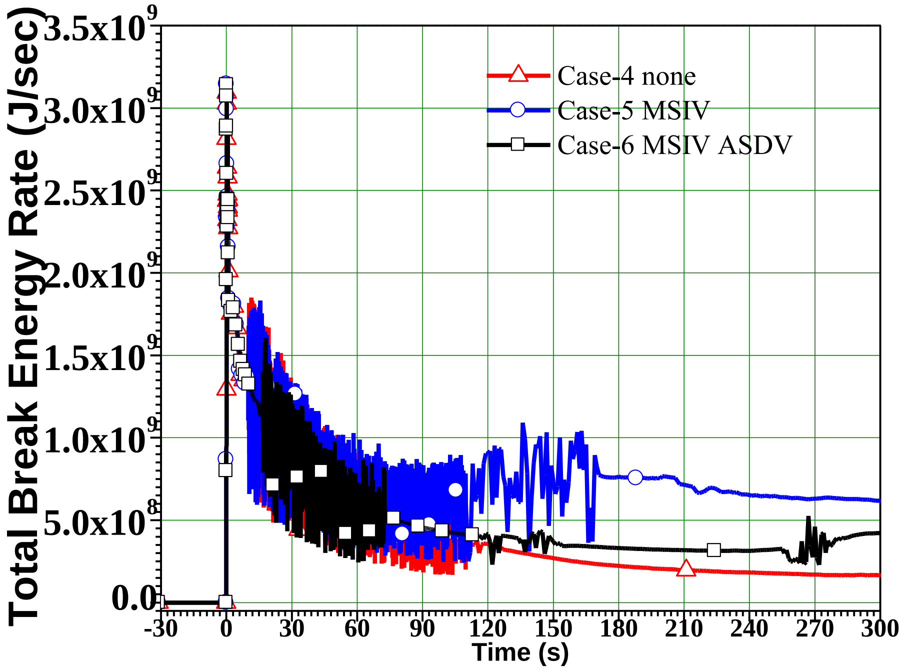

Fig. 16 Total energy discharge to containment, for feed line break all cases-4 to 6

Fig 16 shows that the energy discharge is minimum for the last case. The energy discharge is higher for the 5th case with only MSIV closure. As the ASDVs do not open in this case the break flow remain high. The initial energy discharge to the containment is slightly more that the peak energy discharge, for the, steam line break case.

CONCLUDING REMARKS

All the setting and parameters such as nozzle size, MSIV closure and ASDV opening, affect the outcome of a steam/feed line break were parameterized and their effect on the outcome has been brought out. The design and analysis efforts for steam/feed line breaks are dedicated in minimizing the load on the containment. Though the loads on the SG internals are also important, but they can be tackled by modifying the design after locating the critical areas if any discrepancies are observed. It is seen that many designers rely on the pre-assumed variation of feed flow based on the reactor power. For this study feed water variation during the transient is based on the response of the SG level controller. The values of the feed flow rate encountered in this case simulate the practical scenario without biased assumption. Reactor trip on the containment signal is derived for both shutdown system one and shut down system two as both the trips cannot be ignored the analysis was carried out by taking credit of this trip. This can be justified based on the independent and diversified design based on two-group theory for shutdown systems. It is also noted that energy discharge to the containment can be reduced by closing the MSIVs and the crash cool down through the ASDVs. The steam drum level does not go beyond the separator height for any of the cases analyzed, so the issues related to separator efficiency degradation and enhanced two-phase discharge to the containment are ruled out. The initial peak energy discharge is unaffected by the MSIV and ASDV actions for the steam line break cases. The total energy discharge rate to containment for the third case is minimum which fulfils the objective minimizing the containment energy discharge. The energy discharge is minimum for the 6th case for feed line break. The energy discharge is higher for the 5th case with only MSIV closure, as the ASDVs do not open in this case the break flow remains high. The initial energy discharge to the containment is slightly more that the peak energy discharge, for the, steam line break case.

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