Analysis of small break loss of coolant accident for Chinese CPR1000

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1. Introduction

After Fukushima accident in 2011, the source-term estimation for nuclear power plants of neighboring countries in case of accidents has been drawn attention. This research analyses the small break loss of coolant accident (LOCA) on a Chinese CPR1000 type reactor. LOCA accident is used as benchmark for the PCTRAN/CPR1000 code by comparing the effects and results to the Manshaan FSAR accident analysis.

2. Methods and Results

In order to demonstrate that PCTRAN/CPR1000 simulation software is capable of simulating the Chinese CPR1000 type plant operation, two kinds of runs are conducted. First is to perform normal operation with load change. Second is a loss-of-coolant accident (LOCA). Fig. 1 shows main GUI screen shot of PCTRAN/CPR1000. The user can enter a load demand different from the panel indicated and the reactor will respond to reach the desired load with a ramp rate in percent per minute. For example, by clicking at "M" for manual control for Power Demand in the upper right reactor control panel and entering 40%, the reactor will drop it power output to 40% at a rate of 10%/min. This will be achieved by precise control of the turbine control valve. The neutron flux and thermal power follow the turbine load with a noticeable lag.

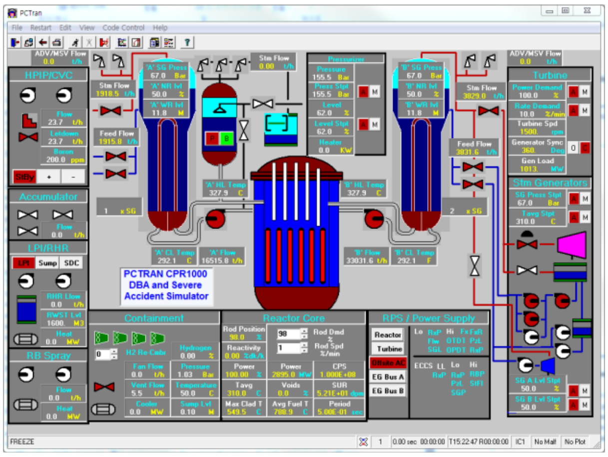

Fig. 1. PCTRAN/CPR1000 windows mimic

The reactor primary pressure after some perturbation returns to the original pressure while the secondary pressure rises to a higher value corresponding to the lower power level. The pressurizer level and reactor coolant T_{avg} will decrease according to the load program. The feedwater flow will be run-back to balance the steam flow. The steam generator narrow-range level returns to the set point of 50% after its initial rise by overfeeding. The rod reactivity is a result of the rod control system that inserts the assemblies into the core. Feedback from Doppler and moderator temperature are also presented, they are combined total reactivity controls the nuclear power in this load reduction process. (Fig. 2)

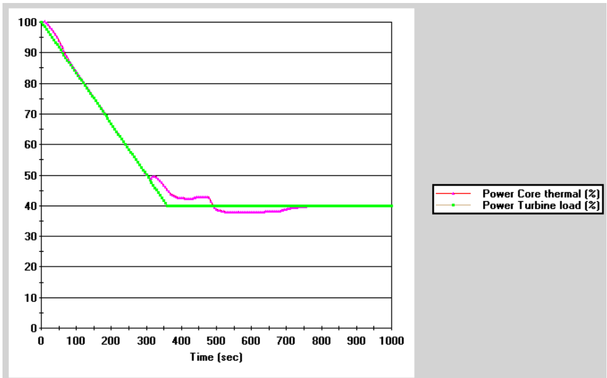

Fig. 2. Power core thermal and power turbine load (%)

LOCA is a design basis accident in which a guillotine break is postulated to occur in one of the cold legs of a pressurized water reactor (PWR). Consequently, the primary system pressure would drop and almost all the reactor coolant would be discharged into the reactor containment. The drop in pressure would activate the reactor protection system and the reactor would trip. The fission chain reaction in the fuel due to the loss of moderator coolant would be terminated. Nevertheless, the heat would continue to be released from the fuel rods by fission product radioactive decay. Subsequently, the emergency core cooling system (ECCS) must provide adequate cooling in time to minimize overheating of fuel cladding, its damage and eventual core meltdown.

Design basis LOCA in a PWR are separated roughly into three periods: (1) blowdown, in which the coolant would be expelled from the reactor vessel, (2) refill, when the emergency coolant water would begin to fill the vessel up to the bottom of the core, and (3) reflood, when the water level would arise enough to cool the core. In contrast with large breaks, the blowdown phase of the small break occurs over a longer time period. Core recovery and long term recirculation then follow a gradual blowdown. The Manshaan nuclear power plant FSAR Section 15.6.5.3 analyzed a spectrum of 3 to 5 inch small breaks. This study will compare the results of a PCTRAN/CPR1000 3-inch SBLOCA with that of the Manshaan FSAR. A small break was simulated by 3-inch (45.6-cm^2) break. The elevation of the break was set to 2 m below the top of the core and all reactor coolant pumps were tripped consistent with the Manshaan FSAR procedure.

The simulation is done for 6000 seconds for the whole accident sequence to play out. The RCP pumps are tripped simultaneously with the occurrence of the break. This immediately triggers the reactor SCRAM. Under normal circumstances the reactor would be able to make up the loss of coolant for some time without a SCRAM. It should be noted that for a break elevation of -2 m the core is uncovered at 399.5 seconds. This closely resembles the Manshaan NPP FSAR results, where the core is uncovered around 500 seconds. The core uncovered triggers the start of fuel heat-up and resulting fuel damage. Fig. 3 depicts the depressurization of the reactor coolant system. Both the simulation results and the Manshaan FSAR data can be compared directly. Both curves follow a similar trend.

Closer inspection reveals that the PCTRAN /CPR1000 code consistently has 20 bars (290 psia) of pressure bias on top of the Manshaan FSAR with the Manshaan FSAR triggering a larger high pressure injection at 100 seconds to make up for the 20 bars shortfall. Typically, such differences are to be expected as the control philosophy of various plants can differ widely. However, this can cause large differences in event sequences due to the cumulative effect of the various operations, as is evident from this case study. In this case the timing of the injection systems initiation down the line will differ by as much as 3000 seconds resulting in the fuel being uncovered for longer and thus larger releases of radioactivity.

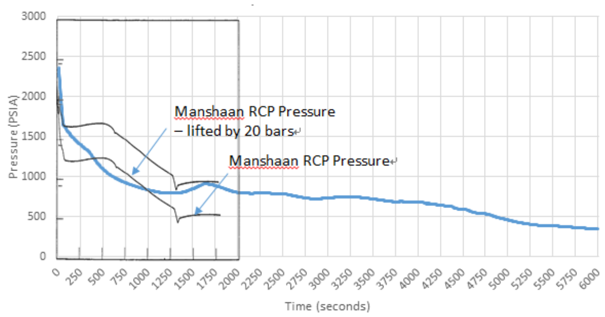

Fig. 3. Reactor coolant system pressure withManshaan NPP data superimposed on simulation results

The fuel clad temperature is directly linked to the coolant level in the reactor. In Fig. 4, it can be seen that at first the fuel temperature drops due to the SCRAM and the stop of the fission chain reaction. The coolant is still required to cool the fuel because of the fission product decay heat it produces. When the core is uncovered at 399.5 seconds, the temperature rises steeply.

The temperature is only reduced once the accumulator starts to reflood the core again at 4384 seconds. Comparing it to the Manshaan FSAR results, it is expected that a similar rise in temperature will be observed until the accumulator floods the core around 1300 seconds, then it is expected that the fuel temperature will drop back to the original water cooled temperature. Superimposing the Manshaan FSAR fuel temperature graph on top of the PCTRAN/CPR1000 simulated fuel temperature in Fig. 4 confirms the expected differences in temperature trends. It also reveals a close correlation in the code results during similar operating sequences.

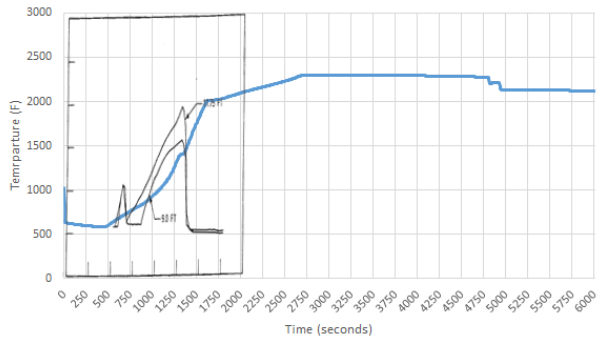

Fig. 4. Fuel clad peak temperature with Manshaan NPP data superimposed on simulation results

To analyze the emergency core cooling system operation, Fig. 5 depicts the high pressure injection initiated at 50 seconds in response to the pressure drop due to the RCS leak. Again the accumulator flow starts at 4384 seconds when pressure drops below 40 bars. The Manshaan FSAR depicts the break flow against the reactor coolant system pressure. Comparing this depiction with the PCTRAN/CPR1000 simulated values can be very effective as the flow rate is a direct function of pressure, break size and break elevation. This comparison is depicted in Fig. 6. The comparison reveals a close link with a variation in the exact break size and elevation. It does however confirm the effectiveness of PCTRAN/CPR1000 to accurately predict the effects of a break on the reactor coolant system.

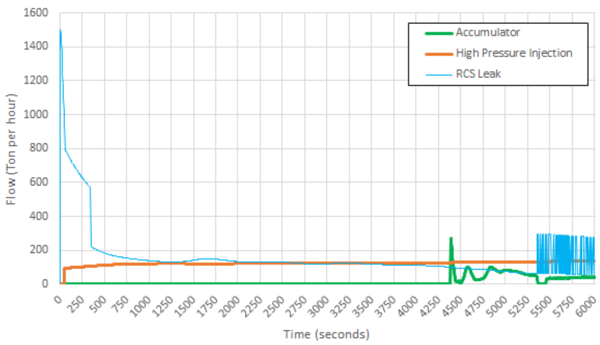

Fig. 5. High pressure and accumulator injection with RCS leak rate

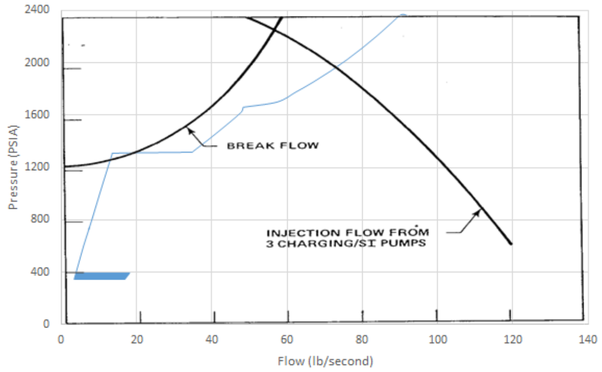

Fig. 6. Break flow against pressure of both PCTRAN/CPR1000 and Manshaan FSAR

3. Conclusions

The simulation of a 3-inch small break loss of coolant accident using the PCTRAN/CPR1000 has revealed this code’s effectiveness as well as weaknesses in specific simulation applications. The code has the ability to run at 16 times real time and produce very accurate results. The results are consistently producing the same trends as licensed codes used in Safety Assessment Reports [1, 2]. It is however able to produce these results in a fraction of the time and also provides a whole plant simulation coupling the various thermal, hydraulic, chemical and neutronic systems together with a plant specific control system. The plant specific control system is possibly code’s greatest weakness. Transfer functions and control philosophies can have a very large impact on the simulator outputs during transients as was seen in the difference in timing of the reflooding of the core. This can be solved by insuring the implementation of the PCTRAN/CPR1000 is customized to exactly copy the plant being analyzed.

ACKNOWLEDGMENTS

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