Safety analysis on steam line break accident of integral PWR

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Abstract: A system analysis of the double-ended rupture plus a 5cm break of main steam line break (MSLB) accident assumed to occur on integral PWR-200 has been conducted. The system and break model have been built using RELAP5 code. In this paper, we present the main operation parameters variation under MSLB, and compare them with Organization for Economic Cooperation and Development (OECD) benchmark problem. The results illustrate that due to SMR's core fuel assembly type and OTSG inherent features, there are significant difference compared to benchmark results. However, the simulation demonstrates IP200 is safe enough under the most severe MSLB accident.

Keyword: main steam line break accident; integral pressurized water reactor; nuclear safety analysis

1 Introduction

Nuclear safety is extremely important for nuclear application especially after the occurrence of Fukushima nuclear accident, after when people become more concerned about the risk of nuclear power [1]. One potential solution is to develop small modular reactor (SMR) with better inherent safety and system behavior under accident conditions, which will be more acceptable to the public when compared to large-scale pressurized water reactor power station. Therefore, safety analysis for SMR is one important step before constructing the real plants.

Main steam line break (MSLB) is an important design basis accident (DBA) that has a wide range of consequence. According to the accident categories made by US Nuclear Regulatory Commission (NRC), which includes operational transients, moderate frequency sequences, rare sequences and limiting accidents, MSLB could classified to the last three categories based on the severity of consequence. When evaluating the reactor safety features, MLSB has been always used as benchmark test. The committee on Safety of Nuclear Installations (CSNI) and the Nuclear Science Committee (NSC) of the OECD are both active in the field of nuclear reactor safety also promoting a variety of International Standard Problems and benchmarks. The two committees jointly proposed the PWR MSLB benchmark. The Babcock and Wilson (B&W) Three Mile Island Unit 1 (TMI-1) reactor, 2772MWt, 900MWe power, equipped with two once-through steam generators (OTSGs) has been taken as reference in the study [2-5]. In the benchmark problem, the accident is initiated by double-ended rupture in one steam line and a small break in the other line, which will be categorized as the most severe limiting accident. After when a lot of scholars from celebrated universities and institutions have devoted themselves using different analysis codes to simulate the MSLB tests and verify their codes. K. Ivanov [6] presented three exercises of the MSLB benchmark and summarized the findings of the participants with regard to the current numerical and computational issues, in addition, he reviewed the details of sensitivity study. In H.G. Joo's paper [7], he used the MSLB benchmark problem to investigate the effects of newly developed high-fidelity MARS/MASTER code, whose results agreed well with the benchmark. Other works based on system codes analyzing MSLB transient behavior have also been conducted in past few years [8-11].

Although there have been plenty of researches focus on MSLB of loop-type PWR, there lacks related insights on SMR behavior under MSLB. Due to the different system configurations, different fuel assembly makeup, there supposes to be different system performance under MSLB accidents. Therefore, in this paper, safety analysis of MSLB on integral PWR IP200 reactor [12] has been carried out. The key operation and safety parameters such as reactor power, pressure and core temperature have

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been monitored. The different system behavior and time sequence have been analyzed.

2 Simulation model

2.1 Integral PWR system description

IP200 is an integral configured SMR, whose concept design has been accomplished by researchers of Harbin Engineering University [12]. The integral compact configuration of IP200 eliminate long and large diameter pipes, hence, eliminating large-break LOCA. Therefore, better inherent safety features as well as simplified structure, light weight features have been obtained for IP200. All the main components, such as four main coolant pumps (MCPs), twelve once-through steam generators (OTSGs) and the core are located inside the reactor vessel (as shown in Fig.1). While, on the secondary side, each group of OTSGs that contains three OTSGs connects one steam line. At last, four steam lines combine into one main steam line. There is also one main feedwater pipe that has four separate feed water pipes. All the above characteristics different from loop-type PWR make IP200 behavior differently under accident conditions which we have been digging in for years. The main operation parameters can be referred from Ref. [12].

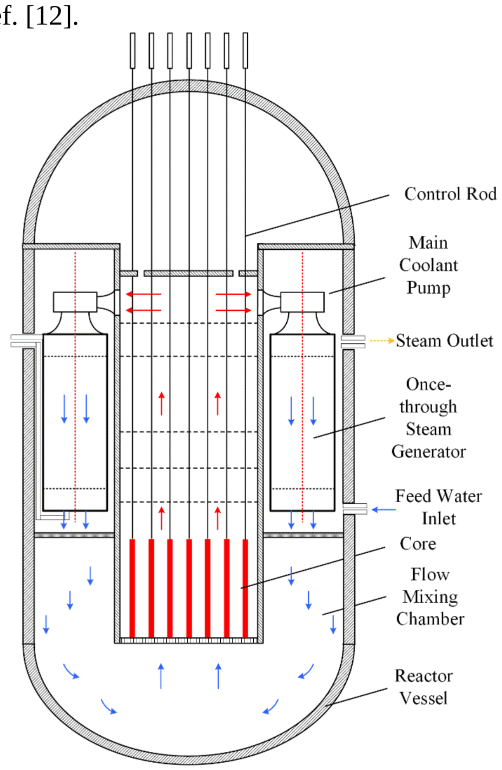


Fig.1 System layout of IP200 reactor.

2.2 RELAP5 simulation model

RELAP5 is highly reliable and widely used safety analysis code, which adopts one-dimensional, two-fluid model for flow of two-phase steam-water mixture. The governing equations for the two-fluid non equilibrium model consists of two phasic continuity equations, two phasic momentum equations, and two phasic energy equations. The whole IP200 reactor including primary system (reactor core, OTSGs, MCPs), OTSGs' secondary side have been modeled using RELAP5 code. While the feed water, steam consumption and steam line breaks have been simplified as boundary conditions. In RELAP5 code nodalization scheme (as shown in Fig.2), the pipe component control volumes 014, 016 and 018 stand for the hot, average and bypass channels of the reactor core, respectively. Pump component control volumes 111, 121, 131, 141 simulates the main coolant pumps, and the area in the red lines simulate one group of OTSGs that contains three ones. Pipe component 026 stands for the integral pressurizer which connects to safety valve component 047, relief valve component 037 and spray valve component 027 to form the high pressure protection system. Ideal steady-state programming control strategy (to keep average temperature of coolant at 577.55**K** and steam pressure at 3.0**MPa**) is adopted for the operation of IP200 [12]. However, when the accident occurs, the control system will partially be out of work.

To simulate the most catastrophic steam line break accident, double-ended rupture on steam line one and one 5cm break on steam line one has been assumed to initiate the accident. Pipe component 252 stands for main steam line, and pipe components 418, 428, 438 and 448 simulate four steam lines on the upstream of the main steam line. The RELAP5 nodalization of double-ended rupture is shown in Fig.3a. Single volume 410 and pipe component 418 stand for the steam line one. The break positions are simulated by trip valve 551 and 561. When the steam line one breaks, trip valve 412 will close immediately to simulate the double ended rupture, meanwhile the valve 551 and 561 will open to simulate the steam loss. Besides, the trip valve 521 simulates the 5cm break, while the time-dependent volume 524 is on behalf of steam loss boundary condition.

3 Results and discussions

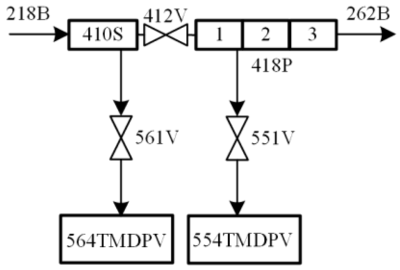
3.1 Steady state conditions

The demonstration of a thermal-hydraulic stable steady state before the initiation of the transient simulation (t=0**s** in the following session) is necessary to achieve reliable results. The initial steady state of MSLB for IP200 reactor is assumed to be at beginning of cycle (BOC), hot full power (HFP) with zero boron concentration. The reactor coolant system (RCS) pressure is at the nominal operation value of 15.5**MPa**. The initial cold leg temperature is 558**K**, and the hot leg one is 592**K**.

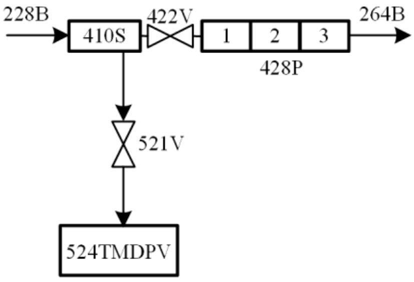
3.2 Transient Scenario

The double-ended rupture of on steam line is assumed to occur upstream of the cross-connect. The double-ended rupture and 5cm cross-connect break result in the highest break flow assumption and maximize the RCS cool-down. The worst single failure is the failure in the open position causes feedwater flow from the intact OTSG to cross over to the broken OTSG through the common header and maximize the feedwater flow to the broke OTSG. The feedwater flow is finally terminated by closure of the feedwater block valve, which is conservatively assumed to close 30 seconds after the break occurs.

Subsequent to break initiation, and following reactor trips, the steam line turbine stop valves are assumed to slam shut, isolating the intact OTSG. Reactor scram is assumed to occur when the power reaches 115% or RCS pressure reaches 13.41**MPa**. All four MCPs are modelled to operate during the event since maximizing the primary system to secondary side heat transfer will accelerate RCS cool-down in a rapid pace. The accident time sequence is listed in detail in Table 1.



(a) Double-ended rupture on steam line one.



(b) 5cm break on steam line two.
Fig.3 Break location simulation model.

Table 1 Time sequence of steam line break accident

Time	Events
0.0	Double-ended rupture on steam line one
0.0	5cm break on steam line two
4.2	Reactor trips
4.2	Turbine trips
4.7	Isolate the intact OTSGs from turbine

3.3 Accident simulation

When the steam line break occurs, the pressure of the broken OTSG drops rapidly, which will cause the flow rate within the SG to increase. The increased flow rate results in enhanced heat transfer and overcooling of the RCS fluid. The decreasing RCS temperature will introduce positive reactivity to the fission reactor, which will lead to an increase of the core power. The power rise continues under it reaches the high neutron flux set point (115% full power), the reactor trips by inserting control rods into the core (as shown in Fig.4). IP200 reactor power runs faster to 115% than benchmark problem, since the fuel assembly type of IP200 is closed, there is no cross flow. Therefore, the power is more likely rising to the peak in local. There is slightly difference in the shutdown period and after 60**s** between IP200 and PWR benchmark results due to the following two reasons: One suggests that in benchmark definition, there is a stuck out control rod assumption which we did not adopt, the other suggests that there is chemical soluble boron to enhance reactor shutdown depth.

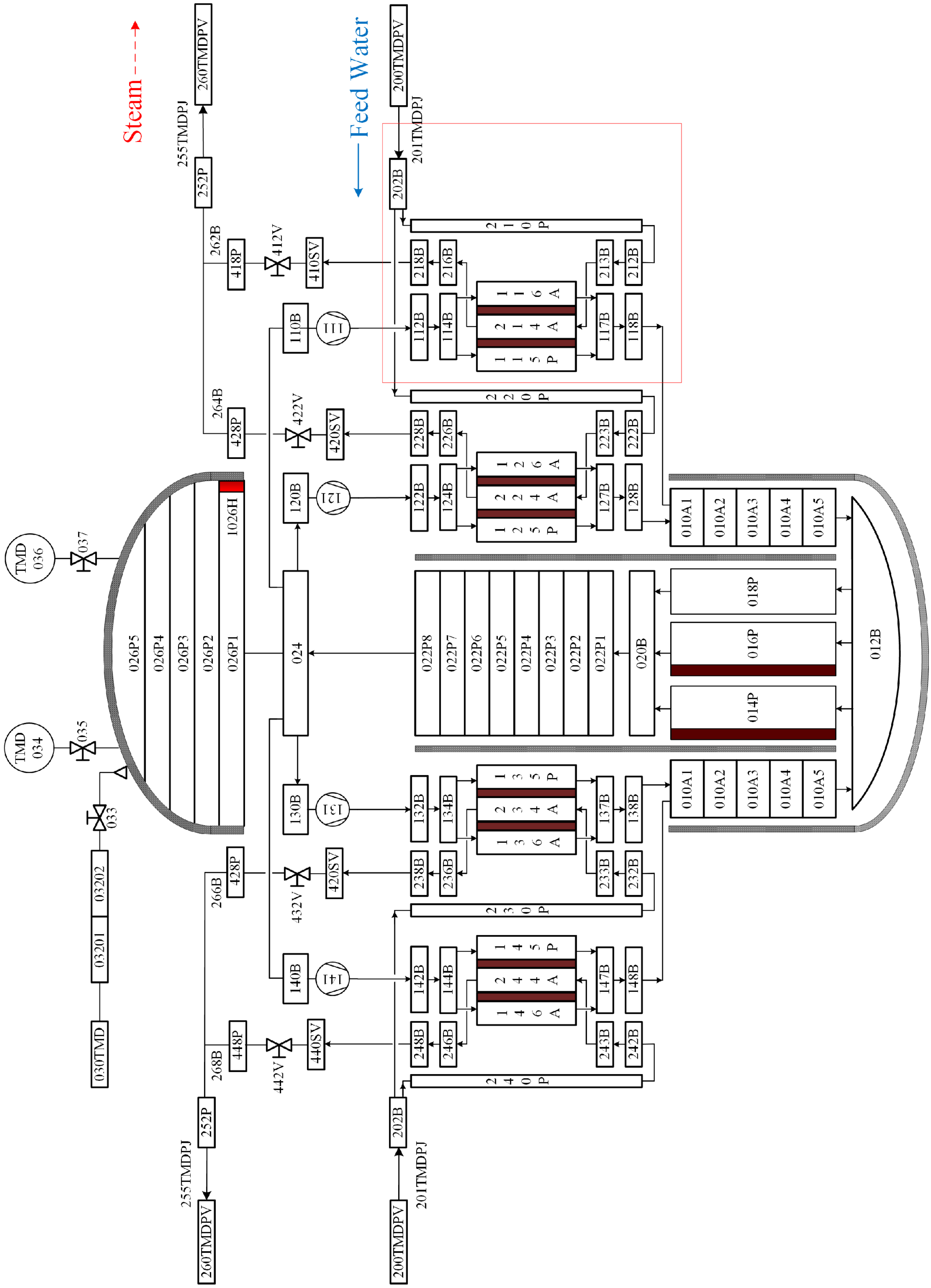


Fig.2 RELAP5 nodalization of IP200 reactor.

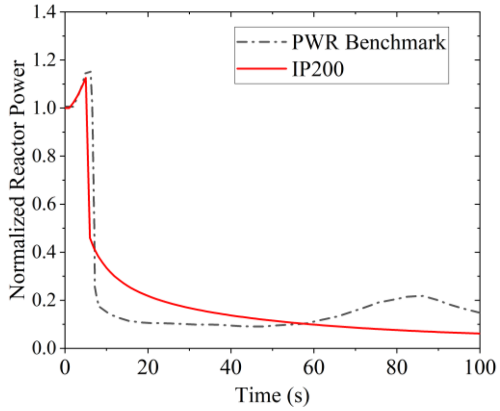


Fig.4 Normalized power variation.

Pressure change follows the reactor power, there is also a small surge following the positive reactivity. The pressure variation (as shown in Fig.5) after the accident varies a lot when compared to the benchmark problem. Due to only 10% coolant inventory of large PWR, IP200 is more likely to be influenced by the power fluctuation. Besides, the break flow rate is only 1/30 of large PWR, the influence to the primary pressure is much smaller. These differences result in a much more gradual reduction of pressure exactly suggest that SMR has better inherent safety performance compare to loop-type PWRs.

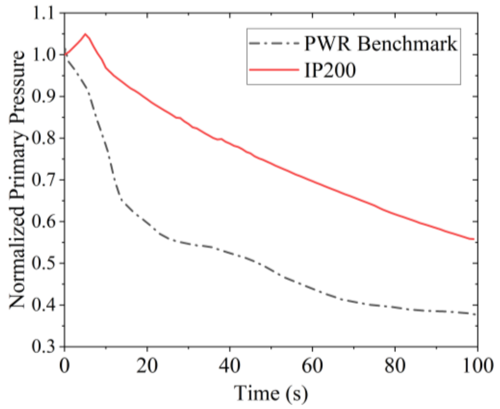


Fig.5 Normalized pressure variation.

At the start of the accident, the enhanced heat transfer leads to average coolant temperature decrease (as shown in Fig.6). The variation of coolant temperature further influence the moderate temperature, density which will also affect the cross section of nuclear fissions. Therefore, the decreased temperature induces positive reactivity. After the reactor trips, the average coolant temperature will decrease gradually. In the benchmark problem, since there is a stuck out control rod, the positive reactivity leads to increase power and coolant temperature.

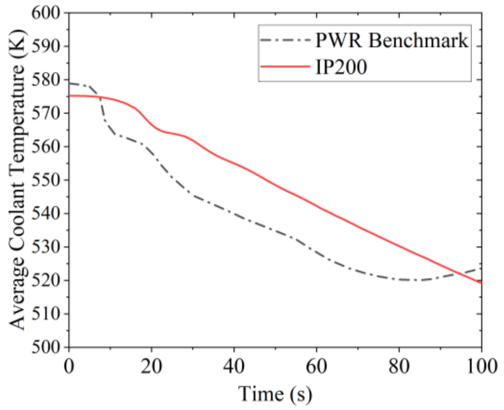


Fig.6 Average coolant temperature variation.

The mixture of superheat steam and saturated water which form critical flow comes out of the break location after the accident occurs. Unlike U-tube steam generators (UTSGs) which produce saturated steam carrying little water, OTSGs produce superheat steam. The break flow of UTSGs are more gradual. However, the OTSG break flow which is two phase critical flow is quite unstable and collides severely (as shown in Fig.7).

Besides, there exist two phase flow instability and collision among heat transfer tubes in OTSGs [13]. All these above features bring about uncertainty to accident consequences. Furthermore, there will also induce vibration in OTSGs that are harmful to the equipment reliability.

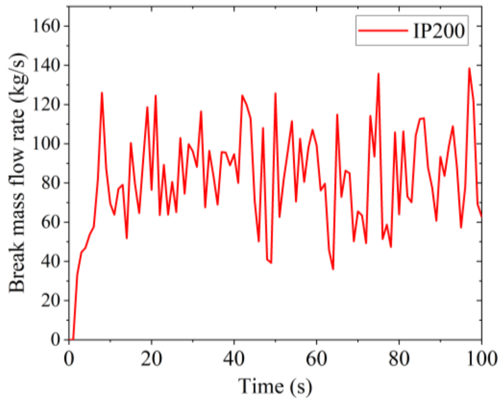


Fig.7 Break mass flow rate variation.

As excepted, the broken steam line pressure drops rapidly following the bFreak as result of the depressurization in the broken SG (as shown in Fig.8). The fluctuation is attributed by the above break mass flow rate instability. However, the fluctuation trends is milder than that of break flow since there is plenty of water inventory in SGs. The intact steam line sees a gradual pressure (as shown in Fig.8) decrease following the reactor trip signal. The much more rapid pressure drop of OTSGs when compared to UTSGs is mainly attributed to the water inventory of the steam generators. In UTSGs, there is a lot of water inventory on the secondary sides, which will absorb and mitigate much more pressure variation and make the pressure decrease more gradually. However, the overcooled water inventory of OTSGs is much less, the pressure change will become so obviously that is full of vibration of rapid falling process.

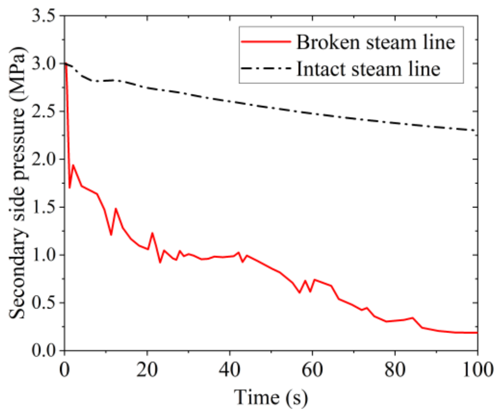


Fig.8 Secondary steam line pressure variation.

4 Conclusions

A comprehensive analysis has been carried out for steam line break accident of IP200 reactor. The considered IP200 reactor is safe enough in the considered transient scenario. However, there exist several differences when compared to loop-type PWR. IP200 triggers reactor trip faster than benchmark test due to core fuel assembly configuration. Furthermore, the primary coolant and secondary SG water inventory is much smaller than PWR, which makes IP200 response more rapidly to the operation parameters' variation. Finally, there exist flow instability and oscillation on the secondary OTSG side when examining the break flow rate and broken line pressure. Above all, IP200 presents reliable inherent safety features under the most limiting main steam line break accident. To further investigate the reactor core behavior under steam line break accident, coupled neutron kinetics and thermal-hydraulic codes are required in future study.

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