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Study on the reactor core barrel instantaneous characteristics in case of Loss of Coolant Accident (LOCA) scenarios for loop-type PWR

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ABSTRACT

The asymmetric pressure wave caused by Loss of Coolant Accident (LOCA) transients in multi-loop type Pressurized Water Reactor (PWR) leads to great shock to the reactor core barrel. The instantaneous shock causes severe damage to the reactor vessel structure integrity. In this paper, a new method for reactor core barrel instantaneous force calculation was proposed based on the widely accepted reactor system analysis code RELAP5. The reliability of RELAP5 code for pipe blow down and pressure wave propagation simulation was assessed against Edward pipe problem firstly. Then the typical three-loop PWR system model was established utilizing Symbolic Nuclear Analysis Package (SNAP) software and the small break LOCA scenarios with different break diameters and break locations were analyzed. The maximum instantaneous asymmetric force applied on the reactor core barrel was achieved under various break conditions, providing the boundary conditions for the core barrel stress distribution and deformation calculation. Results show that the cold leg break LOCA transient leads to great instantaneous asymmetric force, while the hot leg break LOCA transient has no obvious asymmetric force generated. The core barrel instantaneous stress concentration and deformation were discovered. This research could contribute to the reliability assessment of PWR core barrel under LOCA scenarios.

1. Introduction

The reactor core barrel is one of the most important components in a typical Pressurized Water Reactor (PWR). The core barrel and the primary loop coolant interact strongly and the core barrel may suffer great flow impact. The strong coolant flow brings a certain degree of core barrel vibration in the PWR and the vibration of core barrel may cause the structure fatigue, even the equipment damage. So the structure integrity of core barrel must be ensured in the whole design lifetime (Dubyk et al., 2015; Takahashi et al., 2014; Weaver et al., 2000). Currently, the reactor core barrel vibration detectors are usually installed in nuclear power plant in order to control the core barrel structural characteristics. Many core barrel vibration detection systems were designed and invented in different countries for different reactor types (Liewers et al., 1988; Christian et al., 2015; Ansari et al., 2008). They all aimed to master the core barrel small vibrations during the reactor operation conditions and control the reactor operation risk efficiently.

The Small Break Loss of Coolant Accident (SBLOCA) scenarios are typical design basic accidents for nuclear power plant. The SBLOCA transients have attracted more attentions in nuclear engineering field since the Three Mile Island (TMI) nuclear power plant accident in 1979. However, the LOCA was noted by the former US Atomic Energy Commission in early 1970's. They released the Emergency Core Cooling System (ECCS) rule in 10 CFR 50.46 and associated Appendix K (Wang et al., 2010; Nusret, 2007). Since that time, many scholars have contributed great efforts on the LOCA transient study for different reactor types. However, most of the researches were focused on the variations of main reactor thermal hydraulic parameters during the whole process and the reactor status with or without the reactor safety system, while the study of instantaneous shock to the reactor vessel structure, especially to core barrel with the cold leg break, was not paid sufficient attentions. Actually, the first pressure wave in the initial phase of LOCA scenarios somehow causes the most critical conditions. Therefore, the instantaneous effect and shock to core barrel during LOCA transients should be studied carefully.

Usually, the structure integrity analysis tools, such as ANSYS and ABAQUS, are adopted for the core barrel structure analysis in a typical PWR. Hermansky and Krajkovic (2011) presented the preliminary numerical results of the WWER440/V213 reactor vessel internals dynamic response to maximum hypothetical large break LOCA. The finite element model was built by MSC. Patran (Patran, 2010) and dynamic

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response was achieved. The results demonstrated that the acceptance criteria for reactor vessel internals was fulfilled and required nuclear power plant safety standards were satisfied. Faucher et al. (2014) was devoted to the analysis of transient mechanical consequences of LOCA on the internal structures of a PWR. A complete methodology was described and a coupled 1D/3D code was developed for the calculation of fluid-structure interaction inside the main vessel. In these cases, the thermal hydraulic simulation was required to provide necessary boundary conditions. The most popular Computational Fluid Dynamics (CFD) method could handle many reactor single phase thermal hydraulic problems. However, for the thermal hydraulic study of initial blow-down phase during LOCA scenarios, the capacity of CFD method has not been demonstrated in the basic of current models.

Many researches have been performed using system analysis code for the pressure wave propagation process in the simple pipe system. Some classical experiments were carried out to verify the numerical simulation results. Barna et al. (2010) investigated the water hammer phenomena induced by steam condensation. The experiments were performed in the PMK-2 test facility and compared with the WAHA3 (Tiselj et al., 2004; Tiselj and Petelin, 1997) code analysis. Results show that the numerical calculation with WAHA3 code was in good agreement with that of experimental data. Sung-Jae Yi et al. (2016) also used experimental data obtained in PMK-2 test facility to verify the capacity of WAHA3 code for the pressure wave propagation phenomena simulation in a pipe system. Marco Colombo et al. (2012) investigated the density wave instabilities in single channel and two parallel channels by means of the RELAP5/MOD3.3 code. Barten et al. (2008) performed a very comprehensive analysis for the coupled two-phase flow and pressure wave propagations using system code by comparing the numerical results with the UMSICHT PPP cavitation water hammer experiments 329 and 135. Sokolowski and Koszela (2012) also assessed the capacity of RELAP5 code for the pressure wave behavior simulation through the shock tube problems as well as water hammer experiments.

In this paper, a new attempt for the reactor core barrel instantaneous force calculation was carried out based on the worldwide accepted system safety analysis codes, such as RELAP5, RETRAN, TRAC, CATHARE, ATHLET, RAMONA and TRACE. The typical three-loop PWR CPR1000 NPP primary coolant system was modeled and the corresponding core barrel instantaneous force calculation model was developed. The instantaneous force features were studied during SBLOCA transients in case of different break diameters and break locations. At last, the simplified three-dimensional core barrel model was built and the Finite Element Method (FEM) analysis was performed.

2. Methods and verification

2.1. The method description

For a typical multi-loop type PWR, the export, exchanging and transformation of nuclear fission energy are conducted in the reactor coolant system. The basic part of the system should withstand high pressure, constituting the “pressure boundary”, which is one of the three “security barriers” of nuclear power plant. During the primary loop LOCA transients, the pressure decreases sharply and the discharge coolant reaches the critical flow immediately. The pressure wave propagates along the main pipe instantly to the reactor vessel. For a multi-loop type PWR, one loop break would cause asymmetric pressure on the core barrel surface, leading to core barrel great shock. The pressure wave propagation process could be calculated using reactor system code and the asymmetric pressure distribution could be achieved. The schematic diagram of this method is shown in Fig. 1. However, it should be noted that the existed PWR nuclear power plants in the world contain less than six loops.

The maximum force applied on the core barrel could be obtained based on the following equation.

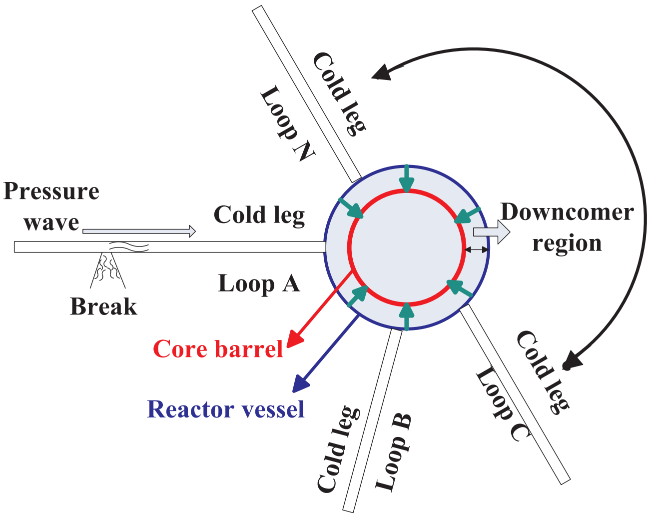

Fig. 1. The schematic of multi-loop type PWR core barrel instantaneous force calculation.

$$F_{barrel} = P_{asymmetry}(t) \times A \tag{1}$$

where $P_{asymmetry}$ is the asymmetric pressure of reactor core barrel, A is the asymmetric loaded force area.

2.2. The model verification

The key models of this method include critical flow model and pressure wave propagation model. The critical flow is important for the safety assessment of a water-cooled nuclear reactor because it determines the system response characteristics under LOCA scenarios (Ardron and Furness, 1976). The critical flow phenomenon has been extensively studied in single-phase and two-phase systems. Meanwhile, the discharge flow makes a great impact on the system depressurization rate, which affects the pressure wave propagation process and the maximum core barrel asymmetric force. Generally, it is accepted that the critical flow rate depends on the fluid stagnation condition, the entrance geometry and the length to diameter ratio of the test section. In RELAP5 code, the default choked flow model is Henry-Fauske critical flow model, which is established based on the following assumptions. It is reasonable to assume that the amount of mass transferred in the expansion is negligible due to the instantaneous effect.

$$x_t \approx x_0 \tag{2}$$

The amount of heat transferred during the expansion process is also negligible.

$$T_{lt} \approx T_{l0} \tag{3}$$

The system entropy during the expansion can be assumed constant.

$$ds_0 = d[(1-x)s_l + xs_g] = 0 \tag{4}$$

The discharge process is regarded as an isentropic process.

$$s_0 = s_t \tag{5}$$

It is assumed that the vapor behavior at the throat could be described by a polytropic process.

$$\left.\frac{dv_g}{dP}\right|_t = \left.\frac{v_0}{nP}\right|_t \tag{6}$$

$$n = \frac{(1-x)c_l/c_{pg} + 1}{(1-x)c_l/c_{pg} + 1/\gamma} \tag{7}$$

Then the critical value of mass flux could be written as: