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Experimental and analytical investigations of primary coolant pump coastdown phenomena for the Jordan Research and Training Reactor

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HIGHLIGHTS

- Core flow coastdown phenomena of a research reactor are investigated experimentally.
- The experimental dataset is well predicted by a simulation software package, MMS.
- The validity and consistency of the experimental dataset are confirmed.
- The designed coastdown half time is confirmed to be well above the design requirement.

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ABSTRACT

Many low-power research reactors including the Jordan Research and Training Reactor (JRTR) are designed to have a downward core flow during a normal operation mode for many convenient operating features. This design feature requires maintaining the downward core flow for a short period of time right after a loss of off-site power (LOOP) accident to guarantee nuclear fuel integrity. In the JRTR, a big flywheel is installed on a primary cooling system (PCS) pump shaft to passively provide the inertial downward core flow at an initial stage of the LOOP accident. The inertial pumping capability during the coastdown period is experimentally investigated to confirm whether the coastdown half time requirement given by safety analyses is being satisfied. The validity and consistency of the experimental dataset are evaluated using a simulation software package, modular modeling system (MMS). In the MMS simulation model, all of the design data that affect the pump coastdown behavior are reflected. The experimental dataset is well predicted by the MMS model, and is confirmed to be valid and consistent. The designed coastdown half time is confirmed to be well above the value required by safety analysis results.wwwyoon@gmail.com

1. Introduction

The Jordan Research and Training Reactor (JRTR) is an open-tank-in-pool type reactor with a downward core flow during a normal operation mode. In this type of a reactor, when the electricity supply to the primary cooling system (PCS) pumps is lost, the downward coolant flow will decrease gradually owing to a flywheel installed on the PCS pump shaft. Before the coolant flow stops completely, flap valves installed on a reactor outlet PCS pipe will be opened by gravity. Residual heat will be reduced to a safe enough level before the flow inversion occurs by the natural circulation through the flap valves and the reactor core. After the flow inversion, core decay heat will be dissipated into a reactor pool, i.e., the ultimate heat sink by the natural circulation. In this paper, flywheel performance is investigated experimentally and analytically.

Many researchers proposed numerical solutions using pump characteristics and analytical solutions based on several assumptions. Yakomura (1969) derived an analytical method for coastdown flow estimation in a case of pump power failure in water-cooled reactor systems. His method was proved to estimate the coastdown flow rate with high accuracy without the need of detailed information on the pump characteristics. He investigated the effect of mechanical friction loss on the coastdown flow rate. In an early stage of a coastdown, this effect is found to be small,

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and a half-decay time of the flow is not affected by the friction loss. However, in a late stage, this effect becomes larger and the time when the flow rate becomes zero depends very much on the friction loss estimation. His method was examined through a comparison with experimental results. Rizwan-uddin (1994) proposed a simple model to simulate fast centrifugal pump transients and stopping periods. The model relates the pump head and flow rate with a shaft speed. It is based on a hypothesis that the rate of head changes is proportional to the rate of changes of the square of a shaft speed. A comparison with experimental data for fast starting and stopping actions showed excellent agreement. Farhadi (2010) developed a mathematical model for analyzing a flow coastdown transient by deriving differential equations for a diminishing fluid flow in a piping system and a retarding motion of rotating parts of a centrifugal pump in case of a loss of electricity. The influence of kinetic energy in a piping system and kinetic energy of a pump is considered in a form of a ratio that he called an effective energy ratio. Analytical results obtained from the model, which does not require detailed pump characteristic curves, are compared with the results obtained using the characteristics of centrifugal pumps, and excellent agreement is observed. The value of the effective energy ratio is found to be an excellent predictor of the pump performance in transient events. Gao et al. (2011) used a moment balance relation of a reactor coolant pump to derive a pump speed transient equation under a flow coastdown condition. The momentum conservation of a primary coolant is used to derive a flow rate transient equation for a flow coastdown. The flow rate transient equation is non-dimensionalized using a pump half-time and a loop half-time. The transient flow rate and pump speed during a flow coastdown are analytically solved. The solution indicates that the transient flow rate is determined by a ratio of the steady-state kinetic energy of the loop coolant fluid to the kinetic energy stored in the rotating parts during the steady-state operation. The transient pump speed is determined by the pump half-time. The pump half-time is affected by the moment of inertia and initial speed of the reactor coolant pump. The model prediction is verified experimentally by comparing a non-dimensional flow rate and a non-dimensional pump speed obtained from the model with published experimental data of two nuclear power plants and a reactor model test on flow coastdown transients. Alatrash et al. (2014) investigated the effect of inertia of the JRTR PCS loop. This effect is found to be small, and the coastdown behavior in the JRTR is found to be mainly determined based on the inertia of the pump flywheel.

In the present paper, the validity and consistency of the coastdown test results for a JRTR PCS pump are evaluated using a simulation software package, modular modeling system (MMS). The pump speed transient equation is derived for a loss of off-site power (LOOP) accident condition in the range where pump affinity rules are valid. In a low flow range where the pump affinity rules are invalid, the measured flow friction loss through the PCS pump is separately taken into account in the model. The calculated coastdown curves agree well with the measured coastdown curves in the whole range of the pump coastdown.

The simulation software package, MMS, has been used for a rapid simulation of power plants by using pre-engineered components called Modules, since it was first released in 1984. The MMS has also been used for the purpose of modeling and simulation of nuclear power plants (Kang et al., 2010; Yoon et al., 2000, 2002; Kang et al., 2000; Godbole and Malan, 1991). More technical papers based on the MMS can be found at www.nhancetech.com.

2. Jordan Research and Training Reactor

The nominal power of the JRTR is 5 MW and it is upgradable to 10 MW. It is under the last stage of construction within the campus of the Jordan University of Science and Technology (JUST) located in the Municipality of Ramtha, the Hashemite Kingdom of Jordan. It is planned to be a hub for the nuclear research and training in the Middle East region.

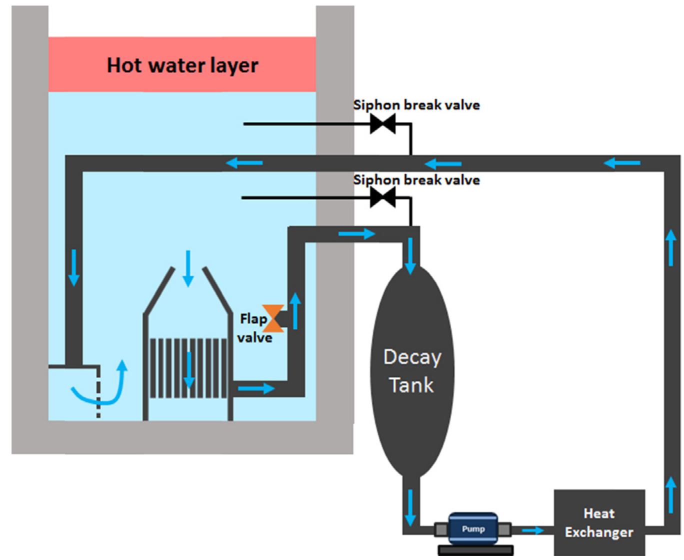

Fig. 1. Schematic diagram of the JRTR PCS system.

Fig. 1 shows a schematic diagram of the JRTR PCS system including major components, i.e., a decay tank, centrifugal pump, plate type heat exchanger, flap valve and siphon breaker valves. During a normal power operation mode, downward core flow is formed by the PCS pumps to cool the heat generated by the reactor core. The coolant flows through a reactor outlet PCS pipe, a decay tank, two PCS pumps, two plate-type heat exchangers and a PCS discharge header to return to the reactor pool.

After the PCS pumps stop due to normal or abnormal conditions, the decay heat is removed by the flow induced by the inertia force of a flywheel attached to each PCS pump shaft. After the PCS inertia flow decreases slowly, the flap valves installed on the reactor outlet PCS pipe inside the pool are passively opened. The openings of these valves provide a flow path for natural circulation from the flap valves through the core to remove the core decay heat.

3. Solution method

The MMS is a Microsoft Windows based visual software system for modeling the dynamic characteristics of power plant systems and studying the various design, performance, operation, training, interface, and maintenance aspects during the entire plant life cycle. Systems that can be modeled with the MMS include, but are not limited to, thermal–hydraulic, nuclear, electrical, and control systems. An MMS model is a collection of component modules graphically configured to represent the dynamic behavior of a system. These modules are connected together to define the inter-relationships. The MMS provides the capability of a real-time simulation, and many flexible features that give users the ability to modify their model and study the effect of adding or removing components on system performance (nHance Technologies Inc., 2005).

3.1. MMS governing equations

Each MMS module typically has an input sheet in which the design and operating data of the modeled component are entered. These data can also be changed during the run-time of the model. The pressure loss coefficient of an individual MMS