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Uncertainty assessment for loss of flow accident of a 5 MW pool-type research reactor

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ABSTRACT

Best estimate plus uncertainty (BEPU) is a promising approach to the safety analysis of nuclear reactors, and the uncertainty calculation is a very important concern about it. BEPU ensures realistic safety margins and secures higher reactor effectiveness by adopting best-estimate codes and realistic input data with uncertainties, whereas the previous conservative analysis generates excessive conservatism by considering each input parameter separately. A loss of flow accident (LOFA) of a 5 MW open-pool type research reactor was selected as a sample problem for a BEPU uncertainty assessment. We selected the failures of all primary cooling system (PCS) pumps, which would cause the abrupt reduction of flow and the reversal of core flow. The significant contributors to the reactor safety were identified and then input sets were sampled. For the uncertainty evaluation, 124 calculations were performed. This is the number of code runs required for a 95%/95% level with the 3rd order Wilk's formula. The MOSAIQUE software developed by Korean Atomic Energy Research Institute (KAERI) was used for automated sampling of the uncertainty parameters, a global uncertainty calculation, and post processing of the results. The critical heat flux ratio (CHFR) and the fuel centerline temperature (FCT) were calculated at the 95%/95% level and were compared with those from conservative analyses. In addition, the impact of each design variables on the safety parameters was estimated by sensitivity analysis.

1. Introduction

In 1989, the U.S. Nuclear Regulatory Commission (U.S. NRC) revised its regulations such that BEPU (best estimate plus uncertainty) was able to replace the previous conservative approach to reactor safety analyses. Meanwhile, the ATUCHA Unit 2 in Argentina recently obtained an operating license using a final safety analysis report based upon the BEPU approach. This shows the global trend of nuclear reactor safety analysis, which is shifting from a conservative analysis to BEPU (IAEA, 2008). The former (conservative approach) depended on conservative assumptions for boundary and initial conditions because of the lack of understanding of the details of the phenomena involved, or of insufficient experimental data (US NRC, 1996). However, the combination of such conservative assumptions often generated excessive conservatism, which resulted in reduction of the operational margin. BEPU estimates safety parameters more realistically, making it possible to responsibly reduce excessive conservatism in a safety analysis and increase the operating margin of a reactor.

Following these international trends, a BEPU safety analysis was conducted for a scenario in which all the PCS pumps of a 5 MW pool-type research reactor failed. The core safety parameters and input variables affecting the consequences were selected first, and the safety parameters were then calculated using a non-parametric uncertainty analysis. The safety parameters including uncertainties are presented herein and the importance of each parameter is ranked based on sensitivity analyses.

2. Introduction

For BEPU uncertainty assessment, a loss of flow accident (LOFA) in a 5 MW open-pool-type research reactor was selected as the sample problem.

The reactor is a multi-purpose open-tank-in-pool type reactor of which the nominal fission power is 5 MW. The reactor assembly and experimental devices are submerged under water and are illustrated in Fig. 1. The fuel is a plate-type and 21 fuel plates constitute one fuel assembly. The reactor core is cooled either by forced convection with PCS pumps or by natural convection via flap valves. The PCS pumps cool the core when the reactor is operated at nominal power. When the PCS pumps are tripped, the

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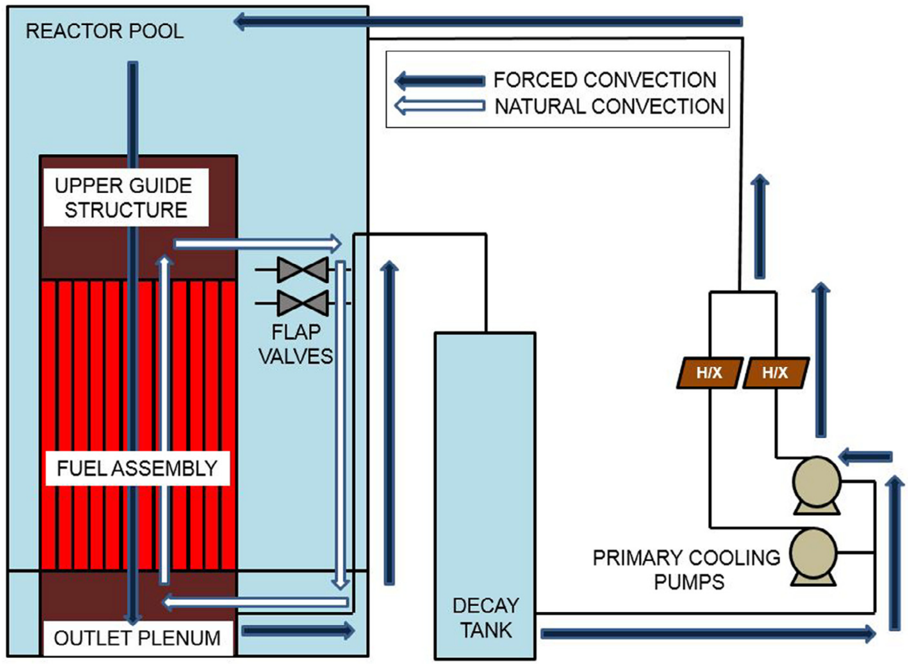

Fig. 1. Forced convection mode and natural convection mode of the reactor.

pumps begin to coastdown and the core flow decreases. After the coastdown ends completely, the core flow reverses due to buoyancy force and the reactor is cooled by natural circulation.

Table 1 summarizes the sequence of events during the failure of all PCS pumps. The failure of all PCS pumps is assumed to be caused by a loss of power to the pumps, or by simultaneous malfunction of pump motors. Once all the PCS pumps stop due to electrical or mechanical failures, the pumps begin to coastdown by the inertia of pump flywheels. The coastdown results in decreased PCS flow and core differential pressure (dP). Then, the reactor is tripped by the low PCS flow signal or the low core dP of the reactor protection system (RPS). As the core flow decreases further, the flap valves open passively when the pressure across the valve decreases below the design value for opening. Then, the core flow reverses to upward, driven by the buoyancy of the coolant in the core. The decay heat from the core is removed by natural circulation via the flap valves.

Fig. 2 shows the typical behavior of the two main safety parameters for fuel integrity, the minimum CHFR (MCHFR) and the maximum FCT (MFCT), during the event. The results are calculated with the best estimate input parameters. The best estimate input represents the average value of the upper and lower limit of the operating range with additionally considering measurement uncertainty. The MCHFR and the MFCT show two distinct peaks, respectively: The first peak takes place at the time of the reactor trip during the forced convection mode and the second peak occurs at the flow reversal during the natural convection mode. A 'failure of all PCS pumps' event was selected for the BEPU assessment because it is

Table 1
Sequence of event.

Time [s]	Events
1	The reactor is in power operation, cooled by forced convection.
2	Loss of 2 PCS pumps occurs and the PCS flow coasts down.
3	Core DP reaches the low setpoint of RPS.
4	PCS flow reaches the low setpoint of RPS Reactor trip signal occurs.
5	The CARs begin to drop into the core.
6	Flap valves are open.Core is cooled by natural circulation.

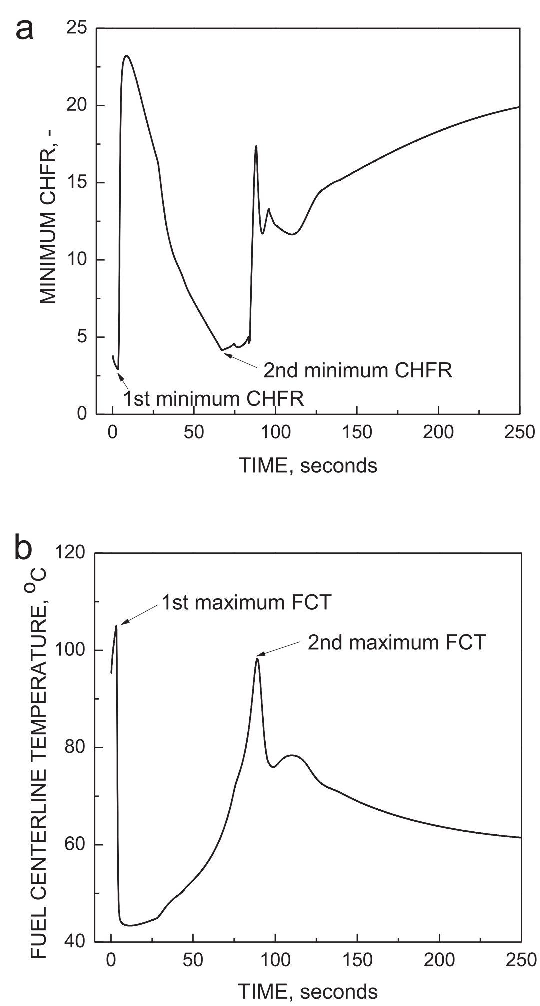

Fig. 2. The behavior of MCHFR and MFCT during loss of flow accident. (a) MCHFR and (b) MFCT.