Application of Probabilistic Safety Assessment to Korea Research Reactor

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1. Introduction

A Probabilistic Safety Assessment (PSA) was performed to assess the level of safety of the design of the Korea research reactor (hereafter referred to as the KRR) and to evaluate whether it is probabilistically safe to operate and reliable to use according to the procedures published by the International Atomic Energy Agency (IAEA) [1][2] and the US Nuclear Regulatory Commission (NRC) [3]. This PSA was undertaken to assess the level of safety for the KRR and to evaluate whether it is probabilistically safe to operate and reliable to use. The technical objectives of this study were to identify the accident sequences causing core damage and to determine the corresponding frequencies.

2. Research Reactor

The KRR is an open tank-in-pool-type multi-purpose research reactor using light water as both the moderator and the coolant, and heavy water as the reflector [4]. The reactor core is submerged in a reactor pool of 4 m in diameter and 13.4 m in depth, and the reactor pool is connected to the service pool by a transfer canal. The reactor core consists of inner and outer cores. Hydraulically the core has a total of 39 separate flow channels, and the inner core is enclosed by the corrugated and parallelepiped inner shell of the reflector tank. The fuel element of the KRR is made from pencils of an extruded uranium-silicon-aluminum dispersion with finned aluminum cladding. The operation cycle length of the reactor is about 28 days at full power conditions [4].

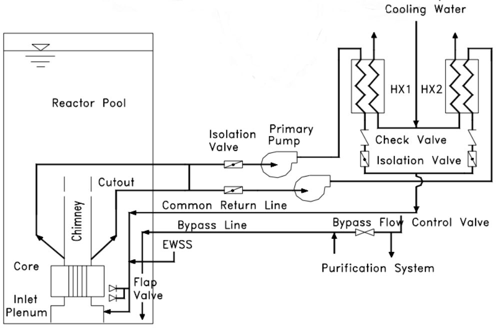

Fig. 1. Schematic Diagram of Cooling and Connected System of KRR [4]

The primary cooling system (PCS) of the KRR is shown in Fig. 1. The PCS consists of one circuit (composed of two parallel 50% capacity pumps (P01 and P02) and heat exchangers (X01 and X02)), flap valves (V003 and V004), and all the necessary interconnecting pipes, valves, flow orifices, and instruments. The PCS outside the reactor pool consists of two parallel circuits, each with a pump and a plate-type heat exchanger with 50% capacity. This water around the outside of the reactor core structure gradually rises from the bottom of the pool, keeping the latter cooled. Under normal operation, the core heat is removed by the forced convection through the two pumps and the two heat exchangers. Approximately 90% of the total PCS flows from the core exit meets the 10% downward bypass, flowing in the lower part of the chimney and flows to the pumps through the chimney nozzles. The main function of the bypass flow is to prevent the core flow containing the activated N16 and other radioisotopes from leaking from the chimney to the pool. During a reactor shutdown, the decay heat is removed via natural circulation by a gravity-driven recirculating flow through the flap valves inside the pool. During loss-of-pool-water incidents, uncovering the core is prohibited by the layout of the PCS pipes and the flap valves [4].

3. KRR PSA

3.1 PSA Software

Advanced Information Management System-Probabilistic Safety Assessment (AIMS-PSA) [5], and Fault Tree Reliability Evaluation eXpert (FTREX) [6] were used to conduct the risk evaluation. AIMS-PSA, developed by the Korea Atomic Energy Research Institute (KAERI), is a software package for PSA. It can be used to construct fault trees and event trees to generate Minimal Cut Sets (MCSs) for each sequence and to subsequently run importance and uncertainty analyses.

3.2 Initiating Events and Accident Sequence Analysis

The initiating events were finally selected and are listed in Table 1. Event trees were developed to identify the accident sequences causing core damage for the selected nine initiating events.

Table 1. Initiating Event List

Initiating Event	
1	Loss of Electric Power: LOEP
2	Reactivity Insertion Accident: RIA
3	Loss of Primary Cooling System: LOPCS
4	Loss of Secondary Cooling System: LOSCS
5	Loss of Coolant Accident: LOCA
6	Beam Tube LOCA
7	Single Channel Flow Blockage: SCFB
8	General Transient by Manual Trip: GTRN-MT
9	General Transient by Automatic Trip: GTRN-AT

3.3 Fault Tree Analysis

The failures of the major components and the dependencies between the systems were considered for the fault tree analysis. The failures of the pumps and supporting systems, such as the electric power system, were modeled, and the failures of the check valve, motor-operated valve, heat exchanger, I&C components involving transmitter, level switch, and relay were also modeled.

3.4 Quantification Results

In this analysis, the dominant MCSs causing core damage are identified from the base model, as listed in Table 2.

Table 2. MCSs Causing Core Damage at Base Model

No	CDF	F-V	Initiating Event	Event 1	Event 2
1	3.67E-06	0.81	%LOEP	PCCVW-FL0304AB	#LOEP-2
2	3.20E-07	0.07	%BT-LOCA	EWVVT-V010	#BT-LOCA-2
3	2.60E-07	0.06	%SCFB	RPOPV-SCFB	#SCFB-4
4	1.18E-07	0.03	%LOPCS	PCCVW-FL0304AB	#LOPCS-2
5	5.33E-08	0.01	%BT-LOCA	EWVVT-V009	#BT-LOCA-2

The first MCS has the greatest impact on the CDF, as understood from Table 2. Following LOEP, this scenario occurs when the two flap valves do not work simultaneously, owing to the same mechanical issue. LOEP is the occurrence of the LOEP initiating event, and PCCVW-FL0304AB is the CCF of the two flap valves. The dominant contributor to the CDF of this sequence is the CCF of the flap valves. The estimated CDF of this sequence is approximately 3.67E-06/yr as a point estimate value, and this MCS provides 80.7% of the CDF. The summary of the results is tabulated in Table 3, including the results of the contributions to the total CDF by initiating events. LOEP is a main contributor to the total CDF by a single initiating event. The final quantification result indicates a point estimate of 4.55E-06/yr for the overall CDF attributable to the internal initiating events for the research reactor. An LOEP initiating event is categorized as the sequence with the highest CDF contribution with a value of 3.68E-06/yr being 80.9% contribution of the CDF. BT-LOCA is the second largest contributor at 9.9%, whereas the contributions of the SCFB, LOPCS, LOSCS, LOCA, GTRN, and RIA are relatively small.

Table 3. Core Damage Frequencies

Initiating Event	Frequency (/yr)	CDF	%
LOEP	1.92E+00	3.68E-06	80.88
RIA	1.67E+00	8.13E-09	0.18
LOPCS	6.20E-02	1.20E-07	2.64
LOSCS	6.20E-02	3.02E-10	0.01
LOCA	9.89E-04	1.90E-09	0.04
SCFB	1.30E-05	2.62E-07	5.76
BT-LOCA	6.85E-06	4.49E-07	9.88
GTRN-AT	5.65E+00	2.75E-08	0.60
GTRN-MT	1.43E+00	4.82E-10	0.01
Total		4.55E-06	100.0

4. Conclusions

This paper describes the work and the results of the PSA for the KRR. The PSA was undertaken to assess the level of safety for the design of the research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use. The principal conclusions from this study are as follows:

- The final quantification result indicates a point estimate of 4.55E-06/yr for the overall CDF attributable to the internal initiating events for the research reactor.
- The KRR is well designed to be sufficiently safe from a safety stand-point.
- The PSA methodology is very effective in improving reactor safety in the full-power operating phase, and, in particular, it is a highly suitable approach to determine the deficiencies of a reactor at full power and improve the reactor safety by overcoming them.

Acknowledgments

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