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JRTR 연구용원자로에 대한 최종 확률론적 안전성평가

이윤환*†

A Study on the Final Probabilistic Safety Assessment for the Jordan Research and Training Reactor

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Abstract : This paper describes the work and the results of the final Probabilistic Safety Assessment (PSA) for the Jordan Research and Training Reactor (JRTR). This final PSA was undertaken to assess the level of safety for the design of a research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use. The scope of the PSA described here is a Level 1 PSA, which addresses the risks associated with core damage. After reviewing the documents and its conceptual design, nine typical initiating events were selected regarding internal events during the normal operation of the reactor. AIMS-PSA (Version 1.2c) was used for the accident quantification, and FTREX was used as the quantification engine. 1.0E-15/yr of the cutoff value was used to deliminate the non-effective Minimal Cut Sets (MCSs) when quantifying the JRTR PSA model. As a result, the final result indicates a point estimate of 2.02E-07/yr for the overall Core Damage Frequency (CDF) attributable to internal initiating events in the core damage state for the JRTR. A Loss of Primary Cooling System Flow (LOPCS) is the dominant contributor to the total CDF by a single initiating event (9.96E-08/yr), and provides 49.4% of the CDF. General Transients (GTRNs) are the second largest contributor, and provide 32.9% (6.65E-08/yr) of the CDF.

Key Words : JRTR, PSA, research reactor, core damage frequency, CDF

1. INTRODUCTION

This paper describes the results of the at-power internal events Level 1 final Probabilistic Safety Assessment (PSA) for the Jordan Research and Training Reactor (JRTR). This final PSA has been performed to assess the level of safety for the design of the JRTR and to evaluate whether it is probabilistically safe to operate and reliable to use according to the procedures published by IAEA[1,2] and the U.S. NRC[3]. The technical objectives of this study were to identify accident sequences leading to core damage and the corresponding frequencies.

1.1. General Description of the JRTR

As shown in Fig. 1, the JRTR is a multi-purpose open-tank-in-pool type reactor, with a nominal power of 5 MW. It has 18 fuel assemblies, each with 21 plate type fuel plates. The fuel is low enriched uranium with a ^{235}U enrichment of 19.75 weight %[4].

There are two kinds of reflectors in the JRTR. Beryllium reflector assemblies are located in the core region as the primary reflector material. Heavy water, as the secondary reflector material, is contained in a heavy water vessel. The beryllium reflector assemblies are supported by and located on a grid plate[4].

There are two kinds of reactivity control mechanisms in the JRTR: Control Rod Drive Mechanism (CRDM) and Second Shutdown Drive Mechanism (SSDM). A CRDM inserts, withdraws, or maintains at a required position Control Absorber Rods (CARs) using a stepping motor. The

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JRTR has four hafnium CARs whereby three of them can successfully shut down the reactor. The SSDM, as an alternate and independent shutdown mechanism, provides a secondary means of reactor shutdown by the gravity drop of the Second Shutdown Rods (SSRs). There are two SSRs in the JRTR, and both of them are needed to shut down the reactor successfully if all of the CARs are assumed to be stuck. All CARs and SSRs are dropped by gravity when a reactor trip is required by the Reactor Protection System (RPS) or by the Alternate Protection System (APS)[4].

For the cooling of the reactor core during normal power operation mode, the Primary Cooling System (PCS) is used to circulate the primary cooling water downward through the core as shown in Fig. 1. The heat is then transferred to the secondary cooling system through heat exchangers to be released to the environment using a cooling tower. After the shutdown of the reactor, decay heat is removed by the establishment of natural circulation through two flap valves that are opened passively after stopping the PCS pumps, which are installed at the PCS outlet pipe inside the reactor pool. The JRTR has two siphon break valves at the PCS inlet and outlet pipes, and these siphon break valves are used to stop any leakage of the primary coolant in the PCS piping system by stopping the siphon effect that results from the elevation difference between the pool and the PCS piping system. Siphon break valves (two on the pool outlet pipe) can be used to establish a natural circulation flow path if the flap valves fail to open[4].

In the case of a large loss of coolant from the pool water due to multiple beam tube ruptures, the Emergency Water Supply System (EWSS) is used to inject water into the reactor core in order to maintain the minimum pool water inventory that is required to prevent core uncovering and to remove the decay heat after reactor shutdown. There are some other connected systems that have non-safety functions, for the example, the Pool Water Management System (PWMS), Hot Water Layer System (HWLS), and Heavy Water System (HWS), as shown in Fig. 1[4].

The electric system of the JRTR works as a support system for the proper functioning of the front line safety systems that directly perform a safety function in addition to its other non-safety functions. It provides the electric power needed to operate the different components of the JRTR systems. The electric system can be categorized

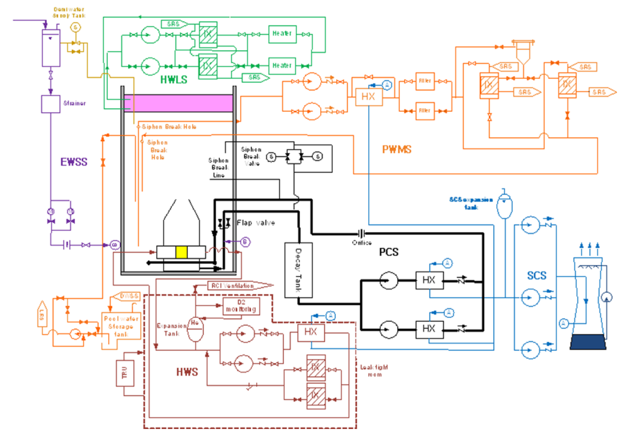

Fig. 1. Schematic diagram of the reactor cooling and connected system of the JRTR.

into three subsystems: normal power supply system, essential power supply system, and uninterruptible AC and DC power supply system. The normal power supply is connected to the electric power company, the essential power is connected to the normal power and has a backup power supply via a diesel generator, and the uninterruptible power supply is connected to the essential power and to an uninterruptible AC system and a battery as an uninterruptible DC power supply system[4].

1.2. PSA Scope

The scope of the PSA reported here is a Level 1 PSA which addresses the risks associated with core damage. It includes an evaluation of the types of accidents that could lead to core damage, and an assessment of their frequencies. The PSA includes only internal initiating events including a loss of offsite power. External initiating events such as earthquakes, floods, fires, and sabotage are not included in this study. All operating modes of the reactor have been considered, but only a full power operation has been assessed as risk significant. Core damage has been conservatively assumed to result in any state of the core where the fuel temperature exceeds the design limit or if the available thermal-hydraulic analyses cannot demonstrate successful cooling of the core.

This paper presents the methodology and software for the PSA (Section 2), the identification of accident initiators (Section 3), event tree analysis (Section 4), fault tree analysis (Section 5), data analysis (Section 6), accident sequence quantification (Section 7), and concluding remarks (Section 8).

2. PSA METHODOLOGY AND SOFTWARE

2.1. PSA Methodology

PSA methodology and approaches for research reactors are in general very similar to those used for power reactors.

The PSA of this study has the following scope: This final PSA for a reactor design uses methodologies consistent with those outlined in the “IAEA PSA Procedures Guide”[1] and “PRA Procedures Guide”[2]. The following subsections describe the major tasks of the PSA. There are six major tasks associated with Level 1 internal event analysis, as shown in Fig. 2.

The first task in the PSA is plant familiarization. The objective of this task is to collect the information necessary for performing the PSA. All information necessary for identification of appropriate initiating events, determination of the success criteria for the systems required to prevent or mitigate the transients and accidents and to identify the dependence between the front line systems and the support systems which are required for a proper functioning of the front line systems.

The second task is to identify and select postulated initiating events. An initiating event is regarded as an event that may lead to core damage if it is combined with the failure of the safety features.

The third task is to develop various accident scenarios (accident sequences) that are combinations of the initiating events and the successes or failures of the systems. This task is accomplished using an event tree.

The fourth task is system modeling. The task includes the construction of fault tree models to identify the causes and probabilities of the system failure. A fault tree of a system is the logical representation in which various causes of a system failure are combined using logic gates, such as OR and AND. Component hardware failures, common cause failures, human errors, and unavailability due to testing and maintenance are included in the fault tree model for a postulated system failure.

The fifth task involves collecting and evaluating the reliability data required for the quantification of event tree and fault tree models. The type of reliability data includes the initiating event frequencies, component hardware failure rates, common cause failure rates, and human error probabilities.

The final task is the accident sequence quantification. The objective of the accident sequence quantification is to evaluate the CDF for each sequence and to find the dominant contributors to the risk. The event tree and fault tree linking approach is used as a basic method of accident sequence quantification. The total CDF is estimated as the sum of the frequencies of the individual accident sequences resulting in core damage.

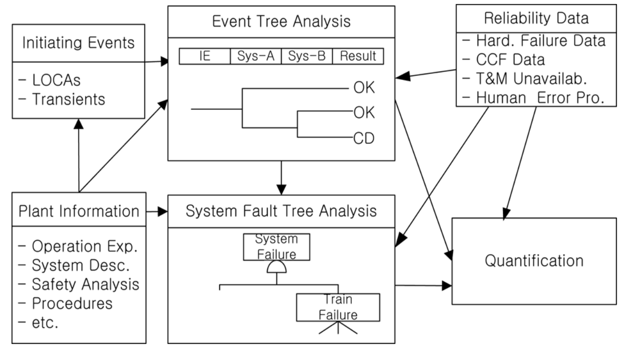

Fig. 2. Major tasks of the PSA.

2.2. PSA Software

AIMS-PSA (Version 1.2c)[5] and FTREX[6] are used for the final PSA of the JRTR. AIMS-PSA, developed by the Korea Atomic Energy Research Institute (KAERI), is software for PSA. It provides a tool to construct fault trees and event trees, to generate minimal cut sets for each sequence, and to perform the importance and uncertainty analyses. AIMS-PSA was developed to simplify PSA work. If a PSA model is provided, AIMS-PSA integrates the PSA model to build one fault tree model for an evaluation of the whole CDF, and also generates minimal cut sets for the CDF. Only a few mouse clicks are required to perform the quantification of the PSA. This helps a PSA analyst to perform PSA easily and quickly. The cut set generation engine FTREX[6], developed by KAERI, is used for AIMS-PSA. FTREX is the most powerful cut set generator that has been successfully used for many PSAs or risk monitors in Korea and the U.S.A. The accident sequence quantification for PSA can be conducted in a few seconds. FTREX has many useful features, and generates MCSs from a fault tree with circular logic and performs rule-based recovery analysis.

3. IDENTIFICATION OF ACCIDENT INITIATORS

3.1. Identification of Initiating Events

An initiating event is an event that leads to an unplanned reactor trip and requires a mitigating action on the part of automatic systems or the operator in order to maintain the safety functions of the plant. Component failures or human errors can be an initiating event. In order to identify the initiating events for PSA, the following approaches were applied after plant familiarization.

The first approach applied to identify initiating events is logical evaluation, which develops the Master Logic Diagram (MLD). The MLD is a high-level fault tree model of the potential causes of a postulated undesirable event and the logical relationships between these potential causes. A set of conceptual core damage general initiators can be identified from the MLD. The second approach for identifying initiating events is to list all postulated events from intensive reviews on design documents including Failure Mode and Effect Analysis (FMEA), safety analysis reports, and other reference to previous lists for research reactors.

3.2. Safety Functions and Corresponding Safety Systems

The design of the JRTR incorporates a number of safety functions aimed at preventing core damage following an initiating event. Table 1 shows three safety functions and corresponding safety systems in order to prevent core damage.

Table 1. Safety functions and corresponding systems

Safety Functions	Corresponding systems
Control reactivity	Reactor protection system & Alternate potection system a. First shutdown system b. Second shutdown system
Maintain primary coolant inventory	Reactor pool isolation and make up a. Siphon break valves b. Emergency water supply system
Remove core decay heat	a. Primary cooling system b. Reactor pool natural convection - Flap valves - Siphon break valves c. Emergency water supply system

3.3. Grouping of Initiating Events

Based on the responses of the corresponding safety systems, the initiating events can be grouped in such a way that all events in the same group impose essentially the same success criteria on the safety systems as well as the same specific conditions.

The following initiating events have been finally selected for the development of an event tree in the final PSA for the JRTR.

- Loss of Electric Power (LOEP)
- Reactivity Insertion Accident (RIA)
- Loss of Primary Cooling System Flow (LOPCS)
- Loss of Secondary Cooling System Flow (LOSCS)
- Loss of Coolant Accident－Out Pool (LOCA-I)
- Small Loss of Coolant Accident－In Pool (SLOCA-II)
- Large Loss of Coolant Accident－In Pool (LLOCA-II)
- Beam Tube LOCA (LOCA-III)
- General Transients (GTRNs)

For each initiating event, challenged safety functions and corresponding systems to prevent core damage are summarized in Table 2.

Table 2. Initiating Events and Relevant Safety Functions/Systems

Initiating Events		Safety function / Corresponding system			
		Reactivity	Coolant Inventory		Core cooling
			Isolation	Makeup	
1	LOEP	RPS/APS			NC
2	RIA	RPS/APS			PCS/NC
3	LOPCS	RPS/APS			NC
4	LOSCS	RPS/APS			PCS/NC
5	LOCA-I	RPS/APS	SBVs		NC
6	SLOCA-II	RPS/APS			PCS/NC
7	LLOCA-II	RPS/APS			PCS/NC
8	LOCA-III	RPS/APS		EWSS	NC/EWSS
9	GTRN	RPS/APS			PCS/NC

Safety Functions / Corresponding Systems
- Reactivity:
  • Reactor Protection System/Alternate Protection System (CRDM/SSDM)
- Coolant Inventory:
  • Isolation: Siphon Break Valves
  • Makeup: Emergency Water Supply System (EWSS)
- Core Cooling:
  • Primary Cooling System (PCS),
  • Natural Convection (NC) via Flap Valves or Siphon Break Valves (SBVs)

4. EVENT TREE ANALYSIS

Event trees are developed for the nine selected initiating events.

4.1. Loss of Electric Power (LOEP)

An event tree for LOEP represents the possible responses of the reactor to a loss of normal electric power as shown

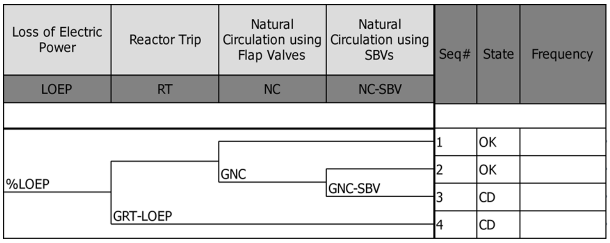

Fig. 3. Event tree for LOEP.

in Fig. 3. A loss of normal electric power is initiated by a loss of off-site grid power. If a LOEP occurs, primary cooling pumps, secondary cooling pumps and cooling tower blowers come to stop. As soon as the electrical power supply to the reactor shutdown system is cut off, the reactor power decreases rapidly by the immediate insertion of CARs and SSRs. In the beginning, the reactor core is cooled by slowing down coolant through the PCS pipe by the inertial force of the pump, flywheel, and coolant itself. As the flow through the PCS line decreases, the pressure differences across the flap valves decrease and meet the opening condition of the valves. When the flap valves open, pool water flows into the pipe that connects to the core, and a natural circulation through the flap valve is established using the pool as a huge heat sink. The siphon break valves help the core cooling by natural circulation flow and provide a diverse means for the core cooling when the two flap valves fail to open[4].

4.2. Reactivity Insertion Accident (RIA)

An event tree for a RIA models the possible responses of the reactor to reactivity insertion events, as shown in Fig. 4. If an excess reactivity is inserted, the reactor core power abnormally increases and may result in damage to the fuel plates. A RIA can be initiated by an event such as an inadvertent ejection of control rod due to operator error or a failure of the CRDM or the reactor regulating system during normal operation. If a reactor trip by the RPS or

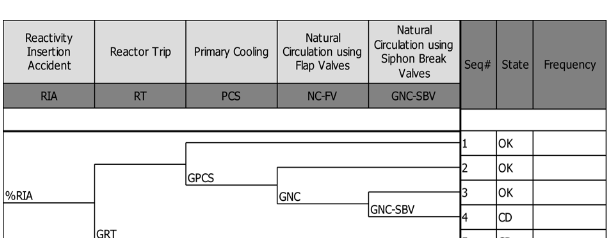

Fig. 4. Event tree for RIA.

APS succeeds, the reactor core will be cooled by the PCS pumps or the pool water natural circulation through flap valves or siphon break valves[4].

4.3. Loss of PCS Flow (LOPCS)

An event tree for a loss of PCS flow models the possible responses of the reactor to a loss of PCS flow, as shown in Fig. 5. A loss of PCS flow can occur when the PCS pumps are malfunctioning owing to a power supply or mechanical problem. The PCS flow can also be reduced when the paths are partially blocked owing to a valve closure or blockage of foreign objects. This initiating event is mainly caused by the instantaneous failure of two PCS pumps, the failure of one PCS pump, or coolant reduction. The reactor will be tripped by either a low PCS flow signal or low core Differential Pressure (DP) signal. If the reactor trip by the RPS or APS succeeds, the core decay heat is removed by forced convection of the PCS pumps. If the PCS pumps fail to run, the decay heat is removed by the natural circulation of the pool water through the flap valves or siphon break valves[4].

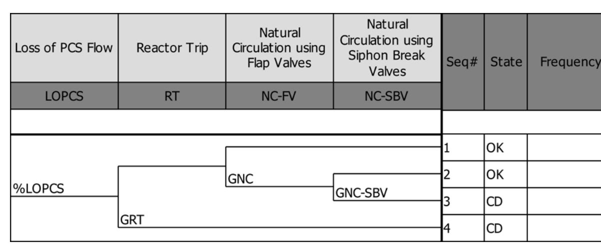

Fig. 5. Event tree for LOPCS.

4.4. Loss of SCS Flow (LOSCS)

An event tree for a loss of SCS flow models the possible responses of the reactor to a loss of SCS flow as shown in Fig. 6. A loss of SCS flow event considers a loss of cooling by the secondary cooling system during normal power operation. A loss of secondary cooling flow can be caused by failures of the secondary cooling pumps or valves, a rupture of the piping, a blockage of the flow path, or a failure of the cooling towers. If a reactor trip by the RPS or APS succeeds, the core decay heat is removed by forced convection of the PCS pumps. If the PCS pumps fail to run, the decay heat is removed by the natural circulation of the pool water through the flap valves or siphon break valves[4].

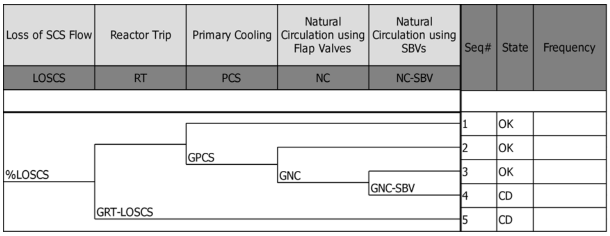

Fig. 6. Event tree for LOSCS.

4.5. Loss of Coolant Outside the Pool (LOCA-I)

An event tree for LOCA-I models the possible responses of the reactor to a loss of coolant event outside the reactor pool, as shown in Fig. 7. For ruptures of the coolant pipe outside the reactor pool, the reactor coolant spills out to the reactor hall through the break location, and subsequently the pool water level and the PCS flow decrease, but the core differential pressure increases owing to the core flow increase. When the pool water level reaches a specified level, the siphon break valves begin to open. Then air is sucked into the reactor outlet PCS pipe, and the discharge flow then decreases. Finally, the discharge flow stops completely before the pool water level reaches a prescribed water level. After the siphon flow is blocked, the reactor core is then cooled by natural circulation of the pool water through the flap valves.

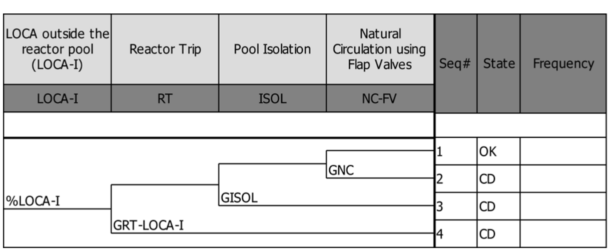

Fig. 7. Event tree for LOCA-I.

4.6. Small Loss of Coolant Inside the Pool (SLOCA-II)

LOCA-II is divided into two groups according to the cross sectional area of the broken pipe and plant response. The first one is a SLOCA-II, as shown in Fig. 8.

During normal operation, the PCS pump generates a downward flow from the pool water through the core. After passing through the core, the coolant circulates through the PCS and is discharged to the bottom of the reactor pool from the PCS discharge header. If the pipe inside the pool is ruptured, the PCS pumps draw the pool water through the rupture plane, and the core differential pressure and core flow are then reduced to a certain level according to the rupture size. However, the normal pool water level remains constant. The expected reactor trip parameter for the accidents is the low core differential pressure of the RPS. The reduction of the core differential pressure depends on the rupture size.

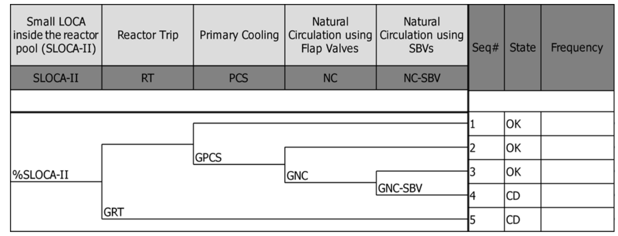

Fig. 8. Event tree for SLOCA-II.

If the rupture size is 5.3 to 9.6 inches on the in-pool PCS pipe, the core flow and core differential pressure decrease rapidly because a large amount of coolant is diverted from the core. This results in a reactor trip by the low core differential pressure signal of the RPS. Since all PCS pumps are operating normally after the reactor trip, the core decay heat is removed by forced convection of the PCS pumps. If the PCS pumps fail to run, the decay heat is removed by the natural circulation of the pool water through the flap valves or siphon break valves[4].

4.7. Large Loss of Coolant Inside the Pool (LLOCA-II)

LOCA-II is divided into two groups according to the cross sectional area of broken pipe and plant response. The second group is a LLOCA-II, as shown in Fig. 9.

It is assumed that during full power operation there is a break of the core outlet pipe inside the reactor pool with rupture sizes of 9.6 to 16 inches. According to the analysis results[4], the parts of the fuel in the core are damaged regardless of the reactor trip. Finally this accident scenario is directly progressed to a partial core damage state. In this analysis, this partial core damage can be conservatively regarded as a whole core damage state owing to the extremely low frequency.

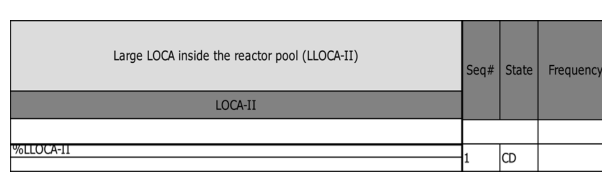

Fig. 9. Event tree for LLOCA-II.

4.8. Beam Tube LOCA (LOCA-III)

An event tree for a loss of coolant due to beam tube rupture models the possible response of the reactor to a loss of coolant owing to beam tube rupture, as shown in Fig. 10. If a beam tube and a beam port flange break down at the same time, reactor pool water is discharged to the reactor building through the broken beam port assembly. The reactor is then tripped by the low pool water level signal of the reactor protection system. The siphon break valves open automatically with the pool water level signal of the RPS, which is activated at a level of 9.0 m from the reactor pool bottom, and prevent excessive loss of the pool water inventory. Therefore, the pool water level is maintained above the flap valve even in the case of a PCS pipe rupture outside the pool. After the siphon break ends, the reactor core is cooled by natural convection of the pool water through the flap valves. When the pool water level reaches an extremely low reactor pool level of 3.8 m from the reactor pool bottom, the EWSS opens valves and injects water passively from the demineralized water supply tank into the reactor outlet PCS pipe[4].

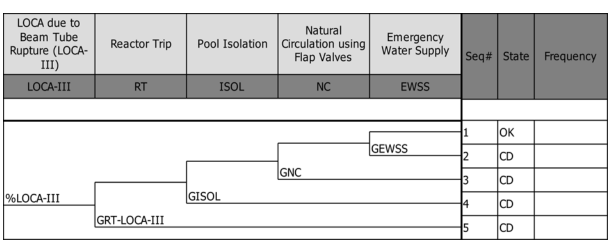

Fig. 10. Event tree for LOCA-III.

4.9. General Transients (GTRNs)

An event tree for GTRN models the possible response of the reactor to general transients, as shown in Fig. 10. GTRN involves a diverse group of non-LOCA and non-accident initiating events in which a process parameter perturbation leads to a reactor trip. If the reactor trip by the RPS or APS succeeds, the core decay heat is removed by forced convection of the PCS pumps. If the PCS pumps fail to run, the decay heat is removed by the natural circulation of the pool water through the flap valves or siphon break valves[4].

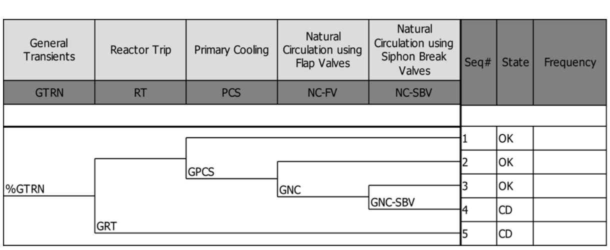

Fig. 11. Event tree for GTRN.

5. FAULT TREE ANALYSIS

Simple fault tree models for the JRTR PSA are developed instead of a detailed model at this conceptual design stage. The failures of the major components and dependencies between systems have been considered for a fault tree analysis. Normal operating trains were assumed to have a pump, a check valve, and a manual valve. The failures of the pumps and support systems such as the electrical power are modeled, and the failure of the check valve or manual valve is also modeled for the train. Of course, the Common Cause Failure (CCF) events and operator error events are modeled. Table 3 shows the system modeled in the fault tree analysis and the modeling descriptions by systems.

Table 3. System modeling

System	Modeling
Primary Cooling System	2 top logics are developed for 1) a failure of decay heat removal mode and 2) a failure of standby PCS train. CCF events, operator error and electric power failure are modeled.
Isolation of broken PCS pipe	The mechanical and signal failures of siphon break valves are modeled.
Natural Circulation in Reactor Pool	The failures of flap valves and siphon break valves are considered. Independent failure and CCF events of Valves are modeled.
RPS & APS	Three major functions are modeled for the RPS (Reactor Protection System), which are the mechanical failure of the control rods, the failure of the trip relays and the failure of trip signal. The APS is also modeled like the RPS.
Emergency Water Supply System	The failures of emergency water injection valves, their signal and electric power are considered. Also, the failure of level transmitters and their CCF are modeled.
Electrical Power System	The main power supply from the offsite grid and emergency diesel generators are considered. The failure of 460V AC or 125V DC is modeled in detail, and is modeled as a part of the 4.16 KV bus. Also, transformer or circuit breaker failure between buses is considered.

6. DATA ANALYSIS

The JRTR is in the design stage and has no operating experience. Therefore, no plant specific failure data and trip history are available. Thus the data used in our PSA are based entirely on generic data.

As component reliability data for research reactors was not available in 1988, the International Atomic Energy Agency (IAEA) published “Generic Component Reliability Data for Research Reactor PSA, IAEA-TECDOC-930[7]” for research reactor PSA in 1993.

The major generic sources used in the JRTR PSA are IAEA-TECDOC-930[7] and ALWR PRA Key Assumptions and Groundrules (KAG) of revision 7[8]. The generic database was compared with several sources during its development. The following subsections describe details of the data analysis.

6.1. Initiating Event Frequencies

This subsection discusses initiating event frequencies and provides details for obtaining each initiating event frequency and its data source. The initiating events used in the PSA were identified and listed in Section 3. The quantification of the initiating event frequency model was based generally on generic initiator frequency data. The results of the initiating event frequencies are summarized in Table 4.

Table 4. Initiating event frequencies

IE	Description	Frequencies (/yr)	Ref.
LOEP	Loss of Normal Electric Power	1.00E+01	[c]
RIA	Reactivity Insertion Accident	2.95E-02	[a]
LOPCS	Loss of PCS Flow	4.26E-01	[a]
LOSCS	Loss of SCS Flow	9.54E-02	[a]
LOCA-I	LOCA Outside the Pool	1.76E-05	[b]
SLOCA-II	SLOCA Inside the Pool	2.31E-06	[b]
LLOCA-II	LLOCA Inside the Pool	2.89E-09	[b]
LOCA-III	LOCA due to Beam Tube Rupture	1.02E-07	[a]
GTRN	General Transients	3.00E+00	[c]

[a] “KOREA Multipurpose Research Reactor Technical Report,” KM-033-RT-K066, KAERI, 1994.
[b] EPRI Generic Data
[c] Expert judgment

6.2. Component Reliability Data

The reliability data used in the final analysis are presented in Table 5. Component hardware failures imply failures of the components to function as required owing to internal defects. The common cause failure probabilities are considered. Because this research reactor is at the design stage and has no plant specific failure experience data, failure probabilities presented herein were assessed based on generic data sources[7,8,9].

Table 5. Component reliability data

Description	Data	Ref.
Bus Failure	2.30E-06/h	[a]
Trip Relay	1.00E-04	[b]
Flap Valve Fails to Open	5.00E-04	[b]
Circuit Breaker Fails to Close on Demand	1.00E-03	[b]
Motor Driven Pump Fails to Run	2.19E-06/h	[a]
Motor Driven Pump Fails to Start	1.00E-03	[a]
Motor Operated Valve Fails to Open	1.00E-04	[a]
Siphon Break Valve Fails to Open	4.50E-06/h	[a]
Heat Exchanger Fails while Operating	1.07E-05/h	[a]

[a] IAEA, Generic Component Reliability Data for Research Reactor PSA, IAEA-TECDOC-930, 1997.
[b] ALWR PRA Key Assumptions and Ground Rules, Rev.7, EPRI, 1997.

6.3. Common Cause Failure Data

The alpha factor method is used to model the Common Cause Failure (CCF) events. For components for which no experience data are available, alpha factors are assumed based on the generic values presented in KAERI/TR-2916/2005[10].

6.4. Human Error Probability Data

Post-initiator events are human errors made in response to the mitigation of an initiating event. These types of errors occur during a situation assessment or task execution after an accident, and are related to operator actions performed in response to an Emergency Operating Procedure (EOP) or recovery actions to resolve a failed safety function. Five post-initiator events were identified in the Human Reliability Analysis (HRA), as shown in Table 6. The quantification of post-initiator events was conducted basically based on the ASEP HRA procedure[11]. A conservative screening value of the Human Error Probability (HEP) was used for post-initiator events in the final PSA.

Table 6. Human error probabilities (HEPs)

Recovery Action	Description	Mean	Event Tree
EWOPV-LV0102	Operator Fails to Open EW Injection LV-001/002	1.00E-01	LOCA-III
RPOPV-RT	Operator Fails to Push Reactor Trip Button	1.00E-01	LOPCS, RIA, LOCA-I, SLOCA-II
RPOPV-RT-GN	Operator Fails to Push Reactor Trip Button at GTRN	1.00E-02	GTRN
RPOPV-RT-LIII	Operator Fails to Push Reactor Trip Button at LOCA-III	1.00E-02	LOCA-III
RPOPV-RT-SC	Operator Fails to Push Reactor Trip Button at LOSCS	1.00E-02	LOSCS

7. ACCIDENT SEQUENCE QUANTIFICATION

Because the CDF by each initiating event is quantified by the sum of all core damage accident sequences contained in each event, this section provides only a summary of their results and a description of the findings.

A total of 38 core damage accident sequences for an internal event analysis were identified and quantified using AIMS-PSA[5]. Of them, only 20 sequences were included in the CDF model for internal events. The criterion for inclusion was all sequences with a point estimate frequency greater than a truncation value of 1.0E-15/yr.

The summary of results is shown in Table 7, including the results of the contributions to the total CDF by initiating events. In addition, Fig. 12 represents the contribution to the total CDF of each initiating event by a phi chart. LOPCS is a dominant contributor to the total CDF by a single initiating event. The final quantification results indicate a point estimate of 2.02E-07/yr for the overall CDF attributable to internal initiating events for a research reactor.

The contributions of initiating events and the findings can be characterized as follows:

(1) LOPCS makes a dominant contribution to the total CDF by a single initiating event.

(2) LOPCS provides 49.4% of the total CDF. The most dominant contributor is the combination of RPS/APS failures and the failure of the operator recovery action

(3) GTRN is the second largest contributor at 32.9% due to a failure of reactor trip using RPS or APS.

(4) The third important initiating event is LOEP, providing 7.1% of the total CDF. This is due to the failure of the natural circulation using flap valves and siphon break valves.

(5) The contributions of LLOCA-II, LOSCS, LOCA-III, and SLOCA-II are relatively small, although they are the next significant contributors. These initiators contribute less than 2% of the total CDF.

The dominant MCSs are described below.

• LOPCS * APBIA-APS-B * RPBIK−PRS-ABC * RPOPV-RT

Following LOPCS, this scenario occurs when APS Ch. B computer module and RPS Ch. A/B/C computer modules containing bistables do not work simultaneously owing to the same mechanical cause. After that, the operator does not manually shutdown the reactor by pushing the manual shutdown button in the MCR. The dominant contributors to the CDF of this sequence are the CCF of the computer modules and the failure of the operator recovery action. The CDF of this sequence is about 1.62E-08/yr as a point estimate value, and this MCS provides about 8% of the total CDF.

• LOPCS * APBIA-APS-A * RPBIK−PRS-ABC * RPOPV-RT

Following LOPCS, this scenario occurs when APS Ch. A computer module and RPS Ch. A/B/C computer modules containing bistables do not work simultaneously owing to the same mechanical cause. After that, the operator does not manually shutdown the reactor by pushing the manual shutdown button in the MCR. The dominant contributors to the CDF of this sequence are the CCF of the computer modules and the failure of the operator recovery action. The CDF of this sequence is about 1.62E-08/yr as a point estimate value, and this MCS provides about 8% of the total CDF.

• LOEP * PCAVW-AV10102 * PCCVW-FLAPV

Following LOEP, this scenario occurs when two siphon break valves AV-101/102 do not work simultaneously owing to the same mechanical cause after the opening failure owing to the CCF of the flap valves. The dominant contributor to the CDF of this sequence is the combination of the opening failure of siphon break valves and the CCF of the flap valves. The CDF of this sequence is about 1.19E-08/yr as a point estimate value, and this MCS provides about 5.9% of the total CDF.

Table 7. Core damage frequencies (CDFs)

Initiating Event	IE Frequency (/yr)	CDF	%
LOPCS	4.26E-01	9.96E-08	49.4
GTRN	3.00E+00	6.65E-08	32.9
LOEP	1.00E+01	1.43E-08	7.1
LOCA-I	1.76E-05	9.85E-09	4.9
RIA	2.95E-02	6.53E-09	3.2
LLOCA-II	2.89E-09	2.89E-09	1.4
LOSCS	9.54E-02	2.11E-09	1.0
LOCA-III	1.02E-07	1.10E-10	0.0
SLOCA-II	2.31E-06	5.12E-13	0.0
Total		2.02E-07	100.0

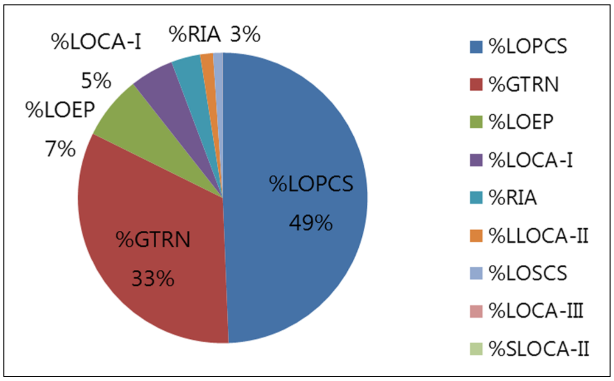

Fig. 12. Initiating event contribution to CDF.

8. CONCLUSIONS

This final PSA was undertaken to assess the level of safety for the design of the JRTR and to evaluate whether it is probabilistically safe to operate and reliable to use. The principal conclusions from this study are as follows:

- The CDF for the representative initiating events is less than 1.0E-6/yr even though conservative assumptions are used in the reliability data.
- The JRTR is well designed to be sufficiently safe from a safety stand-point. The present study indicated that the JRTR has well balanced safety with regard to each initiating event contributing to the CDF.
- The PSA methodology is very effective in improving reactor safety in a design phase, and in particular, Risk Informed Design (RID) is a very good way to find the deficiencies of a reactor under design and to improve the reactor safety by solving them.
  - According to the preliminary JRTR PSA including the relevant sensitivity analysis[12,13], (1) the improvement related to the Pressurizer Safety Valves (PSVs) was reflected in the final JRTR design, (2) the procedure improvements were recommended to the JRTR organization for the risk reduction.

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