Nuclear Engineering and Technology 49 (2017) 313–326

Available online at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

Development of an Accident Sequence Precursor Methodology and its Application to Significant Accident Precursors

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ARTICLE INFO

Article history:
Received 7 December 2016
Received in revised form
13 January 2017
Accepted 13 January 2017
Available online 11 February 2017

Keywords:
Accident Sequence Precursor
Conditional Core Damage Probability
Incremental Core Damage Probability
Operational Events
Probabilistic Risk Assessment

ABSTRACT

The systematic management of plant risk is crucial for enhancing the safety of nuclear power plants and for designing new nuclear power plants. Accident sequence precursor (ASP) analysis may be able to provide risk significance of operational experience by using probabilistic risk assessment to evaluate an operational event quantitatively in terms of its impact on core damage. In this study, an ASP methodology for two operation mode, full power and low power/shutdown operation, has been developed and applied to significant accident precursors that may occur during the operation of nuclear power plants. Two operational events, loss of feedwater and steam generator tube rupture, are identified as ASPs. Therefore, the ASP methodology developed in this study may contribute to identifying plant risk significance as well as to enhancing the safety of nuclear power plants by applying this methodology systematically.

1. Introduction

Operational events occurring at nuclear power plants provide information on the safety and reliability of these plants. Through risk assessment for operational events of a nuclear power plant, vulnerabilities can be identified and the safety of plants can be improved. Systematic management of the results of risk assessments for operational events is essential for improving the safety of plant operation and the design of new models of nuclear power plants.

Accident sequence precursor (ASP) analysis, one of methodologies of quantitative risk assessment for operational events occurring in nuclear power plants, uses probabilistic risk assessment (PRA) to systematically evaluate the risk significance of operational events and to select precursors by applying quantitative criteria. Precursors are the operational events that can cause inadequate core cooling or core damage. Systematic management of the selected precursors plays an important role in improving the safety of nuclear power plants [1].

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http://dx.doi.org/10.1016/j.net.2017.01.014

In this study, recent analyses regarding ASP were surveyed to develop a methodology to reflect the current state of the art; furthermore, the developed analysis methodology was applied to operational events occurring during full power operation as well as low power/shutdown operation.

2. Literature review

Since the establishment of the United States Nuclear Regulatory Commission (U.S. NRC) in 1979, ASP analyses have been performed, with results intended to be utilized for 35 years. Because of the great deal of analysis experience and technical development that has been accumulated, ASP analyses by the U.S. NRC have become a basis for the development of ASP analysis methodologies in other countries.

After the issuance of the WASH-1400 (Reactor Safety Study) [2], the first PRA report, the U.S. NRC formed the Risk Assessment Review Group (Lewis Committee) to provide an independent review of this report. In 1978, the Lewis Committee recommended an assessment of the risks of operational events actually occurring in nuclear power plants using the PRA methodology and, immediately after the occurrence of the TMI-2 accident in 1979, the Division of Risk Analysis of the U.S. NRC established the ASP program. Currently, the ASP program is operated by the Nuclear Operations Analysis Center of Oak Ridge National Laboratory, Oak Ridge, TS, USA; the results of the selection of precursors are documented and posted on the NRC website.

The ASP analysis status reports have been annually documented and open to the public. The first analysis report is "NUREG/CR-2497, Precursors to Potential Severe Core Damage Accidents: 1969–1979" published in 1982 [3]. For this report, Licensee Event Reports on 19,400 operational events at nuclear power plants in the U.S. between 1969 and 1979 were reviewed; 169 events that could cause core damage and severe accidents were selected and ASP analysis for these events was performed. Among the selected events, 52 events turned out to be precursors [3]. In 1984, "NUREG/CR-3591, Precursors to Potential Severe Core Damage Accidents: 1980–1981" was published and 58 precursors were selected [4]. These reports were published every year as a series of "NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents" from 1986; 17 reports were published until 2001 [5]. After the occurrence of the 9/11 attacks in 2001, the U.S. NRC has been annually posting ASP analysis results as commission papers (SECY), removing from the results information that might be sensitive with regard to U.S. security.

In addition, after the introduction of the Risk-informed, Performance-based Regulation by the U.S. NRC, the Reactor Oversight Process has been implemented since 2000 and, as part of the Reactor Oversight Process, the ASP program, the Significance Determine Process, and the MD 8.3 program have been used to assess nuclear power plant operational performance.

In the USA, 63,005 operational events were reported and evaluated from 1969 to 2005. Among them, 262 (0.42%) events were identified with conditional core damage probability (CCDP) values of 1.0×10^{-6} or more; 237 events were identified with CCDP values of 1.0×10^{-5} or more; 166 events were identified with CCDP values of 1.0×10^{-4} or more, 26 events were identified with CCDP values of 1.0×10^{-3} or more, five events were identified with CCDP values of 1.0×10^{-2} or more, and three events, including the fire at the Brown's Ferry nuclear power plant, were identified with CCDP values of 1.0×10^{-1} or more [6].

According to the report on the ASP analyses of operational events occurring over 10 years since 2005, the number of component failure-related precursors that occurred was 104, which was larger than the number of initiating event-related precursors, which was 54 [7]. Representatively, the number of operational events occurring in 2013 that were analyzed was 458 in total. Among these, a total of 17 precursors had a CCDP value greater than 1.0×10^{-6}, consisting of six initiating event-related precursors and 11 system or component failure-related precursors [6].

The AVN, the Belgium regulatory authority, introduced the PSA-based Event Analysis (PSAEA) methodology to analyze power plant operational events. The AVN has performed PSAEA for Belgian nuclear power plants since 1997 and, from its analysis, 13 operational events were selected consisting of eight component failure-related events. Among these eight component failure-related events, five events were assessed to have a CCDP value greater than 1.0×10^{-6} and these were selected as precursors. Similar to the ASP methodology, PSAEA analyzes operational events using the PRA technique. It is mainly used in European countries, including Belgium, Finland, and Switzerland. As with ASP, it selects operational events with CCDP values greater than 1.0×10^{-6} as precursors and those with CCDP values greater than 1.0×10^{-4} as important precursors [8].

In Japan, ASP analyses have been performed since 1994 at the Institute of Nuclear Safety and Nuclear Power Engineering Corporation (INS/NUPEC) with the support of the Ministry of Economy, Trade and Industry. To develop quantification models for the ASP analysis, the INS/NUPEC classified the total of 51 nuclear power plants (BWR: 28, PWR: 23) located in Japan into six types of plant and developed full power and low power/shutdown operation-related quantification models. In addition, through a review of the impact of accidents, such as cases in which the redundancy of the safety system was lost and important single failure events from the viewpoint of severe accidents, the INS/NUPEC selected 12 events from the operational events that had occurred over the past 20 years. When the selected events were analyzed, the CCDP of power operated relief valve (PORV) failure events during the steam generator tube rupture (SGTR) accidents was assessed to have the highest value (7.5×10^{-4}). The CCDP of very small loss of coolant accident (LOCA) (VS-LOCA) accidents was evaluated to have a value of 1.0×10^{-4} and failure of normal bus switching after a manual reactor outage was assessed to have a CCDP value of 1.3×10^{-6} [9].

Therefore, not only the initiating event-related risk significance, but also the component failure-related risk significance have been recognized to be important and, when the ASP methodology was developed, component failure related contents were mainly checked.

3. ASP analysis

The ASP analysis is basically performed through three steps. First, the potential impact and safety of the event should be understood in order to select, from among those events that have occurred at the power plant, events that can cause inadequate core cooling or core damage. Second, after associating the selected events with a PRA model, the PRA model should be changed to reflect the events. To simulate relevant events in the base PRA model, the components of the PRA model, such as the event tree, fault tree, initiating event frequency, component failure rate, human error probability, component restoration probability, and uncertainty parameters, should be newly modeled. Third, the issues to be considered should be derived through the results of the quantification of the changed PRA model. Through these three analysis steps, a precursor database can be constructed and, ultimately, the safety of the relevant nuclear power plant can be systematically checked and managed [10].

3.1. *Conditional CDP and incremental CDP*

A representative result of Level 1 PRA analysis of nuclear power plants is the core damage frequency (CDF), which generally indicates the average risk caused by the operation of a nuclear power plant for 1 year. More specifically, configurations of systems or components of nuclear power plants are changed due to inspections, failures, or maintenance, and the annual average CDF is quantified using the PRA model and considering such changes in the configurations for 1 year.

If it is assumed that, when an initiating event has occurred, *a* numbers of component operation success cases, *b* numbers of component operation failure cases, *c* numbers of operator action success cases, and *d* numbers of operator action failure cases exist simultaneously for a sequence *i* among *n* sequences that cause core damage, the CDF can be obtained by the following Eq. (1):

$$CDF_{IE} = f_{IE} \cdot \sum_i^n \left[\left(\prod_j^a S_{HW,j} \right) \cdot \left(\prod_k^b F_{HW,k} \right) \cdot \left(\prod_l^c S_{HE,l} \right) \cdot \left(\prod_m^d F_{HE,m} \right) \right] \tag{1}$$

where f_{IE}: frequency of the initiating event, $S_{HW,j}$: success probability of j_{th} component, $F_{HW,k}$: failure probability of k_{th} component, $S_{HE,l}$: success probability of l_{th} operator action, and $F_{HE,m}$: failure probability of m_{th} operator action.

Since component and operator action success probabilities have values very close to 1 [see Eqs. (2) and (3)], Eq. (1) can be expressed as shown in Eq. (4):

$$\prod_j^a S_{HW,j} = \prod_j^a (1 - F_{HW,j}) \cong 1 \tag{2}$$

$$\prod_l^c S_{HE,l} = \prod_l^c (1 - F_{HE,l}) \cong 1 \tag{3}$$

$$CDF_{IE} = f_{IE} \cdot \sum_i^n \left[\left(\prod_k^b F_{HW,k} \right) \cdot \left(\prod_m^d F_{HE,m} \right) \right] \tag{4}$$

where $F_{HW,j}$: failure probability of j_{th} component and $F_{HE,l}$: Failure probability of l_{th} operator action.

In the ASP analysis, as risk scales of quantitative analysis for the risk significance of initiating event and component failure event, CCDPs are considered for initiating event-related precursors. For component failure-related precursors, incremental core damage probabilities (ΔCDP) are considered instead of CCDPs, because higher CCDPs may be derived when component unavailability time is longer, and the risk significance may be estimated to be relatively higher.

The CCDPs considered for initiating event precursors can be expressed as shown in Eq. (5), where they are calculated after designating the related initiating event frequency as 1.0 [11]. Likewise, Eq. (5) can be combined with Eq. (4) and expressed as Eq. (6):

$$CCDP = \frac{CDF_{IE}}{f_{IE}} \tag{5}$$

$$CCDP = \sum_i^n \left[\left(\prod_k^b F_{HW,k} \right) \cdot \left(\prod_m^d F_{HE,m} \right) \right] \tag{6}$$

where CDF_{IE}: CDF induced from the initiating event (/y) and f_{IE}: initiating event frequency (/y).

The risk scale, ΔCDP, used for component failure event is quantified after designating the possible component failure-related basic event logic as “true”; the component failure-related precursors can affect multiple components that perform the same functions with common cause failures. Therefore, in the process of risk assessment of component failure events, the common cause failure probability also needs to be considered.

In cases in which a multiple Greek letter model is considered in a PRA model and one of the m components is unavailable, the common cause failure probabilities can be calculated with the following Eqs. (7–9) [12]:

$$(1)\ m = 2 : Q_2 = \beta \tag{7}$$

$$(2) m = 3 : Q_2 = \frac{1}{2}\beta(1-\gamma) \ \text{and}\ Q_3 = \beta\gamma \tag{8}$$

$$(3) m = 4 : Q_2 = \frac{1}{3}\beta(1-\gamma),\ \ Q_3 = \frac{1}{3}\beta\gamma(1-\delta),\ Q_4 = \beta\gamma\delta \tag{9}$$

where β: conditional probability that the common cause of a component failure will be shared by one or more additional components, γ: conditional probability that the common cause of a component failure shared by one or more components will be shared by two or more components in addition to the first, and δ: conditional probability that the common cause of a component failure shared by two or more components will be shared by three or more components in addition to the first.

As shown in the Eq. (10) below, ΔCDP can be expressed as the incremental core damage probabilities due to component failures and can be calculated by the difference between the core damage probability due to component failures and the base core damage probability [13]:

$$\begin{aligned}\Delta CDP &= P(CD|Event) - P_{CD} = CDP_{Event} - CDP_{Base} \\ &= \left(1 - e^{-CDF_{Event} \cdot t}\right) - \left(1 - e^{-CDF_{Base} \cdot t}\right)\end{aligned} \tag{10}$$

where CDF_{event}: CDF during the component failure event (/y) and CDF_{base}: CDF for the base case calculation (/y).

Since Eq. (10) above can be approximately expressed as the product of CDF and component unavailability time using Eq. (11) below, it can be reduced mathematically to Eq. (12):

$$CDP = 1 - e^{-CDF\cdot t} = 1 - \left[1 + (-CDF\cdot t) + \frac{1}{2!}(-CDF\cdot t)^2 + \cdots,\right] \cong CDF\cdot t \quad (11)$$

$$\begin{aligned}\Delta CDP &= \left(1 - e^{-CDF_{Event}\cdot t}\right) - \left(1 - e^{-CDF_{Base}\cdot t}\right) \\ &\cong CDF_{Event}\cdot t - CDF_{Base}\cdot t \\ &= \frac{T_{Event}}{A}\left(CDF_{Event} - CDF_{Base}\right)\end{aligned} \quad (12)$$

where, T_{event}: duration of the operational event (hour) and A: duration of power operation per year (h/y).

The importance of precursors can be determined based on the derived risk scale value. Operational events with CCDP or ΔCDP values greater than 1.0×10^{-6} are selected as precursors and those with CCDP or ΔCDP values greater than 1.0×10^{-3} are also screened as significant precursors.

3.2. *Full power ASP analysis*

As with PRA, the ASP analysis developed in this study was applied to the operational events occurring during full power operation. The full power ASP analysis methodology largely consists of four stages. First, based on the criteria shown in Table 1, operational events to be analyzed are selected from the operational experience of nuclear power plants.

Second, the correlation between the relevant PRA model and the selected operational events needs to be analyzed. To construct an improved PRA model for the ASP analyses, the dependency and correlation between the selected operational events and the PRA models such as the event tree, fault tree, component failure rate, and human error should be established.

Third, the relevant events should be simulated and mapped for the operational events in the PRA. The mapping includes the event tree, fault tree, and failure rate changes. The precursors are selected through quantification of the PRA model on which the operational events to be analyzed are mapped and the results of quantification are checked and reviewed to determine whether the modeling assumptions and the mapping of the PRA model are appropriate. After they are reviewed, the precursors are stored as data to construct a precursor database.

Table 1 – Screening criteria for operational events in full power.

Acceptance criteria	Rejection criteria
Complete failure of a component or a system which is essential for safety shutdown	Component failure without loss of redundancy
Degradation of a redundant system which is essential for safety shutdown	Temporary loss of redundancy in single system
Failure of reactor coolant system, instrument air, instrumentation & control system, & subsystem	An event before critical state
An event which results in core damage (LOOP, SGTR, SLOCA, etc.)	Relatively smaller design errors & quality errors than expected
An event which results in shutdown or LOFW due to degradation of a safety system	An event which does not affect a safety system
An unexpected event or an event which progresses differently on the plant design	An event which affects only core damage

LOOP, loss of off-site power; LOFW, loss of main feedwater; SGTR, steam generator tube rupture; SLOCA, small loss of coolant accident.

3.3. *Low power/shutdown ASP analysis*

The operational events occur not only during the full power operation period, but also during the low power/shutdown operation period, i.e., the period from the time when the system is disconnected from the power grid for refueling and maintenance and overhauled to the time when the reactor is initiated after refueling and the system is connected to the power grid. Plant operating states differ according to operating conditions and possible events may also be diverse. If any operational event occurs during the refueling outage except for during the period of refueling, it may cause the loss of core cooling operation and lead to core damage. Therefore, potential risks should be systematically managed through the ASP analyses, even during low power/shutdown operation. Low power/shutdown precursors should include not only operational events that may cause inadequate core cooling or core damage, but also those operational events that may cause loss of shutdown cooling operation. As with the full power ASP analysis, the low power/shutdown ASP analysis deals with initiating events and component failure events. The operational events that may occur only during the low power/shutdown operation, such as loss of shutdown cooling operation, and loss of reactor coolant system (RCS) inventory, should be additionally considered to select precursors for the ASP analysis.

When the operational events to be analyzed have been selected, the plant operational state (POS) for the relevant events should be defined. The POSs are divided according to reactor power, RCS level and temperature, opening of RCS, whether fuels have been loaded into the core, and state of various safety systems and supporting systems. For the typical OPR 1000, POSs were divided into 15 categories and a brief explanation of individual POSs is provided in Table 2; the RCS levels, pressures, and temperatures for each POS are shown in Figure 1 [14].

The PRA models used to quantify the low power/shutdown precursors are developed with different initiating events, event trees, and fault trees according to the characteristics of the POSs, as shown in Table 2. In cases in which the precursor to be analyzed is an initiating event, the type of the initiating event, unavailability of components, whether human errors should be corrected, duration of plant outage, and whether the event occurred during a forced outage period or a refueling outage period should be checked. In cases in which the precursor to be analyzed is a component failure event,

Table 2 – Plant operational states of the reference plant.

POS	Description
1	Reactor trip & subcritical operation
2	Cool-down with steam generator
3	Cool-down with shutdown cooling system
4	RCS draining for 1st mid-loop operation
5	1st mid-loop operation (installation of nozzle dam)
6	Fill for refueling
7	Fuel unloading
8	Draining for maintenance after fuel unloading
9	Fuel loading
10	RCS draining for 2nd mid-loop operation
11	2nd mid-loop operation (removal of nozzle dam)
12	RCS refilling for start-up
13	Heat-up with shutdown cooling system
14	Heat-up with steam generator
15	Reactor start-up

POS, plant operational state; RCS, reactor coolant system.

operational event-related POSs should be defined and POS duration, unavailability of components, whether the event has occurred during a forced outage period or a refueling outage period, and the time of beginning of the mid-loop operation if the event is related with the mid-loop operation should be checked. Quantification methodology, independent review, and database construction are applied in the same manner as those processes are applied during full power operation.

4. Applications

4.1. *A reference plant*

The ASP analysis was performed with the OPR1000 reactor type, which is the most frequently operated type of reactor in Korea. The OPR1000 is a 1000 MW pressurized water reactor developed in Korea and is a representative reactor type that has recorded world class performance in terms of availability and safety. The OPR1000 has been applied with the concepts of severe accident prevention and accident mitigation, as well as with redundancy, diversity, and independence of safety related facilities, based on the concept of defense-in-depth and the safe operating principle during failure. By adopting digitalized power plant protection systems and the engineered safety features (ESF) actuation system for the first time in the world, the OPR1000 also showed excellent performance in terms of reliability. The OPR1000 has a design life of 40 years and consists of two steam generators in a vertical U-tube type, four reactor coolant pumps, and one pressurizer. Basic design parameters of the OPR1000 are specified in Table 3.

4.2. *Analysis method*

The OPR1000 operation experience was examined to select the operational event to be analyzed [15,16]. As can be seen in Table 4, since 2002, 40 accidents or failures occurred in a total of eight units and, among them, 10 operational events were selected, excluding the events that did not affect the safety system, which is one of the screening criteria in Table 1.

Since it is not necessary to analyze all 10 of the operational events in this study, operational events should be additionally screened. Therefore, taking notice of the fact that the selected

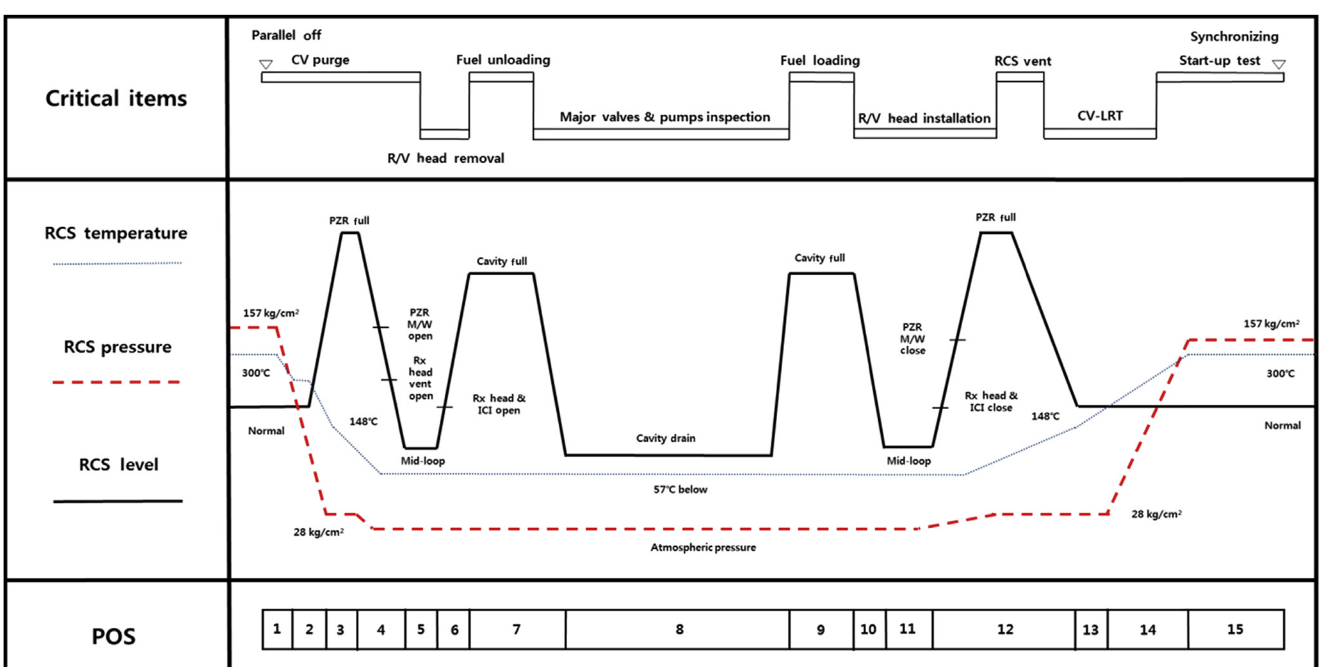

Fig. 1 – Plant operational state (POS) conditions for the reference plant.

Table 3 – Design parameters for the reference plant.

Thermal/electric power		2,825 MWt/1,000 MWe
Design life		40 yr
Seismic acceleration		0.2 g
Fuel		PLUS7
Safety requirements	CDF	$< 10^{-4}$/yr
	Thermal margin	8%
	Operator action time	min. 10 min
	Emergency core cooling	2-Train, cold leg injection
MMIS		Analog
Others	RV head area structure	Independent structure
	Reactor vessel wall cooling during severe accident	Air cooling
	RWST	Outside containment

CDF, core damage frequency; RV, reactor vessel; RWST, refueling water storage tank.

operational events were applied with the same reactor type, only one event was considered in the case of similar events and those events for which it is relatively difficult to modify the PRA model were excluded. Finally, four events were screened out including steam generator-related events and events that can cause inadequate cooling operation, such as when the voltage is too low in the safety bus to perform the ASP analysis. The final list of operational events for the ASP analysis is shown in Table 5.

In this study, the quantification was performed by using the SAREX code, a type of probabilistic safety assessment computation software [17], to reflect the differences between each event scenario in the event tree and fault tree developed during the PRA of the reference nuclear power plant. Although the units in which the individual event scenarios occurred are different, because it is difficult to apply the method to all the units, and because the same OPR1000 reactor type was applied, a model developed during the PRA analysis for Hanul Units 5 and Unit 6 was used as a representative PRA model [14].

Initiating event-related precursors were also analyzed by applying a related initiating event frequency of 1.0. In the case of component failure-related precursors, the logic of the component failure-related basic events was set to “True” and related common cause failure probabilities were recalculated. In addition, the analysis was performed considering the given unavailability time. In cases in which the accurate failure time could not be identified, the unavailability time was assumed to be half of the related test cycle for the quantification.

For the initiating event-related precursor, CCDP was derived to assess the effects of the accidents and, in the case of the component failure-related precursor, ΔCDP was calculated through importance measures instead of through CCDPs to assess the effects of component unavailability.

5. Results

Among the four selected operational events, two were initiating events consisting of one main feedwater flow rate loss accident during full power operation and one steam generator tube rupture accident during low power/shutdown operation; two further operational events were component failure events including a loss of one safety bus event under full power condition and one main steam and main feedwater isolation valve closing event due to the main steam isolation signal. The results of the ASP analysis are provided in Table 6.

5.1. *Case 1 – automatic start of emergency diesel generator due to loss of voltage at 4.16 kV safety bus “B”*

5.1.1. *Event description*

At 13:30 on May 18, 2014, during normal operation, the emergency diesel generator (EDG) started automatically due to loss of voltage (LOV) at the 4.16 kV safety bus “B”. Upon investigation, it was identified that the standby auxiliary transformer (SAT) incoming breaker opened abnormally at 13:30 after power supply for safety bus “B” was transferred from the unit auxiliary transformer (UAT) to the SAT due to abnormal closure of the breaker at 13:05 PM, and LOV at safety bus “B” occurred. Inspection on the breaker revealed that contact failure of an electronic device in a control loop card of the breaker had caused the abnormality of the card.

This event scenario was defined as a component failure event in full power operation because EDG “B” automatically started due to the occurrence of low voltage at 4.16 kV safety bus “B” due to the abnormal operation of the UAT and SAT breakers during full power operation.

5.1.2. *Calculation results*

Since the event occurred during full power operation, the relevant event scenario was applied to the full-power PRA model to perform the analysis. In the fault tree, the basic event logic was changed to “True” to consider the abnormal operation of the SAT breaker; and, since the multiple Greek letter parameters are considered in the base models, the common cause failure probability values of the two SAT breakers were changed to 0.1, according to Eq. (7).

In the case of component failure-related precursors, multiple initiating events should be simultaneously considered, because these precursors do not affect only a single initiating event. In this study, small LOCA, general transient event, loss of offsite power, steam generator tube rupture, and loss of main feedwater were considered as initiating events and full power PRA event trees were applied without any additional change. The initiating event frequencies and the basic events applied to the model are briefly outlined in Table 7; the related fault tree is shown in Figure 2.

For the relevant scenarios, which are component failure precursors, the risk was assessed using incremental core damage probabilities due to component failure. The analysis was performed using the full power model and the event scenario; according to the results, the power system was normalized only about 31 h after the event. Therefore, an unavailability time of 31 h was applied and the overall operation rate was identified as about 90%. The base CDF value was calculated at 3.43×10^{-6} per year and the CDF value, when component failures were applied, was found to be 4.11×10^{-6} per year.

Table 4 – Operational events for first screening.

No.	Reference plant	Date	Description	Power (%)	1st screening
1	#1	Apr 16, 2015	Reactor trip due to abnormal open of an RCP circuit breaker	99	X
2	#1	Oct 17, 2014	Reactor trip during power reduction for maintenance of S/G tube leak	14	X
3	#7	May 18, 2014	Automatic start of EDG due to LOV at 4.16 kV safety bus “B”	100	O
4	#7	Jan 29, 2014	Reactor trip by CEA multiple position deviation	100	X
5	#4	Aug 21, 2013	Death case of workers in a spillway during an overhaul	100	X
6	#7	July 5, 2013	Reactor trip by S/G high level	6	O
7	#3	Oct 2, 2012	Reactor trip due to PCS communication card failure	100	O
8	#4	July 30, 2012	Reactor trip due to the failure of M-G set in control element drive mechanism control system (CEDMCS)	100	X
9	#8	Nov 11, 2011	Reactor trip due to RCP trip caused by protection signal of over current relay	100	X
10	#3	Feb 4, 2011	Reactor trip due to low DNBR caused by RCP trip	100	X
11	#3	Jan 20, 2011	Reactor trip due to steam generator low level	100	O
12	#1	Feb 25, 2010	Indication of boric acid at reactor vessel head vent nozzle due to PWSCC induced crack	0	X
13	#1	Feb 17, 2010	Reactor trip due to RCP stop caused by unsuccessful power transfer while preparing annual overhaul	32	X
14	#3	Oct 23, 2009	Reactor trip due to low DNBR of CPC caused by CEDMCS trouble	100	X
15	#7	Aug 19, 2009	Fire alarm at the vitrification facility	100	X
16	#4	Dec 6, 2008	Reactor trip due to CPC low DNBR caused by RSPT malfunction	100	X
17	#2	May 27, 2008	Indication of boric acid at S/G bowl drain nozzle	0	X
18	#3	May 15, 2008	Gaseous radioactive material leakage due to inadvertent opening of condensate drain valve connected to gaseous radwaste system	100	X
19	#2	Jan 22, 2008	Loss of voltage at train “A” safety bus & subsequent EDG actuation	100	O
20	#3	Nov 22, 2007	Reactor trip due to main steam isolation during a functional test for the ESF	100	O
21	#8	July 29, 2007	Turbine & generator trip by the actuation of the differential relay of main transformer & reactor trip from the RCP trip due to delayed power transfer	100	X
22	#6	June 21, 2007	Turbine-generator trip due to the malfunction of TBN protection relay & subsequent reactor trip from DNBR-low	100	X
23	#1	June 4, 2007	Indication of boric acid leakage at the steam generator bowl drain nozzle	0	X
24	#6	May 30, 2007	Reactor trip due to the DNBR-low signal during load rejection caused by the opening of switchyard breaker	100	X
25	#3	Nov 29, 2006	Loss of off-site power during the refueling outage	0	O
26	#1	Sept 27, 2006	Reactor trip due to loss of feedwater caused by failure of the deaerator level controller	100	O
27	#6	July 18, 2006	Reactor trip due to main steam isolation signal generation during a functional test of engineered safety features	100	O
28	#8	Feb 23, 2006	Reactor trip due to the failure of 2 reactor coolant pumps	87	X
29	#5	Dec 20, 2005	Reactor trip as a result of RSPT failure	100	X
30	#5	Nov 24, 2005	Reactor trip as a result of card failure of CEDMCS	100	X
31	#5	June 23, 2005	Reactor trip due to degradation of the CRDM coil	100	X
32	#7	Jan 6, 2005	Reactor trip due to the RSPT failure	100	X
33	#7	Nov 30, 2004	Reactor trip due to CPC DNBR low caused by CEA drop	100	X
34	#3	July 13, 2004	Reactor trip & actuation of the main steam isolation signal due to the S/G water level high	100	X
35	#1	Nov 25, 2003	Reactor trip due to the DNBR low signal on RCP trip	100	X
36	#3	Aug 3, 2003	Reactor trip due to abnormal drop of part strength control element assembly	100	X
37	#2	Jan 31, 2003	Reactor trip due to the pressurizer high pressure	100	X
38	#6	Dec 29, 2002	Reactor trip due to the failure of CEDM MG set	100	X
39	#6	Apr 5, 2002	Safety injection due to S/G tube leakage	0	O
40	#2	May 3, 2001	Defect on the steam generator tube during a scheduled overhaul	0	X

CEA, control element assembly; CPC, core protection calculator; DNBR, departure from nucleate boiling ratio; EDG, emergency diesel generator; ESF, engineered safety features; LOV, loss of voltage; PCS, plant control system; PWSCC, primary water stress corrosion cracking; RCP, reactor coolant pump; RSPT, reed switch position transmitter; TBN, turbine.

Therefore, ΔCDP due to the relevant event was calculated to have a value of 2.67×10^{-9} according to Eq. (12) and steam generator tube rupture accident sequence number 36 accounted for about 22% of the entire △CDP. The relevant accident sequence is briefly described in both Table 8 and Figure 3. Since the relevant result is smaller than a value of 1.0×10^{-6}, the relevant event cannot be considered to be a precursor.

Table 5 – Final operational events for accident sequence precursor (ASP) analysis.

No.	Reference Plant	Date	Description	Operation
1	#7	May 18, 2014	Automatic start of EDG due to LOV at 4.16 kV safety bus “B”	Full power
2	#1	Sept 27, 2006	Reactor trip due to loss of feedwater caused by failure of the deaerator level controller	Full power
3	#6	July 18, 2006	Reactor trip due to main steam isolation signal generation during a functional test of engineered safety features	Full power
4	#6	Apr 5, 2002	Safety injection due to S/G tube leakage	Low power/shutdown

EDG, emergency diesel generator; LOV, loss of voltage.

Table 6 – The results of accident sequence precursor (ASP) analysis.

No.	Reference Plant	Description	Type	CCDP (ΔCDP)	Precursor
1	#7	Automatic start of EDG due to LOV at 4.16 kV safety bus “B”	Component failure	2.61×10^{-7}	N/A
2	#1	Reactor trip due to loss of feedwater caused by failure of the deaerator level controller	Initiating event	2.30×10^{-6}	Precursor
3	#6	Reactor trip due to main steam isolation signal generation during a functional test of engineered safety features	Component failure	2.67×10^{-9}	N/A
4	#6	Safety injection due to S/G tube leakage	Initiating event	1.04×10^{-4}	Precursor

CCDP, conditional core damage probability; EDG, emergency diesel generator; LOV, loss of voltage.

Table 7 – Modified events for the precursor that involves an inadequate operation of standby auxiliary transformer (SAT)/unit auxiliary transformer (UAT) feed breaker.

Event name	Description	Base probability	Current probability	Modified
%ISL	Small LOCA IE frequency	3.00×10^{-3}	3.00×10^{-3}	No
%IGTRN	General transients IE frequency	1.45	1.45	No
%ILOOP	Loss of offsite power IE frequency	2.20×10^{-2}	2.20×10^{-2}	No
%ISGTR	Steam generator tube rupture IE frequency	4.50×10^{-3}	4.50×10^{-3}	No
%ILOFW	Loss of main feed-water IE frequency	1.86×10^{-1}	1.86×10^{-1}	No
EKHBCSATB	1E 4.16kV bus SW01B FEED BKR from SAT TR02N fails to close	3.00×10^{-4}	True	Yes
EKHBIUATB	FEED BKR between UAT TR01N & 4.16kV bus SW01B open spuriously	1.44×10^{-5}	True	Yes
EKHBWSATAB	CCF of 1E 4.16kV bus FEED BRKs from SATs fail to close	3.00×10^{-5}	0.1	Yes

BKR, breaker; CCF, common cause failure; IE, initiating event; LOCA, loss of coolant accident; SAT, standby auxiliary transformer; UAT, unit auxiliary transformer.

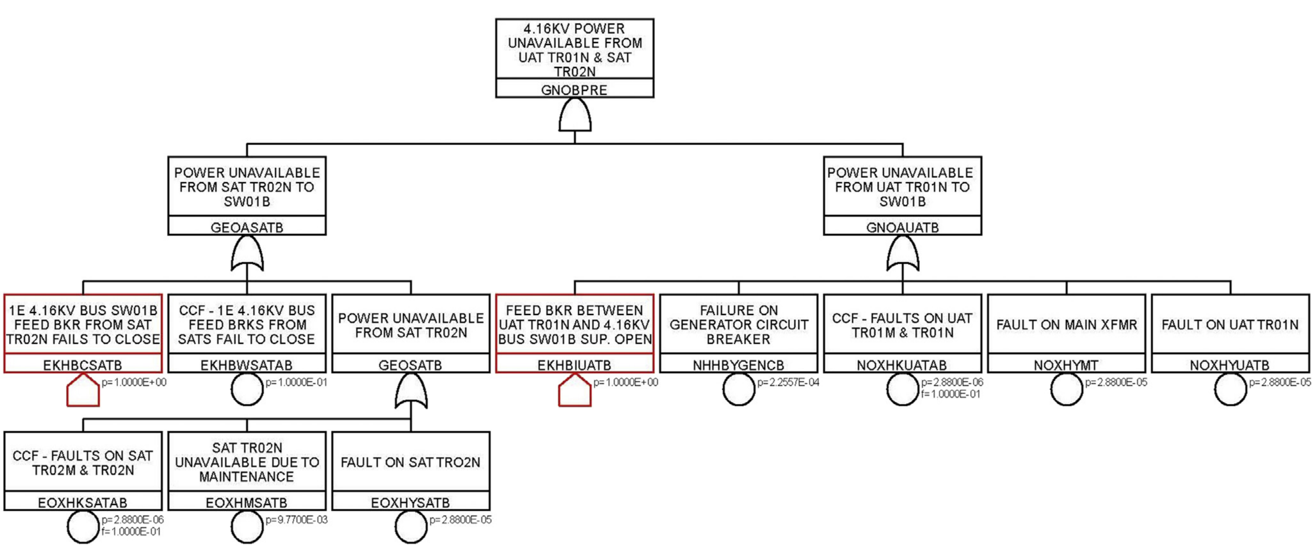

Fig. 2 – Modified fault tree for the precursor that involves an inadequate operation of standby auxiliary transformer/unit auxiliary transformer (SAT/UAT) feed breaker.

Table 8 – Dominant core damage sequence for Case 1.		
Sequence no.	Accident sequence	Contribution
SGTR-36	(1) Reactor trip (2) Failure of HPSIS injection (3) Success of depressurization for LPSIS injection (4) Failure of LPSIS injection	22%

HPSIS, high pressure safety injection system; LPSIS, low pressure safety injection system; SGTR, steam generator tube rupture.

Table 9 – Dominant core damage sequence for Case 2.		
Sequence no.	Accident sequence	Contribution
LOFW-26	1. Reactor trip 2. Failure to deliver auxiliary feedwater 3. Failure of RCS bleeding	83%

LOFW, loss of main feedwater; RCS, reactor coolant system.

5.2. *Case 2 – reactor trip due to loss of feedwater caused by failure of the deaerator level controller*

5.2.1. *Event description*

During full power operation on September 27, 2006, reference plant 1 experienced a reactor trip from the S/G low water level, which was caused by main feedwater pump trip. The initiating event was the trip of the main feedwater pumps due to a low level of the deaerator, which was caused by a failure of the level controller. After the reactor trip, the turbine-generator was also tripped and the auxiliary feedwater actuation signal was actuated as a result of the S/G low-low level, which also caused the EDG to start; however, it was not connected to the bus as the off-site power was available. Upon investigation, the controller failure was found to be due to a fault of the diode in the power supply card of the level controller. The fault of the power supply card caused both of the level transducers to fail or to produce a 100% level, and hence the operator could not take appropriate action, i.e., manual closing of the deaerator bypass valve, etc., in response to an abnormal condition of the deaerator level.

This event scenario was defined as an initiating event in the full power operation because it was related to loss of the main feedwater as an accident in which the reactor was tripped due to low steam generator level. S/G low level was resulted from main feedwater pump trip caused by rapid drops of the deaerator level due to the failure of the level controller during the full power operation.

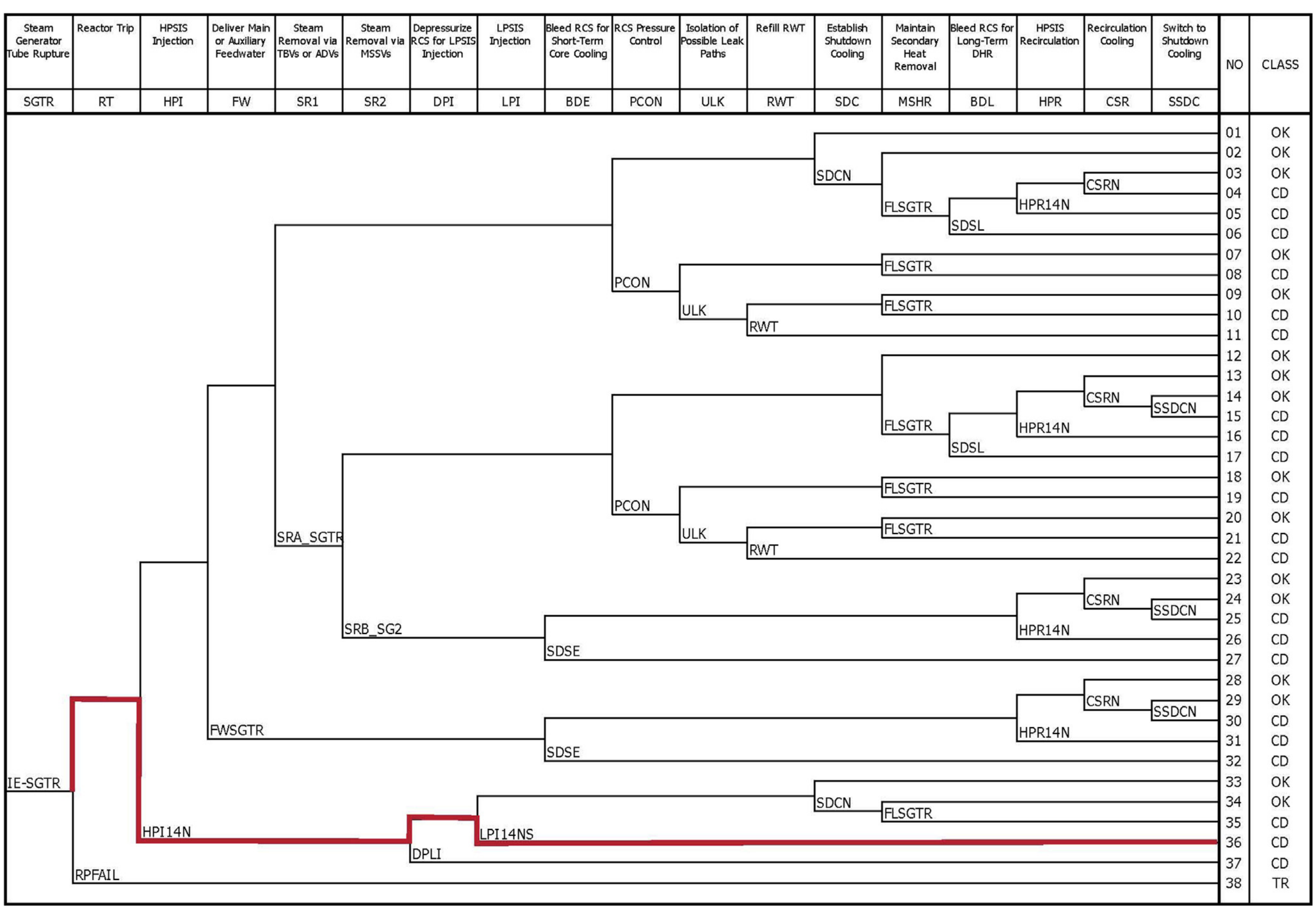

Fig. 3 – Dominant core damage sequence for the precursor that involves an inadequate operation of standby auxiliary transformer/unit auxiliary transformer (SAT/UAT) feed breaker.

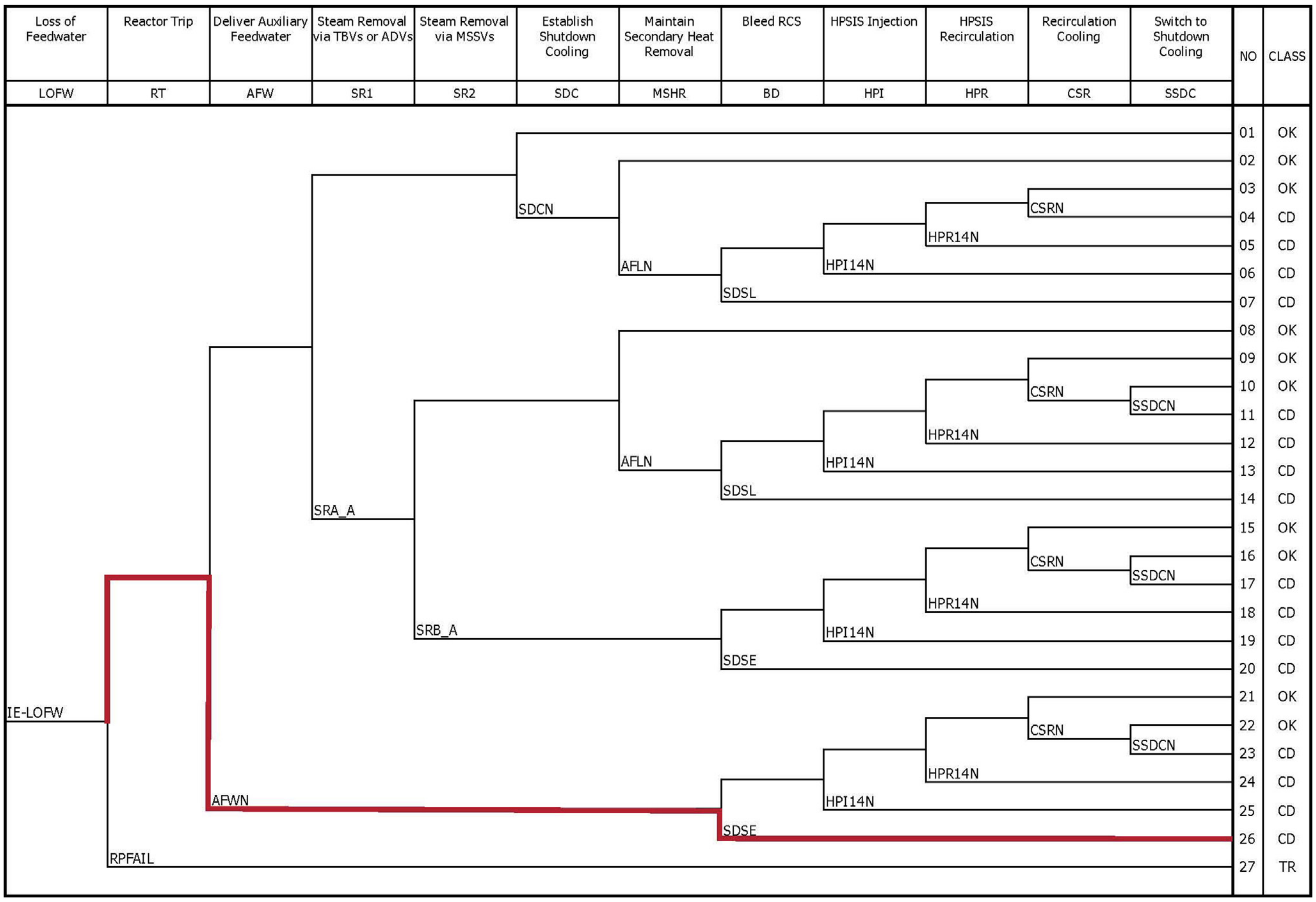

Fig. 4 – Dominant core damage sequence for the precursor that involves an initiating event of loss of main feedwater (LOFW).

5.2.2. *Calculation result*

Since this event occurred during full power operation, the analysis was performed to reflect an event scenario relevant to the full power PRA model. Since no other component failure besides the main feedwater pump trip and reactor trip occurred, the fault tree and the event tree were not additionally changed and the loss of main feedwater initiating event frequency was applied with a value of 1.0 to perform the analysis.

The relevant scenario is for an initiating event; the risk was assessed using the CCDPs. The analysis was performed using the full power model and the event scenario; in the results, the value of CCDP due to the relevant event was calculated as 2.30×10^{-6} according to Eq. (5); loss of main feedwater accident sequence number 26 accounted for about 83% of the entire CCDP. The relevant accident sequence is briefly described in both Table 9 and Figure 4. Since the relevant result is larger than 1.0×10^{-6}, the relevant event was considered to be a precursor.

Table 10 – Modified events for the precursor that involves a closure of main feedwater isolation valves (MFIV) and main steam isolation valves (MSIV).

Event name	Description	Base probability	Current probability	Modified
%ISL	Small LOCA IE frequency	3.00×10^{-3}	3.00×10^{-3}	No
%IGTRN	General transients IE frequency	1.45	1.45	No
%ILOOP	Loss of offsite power IE frequency	2.20×10^{-2}	2.20×10^{-2}	No
%ISGTR	Steam generator tube rupture IE frequency	4.50×10^{-3}	4.50×10^{-3}	No
%ILOFW	Loss of main feed-water IE frequency	1.86×10^{-1}	1.00	Yes
MFEVT131	MFIV V134 transfer closed	4.44×10^{-5}	TRUE	Yes
MFEVT132	MFIV V134 transfer closed	4.44×10^{-5}	TRUE	Yes
MFEVT133	MFIV V134 transfer closed	4.44×10^{-5}	TRUE	Yes
MFEVT134	MFIV V134 transfer closed	4.44×10^{-5}	TRUE	Yes
MSAVZTBVALL	All TBVs fail to open due to MSIV isolation	N/A	TRUE	Yes

IE, initiating event; LOCA, loss of coolant accident; LOFW, loss of main feedwater; MFIV, main feedwater isolation valve; MSIV, main steam isolation valve; TBV, turbine bypass valve.

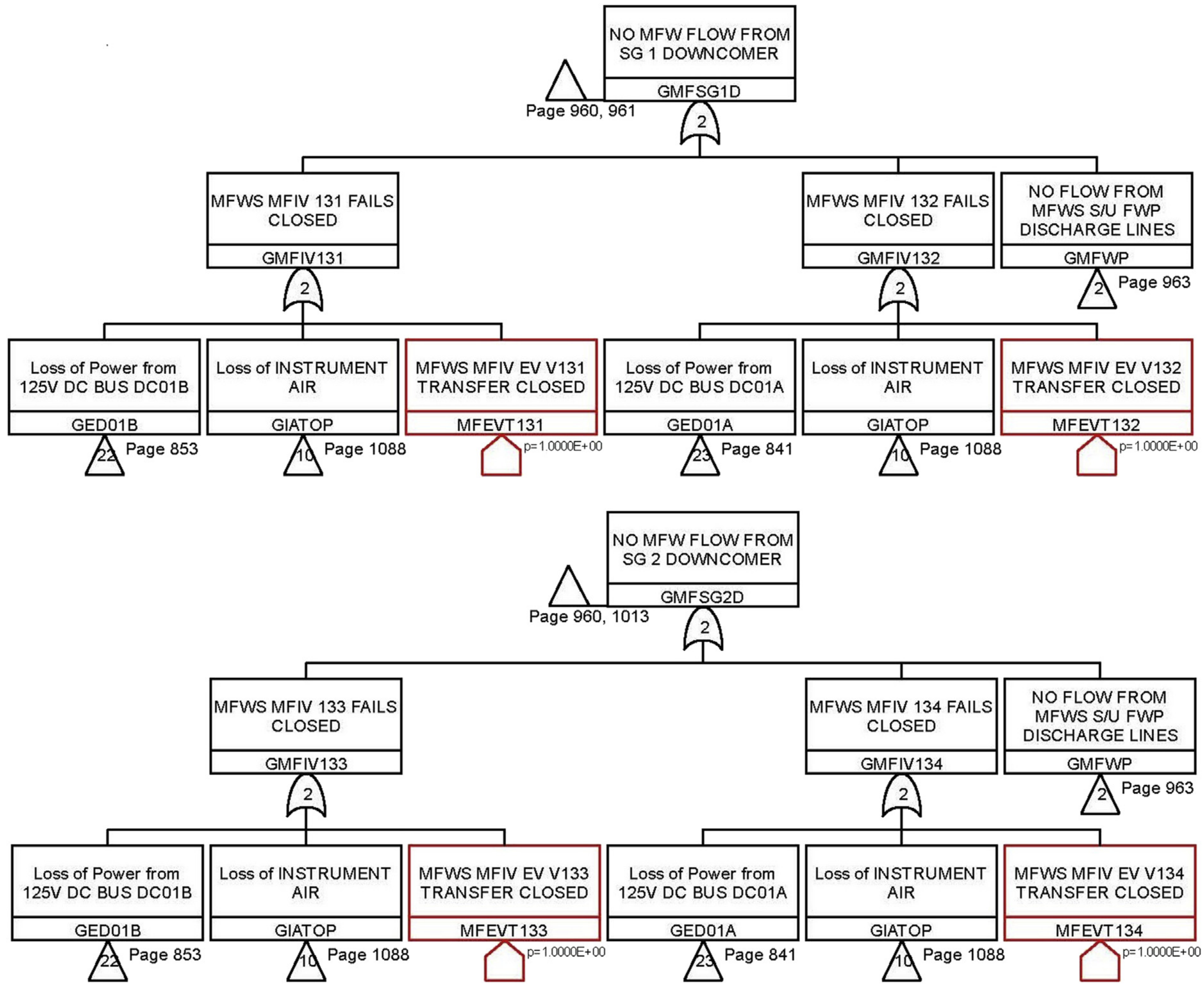

Fig. 5 – Modified fault tree for the precursor that involves a closure of main feedwater isolation valves (MFIV).

5.3. *Case 3 – reactor trip due to main steam isolation signal generation during functional test of ESF*

5.3.1. *Event description*

During a functional test of the ESF actuation logic matrix at full power, reference plant 6 experienced main steam isolation signal generation, resulting in closure of the main steam isolation valves and the main feedwater isolation valves. Subsequently, the RCS cold leg temperature increased due to the decrease of heat removal in the secondary system; the reactor was tripped by an auxiliary trip signal in the core protection calculator system. The postevent investigation results said that the main steam isolation signal was assumed to have been generated due to a bad contact of the test switch in the plant protection system test panel. The direct cause was attributed to a part defect in the test switch.

This event scenario was defined as a component failure event in full power operation because it is an event in which a reactor trip occurred due to an increase of the temperature of the primary system in the cold leg and an auxiliary trip signal in the core protection calculator system. These phenomena were caused by the decrease of heat removal in the secondary system

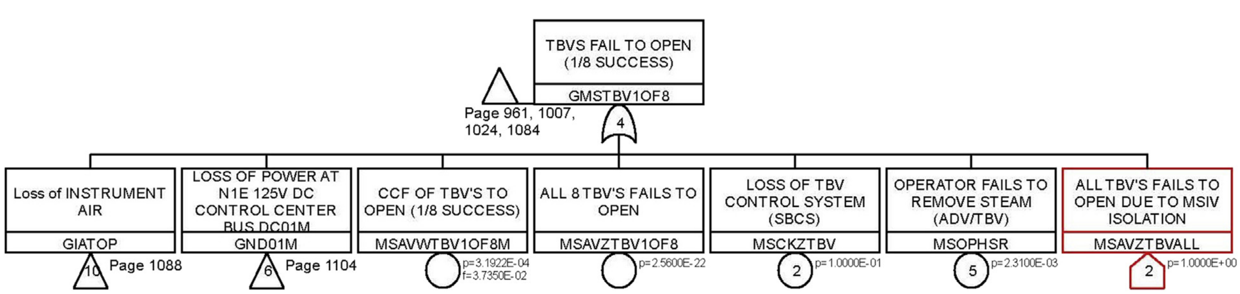

Fig. 6 – Modified fault tree for the precursor that involves a closure of main steam isolation valves (MSIV).

Table 11 – Dominant core damage sequence for Case 3.

Sequence no.	Accident sequence	Contribution
GTRN-26	(1) Reactor trip (2) Failure to deliver main or auxiliary feedwater (3) Failure of RCS bleeding	47%

GTRN, general transient; RCS, reactor coolant system.

Table 12 – Dominant core damage sequence for Case 4.

Sequence no.	Accident sequence	Contribution
SGTR-37	(1) Failure of HPSIS injection (2) Failure of depressurization for LPSIS injection	58%

HPSIS, high pressure safety injection system; LPSIS, low pressure safety injection system; SGTR, steam generator tube rupture.

resulting from the closure of the main steam isolation valve and the main feedwater isolation valve. These valves were closed due to the main steam isolation signals generated in the process of conducting a functional test of ESF actuation logic matrix.

5.3.2. *Calculation result*

Since this event occurred during full power operation, the analysis was performed by considering an event scenario relevant to the full power PRA model. In the fault tree, the basic event logic of related components was changed to "True" in order to consider the closure of the main feedwater isolation valve due to the main steam isolation signals; the basic event logic of the turbine bypass valves was changed to "True" because discharge operation through the turbine bypass valve was not possible due to the closure of the main steam isolation valve. Similarly, since this event could cause a loss of main feedwater event, the relevant initiating event frequency was changed from 0.185 to 1.0 per year.

For the case of component failure-related precursors, multiple initiating events should be simultaneously considered, because these precursors do not affect only a single initiating event. In this study, small LOCA, general transient event, loss of offsite power, steam generator tube rupture, and loss of main feedwater were considered as initiating events and the full power PRA event trees were applied without any additional change. The initiating event frequencies and basic events applied to this model are briefly outlined in Table 10; the related fault trees are shown in Figs. 5 and 6, respectively.

For the relevant scenarios that are component failure precursors, the risk was assessed using incremental core damage probabilities due to component failures. The analysis was performed using the full power model and the event scenario; according to the results, the accident occurred during a monthly periodic test. Therefore, the unavailability time was assumed to be 360 h, which is half of the test cycle; the

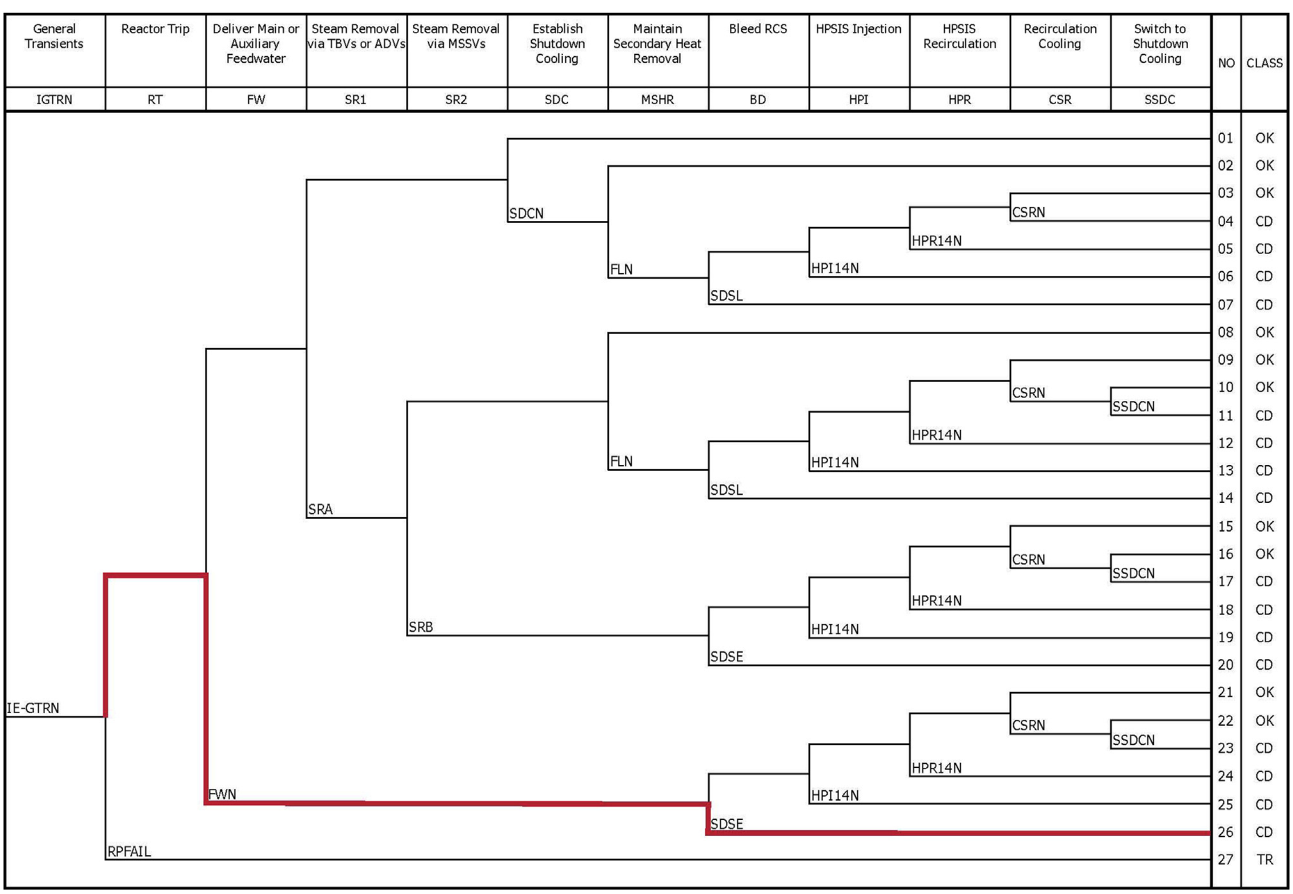

Fig. 7 – Dominant core damage sequence for the precursor that involves a closure of main feedwater isolation valves (MFIV) and main steam isolation valves (MSIV).

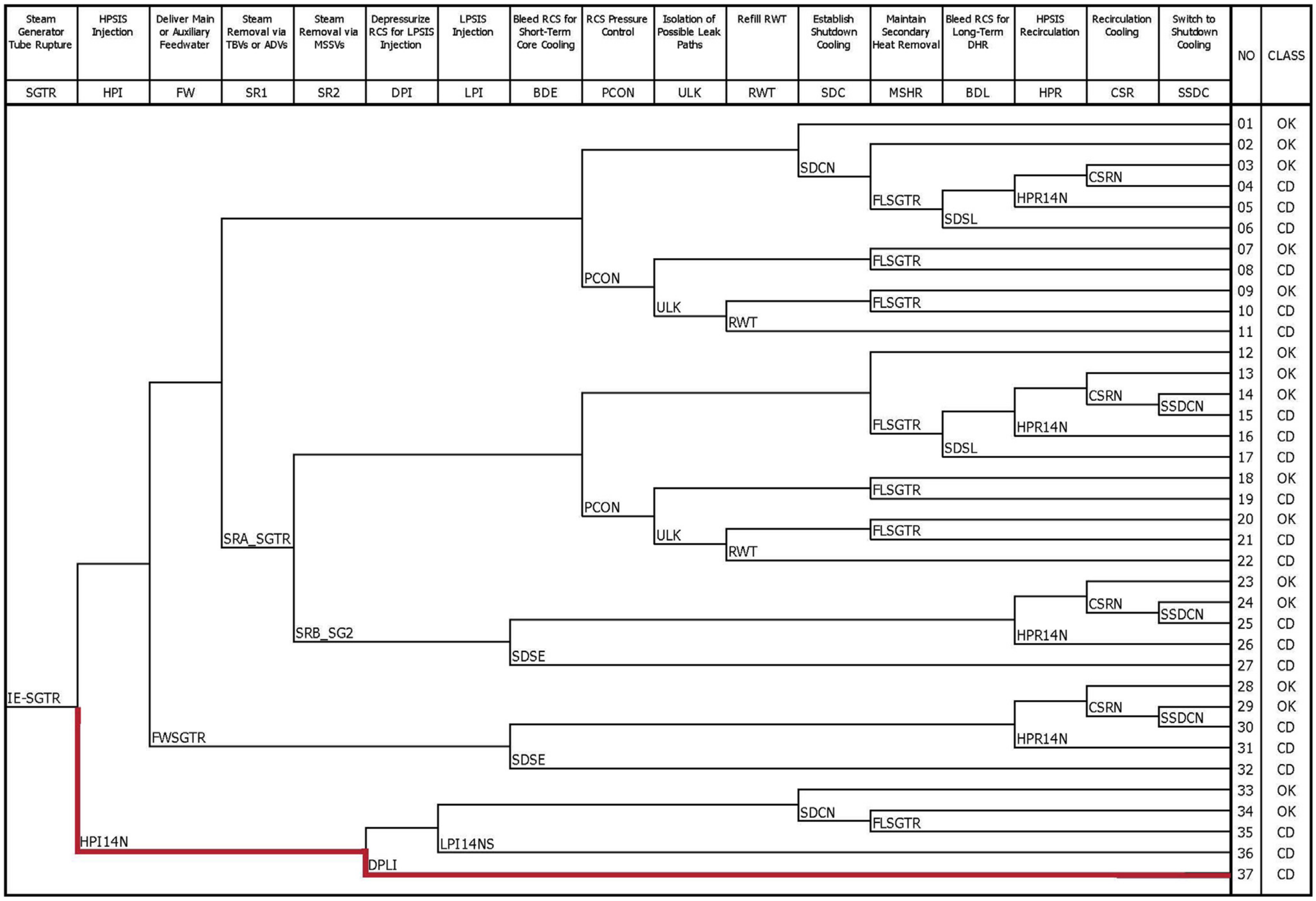

Fig. 8 – Dominant core damage sequence for the precursor that involves an initiating event of steam generator tube rupture (SGTR).

overall operation rate was found to be about 90%. The base CDF was calculated and found to have a value of 3.43×10^{-6} per year; the CDF, at which component failures were applied, was derived and found to have a value of 9.14×10^{-6} per year.

Therefore, the value of ΔCDP due to the relevant event was calculated and found to be 2.61×10^{-7} according to Eq. (12); general transient event accident sequence number 26 accounted for about 47% of the entire ΔCDP. The relevant accident sequence is briefly described in both Table 11 and Figure 7. Since the relevant result is smaller than a value of 1.0×10^{-6}, the relevant event cannot be considered a precursor.

5.4. *Case 4—safety injection due to S/G tube leakage*

5.4.1. *Event description*

Reference plant 6 was shut down on April 5, 2002, for its third refueling outage. At the beginning stage of plant cooldown, at 18:33, the operators observed a notable decrease of pressurizer level and pressure and, in response, the operator isolated the CVCS letdown by starting the third charging pump to compensate for the loss of primary coolant inventory. During the operator counter actions, at 18:46, the radiation monitors of the SG #2 sampling line set off an alarm due to the increase of the SG #2 level; this alarm alerted operators that the event was an SGTR. The operators isolated the affected SG and actuated safety injection (SI) according to the emergency operating procedure (EOP). The pressurizer level recovered with the high pressure safety injection (HPSI) flow, and the reactor was depressurized and cooled down using the intact SG to terminate the leak flow. At 19:59, the RCS and SG pressures were equalized, and the SI was terminated; this was followed by cooling down using the SG backfill operation. At 02:58 AM on the following day, the plant went into the stage of shutdown cooling system operation and exit from EOP. At 13:25, the plant reached cold shutdown condition. After the event, a close investigation was performed and the results showed that a single tube located on the hot-leg side of SG #2 had ruptured.

This event scenario was defined as an initiating event in low power/shutdown operation because it was related to steam generator tube rupture as an accident in which steam generator tubes were ruptured; the reactor was tripped and the cooling operation using the steam generator was performed.

5.4.2. *Calculation result*

Since this event occurred during a refueling outage period, the analysis was performed using the low power/shutdown PRA model to reflect the relevant event scenario. No additional change was made, because no other component failure than the steam generator tube rupture occurred and the steam generator tube rupture initiating event was applied as a value of 1.0 as the IE frequency.

For an initiating event in a relevant scenario, the risk was assessed using CCDPs. The analysis was performed using the low power/shutdown model and the event scenario; in the results, CCDP due to the relevant event was calculated to have a value of 1.04×10^{-4} according to Eq. (5); steam generator tube rupture accident sequence number 37 accounted for about 58% of the entire CCDP. The relevant accident sequence is briefly described in both Table 12 and Figure 8. Since the relevant result is larger than a value of 1.0×10^{-6}, the relevant event is considered to be a precursor.

6. Conclusion

An ASP analysis methodology was developed that can be used to determine the risk significance for nuclear power plants and that can be applied to operational events occurring in nuclear power plants in Korea. Four operational events, selected from operational experience of a reference plant, were analyzed; according to the quantitative evaluation, these events were classified into two initiating event-related operational events and two component failure-related operational events. For the case of the initiating event, the CCDP value of "Reactor Trip Due to Loss of Feedwater Caused by Failure of the Deaerator Level Controller" was estimated to have a value of 2.30×10^{-6}; the CCDP value of "Safety Injection Due to S/G Tube Leakage" was estimated to be 1.04×10^{-4}. Therefore, both events turned out to be precursors. For the case of the component failure event, "Reactor Trip Due to Main Steam Isolation Signal Generation During a Functional test of Engineered Safety Features" and "Automatic Start of EDG Due to LOV at 4.16 kV Safety Bus 'B'" events were analyzed and the corresponding ΔCDP values were calculated and found to have values of 2.61×10^{-7} and 2.67×10^{-9}, respectively. Therefore, those events did not turn out to be precursors.

ASP analysis is an assessment methodology to identify risk significance during both full power operational and low power/shutdown operational events, which may occur in nuclear power plants in Korea; this methodology can ultimately improve the safety of nuclear power plants through the improvement of plant operating procedures and additional installation of related safety systems. Although the regulation organization recognizes the importance of ASP analysis, this methodology is not yet used to properly regulate nuclear power plants. The methodology developed in this study may contribute to enhancing the safety of nuclear power plants by systematically applying ASP analysis to NPPs.

Conflicts of interest

All authors have no conflicts of interest to declare.

Acknowledgments

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety, granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (Number 1305008-0315-CG100).

REFERENCES

[1] D. Markberry, C. Hunter, G. DeMoss, U.S. Nuclear Regulatory Commission Accidents Sequence Precursor (ASP) Program Summary Description, Washington, DC, 2008.

[2] U.S. Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington, DC, 1975.

[3] W. Minarick, C.A. Kukielka, Union Carbide Corp., Nuclear Div., Oak Ridge Natl. Lab., and Science Applications, Inc., Precursors to Potential Severe Core Damage Accidents: 1969–1979, A Status Report, NUREG/CR-2497, Vols. 1 and 2 (ORNL/NSIC-18/V1 and V2), Washington, DC, 1982.

[4] W.B. Cottrell, J.W. Minarick, P.N. Austin, E.W. Hagen, and J.D. Harris, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., Precursors to Potential Severe Core Damage Accidents: 1980–1981, A Status Report, NUREG/CR-3591, Vols. 1 and 2 (ORNL/NSIC-217/V1 and V2), Washington, DC, 1984.

[5] R.J. Belles, J.W. Cletcher, D.A. Copinger, B.W. Dolan, J.W. Minarick, M.D. Muhlheim, Precursors to Potential Severe Core Damage Accidents: 1998, A Status Report, NUREG/CR-4674 (ORNL/NOAC-232, Vol. 27), Washington, DC, 2000.

[6] D. Lochbaum, Accident Sequence Precursors for Nuclear Reactors, Union of Concerned Scientists: Nuclear Energy Activist Toolkit [Internet]. 2015. Available from: http://allthingsnuclear.org/dlochbaum/accident-sequence-precursors-for-nuclear-reactors [cited: Feb 10, 2015].

[7] U.S. Nuclear Regulatory Commission, Status of the Accident Sequence Precursor (ASP) Program and the Development of Standardized Plant Analysis Risk (SPAR) Model, SECY-15-0124, Washington, DC, 2015.

[8] M. Hulsmans, P. De Gelder, Probabilistic analysis of accident precursors in the nuclear industry, J. Hazard. Mater. 111 (2004) 81–87.

[9] T. Uchida, H. Takahashi, M. Fukuda, M. Sugawara, Accident Sequence Precursor Analysis at INS/NUPEC, NEA/CSNI/R (2003) 11, Nuclear Energy Agency of the OECD (NEA), 2003.

[10] International Atomic Energy Agency, Use of Plant Specific PSA to Evaluate Incidents at Nuclear Power Plants, IAEA-TECDOC-611, International Atomic Energy Agency (IAEA), 1991.

[11] R.B. Solanki, S.K. Gupta, Application of PSA Based Operational Event Analysis to Indian Nuclear Power Plants, 2010 2nd International Conference on Reliability, Safety and Hazard - Risk-Based Technologies and Physics-of-Failure Methods (ICRESH-2010), Mumbai, 2010.

[12] R. Beutler, G. Schoen, R. Schulz, K. Theiss, I.S. Kim, A. Perez-Mulas, M. Khatib-Rahbar, Risk-Based Analysis of Operational Events for Swiss Nuclear Power Plants, NEA/CSNI/R (2003) 11, Nuclear Energy Agency of the OECD (NEA), 2003.

[13] M.B. Sattison, T.A. Thatcher, J.K. Knudsen, J.A. Schroeder, N.O. Siu, Advanced Accident Sequence Precursor Analysis Level 1 Models, NUREG/CP-0149 Vol. 2, Washington, DC, 1996.

[14] Korea Hydro & Nuclear Power, Ulchin Units 5&6 Final Probabilistic Safety Assessment Report, KHNP, Korea, 2002.

[15] Korea Institute of Nuclear Safety, 2002–2003 Operational Aspects of Nuclear Power Plants in Korea, KINS/ER-035 Vol. 3, 4, KINS, Korea, 2003–2004.

[16] Korea Institute of Nuclear Safety, 2004–2014 Safety and Operational Status of Nuclear Power Plants in Korea, KINS/ER-035 Vol. 5–15, KINS, Korea, 2005–2015.

[17] KEPCO Engineering & Construction company, Inc., SAREX User Manual Version 1.2, KEPCO-ENC, Korea, 2011.