# Dynamic Level 1 PRA of Seismic-induced Internal Flooding in Nuclear Power Plant

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*The level 1 Probabilistic Risk Analysis Study estimates the frequency of accidents that cause the reactor core damage. In general, the core damage frequency (CDF) is investigated by using event tree which depicts a system that is needed to respond to mitigate the initiating event. The event tree method requires pre-specification of an order of event occurrence which may vary according to the current plant state. Failure probability of a component also vary significantly in current status and the occurrence of each event interacts with each other. Thus, the conventional event tree approach is not applicable to the quantification of an indefinite number of the progression scenarios. In this study, a new methodology using Markov chain and Monte Carlo method is proposed to evaluate the CDF and applied a seismic-induced internal flooding event. The process of seismic-induced degradation is probably unknown in reality. A flooding model is proposed to describe the water level in each room and a propagation of flooding in the turbine building. Then, a continuous Markov chain model is applied to simulate the transition between the states of flood barriers. Also, the common cause failures of two types of the auxiliary feeding water pumps are evaluated by considering the current water level in the turbine building. Monte Carlo method is used to evaluate uncertainties of initial leak rate, broken area of the barrier. As a result, accidents scenarios initiated with a seismic-induced internal flooding are evaluated. The new methodology is a useful approach for quantification of interactive accident scenarios which consider event progression.*

## I. INTRODUCTION

A Level 1 Probabilistic Risk Assessment (PRA) models the various plant responses to an event that challenges plant operation. The plant response paths are called accident sequences. Numerous accident sequences are existed for a given initiating event. The various accident sequences result from whether plant systems operate properly or fail and what actions operators take. Some accident sequences will result in a safe recovery, and some will result in reactor core damage. The accident sequences are graphically depicted with event trees (ET). The event tree consists of an initiating event and subsequent system failure. The fault tree (FT) models the causes of the system failures. Using data on the probability of the causes, the probability of system failure is determined. Thus, the event tree method requires pre-specification of an order of event occurrence which may vary as the event progress. Also, the failure probability of an event changes significantly in current system status and the probability of a single event interacts with each other. For example, in the case of an internal flooding event, the failure probability for auxiliary feed water pump depends on the water level in the turbine building room where the pump is installed. Thus, the conventional event tree approach is not applicable to the quantification of an indefinite number of the progression scenarios.

In this study, we propose a new method of coupling Markov Chain theory to Monte Carlo method for quantification of time-dependent accident scenario and uncertainty. The new methodology is applied to a seismic-induced internal flooding event for quantification of accident scenarios which result in the core damage.

A flooding model is proposed to describe the water level in each room and a propagation of flooding in the turbine building. Also, the common cause failures of two types of the auxiliary feeding water pumps are evaluated. However, the process of seismic-induced degradation is probably unknown in the real analysis. A continuous Markov chain model is applied to simulate the transition between the states of flood barriers. Monte Carlo method is used to evaluate uncertainties of initial leak rate, broken area of the barrier.

## II. INTERNAL FLOODING PRA

### II. Outline of Internal Flooding PRA

Internal Flooding (Pipe rupture, valve rupture, etc.) is one of the initiating events which may cause the core damage in a nuclear power plant(NPP). Thus, the internal flooding event initiated by a fire hydrant and failures of equipment has been carried out by operators in worldwide [Ref.1,2]. However, a seismic-induced internal flooding PRA has less performance.

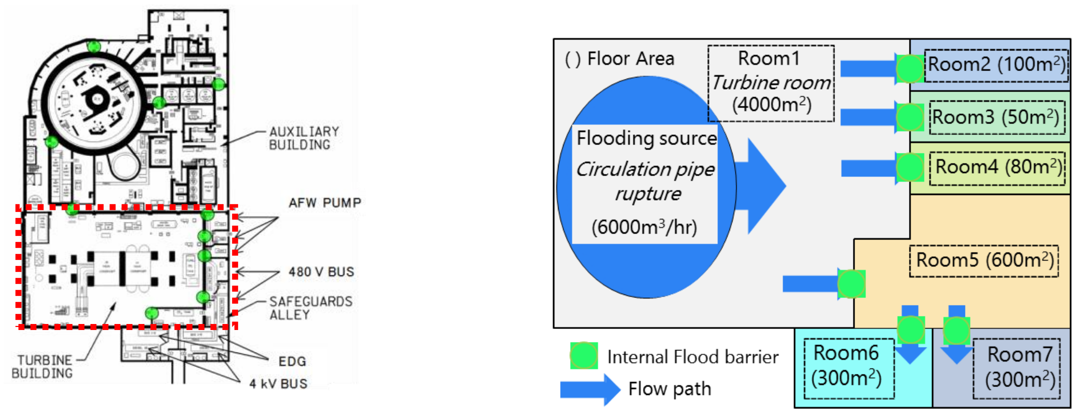
U.S. NRC report indicated a vulnerability in the NPP when an internal flooding is caused by failures of non-safety related systems such as a non-seismic piping system [Ref.3]. Thus, in this study, we estimate accident sequences, which are initiated by a seismic-induced internal flooding, result in the reactor core damage.

II.B Internal Floods Technical Elements

An important step in the internal flooding PRA process is to define flooding areas which are done in the flood are partitioning. Flood scenarios are developed by the process of flooding source analysis, flooding scenario analysis, and subsequent flooding scenario delineation and quantification.

*II.B.1 Flood Area Partitioning*

Kewaunee Power Station (KPS) is the focus of this internal flooding PRA study [Ref.4]. Figure 1-a shows the floor plan of reactor, auxiliary and turbine building. In the turbine building, various equipment such as a turbine, main steam system, condenser, and pumps is located. Thus, we focus on the internal flooding scenario in the turbine building. As shown in Fig.1-b, we partition the flood area inside the turbine building into several parts. The area of turbine room is postulated as 4,000m$^2$.




(a) Floor Plan of Kewaunee Power Station (b) Floor area partition of turbine building

Fig.1 Floor area modeling

*II.B.2 Flood Source Analysis*

A postulated flood source is a rupture of circulating system pipe which has low earthquake resistance. The leak rate is decided as 1/4 rupture of circulation system pipe in the turbine room. Based on the past research, the leak rate is 6,000 m$^3$/hr [Ref.5]. The analytical time is 1 hour. NRC's inspection on KPS indicates that some equipment such as turbine driven auxiliary feedwater pump (TDAFWP), motor driven auxiliary feed water pump (MDAFWP), emergency diesel generator (EDG), 480v BUS and 4160v is vulnerable to internal floods [Ref.6]. In this study, we focus on failures of the one TDAFWP, two MDAFWPs and one EDG installed in the turbine building by floods. The detail criteria for failure of each component will be discussed in the following chapter.

*II.B.3 Flood Scenario Analysis*

The initiating event is a loss of offsite power by the earthquake. After the loss of offsite power occurs, the reactor trip is successful. However, continuous cooling down of the primary system is necessary. Thus, a cooling down through steam generator from the secondary cooling system is attempted. The following success criteria are considered.

1) Reactor scram and operation of EDG are successful, under operation of operation of the emergency power supply system in the plant, the secondary loop system is cooled down by feeding water from the MDAFWP or TDAFWPs
2) Reactor scram is successful. However, the emergency power supply system fails, then the station blackout takes place. However, a cooling down of the reactor is possible by feeding water to the SG using the TDAFWP.

The postulated flooding paths are listed in Table I.

TABLE I. Floods propagation paths (Ref. 3)

| | |
|---|---|
| Waterproof sealing | Peeling out, Crack |

| | |
|---|---|
| Sump | Blocked |
| Flood barrier, Watertight door | Damage, Distortion |
| Floor, Wall | Creep, Crack |

There exist three major postulated internal flooding scenarios.
1. Internal flooding scenarios that can be terminated by operator action before critical flood height for equipment damage is reached.
2. Internal flooding scenarios that are not terminated early, but are limited to a single flood area.
3. Internal flooding scenarios that are not terminated early and can propagate to additional flood areas.
The 3rd scenario is adopted for accident scenario analysis. As a failure mode for AFWPs and EDG, submergence of the component by floods. By adding the failure of AFWPs and EDG by submergence to conventional event trees for internal PRA and Seismic PRA, accident scenarios which lead to reactor core damage are evaluated.

## III. FLOODS PROPAGATION MODEL

In the KPS, 30-inch-height flood barriers are installed to protect important safety-related equipment from floods. In this study, we focus on the flood barrier as flood propagation paths. As shown in Fig.2, a leak, and an overflow are the main source of floods. The leak takes place through a crack induced by seismic damage. And the overflow occurs when the water level exceeds the height of the flood barriers.

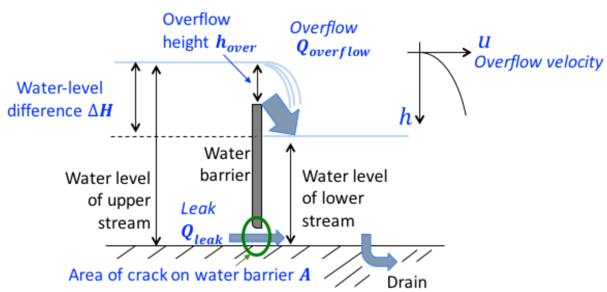



Fig.2 Internal floods modeling

### III. A. Overflow Rate

The overflow can be calculated by using overflow velocity ($u$) and height ($h_{over}$) based on the following equation.

$$Q_{overflow} = w \times \int_0^{h_{over}} u dh_{over} \quad (1)$$

Where, $Q_{overflow}$: overflow rate [m3/s], $w$: open channel width [m], $h_{over}$: overflow height [m], $u$: overflow velocity [m]
In this study, the open channel width indicates the contact are of a room of upper and downstream. The maximum width is 20m and minimum width is 5m. The overflow height($h_{over}$) is decided as following;

$$h_{over} = h_o - h_b \quad (2)$$

Where, $h_o$: water level of upper steam [m], $h_b$: height of the water barrier [m]
The overflow velocity can be derived from Bernoulli's equation as the following equation

$$u' = \sqrt{2gh} \quad (3)$$

Thus, the equation 1 can be written as follow

$$Q_{overflow} = \frac{2}{3} \times W \times \sqrt{2g} \times h_{over}{}^{3/2} \quad (4)$$

### III. B. Leak Rate

When a crack occurs at the bottom of the water barrier, like equation 4 the leak rate can be calculated according to Internal PRA guideline of Atomic energy society of Japan (Ref.1).

$$Q_{leak} = cA_{leak}\sqrt{2g(h_0\text{-}h_1)} \tag{5}$$

$Q_{leak}$: leak rate [m$^3$/s], $A_{leak}$: area of crack on water barrier [m$^2$], $g$: gravity [m/s$^2$] , c: flow rate coefficient [-], $h_0$: upper stream water level [m], $h_1$: down stream water level [m]
For the flow rate coefficient, 0.6 is used.

### III. C. Breakage of Water Barrier

Another floods source is a flow comes from a breakage of water barrier. It is assumed that breakage takes place on the water barrier by the earthquake. The postulated leak rate from the breakage of water barrier is decided based on breakage area according to experiment on leakage rate of sealed door [ref]
The Leak rate can be described based on the Bernoulli equation as follow

$$Q = CA\times\sqrt{2gh} \tag{6}$$

Where, $Q$: leakage rate from breakage of water barrier [m$^3$/sec], A: leak area, $C$: flow rate coefficient [-], $h$: height above water barrier in upper stream [m]

The mass balance of the flood propagation model between two room can be described as Eqn. 7. Based on the Eqn. 7, a flow network code is prepared.

$$\frac{\partial W}{\partial t} = G_{in}\text{-}G_{overflow}\text{-}G_{leak}\text{-}Drain \tag{7}$$

## IV. FLOOD PROPAGATION ANALYSIS

### IV.A Event Tree Analysis

The quantification of internal flooding accident scenarios is carried out by analyzing an internal flooding event tree. Based on domestic internal flooding PRA reports, the flowing event tree is prepared (Ref.1). In this study, we focus on whether the reactor core gets damaged or not by propagation floods. Thus, the accident sequences are divided into the reactor core damage and reactor core cooling. As shown in Fig.3different accident sequences are expected. Among these sequences, two paths (Sequence 3 and Sequence 5) reach to reactor core damage. The sum of the probability of each result become 1.

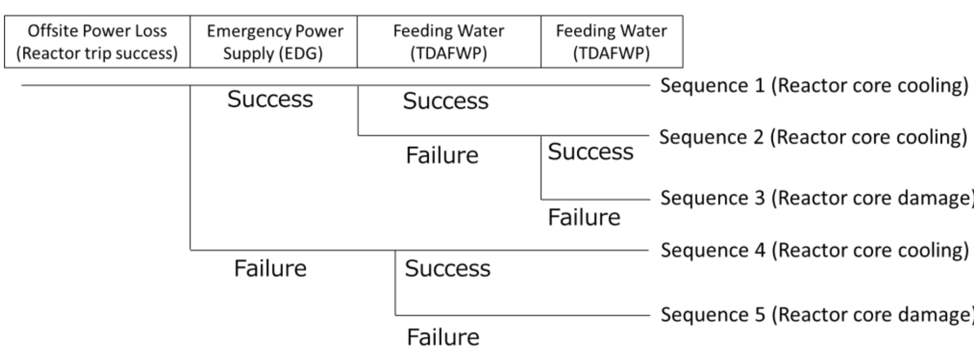



Fig.3 Event tree for internal flood

The new methodology evaluates the current system state probability sequentially using the current plant state. The current system state probability decides the system failure on the next time step. Thus, the ET is not required to determine a branch probability (Whether a system fails or not). By repeating these procedures until the end of the analysis, the accident scenario can be evaluated. The following assumptions are considered to create the event tree

- There is no equipment which gives influence on reactor trip failure in the partition of the flood area.
- Recovery of the offsite power is not considered. Once the EDG is submerged, the recovery of the EDG is impossible.

**IV.B Fault Tree Analysis**

By referring the domestic report of internal PRA, the fault trees are prepared (Ref.7). As basic events, random failure, seismic-induced failure and flood failure by submerging.

*IV.B.1 Emergency On-site Power*

Figure 4 shows a fault tree for the emergency on-site power. Two emergency diesel generators are in waiting. Since the off-site power loss is postulated, thus, the on-site power loss is regarded same as the functional loss of the EDG. Also, only the floods (submerging by water) is considered as failure causes, instrumentation and control systems are neglected. The functional failure and electrical panel failure are connected by OR gate.

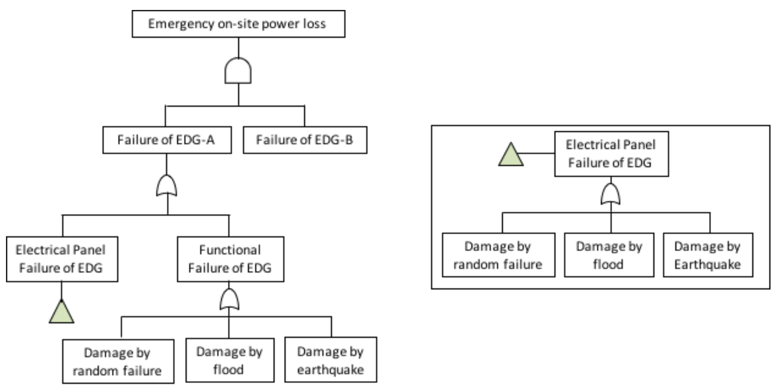



Fig. 4 Fault tree for emergency on-site pwer loss

*IV.B.2 Auxiliary Feed Water System (Ref.8)*

Auxiliary feedwater system consists of two MDAFWP and one TDAFWP. Opening operation of the electrical panel is necessary to operate the AFWP. Thus the failure of electrical panel is related to the functional failure as the or gate. Figure 5 shows a fault tree for MDAFWP and TDAF. There exist two reasons for AFWP tripping. The one is a failure of the motor operated the valve and the other is a functional failure of AFWP. Each failure mode has basic events such as damage by random failure, floods and earthquake.

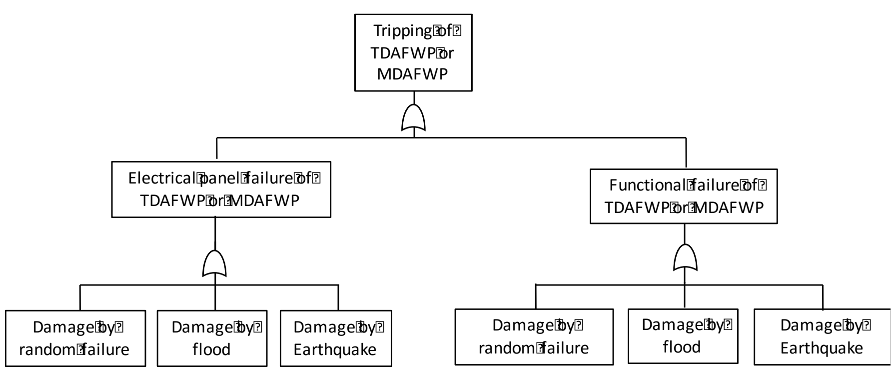



Fig.5 Fault tree for auxiliary feed water pump

*IV.B.3 Component Failure Probability*

(Seismic fragility)

The seismic failure probabilities are determined according to the domestic PRA reports (Ref.1). Seismic failure fragilities for five different ground motions (300gal, 500gal, 700gal, 1000gal and 1200gal)

(Flood failure)
When water level exceeds the flood level for each component, it is decided that the flood failure probability is 1. The flood failure probability is kept '0' until the water level reaches to the flood level.

(Random failure)
According to the reference materials, the random failure probability can be evaluated as follow (Ref.9).

$$\frac{\partial p}{\partial \mathrm{t}} = (1-p)\lambda \tag{8}$$

Where, $P$: failure probability of equipment at a certain time [-], $\lambda$: failure rate [1/sec]

### IV.C Water Level Calculation

Using the flow rate calculation model, water level in each room of the turbine room and core damage probability are evaluated for quantification of internal flooding accident scenarios. Table II shows equipment installed in the turbine building. Table III shows the height of water barrier in each room.

TABLE II List of components in turbine building rooms (Ref.3)

| | |
|---|---|
| Room1 | Turbine building room, Flood source |
| Room2 | MDAFWP-1, Motor-operated valve-1 (MOV-1) |
| Room3 | MDAFWP-2, Motor-operated valve-2 (MOV-2) |
| Room4 | TDAFWP-1, Motor-operated valve-3 (MOV-3) |
| Room5 | Electrical panel |
| Room6 | EDG-1 |
| Room7 | EDG-2, Electrical panel |

TABLE III Height of water barrier between turbine building rooms (Ref.3)

| | |
|---|---|
| Room 1 ⇒ Room 2 | 0.3 m |
| Room 1 ⇒ Room 3 | 0.8 m |
| Room 1 ⇒ Room 4 | 0.5 m |
| Room 1 ⇒ Room 5 | 0.5 m |
| Room 5 ⇒ Room 6 | 0.1 m |
| Room 5 ⇒ Room 7 | 0.2 m |

Table IV shows the water level that leads functional failure of these components. When the water level reaches to the below level, it is decided that the function of the component fall into fails.

Table IV. Flood levels impacting class I equipment (Ref.3)

| | |
|---|---|
| Turbine-driven AFW pump | 9" flood level auxiliary lube oil pump fails |
| | 18" flood level pump fails |
| Motor-driven AFW pumps | 9" flood level auxiliary lube oil pump fails |
| | 13" flood level pump fails |
| Emergency diesel generators and dedicated shutdown panel | Equipment is above 6" flood level |
| | Associated 4kV buses fail at 6" flood level |

Fig 6 shows the water level analysis in the turbine room. 300 gal is used as the input ground motion. As shown in the figure, in early stage leakage is the main source for water level increase in each room. After 2600 sec, however, the water level in room 3 rapidly increase by the overflow. It is known that the dominant source of water level increase is overflow. The analysis result well describes water level increase by internal floods.

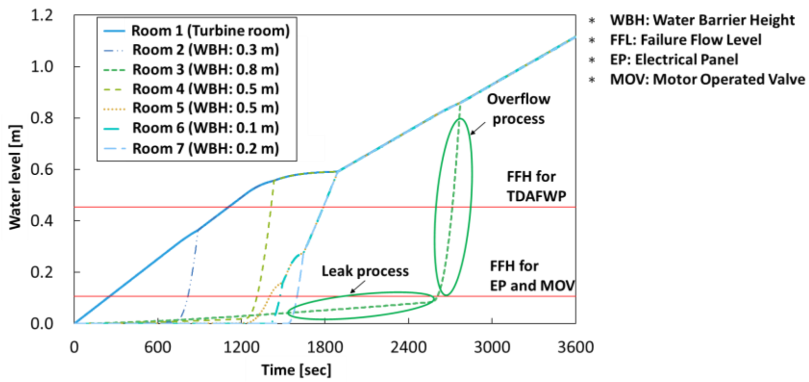
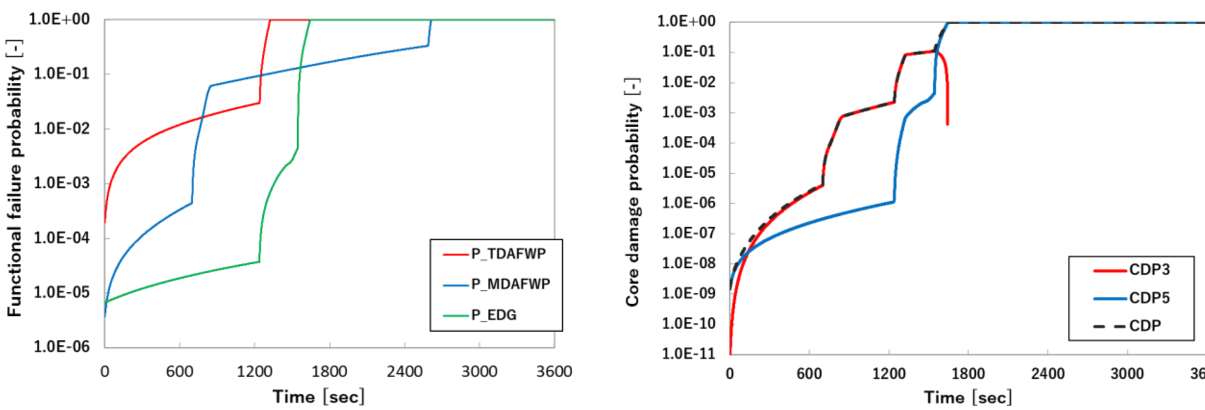



Fig 6. Flow level analysis in turbine building

Figure 7-(a) shows the expected functional failure probability (FFP) for the TDAFWP, MDAFWP and EDG. The loss of the emergency onsite power loss takes place about 1600 sec. At the same time, the TDAFW lost its function by floods. However, the MDAFW installed in the room3 is under operation until 2500 sec if the electricity is available. Figure 7-(b) indicates the reactor core damage probability which are CDP3, CDP5 and CDP (the sum of CDP3 and CDP5). When the emergency onsite power loss (Failure of the EDG) takes place, the CDP3 decreases and become zero. However, the CDP5 definitely occurs. Because the TDAFWP is already lost its function by floods. If an action such draining water from the room4 to postpone the functional failure of the TDAFWP, it can lead a mitigation of the occurrence of the reactor core damage.

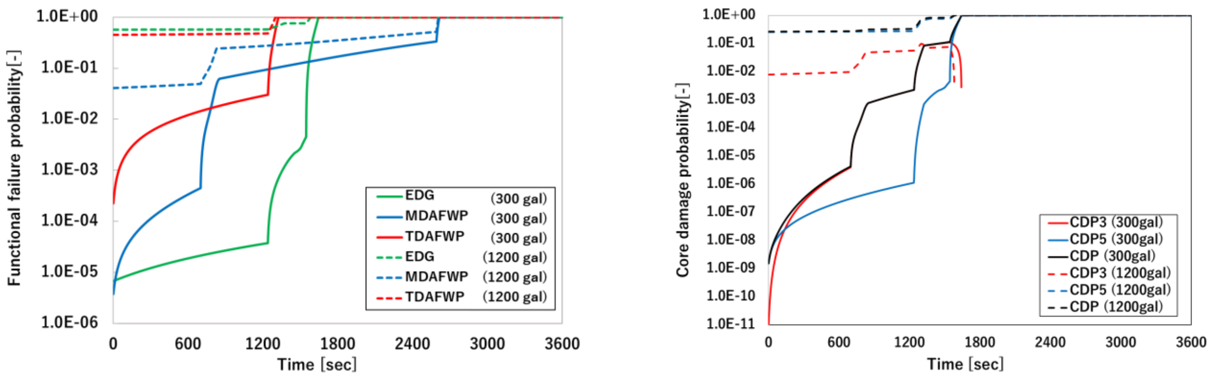



(a) Functional failure probability of each component (b) Core damage probability

Fig. 7 Functional failure probability and core damage probability (300 gal)

Then we consider how does ground motion give influence on accident progression. Figure 9 shows the function failure probability of each component and core damage probability under different ground motions (300gal and 1200gal). As shown in Fig.8-(a), under 300gal, the FFP of the MDAFWP is higher than that of EDG. However, under 1200gal, the FFP of the EDG is always higher than that of the MDAFWP. Also from the Fig.8-(b), it is known that the CDP 5 is lower than the CDP3 until the EDG failure (1600 sec) under 300 gal, however, the CDP 5 is always higher than the CDP 3 under 1200 gal. These results come out from the reason that the seismic failure is a more dominant factor of the FFH for the EDG. From these results, it is suggested that the failure mode differs under different ground motion. Accident management for seismic damage of the EDG should be considered. Also, it is known that an order of occurrence of basic events depends on plant state.

Figure 9 shows the CDP3 and CDP5 under various ground motions. When the ground motion is relatively small (300 and 500 gal), the CDP 3 is dominant accident sequence. However, when the ground motion is higher than 700 gal, the CDP 5 is dominant accident sequence. These results suggest that the EDG is vulnerable to seismic damage, thus to mitigate reactor core damage under high ground motion, an accident management for seismic damage is necessary.




(a) Functional failure probability of each component (b) Core damage probability

Fig. 8 Functional failure probability and core damage probability under various ground motions

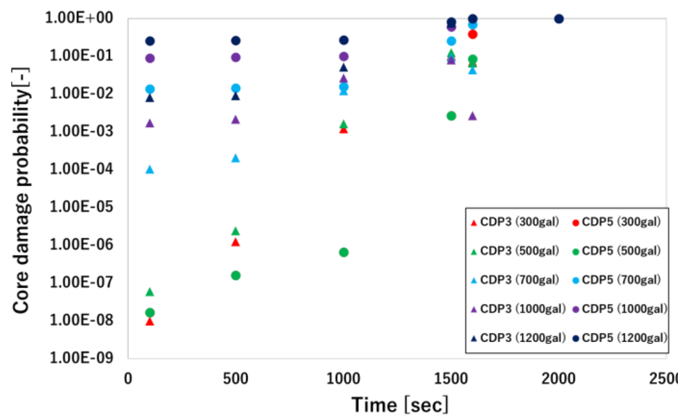



Fig. 9 Core damage probability under various ground motion

## V. CONCLUSIONS

A flow propagation model is proposed to evaluate interactive seismic-induced internal flooding accident scenario which reflects time progression. The water level in turbine building rooms is calculated when a circulating system pipe is ruptured in the turbine room. Time-dependent failure rates of some safety components are evaluated based on water level increase with time progression. Based on this information, quantification of accident scenario is carried out by evaluating the scenarios which result in core damage. Also, the new methodology gives an opportunity to evaluate the impact of safety components to accident progression. It reveals the vulnerability of each safety component that is necessary to be followed up by accident management.

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