Tsunami PRA and Its Protective Measures at the Kashiwazaki-Kariwa Nuclear Power Plant Unit 7

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Abstract: Tokyo Electric Power Company (TEPCO) has developed a probabilistic risk assessment (PRA) model to enhance safety by identifying significant accident sequences and utilizing risk information. The purpose of PRA is to analyze evaluation results to identify plant vulnerabilities and to formulate measures for improving safety.

The accident at TEPCO Fukushima Daiichi Nuclear Power Plant on March 11, 2011, resulted in a severe incident caused by the impact of a Tsunami. Drawing lessons from this accident, We have implemented various countermeasures.

This paper presents the evolution of Tsunami PRA at TEPCO, efforts to enhance the Tsunami PRA model, and the examination of protective measures based on evaluation results. As part of our recent initiatives, we also introduce case studies in which risk information has been effectively utilized.

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1. INTRODUCTION

1.1. Overview of the Kashiwazaki-Kariwa Nuclear Power Station (KK NPS)

The Kashiwazaki-Kariwa Nuclear Power Station (KK NPS) is one of the largest nuclear power plants in the world, consisting of seven reactors with a total generating capacity of 8,212 MWe. Following the 2011 accident at the TEPCO Fukushima Daiichi Nuclear Power Station (1F accident), all units were shut down. Based on lessons learned from the disaster, various safety measures—both hardware and software—have been implemented in accordance with the new regulatory standards established and enforced in 2013. Unit 6 has restarted in 2026 and began commercial operation in April 2026.

1.2. Lessons Learned from the Fukushima Daiichi Nuclear Power Plant Accident

On March 11, 2011, a magnitude 9.0 earthquake occurred, followed by a Tsunami exceeding 13 meters in height that struck the Fukushima Daiichi Nuclear Power Plant. As a result, outdoor equipment was submerged, and indoor equipment was also flooded, leading to a loss of safety-critical functions such as reactor cooling water injection and condition monitoring.

Units 1,2 and 3, which had been in operation at the time, failed to adequately cool the reactor cores after shutdown, ultimately resulting in a severe accident. A direct cause of the failure to achieve cooling was the loss of power, which made it impossible to operate and control the systems responsible for cooling.

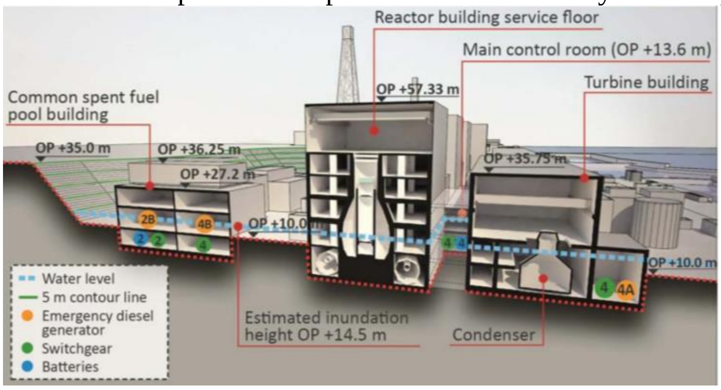

Fig.1: The elevations and locations of structures and components at the Fukushima Daiichi NPP [1]

The lessons learned are as follows:

(1) Protection against Tsunamis was inadequate.
(2) Sufficient measures for power restoration, water injection into the reactor, and cooling were not adequately prepared.
(3) Measures to prevent hydrogen explosions and to reduce the release of radioactive materials after core damage were not sufficiently in place.
Based on these lessons, efforts are being made to enhance safety from both hardware and software perspectives. [2]
(1) Strengthening preparedness for natural disasters
In addition to installing seawalls and coastal barriers, relocating important equipment to higher elevations, and improving water tightness, the spatial distribution of equipment has been enhanced to improve protection capabilities against external events, including Tsunamis beyond prior assumptions.
(2) Improving the reliability of power supply and cooling functions
Assuming severe accidents such as Station Blackout (SBO), diversification of power sources and water injection methods, along with the deployment of portable equipment, has been promoted to establish a system capable of ensuring core cooling through multiple means.
(3) Enhancing mitigation measures during severe accidents
Measures have been implemented to prevent overpressure failure of the containment vessel (e.g., hydrogen control measures and filtered venting systems). In addition, based on the concept of defense-in-depth, efforts are made to reduce the release of radioactive materials into the environment.
(4) Reforming safety culture and organizational climate
Efforts are underway to improve safety awareness at all levels—from management to frontline workers—enhance technical capabilities, and foster a proactive attitude toward identifying and sharing risks. Continuous safety improvement is pursued while incorporating external expertise.

2. Tsunami PRA

Since Japan is one of the countries where earthquakes occur frequently, research in the field of seismology has been actively conducted from an early stage. On the other hand, although the importance of guidelines for Tsunami PRA had been recognized among PRA experts, guidelines for Tsunami PRA had not been established. The Atomic Energy Society of Japan (AESJ), incorporating the opinions of experts in related fields, developed a standard for Tsunami PRA (AESJ-SC-RK004:2011) in December 2011 to define basic concepts, requirements, and standard procedures reflecting the results of methodological studies. In 2017, following further expert review, an updated standard (AESJ-SC-RK004:2016) was issued.
PRA is conducted with the aim of identifying important accident sequences in the Establishment Permission (EP) (as described in Section 2.1, “EP Model”) and enhancing safety through the effective use of risk information. Among PRA models, Tsunami PRA enables the calculation of the probability of core damage due to Tsunami events, and through analysis of the evaluation results, plant vulnerabilities can be identified. Tsunami PRA is carried out in accordance with the evaluation procedure shown in Fig. 2, which includes investigation of plant configuration and characteristics as well as site conditions, identification of accident scenarios, Tsunami hazard assessment, building and equipment fragility assessment, and accident sequence evaluation [3].
At our company, a PRA model (EP model) was developed for Units 6 and 7 of KK at the time of the installation permit application in 2013. Subsequently, in 2024, an advanced model (As-Is model) for Unit 7 incorporating the latest plant information was developed. Here, an overview of both the EP model and As-Is model is presented.

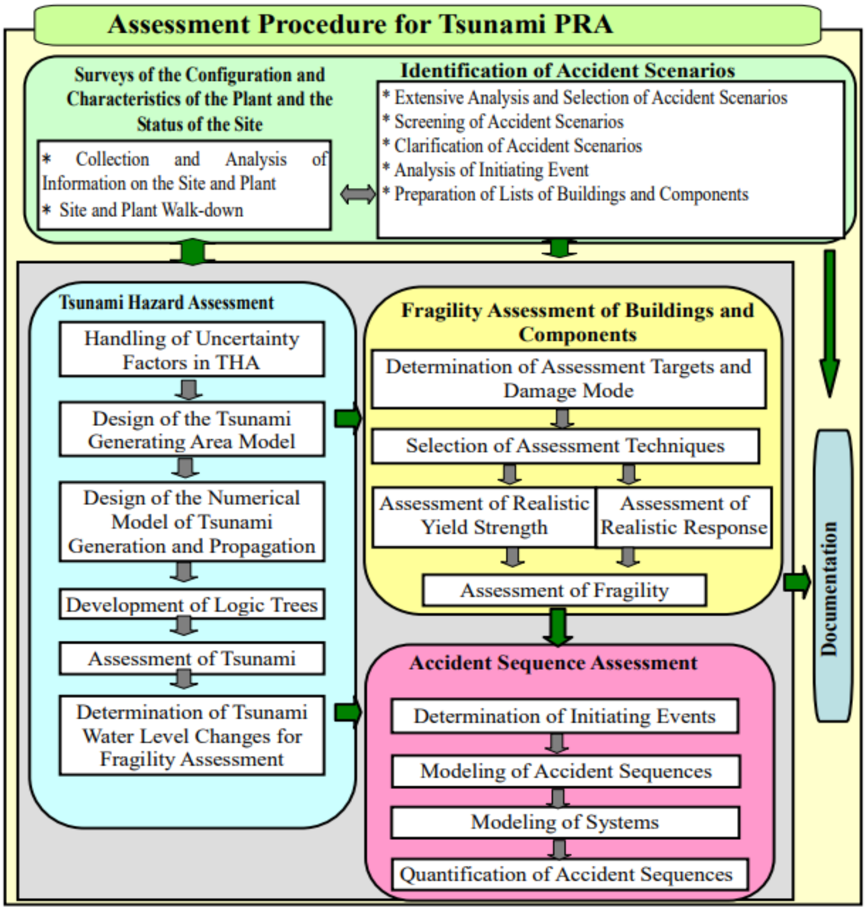

Fig.2: Assessment Procedure for Tsunami PRA[3]

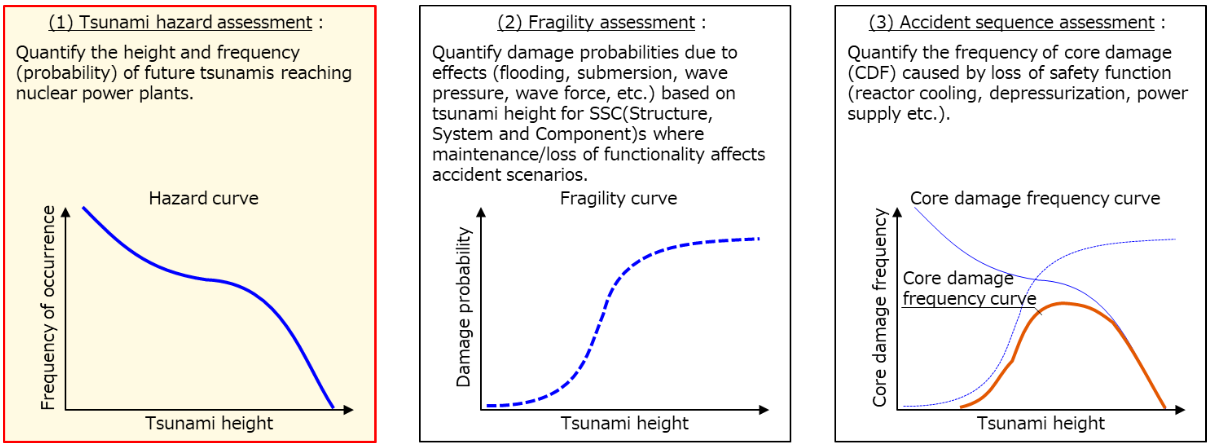

Fig. 3: Hazard assessment (left), fragility assessment (center), and accident sequence assessment (right).

2.1. EP model

In Japan, when an operator intends to install a nuclear power plant for power generation, it must submit an application for the establishment permission of a nuclear reactor to the Nuclear Regulation Authority (NRA) and obtain its approval in accordance with the "Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors."

NRA examines conformity with the following five criteria:

1.There is no risk that the nuclear reactor for power generation will be used for purposes other than peaceful ones.
2.The applicant possesses the necessary technical capability and financial basis required to install the nuclear reactor for power generation.
3.The applicant has the technical capability to implement measures necessary to prevent the occurrence and escalation of severe accidents, as well as other capabilities sufficient to properly carry out the operation of the nuclear reactor for power generation.
4.The location, structure, and equipment of the nuclear reactor facility conform to the standards specified by NRA regulations to ensure that they do not hinder the prevention of disasters caused by nuclear fuel material, materials contaminated by nuclear fuel material, or the nuclear reactor itself.
5.The quality management system conforms to the standards specified by NRA regulations.

In 2013, we submitted an application for EP and conducted PRA for internal events, seismic events, Tsunamis, and shutdown conditions with the objective of selecting significant accident sequences. For the Tsunami PRA, a PRA model (EP model) was developed in accordance with the AESJ 2011 standard.

(1) Investigation of Plant Configuration, Characteristics, and Site Conditions

The plant configuration, characteristics, and site conditions were investigated through document reviews and plant walkdowns. In the EP application, only the equipment included in the application was subject to evaluation, with the assumption that accident management measures would not be credited.

Therefore, even measures considered effective for Tsunami countermeasures—such as seawalls, watertight doors, and sealing of penetrations—were not taken into account.

(2) Identification of Accident Scenarios

Based on the collected information, accident scenarios leading to core damage due to Tsunami events were identified. As a result of the analysis, the initiating events induced by Tsunamis were identified as the following five events:

・Loss of offsite power

・Loss of DC power supply

・SBO

・Loss of ultimate heat sink (LUHS)

・Transient events (loss of all feedwater functions)

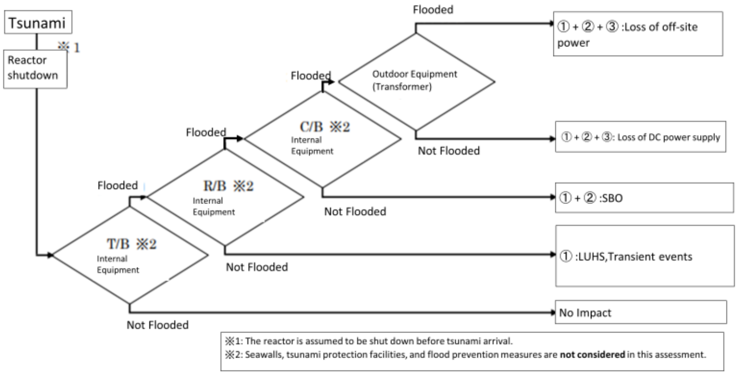

Fig.4: Flow of Initiating Event Identification

(3) Tsunami Hazard Assessment

In Tsunami PRA, a hazard assessment is conducted to calculate the frequency of Tsunami occurrence. At KK NPS, the frequency at which Tsunamis of arbitrary heights strike the site is evaluated in terms of annual exceedance probability.

Tsunamis generated by earthquakes are considered in the assessment, and the analytical procedures are based on the AESJ Tsunami PRA standard as well as the "Probabilistic Tsunami Hazard Analysis Method" (Japan Society of Civil Engineers, 2011).

Epistemic uncertainties, such as earthquake magnitude, mean recurrence interval, and source modeling, are taken into account. For aleatory uncertainties, the distribution of variability in Tsunami water levels is modeled using a lognormal distribution.

The logic tree and branch weights were established by incorporating insights gained from 1F accident.

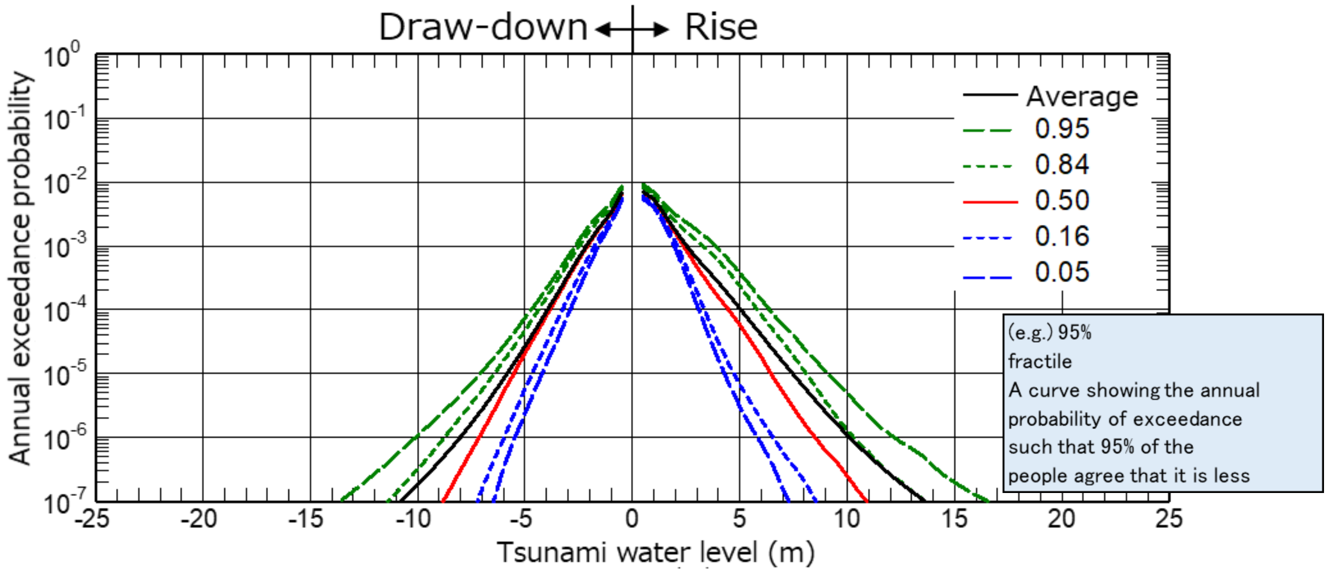

Fig.5: Tsunami Hazard Curve

(4) Building and Equipment Fragility Assessment

A fragility assessment was conducted to evaluate the capacity of equipment to withstand Tsunami impacts. Based on the information obtained through the collection and analysis of plant data, the damage modes leading to functional failure for selected buildings and equipment were examined. As a result, the damage modes of "water exposure and submergence" for dynamic and electrical equipment were identified as evaluation targets.

For these damage modes, it was assumed that "water exposure and submergence" occur when the Tsunami height exceeds both the building opening elevation (points where water can enter) and the installation elevation of the equipment. The equipment fragility is shown in Fig. 6.

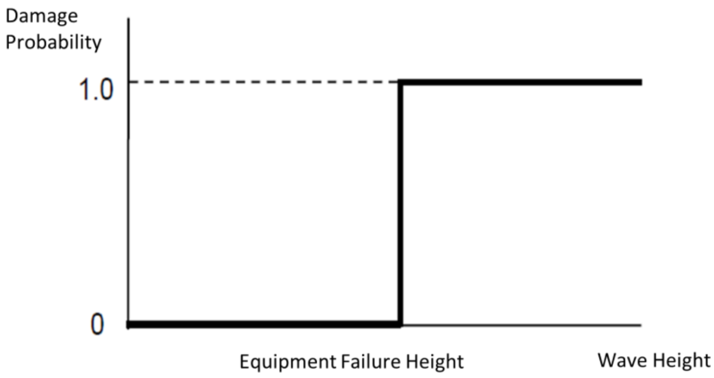

Fig.6: Equipment Fragility

(5) Accident Sequence Evaluation

After conducting the hazard assessment and fragility assessment, initiating events and the SSCs (Structures, Systems, and Components) that lead to these events were identified based on damage to SSCs caused by the Tsunami. The initiating events induced by Tsunami-related facility damage are as follows:

・Loss of offsite power: LOPA

・Loss of ultimate heat sink: LUHS

・Station blackout (reactor auxiliary cooling system: intact): SBO1

・Station blackout (reactor auxiliary cooling system: failed): LUHSSBO

・Station blackout + loss of RCIC and HPAC functions: SBO2

・Loss of DC power and loss of instrumentation and control functions: SBO3

For the defined initiating events, hierarchical structuring (evaluation using hierarchical event trees) was performed to rationally assess Tsunami-specific effects. In this structuring, attention is focused on the extent of impact associated with each initiating event.

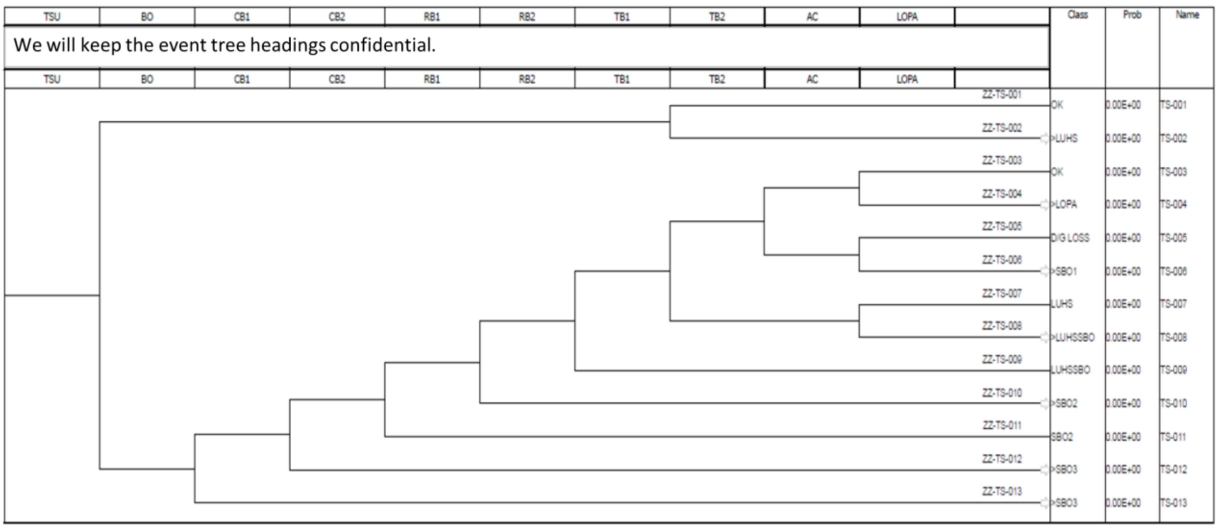

Fig.7: Hierarchical Event Tree

Fig. 8 shows the evaluation results of the EP model. As part of a sensitivity analysis, the Tsunami PRA evaluated cases in which severe accident (SA) measures and Tsunami countermeasures (such as seawalls and watertight doors) are assumed to be effective.
By incorporating SA measures, Tsunami protection measures—such as seawalls and watertight doors— the core damage frequency (CDF) decreased to about one-thousandth.

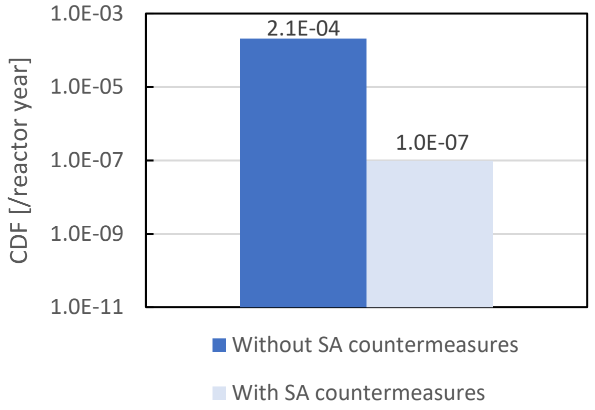

Fig.8: PRA Results

2-2 As-Is Model

Currently, our company is making various efforts to enhance PRA as an important method for plant risk assessment. In the past, a gap in the level of detail between PRA models used by domestic operators, including our company, and those used in the United States was identified, and efforts have been made to bridge this gap. In 2018, our company began enhancing and developing an "as-is" Level 1 PRA model for internal events during operation.
Based on the knowledge gained through the enhancement and as-is development of the internal events Level 1 PRA, as well as the "Implementation Standard for Probabilistic Risk Assessment for Nuclear Power Plants Initiated by Tsunami" issued by the Atomic Energy Society of Japan, a Tsunami Level 1 PRA model (As-Is model) has been developed. As-Is model incorporates accident management (AM) measures, as well as Tsunami countermeasures introduced after 1F accident—such as seawalls and watertight doors—and severe accident (SA) equipment.
(1) Enhancement of Fragility Assessment
One notable difference from the EP model is the enhanced fragility assessment. Specifically, run-up analysis has been incorporated, and the scope of evaluation has been expanded to include penetration areas.
In the fragility assessment of buildings and equipment, fragility curves were developed by evaluating the probability distributions of realistic capacities and responses leading to functional failure due to Tsunami effects such as water exposure and submergence, hydrodynamic forces, scouring, debris impact, and seabed sediment movement.
For the probability distributions, bounded distributions are assumed for functional damage modes due to water exposure and submergence, while lognormal distributions are assumed for structural and functional damage modes caused by hydrodynamic forces, scouring, debris impact, and seabed sediment movement.
Since the damage limit is evaluated as a random variable, uncertainty in the damage limit is assessed using probabilistic analysis methods in which material properties and similar parameters are treated as random variables. These uncertainties are categorized into those associated with aleatory uncertainty factors and those associated with epistemic uncertainty factors.

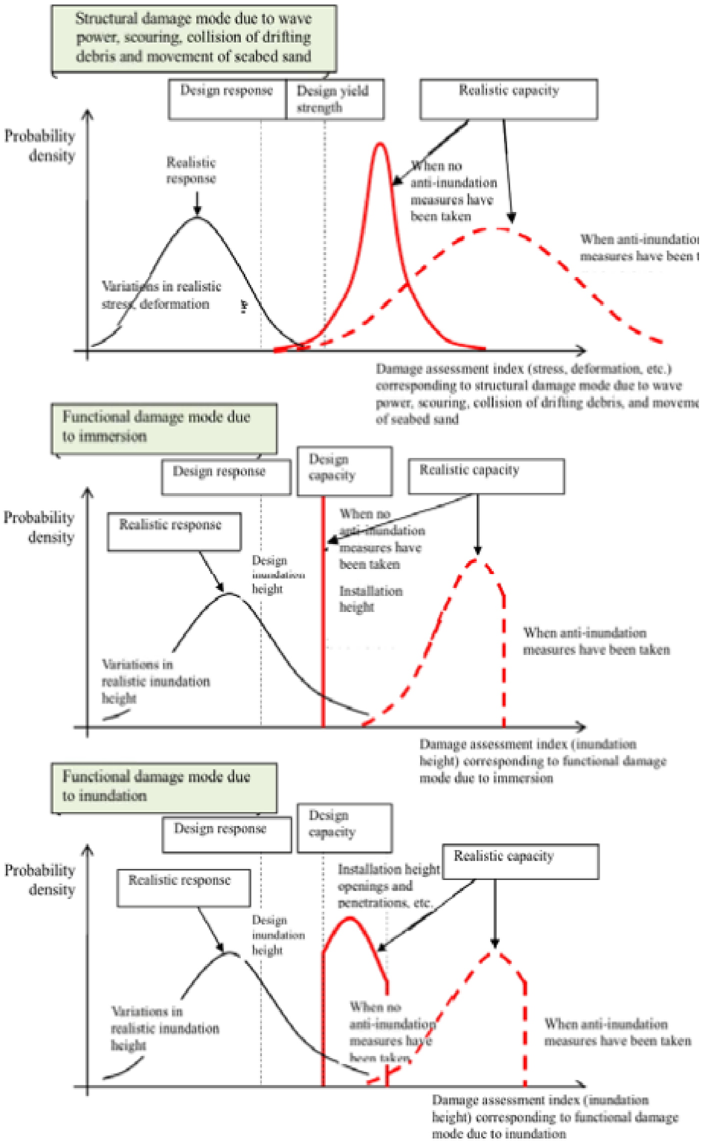

Fig.9: General Concept of Realistic Capacity

(2) Quantitative Results of As-Is Model

The result of the EP model* (*for comparison with As-Is model, a model including SA measures) is 1.0×10^{-7} [per reactor-year], while the result of As-Is model is 2.3×10^{-7} [per reactor-year].

In the EP model evaluation, fragility curves for each building rise almost simultaneously starting from a Tsunami height of approximately 15 m (site elevation) + 2 m, and inundation within the site progresses in the order: site ⇒ R/B ⇒ T/B ⇒ C/B, depending on Tsunami height.

In contrast, in the Safety Assessment Report (SAR) evaluation, fragility curves begin to rise from lower Tsunami heights, and inundation within the site progresses in the order: site ⇒ C/B ⇒ R/B ⇒ T/B, depending on Tsunami height.

Scenarios in which reactor building (R/B) flooding serves as the initiating event and leads directly to core damage are the dominant scenarios. Since these occur at lower Tsunami heights compared to the EP model, the core damage frequency is considered to be higher.

Fig. 10 shows the relationship between core damage frequency (CDF) and conditional core damage probability for each Tsunami height. The CDF reaches its maximum around Tsunami heights of 16.5 to 17.5 m. The maximum CDF across all Tsunami heights occurs at a Tsunami height of 17 m, contributing approximately 28% to the total CDF.

For Tsunami heights below 15.3 m, the CDF is kept relatively low due to the effectiveness of inundation prevention measures such as seawalls and intake structure closure panels (T/B). However, once the Tsunami height exceeds approximately 17 m, the probability of R/B flooding increases, and there are no effective means to prevent core damage, leading directly to core damage.

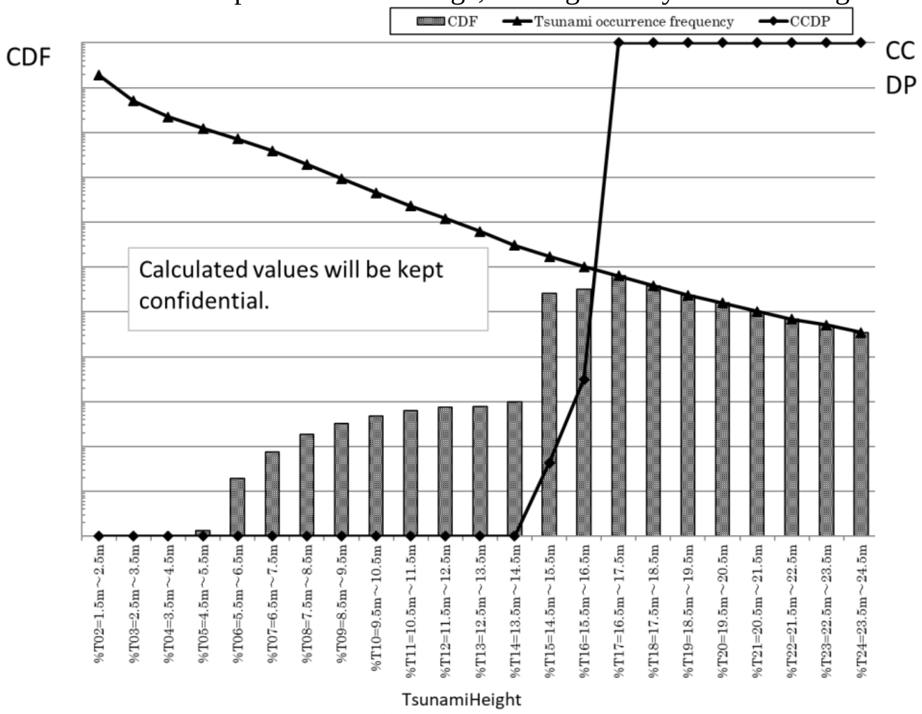

Fig.10: Core Damage Frequency and Conditional Core Damage Probability by Tsunami height

2-3 Examination of Protective Measures Based on the Results of As-Is Model

Protective measures for improving safety were examined based on the results obtained from As-Is model.

As an indicator for selecting protective measures, the Fussell–Vesely (FV) importance measure was calculated. The FV is determined using Equation (1), and Table 1 presents the results of the FV importance analysis with respect to the total CDF.

$$\mathrm{FV} = \frac{\mathrm{R_0} - \mathrm{R_n}}{\mathrm{R_0}} \qquad (1)$$

$\mathrm{R_0}$: CDF (Current risk)
$\mathrm{R_n}$: CDF (Risk assuming that the target component does not fail)

The FV importance analysis results indicate that failure to properly close watertight doors between buildings and watertight doors leading to the process computer room ranks among the dominant contributors.
As hardware countermeasures, installing buzzers that activate when the doors are open can draw operators’ attention and help ensure that the doors are properly closed. Additionally, installing sensors on the doors and implementing physical barriers that prevent access when the doors are open can be considered. Specifically, rather than installing the sensor immediately adjacent to the door, it should be placed slightly away from the door along the passage route, arranged such that any worker attempting to proceed will inevitably come into contact with or be detected by the sensor. When this sensor is activated, if the watertight door has not been closed within a certain period of time, a signal will be

automatically sent to the central control room, triggering an alarm or warning. This mechanism ensures that any failure to close the door is reliably detected and corrected.
As operational countermeasures, improving operator awareness through training is also important. Moving forward, the effectiveness of these proposed protective measures will be verified through sensitivity analyses, and if their effectiveness is confirmed, their application to actual plant systems will be considered.

Table 1: Results of FV Importance Analysis for Total CDF

Rank	Event	FV Importance	Factor
1	Seawall damage due to Tsunami	We will keep FV Importance confidential.	Tsunami Damage
1	6 unit startup transformers (6SA, 6SB) damaged by Tsunami		Tsunami Damage
2	Watertight door left open (S/B-C/B), failure in closing operation		Human Error
3	Watertight door left open (No.22)		Human Error
3	Watertight door left open (No.66)		Human Error
4	C/B flooding due to Tsunami		Tsunami Damage
5	SPC system configuration error (excess)		Human Error
6	Power supply operation using portable generator		Human Error
7	ADS SV1 rapid depressurization operation (excess)		Human Error
7	ADS SV2 rapid depressurization operation (excess)		Human Error
8	Fuel supply return operation using tank lorry (4kL)		Human Error
9	Debris removal using wheel loader		Human Error
10	Water supply to fresh water tank using portable pump (A-2 class)		Human Error

3. Recent Initiatives

3.1. Case Study of Tsunami Risk Assessment

At KK7, there has been a case in which Tsunami risk was evaluated using internal PRA. One example is the work involving the removal of a funnel backflow prevention device.

・Work description:

Removal (release) of the backflow prevention device in the Ferrum Ion Injection System (FEI) room

・Assumed events:

Earthquake and earthquake-induced Tsunami

・Evaluation assumptions:

In the seismic flooding assessment for the turbine building, it is assumed that expansion joints in all circulating water piping of seismic class B or lower fail. As a result, seawater flows into the Circulating Water Pump (CWP) room and the condenser room.

In addition, if a Tsunami is generated due to an earthquake, seawater flows into the CWP room. Furthermore, if the funnel in the FEI room—connected to the Sea Water Storm Drain System (SWSD) sump installed in the CWP room—is in an open state, seawater flows into the condenser room.
If seawater continues to flow in and the inundation level reaches the compartments where the Reactor Sea Water System (RSW) and Reactor Building Cooling Water System (RCW) pumps (systems A, B, and C) are installed, all three trains of safety-related equipment will lose their functions.

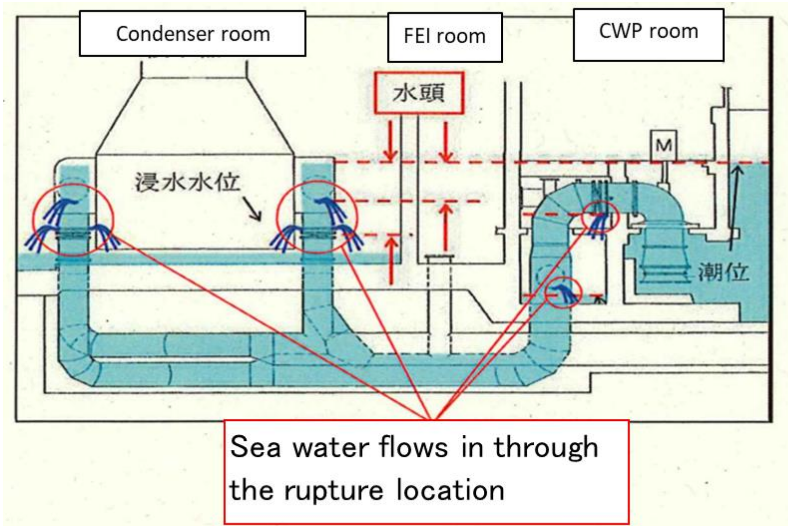

Fig 11: Locations Where Flooding Is Assumed in the Event of Expansion Joint Failure

The impact of releasing the funnel backflow prevention device was evaluated in terms of the incremental increase in core damage.

ICDP=IEF※1×CCDP※2×Δt/365

The allowable work duration Δt/365 was calculated such that ICDP=1.0×10^{-5}※3

IEF: Initiating event frequency [/year]
CCDP: Conditional core damage probability
Δt: Work duration [days]

※1 Since all three trains of RCW and RSW would be lost due to a Tsunami exceeding 3500mm above sea level (height of the protected equipment area), a conservative annual exceedance frequency corresponding to a 3 m Tsunami was used.

※2 From the internal events Level 1 PRA model during operation at KK7, the CCDP for the loss of RCW/RSW systems A, B, and C (loss of two units in one train) is 2.3×10^{-1}, based on the core damage frequency for each initiating event.

※3 If ICDP exceeds 1.0×10^{-5}, the risk level becomes "red," and online maintenance (OLM) is not performed.

It was determined that if the work duration is limited to 23 days, the increase in risk remains small and the work can be permitted. Currently, the evaluation is performed using internal PRA; however, once preparations for Tsunami PRA are completed, a phased transition to evaluation using Tsunami PRA is planned.

3.2 Installation of a Risk Monitor in the Main Control Room

Since 2026, a risk monitor has been installed in the Main Control Room.

In preparation for the implementation of online maintenance (OLM), efforts are underway to enable operators to perform risk assessments themselves in order to better understand and manage risks during planning (Figure 12). By allowing operators to conduct evaluations on their own, it becomes possible to achieve assessments that more closely reflect actual on-site conditions.

Currently, the risk monitor is limited to internal PRA; however, the aim is to implement a full-scope PRA that also includes external events. Furthermore, efforts are being made not only to develop models but also to ensure that these models are designed with practical on-site application in mind.

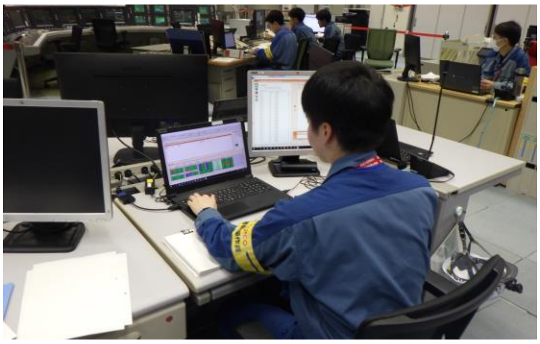

Fig. 12: Risk Monitor Installed in the Main Control Room

4. Conclusion

This paper has described the significance of Tsunami PRA at KK7 and its implementation status. In light of 1F accident, it is essential to appropriately evaluate Tsunami-induced risks and to identify plant vulnerabilities going forward. Through Tsunami PRA, clarifying accident scenarios enables the formulation of effective protective measures.

In particular, the comparison between the EP model* and As-Is model allows for a more detailed evaluation of Tsunami impacts and establishes a foundation for examining specific measures aimed at risk reduction. Moving forward, it is important to utilize the knowledge gained to further enhance safety and to implement measures that rationally reduce Tsunami-related risks.

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