Seismic PRA for Kashiwazaki-Kariwa NPP

Keiichiro Saito[a], Masanori Takeuchi[b],Takashi Uemura[c],Yasunori Yamanaka[d]
[a bcd]Tokyo Electric Power Company Inc, Tokyo, Japan

Abstract : A seismic PRA was carried out in order to confirm the effectiveness of measures related to Kashiwazaki-Kariwa NPS (KK-NPS) that are aimed at improving safety and founded upon lessons learned from Niigata-Chuetu-Oki Earthquake (NCO) in 2007 and The 2011 off the Pacific coast of Tohoku Earthquake (the Tohoku Earthquake)) in 2011 as well as from understanding our plant vulnerability to Earthquakes. The lessons learned from the Fukushima Daiichi (F1) accident and findings gathered from the Great Eastern Japan Earthquake were reflected in both the hazard evaluation and the sequence evaluation during the seismic PRA. In this evaluation, we were able to confirm the effectiveness of safety measures carried out towards plant vulnerabilities that were found before these measures were implemented. Additionally, our objective is to continually work towards improving the level of safety through utilizing risk which also accounts for results from seismic and other PRA in order to assess effective countermeasures. Here, we will also evaluate the findings extracted from the seismic PRA carried out this time in studying how to "improve the accuracy of fragility evaluations of portable equipments" and "methods for implementing evaluation conditions for redundancy".

Keywords : seismic PRA, Kashiwazaki-Kariwa, Fukushima Daiichi accident

1. Introduction

We have implemented various safety measures while accounting for information gathered from lessons learned from the Tohoku Earthquake that took place in 2011 and NCO in 2007. In this study, we carried out a seismic PRA for the KK Unit7 (ABWR,1356MWe) in order to quantitatively-confirm the effectiveness of safety measures while understanding the risk associated with earthquakes for the KK-NPS.

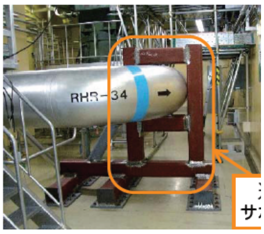

(KK-NPS whole view)

We carried out the PRA in accordance with "A standard for Procedure of Seismic Probabilistic Safety Assessment (PSA) for nuclear power plants issued by the Atomic energy Society of Japan (AESJ))"

However, revisions to this standard started to be carried out last year, and knowledge gained from lessons learned from 2007 onwards, with regard to the Tohoku Earthquake, is currently being incorporated into revisions. Therefore, some contents including ones about the seismic hazard evaluation that are planned to be incorporated into the revisions of the aforementioned standards have already been reflected in this evaluation.

2. Outline of Safety Measures for KK-NPS

KK-NPS has already experienced a large earthquake referred to as the NCO (which occurred approx. 16km from KK-NPS with a magnitude 6.8Mw). In total, KK-NPS is comprised of seven units. When this earthquake occurred, Unit3, 4, and 7 were in operation and Unit2 was undergoing preparation for start-up. All four of these units stopped automatically as per their design, and the remaining three reactors (Unit1, 5, and 6) were able to preserve their cold shutdown condition. The damage to the safety facilities was not confirmed in a detailed investigation that was carried out after the earthquake. However, seismic ground motion far exceeded the design basis of KK-NPS, so after the NCO we revised the design base earthquake of the KK NPPs and reinforced the earthquake-resistance of many different types of equipment such as pipes and electrical conduits (i.e. as for the pipes, target points are over 1,000 locations for each reactor facility). After implementing measures for reinforcing the earthquake-resistance capabilities of Unit7 as well, the Unit resumed operation in 2009, two years after the NCO.



（additional piping support）

Afterwards, the massive tsunami induced by the Tohoku Earthquake in 2011 hit F1-NPS, Unit1 `3 of F1 experienced core damage. Many countermeasures that are based on lessons learned from the F1 accident have been implemented for the KK-NPS as well to improve the robustness of the plants against extreme external events. Currently, additional safety measures are also underway.

Because the tsunami was the primary factor for the F1 incident, measures for flooding, such as the installation of water-tight doors and seawalls, and measures for severe accident after core damage account for a large proportion of the countermeasures implemented thus far. However, other measures that are effective against earthquakes are also being implemented. These include improving the earthquake-resistance of some facilities associated with off-site power supply (i.e. switchyards, transformers, cable tunnels, etc.)

Additionally, the implementation of various accident management countermeasures that provide flexibility in times where the design threshold is exceeded may lead to improved levels of safety for plants against earthquakes.

The following represents the primary safety measures that are considered to be able to contribute to improving the levels of safety of plants in the event of an earthquake.

(Primary Safety Measures)

(power-supply cars)

(freshwater reservoirs)

Strengthening Electrical Capabilities (SBO measures)
- Securing emergency AC power using power-supply cars and/or Air-Cooling GTG
- Enhancing the capacity of standby batteries

Securing Redundancy and Diversity of RPV Water Injection Methods
- Reinforcing the earthquake-resistance capabilities of the condensate water makeup system (Permanent installations of methods for low-pressure coolant injection)
- Deployment of fire trucks (securing a portable means for low-pressure coolant injection)
- Building Freshwater Reservoirs as a water source for injection.
- Reinforcing the earthquake-resistance capabilities of the freshwater tanks

Reinforcing Heat-removal Capabilities

-Installation of an Alternative Seawater Heat Exchanging System

-Other (Improving Workability of AM Countermeasures)

-Reinforcing roads and installing heavy machines in order to secure access road for power-supply cars and fire truces

-Enhancement of function for monitoring plant states (i.e measurement of water level in the reactor) in times when off-site power supply is lost

3. Outline of Seismic PRA

3.1 Seismic PRA Regulatory Procedures

The regulatory procedures for the seismic PRA were carried out while adhering to the AESJ standards. These procedures largely consist of the following four steps.

(i)Collection of site and plant information , the setting of accident scenarios

(ii)Seismic hazard evaluation

(iii)Fragility evaluation of buildings and facilities

(iv)Accident sequence evaluation

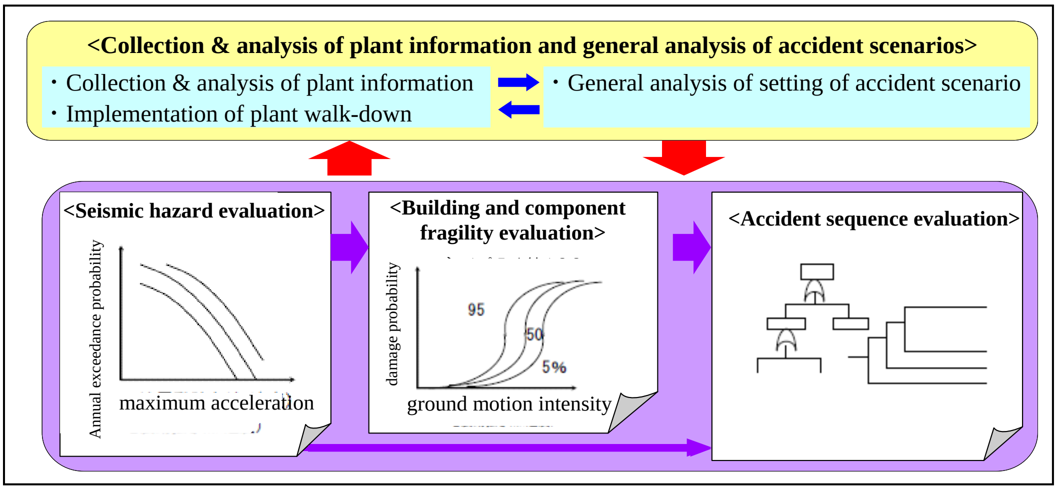


Fig.1 Seismic PRA Evaluation Process

3.2 Collection of site and plant information, the setting of accident scenarios

During this seismic PRA, we first collected information on a very wide scale that relates to the system design and the management of operations in KK-NPS. These were collected at the time the Level 1 PRA for internal events was carried out. In addition, we also collected information from viewpoints that are unique to earthquakes, such as books about calculating the resistance to earthquakes.

1) Establishment Permit of Nuclear Power Station
2) piping and instrumentation diagrams
3) electrical system diagrams
4) plant component layout diagrams
5) System Specification
6) Equipment Specification
7) Tech. Spec.
8) operating procedure manuals

9) internal events L1RRA report
10) stress test report
etc

Next, we analyzed the plausible Initiating Events that could lead to severe damage of the reactor fuel in the event of an earthquake while focusing on accident scenarios that were particular to that found at the time of an earthquake. Additionally, we also considered the initiating events that were used in the PRA for internal events in estimating the effects that seismic ground motion would have on the CDF. Here, we selected scenarios that were appropriate for analyzing accident scenarios.

Items for consideration in addition to the collection of information include reflecting the implementation status of the countermeasures for reinforcing the earthquake-resistance and the most recent condition of the plants which includes the implementation additional of portable accident-management equipment, such as fire trucks and power-supply cars that were used widely after the Tohoku Earthquake, and the new procedures used for such.

Additionally, instead of just simply confirming the operation procedures and equipment specifications with regard to the portable AM measures, careful attention is also needed for utilization and confirmation of the results from implementing the findings gathered from walk downs and lessons learned, because the feasibility of countermeasures would be difficult to understand such as "the effectiveness of emergency organization" or "'on-site accessibility and operability" when evaluating them from a textbook approach.

3.3 Seismic hazard evaluation

In the seismic hazard evaluation, we firstly established seismic source models (specified source models and zone source models) for the seismic source based on results from studies of active faults investigations around NPS, then the propagation of the seismic ground motion from such seismic sources.

Although we did not consider the simultaneous movement of faults in the past evaluations, in this time , we considered it in the light of Fukushima Daiichi accident.

In establishing our seismic ground motion propagation model, we considered the properties regarding to seismic ground motion propagation and the seismic source around KK-NPS, in order to establish a model that can evaluate the probability distribution of the seismic ground motion intensity at KK-NPS when an earthquake with certain scale occurs at some place.

Additionally, the probability distribution of the seismic ground motion intensity produced by earthquakes of a specific scale represents the aleatory uncertainty in this seismic ground motion propagation model, and the epistemic uncertainty produced by insufficient information and awareness was evaluated using the logic tree divergence related to the parameters for the probability distribution and selection of the evaluation model. Here, the seismic hazards were evaluated, and Fig.2 shows the results of the seismic hazard evaluation for the Unit7 at KK-NPS.

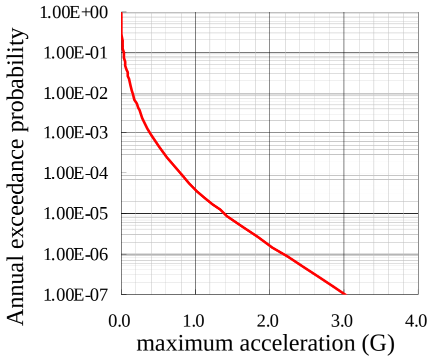


Fig.2 hazard curve

3.4 Fragility evaluation of buildings and facilities

A fragility evaluation was carried out based on the analysis of the accident scenario mentioned in Section 3.2. The evaluation objects were the same buildings and facilities that were evaluated in the initiating events and the accident sequence evaluation based on the findings gathered from clarifying the accident scenario post-fact. There were approx. 600 items of equipment evaluated in this seismic PRA for Unit7 of KK-NPS.

However, because it would be difficult in this stage to evaluate mobile-type equipment, such as fire trucks and power-supply cars that were deployed after the F1 accident, we applied a deterministic method for evaluating their integrity in place of the fragility evaluation. More specifically, in this fragility evaluation, we assume that functionality can be preserved all the way up to the seismic ground motion (Ss) as outlaid in the design criteria, and in contrast we will assume that functionality will be lost when seismic ground motion occurs that exceeds this level. Hereafter, we plan to research formulations for the fragility evaluation method for such mobile equipment types.

In addition to the random failure mode which are considered in the PRA for internal events, "structural damage" and "functional damage" caused by the earthquake are considered..

These areas that assume perfect correlations for the damage caused by earthquakes also differed with those areas in the PRA for internal events. In other words, when one piece of equipment in a series is damaged, we assume that all equipment that is similar in type will be damaged as well. As such, the effects gained from improving the reliability redundancy of the system are not expected conservatively.

3.5 Accident sequence evaluation

We extracted the buildings, structures, systems and components (SSCs) that are necessary for preventing severe damage to the core in relation to the accident sequence (initiating events and accident scenarios) that was the target of the analysis. We then created a model for the system therein and for an accident sequence that will lead to the core damage based on the analysis results for the accident scenario in Section 3.2. Using these models we quantify the accident sequence and evaluate the CDF, in the event of earthquake. Then, we performed analysis of the primary accident sequences.

A revision of the “Mission Time" and the "operator handling failure probability" was made to reflect the lessons learned from the F1 accident. Although the THERP technique

(NUREG/CR-1278) was used for establishing the operator handling failure probability, a stress factor (x5) was assumed for the Human Error Probability(HEP) used in the PRA for internal events while assuming a high-stress state that accompanies the confusion that manifests after an earthquake occurs in relation to the operations performed over a relatively short period of time (within a few hours) (However, it is assumed to be constant regardless of the size of the seismic ground motion). A specific example of such operations include“ ECCS manual start-up in case of ECCS automatic start-up failure” and "manual reactor water level control operations performed in case of the high-pressure coolant injection success".

Additionally, the primary contributing factor of damages was more so found to be the hydrogen explosion accident and the submersion effects caused by the tsunami, rather than the damages associated with the earthquake during the F1 accident. However, based on previous experiences of incurring difficulty in restoring works, the mission time set to 72 hours compared with only 24 hours in the Level 1 PRA for internal events.

Attention has also been given to the integrity of the fuel in the spent fuel pool (SFP) in the standard revision activity to reflect lessons learned from the F1 accident. Therefore, although the objective of this seismic PRA was to evaluate the CDF in the RPV while the plant is in operation, we also plan on further evaluating the SFP hereafter.

4. Evaluation

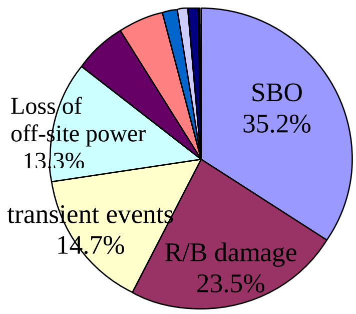
4.1 Summary of the Evaluation Results

In order to confirm the effectiveness of the various safety countermeasures and AM countermeasures implemented in the past and provide for more effective countermeasures for safety enhancement, we first performed the CDF evaluation of the plant the state before the implementation of safety measures, which were implemented based on the lessons learned from F1 accident.

The total CDF is 1.6E-05(/RY) , and shows the results of the CDF evaluation categorized by initiating events and by core damage sequence (as found below), and also shows the analysis results for the primary accident sequences.

i. CDF by initiating events and Analysis of the primary accident sequences

The CDF categorized by initiating events is shown in Fig.3 and Table 1.



Initiating event	CDF[/RY]	contribution ratio
SBO	5.5E-6	35.2 %
R/B damage	3.8E-6	23.5 %
transient events	2.4E-6	14.7 %
loss of off-site power	2.1E-6	13.3 %
RPV/PCV damage	8.9E-7	5.5 %
LOCA(E-LOCA)	7.7E-7	4.8 %
loss of the instrumentation/control systems	2.8E-7	1.7 %
PCV bypass	1.5E-7	0.9 %
loss of DC power source	5.6E-8	0.3 %

Fig.3/Table.1 CDF categorized by initiating events

The dominant accident sequence that has the largest contribution to the CDF is the scenario where all AC power is lost (SBO) due to failure of D/G support systems and off-site power sources. The CDF is 5.5E-06(/RY), and comprises of about 34% as a whole.

Although the pressure control by safety relief valves and the water injection into the RPV by the reactor core isolation cooling systems (RCIC) are successful after the SBO occurs, this is a scenario where the core damage occurs as a result of failure of continuous water injection due to the depletion of water source or the batteries.

The accident sequence which has second largest contribution to the total CDF is the damage of Reactor building (R/B) by the earthquake. The CDF herein is 3.8E-06(/RY) and comprises of about 24% of the entire sequence (RBR sequence). Here, the Slippage of Reactor Building foundation ground is damaged, or in other words, the reactor building is destructed. This causes the damage of the structures and equipments like primary containment vessel and the reactor pressure vessel which are located inside the reactor building. This leads to a scenario where the many of the mitigation components also become damaged. Although correspondence to restore the situation is still possible depending on the degree of damage incurred by the mitigation components, conservative assumptions such as complete loss of functionality in all of the mitigation components is made due to the difficulty in identifying the accident scenario in detail. Therefore, in this case, it is assumed that the failure of the reactor shutdown and the reactor cooling and it leads to reactor core damage.

If we take a look at the relationship of the seismic acceleration and the CDF, we can see some values in regions with large seismic ground motion that exceeded 1600gal (Fig. 4), and the RBR sequence is also found to be the dominant sequence in regions with these high levels of seismic ground motion (Fig.5). Here, the seismic hazards that are estimated for the KK-NPS (as shown in Fig.1) show that there are some values in regions that exceeded 2000gal as well, which means that the CDF of the RBR sequence is even higher.

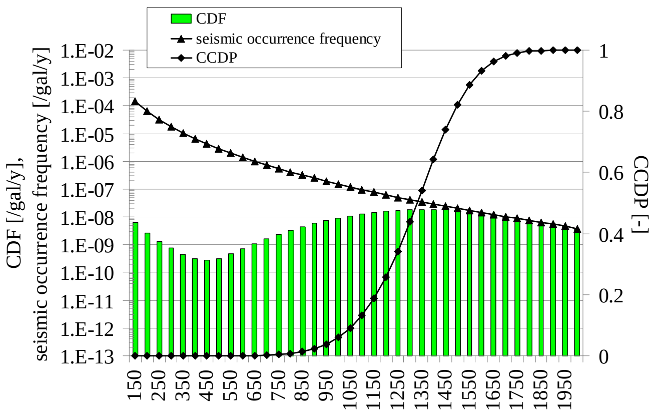


Fig.4 seismic acceleration and CDF

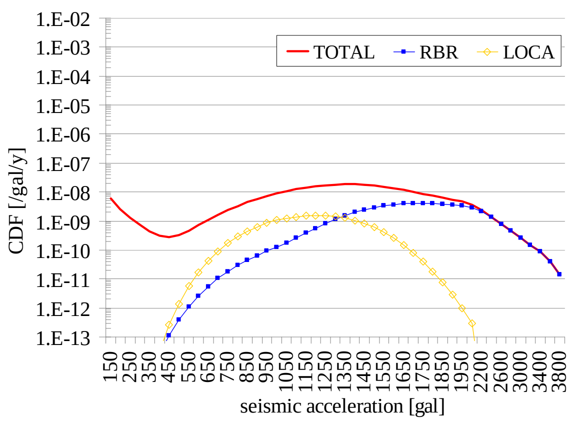
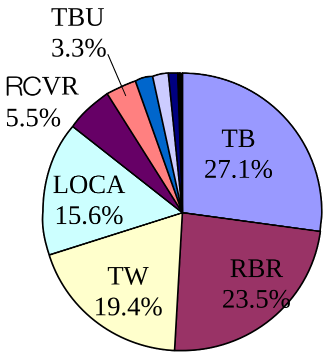


Fig.5 seismic acceleration and CDF of the primary accident sequences

However, in reality, when massive earthquakes occur that lead to large-scale damage of the R/B, it would be difficult to imagine cases where the all mitigation systems had been completely damaged. If we assume that the R/B is robust, then only the relatively-weak part of the building will incur damage. Although there is a possibility that this will lead to a loss of function in some of the equipment as a result, the possibility that at least some of the methods that can preserve the functionality of the equipment such as removing heat or injecting water will remain is not small. Therefore, in the future, we plan on pursuing evaluations of detailed analyses related to structural damage and accuracy improvement for ground sliding analyses, and we think that it will be necessary to link that to the understanding actual plant risk and the consideration for countermeasures aimed at improving the levels of safety.

ii.CDF classified by reactor core damage sequence and Analysis of the primary accident sequences

Fig.6 and Table 2 both show the CDF categorized by the accident sequence for the reactor core damage. Here, the power failure sequence (TB sequence) consists of the largest contributing proportion at approx. 27%, and the CDF was 4.4E-6(/RY). The damage to the building/structural components(R/B) (RBR sequence) had the next largest contributing proportion at approx. 23%, and the CDF was 3.8E-6(/RY). These two accident sequences alone account for approx. 50% of the entire scenario, and the decay heat removal failure sequence (TW sequence) (CDF: 3.1E-6(/RY), contributing proportion: approx. 19%) and the reactor coolant pressure boundary failure sequence (LOCA sequence) (CDF: 2.5E-6(/RY)), contributing proportion: approx. 16%) all follow accordingly.



core damage sequence	CDF[/RY]	contribution ratio
TB: Loss of power supply:	4.4E-6	27.1 %
RBR: Reactor building damage	3.8E-6	23.5 %
TW: Decay heat removal failure	3.1E-6	19.4 %
LOCA: primary coolant pressure boundary failure	2.5E-6	15.6 %
PCVR:PCV/RPV failure	8.9E-7	5.5 %
TBU: Loss of power supply	5.4E-7	3.3 %
TC: Failure to ensure reactor sub-criticality	3.6E-7	2.2 %
CI: loss of the instrumentation/ control systems	2.8E-7	1.7 %

Fig.6/Table.2 CDF categorized by reactor core damage sequence

The LOCA sequence assumes a scenario where the primary coolant boundary in the PCV is damaged as a result of a direct load caused by the earthquake or a rise of PRV pressure due to failure to open safety relief valves (S/R valves). However, we take a conservative approach towards both of the cases by organizing this accident scenario as one where core damage is eventually occurred and set this sequence equal to the Excessive-LOCA (E-LOCA) due to the difficulty in evaluating the possibility of the situation could be taken back under control depending on degree of the damage to the boundaries of primary coolant.

The LOCA sequence due to failures of S/R valves accounts for approx. 70% of the total LOCA sequence, and a perfect correlation is assumed for failures of the 18 total S/R valve. However, in reality, there are differences in the installed location, the piping that the S/R valve is mounted onto, and in the response of the valve in the event of an earthquake. Given that high CDF-contributing regions where seismic ground motion occurred for the LOCA sequence were around 1200gal (seismic design basis level) (Fig. 4), we can say that our approach is fairly conservative based on our assumption that if 1 valve is damaged that all of the other valves will also fail.

Additionally, in the same way as mentioned about the S/R valves, we can also say that our assumption is fairly conservative for the LOCA sequence due to damages of primary coolant pipings. E-LOCA is caused by the damages of plural pipings (the main stream system, feed-water system, and the ECCS systems (RHR, SLC,RCIC, and HPCF)), but we set the damage of RHR piping that has the lowest HCLPF as E-LOCA.

Considering the conservative approaches of the LOCA sequence and the RBR sequence mentioned in Section i, the differences in the size of the CDF value do not always show the true risk for the plant.

Therefore, it is also important to consider the pre-defined conditions for the accident sequence evaluation and the fragility evaluation rather than just focusing on the CDF alone when considering effective safety measures.

4.2 Importance Analysis

Table.3 shows the results from the importance analysis (Fussell-Vesely index). The FV index for the components related to the SBO, RBR, TW, and the LOCA constitutes the upper ranks just as shown in the results of the analysis of the primary accident sequences in Section 4.1.

These results also show that countermeasures for the TW sequence and SBO sequence are effective for reducing risk. Although the original primary objective of the reinforcement of the heat-removal and AC-power source capabilities shown in Section 2 is to implement countermeasures for tsunami, we can also see that they are also effective countermeasures for earthquakes as well.

Building/Component	FV	accident sequences
RCW Heat Exchanger	1.1E-1	SBO, loss of off-site power, transient events
RSW pump	8.1E-2	SBO, loss of off-site power, transient events
S/R valve	6.9E-2	LOCA(E-LOCA)
RCW piping	3.2E-2	SBO, loss of off-site power, transient events
RHR valve	3.2E-2	loss of off-site power, transient events
RHR/LPFL valve	2.9E-2	loss of off-site power, transient events
RHR piping	2.9E-2	loss of off-site power, transient events
Piping in PCV	1.5E-2	LOCA(E-LOCA)
R/B foundation ground slip line	1.5E-2	R/B damage
PRV pedestal	1.3E-2	RPV/PCV damage

Table.3 Fussell-Vesely index

4.3 Efficacy of accident management countermeasures

Although various AM countermeasures were implemented for the KK-NPS, we consider only countermeasures that were able to secure earthquake-resistance in this evaluation for the Unit7.

These countermeasures primarily consist of securing an AC power source via power-supply cars or GTG, installing alternative seawater heat-exchanging system, or building freshwater reservoirs. Conversely, although the earthquake-resistance is secured for the fire trucks that have been deployed as a portable means for low-pressure coolant injection, the earthquake-resistance for the fire extinguishing system piping that the fire truck connects to is never confirmed. Therefore, we do not expect consider the water injection using fire trucks in this seismic PRA evaluation.

However, there are plans to change the fire extinguishing system piping connection for the fire truck to a line where the earthquake-resistance has been confirmed, from which in the future we can expect that water injection using the fire trucks will become a plausible means of correspondence.

By the implementation of AM measures, the evaluated CDF value was reduced approx. 20%. If we take a look at the CDF from the accident sequences while removing the events that are directly-linked with the core damage such as the RBR or the LOCA, we can see that the values reduced by approx. 50% and thereby say that this shows the efficacy of the AM measures.

In this study, we set the aforementioned fragility conditions based on a deterministic approach because we are unable to evaluate fragility curve for the mobile-type equipment, such as the power-supply cars and the alternative seawater heat-exchanging system. So it is necessary to continue the examination of the fragility evaluation method for them.

5. Conclusion

We confirmed the vulnerable areas of the plant through performing a seismic PRA for Unit7 at KK-NPS before the AM measures. Based on the characteristics and trends related to the risk of Unit7 found in the results, we were able to confirm that the various AM measures that had been implemented based on lessons learned from the F1 accident were also effective for earthquakes.

Hereafter, in order to implement even more effective countermeasures when aiming to continuous enhancement of safety, rather than just simply looking at the values for the CDF, it is also important to give due consideration for the validity of the evaluation conditions and the details of the accident scenario in the same way that we considered the RBR and LOCA sequences.

Additionally, we continue to examine the fragility evaluation method for portable-AM equipments and to evaluate a SPRA for other Unit at KK-NPS in consideration of countermeasures which are currently being implemented and planned hereafter, and we also plan to focus on evaluations for SFP, as well.

References

[1] “A standard for Procedure of Seismic Probabilistic Safety Assessment for nuclear power plants issued” the Atomic energy Society of Japan (2007)