Seismic PRA for Hamaoka NPP Unit 4 (Including severe accident countermeasures)

Tomoki Inokuchi[1], Hiroyasu Takeyama[2], Yoshitaka Tsutsumi[3], Yasuhiro Iwaya[4]

[1] *Chubu Electric Power Co.,Inc, Nuclear Power Dept., Safety Review Group: 1,Higashi-shincho,Higashi-ku, Nagoya, Japan, 461-0001, Inokuchi.Tomoki@chuden.co.jp*
[2] *Chubu Electric Power Co.,Inc, Nuclear Power Dept., Safety Review Group: 1,Higashi-shincho,Higashi-ku, Nagoya, Japan, 461-0001, Takeyama.Hiroyasu@chuden.co.jp*
[3] *Chubu Electric Power Co.,Inc, Nuclear Power Dept., Design And Engineering Group: 1,Higashi-shincho,Higashi-ku, Nagoya, Japan, 461-0001, Tsutsumi.Yoshitaka@chuden.co.jp*
[4] *Chubu Electric Power Co.,Inc, Nuclear Power Dept., Safety Review Group: 1,Higashi-shincho,Higashi-ku, Nagoya, Japan, 461-0001, Iwaya.Yasuhiro@chuden.co.jp*

***Abstract*：**

Objectives: New enhanced safety regulatory requirements, which are based on the lessons learned from the Fukushima-Daiichi NPP accident, require Japanese utilities to install adequate severe accident countermeasures and to evaluate the effectiveness of these countermeasures in the deterministic manner based on scenarios derived in consideration of the Probabilistic Risk Assessment (PRA). In addition, Japanese utilities are required to submit the safety improvement evaluation, which includes evaluation of the effectiveness of the severe accident countermeasures by means of PRA, by 6 months after the end of the first refueling outage following restarting NPP operation. On the other hand, we have been enhancing our Hamaoka NPP's seismic resistance even before the Fukushima-Daiichi NPP accident, and then, considering the new enhanced safety requirements, we now have been studying the seismic PRA in order to evaluate the effectiveness of our severe accident countermeasures.

Methods: The seismic PRA is implemented following the guideline "Implementation Standard for Seismic Probabilistic Safety Assessment of Nuclear Power Plants: 2007" (Ref. 1) issued by Atomic Energy Society of Japan (AESJ) in September 2007.

Results: Core Damage Frequency (CDF of the case without the countermeasures is 1.9E-06 (/RY). CDF of the case with the countermeasures is under evaluation

Conclusions: The effectiveness of the countermeasures in terms of the seismic PRA of the cases with/without the countermeasures is discussed.

Keywords: Seismic PRA Hamaoka, Safety Improvement Evaluation

I. Introduction

We have implemented various safety countermeasures based on the lessons learned from expected the Tokai earthquake and Suruga-Bay Earthquake in 2009. In this study, the seismic PRA for the Hamaoka Unit-4 (BWR-5, 1,137MWe) is carried out in order to quantitatively-confirm the effectiveness of safety countermeasures and to obtain the risk information associated with earthquakes for the Hamaoka NPP. We carried out the PRA in accordance with "Seismic PSA Standards: 2007" (Ref. 1) published by the Atomic Energy Society of Japan (AESJ) in September 2007." This standard was revised in December 2015, adding the knowledge gained after 2007 including the lessons learned from the Suruga-Bay Earthquake and the Tohoku Earthquake. Therefore, some methods used in this analysis including the seismic hazard evaluation has been already reflected in the latest version of the standard.

Hamaoka NPP

Ⅱ. Outline of Safety Countermeasures for Hamaoka NPP (As shown in Fig.1)

Hamaoka-NPP has experienced a large earthquake called the Suruga-Bay Earthquake (the hypocenter was located at approx. 43.5km from Hamaoka-NPP, the magnitude was assessed at 6.5Mw). Hamaoka-NPP comprises a total of five units. When this earthquake occurred, Unit 4 and 5 were in operation, and Unit 3 was the reactor shutdown state for periodic inspection (Unit 1 and 2 have been under decommissioning since Jan 2009). Both two units which were in operation stopped automatically as designed, and the remaining three reactors (Unit-1, Unit-2, and Unit-3) were able to preserve their cold shutdown condition. The all safety facilities was confirmed intact in detailed investigations carried out after the earthquake. In addition, after the Fukushima-Daiichi NPP accident, we revised the design base earthquake of the Hamaoka NPP (The anticipated triple-interlocked Tokai/ Tonankai/ Nankai earthquake that we envision at the Hamaoka NPP would have maximum tremor strength of 1,200 gals in the bedrock), and we have reinforcing the earthquake-resistance of many different types of equipment such as pipes and electrical conduits. Many countermeasures those have been implemented in the Hamaoka NPP based on lessons learned from the Fukushima Daiichi-NPP accident improved the robustness of the plants against extreme external hazards. Currently, additional safety countermeasures are also being implemented. Since the tsunami was the one of the most important factor in the Fukushima Daiichi-NPP accident, countermeasures for flooding, such as the installation of water-tight doors, tsunami protection wall and countermeasures for severe accident after core damage account for a large proportion of the countermeasures implemented thus far. However, other countermeasures those are effective against earthquakes have also been implemented. Various severe accident management countermeasures are to be implemented to provide flexibility in case that earthquakes exceed the plant design. The following sessions are discussing the major safety countermeasures which improve the safety of our plants in the event of an earthquake.

Ⅱ.A. Our Strategy against Tsunami

Ⅱ.A.1 Flooding Prevention Countermeasures 1: Preventing tsunami from flooding the station site

〈Tsunami Protection Wall〉

・We constructed a tsunami protection wall, which crosses approximately 1.6 km from east to west of the sea side of the station site. This will maximize the effectiveness of our measures to prevent flooding on the station site.

・The height of the tsunami protection wall is 22 meters above sea level. This height was chosen based on the analysis of the anticipated Nankai Trough Mega-quake.

・In contrast to conventional breakwaters, the new tsunami protection wall combines reinforced concrete foundations embedded into the bedrock with an L-shaped wall consisting of structural steel and steel-framed reinforced concrete in order to obtain high resistance to earthquakes and tsunamis.

・ The east and west sides of the Hamaoka NPP is protected from tsunami flooding by "cement-mixed soil embankments."

・These embankments have 22 - 24 meters height above sea level. Mixing cement into embankments of soil will create a more solid structure.

〈Flood Protection Wall〉

・Since the intake ponds are connected to the sea, the rise of sea level could cause the overflow from these ponds. Preparing against the overflows of seawater onto the site from the openings of the intake ponds, we are building approximately 4-meter-high flood protection walls around the seawater pumps that are needed to cool the reactor equipment.

Ⅱ.A.2 Flooding Prevention Countermeasures 2: Preventing flooding inside reactor buildings when sea water floods on the station site

〈Emergency Sea Water Cooling System〉

・Since seawater pumps located outdoors could lose their functions as a result of the flooding, we have installed a new Emergency Sea Water Cooling System (EWS) which have robustness to the flooding.

・Although EWS contains two pumps which have the same function of the outdoor seawater pump, these were installed inside a watertight building.

・In addition, the suction line of EWS consists of multiple routes by using a seawater tunnel connected with each reactor's intake water ponds so that seawater can also be taken from intakes of other reactors.

・By taking account of the drawback of tsunami, EWS has pit structure which can store seawater needed for about 20 minutes of cooling. (5min. is maximum for opening the intake tower during drawback tsunami)

〈Protection Doors and waterproof doors〉

・Even if the site is flooded by events such as the tsunami overtops, protection doors and waterproof doors will protect safety-related equipment, such as emergency diesel generators located in reactor buildings.

・External wall doors is doubled: the inner door is watertight, and the outer tsunami protection door is built to resist wave impact.

〈Measures to Prevent Inundation of Building〉

・We have already introduced measures to prevent flooding within the site such as the installation of a tsunami protection wall, and introduced measures to prevent inundation of tsunamis up to a height of T.P. (Tokyo Peil) +15m to buildings. In order to increase safety and to prevent severe accidents or other major events, we are now introducing further measures to prevent inundation of tsunamis to buildings as part of our additional safety measures. For example, by introducing the gates which automatically closes to the openings of buildings up to the height of the intermediate roofs of the reactor buildings (about T.P.+20m).

・The air intakes and vents on the external walls of reactor buildings, used for ventilation and air conditioning, ware converted to snorkel-tubes-like shape.

・In places where the building walls have penetration with such as piping, we installed sealing material and sealing plates between the gaps to further enhance of waterproofing.

・ We enhanced the waterproofing of equipment rooms where safety-related equipments are installed. This countermeasures includes reinforcing watertight doors that had already been installed and adding new ones.

・In addition, to prevent flooding of safety-related equipments inside buildings, we are aiming at maintaining the function of all safety-related equipment.

Ⅱ.A.3 Enhanced Emergency Countermeasures

〈GTG: Gas Turbine Generator〉

・On the high ground which has elevation of 40 meters above sea level, we have installed gas turbine generators and underground fuel tanks. The gas turbine generators and safety facilities such as reactor cooling functions are connected to each other by the exclusive electric lines.

・By providing necessary power immediately by the generators, the cooling function of from the unit 3 to the unit 5 are maintained.

・6 gas turbine generators have a total of 19,200kW (3,200kW/unit) output, which is equivalent to the output of middle-size hydraulic power stations.

・The fuel tanks for these generators are also installed on the high ground, which can provide enough amount of fuel which is required at least a seven-day.

〈FCVS〉

・The filtered containment ventilating system (FCVS) is installed as a measure to remove the decay heat when RHRs are not available in an accident conditions. FCVS can release high-temperature and high-pressure gas from the containment vessel to prevent damage to it, while reducing the release of cesium and other radioactive particles to 1/1,000 or less compared with release without filters.

〈Others〉

・Enhancing the capacity of the standby batteries.

・Water Storage Tank: Provides at least necessary amount of water for a week. The tank is a back-up water storage for CST and intake water pond, and can provide enough amount of water in the case of severe accidents. The Emergency Water Storage Tank Capacity: 9,000 m^3. 2 tank structure with 4,500 m^3 each.

Ⅱ.A.4 Other Countermeasures

・The Mobile Pumps and Pipings for Feeding Water: Intake Water Pumps Truck, Feed Water Pumps Truck, Hose Cars

・The Mobile Power Supply Equipments: AC Cars, DC Cars

・ The Heavy Construction Equipments: Wheel Loaders, Bulldozer, Hydraulic Shovel, Crawler Carrier, Folk Lift

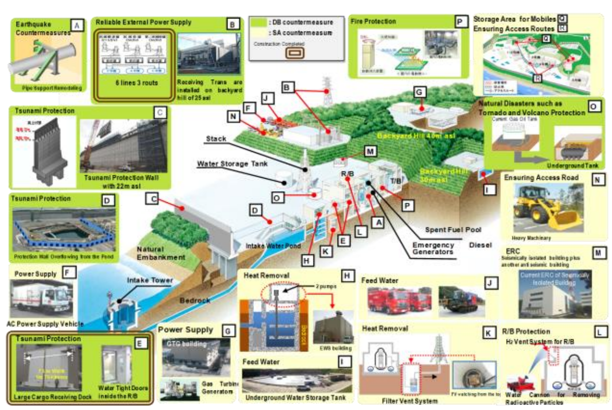


Fig. 1. Overview of Hamaoka NPP Construction Works

Ⅱ.B The countermeasures against Earthquakes

・ We have taken a number of countermeasures against earthquakes.

・ We have been considering not only a anticipated Tokai earthquake which already have been considered to happen, but also the anticipated triple-source-interlocked megaquake which is assumed to have magnitude 8.7 and to includes the anticipated Tokai earthquake as well as the anticipated Tonankai and Nankai earthquakes whose sources are located at the west side of the Nankai Trough.

・The anticipated triple-interlocked Tokai/ Tonankai/ Nankai megaquake that we envision at the Hamaoka NPP would have maximum tremor strength of 1200 gals in the bedrock.

・The Hamaoka NPP need to have high seismic resistance. Reflecting new regulations, we have considered various ground motion amplifications. As a result, we introduced several reinforcement of piping to ensure to protect safety functions from the ground motion of 1,200 gals.

Ⅲ. Outline of Seismic PRA

Ⅲ.A. Seismic PRA Regulatory Procedures

The regulatory procedures of the seismic PRA are carried out in accordance with the Atomic Energy Society of Japan Standards “Seismic PSA Standards: 2007”. These procedures mainly consist of the following four steps (As shown in Fig.2).

a. Collection and Analysis of Plant Information and General Analysis of Accident Scenarios
b. Seismic Hazard Evaluation
c. Building and Component Fragility Evaluation
d. Accident Sequence Evaluation

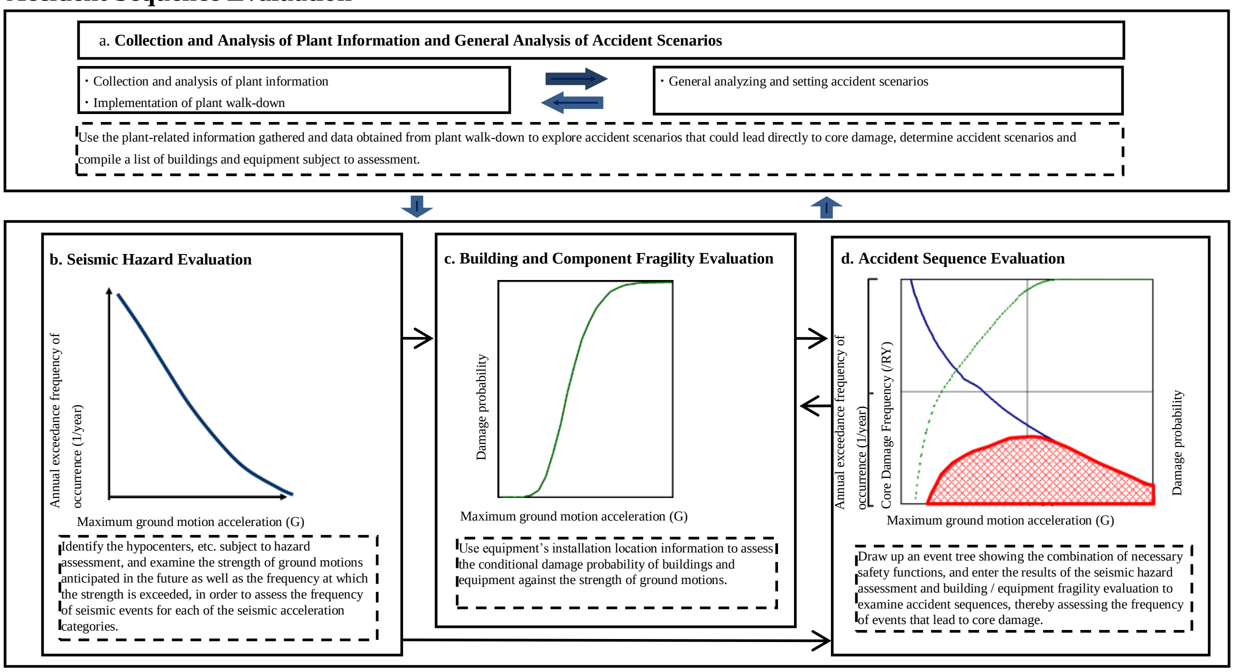


Fig. 2. Seismic PRA Evaluation Process

Ⅲ.B. Collection and Analysis of Plant Information and General Analysis of Accident Scenarios

Ⅲ.B.1. Collection and Analysis of Plant Information

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 Hamaoka NPP. These were collected while implementing the Level 1 PRA for internal events. In addition, we also collected information from documents about such as the evaluation of the earthquake-resistance, which are unique to the seismic PRA.

1) Establishment Permit of NPP
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 Level-1RRA report
10) Plant walk-down report
11) Report on human reliability analysis (NUREG/CR-1278)
12) General equipment failure rate database in Japan, etc

Ⅲ.B.2. Scenario and analysis of accidents which leads to the core damage due to earthquakes

We have analyzed the accident scenarios peculiar to earthquakes and accident scenarios were identified and selected using the gathered plant-related information and information collected by plant walk down. In this assessment, the following four wide-ranging accident scenarios illustrated in the “Seismic PSA Standards: 2007”, were analyzed and specific accident scenarios were identified.

・Accident scenarios that can directly lead to core damage accident due to the main shock: “Damage to the equipments categorized S class in seismic design”, “Damage to or random failure of equipments available during loss of external power”, “Storage battery for the seismic design S emergency generator is over discharged and fails to start”.

・ Accident scenarios that can indirectly lead to core damage accident due to the main shock: “Impact on the Reactor Pressure Vessel and Primary Containment Vessel due to the falling or dropping of the overhead crane”, “Damage to S class equipment caused by the damage of the facilities categorized B or C class in seismic design”, “Cascading effects on the reactor equipment cooling system (including the seawater system) due to the collapse of the outdoor cranes”, “Impact on the safety related equipments due to the sloshing of spent fuel pool water”, “Impact on the related equipment in the adjacent reactor buildings due to turbine missile associated with the damage of the main turbine bearing”, “Impact on reactor buildings or surrounding structures due to falling of stack”, “Impact on reactor buildings or surrounding structures due to slope failure”, “Impact on external power supply associated with the damage to transmission network towers”, “Erroneous operations of operators, site workers, and other concerned persons in the vicinity (such as who related to power supply) during or after earthquakes”.

・Accident scenarios related to aftershocks

・Accident scenarios related to aging

We analyzed the accident scenarios as follows. For the selected accident scenarios, the possibility of leading to core damage was qualitatively or quantitatively judged in from viewpoints such as negligibility of the impact of the accident or the negligibility of the frequency of the accident, the scenarios were screened and it was determined whether or not the scenarios must be considered as accident scenarios for this seismic PRA.

・Damage to the equipments categorized S class in seismic design due to the main shock

・Damage to or random failure of equipments available during loss of external power due to the main shock

・Erroneous operations of operators, site workers, and other concerned persons in the vicinity (such as who related to power supply) after earthquakes

Ⅲ.C. Seismic Hazard Evaluation

Ⅲ.C.1. Method of probabilistic seismic hazard assessment

・Perform assessment based on the method described in the “Seismic PSA Standards: 2007”.

・In view of amplification characteristics of ground motions vary with the place on the site, we set both observation points those do not show significant amplification of ground motions as well as those which show significant amplification of ground motions.

Ⅲ.C.2. Main hypothesis of the assessment

〈Setting the hypocenter model and ground motion propagation model〉

・ Set the anticipated Nankai Trough earthquake and the anticipated inland crustal earthquake as specific hypocenter models, and the anticipated oceanic intraplate earthquake and the anticipated inland crustal earthquake as area hypocenter models.

・Use the distance attenuation formula shown by Noda et al. (2002) as the seismic motion propagation model.

〈Compiling a logic tree〉

・ Consider the anticipated Nankai Trough earthquake and the anticipated inland crustal earthquake as specific hypocenter models, and the anticipated oceanic intraplate earthquake and the anticipated inland crustal earthquake as area hypocenter models. Especially, map the anticipated Nankai Trough earthquake, which is considered to have a major impact on the site, to a detailed logic tree diagram for assessment.

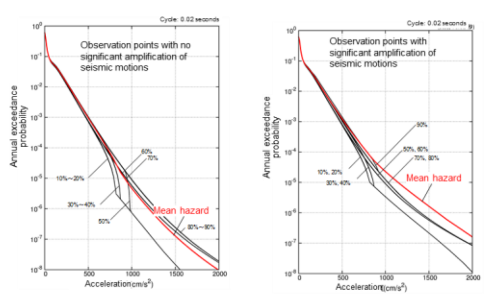


Fig. 3. Fractile seismic hazard curves (horizontal motions)

・In the assessment of observation points which show significant amplification of ground motions, reflect the significant amplification of ground motions to the logic tree diagram.

・The fractile seismic hazard curves (horizontal motions), assessed with the logic trees are shown in Fig.3.

Ⅲ.D. Building and Component Fragility Evaluation

Ⅲ.D.1. Facilities subject to assessment

・We set facilities included in the building / equipment list of the Hamaoka NPP Unit 4 as subjects to assessment.

Ⅲ.D.2. Input ground motions

・ Hamaoka NPP Unit 4 is located where's closest observation points show no significant amplification of ground motions. For this reason, the ground motions for fragility evaluations are based on the target spectrum defined in view of the probabilistic seismic hazard evaluation at observation points with no significant amplification of ground motions. As shown in Fig.4, suitable seismic waves are simulated accordingly.

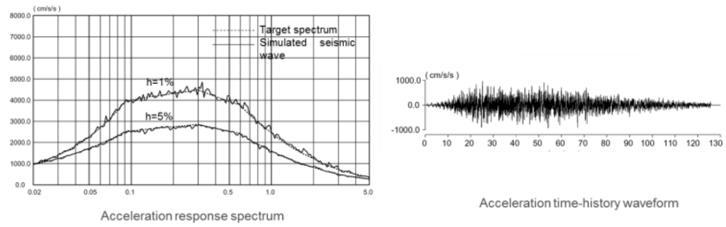


Fig. 4. Seismic waves matching the uniform hazard spectrum

Ⅲ.D.3. Approach to selecting the fragility evaluation method

・The AESJ Seismic PSA Standards stipulate that fragility evaluation should be carried out in one of the following methods (1), (2) and (3). The suitable evaluation methods for buildings, important outdoor structures and equipment, differences between the evaluation methods and reasons for selecting the methods are described in Table. 1.

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

Table 1. Each fragility evaluation method

	Application	Differences between methods	Reasons for selection
(1) Method based on response analysis: Using realistic resistance and realistic response	・Buildings ・Important outdoor civil engineering structures	Direct evaluation up to non-linear response	Building response examines story collapse, and the response of important outdoor civil engineering structures examines structural collapse. Due to the need to examine up to the strongly non-linear region, this method uses non-linear response analysis to evaluate direct realistic response.
(2) JAEA method: Using realistic resistance and response coefficient	–	Expressing response in the non-linear region in linear response, and using the energy absorption coefficient to take non-linear characteristic into account in the response term	–
(3) Safety coefficient method: Using resistance coefficient and response coefficient	・Equipment	Expressing response in the non-linear region in linear response, and using the energy absorption coefficient to add extra realistic resistance and take non-linear characteristic into account in the resistance term	Existing insight is used to determine the median value and uncertainty of the resistance coefficient, but this method has the level of accuracy equivalent to (1) – (3) in principle. The response coefficient is evaluated based on the results of response analysis with track records in existing processes. As the results demonstrate accuracy comparable to the method based on response analysis (1), the safety coefficient method (3) is believed to be capable of performing appropriate evaluation.

Ⅲ.D.4. Fragility evaluation results

・The Fragility of approximately 320 items of Buildings and Components were evaluated in this seismic PRA for the Hamaoka NPP Unit 4.

・Since it is thought to be difficult at this stage to evaluate the fragility of mobile-type equipment, such as fire trucks and power-supply cars which were deployed after the Fukushima-Daiichi NPP accident, however, we now plan to research formulations for the fragility evaluation method for such mobile equipment types.

Ⅲ.E. 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 models for the system and for an accident sequence that will lead to the core damage based on the analysis results for the accident scenario in Section Ⅲ.B. Using these models, we quantify the CDF related to earthquakes. Then, we performed analysis of the important accident sequences. The point that our evaluation is based on assumption of perfect correlations for the damage caused by earthquakes is different from the internal event PRA at power. In other words, we assume that all equipment which is similar in type will be damaged when one of them is damaged by earthquakes. Therefore, for conservative evaluation, the improvements of reliability against earthquakes gained from improving the redundancy of the system are not expected. Although the THERP (NUREG/CR-1278) was used to evaluate the Human Error Probability (HEP) as in the internal PRA, a stress factor of 5 was assumed for the HEP used in the seismic PRA which related to the operator actions those should be done in a relatively short period of time (within a few hours) after earthquakes since high-stress state due to the confusion caused by the earthquakes and the aftershocks. (However, it is assumed to be constant regardless of the scale of the ground motion). We set 24 hours as the mission time in order to confirm the effect of random failure on the same conditions as of internal events PRA. In addition, for the evaluation of CDF before the implementation of additional safety countermeasures, recovery actions of component which damaged by the ground motions is not expected. We should be care about that it is discussed in the revising actions of AESJ seismic PRA Standards that the evaluation of integrity of the fuel in the spent fuel pool (SFP) could be important in view of lessons learned from the Fukushima-Daiichi NPP accident. Therefore, although the objective of this seismic PRA was to evaluate the frequency of the damage of the fuels in the Reactor Pressure

Vessel (RPV) during the plant operation, we also plan on further evaluating the fuel damage frequency of SFP due to earthquakes.

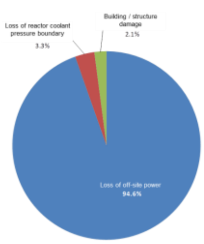
Ⅳ. Evaluation

Ⅳ.A. Summary of the Evaluation Results

In order to confirm the effectiveness of the various safety countermeasures and Severe Accident (SA) countermeasures implemented before and after the Fukushima-Daiichi NPP accident and to provide knowledge for the further safety enhancement, we first performed the CDF evaluation of the plant state before the implementation of safety countermeasures, which were implemented based on the lessons learned from Fukushima-Daiichi NPP accident. The total CDF is 1.9E-06(/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.

Ⅳ.A.1. CDF by initiating events and Analysis of the primary accident sequences

The CDF categorized by initiating events is shown in Fig.5 and Table.2. The dominant accident sequence which has the largest contribution to the CDF is the scenario of Loss of off-site power. The CDF is 1.8E-06(/RY), which comprises about 95% of the total CDF. That is, this is the accident scenario caused by random failure after Loss off-site power due to the earthquake, such as the failure of emergency D/Gs or RHRs. The accident sequence which has second largest contribution to the total CDF is the accident sequence of Loss of reactor coolant pressure boundary integrity due to the earthquakes. The CDF is 6.2E-08(/RY), which comprises about 3% of the total CDF. Here, Loss of reactor coolant pressure boundary integrity is assumed to occur when the most fragile piping of each system in the PCV fails due to the earthquakes, and conservatively assumed to be Excessive LOCA leading directly to the core damage. The accident sequence which has third largest contribution to the total CDF is the scenario of the damage to buildings and structures (R/B) due to the earthquake. The CDF is 3.8E-08(/RY), and comprises about 2% of the total CDF. Here, the damage to buildings and structures (R/B) is assumed to occur when damage to weakest portion of each earthquake-resisting wall of the reactor building, and conservatively assumed to lead directly to core damage. Since contribution to the total CDF of “Loss of reactor coolant pressure boundary integrity” and “Damage to buildings and structures (R/B)” is very low, the earthquake-resistant structures, systems and components are considered as a result that there is sufficient seismic resistance, because we have been carried out continuously the construction work for improvement of the seismic safety margin in the past. (As shown in Fig.6 and Fig.7)



Initiating event	Core damage frequency (/reactor year)	Contribution (%)
Loss of off-site power	1.8E-06	94.6%
Building / structure damage	3.8E-08	2.1%
Loss of reactor coolant pressure boundary	6.2E-08	3.3%
Total	1.9E-06	100.0%

Fig.5. / Table 2. CDF categorized by initiating events

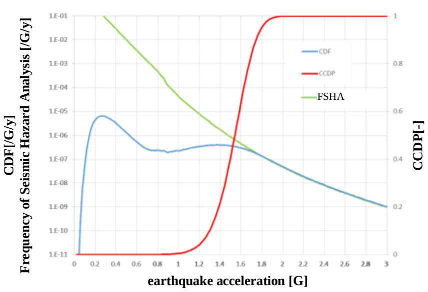


Fig. 6. Seismic acceleration and CDF

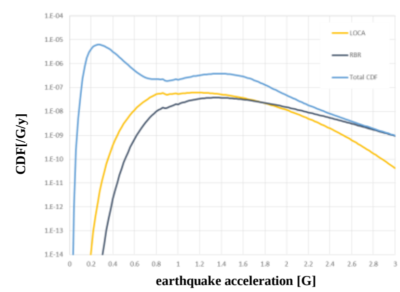
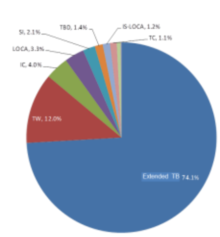


Fig. 7. Seismic acceleration and CDF of the primary accident sequences

Ⅳ.A.2. CDF classified by reactor core damage sequence and Analysis of the important accident sequences

Fig. 8 and Table 3. both show the CDF categorized by the accident sequence of the reactor core damage. Here, the loss of AC power sequence (Extended TB sequence) has the largest contribution account for approximately 74% of total CDF, and the CDF is 1.4E-06(/RY). The dominant accident sequence that has the largest contribution to the CDF is the accident sequence where all AC power is lost (Extended TB sequence) due to random failure of emergency D/Gs, by fail to run and fail to start-up. The loss of decay heat removal sequence (TW sequence) had the second largest contribution account for approximately 12% of total CDF, and the CDF is 2.2E-07(/RY). The accident sequence that has second largest contribution to the total CDF is the accident sequence where failure of decay heat removal is lost (TW sequence) due to random failure of RHRs.



Accident sequence group	Core damage frequency (/reactor year)	Contribution
Extended TB	1.4E-06	74.1%
TW	2.2E-07	12.0%
IC	7.4E-08	4.0%
LOCA	6.2E-08	3.3%
SI	3.8E-08	2.1%
TBD	2.5E-08	1.4%
IS-LOCA	2.3E-08	1.2%
TC	2.1E-08	1.1%
TBU	8.5E-09	0.5%
TBP	4.1E-09	0.2%
TQUX	2.5E-09	0.1%
TQUV	9.5E-13	⟨0.1%
Total	1.9E-06	100.0%

Fig. 8. / Table 3. CDF categorized by reactor core damage sequence

Although the pressure control by safety relief valves and the water injection into the RPV by the Reactor Core Isolation Cooling systems (RCIC) success after the SBO occurs, the core damage occurs as a result of failure of continuous water injection due to the depletion of water source or the batteries. These two accident sequences described above are both random failure due to the earthquake, these sequences account for approximately 86% of the total CDF. Accident sequences other than that above these two are, as shown in Table 3. and Fig. 8., loss of the Instrumentation and Control system sequence (IC sequence) (CDF: 7.4E-08(/RY), contributing proportion: approx. 4%) and loss of reactor coolant pressure boundary sequence (LOCA sequence) (CDF: 6.2E-08(/RY)), contributing proportion: approx. 3%), and damage of building and structure sequence (SI sequence) (CDF: 3.8E-08(/RY)), contributing proportion: approx. 2%). Here, “IC sequence”, “LOCA sequence”, “SI sequence”, “TBD sequence”, “IS-LOCA sequence” and “TC sequence” are assumed to be accident sequences that leads directly to the core damage. Additionally, our approach for Seismic PRA is considerably conservative since our evaluation was based on assumption that all safety equipment which is similar in type will be damaged when one of them is damaged by earthquakes.

Ⅳ.B. Importance Analysis

Table 4. shows the results from the importance analysis (Fussell-Vesely index). Among the top SSCs in terms of FV index, the random failure of emergency D/Gs by fail to run and fail to start-up are rated the top, scoring total 0.70.

They are followed by the random failure of the residual heat removal system which scores total 0.12, cable tray damage, and RCIC pipe damage. The FV index results constitutes the upper ranks just as shown in the results of the analysis of the primary accident sequences in Section Ⅳ.A. The results also show that countermeasures for Extend TB sequence and TW sequence are effective for reducing risk. Although the original objective of the improvement of diversity of the AC-power source, heat-removal and other capabilities shown in Section Ⅱ. is to implement additional countermeasures against tsunami, we can see that they are also effective to reduce CDF due to earthquakes as well.

Table 4. Fussell-Vesely index

Order	FV index	Component
1	4.1E-01	Random failure of emergency D/G by fail to run
2	2.9E-01	Random failure of emergency D/G by fail to start-up
3	1.2E-01	RHR random failure
4	6.3E-03	Cable tray damage
5	5.0E-03	RCIC pipe damage
6	4.7E-03	Function loss of the seawater heat exchanger building's air supply fans
7	4.7E-03	Function loss of the seawater heat exchanger building's exhaust fans
8	4.5E-03	S/P box support damage
9	4.2E-03	Damage to RPV brackets, etc.
10	4.1E-03	Damage to the control rod drive mechanism's housing

Ⅳ.C. Efficacy of severe accident management countermeasures

Although various SA countermeasures were implemented in the Hamaoka NPP, we consider only countermeasures that were able to secure seismic-resistance in this evaluation for the Unit 4. Main SA countermeasures consist of installing the AC power source via GTG, the new cooling system called Emergency Sea Water Intake System (EWS), and the back-up water storage. We can reduce the CDF of the low acceleration range of the ground motion by reducing the random factor which can be seen in the Extend TB sequence and the TW sequence by SA countermeasures. In addition, the result also show that introducing SA countermeasures reduces the factors of random failure (the random failure of emergency D/Gs by fail to run and fail to start-up, the random failure of the residual heat removal system) of FV index. By the implementation of SA measures, the CDF value will be reduced approx. 90%. In the future, we will continue to establish the fragility assessment methods of mobile equipment (such as the power supply cars and the feeding water trucks) for further safety improvement countermeasures.

Ⅴ. Conclusion

We confirmed the vulnerable areas of the plant through performing a seismic PRA for Unit 4 at Hamaoka NPP before the SA countermeasures installed. Based on the characteristics and trends related to the risk of Unit 4 found in the results, we confirmed that the various SA countermeasures that had been implemented based on lessons learned from the Fukushima-Daiichi NPP accident were also effective for earthquakes. Hereafter, in order to implement additional effective countermeasures for 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.

Additionally, we continue to establish the fragility evaluation method for mobile SA-countermeasures and establish seismic PRA for Unit 3 and Unit 5 at Hamaoka NPP in consideration of countermeasures which are currently being implemented and planned hereafter, and we also plan to focus on the tsunami PRA and SFP evaluations, as well.

REFERENCES

1. *Implementation Standard for Seismic Probabilistic Safety Assessment of Nuclear Power Plants: 2007*, Atomic Energy Society of Japan (AESJ), Tokyo, Japan (2007).