Effectiveness Evaluation of the Measures for Improving Resilience of Nuclear Structures against Excessive Earthquake (2) Accident Sequences Analysis

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Abstract: The objective of this study is to implement an effectiveness evaluation of the measures for improving resilience of nuclear structures against excessive earthquake beyond a design basis ground motion. In this study, those measures for improving resilience have an effect to enlarge their seismic safety margin. To evaluate effectiveness of those measures, seismic core damage frequency (CDF) is selected as an index. Reduction of CDF as an effectiveness index is quantified by applying seismic PRA technology. Target system is a loop-type next-generation sodium-cooled fast reactor, which adopts the building isolated from horizontal seismic ground motion. Even if the reactor vessel (RV) is buckled due to seismic shaking, it is expected that the RV maintains stable state without unstable failure such as rupture, collapse. Realistic consideration of the post-buckling behavior is regarded as a measure for improving resilience in this study. Two cases were set for improving the resilience in the accident sequences analysis, in addition to a base case assuming bucking failure. The first case assumes low-cycle fatigue failure after buckling as the realistic failure mode of the RV for the fragility evaluation in this study. After the RV fatigue failure, the behavior of failure propagation is very uncertain. As the second case, the median seismic capacity to loss of reactor coolant level is assumed to be slightly larger than that of fatigue failure of the RV. Analyses for both cases were performed, and the results were compared to the base case indicating significant reduction of CDF. Within the assumption, the measures for improving the resilience were significantly effective for decreasing CDF in excessive earthquake up to several times of the design basis ground motion. The seismic PRA technology could serve to the effectiveness evaluation of the measures for improving resilience of nuclear structures against excessive earthquake.

Keywords: Sodium-cooled fast reactor, probabilistic risk assessment, excessive earthquake, accident sequence.

1. INTRODUCTION

As part of development of next-generation advanced reactors, Japan Atomic Energy Agency (JAEA) has been developing sodium-cooled fast reactors (SFRs) in Japan. Unlike light water reactors, the boiling temperature of sodium as the reactor coolant is very high at the atmospheric pressure, so that a loss of coolant accident due to depressurization boiling could not happen at a normal operating temperature even if reactor coolant boundary fails as long as reactor coolant level is maintained. In SFRs, the reactor coolant level required for core cooling is maintained by the static components such as guard vessels so that active safety function such as core coolant injection is not necessary. Decay heat removal in typical SFRs can be achieved by natural circulation of coolant sodium and natural air draft at the air cooler of the final heat sink so that active components such as circulating pumps and air blowers are not needed. Safety in typical SFRs could be achieved less or no dependent on electric power. Integrity of static components which are needed for decay heat removal is important for safety.

Coolant sodium boundary consists of thin-walled structure to avoid excessive stress due to anticipated thermal transient at high temperature. Mechanical load on thin-walled structure caused by an earthquake should be limited more strictly than thick-walled structure. SFR-specific efforts have been made on seismic design of sodium boundary structures such as reactor vessel (RV), pipes, etc. Next-generation SFR in Japan is designed to enhance seismic resistance by introducing the seismic isolation system for the reactor building. When the seismic ground motion becomes larger beyond design basis ground motion, seismic load on SSCs increases non-linearly with causing damage or failure of SSCs in the seismic-isolated building. This arises from hardening of the laminated rubber bearings which the isolation system consists of. The characteristics of the laminated rubber bearings should be understood quantitatively for more realistic estimation of the fragility. The next-generation SFR adopts not the conventional laminated rubber bearings but thick ones.

Recent year, there is progress in experimental studies to obtain the various data of the thick laminated rubber bearings considering aging effect: e.g., linear strain limit, breaking shear strain, etc. [1][2][3].

Based on these safety features, JAEA has been implementing studies on level-1 probabilistic risk assessment (PRA) for SFRs to evaluate safety of SFRs quantitatively. Particularly an earthquake is recognized as a significant risk contributor in reactor safety among the external hazards in Japan. In 1990s, JAEA implemented the seismic level-1 PRA for the existing typical loop-type SFR in Japan, which has three main cooling loops and has no seismic isolation system. [4][5] This PRA indicated that seismic-specific common cause failure was a significant contributor to the core damage frequency (CDF): i.e., failure of the reactor building which could cause loss of all structures, systems, and components (SSCs) needed for the decay heat removal in a natural circulation mode. In 2010s, JAEA studied on the seismic PRA for the next-generation SFR having seismic-isolated building. As part of this study, fragility of representative SSCs were evaluated by considering seismic response in the seismic-isolated building. The result showed that fragility of the reactor building was lower than that of other SSCs and that of buckling of RV was relatively high. [6] The seismic PRA for this SFR showed that buckling of RV was a dominant contributor to the seismic CDF and failure of the reactor building was negligible contributor. [7] This implicates that seismic isolation system is effective for improvement of seismic resistance of the building. However, seismic CDF is not small negligibly yet. The sensitivity study showed that for further reduction of CDF is needed to increase seismic margin of for maintaining reactor coolant level by additional safety provisions: e.g., third vessel surrounding RV and its guard vessel. [7]

There are different studies on the seismic PRA for pool-type SFRs in U.S.A. One is a seismic PRA for the existing SFR of EBR-II. [8][9] Seismic CDF is larger than CDF due to internal events and fire. Dominant accident sequence in seismic core damage is structural failure of the primary tank hangers, which is a common cause of loss of reactor scram and loss of decay heat removal. Another is a seismic PRA for the next-generation SFR of PRISM which has seismic isolation system. [10] Seismic CDF is comparable to CDF due to internal events. Seismic dominant accident sequence is common cause failure to insert all control rods due to buckling of core support platform which is followed by loss of inherent feedback system while heat removal could continue by the reactor vessel auxiliary cooling system under the earthquake beyond the capability of seismic isolation system. Reduction of seismic fragility of the reactor structures against such an earthquake would be effective to reduce seismic CDF.

The seismic failure mode of SSCs in the past study was assumed by considering the design limit, i.e., assuming buckling for RV and shell of intermediate heat exchangers, bending failure for pipes. [7] On the one hand, experimental and analytical study on the thin-walled cylindrical vessel such as RV for SFRs indicated that failure under the earthquake is caused by crack penetration due to cyclic fatigue and it is predictable by estimation of the fatigue damage if unstable behavior on the entire structure is prevented even after shaking-induced buckling. [11] Another experimental and analytical study on the thin-walled pipes for SFRs which are made of stainless-steel material indicated that failure mode under seismic load is not unstable failure such as collapse but crack penetration due to cyclic fatigue and it is predictable by estimation of the fatigue damage (i.e., usage factor). [12][13] Hence, more realistic (i.e., less conservative) estimation of seismic fragility of vessel and pipes needs more realistic assumption of failure mode.

In conventional structural designs, efforts have been focused on prevention of failure of SSCs for design basis events. Kasahara et al. newly proposed the fracture control technology that can mitigate failure consequences (hereafter, failure mitigation technology) against beyond-design-basis events.[14][15][16] In this technology, unstable failure modes, like fracture or collapse, are prevented by allowing small failure modes. Introduction of this technology combined with conventional accident management measures is expected to become one of promising measures for improving resilience of nuclear structures. According to application of the failure mitigation technology to seismic design,[15][16] if small breakage or deformation occurs on the SSCs in an excessive earthquake that exceeds significantly design basis earthquake which is considered as one of beyond-design-basis events, the natural frequency of the SSCs would decrease from the dominant input seismic frequency range. As the result, earthquake-induced catastrophic failure would be prevented. Thus, by introducing the failure mitigation technology against excessive earthquake, it is expected to increase seismic margin of the SSCs and then to improve resilience of nuclear structures.

Currently, there is no methodology to evaluate effectiveness of the measures for improving resilience of nuclear structures against excessive earthquakes. Therefore, it is necessary to develop an effectiveness evaluation methodology against excessive earthquakes. From the above-mentioned background, actual failure under the earthquake would be caused by not the buckling but the crack penetration due to cyclic fatigue and it is predictable by estimation of the fatigue damage. This might be explained by the failure mitigation technology against the excessive earthquake for improving the resilience. Realistic evaluation of seismic failure of nuclear structure such as RV could be regarded as one of measures for improving resilience, i.e., enhancing safety in-depth.

The project of research and development is being implemented for four years since 2020 [16][17]. One of the subjects in the project is to develop an evaluation methodology for the effectiveness of the measures for improving the resilience after applying the failure mitigation technology to nuclear structures. The general concept of this evaluation methodology was developed in the previous paper.[18] Using the methodology, this study aims to evaluate the effectiveness of the measures that improve structural resilience.

2. METHODOLOGY OF EFFECTIVENESS EVALUATION

This study postulates occurrence of an initiating event induced from the earthquake and failure of existing measures, which include measures against design basis accident and severe accident. This study assumes the measures for improving resilience, and the success path of those measures is drawn as red line in the event tree shown on the left-hand side of Fig. 1.

In our four-year joint research project, Kasahara et al. introduced the resilience index quantitatively to evaluate the resilience-enhancing effects caused by the failure mitigation technology.[16] The resilience index is the expected value of the margin for the minimum requirement of time and safety performance, which is evaluated by considering the time change of the safety performance. This margin could represent well the characteristics primarily associated with success path in the event tree shown in Fig. 1. When the margin is large, the resilience-enhancing effects is evaluated as large.

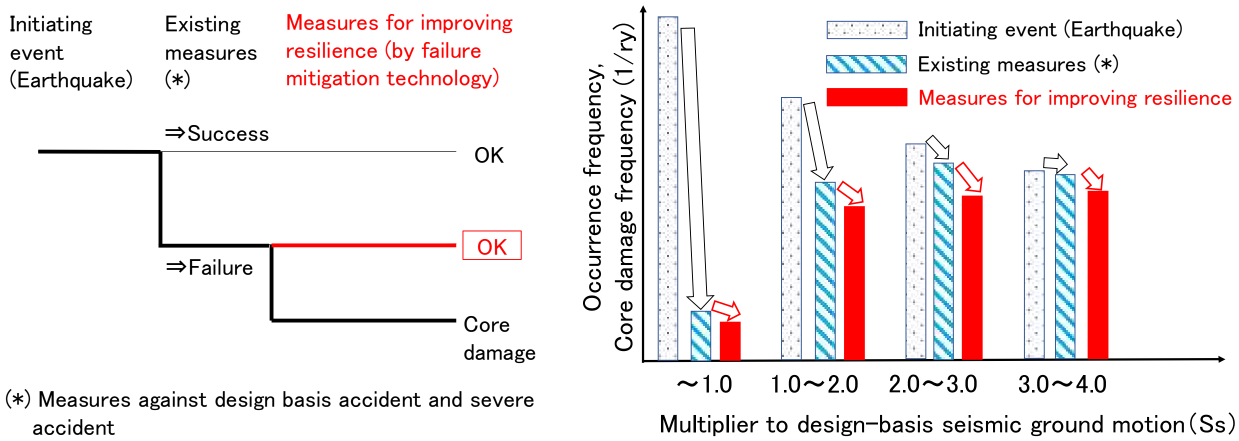


Figure 1. General Concept of the Effectiveness Evaluation of Resilience-improvement Measure against an Excessive Earthquake

To complement this resilience index, this study paid attention to the failure path in the event tree in Fig. 1. We presented the general concept that the effectiveness of the measures to improve resilience is evaluated by quantifying reduction of the CDF by implementing those measures for each intensity of seismic ground motion as shown on the right-hand side of Fig. 1.[18] To evaluate effectiveness of those measures, seismic CDF is selected as an index. Reduction of CDF as an effectiveness index is quantified by applying seismic PRA technology. This is the methodology of the effectiveness evaluation.

3. APPLICATION OF THE METHODOLOGY TO SFR PLANT

To examine applicability of the proposed methodology of the effectiveness evaluation of the measure for improving resilience by using this event tree model, the effectiveness evaluation was performed targeting a typical loop-type next-generation SFR designed in Japan as an example. [19]

3.1. Outline of Target SFR Plant

JAEA has promoted conceptual design study of the SFR. As shown in Fig. 2, the SFR has a double walled structure, which means components and pipes containing primary coolant sodium are covered by guard vessels and guard pipes to maintain sodium levels for decay heat removal in the accident or unlikely event of coolant leakage in the primary system. Also, the SFR has three circuits for decay heat removal system (DHRS) using natural circulation: a Direct Reactor Auxiliary Cooling System (DRACS) and two Primary Reactor Auxiliary Cooling Systems (PRACS) (Fig. 2). Decay heat can be removed if any one of the three systems is available. As for the seismic design, a seismic isolation system is introduced at the base mat of the reactor building to be isolated from the seismic ground motion as shown in Fig. 2. The seismic isolation system of the SFR is made of laminated natural rubber bearings thicker than industry-used ones and oil dampers.[20]

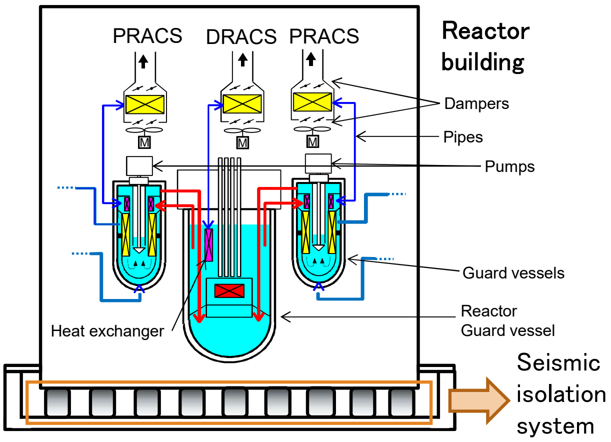


Figure 2. Illustration of Representative Next-generation SFR Plant

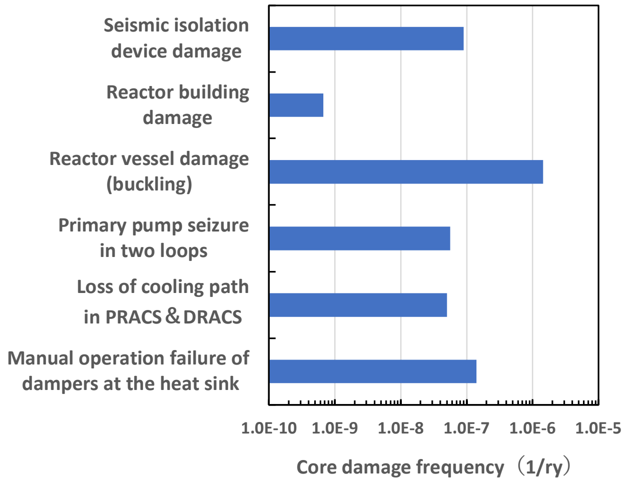


Figure 3. Comparison of Occurrence Frequency between Seismic Core Damage Sequences of the SFR

3.2. Accident Sequences

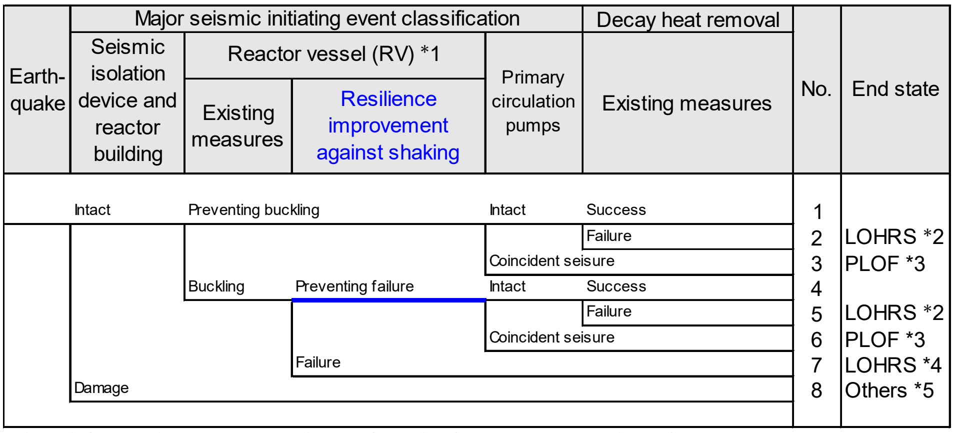
According to the previous study on the seismic PRA of the SFR, [7] the occurrence frequency of significant accident sequences to core damage is summarized as shown in Fig. 3. A dominant contributor to core damage sequences is structural damage (i.e., buckling) of the RV, and this occupies 81% of total CDF. The previous study assumed that when the RV is damaged (i.e., buckled), the reactor guard vessel also becomes dependently failure and the reactor coolant leak out of both those vessels so that the reactor core becomes exposed to cover gas leading to core damage. This study focuses on the dominant contributor, RV damage. Damage of the seismic-isolation device and damage of the reactor building are conservatively regarded as initiators directly causing core damage because the consequence is uncertain. Simultaneous seizure of the two pumps in the primary cooling system is an initiator that causes rapid decrease of the core coolant flow which could result in core damage. The other accident sequences are failures in decay heat removal after reactor scram.

3.3. Measures for Improving Structural Resilience

The failure mitigation technology is introduced to the measure for improving structural resilience against strong shaking due to excessive earthquake. Currently the failure mitigation technology against strong shaking aims to utilize inherent characteristics of the metallic structure such as the RV: i.e., reduction of the stiffness after plastic deformation caused by strong shaking. Significant reduction of stiffness would decrease the input energy from the shaking to the deformed structure so that it might prevent catastrophic failure (e.g., rupture) of the deformed structure. As mentioned above, in the previous studies, [6][7] buckling distortion of RV was considered as core damage. However, even if the RV is buckled due to seismic shaking, it is expected that the RV maintains stable state without unstable failure such as rupture, collapse. Realistic consideration of the post-buckling behavior is regarded as the measure for improving the resilience in this study.

3.4. Effectiveness Evaluation

The methodology of effectiveness evaluation was applied to the SFR for an effectiveness evaluation of the above-mentioned measure for improving resilience. Figure 4 shows a detailed event tree model, which was modified from the previous seismic PRA study.[7] This tree model assumes that an excessive earthquake does not cause losses of reactor shutdown functions. Addition of the tree branch at the measure for improving resilience is a change from the previous study. Damage of the seismic-isolation device, damage of the reactor building and simultaneous seizure of the two pumps in the primary cooling system are included in the accident sequences, and these are differentiated from loss of heat removal system because the measure for improving resilience is not applied to these accident sequences in the present study. This study assumed that the seismic hazard and various seismic failure probability of SSCs other than RV take the same numerical values as those in the previous seismic PRA study. [7]



LOHRS (Loss of Heat Removal System), PLOF (Protected Loss of Flow)
*1 Guard vessel assumed to become dependent loss of function when reactor vessel becomes damaged
*2 This results in core damage at the ultra-high temperature condition.
*3 PLOF could cause rapid loss of core flow resulting in core damage due to core temperature rising.
*4 It is assumed that reactor coolant leak causes lowering reactor coolant level where the core is exposed resulting in core damage.
*5 Since effect of damage is unknown, core damage is assumed conservatively. Realistic evaluation is future work.

Figure 4. Detailed Event Tree against an Excessive Earthquake in the SFR

Since damage of the RV is dominant in Fig. 3, failure mitigation technology assumed to be applied not to DHRS but to the RV for resilience improvement against shaking by excessive earthquake. The possibility of its success and failure is considered in the event tree structure of Fig. 4. The RV is conventionally designed to prevent from buckling. In this point of view, when the RV is regarded as existing measure in Fig. 4, damage probability of the RV is calculated assuming buckling failure mode, and the reactor guard vessel assumed to be dependently damaged when the RV is damaged by strong shaking in the same way as the previous study. [7]

Through the conceptual design work, RV thickness was increased to enhance seismic capacity after the previous study. [7] Based on this, a seismic fragility of the RV buckling was newly evaluated by implementing the RV structural analysis with the finite element method [21] for the base case in this study: i.e., Case 1 in Table 1. The evaluated mean fragility curve was compared to that in the previous one, and we confirmed that median seismic capacity is improved from 3.5 times of Ss to around 5 times of Ss as shown in Fig.5. These RV buckling probability curves were obtained by considering the aleatory and epistemic uncertainties which are included in the seismic response of RV and the seismically isolated building and in the seismic capacity of RV. The evaluated percentile fragility curves are drawn in Fig. 6 (a) in addition to the mean fragility curve.

For considering the measure for improving resilience, post-buckling behavior of the RV is also analyzed with the finite element method, and the RV fatigue failure fragility was evaluated. [21] This study considered fatigue damage only from main quake. However, when afterquake is given, fatigue damage accumulation due to the afterquake can be considered. The result is shown in Fig. 6 (b). The median capacity is increased from around 5 times of Ss to around 6 times of Ss. This increase is significant for improving resilience. On the one hand, it seems smaller than the uncertainty in the fragility curves. Since the seismic response behavior of the RV given the seismic floor response wave is realistically analyzed, the uncertainty in the seismic response evaluation of the RV would be small and the uncertainty in the fragility would be mainly determined from the uncertainty in the response analysis for the building. Safety factor method is used in our fragility evaluation. The uncertainty in the building response factor, which is an element of the safety factor, was determined by the expert.

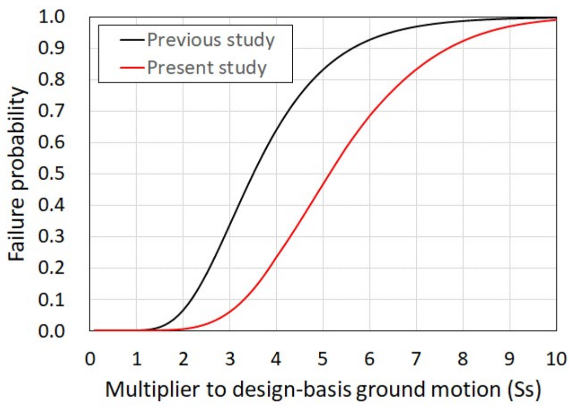
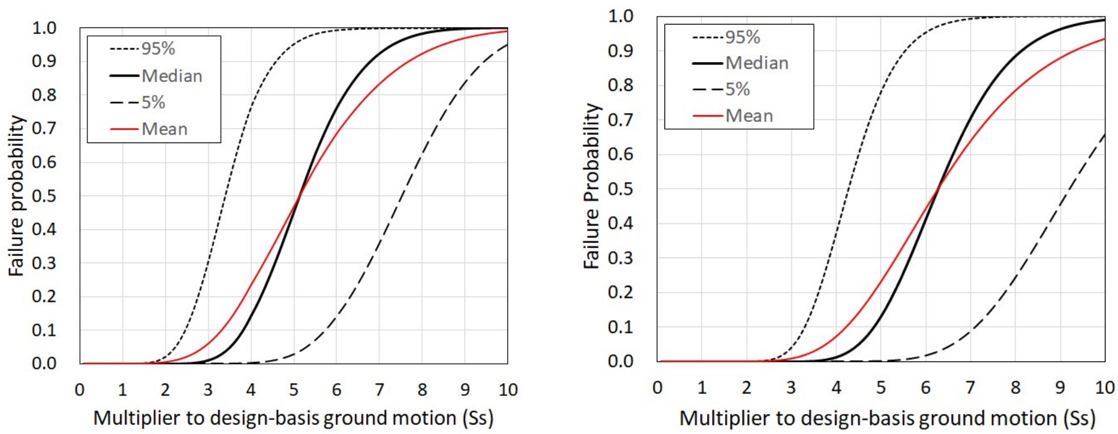


Figure 5. Comparison of Mean Fragility Curves for RV Buckling in the SFR
Previous study: Naruto et al., NTHAS9 (2014)



(a) RV Buckling Fragility in This Study (b) RV Post-Buckling Fatigue Failure Fragility
Figure 6. Comparison of RV Seismic Fragility between Different Failure Modes

As for the fatigue failure fragility evaluation, some factors included in the safety factor are not yet analyzed and evaluated in this study, so they are conservatively assumed. For example, a conservative value of yield stress for design work was applied for the RV structural analysis, and this value was regarded as a realistic one in quantifying the related safety factor, which means the resultant fragility still includes conservativeness. The effectiveness evaluation of the measure for improving resilience needs comparison of the fragilities between buckling and fatigue failure which are evaluated under the same analytical condition. So, the same factors of buckling evaluation were also conservatively assumed in the same way as fatigue failure evaluation.

The behavior of failure propagation after the RV fatigue failure is very uncertain. Under the excessive earthquake, after the RV fatigue failure, the failure might propagate to unstable one such as break, or the failure propagation might stop within the stable state preventing unstable failure such as rupture. The RV is hung from the upper deck floor. If the RV failure propagates to circumferential break, the RV would drop onto the guard vessel then the guard vessel, which is also hung from the upper floor, would be broken resulting in loss of reactor coolant level and/or loss of heat removal path in the RV. This study assumed that the median seismic capacity to loss of reactor coolant level and/or loss of heat removal path in the RV is 1.1 times of that to the RV fatigue failure for the analysis case considering success possibility in maintaining the reactor coolant level by the guard vessel: i.e., Case 3 in Table 1. The assumed mean fragility curve is plotted as red solid line in Fig. 7. Numerical value of 1.1 times is just an assumption. Evaluation is a future subject.

If the guard vessel is placed on the bottom floor, even if the RV would drop onto the guard vessel, the guard vessel might not be broken, and the reactor coolant level might be maintained for decay heat removal. However, in this case we should examine the integrity of the reactor internals such as pipes, heat exchangers inside the RV that form decay heat removal path.

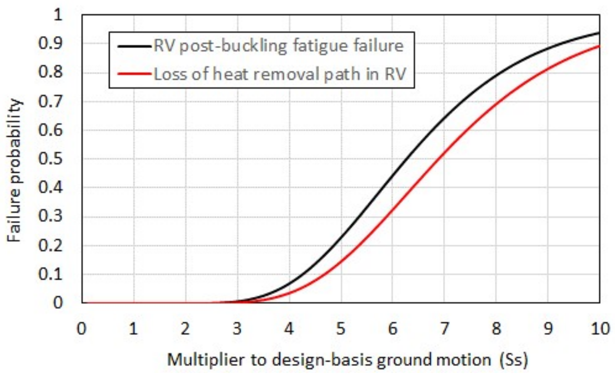


Figure 7. Assumed Mean Fragility Curve for Loss of Heat Removal Path in RV

The CDFs were calculated for various cases as shown in Table 1. In Case 1, the existing measures against design basis accident and severe accident are taken, but the measure for improving resilience is not considered. The RV buckling distortion is regarded as the RV failure. In Cases 2 and 3, the measure for improving resilience against shaking by excessive earthquake is considered in addition to the existing measures. Not the RV buckling but the RV post-buckling fatigue failure is regarded as the RV failure. In Cases 1 and 2, we assumed that the seismic RV failure results in loss of reactor coolant level: i.e., the guard vessel loses its function. In contrast, Case 3 considers a possibility to maintain the reactor coolant level by the guard vessel after the RV failure, and Case 3 assumed a probability to prevent failure propagation from fatigue failure to loss of reactor coolant level.

Table 1. Analysis Cases

Case	Measure to improve resilience	RV failure criterion	Maintaining reactor coolant level by guard vessel after RV failure
1	Not considered	Buckling due to shaking	Not considered conservatively
2	Considered	Fatigue failure after buckling	Not considered conservatively
3	Considered	Fatigue failure after buckling	Assumed a probability to prevent failure propagation from fatigue failure to loss of reactor coolant level

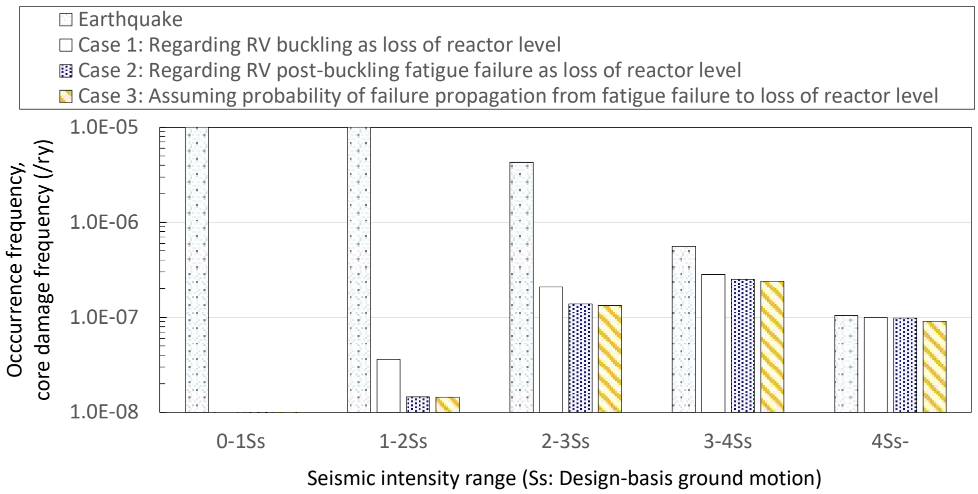


Figure 8. Comparison of Earthquake Frequency and CDF between Cases 1 to 3 at Each Seismic Intensity Range in the SFR

Figures 8 and 9 show the CDF calculation results for each seismic intensity range. The design basis seismic ground motion (Ss) is assumed 565 gal for the SFR site in this study. However, it is noted that Ss is assumed 800 gal in the RV structural analysis for the RV fragility evaluation based on the latest seismic design condition of the SFR. The CDF in all the cases below 1.0 Ss is negligibly low owing to the existing measures and the measure for improving resilience. According to Cases 1 to 3 in Figs. 8 and 9, the most dominant seismic intensity range is from 3 to 4 times of Ss, and the second and third dominant are the ranges from 2 to 3 times of Ss and larger than 4 times of Ss, respectively. In the seismic intensity range from Ss to 3 times of Ss, CDF is significantly reduced by the measure for improving resilience and there is significant difference in CDF between Cases 1 and 2. From Fig. 10, the reduction rate of CDF due to the measure for improving resilience becomes remarkable as the seismic intensity becomes small. In addition, the reduction rate of CDF in Case 3 is slightly lower than that in Case 2 thanks to consideration of maintaining the reactor coolant level by the guard vessel after the RV fatigue failure.

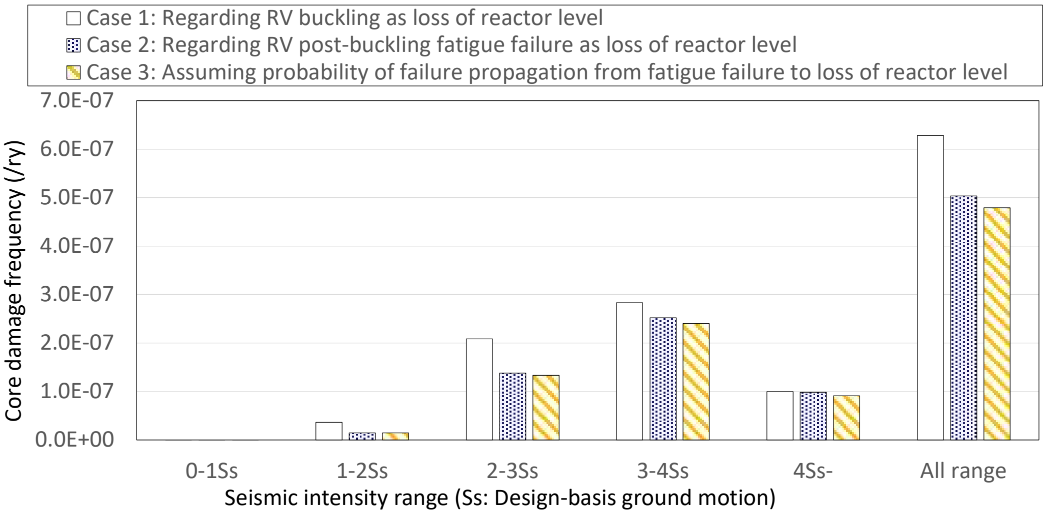


Figure 9. Comparison of CDF between Cases 1 to 3 at Each Seismic Intensity Range in the SFR

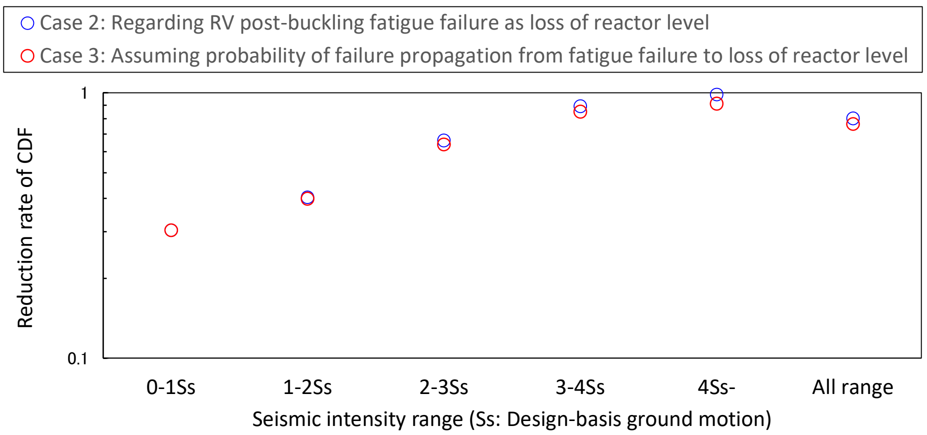


Figure 10. Comparison of Reduction Rate of CDF between Cases 2 and 3 at Each Seismic Intensity Range in the SFR

Among the various accident sequences, loss of reactor coolant level due to RV damage is a significant contributor to CDF in Case 1. As shown in Fig. 11, this sequence becomes a small contributor to CDF thanks to the measure for improving resilience: i.e., consideration of the RV post-buckling fatigue failure in Case 2. As a result, there is less difference in CDF between Cases 2 and 3 as shown in Figs. 8 and 9.

When we consider the measure for improving resilience of the RV integrity against excessive earthquake, the seismic capacity of the RV increases significantly, and CDF decreases significantly within the assumption of this study. Thus, the effectiveness of the measure for improving resilience is quantified.

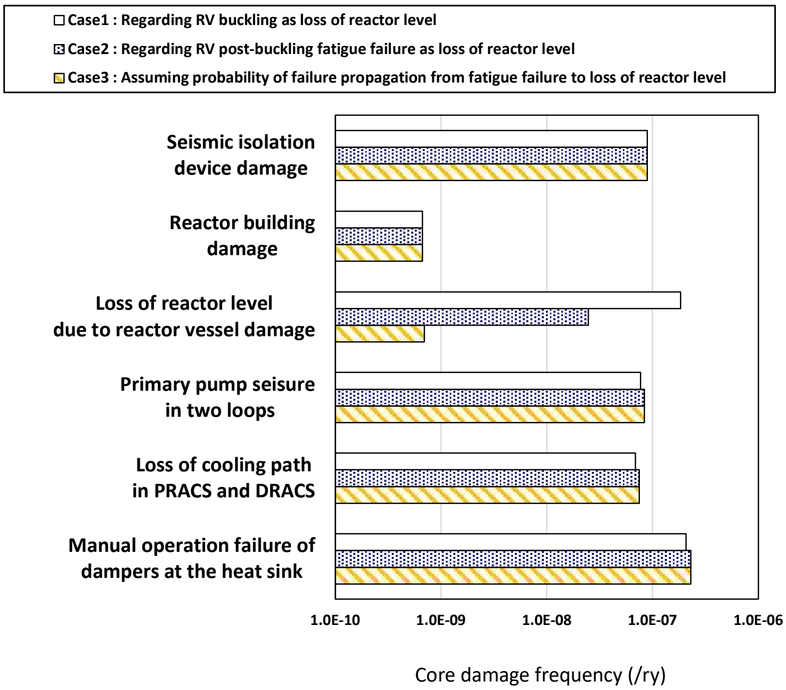


Figure 11. Comparison of Accident Sequence Frequencies between Cases 1 to 3

4. CONCLUSION

This study evaluated the effectiveness of the measure for improving resilience of nuclear structures against excessive earthquake. To evaluate its effectiveness, the reduction of seismic CDF is quantified by applying seismic PRA technology. Realistic consideration of the RV post-buckling behavior is regarded as the measure for improving resilience in this study. Two cases defined for improving resilience were compared to the base case without the measure for improving resilience. The first case assumed post-buckling fatigue failure as the realistic failure mode of the RV for the fragility evaluation. The second case assumes the median seismic capacity to loss of reactor coolant level to be slightly larger than that of fatigue failure of the RV by considering uncertainty in the behavior of failure propagation after the RV fatigue failure. Within the assumption, the measure for improving resilience were significantly effective for decreasing CDF in excessive earthquake up to several times of a design basis ground motion. The seismic PRA technology could serve to the effectiveness evaluation of the measure for improving resilience of nuclear structures against excessive earthquake. The seismic fragility evaluation method in this study could consider the fatigue damage due to not only main quake but also afterquake. To refine the RV fragility evaluation, a further study on the uncertainty in seismic response of the building including the seismic isolation system would be needed.

Acknowledgements

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