LEVEL-1 SEISMIC PROBABILISTIC RISK ASSESSMENT FOR A PWR PLANT

Authors; Keisuke Kondo*, Akihiro Ichitsuka**, Masahide Nishio*, Haruo Fujimoto*
*Japan Nuclear Energy Safety Organization (JNES)
** Japan Systems Corporation

Abstract

In Japan, revised Seismic Design Guidelines for the domestic light water reactors was published on September 19, 2006. These new guidelines have introduced the purpose to confirm that residual risk resulting from earthquake that exceeds the design limit seismic ground motion (Ss) is sufficiently small, based on the probabilistic risk assessment (PRA) method, in addition to conventional deterministic design base methodology. In response to this situation, JNES had been working to improve seismic PRA (SPRA) models for individual domestic light water reactors. In case of PWR in Japan, total of 24 plants were grouped into 11 categories to develop individual SPRA model. The new regulatory rules against the Fukushima dai-ichi nuclear power plants' severe accidents occurred on March 11, 2011, are going to be enforced in July 2013 and utilities are necessary to implement additional safety measures to avoid and mitigate severe accident occurrence due to external events such as earthquake and tsunami, by referring to the results of severe accident study including SPRA.

In this paper a SPRA model development for a domestic 3-loop PWR plant as part of the above-mentioned 11 categories is described. We paid special attention to how to categorize initiating events that are specific to seismic phenomena and how to confirm the effect of the simultaneous failure probability calculation model for the multiple components on the result of core damage frequency evaluation.

Simultaneous failure probability for multiple components has been evaluated by power multiplier method. Then tentative level-1 seismic probabilistic risk assessment (SPRA) has been performed by the developed SPSA model with seismic hazard and fragility data. The base case was evaluated under the condition with calculated fragility data and conventional power multiplier. The difference in CDF between the case of conventional power multiplier and that of power multiplier=1 (complete dependence) was estimated to be quite small as only 3 percent. However, according to the sensitivity study, it turned out that the effect of power multiplier increases as much as 20 percent if the effect of higher ECCS piping fragility is not considered.

In the future, more realistic evaluation of correlated simultaneous failure probability would be preferred to perform, however, we confirmed in this study, that effect of this failure probability is up to 20percent.

Key words; PRA, SPRA, CDF, initiating event, power multiplier, HPI, LPI, CSI, SSC

1. Introduction

In Japan, revised Seismic Design Guidelines for the domestic light water reactors was published on September 19, 2006. This new guideline has introduced the purpose to confirm that residual risk resulting from earthquake that exceeds the design limit seismic ground motion is sufficiently low level, based on the probabilistic risk assessment (PRA) method, in addition to conventional deterministic design base methodology. In response to this situation, JNES has been working to improve seismic PRA (SPRA) models for individual domestic light water reactors since before the Fukushima dai-ichi nuclear power plants' severe accidents in March 11, 2011. In case of PWR in Japan, total of 24 plants were grouped into

11 categories to develop individual SPRA model. Following the Fukushima accident, the new regulatory rules are going to be enforced in July 2013 and utilities are necessary to implement additional safety measures including back-fits to existing plants to avoid and mitigate severe accident occurrence due to external events such as earthquake and tsunami, by referring to the results of severe accident study including SPRA, as prerequisite for plant restart.

In this paper a SPRA model development for a domestic 3-loop PWR plant as part of the above-mentioned 11 categories is described. We paid special attention to how to categorize initiating events that are specific to seismic phenomena and how to confirm the effect of the simultaneous failure calculation probability model for the multiple components on the result of core damage frequency evaluation.

In the following sections, outline of the categorizing of initiating events, development of event tree (ET), fault tree (FT) models including correlated simultaneous failure calculation model using power multiplier and results of quantification and sensitivity study for simultaneous failure calculation model are described.

2. Seismic PRA model development

Target plant for SPRA is a 3-loop PWR plant whose major design features are shown in Table-1. Remarkable design features of this plant are as follows.

- Function of charging and safety injection are made mutually independent so that separate pumps for each function, i.e. three charging pumps and two high pressure injection (HPI) pumps, are installed.

- Recirculation operation of high pressure injection (HPI) system is made independent of boosting by residual heat removal (RHR) system pumps and coolers, while residual heat removal function is available for HPI solely recirculation operation mode by the coolers of the containment spray (CS) system.

Table-1 Major design features of the target plant

Major design item		unti	Notes
Thermal output	2660	MWt	
Electrical output	890	MWe	
Redundancy of safety system	2	trains	
Charging pump	3	pumps	Separated from HPI function
HPI pump	2	pumps	1/train
HPI recirculation	2	trains	Separated from RHR boosting
RHR pump	2	pumps	1/train
RHR cooler	2	coolers	1/train
LPI recirculation	2	trains	Manual switchover
Containment spray pump	2	pumps	1/train
Containment spray coolers	2	coolers	1/train
EDG	2	units	1/train
Auxiliary feedwater pump	2	pumps	motor-driven
	1	pump	turbine-driven
CCW pump	4	pumps	2/train
CCW cooler	4	coolers	2/train
Sea water pump	4	pumps	2/train

2.1 Modeling of initiating events

2.1.1 Initiating event categorization

Seismic initiating events can be categorized into two groups, one of which is so catastrophic to directly lead to core damage and the other is relatively mild to be able to expect the function of various accident mitigation systems. In the SPRA model for subject PWR plant, the initiating events consist of the former category are

- Containment building (CB) failure,
- Auxiliary building (AB) failure, and
- Reactor vessel (RV) failure.

And consist of the latter category are

- Interface system LOCA (ISLOCA),
- Large break LOCA (L-LOCA),
- Intermediate break LOCA (I-LOCA),
- Small break LOCA (S-LOCA),
- Secondary system break LOCA (SB-LOCA),
- Loss of components cooling water system (L-CCW)
- Loss of offsite power (LOOP), and
- Other transients.

These are defined referring to the internal events PRA. Special considerations in the context of SPRA are described below.

2.1.2 Special considerations for SPRA

(1) CB failure

In this initiating event, two events are included, one is the failure of CB itself that contains pressure boundary of the primary cooling system, and the other is total failure of all Steam Generators (SGs) that result in total failure of both primary coolant piping and penetrations of steam line through CB wall thus leading to simultaneous occurrence of core damage and breach of containment isolation function. Considering the severity, this initiating event is assumed to directly go to core damage.

(2) AB failure

AB contains main control room (MCR) and accident mitigation systems such as ECCS and its supporting systems so that this initiating event is also assumed to directly go to core damage.

(3) RV failure

RV failure itself and extended LOCA (E-LOCA) that exceeds the cooling capacity of ECCS are categorized in this initiating event category. E-LOCA involves multiple failures of primary coolant piping and heavy components such as SGs and reactor Coolant Pumps (RCPs), except for total failure of SGs included in CB failure.

(4) L-LOCA

L-LOCA includes failure of single primary coolant system piping whose diameter equals or more than 6 inches, single SG, single RCP or pressurizer.

(5) L-CCW

L-CCW is assumed to be triggered by single failure of one of the two safety-class main headers (A, B) and one non safety-class main header (C).

(6) Other transients

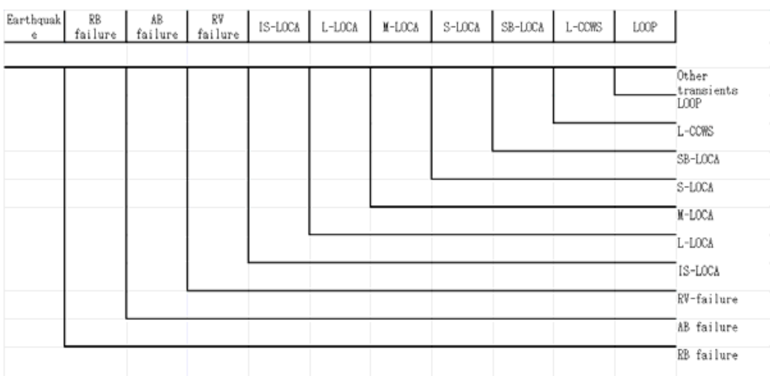
In SPSA, non-safety class power conversion system (PCS) such as feedwater and condensing system, etc. is assumed to be lost unconditionally. If no initiating event occurs by seismic failure of systems, structures and components (SSCs), other transients are assumed to occur as the remainder of the sum of the probabilities of all the seismically induced events to 1.0 calculated by the hierarchical event tree shown in the next section.

2.1.3 Calculation of initiating event probability

Occurrence probability for each initiating event is calculated using so called "hierarchical event tree[(1)(2)] " shown in Figure-1.

Occurrence probability is calculated such that sum of probability of each initiating event gets equal to 1.0 at each seismic motion level.

Figure-1 Hierarchical tree of initiating events



2.2 Modeling of event tree and fault tree

Event trees (ETs) and fault trees (FTs) are developed referring to those of internal events PRA, and in the FT analysis, we considered correlated simultaneous failure of multiple components of the same design within the same system and on the same floor by the power multiplier method[(1)(2)]. In the following subsection, at first, special considerations on ET and FT model development are described, followed by the outline of the power multiplier method.

2.2.1 ET model development

ETs for those initiating events, which do not lead directly to core damage and contained in the latter category in subsection 2.1.1 above, have been developed. Special considerations associated to ET model development are as follows.

In the initiating event of CCWS failure, isolation of affected main header (header A or B) is essential to recover CCWS function and avoid the accident progression through RCP seal LOCA etc. For this recover operation, short term and long term recovery are included in the ET headings, considering automatic isolation function of affected main header which is unique to this plant.

In the event of LOOP, recovery of the failed emergency diesel generators (EDG) due to earthquake is not modeled considering the difficulty of the local recovery actions by crew under the influence of the earthquake. Thus, once this event happen, and both diesel generators are failed due to earthquake or other reason, it is assumed to lead directly to core damage

In the event of ATWS after LOOP, insertion of control rods by gravity after unlatch of holding device of CRDM due to loss of normal power is modeled irrespective of the soundness of the safety protection system, if the deformation of reactor core is within permissible range for control rods insertion.

2.2.2 FT model development

FTs for major safety systems including both front line systems and supporting systems were developed. Special considerations for FT development are as follows.

1) Seismic failure of various boards for such as emergency electric power supply, signal processing and transmission, operator action in MCR or local, etc. are modeled.

2) Total loss of injection water source due to failure of any segment of the ECCS piping of the HPI, LPI and CSI outside containment structure was modeled as the cause for total loss of injection function from three systems.

3) Total loss of CCW cooling water source of both trains due to failure of any one of the cooler bodies of CCWS, RHR and CS system coolers was assumed resulting in loss of cooling function of CCWS.

4) Operator actions, such as feed and bleed and so on are modeled just as internal PRA.

5) Correlated simultaneous failure of components is modeled based on the power multiplier model as described in the next subsection.

2.2.3 Calculation of simultaneous failure of components

Correlated seismic simultaneous failure was considered between such components as boards, heavy components like SGs, RCPs and valves of the same design, within the same system and on the same floor.

Simultaneous failure probability of multiple components are calculated by the following equation (1),

$$P_{1,2,\cdots,N}(\alpha) = P_0(\alpha)^{n(\alpha)} \cdots\cdots \quad (1)$$

Where, $P_{1,2,\cdots,N}$: simultaneous failure probability of multiple components 1,2,・・・,N

P_0 : geometrical average of failure probabilities of N components is given by equation (2) below,

$$P_0(\alpha) = \left(\prod_{k=1}^{N} P_k(\alpha) \right)^{1/N} \cdots\cdots \quad (2)$$

n :power multiplier representing the effect of correlation, n=1 at complete dependence, n=N at complete independence between multiple components

α :seismic ground motion level (gal)

Thus power multiplier (n) is derived by the following equation (3)

$$n(\alpha) = \frac{\ln P_{1,2,\cdots N}(\alpha)}{\ln P_0(\alpha)} \cdots\cdots \quad (3)$$

Failure probability of single component k can be calculated by the following equation,

$$P_k(\alpha) = \Phi\left[\frac{\ln\left(M_{rk}(\alpha)/M_{ck}\right)}{\sqrt{\beta_{rr}^2(\alpha) + \beta_{ru}^2(\alpha) + \beta_{crk}^2 + \beta_{cuk}^2}} \right] \cdots\cdots \quad (4)$$

where,

$P_k(\alpha)$:average (point estimate) failure probability of component k

$\Phi[]$:standard log normal cumulative distribution function

$M_{rk}(\alpha)$:median response of component k

M_{ck} : median capacity of component k

$\beta_{rr}(\alpha)$:log-normal standard deviation of aleatory uncertainty of response of component k

$\beta_{ru}(\alpha)$:log-normal standard deviation of epistemic uncertainty of response of component k

β_{crk} : log-normal standard deviation of aleatory uncertainty of capacity of component k

β_{cuk} :log-normal standard deviation of epistemic uncertainty of capacity of component k

Simultaneous failure probability $P_{1,2,\cdots,N}$ is calculated by introducing the factor, Q, which represents confidence level, as equation (5), where common response and capacity distributions for all the components 1,2,・・・N, namely complete dependence, are assumed.

$$P_{1,2,\cdots,N}(\alpha) = \int_0^1 \prod_{k=1}^{N} \Phi\left[\frac{\ln\left(M_{rk}(\alpha)/M_{ck}\right) + \sqrt{\beta_{rr}^2(\alpha)+\beta_{ru}^2(\alpha)}\Phi^{-1}[Q]}{\sqrt{\beta_{crk}^2+\beta_{cuk}^2}}\right] dQ \cdots\cdots\cdots\cdots\cdots\cdots (5)$$

Equation (5) is transformed to the following equation to calculate numerically.

$$P_{1,2,\cdots,N}(\alpha) = \frac{1}{L}\sum_{i=1}^{L-1} \prod_{k=1}^{N} \Phi\left[\frac{\ln\left(M_{rk}(\alpha)/M_{ck}\right) + \sqrt{\beta_{rr}^2(\alpha)+\beta_{ru}^2(\alpha)}\Phi^{-1}[i/L]}{\sqrt{\beta_{crk}^2+\beta_{cuk}^2}}\right] \cdots\cdots\cdots\cdots\cdots\cdots (6)$$

Equation (6) is numerically calculated by rectangular integral calculus, where L is the total number of segmenting over integration interval, and i is the segment number over the integration interval.

An example of calculated power multiplier is shown in Figure-2 for the case of redundancy of 3.

While power multiplier varies depending on the mean failure probability of single component, the higher the failure probability of single component, the higher the power multiplier results. Results for redundancy of 2, 3 and 4 are summarized in Table-2. Newly calculated power multiplier fell below the conventional one's that are general input in our SPRA which are set based on the engineering judgment. In the following section we consider the effect of this difference.

Figure-2 Example of calculated power multiplier

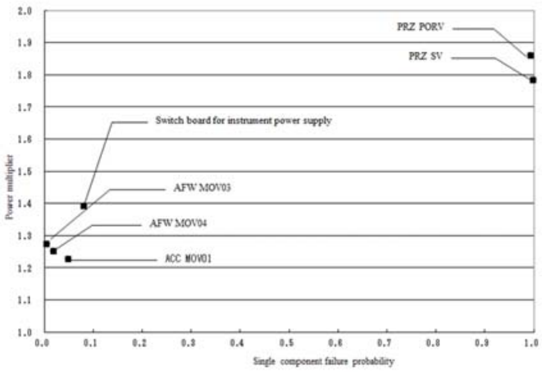


Table-2 Result of power multiplier calculation

Redundancy	2	3	4
n: old*1	1.5	2.0	2.3
n: new*2	1.1～ 1.3	1.2～ 2.0	1.4～ 1.6

*1 Conventional values used in SPRA

*2 Newly calculated values

3. Results of SPSA

SPSA was performed for the following cases.

1) Power multipliers for simultaneous multiple components failure is set to the values of n: old in Table-2.

2) Power multipliers for simultaneous multiple components failure is set to 1, assuming complete dependency.

Figure-3 shows the result of CDF calculation for two cases. It turned out that the difference of calculated CDF between two cases is as much as only three percent. Reason for this small difference is described below and in the next section.

Figure-4 shows the contribution of initiating events to the calculated total CDF for case 1) i.e. with power multiplier not equal to 1. It can be seen from Figure-4 that contribution from M-LOCA is the most dominant showing about 64% of the total CDF, followed by S-LOCA, 25% and L-LOCA, 9%. These three LOCA events account for about 98% of the total CDF.

Figure-3 Example of calculated power multiplier

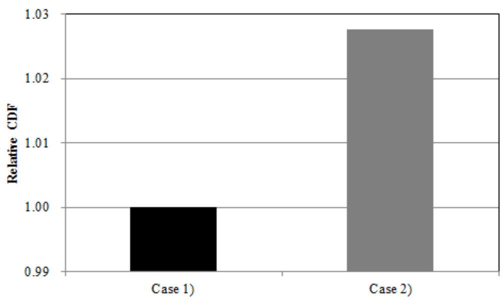
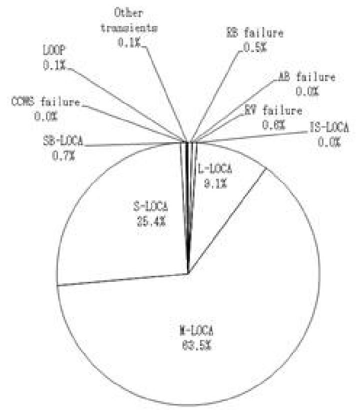


Figure-4 Contribution of initiating events



Major accident sequences for case 1) and F-V importance are shown in Table-3 and Table-4, respectively. It can be seen from Table-3 that most of the dominant accident sequences involve loss of ECCS injection i.e. HPI, LPI and CSI.

These results are related with calculated higher fragility of ECCS piping and the assumption 2) in FT development, described in 2.2.2 above. According to Table-4, LPI piping shows extremely high F-V importance.

Table-3 Dominant accident sequences

No.	Initiating event	Accident sequences	Contri-bution	Cumul-ative
1	M-LOCA	Loss of HPI + Loss of LPI + Loss of CSI	52%	52%
2	S-LOCA	Loss of HPI + Loss of LPI + Loss of CSI	23%	75%
3	M-LOCA	Loss of HPI + Loss of 2'ry cooling + Loss of CSI	11%	86%
4	L-LOCA	Loss of LPI + Loss of CSI	7%	93%
5	L-LOCA	Loss of ACC + Loss of CSI	2%	95%

Table-4 Dominant component categories of F-V importance

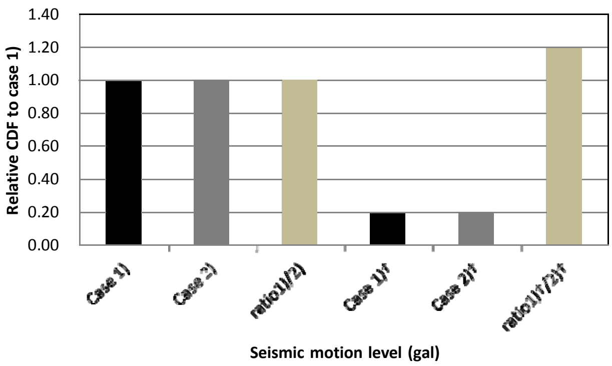
No.	Components category	F-V importance
1	LPI piping	6.7E-01
2	Prz PORV piping	2.1E-01
3	Prz spray piping	1.6E-01
4	CS piping	1.0E-01
5	Power center	5.4E-02

4. Discussion

As shown in the Table-2 power multiplier for correlated simultaneous failure probability of components by equation (1) went below the conventional input values. As shown in Figure-2, the difference in CDF is quite small between conventional n-values and n=1 (complete dependency), thus for the case of target plant of SPRA, the effect of change of n-values seems to be almost negligible, this is because CDF is dominated by the failure of piping of ECCS whose failure probabilities are calculated significantly higher, especially for the case of LPI.

In order to examine this effect, we performed sensitivity analysis by enhancing the seismic capacity of the ECCS piping (i.e. HPI, LPI and CSI) by ten-fold to hypothetically remove the effect of piping failure and then re-evaluating the case 1) and case 2) of section 3 above. Results are shown in Figure-5. As shown in this figure, after enhancing capacity of ECCS piping, CDF for both case 1) and case 2) has reduced to about one-fifth, shown as 1)†and 2)†in the figure, respectively, however, the ratio of CDF has increased from 1.03 to 1.20. This indicates that importance of power multiplier increases after removing the effect of fragility of piping to which power multipliers are not applied.

Figure-5 Sensitivity study by removing piping fragility of ECCS



5. Conclusions

SPSA model for a domestic PWR plant has been established and tentative quantification was successfully performed, and power multiplier to evaluate correlated simultaneous failure probabilities for multiple components are calculated. In the base case where calculated fragility data and conventional power multiplier was applied, the difference in CDF between the case of conventional power multiplier (case 1)) and that of power multiplier=1 (case 2)) was calculated to be quite small as only 3 percent. However, because power multipliers were calculated lower than the conventional values, we performed a sensitivity study to see if this reduction of multipliers becomes important, for the case fragility of ECCS piping, to which power multipliers are not applied in the current SPSA model, is small enough. It turned out that the effect of power multiplier increases up to 20 percent if the effect of ECCS piping fragility is removed.

In the future, more realistic evaluation of correlated simultaneous failure probability would be preferred to perform, however we confirmed in this study, that effect of this failure probability is up to 20 percent.

6. References

(1) "Implementation Standard Concerning the Seismic-Induced Probabilistic Risk Assessment on Nuclear Power Plant: 2007," AESJ-SC-P0006:2007.

(2) "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150,"NUREG/CR-4840, SAND88-3102, Sandia National Laboratories, November, 1990.