Updating Seismic PSA for OPR-1000

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1. Introduction

Since Fukushima accident, the full scope PSA (Probabilistic Safety Assessment) for OPR-1000 (Optimized Power Reactor) reactor was performed for all types of risks from internal and external events and for all plant operational modes. As a part of this project, the seismic PSA (SPSA) for OPR-1000 reactor has been also updated based on new SPAS methodology and key elements of SPSA updating results. The seismic hazard evaluation was re-performed to develop the specific frequencies for the pilot OPR-1000 reactor site. The fragility evaluation was re-performed to estimate the conditional failure probability of SSCs (Structure, System & Component) on the seismic equipment list. To develop the seismic induced accident sequence model, initiating events from the internal events were reviewed to determine the appropriate response to an earthquake. Through this process, eleven seismic induced initiating events were identified for the five intensity earthquake levels. The internal fault trees for SSCs were modified to apply the seismic induced failure probability for each intensity levels. The seismic HRA was also modified and applied the seismic accident sequence analysis process. The quantification is performed using AIMS and PRASSE code.

2. Seismic Hazard Analysis

The seismic hazard analysis re-evaluated to develop the frequencies of occurrence of different levels of earthquake ground motion (PGA: peak ground acceleration) at a pilot OPR-1000 reactor site. For the improvement of the seismic input motion, uncertainty reduction of a probabilistic seismic hazard analysis (PSHA) and methodology of constructing response spectra considering probabilistic site amplification effect was studied. Input data of a PSHA was analyzed, and used for sensitivity analysis to see the effect of each parameter on the uncertainty of seismic hazard. So, one of the seismic hazard curves developed for pilot OPR-1000 reactor site as shown in Figure 1. The probabilistic method to evaluate the uniform hazard spectra of soil nuclear power plant sites corresponding to that of bedrock site was developed for considering local site effect. For the evaluation of an input ground motion caused by a causative fault, the response spectra considering probabilistic fault parameters were estimated by using the finite fault model.

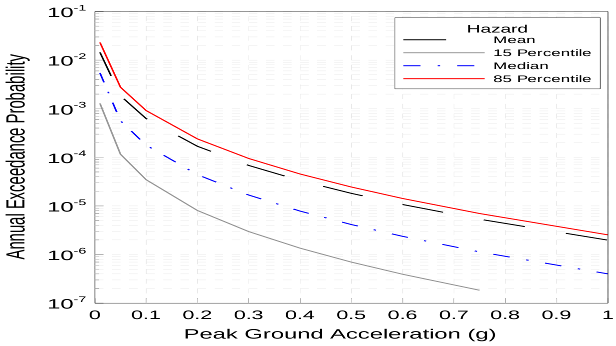

Fig. 1. Seismic Hazard Curve

3. Fragility Analysis

The Seismic Fragility analysis was performed to estimate the conditional probability of seismic induces failure for SSCs for a pilot OPR-1000. The fragility of SSCs, representing the seismic capacities and the associated uncertainties, are the basic input for SPSA model. Before the fragility analysis, the seismic equipment list (SEL) was developed. The list included the equipment and system required to provide protection and needed to mitigate for the seismically induced initiating events and the structure that house them. Some non-seismically qualified systems were screened out based on conservative assumption. The example fragility and FMEA results of SELs respectively for OPR-1000 is summarized in Table 1.

Table 1. Fragility Analysis Results

SEL	Fragility				FMEA
	A_m	β_R	β_U	HCLPF	
Rx Building	3.8g	0.26	0.35	1.4	Initiating Event
Aux Building	2.0g	0.32	0.37	0.64	Initiating Event
Intake Structure	2.74	0.34	0.36	0.87	Initiating Event
RCP	0.92	0.2	0.21	0.47	Initiating Event
HPSI Pump	1.97	0.26	0.28	1.35	Mitigation Function
C1E Switchgear	2.92	0.25	0.43	0.95	Mitigation Failure
RCS Piping	13.69	0.27	0.3	5.4	Initiating Event
Other Pipe	2.27	0.29	0.4	0.74	Mitigation Function

The SEL was developed based on the internal PSA results, former SPRA, site walkdown and expert judgment. After the fragility analysis for the SEL, the seismically FMEA(Failure Mode and Effect Analysis) was also performed to identify the effect (cause seismic induced initiating events or affect mitigation function) of the seismic induced failure of SSCs.

4. Accident Sequence Analysis for Seismic Event

The seismic induced accident sequence model accounts for the unique failure modes caused by seismically induced ground motion. The hierarchy logic tree, was developed to identify the various seismically induced initiating events (IE), is presented in Fig 2.

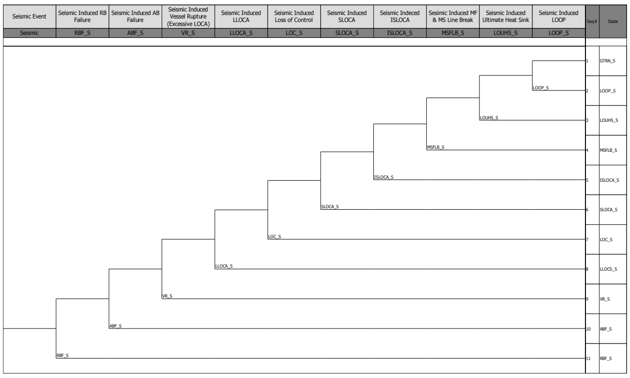

Fig. 2 Seismic Hierarchy Logic Tree

The hierarchy in this logic tree is defined from left to right. If a selected IE occurs (lower branch), the occurrence of IE further to the light on the hierarchy tree is of no significance with respect the plant response. Given the occurrence of seismic event, the hierarchy logic tree is developed such that the seismically induced IE with the most challenge to the plant safety system is considered in the following order: essential structure collapse, excessive LOCA(including reactor vessel rupture), loss of coolant accident, loss of plant control, loss of off-site power and transient. The eleven IEs are identified and the six IEs could cause direct core damage without any further analysis by conservative assumption. The five specific event trees are developed for further analysis.

The frequencies of each IEs results according to the five intensity earthquake levels were estimated by PRASE code, the SSCs could cause each IEs and the effect are summarized in Table 2.

Table 2. Seismic induced IE Frequency

IE	SSCs causing IE	IE Frequency for intensity levels		Remarks
		0.6~0.8g	0.8~1.0g	
RBF_S	Rx Building	2.51E-10	8.84E-10	Direct CD
ABF_S	Aux. Building	9.37E-08	1.05E-07	Direct CD
VR_S	Rx Vessel, Rx Internal, RCS Pipe, RCP, S/G, SIS Pipe, RX Shield Wall	8.09E-07	8.62E-07	Direct CD
LLOCA_S	PZR	3.26E-12	1.85E-11	Specific ET
LOC_S	MCR Board, PPS/ESFAS/PCS Sys.	2.12E-07	1.53E-07	Direct CD
SLOCA_S	PZR Safety V/V, CVCS Pipe, Instrument , REG Hx	2.14E-07	1.49E-07	Specific ET
ISLOCA_S	CVCS Pipe & Iso. Vv	4.92E-14	1.19E-11	Direct CD
MSMFLB_S	MF& MS Pipe	2.38E-07	1.49E-07	Specific ET
LOUHS_S	CCW&ESW Pipe/Pump, CCW Surge Tank, Intake	1.72E-12	5.26E-12	Direct CD
LOOP_S	Switchyard Insulator	2.36E-06	1.64E-07	Specific ET
GTRN_S	Occur above SSE	9.80E-09	3.62E-11	Specific ET

4.1 Specific Event Tree for each Initiating Event

The five specific seismic event trees for each IEs according to five intensity earthquake levels were developed to analyze the accident sequence based on internal PSA accident sequence model. Initiating events from the internal PSA were reviewed to determine the appropriate response to an earthquake. This analysis assumed that plant response to an initiating event from the earthquake would be similar to that internal event accident sequence. But, some recovery actions such as off-site recovery after SBO (Station Black-out) accident and the non-safety related system model such as main feed water system or condenser dump system were excluded in seismic event tree based on conservative assumption. The seismic induced loss of off-site power event tree is presented for reference in Fig 3.

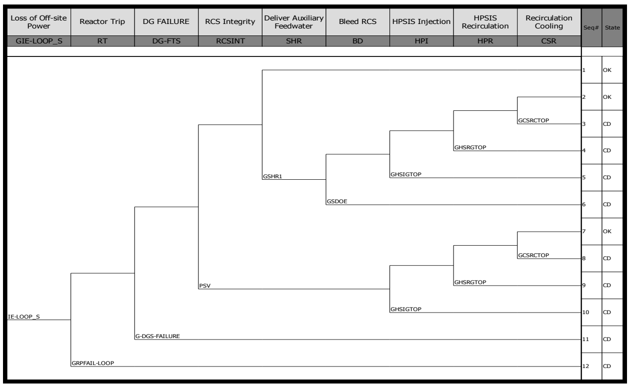

Fig. 3 Seismic induced LOOP Even Tree

4.2 Fault Tree for Seismic Event

The seismic fault tree was developed to quantify the accident sequence model also. There are two main modifications to internal PSA fault trees which must be made to apply them to the seismic PSA. The first one is the conditional failure probability for seismic induced failure mode for each SSCs were added to system fault tree for five discrete levels by the intensity of earthquake respectively. The second one is a process to evaluate the fragility correlation (simultaneous seismic induced failure of redundant safety system) for SSCs in seismic fault trees. For the that purpose, the mapping table was developed in which there are the fragility do each SSCs, a basic event name for seismic induced failure probability and the basic name for simultaneous failure probability for redundant SSCs. The correlation for redundant SSCs on the same floor slab assumed same failure probability based on conservative assumption. The seismic module of AIMS code and PRASSE code calculated seismic induced failure probability for each SSC according to the five intensity earthquake levels and added automatically to internal PSA fault trees. The sample mapping table and modified fault for seismic PSA is presented in Table 3 and Fig 4.

Table. 3 Mapping Table for SPSA fault tree

SSC	Basic Event Name(Internal)	Fragility				Seismic induced Basic Event Name	Simulateneous Failure Basic Event Name
		Am	Br	Bu	HCLPF		
LPSI Pump 1	LSMPRLPSI1	2.46	0.24	0.37	0.9	SS_LSMPRLPSI1	ALL-SS_LSMPRLPSI1
LPSI Pump 2	LSMPRLPSI2	2.46	0.24	0.37	0.9	SS_LSMPRLPSI2	
HPSI Pump 1	HSMPR0001A	1.41	0.24	0.28	0.6	SF_HSMPR0001A	ALL-SS_HSMPRLPS1A
HPSI Pump 2	HSMPR0002B	1.41	0.24	0.28	0.6	SF_HSMPR0002B	
SI TANK 1A	STTKBLPF1A	0.58	0.23	0.49	0.18	SS_STTKBLPF1A	ALL-SS_STTKBLPF1A
SI TANK 1B	STTKBLPF1B	0.58	0.23	0.49	0.18	SS_STTKBLPF1B	
SI TANK 1C	STTKBLPF2A	0.58	0.23	0.49	0.18	SS_STTKBLPF2A	
SI TANK 1D	STTKBLPF2B	0.58	0.23	0.49	0.18	SS_STTKBLPF2B	

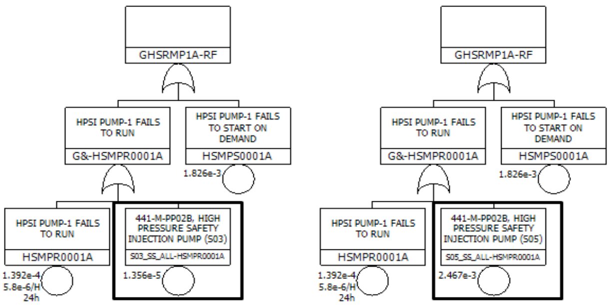

Fig. 4 Modified Fault Tree for SPSA fault tree

The seismic human reliability analysis (HRA) was also modified and applied the seismic accident sequence analysis process.

5. Quantification

To quantify the core damage frequency (CDF) during an earthquake, five one-top quantification models was developed that fault trees and event trees were linked according the intensity of earthquake interval. The CDF induced seismic initiating events estimated for the five discrete levels by the intensity of earthquake in OPR-100 reactor by the AIMS & PRASSE PSA code. Seismically induced CDF is given by the integration of the annual probability of occurrence of the five earthquake intensity level. The CDF results caused by seismic event based on three quantification methods in AIMS are summarized in Table 5. As shown in Table 5, the basic classical quantification method (rare event approximation) is not adoptable for SPAS because of seismic induced high failure probability of SSCs.

Table 5. CDF results for Each Methods

Seismic Intensity Interval		CDF(/year)		
		(REA)	(MCUB)	(FTeMC)
Interval 1	0.2~0.4g	1.71E-06	1.70E-06	1.66E-06
Interval 2	0.4~0.6g	6.05E-06	5.81E-06	4.93E-06
Interval 3	0.6~0.8g	9.08E-06	8.03E-06	4.76E-06
Interval 4	08~1.0g	1.27E-05	9.55E-06	2.11E-06
Interval 5	1.0~1.2g	2.21E-05	1.06E-05	9.65E-07

The 67% of CDF is occurred in intensity interval 2 and 3 as shown in Fig 5.

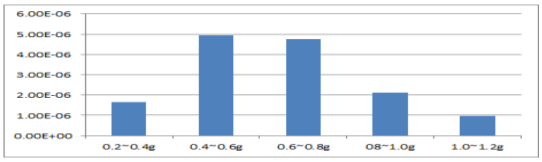

Fig 5. CDF results for Intensity Interval

The CDF evaluation results categorized by initiating events and each earthquake intensity intervals are presented in Table 6.

Table 6. CDF Results for each Intensity Intervals

IE	CDF for intensity interval		
	0.4~0.6g	0.6~0.8g	08~1.0g
%SE_LLOCA_S	negligible	negligible	1.83E-11
%SE_SLOCA_S	9.45E-08	1.75E-07	7.65E-08
%SE_ISLOCA_S	negligible	negligible	negligible
%SE_VR_S	2.39E-07	8.13E-07	8.70E-07
%SE_MFSLB_S	1.02E-08	9.77E-08	1.34E-07
%SE_GTRN_S	6.03E-08	3.08E-09	3.02E-11
%SE_LOUHS_S	3.13E-06	2.40E-06	6.35E-07
%SE_LOOP_S	1.24E-06	9.64E-07	1.29E-07
%SE_RBF_S	1.29E-11	2.51E-10	8.84E-10
%SE_ABF_S	3.98E-08	9.37E-08	1.05E-07
%SE_LOC_S	1.22E-07	2.13E-07	1.55E-07

The main contributors to the seismic induced CDF are direct core damage caused by loss of ultimate heat sink initiating event and all loss of AC power event (SBO) accompanied by seismic induced emergency diesel generator failure following seismic induced LOOP.

6. Conclusions

The seismic PSA model for OPR-1000 reactor was updated based on recent research project results as follows :

- updated hazard analysis for OPR-1000 site
- updated fragility analysis results for SSCs

- test new seismic quantification code
- updated HRA analysis
- updated based on internal PSA result

Seismically induced CDF is given by the integration of the annual probability of occurrence of the five earthquake intensity level.

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