Evaluation of a Containment Failure Frequency Considering Mitigation Accident Managements for a Japanese PWR Plant *

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ABSTRACT

Institute of Nuclear Safety of NUPEC is carrying out a program of developing methodology of Probabilistic Safety Assessment (PSA) and severe accident analysis. For a Japanese 4-loop PWR plant with a pre-stressed concrete containment, containment failure frequency evaluation methods by point-estimate and uncertainty estimate were established for internal initiating events during full power operation assuming mitigation accident management.

In the point-estimate evaluation, core damage sequences were categorized for every plant damage state (PDS) using the results of Level 1 PSA that evaluated core damage frequencies. Sequences both without and with mitigation AM countermeasures were analyzed with the MELCOR code: (1) the natural convection cooling by containment cooling units for normal operation, (2) fire water injection into the containment, (3) the forced depressurization of primary system by pressurizer PORVs, (4) the restoration of containment spray system, and (5) water injection into the primary system by charging pumps. A containment event tree including the AM measures was made, and the simplified reliability evaluation on equipment failure and human factor at AM operation was executed. Severe accident sequence representing each PDS was analyzed with MELCOR code in order to quantify the branch probability of a containment event tree. The quantification of containment event tree was done for each plant damage state. Consequently, in the case of AMs included, the total containment failure frequency was obtained to be 1.1×10^{-7} / reactor year comparing 2.2×10^{-7} / reactor year with without AMs. A dominant sequence of containment failure when the AM plan is implemented is an interface system LOCA sequence that a pipe of the residual heat removal system breaks loaded primary system pressure.

I. INTRODUCTION

The severe accident analysis method at NUPEC/INS has been furbished in according with the PSA methodology in order to provide the deterministic safety review with the supplemental information. With a primary objective of estimating

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containment performance, the level 2 PSA [1] by point-estimate and uncertainty estimate was executed for a typical Japanese 1,100 MWe PWR plant ; the procedures were consisted of plant familiarization, accident sequence grouping and definition of PDSs, construction of CETs, accident progression analysis with the MELCOR code [2], and quantification of CETs for obtaining the level 2 PSA results, reflecting the core damage frequency results obtained from the level 1 PSA activities for the same plant that had been done previously.

II. CONTAINMENT FAILURE FREQUENCY EVALUATION BY POINT ESTIMATE METHOD

The AM mitigation [3] measures were considered (1) the natural convection cooling by containment cooling units for normal operation, (2) fire water injection into the containment, (3) the forced depressurization of primary system by pressurizer PORVs, (4) the restoration of containment spray system, and (5) water injection into the primary system by charging pumps. In addition to those countermeasures [3], cooling down and re-circulation as the SGTR measurement was considered in the present study. These AMs are shown in Figure 1.

A. Natural Convection cooling by Containment Cooling Units

The heat removal from containment atmosphere can be achieved through natural convection by supplying water to the chillier of containment cooling system in the case of the failure of emergency containment cooling system, etc. The analysis of the failure of emergency containment cooling system in the event of large LOCA, that causes most severe effects, shows that the increase in containment pressure is suppressed by starting the cooling water supply to the chiller, and the pressure is gradually reduced.

B. Fire Water Injection into the Containment

This measure injects the water from raw water tank, etc. into the reactor cavity in case the water injection to the core is not sufficient and the core melt and subsequent vessel bottom melt-through should occur. Submergence of the melted core in the reactor cavity region with water mitigates the erosion of concrete while preventing the overheating of containment penetration by keeping the containment atmosphere at saturated condition.

C. Forced Depressurization of Primary System

In case of the simultaneous failure of high pressure injection system and heat removal function via secondary loop, the integrity of core is threatened while a high reactor vessel pressure. The dispersion of melted debris can be suppressed by reducing the reactor pressure through the opening of pressurizer relief valves, in case the melt-through of reactor vessel should occur. This measure prevents the direct heating of containment atmosphere and the direct contact of the melted debris to the containment vessel. Additionally, the measure provides the chance to utilize low pressure injection system, which increases the possibility of suppressing and/or delaying the core degradation and the reactor vessel melt-through.

D. Cooling down and Re-circulation

The cooling down and re-circulation measure was adopted to minimize the leakage of primary coolant to the exterior of containment vessel when operator failed the leakage isolation at SGTR, etc. First, the reactor core cooling is established via secondary loops through releasing main steam while keeping water injection to the reactor. Secondly, the reactor pressure is reduced by opening pressurizer relief valves, etc. to suppress the leakage. Finally, long-term leakage reduction and heat removal are secured by putting residual heat removal system into service.

III. CONTAINMENT FAILURE FREQUENCY EVALUATION BY POINT ESTIMATE METHOD

A. Outline of the Reference PWR Plant

The reference plant is a 4-loop PWR plant with 1,100 Mwe which is the typical PWR plant in Japan. The design characteristics are as follows:

1) two-train safety injection system and three auxiliary feed water pumps,
2) pumps of the containment spray system and the safety injection system take suction from the in-containment, re-circulation sump, and the water resource is necessary to switch from the refueling water storage pit,
3) pre-stressed concrete containment with design pressure of 439kPa and volume of 73,700 m3.

B. Plant Damage States (PDSs)

The accident sequences obtained from the level 1 PSA were grouped into thirteen (16) PDSs by considering the similarities of accident progression, and containment response. Figure 2 shows the ratio of each PDS and the total core damage frequency with the limited preventive AMs. A fraction of about 48% of total core damage frequency is caused by the re-circulation failure after a LOCA.

C. Containment Event Trees (CETs) and Quantification

The CETs with the AMs were developed to trace the interdependent physico-chemical processes influencing severe accident progression in the reactor system and the containment as shown in Figure 3 as an example of late accident phase. A large CET approach was selected to present accident progression with thirty three (33) top events. The end points of CETs that were relevant to the integrity and retention capability of the containment were attributed to containment failure modes. A dominant accident sequences of each PDS were analyzed for obtaining quantify the containment event tree. The quantification of branching probabilities in the CETs was performed using analytical results of the MELCOR code, engineering judgment, and unavailability evaluation regarding to AMs.

D. Point-Estimate of Containment Failure Frequency

Point-estimate values of the containment failure frequency for each containment failure mode considered with and without AMs were obtained as shown in Figure 4 and Figure 5. The major results are as follows.

1) In the case of AMs included, the total containment failure frequency was estimated to be 1.1×10^{-7} / reactor year comparing with that the case without AMs was estimated to be 2.2×10^{-7} / reactor year.
2) The containment failure frequencies with overpressure and concrete penetration during core/concrete interaction were reduced by mitigative AMs. However, the containment failure frequency with ex-vessel steam explosion increased by AMs such as the fire water injection into the containment.
3) A dominant sequence of containment failure when the AM plan is implemented is an interface system LOCA sequence that amounts to about 81% of total containment failure frequency.
4) From the viewpoint of large fission product release in the case of AMs included, the containment bypass mode including core damage accidents resulted from initiating events of interfacing system LOCA and SGTR is the most important.

IV. CONTAINMENT FAILURE FREQUENCY EVALUATION BY UNCERTAINTY ESTIMATION METHOD

A. Uncertainty Estimation Method

The Latin Hypercube sampling (LHS) sampling method was applied to perform uncertainty analysis with a sample size of 200 to each probability distribution function related to uncertainty parameters in the CET. The uncertainties in the CETs were combined and propagated by the PREP/SPOP code [4] in which credit is given for statistically correlated parameters.

B. CET Headings Considered in Uncertainty and Quantification

In the uncertainty evaluation, the uncertainty probability distributions were examined based on level 2 PSA that carried out for the Zion plant [5] in NUREG-1150 [6] . The probability distribution functions that were used in the present study were generated for seven parameters of the severe accident phenomena: (1) over-temperature failure of primary system, (2) over-temperature failure of steam generator tube, (3) oxidization rate of zirconium in reactor vessel, (4) discharge rate of melted core debris at reactor vessel failure, (5) failure mode of reactor vessel, (6) pressure increase of containment atmosphere at reactor vessel failure, and (7) containment failure probability. In addition, the uncertainty probability distributions of in-vessel steam explosion and debris coolability in the reactor cavity were established in the present study as shown in Figure 6 and 7 by ROAAM [7] method. These probability distributions were taken into account in the containment event tree, the probabilistic propagation calculation was performed, and the uncertainty of containment failure frequency was estimated while was not included the uncertainty of level 1PSA.

C. Results of Uncertainty Analysis

Results of uncertainty analysis including mitigation AMs are shown in Figure 8 with 5%, mean, and 95% values of containment failure frequencies. Containment failure modes that lead to late containment failure have not large uncertainties, such as containment bypass, and late over-pressurization excluding base-mat melt-through. Early containment failure modes of in-vessel and ex-vessel steam explosion, and hydrogen detonation failure that have large uncertainties do not contribute much to the total containment failure frequency. Uncertainty bound of base-mat melt-through method was estimated to be about one and half digit from 6.E-10/RY to 3.E-08/RY, and mean value of the bound was denoted to be 8.E-09/RY used ROAAM method.

V. CONCLUSIONS

The present study addressed to establish the containment failure frequency evaluation method by the point-estimate and uncertainty evaluation method including for internal initiating events during full power operation for a Japanese 4-loop PWR plant with pre-stressed concrete containment. In the present study, the calculated result showed that containment failure frequency in the case without AMs was reduced about 50% by implementation of AMs. A dominant sequence that leads to the containment failure with AMs was interfacing system LOCA sequence that a pipe of the residual heat removal system breaks loaded primary system pressure. The results of the present study indicated the containment failure frequency band of base-mat melt through has large uncertainty, but does contribute on a limited total containment failure frequency.

In the current program at NUPEC/INS, the evaluation of containment failure frequencies are being carrying out taking into account the effectiveness of accident management measures after core damage on containment failure frequencies and mitigation of accident progression based the operating manual obtained by utilities.

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REFERENCES

1. INS/NUPEC, "Level 2 PSA for 1100MWe, 4-Loop PWR Plant with a Large Dry Containment," (in Japanese) INS/M99-08 (2000)
2. R. M. Summers, et al., "MELCOR Computer Code Manuals," NUREG/CR-6119, Vol.1 & Vol.2, (1995)
3. M. Sobajima, "Current Status of the Implementation Plan of Accident Management to Nuclear Power Plants," Journal of the Atomic Energy Society of Japan. Vol. 37 No. 5, 1995
4. T. Homma and A. Saltelli, " LISA Package User Guide Part I, PREP(Statistical PRE Processer), Preparation of Input Sample for Monte Carlo Simulations Program Description and User Guide," EUR 13922 EN(1992)
5. M. B. Sattison, et al., "Analysis of Core Damage Frequency: Zion, Unit 1 Internal Events," NUREG/CR-4550, Vol. 7, Rev. 1 (1990)
6. USNRC, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," NUREG-1150, Final Summary Report (1990)
7. T. G. Theofanous, et. al., "An Assessment of Steam-Explosion-Induced Containment Failure," NURG/CR-5030, February,

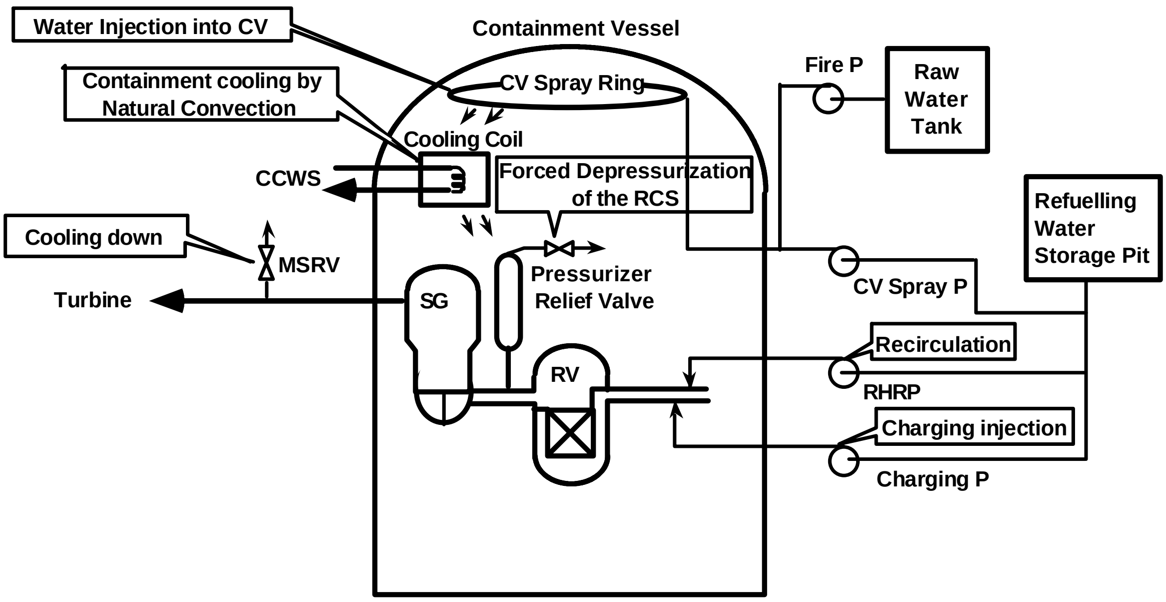
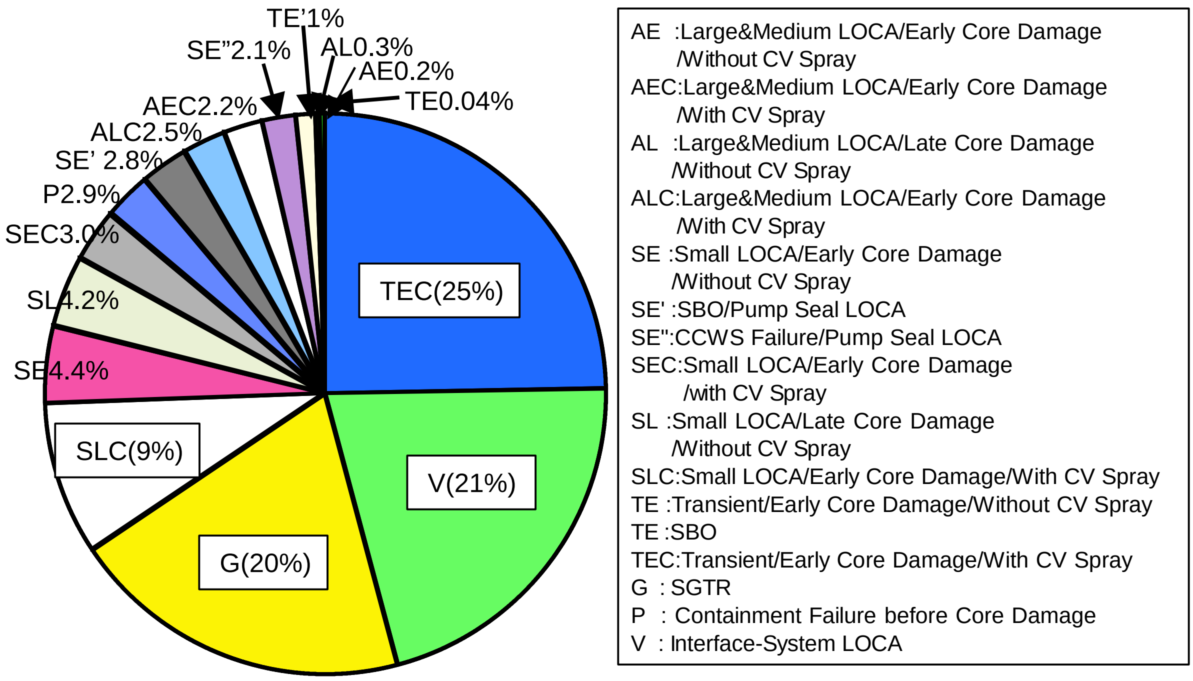


Figure 1 Accident Management Measures



Core Damage Frequency 4.3E-07/RY

Figure 2 Core Damage Frequencies (Non AM)

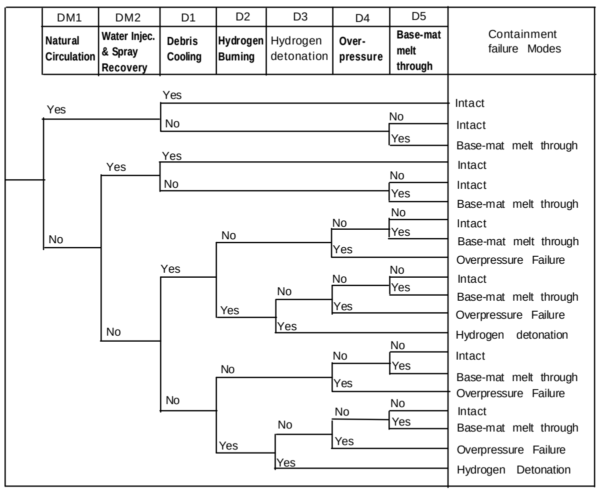
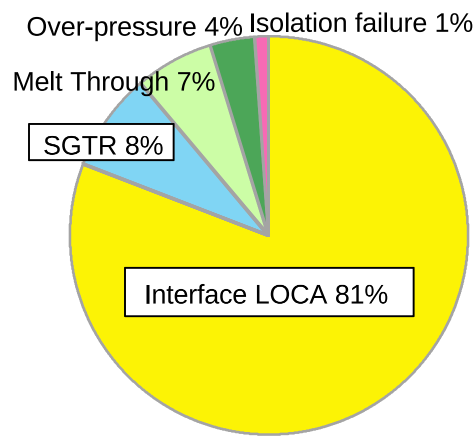


Figure 3 Containment Event Tree for a long term period after the reactor vessel failure



Failure Frequency 1.1E-07/RY

Figure 5 Containment Failure Fraction (With AM)

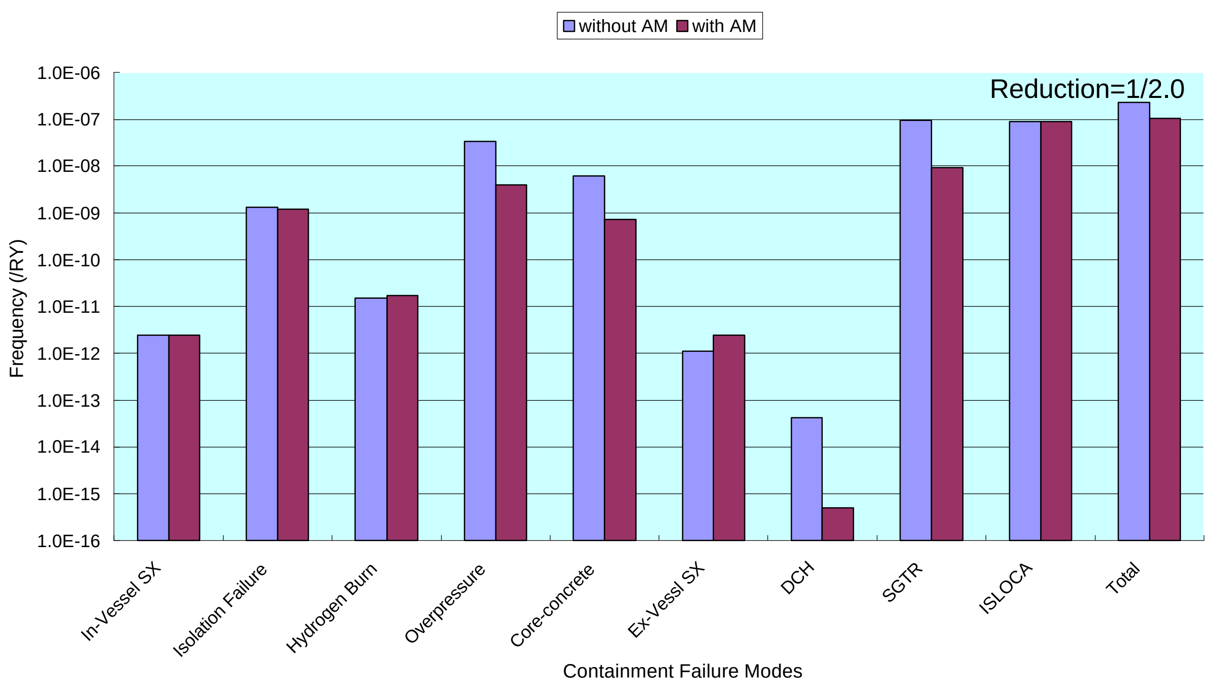


Figure 4 Containment Failure Frequencies (Point Estimate)

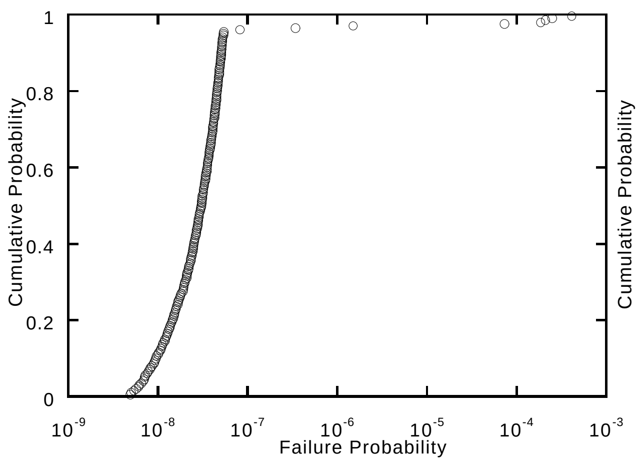


Figure 6 Probability Distribution of In-Vessel Steam Explosion

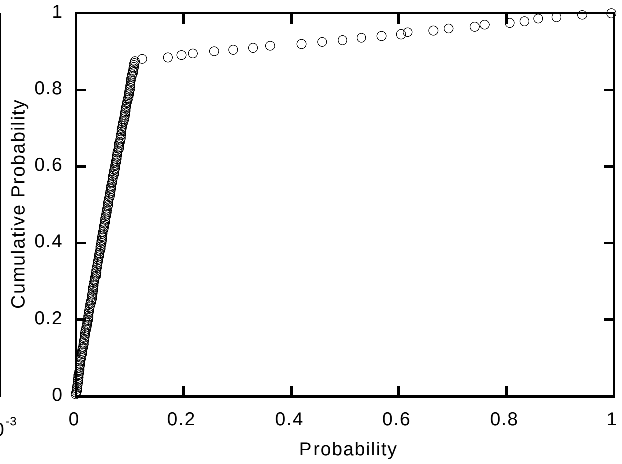


Figure 7 Probability Distribution of Debris Cooling (Low Pressure, Sub-cool water & Water injection)

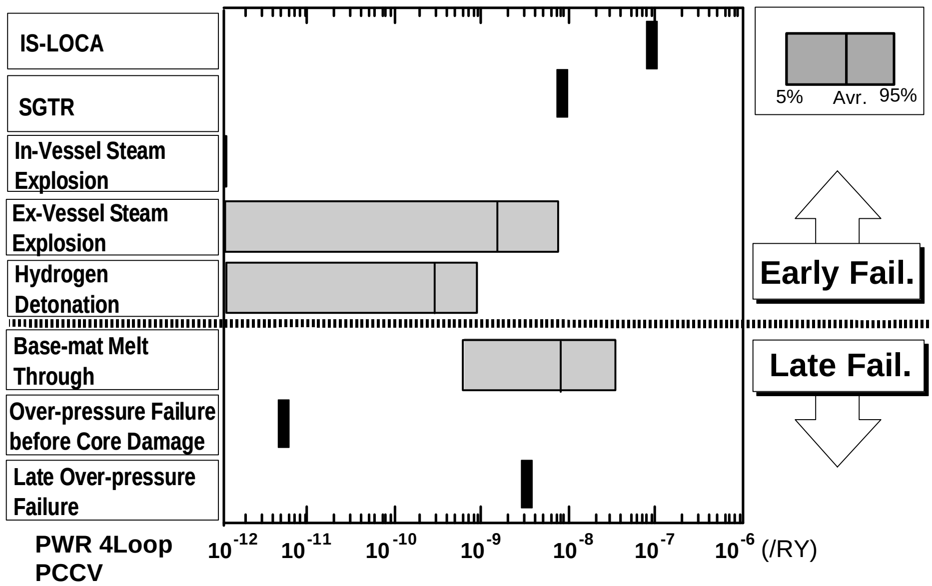


Figure 8 Uncertainty Band of Containment failure frequency (With AM)