Proceedings of the Korean Nuclear Society Autumn Meeting
Taejon, Korea, October, 2000

Efficiency of Severe Accident Management Options for In-Vessel Core Melt Retention in a Large Power Reactor

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Abstract

A simple probabilistic analysis method has been proposed to estimate efficiency of external reactor vessel cooling and late in-vessel injection as severe accident management (SAM) actions for retention of molten core inside the reactor vessel (In-Vessel Retention, IVR) under hypothetical core-melting severe accidents. This method defines fractional core damage frequency where molten core can be retained inside the vessel with severe accident management actions of primary system depressurization, external reactor vessel cooling and/or late in-vessel injection. The method applied to the Korean Next Generation Reactor (KNGR), a large evolutionary pressurized water reactor shows that inclusion of external reactor vessel cooling strategy significantly increases the frequency of in-vessel retention even with conservative assumptions of system availability and conditional probabilities.

1. Introduction

Since the Three Mile Island (TMI) nuclear power plant incident, there have been extensive research activities to develop accident management strategies to mitigate the consequences from core-melting severe accidents.

Considering very low occurrence frequency of a severe accident, prevention through more careful maintenance and operator training is pursued in existing nuclear power plants. In contrast, for the advanced plants, mitigation of severe accident is reflected in the design itself. The severe accident phenomena considered are hydrogen combustion, direct containment heating, steam explosion, and core concrete interaction that may threaten the containment integrity.

Phenomenological issues such as hydrogen combustion, direct containment heating, and steam explosion are being closed through research activities and plant-specific design features. As for core melt coolability, it has been successfully shown for small and medium power reactors such as AP600 and Loviisa that external reactor vessel cooling (ERVC) can retain corium inside the reactor vessel (In-Vessel Retention, IVR) and reactor vessel failure is physically unreasonable. However, core melt coolability inside and outside of reactor vessel is still one of the remaining phenomenological issues for large power reactors

For large pressurized water reactors, without an issue resolution by innovative design features, a consensus is that ERVC-only strategy has an uncertainty in showing reactor vessel failure is physically unreasonable and thus it would be better approach to credit late in-vessel injection (IVI) complementing ERVC. Thus it can be stated that SAM activities will be a more important factor for a large power reactor at least with respect to core melt coolability.

In this paper, therefore, in order to estimate the efficiency of external reactor vessel cooling as a severe accident management (SAM) action for in-vessel retention of a core melt, a simple probabilistic analysis method has been proposed and applied to the Korean Next Generation Reactor (KNGR). The method resulted in fractional core damage frequency as a measure of the efficiency for candidate three SAM options for in-vessel retention using conditional success probability as sensitivity parameters.

2. SAM Actions Relevant to In-Vessel Core Melt Retention

For a large pressurized water reactor, in order to cool the corium inside the reactor vessel after core damage, primary system depressurization is essential to prevent reactor vessel creep rupture, and subsequently, we can consider following three SAM options:

(1) External Reactor Vessel Cooling only

(2) Late In-Vessel Injection only

(3) Both External Reactor Vessel Cooling and Late In-Vessel Injection

For primary depressurization, pressurizer relief valves would be needed. For external reactor vessel cooling, the relevant system would be containment spray system, or any other particular system dependent on the plant design. For late in-vessel injection, the relevant system would be safety injection system (high and low pressure safety injection).

3. Description of the Method

We already have an established professional analysis tool, level II probabilistic safety assessment (PSA) that has been generally used to estimate containment integrity using a measure of large release frequency (LRF) to severe accident mitigation design features and SAM actions. The level II PSA, however, does not effectively combine system operation before core damage and after core damage.

The measure of the method described in this paper is the efficiency of specific SAM actions for specific severe accident mitigation purpose, what will be called *fractional CDF, FCDF*) and defined in this section.

In this paper, equations to estimate the *FCDF* focused on the SAM actions relevant to in-vessel core melt retention (IVR) for a large powered pressurized water reactor will be derived. However it could be applicable to any other severe accident phenomenology such as steam explosion, etc.

The *FCDF* of IVR is defined as following:

$$FCDF(IVR) = \frac{CDF(IVR)}{CDF(total)} \tag{1}$$

where *CDF(IVR)* means sum of *CDFs* where the molten core material is cooled inside the vessel and CDF(total) is the total domain CDF.

For the *CDF(IVR)*, the system status and SAM action can be considered in two stages: before core damage and after core damage. For the state before core damage, we are interested in the status of specific systems relevant to IVR. The systems used in a general large evolutionary pressurized water reactor are Safety Depressurization System (SDS), Cavity Flooding System (CFS), and Safety Injection System (SIS). After core damage, we are interested in such SAM actions of Rapid Depressurization (RD), External Reactor Vessel Cooling (ERVC) and In-Vessel Injection (IVI).

After primary system depressurization (RD) essentially necessary in common for such three SAM options, (1) ERVC only, (2) IVI only, and (3) both ERVC and IVI as defined in section 2, *CDF(IVR)* in Eq.(1) can be estimated as following:

For the SAM option (1), the *CDF(IVR)* is

$$CDF\,(IVR) = \sum_i CDF_i \left\{ A_i\,(RD) A_i\,(ERVC) P\,(IVR \mid ERVC \mid RD) \right\} \tag{2}$$

For the SAM option (2), the *CDF(IVR)* is

$$CDF\,(IVR) = \sum_i CDF_i \left\{ A_i\,(RD) A_i\,(IVI) P(IVR \mid IVI \mid RD) \right\} \tag{3}$$

For the SAM option (3), the *CDF(IVR)* is

$$CDF\,(IVR) = \sum_i CDF_i \left\{ \begin{array}{l} A_i\,(RD) A_i\,(ERVC) A_i\,(IVI) P(IVR \mid IVI \mid ERVC \mid RD) \\ + A_i\,(RD) A_i\,(ERVC) [1 - A_i\,(IVI)] P(IVR \mid ERVC \mid RD) \\ + A_i\,(RD) [[1 - A_i\,(ERVC)] A_i\,(IVI) P(IVR \mid IVI \mid RD) \end{array} \right\} \tag{4}$$

In Eqs(2)-(4), i represents the specific core damage accident sequence, CDF_i means the core damage frequency of an accident sequence, $A_i(RD)$, $A_i(ERVC)$, $A_i(IVI)$ are the values of availability for RD, ERVC, and IVI after core damage, respectively, and *P(IVR|ERVC|RD)* is the conditional probability of in-vessel core melt retention once SAM actions of ERVC and RD are successful, *P(IVR|IVI|RD)* is the conditional probability of in-vessel core melt retention once SAM actions of IVI and RD are successful, and *P(IVR|ERVC|IVI|RD)* is the conditional probability of in-vessel core melt retention once SAM actions of IVI, ERVC and RD are successful.

4. Application to KNGR design

In this section, the *FCDF(IVR)* of the KNGR is estimated using the results of the KNGR level 1 PSA results and assumptions for the availability of systems dependent on the status before core damage. Table 1 shows the core damage sequence names, frequencies, system status before core damage, and RCS pressure ranges of the KNGR severe accident sequences whose CDF is greater than reactor vessel spontaneous rupture.

4.1 System Availability after Core Damage

The values of system availability after core damage are assumed according to the system status before core damage as following:

Availability of Rapid Depressurization (RD)

For low pressure sequences such as large and medium LOCA and for high pressure sequences with successful secondary cooling A_i *(RD)* =1. For a high-pressure sequence, A_i *(RD)* = 0.9284 when SDS is available before core damage, and A_i *(RD)* = 0.8430 considering high stress condition when SDS is unavailable before core damage. The reason why RD is available after core damage even though unavailable before core damage is that large fraction of unavailability of SDS is due to operator human error, and when it is not mechanically failed, operator can try RD action again as a SAM after core damage.

Availability of External Reactor Vessel Cooling (ERVC)

For the KNGR, it is assumed that one train of shutdown cooling system (SCS) pump can be used for ERVC. Availability of ERVC, A_i *(ERVC)*, is independent of RCS pressure. When SCS pump is available before core damage, A_i *(ERVC)* is assumed 0.9 considering high stress condition. However, when the system is unavailable before core damage, A_i *(ERVC)* = 0.7794 is used which is obtained by multiplying a recovery factor of 0.866 to 0.9.

Availability of In-Vessel Injection (IVI)

For the KNGR, safety injection into RCS can be accomplished by using SIS or SCS. We used 0.99 as A_i *(IVI)* when SIS and SCS is available before core damage, and 0.9 when SIS is unavailable and 0.7794, in the same way as ERVC, when SIS and SCS are unavailable before core damage.

Table 1 Core damage frequencies and system status before core damage of the KNGR severe accidents

Sequence	CDF (/R-Y)	System Status before Core Damage			RCS Pressure at core Damage
		SDS	SIS	SCS	
LOFW-S17	3.49E-07	Available	Unavailable	Available	Low
SLOCA-S23	2.61E-07	Available	Unavailable	Available	High
SGTR-S28	1.67E-07	Available	Unavailable	Unavailable for RCS injection	Low
SLOCA-S22	1.17E-07	Available	Unavailable	Unavailable for RCS injection	Low
ATWS-S12	1.12E-07	Unavailable	Available	Available	Low
MLOCA-S23	1.07E-07	Available	Available	Available	Low
Vessel Rupture	8.87E-08	N/A	N/A	N/A	N/A

4.2 Conditional Probability of IVR after Successful SAM actions

Now we have to determine the three conditional probabilities *P(IVR|ERVC|RD), P(IVR|IVI|RD), and P(IVR|ERVC|IVI|RD).* Once RD, ERVC and IVI are successfully done, the conditional probability of IVR, *P(IVR|ERVC|IVI|RD)*, would be about 99% for a large power reactor.

However, there are somewhat large uncertainty in the two conditional probabilities *P(IVR|ERVC|RD)* and *P(IVR|IVI|RD)* for a large power reactor. Thus, in this paper we used the two values as sensitivity parameters to obtain *FCDF(IVR)* as a measure of effectiveness of 3 SAM actions for IVR in the KNGR. For *P(IVR|ERVC|RD)* and *P(IVR|IVI|RD)*, we used a series of values from 0.5 to 0.9.

4.3 *FCDF(IVR)* for each SAM options in the KNGR Design

In order to calculate the values of *FCDF(IVR)* with the three SAM options for the KNGR as a function of the two conditional probabilities, *P(IVR|ERVC|RD)* and *P(IVR|IVI|RD*), a spread sheet for event tree is constructed and the resulting values of *FCDF* are shown in Tables 2, 3, and 4.

SAM Option 1 (ERVC only)

As shown in Table 2, when we use *P(IVR|ERVD|RD)= 0.8* considering uncertainty of IVR with only ERVC for the KNGR, the fraction of core damage frequency where IVR is successful, *FCDF(IVR)*, is about 68%.

SAM Option 2 (IVI only)

The coolability of corium in-vessel with only IVI is more uncertain for a large power reactor. Thus, when we assume *P(IVR|ERVD|RD)= 0.7*, the percent of core damage frequency where IVR is successful, *FCDF(IVR)*, is about 61% for the KNGR severe accident sequences as shown in Table 3.

SAM Option 3 (ERVC and IVI)

For SAM option (3), *FCDF(IVR)* is estimated as a function of two dimensional matrix of *P(IVR|ERVC|RD)* and *P(IVR|IVI|RD*) and it is shown in Table 4. We have *FCDF(IVR)* of about 91% when we assume 0.8 and 0.7 for *P(IVR|ERVC|RD)* and *P(IVR|IVI|RD*), respectively. In this case, the *CDF* where IVR fails is 1.05×10^{-7}/R-Y. This value is comparable to the spontaneous reactor vessel failure frequency of 8.87×10^{-8}/R-Y.

Table 2 FCDF(IVR) for Option 1 (ERVC only)

P(IVR|ERVC|RD)	0.9	**0.8**	0.7	0.6	0.5
FCDF(IVR)	0.7687	**0.6833**	0.5979	0.5125	0.4271

Table 3 FCDF(IVR) for Option 2 (IVI only)

P(IVR|IVI|RD)	0.9	0.8	**0.7**	0.6	0.5
FCDF(IVR)	0.7846	0.6975	**0.6103**	0.5231	0.4359

Table 4 FCDF(IVR) for Option 3 (ERVC and IVI) for *P(IVR|ERVC|IVI|RD)* = 0.99

		P(IVR|IVI|RD)				
		0.9	0.8	**0.7**	0.6	0.5
P(IVR|ERVC|RD)	0.9	0.9372	0.9262	0.9150	0.9039	0.8927
	0.8	0.9279	0.9168	**0.9056**	0.8945	0.8834
	0.7	0.9185	0.9074	0.8963	0.8852	0.8741
	0.6	0.9092	0.8981	0.8869	0.8758	0.8647
	0.5	0.8998	0.8887	0.8776	0.8665	**0.8554**

Comparison of the results shown in Tables 2 and 3 shows that the most contributing factor to *FCDFs* for KNGR SAM option 1 (ERVC only) and SAM option 2 (IVI only) is the conditional probabilities of in-vessel retention after SAM actions.

According to Tables 3 and 4, *FCDF* of SAM option 3 (with both ERVC and IVI), even though we assume 0.5 for the two conditional probabilities, we obtain *FCDF(IVR)* of 86%. However, the value is considerably greater than *FCDF* of SAM option 2 (in-vessel injection only), 44%.

5. Conclusion

In order to get efficiency of external reactor vessel cooling in the Korean Next Generation Reactor design for in-vessel core melt retention under hypothetical severe accident, a simple probabilistic analysis method has been proposed which allows to estimate fractional core damage frequency where the in-vessel retention can be achieved.

The method is applied to the KNGR severe accident sequences whose CDF is greater than spontaneous vessel rupture and the fractional CDF is calculated using conditional probabilities of in-vessel retention under each SAM option as sensitivity parameters. It is found that for about 90% of severe accidents in-vessel retention could be achieved with severe accident management actions of primary depressurization, external reactor vessel cooling and late in-vessel injection.

Inclusion of SAM action of external reactor vessel cooling increases the fractional core damage frequency of the KNGR where in-vessel retention is attained from 44% to 86% even with conservative assumptions for conditional probabilities of in-vessel retention as 0.5. This implies that, for the KNGR design, which is a large evolutionary pressurized water reactor, it is worthwhile to include external reactor vessel cooling strategy in pursuit of in-vessel retention of molten core under severe accidents.