Evaluation of Technological Appropriateness of the Implemented Accident Management Measures for BWR by Level 1 and Level 2 PSA Methods

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1. Introduction

After the TMI-2 accident, the researches on severe accidents and PSA have made extensive progresses worldwide. With these backgrounds in Japan, electric utilities have steadily implemented by their own initiatives countermeasures both for preventing the occurrence of severe accidents and mitigating their consequence. In addition, in July of 1992, the MITI (Ministry of International Trade and Industries) requested the utilities to prepare the accident management (AM) plans responding to policy based on the discussions for severe accident by Nuclear Safety Commission.

In March of 1994, the utilities submitted AM study reports on light water reactor NPPs to the MITI. The AM measures include newly chosen candidates as well as already implemental ones, and vary depending on the type of reactors. The MITI reviewed(1) the technical adequacy of the reports, and encouraged the utilities to proceed implementation of AM measures, operational procedures etc. The implementation is expected to complete by February 2002 for all PWRs and BWRs.

In May of 2001, the “Review Programs of Accident Management” by METI (former MITI) was initiated, and the associated technical supports have been performed by the NUPEC(2).

The present paper deals with results on examinations of effectiveness of preventive and mitigative AM countermeasures by PSAs for a typical BWR-3 with Mark-I, BWR-4 with Mark-I, BWR-5 with Mark-II and ABWR. These four types of BWR classification cover all the BWRs because a policy of standardized design is undertaken in Japan.

2. Accident Management Measures for BWR Plants

2.1 Preventive AM Measures

Table 2.1(a) summarized the AMs for prevention of core damage for all BWRs. The ARI (Alternative Rod Insertion) is activated with signals for high pressure of the reactor coolant system, or low liquid level in the reactor pressure vessel. The RPT (Re-circulation Pump Trip) is activated with the same signals as for the ARI. The logic control systems of these new signals are independent of conventional scram & ECCS.

The reinforcement of the automatic depressurization system (ADS) is activated by a signal with low liquid level of the reactor vessel to prevent core damage for a transient with failure of the high pressure core cooling system. The alternative water injection is achieved by use of the make-up line or water supply from the fire protection system.

The containment hardened vent is introduced to prevent accident sequence with loss of decay heat removal that leads to the containment failure before core melting. The power supply in station blackout sequences is achieved by the accommodation of 6.9kV & 480V from adjacent plants. In addition, the power supply is changed from the swing type to the exclusive type of the emergency diesel generator (EDG) for BWR-3 and BWR-4 plants.

2.2 Mitigative AM Measures

Table 2.1(b) shows the AMs for mitigation of accident progression for all BWRs. The alternative water injection to the containment is achieved with use of the make-up line and also water supply from the fire protection system. The alternative heat removal is achieved with use of the drywell cooler, use of the heat exchanger in the make-up line, recovery of the RHR system and the containment hardened vent.

3. Evaluation of Effectiveness of Accident Management Countermeasures

3.1 Prevention of Core Damage

Figures 3.1 compiles core damage frequencies (CDFs) in each core damage state for BWR-3, BWR-4, BWR-5 and ABWR. The CDFs were effectively reduced with the implementation of AMs.

In BWR-3, the reduction of the CDF by AMs was not extensive compared with the other types, because the accident sequence that leads to core damage was dominated by large or medium size break LOCAs. In BWR-4, the reinforcement of the ADS was effective to reduce the CDF. In BWR-5, the CDF was reduced effectively by the containment vent for TW, and by the ARI and RPT for TC. In ABWR, since highly reliable ECCSs, RHR and scram system are designed into ABWR, the CDF was intrinsically small compared with the other types.

3.1.1 Dominant Accident Sequences and Core Damage Frequencies

(1) ATWS sequence (TC)

The reductions of the CDFs in BWR-3, BWR-4 and BWR-5 for TC sequences were significantly large because common cause failure of the scram logic control system in the dominant sequence was effectively reduced by the implementations of ARI and RPT (Figures 3.1(a), (b), (c)). The no reduction of the CDF for the TC sequence of the ABWR type means that ARI and RPT were already implemented in the design stage (Figure 3.1(d)).

(2) LOCA with loss of depressurization (LOCA-X) & LOCA with loss of water injection (LOCA-V)

In BWR-3 for LOCA-X, since AM for depressurization was not implemented, there is no reduction of CDF (Figure 3.1(a)). As for LOCA-V, since there are no enough time to depressurize due to rapid accident progressions to the core damage, reductions of CDFs were not expected for BWR-3, BWR-4 and BWR-5. In ABWR for LOCA-X and LOCA-V, contributions of alternative water injection (AM) to prevent core damage were not significant, because reliabilities of high pressure & low pressure injection systems were kept high by means of independent redundant systems.

(3) Transient with Loss of depressurization (TQUX)

In BWR-4 and BWR-5, the AM with reinforcement of ADS was significantly effective to reduce the CDF (Figures 3.1(b), (c)). Since the isolation condenser is installed, depressurization AM is not applied to BWR-3. ABWR has no AMs for depressurization because of high reliability of high and low pressure injection systems.

(4) Transient with loss of ECCS function (TQUV)

The alternative water injection is achieved using a pathway of the containment spray line for BWR-3, and the Low Pressure Core Injection (LPCI) lines for BWR-4 and BWR-5. The core damage frequencies were reduced effectively by the alternative water injection (Figures 3.1(a), (b), (c)). In ABWR, reduction of the CDF was larger than those of BWR-3, BWR-4 and BWR-5 because of the multiple-injection pathways (Figure 3.1(d)).

(5) Loss of Decay heat Removal function (TW)

Since the containment vent system was implemented to the all of BWR types, CDFs were reduced with reliabilities of vent systems (Figure 3.1).

(6) Loss of all AC power (TB)

In BWR-3 and BWR-4, the reductions of the CDFs were not extensive because power supply from the DG in the adjacent plant to the High Pressure Core Injection system (HPCI) or the Reactor Core Isolation Cooling system (RCIC) were not effective by 8 hours (exhaust time of DC battery) after station blackout (Figures 3.1(a), (b)). In BWR-5, the dominant accident sequence was a transient with loss of recovery of AC power within 24 hours. The AM of power supply from adjacent plant through 6.9kV buss became effective in BWR-5 (Figure 3.1(c)). The failure probability of power supply was estimated to be 0.18. In ABWR, the reduction of the CDF was not extensive compared with the other type BWR (Figure 3.1(d)), because the core damage during loss of AC power was dominated by mechanical failures of RCIC.

3.1.2 Effects of Accident Management Countermeasures on Core Damage

The results of the present study indicated that the reductions of the CDFs were estimated to be 2/3 for BWR-3, 1/2 for BWR-4, 1/4 for BWR-5 and 1/6 for ABWR. In addition, the CDF for a BWR plant was estimated to be lower than $3x10^{-7}$ (1/R.y) considering with AMs.

On the bases of the results in the present study, CDFs for BWRs were effectively reduced by the implemented AMs.

3.2 Prevention of Containment Failure

In the Level 2 PSA, plant damage states (PDSs) were defined based on the results of the Level 1 PSA. Figure 3.2 and Table 3.1 show the PDSs used in the present study for BWR-5 with Mark-II. The PDSs were used in the other type BWRs in the present study. The AMs become effective to reduce the CDFs such as the alternative water injection for TQUV, the reinforcement of the ADS for TQUX, AC power accommodation for TB, the containment vent and the alternative water injection for TW and ARI & RPT for TC (Figure 3.2) as discussed in Chapter 3.1.

The containment event trees were developed for each plant damage state, and the end states of the containment event trees were classified into containment failure modes. Table 3.2 shows the containment failure modes used in the present study. In Table 3.2, the containment failure occurs before core melt for the over-pressurization due to loss of decay heat removal and the over-pressurization with failure of reactor scram.

3.2.1 Containment Failure Frequencies for 4 types of BWR

Figure 3.3 compiles the containment failure frequencies (CFFs) for BWR-3, BWR-4, BWR-5 and ABWR. The containment failure frequencies were effectively reduced by the implemented AMs.

(1) Drywell Wall Melt-Through

In the BWR-3 with Mark-I, the CFF due to TQUV, which was dominant sequence, was effectively reduced by the alternative water injection, the recovery of the component cooling system (CCS) and the containment vent (Figure 3.2(a)). In the BWR-4 with Mark-I, the CFF due to AE, which was the dominant sequence, was effectively reduced by the alternative water injection, the recovery of the decay heat removal system (RHR) and the containment vent (Figure 3.2(b)). In the BWR-5 with Mark-II and ABWR, the drywell wall melt-through would not occur, because the molten debris moves downward in the reactor pedestal (Figure 3.2(c), (d)).

(2) Over-pressurization with Steam/Non-Condensable Gases during MCCI

In the BWR-3 with Mark-I, the CFFs were reduced by the AMs such as the alternative water injection, recovery of the CCS and the containment vent for TQUV and AE (Figure 3.2(a)). Especially, the alternative water injection to the reactor vessel using the RHR line became effective to reduce the CFF for AE. In the other BWR types, the alternative water injection, recovery of the RHR and the containment vent became effective to reduce the CFFs (Figure 3.2(b), (c), (d)).

(3) High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH)

In the BWR-3 with Mark-I, the alternative water injection, recovery of the CCS and recovery of AC power became effective to reduce the CFFs for TB, which was dominant accident sequence that leads to HPME and DCH (Figure 3.2(a)). In the BWR-4 with Mark-I, BWR-5 Mark-II and ABWR, recovery of AC power became effective to reduce the CFFs (Figure 3.2(b), (c), (d)).

3.2.2 Effects of Accident Management Countermeasures on Containment Failure

The results of the present study indicated that the reductions of the CFFs were estimated to be 1/18 for BWR-3 Mark-I, 1/5 for BWR-4 Mark-I, 1/5 for BWR-5 Mark-II and 1/12 for ABWR. In addition, the CFF for a BWR plant was estimated to be lower than $6x10^{-8}$ (1/R.y) considering with AMs. In addition, frequencies of early containment failure and containment bypass sequences that lead to the early large releases were significantly reduced to 1/22 for BWR-3 Mark-I, 1/82 for BWR-4 Mark-I, 1/50 for BWR-5 Mark-II and 1/1.1 for ABWR (Figure 3.2).

On the bases of the results in the present study, CFFs for BWRs were effectively reduced by the implemented AMs.

3.2.3 Analysis of Severe Accident Progressions

In the present study, the MELCOR1.8.3 code was used to examine the effects of mitigation on the accident progressions with AMs. Figure 3.3 shows the calculated results of core liquid level regarding with the water injection flow rate of the alternative water injection. The calculated results indicated that the core is cooled down by the alternative water injection even if the after core damage.

Figures 3.4 shows the pressure behavior in the containment. The pressure in the drywell was suppressed by the alternative water injection from the pool in the fire protection system to the containment. In the case of failure to recovery of the RHR system, the calculated results showed that containment vent would be operated at about 1 day after from accident initiation. Table 3.3 shows allowable times of the recovery actions for AC power and RHR including other actions to prevent failures of the reactor vessel and the containment failure.

4. Conclusion

The present study examined the effectiveness of accident management countermeasures in terms of the Level 1 PSA and Level 2 PSA for the BWR-3 Mark-I, BWR-4 Mark-I, BWR-5 Mark-II and ABWR in Japan. The results indicated that accident management countermeasures implemented to BWRs in Japan were effective to reduce core damage frequency and containment failure frequency.

The core damage frequencies and containment failure frequencies for BWRs were estimated to be lower than $3x10^{-7}$ (1/R.y) and $6x10^{-8}$ (1/R.y), respectively. In addition, containment failure frequencies that lead to early large release were significantly reduced with AMs for BWRs.

References

[1] M.Sobajima, et al., "Current Status of the Implementation Plan of Accident Management to Nuclear Power Plants," (in Japanese) J. of JAES, Vol.37, No.5 (1995)

[2] INS/NUPEC, "Review of Accident Management for Typical PWRs and BWRs in Japan," (in Japanese), INS/M00-27 (2001)

Table 2.1 Accident Management Countermeasure for BWR

(a) Prevention of Core Damage

AM Functions	Equipment & Systems	Accident Sequences	Comments
Reactor Scram	- ARI (Alternative Rod Insertion: activation with signals for high pressure of the reactor coolant system, or Low core liquid level) - RPT(Re-circulation Pump Trip: activation with the same signals above)	transient without scram (TC)	New signals are independent of conventional scram & ECCS signals. These systems are already designed into ABWR.
Depressurization	- Automatic depressurization system (ADS) is activated by a signal with low liquid level of the reactor vessel.	transient with failure to depressurization (TQUX)	This AM is not applied to BWR-3 and ABWR. BWR-3: Isolation Condenser is implemented. ABWR: high pressure ECCSs are already reinforced.
Alternative Water Injection	- Use of the make-up line. - Water supply from the fire protection system	transient with loss of ECCS injection (TQUV)	
Alternative Heat Removal	- The containment hardened vent	transient with loss of decay heat removal (TW)	
Supply AC Power	- Accommodation of 6.9kV & 480V from adjacent plant - Power supply from emergency diesel generator (EDG)	loss of all AC power (TB)	This AM is applied to BWR-3 & BWR-4 plants.

Table 2.1 Accident Management Countermeasure for BWR (Continued)

(b) Mitigation of Accident Progression (Applied to Core Damage)

AM Functions	Equipment & Systems	Accident Sequences
Depressurization (same as prevention)	- Automatic depressurization system (ADS) is activated by a signal with low liquid level of the reactor vessel.	transient with failure to depressurization (TQUX)
Alternative Water Injection	- Use of the make-up line. - Water supply from the fire protection system	transient with loss of ECCS injection (TQUV) transient with failure to depressurization (TQUX) loss of all AC power (TB, TBU) LOCA with loss of ECCS injection (AE)
Alternative Water Injection to Containment	- Use of the make-up line. - Water supply from the fire protection system	same as above
Alternative Heat Removal	- Use of the drywell cooler, and use of the heat exchanger in the make-up line. - Recovery of the RHR system - The containment hardened vent	same as above
Supply AC Power	- Accommodation of 6.9kV & 480V from adjacent plant - Power supply from emergency diesel generator (EDG)	loss of all AC power (TB, TBU)

Table 3.1 Plant Damage States

Designators	Contents
TQUV	Transient with loss of all ECCS injections (including small break LOCA)
TQUX	Transient with failure to depressurization of the reactor coolant system
TB	Transient with loss of all AC powers
TBU	Transient with loss of all AC & DC powers
TW	Transient with loss of decay heat removal
TC	Transient without scram
AE	LOCA with loss of all ECCS injections
V	Interface-systems LOCA

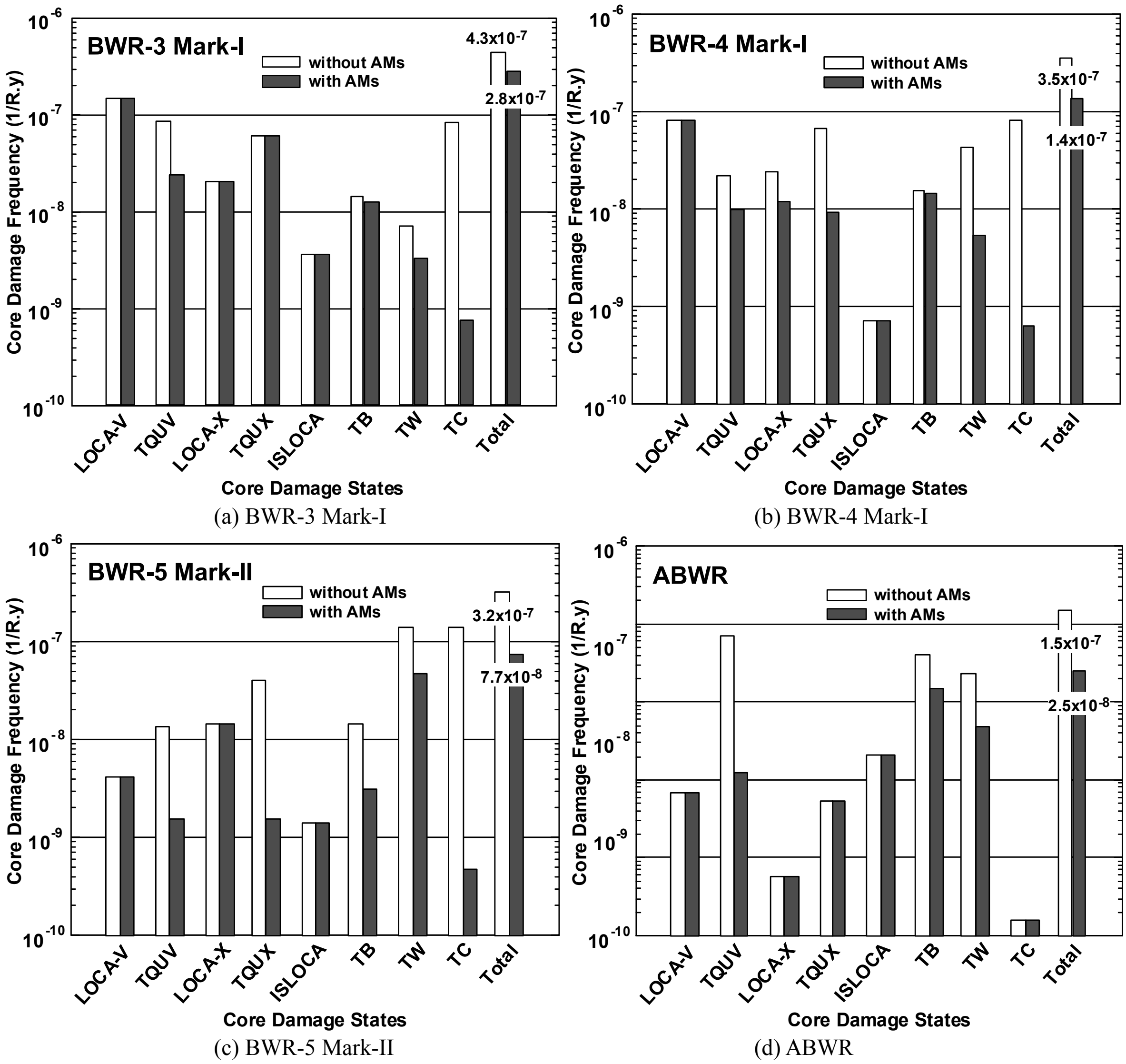
Table 3.2 Containment Failure Modes

Designators	
τ	Wall melt-through
σ	Direct containment heating (DCH)
μ	High pressure melt ejection (HPME)
δ	Over-pressurization with steam/non-condensable gases
TW–θ	Over-pressurization with steam in transient with loss of decay heat removal
TC–θ	Over-pressurization with steam in ATWS
α	In-vessel steam explosion
ν	Interface-systems LOCA & Containment bypass

Table 3.3 Allowable Times for Recovery Actions

(Example of the BWR-5 with Mark-II containment)

Accident Management Countermeasures	PDS	Allowable Time (hr)
-Alternative Water Injection to the Reactor Coolant System	TQUV	1 - 2
	TQUX,TB,TBU	1 - 2
	AE	1 - 2
-Alternative Water Injection to the Containment (Water injection to core debris)	all PDSs	about 1 hour after the reactor vessel failure
-Alternative Heat Removal (drywell cooler, use of heat exchanger in the Make-up system)	all PDSs	about 1 hour
-Recovery of RHR		
recovery before the reactor vessel failure with successful of the Alternative Heat Removal	all PDSs	about 50 hours
recovery before the reactor vessel failure with failure of the Alternative Heat Removal	all PDSs	about 35 hours
after the reactor vessel failure		about 8 hours
-Containment Venting		about 20 hours
-Accommodation of AC Power or Recovery of DG		
before the reactor vessel failure	TBU	2 – 3 hours
	TB	6 – 7 hours
after the reactor vessel failure	TBU	about 20 hours
	TB	about 20 hours



(a) BWR-3 Mark-I

(b) BWR-4 Mark-I

(c) BWR-5 Mark-II

(d) ABWR

Figure 3.1 Core Damage Frequencies

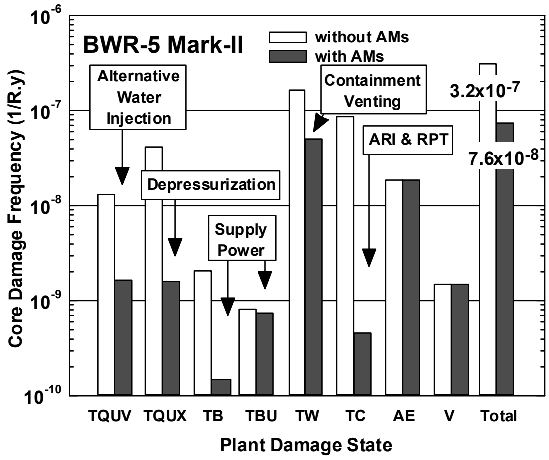
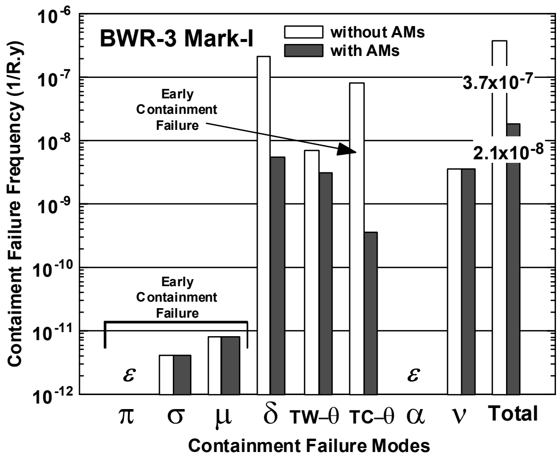
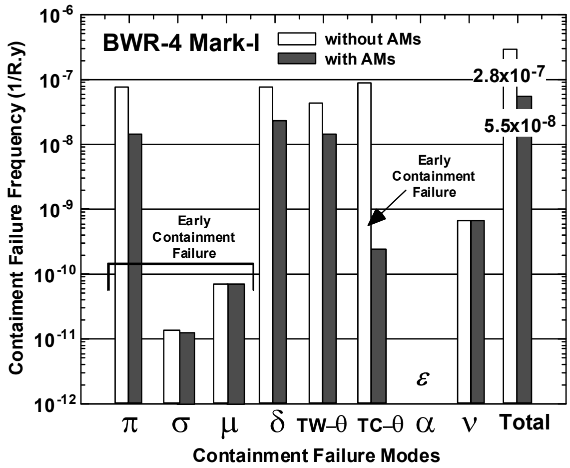


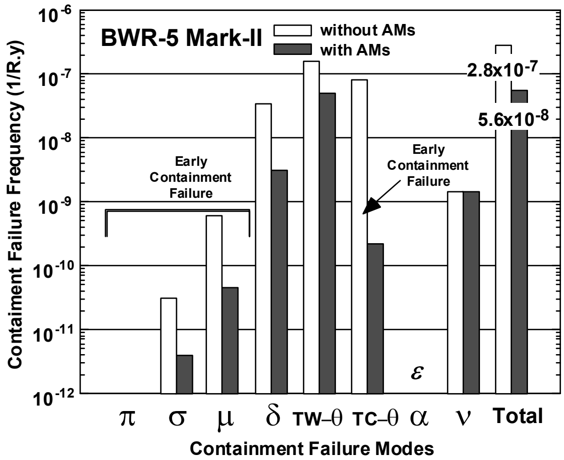
Figure 3.2 Plant Damage States (Example of the BWR-5 Mark-II)



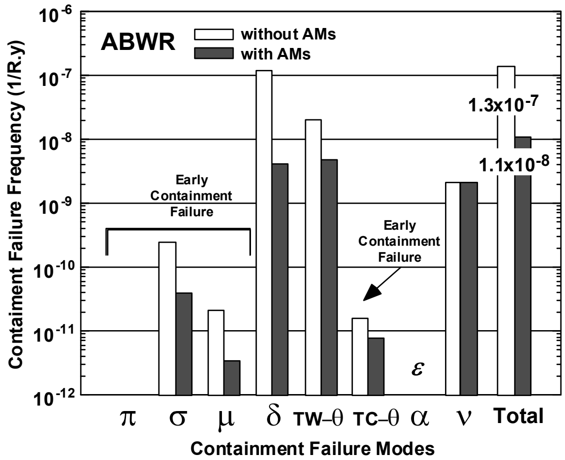
(a) BWR-3 Mark-I



(b) BWR-4 Mark-I



(c) BWR-5 Mark-II



(d) ABWR

Figure 3.2 Containment Failure Frequencies

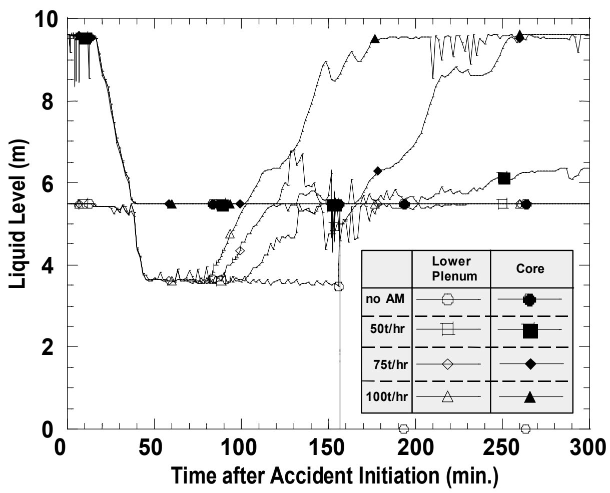


Figure3.3 Liquid level in the Reactor Vessel with various timings of the Alternative Water Injection

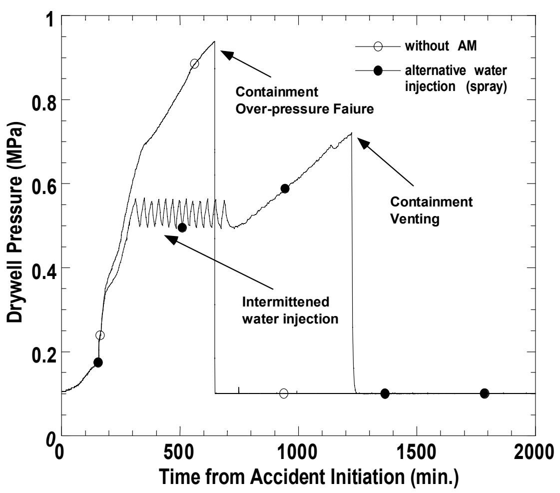


Figure3.4 Pressure Behavior in the Drywell with the Alternative Water Injection to the Containment