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A PROPOSAL FOR ACCIDENT MANAGEMENT OPTIMIZATION BASED ON THE STUDY OF ACCIDENT SEQUENCE ANALYSIS FOR A BWR

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Abstract

The paper describes a proposal for accident management optimization based on the study of accident sequence and source term analyses for a BWR. In Japan, accident management measures are to be implemented in all LWRs by the year 2000 in accordance with the recommendation of the regulatory organization and based on the PSAs carried out by the utilities. Source terms were evaluated by the Japan Atomic Energy Research Institute (JAERI) with the THALES code for all BWR sequences in which loss of decay heat removal resulted in the largest release. Identification of the priority and importance of accident management measures was carried out for the sequences with larger risk contributions. Considerations for optimizing emergency operation guides are believed to be essential for risk reduction.

I. Introduction

Accident management (AM) measures are to be implemented in all light water power reactors in Japan by the year 2000 [1]. Since the examination of accident management was requested to the utilities by the regulatory authorities, the Nuclear Safety Commission (NSC) and the Ministry of International Trade and Industry (MITI), each utility planned their strategy of implementing accident management in their power reactors [2], [3]. These measures were proposed on the basis of their probabilistic safety assessments (PSAs). They were approved after reviews by the regulatory bodies to be appropriate as a whole finally in November 1995.

JAERI is performing PSA studies for general purposes which include the examination of accident management. Analytical codes for evaluating core damage sequences are being developed and applied to power reactors for studying the consequences of severe accident sequences. Source term analysis with the THALES/ART [4] and THALES-2 [5] codes developed for this purpose is available, as source term models for release, transportation, deposition and revaporization of radioactive materials are incorporated in them. Since these codes run relatively fast, overall calculation of many sequences is feasible. Based on comparison of various sequences, numerous insights and knowledge concerning categorization of similar sequences could be gained.

With this knowledge and based on the study on the AM measures proposed by the utilities, the author proposes some important aspects which are desired to be incorporated in the emergency plant operation procedure guide for optimizing the effect of each AM measure.

II. Preceding Work

Kajimoto et al. [6] performed severe accident sequence analysis of a BWR and grouped the various sequences according to the scenario of pressure vessel failure and containment failure due to overpressure. They clarified the effects of core melt relocation, coupling of thermal-hydraulics and fission product (FP) vaporization, and containment failure location on the release fraction of CsI to the environment for a variety of sequences shown in Table 1 as sensitivity studies by using the THALES/ART (THALES-2) code.

The major findings of the work can be summarized as follows:

(1) The analyzed sequences initiated by transients and LOCA are basically categorized in 5 groups according to the similarity of the scenario from core damage to containment failure.

(2) The released mass of CsI also exhibits similar behavior in each group. It is affected by the melt relocation modes in which the core melt is once trapped at the core support plate at the bottom of the core or not. If the melt experiences high temperature there due to lack of coolant, the major part of FPs is released into the reactor coolant system (RCS) and deposits in the RCS, more or less, depending on the time duration in the sequence.

(3) Melt having not previously experienced high temperature may later release major amounts of FPs during the process to containment failure. This could result in larger source term release into the environment. An example of FP release in transient sequences is as given in Fig. 1 which shows larger release for such sequence groups having shorter FP deposition time like the TW groups named here Groups A and B, and the TC Group E.

(4) The effect of containment failure location on source term release is also significant as shown in Fig. 2 in which drywell space and wetwell liquid space failures with no water scrubbing effect show relatively large source terms.

(5) When thermal-hydraulics and FP revaporization are coupled in the code model, the revaporization of FPs in the reactor coolant system (RCS) during the containment depressurization process increases the source term to the environment by a factor of 10 at maximum compared with the source term without revaporization.

Watanabe et al. [7] performed further sequence analysis for the study of containment failure modes. They regrouped all significant accident sequences into 8 groups including interfacing system LOCA and reactor pressure vessel (RPV) rupture, and separated the loss of containment heat removal group into long-term and short-term as summarized in Table 2.

They obtained the conditional containment failure probabilities (CCFP) for various failure modes, for four separate accident progression stages in each sequence group and clarified the dominant containment failure modes and stages for each sequence group as summarized in Table 2. The dominant failure modes were mostly over-pressurization and over-temperature in this study. However, the dominating stage was different group by group. For example, containment failure due to overpressurization dominantly occurs in the 'pre-stage for core-melt' in Group 3 with loss of long-term containment heat removal, whereas containment failure occurs in the 'long-term progression stage' in Group 7 with loss of short-term containment heat removal. Though they also proposed mitigative measures by the use of conventional systems as shown in the table, these measures now should be replaced by the AM measures planned lately.

III. Discussions

The author reviewed the AM measures proposed by the utilities in the light of the correspondence with the 8 groups proposed in the previous work by Watanabe et al. and gave some consideration for optimizing those measures in timing and conditions of activation based on the above analytical results and other knowledge obtained through various experiments and analyses of severe accidents.

For a representative BWR-5 with Mark-II containment [1], [2] Table 3 summarizes the relation among each fundamental safety function, accident sequences contributing to core damage frequency (CDF) and containment failure probability and AM measures which are currently adopted and to be implemented together with the AM measures already implemented. The corresponding sequence groups G1 through G8 in the previous study are also indicated in the column of 'accident sequence' of this table. It can be confirmed that all significant sequence groups correspond to either of the individual AM measures in the table except for the interfacing system LOCA group, G1 which has a negligibly small contribution to CDF. Over-pressurization and over-heating scenarios of the containment which were dominant in most of the sequence groups in Table 2, are prevented or mitigated by the adopted AM measures such as 'alternative reactivity control', 'alternative water injection to reactor and containment', 'drywell cooler' or 'hardened vent'. Thus, all sequence groups can be covered by any of the AM measures already implemented and those adopted this time. The level of defence in depth for the core, pressure vessel and containment will be significantly increased by implementing all the AM measures.

Table 1. Accident Sequences Analyzed by Kajimoto et al. [6]

Transient			LOCA		
Transient (other than IORV)		IORV	Large Break	Medium Break	Small Break
TW	TB	T_iW	AW	S_1W	S_2W
TQW	TBU	T_iQW	AUW	S_1UW	S_2QW
TQU_1W	TBP	T_iQU_1W	AUV_1W	S_1UV_1W	S_2QU_1W
TQUW	TBPU	T_iQUW	AB	S_1B	S_2QUW
$TQUV_1W$	TQUX	T_iQUV_1W	AUV	S_1UX	S_2QUV_1W
TPW	TQUV	T_iB	AC	S_1UV	S_2B
TPQW TC		T_iBU		S_1C	S_2BU
$TPQU_1W$		T_iQUX			S_2QUX
TPQUW		T_iQUV			S_2QUV
$TPQUV_1W$					S_2C
TPQUX					
TPQUV					

A : Large break LOCA
B : Loss of all AC power
C : Failure to reactor scram
P : Failure to reclose relief valve
Q : Loss of feedwater
S1 : Medium break LOCA
S2 : Small break LOCA
T : Transient
Ti : Inadvertent opening of relief valve

U : Loss of all high pressure system
U1 : Loss of high pressure core spray
U2 : Loss of RCIC
V : Loss of all low pressure system
V1: Loss of low pressure core spray
V2: Loss of low pressure injection
W : Loss of residual heat removal
X : Failure to depressurization

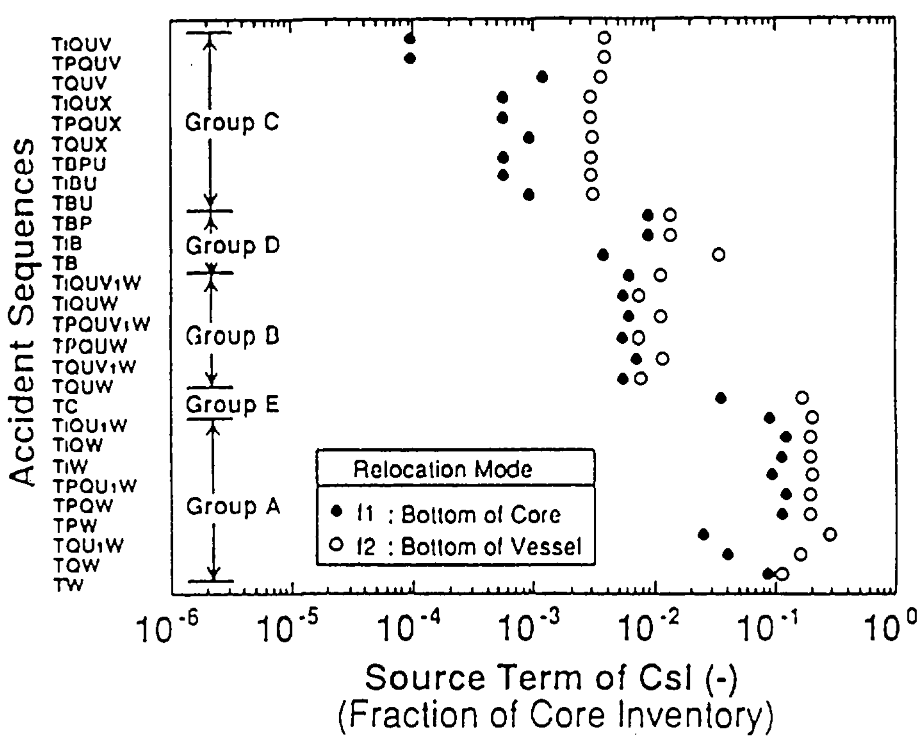

Fig. 1 Effect of Melt Relocation Modes on Source Term of CsI

For enhancing the effectiveness of these AM measures, some further consideration is necessary. Optimization of procedures is useful for establishing an effective emergency operation guide. Since risk is the product of CDF or resultant CCFP shown in Fig. 3(a) and 3(b) and the consequences of source terms, the importance of AM measures naturally focusses more on those which are effective for preventing and mitigating accident sequences with a larger risk contribution. It should be noted here that the contribution of each sequence to CDF and CCFP is different depending on the type of BWR as seen in Fig. 3 because of differences in system and safety function configuration. Nevertheless, there is a commonality for each type and therefore the discussion below will be mainly done based on the results for the BWR-5.

(1) Failure of Decay Heat Removal and Resulting Overpressure

One important recognition is that such sequence groups, having a dominant contribution to CCFP, as the residual heat removal system failure in a BWR-5 also shows a relatively large CsI release consequence as indicated in Fig. 1 and Fig. 2. Therefore the prevention and mitigation of those sequence groups, such as loss of containment heat removal (T--W) and anticipated transients without scram (ATWS or TC), have priority and need to be ensured for total risk reduction. The AM measures for containment heat removal are illustrated in Fig. 4(b). The 'hardened vent' is expected to ultimately ensure containment heat removal in the case the implemented 'manual operation of containment spray system' measure fails.

The determination of the optimum vent pressure is separated into two cases. In a sequence such as TW wherein containment overpressurization takes place earlier than core damage, the vent pressure can be set just above the maximum operating pressure of the containment to expedite the depressurization by releasing steam without FPs. In this case, confirmation of no core damage with radiation monitors and of the unavailability of containment spray and other cooling means is required before the operation of the vent system.

In the other overpressurization sequences which take place after core damage, a larger risk reduction can be achieved by setting the vent pressure as high as just below the endurable limit pressure of the containment and by extending the time of venting as long as possible allowing for FP decay. In this case, the 'drywell cooler' will contribute to suppressing pressurization and extending the time of venting. The wind direction may also be taken into consideration for risk reduction by a potential early venting in the process.

It is already known that water at saturated temperature reduces the scrubbing effect for FPs such as CsI [8]. Since, in the overpressure sequences, the water temperature is in saturation, minimum decontamination factors should be expected. Therefore, for deciding on the venting time, the temperature of the pressure suppression water must also be watched with instrumentation made reliable even in such an accident environment.

Recovery of heat removal is required in a long range in all cases. For determining the containment durability, demonstration tests are planned to be conducted by NUPEC [9]. This result will be useful for optimizing the venting conditions.

(2) ATWS

Alternative reactivity control measures such as reactor pump trip (RPT) and auxiliary control rod insertion (ARI) are adopted for the case of ATWS leading to core damage and containment overpressure. Although the contribution of ATWS to CDF is relatively small, source term release consequences of Group E (TC group) are large. The above measures will further ensure safe reactor shutdown and largely reduce risk. Since both of them are envisaged to be automatically activated in the case of the failure of the emergency scram signal, an operator is only needed to confirm the successful activation and for shutdown cooling afterwards.

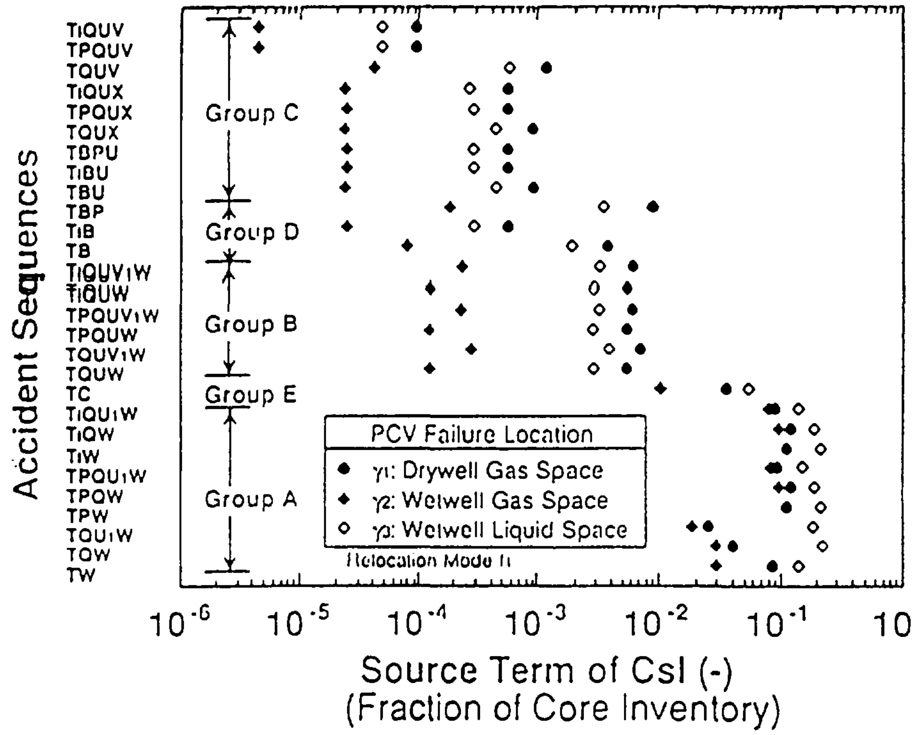

Fig. 2 Effect of Containment Failure Location on Source Term of CsI

(3) Loss-of-Electric Power Supply and Resulting Direct Containment Heating

In addition to the supply of high voltage electric power from the adjacent plant with a cross-tie already implemented, the 'supply of low voltage power' from the adjacent plant with another cross-tie and 'supply of power from D/G for HPCS' are adopted, as the BWR-5 is provided with a special D/G for HPCS. The operator who connects the cross-tie needs to confirm the activation of the D/G of the adjacent plant or of the D/G for HPCS for supplying power. The 'supply of low voltage power' from the adjacent plant will be used to activate D/Gs in case of battery system failure in the accident plant. All these measures for terminating SBO sequences, by establishing the emergency guide and training of the operators, will contribute to eliminate the DCH sequence in Fig. 3(b) and the large source term release of Group D (TB sequences).

(4) Failure of Water Injection or Depressurization and Resulting Over-heating

The contribution of those sequence groups initiated by failure of water injection and depressurization of RPV to CDF is relatively small. The consequential source term release mass is also the smallest of all the groups. However, the probability and consequence of a steam explosion still have a large uncertainty. If it occurs violently in the pressure vessel or in the containment, the source term mass release out of the containment drywell may significantly increase as can be seen in Fig. 2 for Group C.

It is already known that dropping of the melt in water at relatively high pressure (> 1MPa) or with a temperature near saturation does not cause significant steam explosion [10]. Therefore, water mass distribution and the pressure and temperature of the water are crucial for the prediction of steam explosion occurrence. Melt relocation and water conditions must be carefully monitored with required instruments to avoid a steam explosion. When water injection is required for melt cooling both in the pressure vessel and containment, spraying water on the melt is more desirable than filling plenty of water in a place beneath the melt where it finally slumps (see Fig.4(a)). Effective melt cooling conditions also need to be established through research and reflected in the emergency guide to prevent the melt spreading on pedestal floor and directly attacking the containment shell as well as the concrete floor.

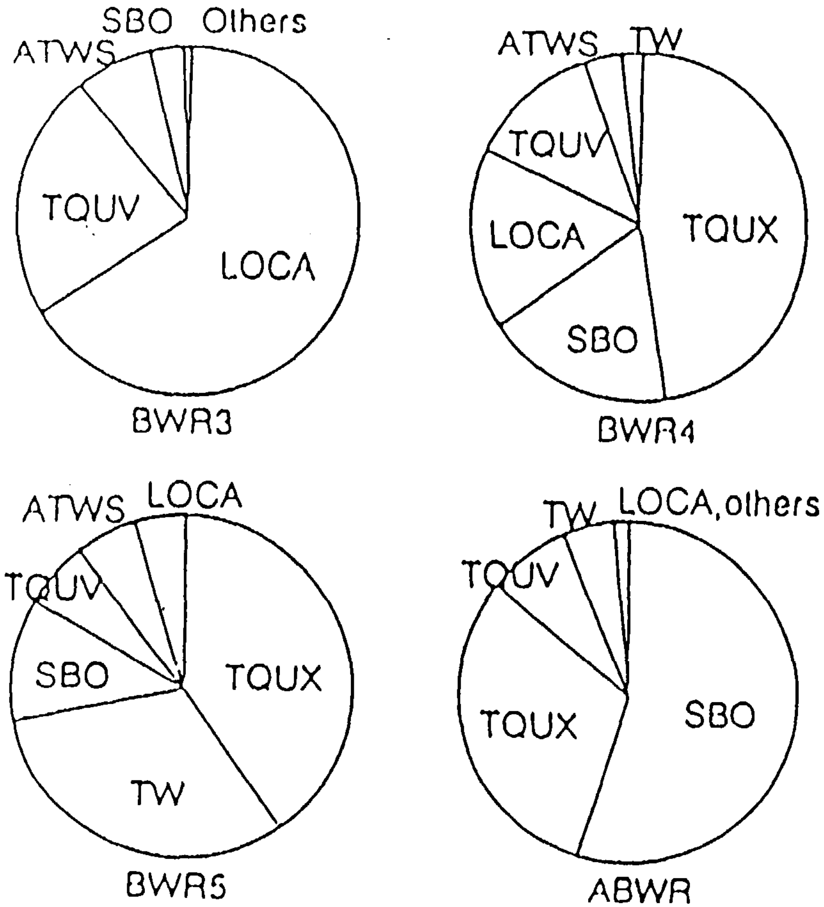

Figure 3(a) level-1 PSA results [2]

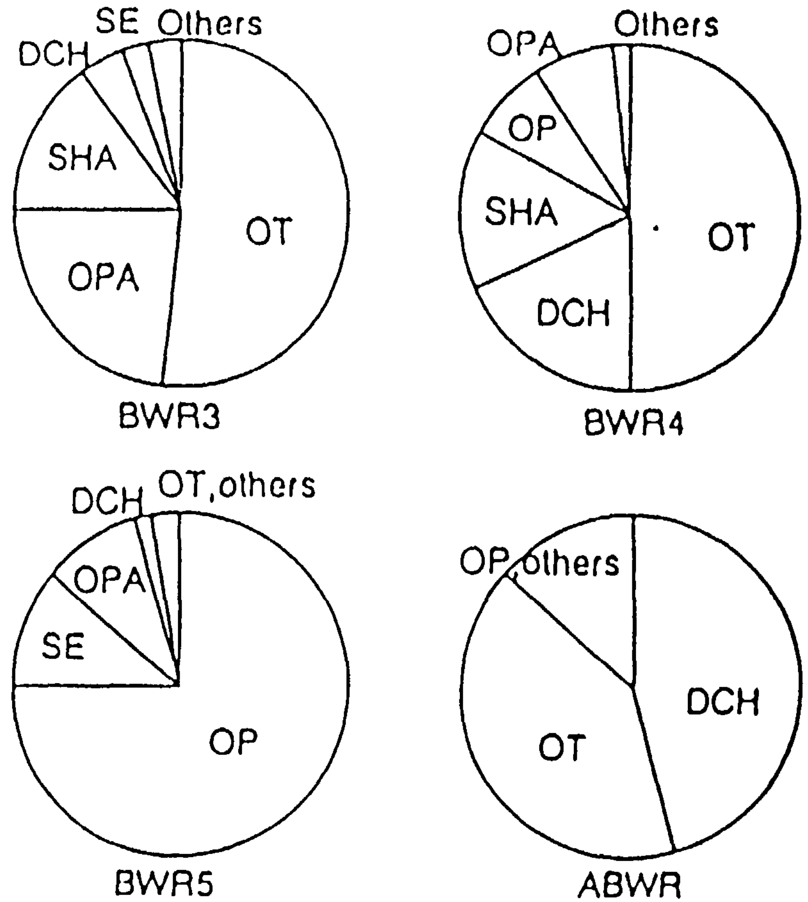

Figure 3(b) level-1.5 PSA results [2]

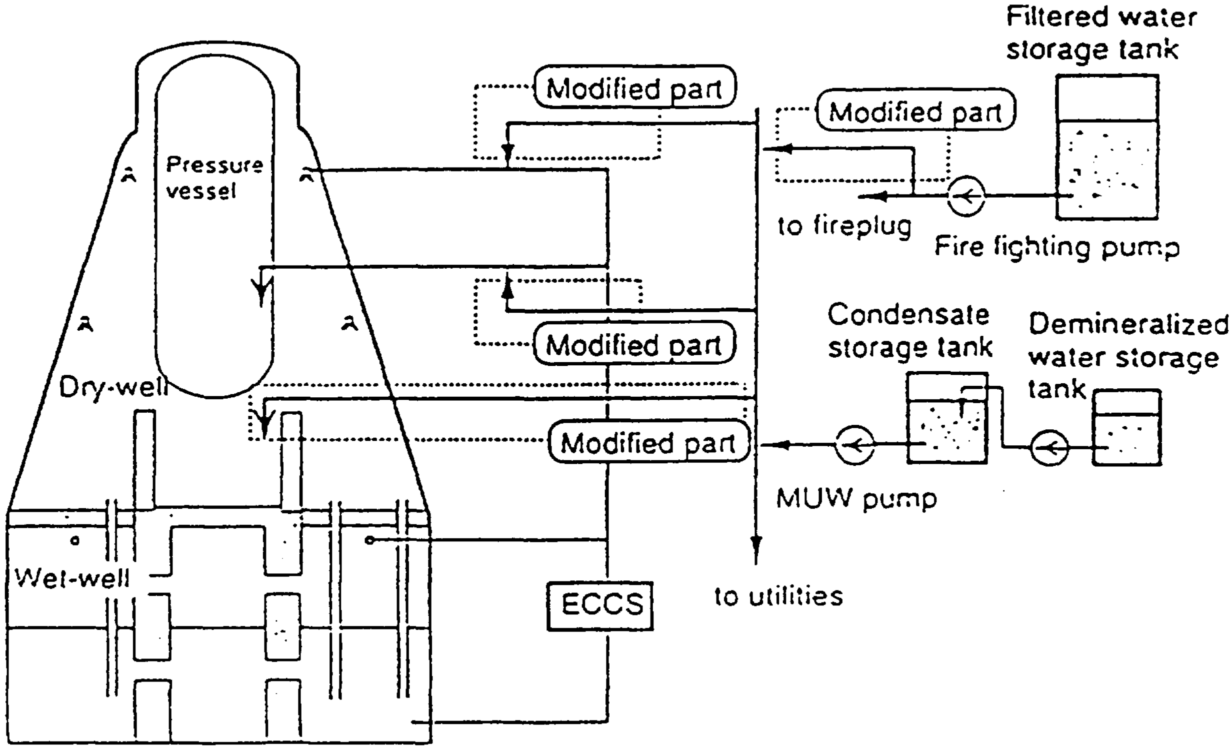

Fig. 4(a) Alternative water injection [1]

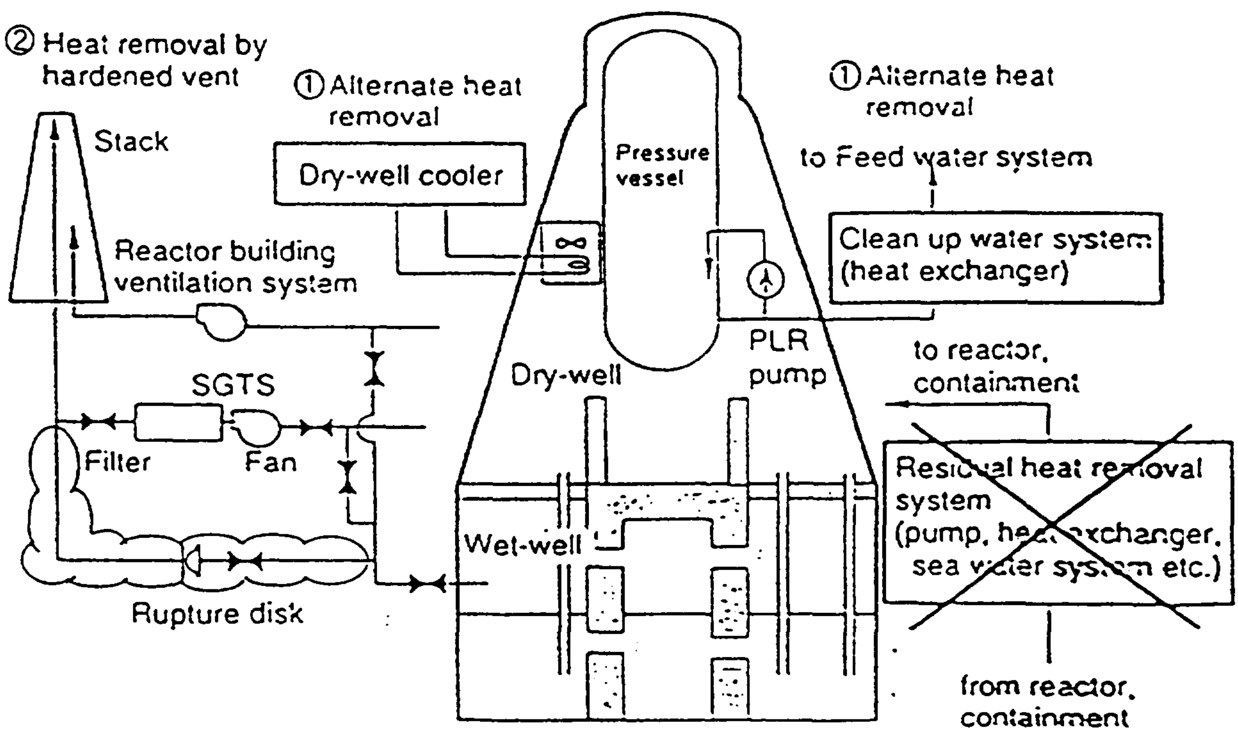

Fig. 4(b) Equipment for containment heat removal [1]

Table 2. Summary of Sequence Categorization by Watanabe et al. [7]

Sequence Group	Core Damage Sequences	Dominating Failure Mode	Effectiveness of Mitigative Measures Considered
Interfacing System LOCA (Group 1)	V-sequence	Containment Bypass (Pre-stage for Core-melt)	Not Applicable
ATWS with High Pressure Injection Available (Group 2)	TC,S2C,S1C,AC	Overpressurization (Pre-stage for Core-melt)	Not Applicable
Loss of Long-term Containment Heat Removal (Group 3)	TW,TQW,TQU1W,TPW, TPQW,TPQU1W,S2W, S2QW,S2QU1W,S1W,AW	Overpressurization (Pre-stage for Core-melt)	Containment Venting (Pre-stage for Core-melt)
Station Blackout (Group 4)	TB,TPB,TBU,TPBU	Overpressurization Over-temperature (Long-term Prog.Stage) Direct Containment Heating (Debris Exit Stage)	Off-site Power Recovery (Core-melt Prog.Stage) (Long-term Prog.Stage)
Loss of Coolant Injection with Reactor Not Depressurized (Group 5)	TQUX,TPQUX,S2QUX, S1UX,TCU,S2CU,S1CU	Overpressurization (Core-melt Prog.Stage)	Reactor Depressurization (Core-melt Prog.Stage) Debris Cooling by RHR (Long-term Prog.Stage)
Loss of Coolant Injection with Reactor Depressurized (Group 6)	TQUV,TPQUV,S2QUV, S1UV,AUV	Overpressurization Over-temperature (Long-term Prog.Stage)	Not Applicable (It would be possible to repair the failed RHR components in Long-term Prog.Stage)
Loss of Short-term Containment Heat Removal (Group 7)	TQUW,TQUV1W,TPQUW, TPQUV1W,S2QUW, S2QUV1W,S1UW,S1UV1W, AUW,AUV1W	Overpressurization Over-temperature (Long-term Prog.Stage)	Not Applicable (It would be possible to recover the containment cooling in Long-term Prog.Stage)
Reactor Pressure Vessel Rupture (Group 8)	R-sequence	Overpressurization (Pre-stage for Core-melt)	Debris Cooling by RHR (Long-term Prog.Stage)

Table 3. Accident Management Measures of the Representative BWR

Function	Accident Sequence :Corresponding Group	Up: AM Measures Adopted This Time Lo: AM Measures Implemented
Reactor Shutdown Function	-ATWS : G2 =Overpressure after ATWS : G2	>Alternative reactivity control
		>Manual operation of boric acid injection system
Water Injection into Reactor and Containment	-Failure of high pressure water injection and depressurization : G5 -Failure of high and low pressure injection : G6 =Over-heating of penetration (steam explosion) : G7	>Automatic depressurization by the signal of transient (low reactor water level) >Alternative water injection (water injection to reactor and containment using pumps with condenser makeup water system or fire fighting system)
		> Alternative water injection (>Water injection to reactor through feed water system or control rod drive hydraul. system >Water injection to reactor and containment by sea water system)
Heat Removal from Containment	-Failure of decay heat removal : G3, G7 =Overpressure by steam (decay heat) : G2- G8	>Drywell cooler >Hardened vent
		>Manual operation of containment spray system
Support of Safety Function	-Loss-of-electric-power supply : G4 =Direct containment heating : G4	>Supply of electric power (cross-tie of power supply in low voltage from adjacent plant and high voltage from D/G for HPCS)
		>Supply of electric power (cross-tie of power supply in high voltage from adjacent plant)

IV. Summary

Through the review of the source term analysis study, the sequence grouping study and recent experimental knowledge in severe accident phenomena, the relativity of the importance of AM measures adopted for BWRs was discussed. Some proposals to be considered in emergency operation procedures were made in the viewpoint of the reduction of risk as the product of CDF and source term release consequences.

This examination also addressed the area where further clarification in the phenomena for reducing uncertainty is required to monitor accident progression in a reliable way and for taking appropriate measures for mitigating consequences. Although the present analytical models and results include some uncertainty, general insights obtained through this review are considered to be valid.

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