Circumstances and Present Situation of Accident Management Implementation in Japan

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1. Introduction

The beginning of accident management (AM) implementation in Japan can be traced back to 1992. Through in-depth researches and discussions regarding the severe accidents and AM, the Nuclear Safety Commission (NSC) of Japan issued a decision entitled "Accident Management as a Measure against Severe Accidents at Power Generating Light Water Reactors"[1] in May 1992. In this decision, the NSC strongly recommended the regulatory body and utilities to introduce AM measures to nuclear power plants (NPPs), although sufficient safety level has been maintained by current safety systems at operating NPPs.

Responding to the decision issued by the NSC, the Ministry of International Trade and Industry (MITI), which was the regulatory body of NPPs at that time, encouraged the utilities to establish AM implementation plans using benefit of insights obtained from PSA in July 1992. With an investigation period of one year, the utilities submitted their plans of AM implementation to MITI in March 1994. MITI reviewed these utilities' plans from the technical point of view and made a report of "AM for Light Water NPPs"[2] in October 1994, in which MITI recommended the utilities to undertake the AM implementation plans with preparation of AM operating procedures and establishment of administrative framework toward 2000.

The utilities completed implementation of AM to their NPPs by February 2002 and reported to the Nuclear and Industrial Safety Agency (NISA), which is the new regulatory body of NPPs founded in

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[1] Nuclear Safety Commission, "Accident Management as a Measure against Severe Accidents at Power Generating Light Water Reactors," May 28, 1992

[2] Ministry of International Trade and Industry, "Accident Management for Light Water Nuclear Power Reactors," October, 1994

January 2001. In addition, the utilities submitted evaluation of effectiveness of AM measures for eight representative BWR and PWR plants to NISA. NISA reviewed those results with the assistant of the Japan Nuclear Energy Safety Organization (JNES) and confirmed validity of them. The results of evaluation performed by JNES were presented in the previous SAMM conference held in 2001.[3, 4]
Meanwhile, NISA recognized it was also important to evaluate effectiveness of AM measures for NPPs other than eight representative plants and requested utilities to perform evaluation on them. Following this request, the utilities performed evaluation of effectiveness of AM measures for each NPP and submitted the results to NISA as "PSA Evaluation Report following AM Implementation" in March 2004. NISA also reviewed these results with the help of JNES and confirmed appropriateness of these evaluation. This paper presents the results of this review.
Besides fifty-two operating NPPs, AM have been studied and implemented to four newly constructed NPPs up to now. This paper also presents current situation of AM implementation for these newly constructed NPPs.

2. Accident management measures and their effectiveness at the representative plants

The utilities selected AM measures focusing on essential safety functions of NPPs. Specifically, reactor shutdown, coolant injection to the reactor vessel and the containment vessel, heat removal from the containment vessel, and supporting function to the safety systems are chosen as the four essential functions for BWR and then relevant AM measures were selected for each safety function. Table 1 summarizes those AM measures adopted for BWRs. Similarly, reactor shutdown, core cooling, confinement of fission products, and supporting function to the safety systems are chosen as the essential safety functions for PWRs and AM measures were selected, which are shown in Table 2.
Although, similar AM strategies are used for BWR or PWR, respectively, regardless of varieties of plant design, specific AM plans depend on the design of plant as well as the preference of the utilities. For example, as an enhancement of electric power supply, which is categorized as one of AM measures of supporting functions to the safety systems, it is realized in various ways for BWRs as follows;

- Electric power from the adjacent unit is used by connecting safety buses of both units, in case of both offsite power and emergency diesel generators (EDG) being unavailable simultaneously.

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[3] M. Kajimoto et. al., "Evaluation of Technological Appropriateness of the Implemented Accident Management Measures for BWR by Level 1 and Level 2 PSA Methods," Workshop on the Implementation of Severe Accident Management Measures, September 2001

[4] H. Takahashi et. al., "Evaluation of Technological Appropriateness of the Implemented Accident Management Measures for PWRs by Level 1 and Level 2 PSA Methods," Workshop on the Implementation of Severe Accident Management Measures, September 2001

- For the single-unit site, HPCS-DG is used as an alternate AC source to another safety bus.
- In case that one EDG is shared by two adjacent units, an additional EDG is installed so that each unit is equipped with two dedicated EDGs.

With regard to PWRs, following three alternatives are used for ECCS recirculation;

- Cross-tie between the low pressure injection line and the containment vessel spray injection line, which can make the low pressure recirculation using a containment spray pump in case of ECCS recirculation failure
- Alternative recirculation pump put in place in the recirculation sump
- A redundant valve to the recirculation sump isolation valve

Effectiveness of AM measures is assessed using level 1 and level 2 PSA. Reflecting highly standardization of plant designs in Japan and considering commonality of them, all BWR plants and all PWR plants are divided into eight groups, four for BWRs and another four for PWRs, and then PSA was performed for a representative NPP in each group. Categorization of BWRs and PWRs as well as their safety features are presented in Table 3 and 4, respectively. Studies on effectiveness of AM measures were conducted both by utilities and JNES. The result performed by JNES were presented previous ISAMM meeting held in 2001.

3. Effectiveness of accident management measures of individual plant

Upon the request from NISA, the utilities performed evaluation of effectiveness of AM measures for individual plant other than the eight representative plants, and submitted the results of these evaluation to NISA in March 2004 as “PSA Evaluation Report after AM Implementation.” NISA reviewed these reports with assistance from JNES. In the course of this effort, JNES made an investigation focusing on the large differences in the core damage frequencies (CDFs) between individual plant and the representative plant in the same group. In addition, PSA models of the representative plant were modified and sensitivity studies were done in order to clarify the causes of these large differences. The results of studies on the effectiveness of AM measures of individual plant are shown below.

3.1 BWR plants

Figure 1 shows the comparison of CDFs of individual BWR plant before and after AM measures implementation. Those values are normalized by CDF of type C (BWR5) representative plant before AM implementation. Figure 1 also shows reduction ratio of CDF by AM measures in each plant. This value is defined by the ratio of CDF after AM implementation to CDF before AM implementation in each plant. Similarly, Figure 2 shows the comparison of the containment functional failure frequencies (CFFs) of individual NPP and reduction ratios of CFF. These CDFs and CFFs are the results evaluated by the utilities.

When comparing CDFs among plant types, CDFs of type D plants before AM implementation are much less than CDFs of type A, type B and type C plants, while the reduction ratios by AM of type D plants are greater than the ratios of other plant types. For type D plants, the alternate rod insertion (ARI) and recirculation pump trip functions, which are designated as AM measures for the other types of plants, are adopted in the basic design of the plant for the purpose of additional reactor shutdown. In addition, highly redundant systems are used for the coolant injection and residual heat removal functions in the basic design of type D plants. These factors make CDFs before AM implementation much smaller than CDFs of other types. On the other hand, because additional reactor shutdown measures are already installed and additional AM measures are considered unnecessary for the highly redundant coolant injection and residual heat removal function, overall reduction ratios of CDF by AM measures of type D plants are greater than the other.

Some differences can be found among CDFs and CFFs of individual NPP before AM implementation and the reduction ratios by AM measures even in the same plant type. This is because there are some small differences in the design and operation of plants and AM measures adopted are sometimes unique to individual plant. One typical example of this difference is the design and operational of CCWS. While there are a lot of plants which belong to type C, they can be further divided into three subgroups. The plants in the first subgroup have a similar design of CCWS to the representative plant of the group. The design and operation of CCWS in the second subgroup, such as Kashiwazaki-Kariwa-2, and the third subgroup, such as Hamaoka-3, is not same as the first subgroup. This difference yield low unavailability of ECCS and, thus, smaller CDFs of the plants. On the other hand, an example of difference in AM measures can be found in Onagawa-1 in type B. In Onagawa-1, redundant CCWS pumps are installed as an AM measure, which makes a large reduction of CDF after AM implementation comparing the other plants in type B.

Because the differences in CFFs chiefly come from the differences in CDFs, thorough investigation on the differences of CFFs are not performed.

Reduction ratios range from 0.02 to 0.6 for CDFs and from 0.01 to 0.08 for CFFs. The effectiveness of AM measures can well be confirmed.

3.2 PWR plants

Figure 3 shows the comparison of CDFs of individual PWR plant before and after AM implementation. These values are normalized by CDF of type D (four-loop PWR with large dry containment vessel) representative plant before AM implementation as is in the BWR case. Figure 3 also shows the reduction ratios of CDF by implementing AM measures. Similarly, Figure 4 shows the comparison of CFFs of individual NPP and the reduction ratios. These results are evaluated by the utilities as well.

Some differences can be observed among CDFs of individual NPP and their reduction ratios. They are originated from the difference in the plant design or AM measures adopted, as discussed in the BWR case.

An example of the variation of the plant design which causes the difference in CDFs and CFFs is ECCS system design. CDF of Ikata-3 in type B group is much smaller than CDFs of other NPPs in the same group. In Ikata-3, the high pressure injection (HPI) pumps do not require boosting by the low pressure injection (LPI) pumps during ECCS recirculation mode while the other NPPs in the same group require boosting by LPI pumps. This plant design of Ikata-3 leads to smaller overall unreliability of ECCS during recirculation mode and thus smaller CDF of the plant. Same situation also can be found in the type D plants. Amongst type D plants, Turuga-2 is the only one plant which needs the boosting by LPI pump to HPI pump and, therefore, CDF of Turuga-2 is higher than CDFs of the other plants in type D group.

Another example can be seen in type A group. ECCS switch-over from the injection mode to the recirculation mode is done automatically for Tomari-1 and 2, while this operation is done by operator manually in other NPPs of type A group. This design difference makes CDFs of Tomari-1 and 2 smaller than CDFs of the other plants in the type A group.

In contrast, an example of the variation of AM measures which causes the differences in CDFs can be found in a measure of alternative ECCS recirculation. The CDF reduction ratio of Turuga-2 in type D group is smaller than the reduction ratios of other plants in the same group. The cross-tie between LPI line and CSI line is adopted as an AM measure for the alternative ECCS recirculation in type D plants generally, and this AM measure is applied to only one train for the plants other than Turuga-2. On the other hand, this AM measure is applied to both two trains of LPI and CSI at Turuga-2, and thus CDF reduction ratio of this plant is lower than the other.

The differences in CFFs chiefly come from the differences in CDFs and a thorough investigation is not performed for CFFs.

Although there are some differences in CDFs and CDF reduction ratios among plants according to the difference in the design of plants and AM measures adopted as mentioned above, reduction ratios of CDF and CFF lie in the range of 0.3 to 0.6 and 0.1 to 0.6, respectively, and the effectiveness of AM measures can well be confirmed.

In general, a large variation of CDFs can be found among the types of BWRs even before AM implementation comparing to CDFs of PWRs. This is because the basic design concept of ECCS is basically similar even in the different types of the plants for PWRs, whereas it depends on the types of the plants for BWRs. For PWRs, necessity of boosting by LPI pumps to HPI pumps during recirculation mode has a large effect. In addition, there is a tendency that the reduction ratios by AM measures are large for BWR plants.

4. Implementation of accident management measures for the newly constructed NPPs

Implementation of AM measures to the operating fifty-two NPPs had been completed by 2002 involving plant modifications. Meanwhile, for the newly constructed NPPs which begin commercial operation in 2002 or later, it is recommended by the NSC to establish an AM implementation plan and to submit the plan to the regulatory body for review soon after the detailed design of the plant was accomplished, and to complete AM implementation before the first fuel loading to the core.[5] According to this process, AM measures for Higashidori-1, Hamaoka-5, Shika-2, and Tomari-3 have been investigated, reported to NISA and reviewed by NISA and the NSC until now.

AM implementation plan and evaluation of effectiveness of AM measures for Tomari-3 were reported to NISA last year and they were reviewed by NISA and the NSC until the beginning of this year.[6, 7] Similar AM measures to the operating plants shown in Table 4 are used for this plant, but some of them, i.e. train separation of CCWS actuated by a low CCW surge tank level signal against loss of CCWS function and redundant intake lines from CV recirculation sump, are incorporated as a part of basic design. The reduction ratio of CDF and CFF taking a credit of AM measures including the measures considered as the basic design described above are 0.4 and 0.1, respectively. Although Tomari-3 belongs to type B group in Table 2, the design of the plant and the results of CDF, CFF, and the reduction ratio of CDF and CFF are not close to those of the representative plant of the group, rather close to those of Ikata-3.

In AM review, possibility of adverse effects on the essential safety functions of the plant and conformance to the basic requisites stipulated by NISA are examined in addition to the evaluation of the effectiveness of AM measures using PSA. These reviews are performed for the newly constructed plants in a similar way to the operating plants.

Specifically, the adverse effects on the essential safety functions of the plant are reviewed from the following points;

- Conformance to the safety guidelines of NPPs

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5 Nuclear Safety Committee, “Future Policy on Implementation of Accident Management for Light Water Nuclear Power Reactor Facillities,” Ocotrber 20, 1997

6 Nuclear and Industrial Safety Agency, “Report for Studies on Accident Management of Hokkaido Electric Power Company Tomari Nuclear Power Plant Unit No.3,” October 6, 2008

7 Nuclear Safety Committee, “Implementation of Accident Management for Hokkaido Electric Power Company Tomari Nuclear Power Plant Unit No.3,” January 19, 2009

Conformance to the safety design guidelines, the safety analysis guidelines, and the seismic design guidelines was reviewed to check if there is no adverse effect by implementing AM measures.

- Adverse effect on the safety systems

  To check if there is no adverse effect on redundancy, independence, and essential functions of the safety systems in case of modification of these systems being made in order to incorporate AM measures.
- Effect to the results of safety analyses

  To check if there is no effect to the results of safety analyses which are reviewed in the plant licensing in case of any failure in AM features being assumed in normal operation.

With regard to AM basic requisites, the following five points are reviewed by NISA;

- AM enforcement structure (organization, roles of staffs)
- Facilities and equipments (communication system, plant information transmission system, data acquisition system like radiation monitors, emergency dose prediction system, manuals (operating manuals and AM guidelines))
- Knowledgebase of AM
- Notification and communication
- Training of staffs

The results of AM review for Tomari-3 by NISA were reported to the NSC in October, 2008. Upon receiving this report, the NSC reviewed the results and corroborated adequacy of AM measures for Tomari-3. The NSC also raised the followings as the future issues of AM implementation;[8]

- Reconsideration of the treatment of AM in the nuclear safety regulatory framework
- Efficient scheme of AM development
- Improvement of quality to confirm the effectiveness of AM measures
- Points of concern to use PSA
- Consideration of external events
- Contribution to grow up the security of public to the nuclear safety

5. Concluding Remarks

Introduction of AM measures to the Japanese NPPs began with the decision by the NSC issued in 1992, followed by the study of AM measures for the operating plants. Modifications of the plants as well as the establishment of AM enforcement framework and the preparation of the relevant AM

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[8] Nuclear Safety Committee, "Future Issues for Implementation of Accident Management," January 19, 2009

procedures have been completed by 2002. The effectiveness of AM measures is evaluated by utilities and results of these evaluations are reported to the regulatory body. The effectiveness of AM measures was confirmed through the reviews on these reports performed by the regulatory body.

Meanwhile, for the newly constructed NPPs, it is recommended to establish AM measures and to complete installation of AM measures by the first fuel loading to the core of the plant. Up to now, AM plans for four newly constructed plants are studied and reviewed in this process. In some cases, AM measures are incorporated as a part of basic design of the plant, reflecting the outcomes achieved by the AM studies for the operating plants.

In the latest AM review, the NSC pointed out some future issues for AM implementation; i.e. reconsideration of the treatment of AM in the nuclear safety regulatory framework, improvement of the quality of PSA, AM for external events and others.

Table 1 Reactor types and safety systems (BWR)

		type A	type B	type C	type D
Type of reactor		BWR2, 3	BWR4	BWR5	ABWR
Type of containment vessel		MARK-I	MARK-I	Mod. MARK-I MARK-II Mod. MARK-II	RCCV
Name of plant (Bold : representative plant)		**Fukushima1-1** Turuga-1	Onagawa-1 **Fukushima1-2** Fukushima1-3, 4, 5 Hamakoka-1, 2 Shimane-1	Onagawa-2, 3 Fukushima1-6 **Fukushima2-1** Fukushima2-2, 3, 4 Tokai-2 Kashiwaza -Kikariwa-1, 2, 3, 4, 5 Hamaoka-3, 4 Shika-1 Shimane-2	**Kashiwazaki** **-Kariwa-6** Kashiwaza -Kikariwa-7
Safety systems					
Reactor scram		CRDHS SLCS	CRDHS SLCS	CRDHS SLCS	CRDHS SLCS ARI FMCRD
ECCS	High press.	HPCI IC(2 trains)	HPCI RCIC	HPCS RCIC	HPCF(2 trains) RCIC
	Low press.	CS(2 trains)	CS(2 trains) LPCI(2 trains)	LPCS LPCI(3 trains)	LPFL(3 trains)
Containment heat removal		SHC(2 trains) CCS(2 trains)	RHR(2 trains)	RHR(2 trains)	RHR(3 trains)

RCCV: Reinforced concrete containment vessel
Fukushima1: Fukushima Site No.1
Fukushima2: Fukushima Site No.2
CRDHS: Control rod drive hydraulic control system
SLCS: Standby liquid control system
ARI: Alternate rod insertion
FMCRD: Fine motion control rod drive
HPCI: High pressure core injection (system)
IC: Isolation condenser
RCIC: Reactor core isolation cooling (system)
HPCF: High pressure core flooder
CS: Core spray (system)
LPCI: Low pressure coolant injection (system)
LPFL: Low pressure flooder
SHC: Shutdown reactor cooling (system)
CCS: Containment cooling system

Table 2 Reactor types and safety systems (PWR)

		type A	type B	type C	type D
Type of reactor		Two-loop	Three-loop	Four-loop	Four-loop
Type of containment vessel		Large dry SSCV	Large dry SSCV	Ice condenser	Large dry PCCV
Name of plant (Bold: representative plant)		Tomari-1, 2 Mihama-1, 2 Ikata-1 **Ikata-2** Genkai-1, 2	Mihama-3 Takahama-1, 2 **Takahama-3, 4** Ikata-3 Sendai-1, 2	**Ohi-1, 2**	Turuga-2 **Ohi-3, 4** Genkai-3, 4
Safety systems					
Reactor protection system		2 trains, Relay type	2 trains, SSPS	2 trains, SSPS	4 trains, SSPS
ECCS	High press. injection (No. of pumps)	2(High press. injection pump), Boosted by LPI pump during recirculation mode	3(Charging SI pump), Boosted by LPI pump during recirculation mode	2(Charging SI pump), 2(High press. injection pump), Boosted by LPI pump during recirculation mode	2(High press. injection pump)
	Low press. injection (No. of pumps)	2	2	2	2
	No. of accumulators	2	3	4	4
Auxiliary feedwater No. of M/D pumps No. of T/D pumps		 2 1	 2 1	 2 2	 2 1
Containment vessel spray (No. of pumps)		2	2	2 with 2 RHR spray pumps	2

SSCV: Steel containment vessel
PCCV: Pre-stressed concrete containment vessel
SSPS: Solid state protection system
ECCS: Emergency core cooling system
M/D: Motor-driven
T/D: Turbine-driven
RHR: Residual heat removal (system)

Table 3 Accident management measures (BWR)

<table>
<tr><th>Safety function</th><th>Purpose</th><th>Accident management measures to prevent core damage</th><th>Accident management measures to mitigate core damage</th></tr>
<tr><td>Reactor shutdown</td><td>Alternate reactivity control</td><td>● ARI(Control rod insertion by high reactor pressure or low reactor level)
● RPT(same signal)
*These signals are independent to current scram signals or ECCS actuation signals
ABWR adopts alternate reactivity control as the basic design.</td><td>-</td></tr>
<tr><td rowspan="2">Coolant injection to reactor and containment vessel</td><td>Automatic reactor depressurization</td><td>● ADS automatic actuation by low reactor level(L-1) with delay (except BWR2,3 and ABWR)</td><td>-</td></tr>
<tr><td>Alternate coolant injection</td><td colspan="2">● MUWC
● Fire extinguish system (except Onagawa), Filtrate water system (Onagawa)</td></tr>
<tr><td rowspan="3">Heat removal from containment vessel</td><td>Hard vent system</td><td colspan="2">● Hard vent system</td></tr>
<tr><td>Alternate cooling</td><td>-</td><td>● Alternate cooling by dry-well cooler or CUW</td></tr>
<tr><td>Recovery of RHR</td><td colspan="2">● Recovery of RHR</td></tr>
<tr><td rowspan="2">Supporting function</td><td>Electric power supply</td><td colspan="2">● Electric power supply from adjacent unit on 6.9 kV and 480 V (Fukushima Site No.1, Fukushima Site No.2, Kashiwazaki-Kariwa, Tokai-2, Tsuruga-1) or 460 V (Other Plants)
● Electric power supply from HPCS-DG (Single-unit site: Shika-1 and Tokai-2)
● Installation of dedicated emergency diesel generators (Fukushima Site No.1)</td></tr>
<tr><td>Recovery of emergency diesel generator</td><td colspan="2">● Recovery of emergency diesel generator</td></tr>
</table>

ARI: Alternate rod insertion

RPT: Recirculation pump trip

MUWC: Makeup water system condensated

CUW: Reactor water cleanup (system)

Table 4 Accident management measures (PWR)

Safety function	Purpose	Accident management measures to prevent core damage	Accident management measures to mitigate core damage
Reactor shutdown	Reactor shutdown	● Diversity of emergency secondary cooling (use of main feedwater in case of ATWS)	-
Core cooling	ECCS injection	● Use of LPI with depressurization by turbine bypass valves	-
	ECCS recirculation	● Alternative recirculation ➢ Tie-line between LPI and CSI ➢ Alternate recirculation pump ➢ Recirculation sump isolation valve bypass line	-
	Isolation of coolant leakage	● Cooldown and recirculation	-
Confinement of radioactive materials	Heat removal from containment vessel	● Natural convection heat removal ➢ Use of non-safety CV heat removal system ➢ Outside CV spray	● Natural convection heat removal ● Coolant injection to CV ● Forced depressurization of primary system ● Hydrogen igniter (Ice condenser CV plant)
Supporting function	Supporting function	● Alternate component cooling ➢ Air conditioning system ➢ BOP CCWS ➢ CV cooling system ➢ Fire extinguish system	-
		● Electric power supply from adjacent unit ➢ Connection between high voltage buses ➢ Connection between low voltage buses	-

LPI: Low pressure injection
CSI: Containment spray injection
BOP: Balance of plant

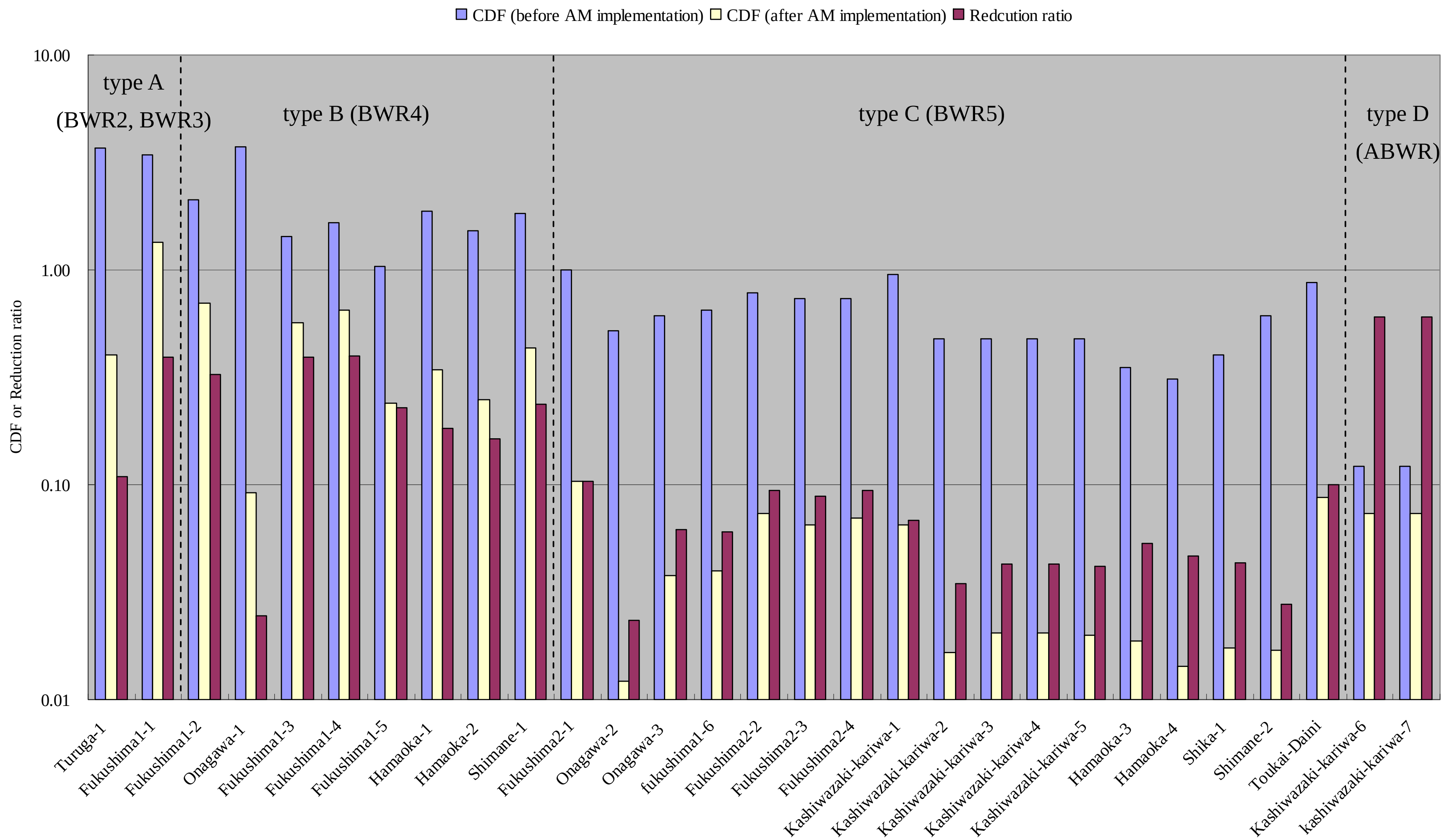

Figure 1 Comparison of core damage frequencies before and after AM implementation (BWR)

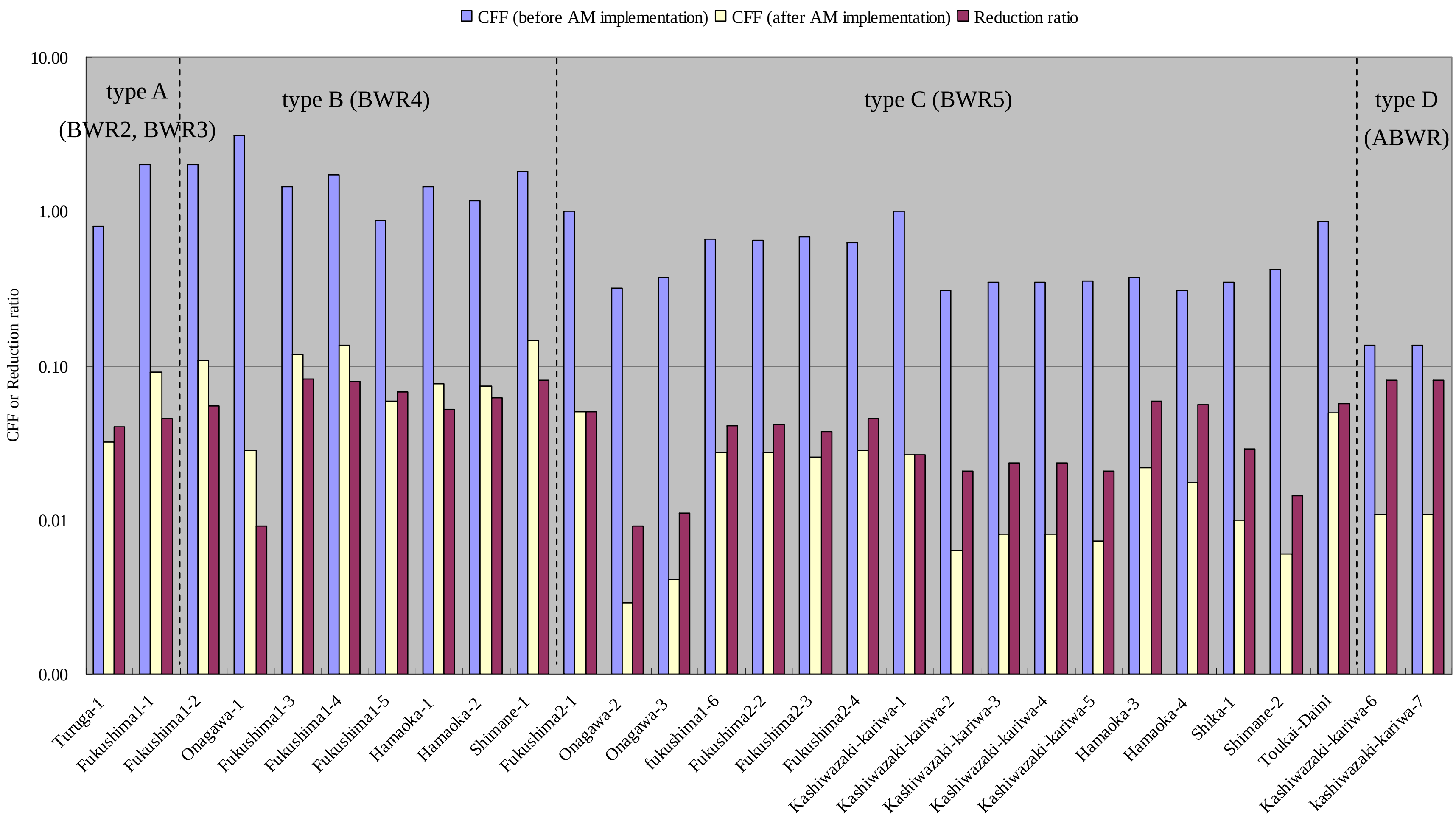

Figure 2 Comparison of containment functional failure frequencies before and after AM implementation (BWR)

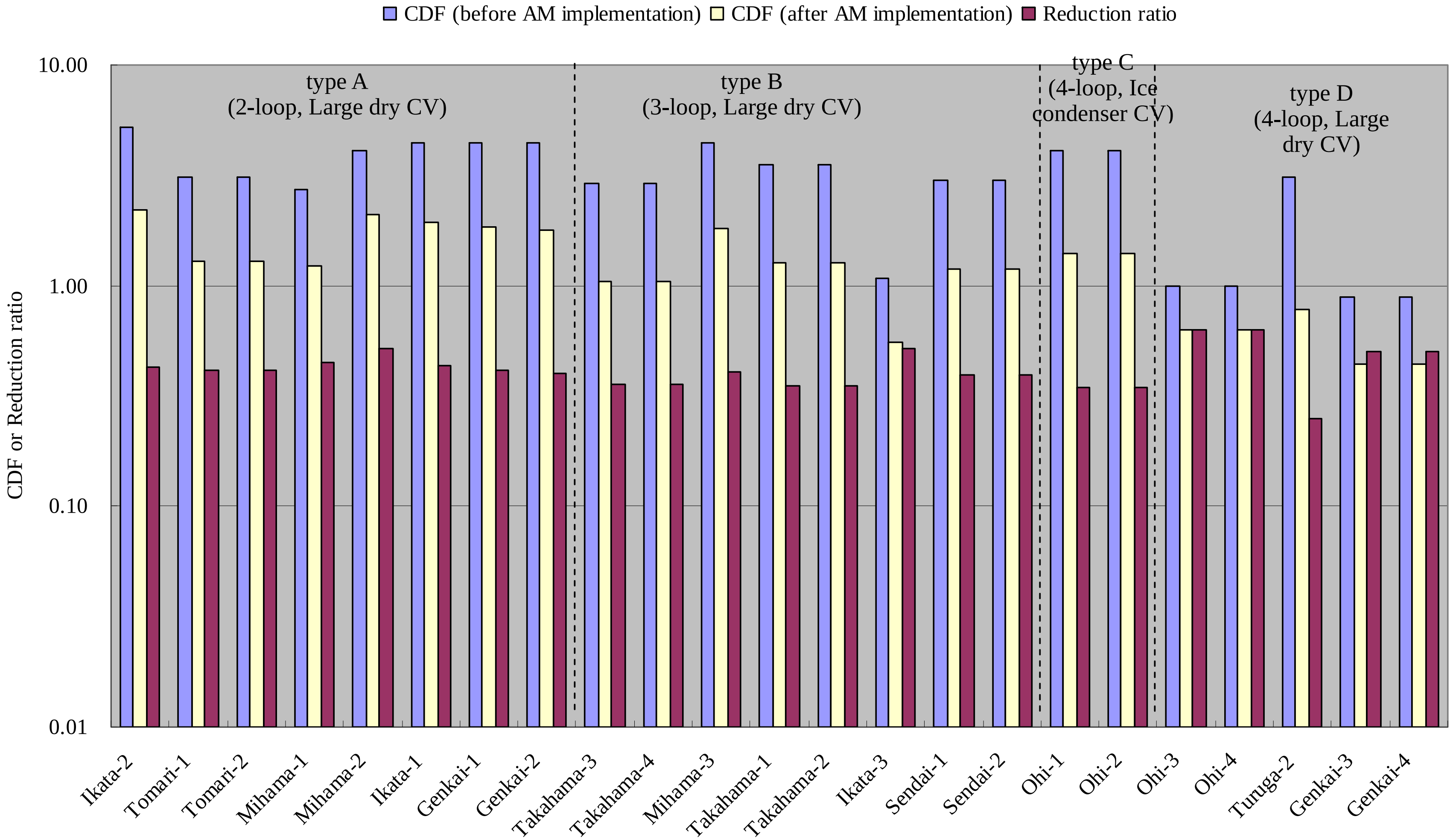

Figure 3 Comparison of core damage frequencies before and after AM implementation (PWR)

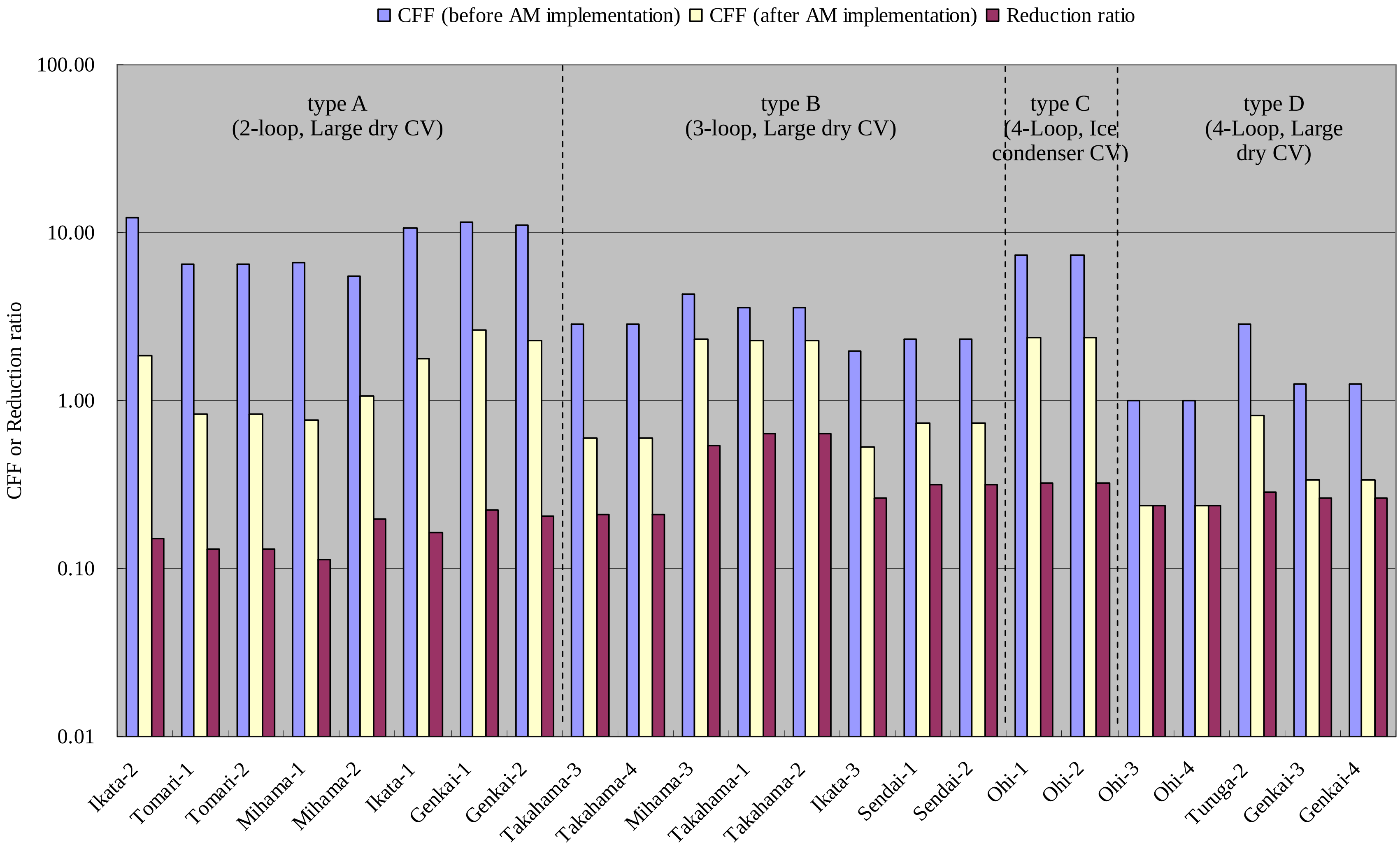

Figure 4 Comparison of containment functional failure frequencies before and after AM implementation (PWR)