International Review on Safety Requirements

for the Prototype Fast Breeder Reactor “Monju”

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Japan Atomic Energy Agency

Abstract

In response to the lessons learned from the serious nuclear accidents at the TEPCO's Fukushima Daiichi Nuclear Power Plants, an advisory committee, which was set up by the Japan Atomic Energy Agency, issued the report "Safety Requirements Expected to the Prototype Fast Breeder Reactor Monju" taking into account the SFR specific safety characteristics in July 2014.

The report was reviewed by the leading international experts on SFR safety from five countries and one international organization in order to obtain independent and objective evaluation. The international review comments on each subsection were collected and compiled, and then a summary of results was derived through the discussion at the review meeting and individual feedbacks.

As a result the basic concept for prevention of severe accidents and mitigation of their consequences of Monju is appropriate in consideration of SFR specific safety characteristics, and is in accordance with international common understanding.

Contents

1. Introduction

Since the Great East Japan Earthquake on March 11, 2011 and the subsequent serious nuclear accident at the TEPCO's Fukushima Daiichi Nuclear Power Plants (hereinafter referred to as "1F Accident"), almost all the nuclear reactors in Japan have stopped their operation. Every reactor facility needs to go through a licensing procedure by a newly founded Nuclear Regulation Authority (NRA) based on new regulatory standards established taking lessons learned from the 1F Accident. In the New Regulatory Requirements for Commercial Power Reactors enforced by NRA in July 2013, stricter considerations are given to severe accidents and natural hazards. In addition to the Requirements for commercial light water reactors (LWRs), the NRA also enforced the Requirements for power-generating reactors under research and development, which are applied to Japan's prototype sodium-cooled fast reactor (SFR) facility Monju.

The latter regulatory requirements to be applied to Monju are said to be further revised by NRA, by reflecting the public comments which assert to consider safety characteristics of SFRs more carefully. Under such circumstances, the Japan Atomic Energy Agency (JAEA) set up an advisory committee on safety requirements for Monju, in order to establish the safety requirements for Monju independently, taking into account the severe accidents from the viewpoints of scientific and technical insights. The committee consists of the experts on fast reactor technology and safety research. The committee issued a report titled "Safety Requirements Expected to the Prototype Fast Breeder Reactor Monju" (hereinafter called as "the Report") and submitted it to NRA in July 2014.

The Report has been reviewed in detail by both the domestic and international experts on SFR safety, in parallel, in order to obtain independent and objective evaluation and also to confirm whether the basic concept for SFR safety and safety requirements written in the Report is in accordance with international common understanding. The present report describes the process and results of the review by the international experts.

2. Method of International Review

2.1 Reviewers

The review was entrusted to the leading specialists in SFR safety in the international community. As listed in Appendix 1, the nine reviewers selected this time are from China, France, Korea, Russia, the United States and the European Union, and they are responsible for the SFR safety design and evaluation in their own national SFR programs or represent international organization. All of them are active and key members of international cooperation programs such as in Generation IV International Forum (GIF) and International Atomic Energy Agency (IAEA).

2.2 Procedure of the Review

(1) Scope of the Review

The Report (in Japanese) is over 270 pages and describes the entire safety concepts and approaches for Monju, covering: safety characteristics of SFR and safety approach (Chapter 2), design basis accidents (Chapter 3), severe accidents (Chapter 4), significant external events (Chapter 5), specific considerations of lesson learned from the 1F Accident (Chapter 6) and safety requirements (Chapter 7), To make the international review more efficient and effective, the two important chapters of the Report, Chapter 4 (Concept for Prevention of Severe Accidents and Mitigation of Consequences) and Chapter 7 (Concept for Securing Safety of SFR), were selected for detailed evaluation by the reviewers . The excerpt of the Report translated in English (see Appendix 2) was distributed to the reviewers in advance and the results of the review were compiled for further discussions at the review meeting (see Section 2.3) and upon drafting of the present report.

(2) Rating by Reviewers

The reviewers were asked to evaluate the individual contents of the Report item by item and to rate with the ranking from 1 to 5 as shown below. Here each item for rating corresponds to a section or sub-section of the Report. A complete list of the review items is given in the Review Sheet (see Table 1). The reviewers are also asked to describe the reasons of their ratings and, more importantly, scientific and technical comments based on their own experience and expertise.

Ranking

5: Completely supportive

4: Rather supportive

3: Neutral

2: Rather negative

1: Completely negative

The comments by the reviewers are given in Appendix 3 and the major supportive and negative comments were compiled as a preliminary summary of the review, which was then further refined and revised through the continued refinement process including the Review Meeting (see Section 2.3). The resulted final version of the summary, agreed among the reviewers, is given later in Chapter 3 of this report. The score sheet of the ratings by the reviewers is shown in Appendix 4. The average scores are between 4 and 5, and this means the Report has been evaluated, by the international experts, to be very supportive. It is inappropriate, however, to simply conclude an international consensus is reached. Nevertheless the scientific and technical inputs from the reviewers are essential and helpful.

2.3 Review Meeting

A review meeting was held in Tokyo, Japan on May 13, 2015 in collaboration with MEXT (Ministry of Education, Culture, Sports, Science and Technology), in which the international reviewers are invited to participate. Although only a few of the reviewers were actually join the meeting, all the major comments compiled in the preliminary summary of the review were put on a table for detailed discussions. The contents of the summary were either approved, deleted or refined. A revised version of the summary was distributed to the reviewers for their comments *(as a draft version of this report)* and the final version of the summary, which has been agreed among the reviewers, is presented in Chapter 3.

Table 1　Review Sheet

Ranking	5	Completely supportive
	4	Rather supportive
	3	Neutral
	2	Rather negative
	1	Completely negative

No.	Chapter/ Section	Check Item	Score	Reason of the choice and/or comments
1	4.1	Basic concepts and procedure of assessments on the beyond design basis accidents corresponding to the level 4 of the defense-in-depth concept		
2	4.2.1	Methods and results of selecting the accident sequence groups extracted among ones induced by internal events of the plant		
3	4.2.1	Methods and results of selecting the accident sequence groups from ones caused by earthquakes and tsunamis		
4	4.2.1	Candidates for postulated initiators specific to SFRs to be examined		
5	4.2.2	Postulated accident sequence groups in consideration of the features of EVST in Monju		
6	4.2.2	Postulated accident sequence groups in consideration of the features of spent fuel pool in Monju		
7	4.2.3	Postulated accident sequence groups for the reactor at shutdown state		
8	4.2.4	Methods and results of selecting the accident sequence groups for assessments on effectiveness of measures for ensuring containment function		

9	4.2.4	Methods and results of selecting the representative accident sequence groups for ATWS events		
10	4.2.4	Methods and results of selecting the representative accident sequence groups for LOHRS events		
11	4.3.1	Concepts for preventing significant core damage		
12	4.3.2	Concepts for preventive measures against spent fuel damage		
13	4.3.3	Concepts for preventive measures against damage of fuels stored in the core during reactor shutdown		
14	4.4.1	Basic concepts (Philosophy)		
15	4.4.1	Most likely progress of events identified for ATWS (initiating phase, transition phase, PAMR and PAHR phase)		
16	4.4.1	Methods and results of assessing the influences of uncertainties in dominant factors on the most likely progress of events for ATWS (initiating phase, transition phase, RV response phase, PAMR and PAHR phase)		
17	4.4.2	Basic concepts (Philosophy)		
18	4.4.2	Concepts for ensuring containment function during LOHRS-type events		
19	4.4.2	Summarized results of concepts for ensuring containment function in SFRs		
20	4.4.2	Concepts for instrumentation and monitoring during SAs		
21	7	Requirement 1		
22	7	Requirement 2		
23	7	Requirement 3		
24	7	Requirement 4		
25	7	Requirement 5		
26	7	Requirement 6		

27	7	Requirement 7		
28	7	Requirement 8		
29	7	Requirement 9		
30	7	Requirement 10		
31	7	Requirement 11		
32	7	Requirement 12		
33	7	Requirement 13		
34	7	Requirement 14		
35	7	Requirement 15		
36	7	Requirement 16		
37	-	Any opinions or comments on safety requirements for SFRs as a whole (No choice is needed.)		

3. Results of International Review

The final version of the summary, agreed among the reviewers, is presented in this chapter as the results of the international review. The review was focused, as already explained, on Chapters 4 and 7 of the Report, “Safety Requirements Expected to the Prototype Fast Breeder Reactor ‘Monju’”, which was sent to the nine international experts in advance for their review. The results of the review as well as comments and opinions were collected and complied, and they are further revised based on the discussions made at the review meeting. In the followings, the outcome of this review is summarized with referring to the original *chapter and section numbers and headings* of the Report.

For Chapter 4 of the Report

4. Concept for Prevention of Severe Accidents and Mitigation of Consequences

4.1 Basic Concept

The basic concept for prevention of severe accidents and mitigation of their consequences are appropriate and the selection of accident sequence groups and representative accident sequences are made systematically and comprehensively, with taking into account of the safety characteristics of SFRs and considering the insight from probabilistic risk assessment (PRA) and the knowledge base from safety research on SFR severe accidents. Although PRAs provide valuable information in the selection process, it is desirable to give careful consideration on uncertainties involved in the process.

4.2 Selection of Accident Sequence Groups

4.2.1 Prevention of Severe Core Damage

(1) Accident sequence groups selected among plant internal events

The accident sequences induced by plant internal events, including the sequences specific to the design characteristic of Monju, are identified systematically based on the comprehensive PRA and deterministic assessment, and thus the procedure and results of selecting accident sequence groups are judged appropriate. They are basically in accordance with the current international standards, such as the Safety Standards series developed by the IAEA.

(2) Accident sequence groups selected among earthquake and tsunami hits

Accident sequence groups initiated from external hazards have been properly identified based on the results of event tree analyses of earthquakes and tsunamis. Selecting the station blackout (SBO) scenario as the consequence of representative event progression from severe external hazards is justifiable for addressing all major external initiators. External hazards other than earthquakes and tsunamis are to be investigated as well, according to the concept which was discussed in the other chapters of the Report, which was out of scope of the this review.

(3) Investigations concerning postulated initiators specific to SFRs

Local faults and passage of gas bubble through the core, and sodium leakage or other sodium reaction events are important accident initiator candidates specific to SFRs, and they are comprehensively treated in the category of design-basis accidents (DBAs) together with the safety design to prevent and mitigate these accidents.

For severe accident consideration here, however, selecting rather significant local faults, such as the total subassembly inlet blockage (TIB) is considered inappropriate, because of a highly hypothetical nature of the events and extremely remote likelihood of occurrence. Therefore the TIB events are judged practically eliminated in accident sequence groups and this concept is in agreement with the current SFR safety approach commonly adopted in the world.

There might be another approach, where the TIB event is selected in the event category of beyond DBA, for safety assessment to represent and bound various local-fault sequences, in a non-mechanistic and deterministic way. For Monju, the TIB was not selected as an accident sequence group, since it was evaluated that the occurrence probability was negligibly small and its consequence was well enveloped by whole-core accidents represented by an unprotected loss-of-flow (ULOF) accident.

4.2.2 Prevention of Fuel Failure in Spent Fuel Storage Tanks

(1) Ex-vessel fuel storage tank (EVST)

The concept and results in selecting accident sequence groups for the EVST are appropriate, considering the design characteristics of Monju. Namely the selection is based on the redundant and diverse design of decay heat removal systems for EVST and the results of accident analyses to demonstrate their adequacy. The considerations are also given to potential events during fuel transfer and handling, and criticality accidents.

(2) Spent fuel pool

The concept and results for selecting accident sequence groups of the spent fuel pool are also appropriate considering the design characteristics of Monju, which provides a sufficient grace period because a decay heating level of the spent fuel transferred to the water pool is sufficiently lowered.

4.2.3 Prevention of Core Damage during Reactor Shutdown

The selected accident sequence groups adequately cover the spectrum of accidents during reactor shutdown in Monju, and design provisions already available are judged sufficient to avoid fuel damage during such sequences.

4.2.4 Ensuring containment function

(1) Selection of accident sequence groups

The accident sequence groups, from which the representative accident sequences are selected for evaluating the effectiveness of the measures to ensure the containment function, are appropriate, since the groups are essentially the same as those selected for evaluating the prevention of significant core damage, covering the accident spectrum caused by both the internal and external events. The design characteristics and event progression specific to SFRs are also considered.

(2) Selection of representative accident sequences.

(a) Anticipated Transients without Scram (ATWS)

Proper representative accident sequences to be assessed have been identified systematically and consistently for the three accident groups, ULOF, unprotected transient overpower (UTOP) and unprotected loss of heat sink (ULOHS), which are classified as ATWS. The procedure and results in selecting the representative accident sequences are judged appropriate. As for ULOHS accidents in Monju, addition of a new interlock is being proposed to strengthen the measures for prevention of core damage. With this design measure, together with a longer grace time in typical in ULOHS, this accident group may well be practically eliminated.

(b) Loss of Heat Removal System (LOHRS)

Proper representative accident sequences to be assessed have been identified systematically and consistently for the two accident groups, protected loss of heat sink (PLOHS) and loss of reactor level (LORL), which are classified as LOHRS. It should be reminded that an SBO accident results in loss of core cooling and hence included in the group PLOHS.

4.3 Concept for Preventive Measures against Significant Core Damage

4.3.1 Prevention of Significant Core Damage

An overall approach to provide design measures and to evaluate their effectiveness for preventing significant core damage, with adequate consideration of the design characteristics specific to SFRs is judged to be appropriate. Representative accident sequences, which might lead to core damage in SFRs, have been properly selected.

It is crucial to provide redundancy or diversity, and independence in the reactor shutdown and decay heat removal systems to prevent significant core damage. At least one of the systems must be surely activated even under the assumption of loss of off-site electric power. In addition, it is necessary for the decay heat removal systems not to lose its functions in accident sequences initiated from external hazards.

The preventive measures against loss of heat removal in Monju provide highly reliable safety function, taking advantage of natural circulation of single-phase sodium over a wide range of initiating events to avoid significant core damage. The safety design features provided in Monju ensure automatic transition to a natural circulation mode, with additional measures, such as batteries or manual operation, being provided, which then can add an extra layer of defense for external events with magnitude beyond the design basis as well.

4.3.2 Prevention of Fuel Damage in Spent Fuel Storage Tanks

An overall approach to provide design measures and to evaluate their effectiveness for preventing fuel damage in the spent fuel storage tanks with adequate consideration of the design features of Monju, which has diversity in the coolant circulation modes, which is forced and natural circulation is judged to be appropriate. Although the natural circulation is effective for preventing fuel damage in accident sequences beyond design basis, providing additional accident management measures, such as the cooling of air cooler by an auxiliary portable blower, may further strengthen the diversity in the preventive measure in

Monju.

4.3.3 Prevention of Core Damage during Reactor Shutdown

The preventive measures against significant core damage during reactor shutdown are judged to be sufficient to avoid core damage.

4.4 Concept for Ensuring Containment Function

4.4.1 ATWS

(1) Basic concept

The basic concept is judged to be in accordance with the commonly practiced safety assessment approach for SFRs with considering a ULOF event as a representative for ATWS events for the assessment to ensure containment function. In addition, it is a common understanding that ensuring containment function with an emphasis on in-vessel retention of degraded core materials is the most important approach in SFRs with taking full advantage of inherent safety features.

(2) Most likely progression of events in ATWS

Since the reactor core in SFRs is not designed in its most reactive configuration, relocation of core materials may have positive reactivity effects. It should be also considered that sodium void reactivity is positive in the center of reactor core. Historically a potential of recriticality events and the resultant mechanical energy release is one of the central safety concerns in the SFRs. The advances in mechanistic analysis codes and acquirement of new experimental knowledge base, from the CABRI and EAGLE experiments, have been significant in this field in the last few decades. Using the state-of-the-art analysis codes with best-estimate assumptions, the accident progression becomes much milder with no prompt recriticality and almost null mechanical energy release, and this is the most likely accident progression of ATWSs.

(3) Evaluation of the influences of uncertainties on event progression

The state-of-the-art analysis codes used in the ATWS analyses can accurately model the reactor core and heat transport systems of Monju in detail and simulate various physical phenomena during the accident progression. Unlike in the DBA analyses, conservative assumptions are not necessarily required for the severe accident analyses, but it must be recognized that there still remain relatively large uncertainties especially in those phenomena that occur beyond fuel damage. The evaluation of influences of these uncertainties is therefore very important to make the evaluated results more reliable and robust.

Within the ranges of uncertainties, which also must reflect the up-to-date knowledge and be rationalized, the mechanical and thermal consequences are to be evaluated. For the former, it has been confirmed that the integrity of reactor coolant boundary is maintained against the mechanical energy release even in an upper bound scenario in which a recriticality event takes place. For the thermal consequence, it has been evaluated that degraded core material may well be retained on the lower-core structures inside the reactor vessel. This scenario of in-vessel retention (IVR) reflects the inherently safe and reliable decay heat removal capability of SFRs.

4.4.2 LOHRS

(1) Basic concept

The basic concept is appropriate to cope with LOHRS-type events by providing redundant measures to ensure containment function by preventing core damage. After the failure of the preventive measures against core damage, it is important that the integrity of the reactor coolant boundary is maintained to ensure the containment function by further preventing core damage by providing additional backup measures in Monju.

(2) Accident sequences

The concept for ensuring containment function during LOHRS-type events employ multiple redundant measures including heat removal by natural circulation using the auxiliary cooling systems (ACSs) and the use of the maintenance cooling system. These multiple measures provide adequate protection with high reliability against core damage and reactor coolant boundary failure even in the case of the failure of DBA mitigation measures. These justify the concept for ensuring containment function against LOHRS-type events.

(3) Summarized results of concept for ensuring containment function in SFRs

Inherent & unique design features of SFR and key design measure to ensure the containment function are well addressed. In SFRs, a long grace periods before core damage in LOHRS-type events has large safety margin and therefore multiple protective measures prevent core damage to ensure containment function by avoiding reactor coolant boundary failure. Contrary to the pressurized system in LWR, a guard vessel is sufficient in

SFRs to maintain the reactor coolant level needed for core cooling, and the pressure relief systems and recirculator units are not needed for SFRs to maintain containment function.

Concerning the SBO accident, which was the real severe accident in 1F Accident and one of the most representative LOHRS-type sequences, the likelihood of loss of containment function is extremely remote as long as one of the multiple cooling paths is available.

(4) Instrumentation and monitoring during severe accidents (SAs)

The concept for instrumentation and monitoring during severe accidents exhibited proper measures for decision-making in the course of severe accident progression. It is recommended to provide instrumentation and monitoring during SAs not only in the central control room, but also in the emergency operation centers.

For Chapter 7 of the Report

7. Concept for Securing Safety of SFRs

Requirement 1

1. *In order to secure the safety of people and to protect environment from accidents in nuclear facility, any risk shall be limited below a socially acceptable level. This shall be accomplished by preventing occurrence of accidents and mitigating their consequences based on the defense-in-depth concept.*

This requirement is a fundamental requirement and is appropriate for defining general and comprehensive framework of safety assurance for design and operation of nuclear facilities by prevention and mitigation of consequences of accidents based on the defense-in-depth concept, and therefore is consistent with the safety design criteria developed for the Generation-IV SFRs.

Requirement 2

2. *Reactor Shutdown Systems: Reactor shutdown systems shall be equipped according to the concept of redundancy, diversity and independence. SFRs shall have multiple and independent reactor shutdown systems using control rods. At least one of those systems shall be able to shut the reactor down to a sub-critical condition and maintain sub-criticality during DBAs.*

The requirement to assure reactor shutdown capabilities with redundancy or diversity and independence is appropriate. The safety requirement can be properly fulfilled by the two independent reactor shutdown systems, each of which has capability sufficient to rapidly and reliably shut down the reactor with large safety margins under normal operation, anticipated operational occurrence and accident conditions.

Requirement 3

3. *Decay Heat Removal Systems: Decay heat removal systems shall be equipped in order to transport the decay heat generated in the core to the ultimate heat sinks. The heat transport systems and ultimate heat sinks shall not lose their functions and integrity.*

The requirement for the decay heat removal system at the level III of the defense-in-depth is properly described considering the features of SFR. It is recommended to explicitly require the redundancy, diversity, and independence, and importance of maintaining the needed sodium inventory.

Requirement 4

4. *Avoid the Common Cause Failure: Needed preventive measures shall be deployed against occurrence and expansion of internal fires and flooding that may lead to common cause failure. The deployment of the measures shall be confirmed, and additional measures shall be implemented if the existing ones are insufficient.*

The requirement against internal fires and flooding is appropriate since it enhances the safety of the plant to prevent a loss of safety function caused by common cause failure. To meet this requirement, the effectiveness of the safety measures needs to be evaluated and additional measures are implemented if necessary.

Requirement 5

5. *Specific accidents concerning SFRs: Sufficient measures shall be taken against the sodium leakage and sodium-water reactions. Present facilities and measures which have already been implemented shall be reviewed to be effective for the sodium leakage in the secondary cooling system and the water leakage in steam generator, and these events shall be investigated as to whether or not they may progress into SAs beyond the design-basis accidents. Additional measures shall be adequately taken, if necessary.*

The requirement is appropriate since it calls for providing sufficient safety measures against the sodium leakage in secondary system and water leakage in steam generator. It also requires the evaluation of possibility of progression into severe accidents beyond design basis, and to provide additional measures if needed. It is also recommended to require measures for sodium leakage in primary system, such as inert atmosphere, to prevent the progression into severe accidents. The influence of toxic materials generated from sodium reactions should be considered in the operation procedures such as fire extinction.

Requirement 6

6. *Accident management (AM) measures shall be adequately implemented against ATWS and LOHRS events which may lead to possible core damage. The measures shall be determined considering the aspects of equipment and facilities (hardware), and operation, management, and system preparedness (software). Loss of safety functions and progress of events shall be adequately considered by referring to PRAs and others.*

It is adequate to require effective AM measures, systematically from both the hardware and software aspects, against ATWS and LOHRS, considering the insight from PRA and the assessment of event progression.

Requirement 7

7. *It shall be considered the ATWS events in SFRs may proceed fast due to its core characteristics, and therefore the safety margins in design shall be carefully confirmed based on updated knowledge and experiences. Adequate measures shall be taken considering the aspects of needed equipment and facilities (hardware), and operation, management, and system preparedness (software).*

It is judged appropriate, considering the design features of SFRs, to require redundant and independent and thereby highly reliable reactor shutdown systems to prevent ATWS events. The AM measures, from both hardware and software aspect, shall be adequately provided for mitigating the event progression and thereby ensuring containment function by attaining in-vessel retention.

Requirement 8

8. *AM measures against LOHRS events are extremely important considering the safety features of Monju, and therefore the adequate measures shall be established from the viewpoints of equipment and facilities (hardware), and operation, management, and system preparedness (software).*

The requirement for AM measures against LOHRS-type events is appropriately defined considering the design features of SFRs. Safety can be enhanced by providing effective AM measures, taking advantage of a wide temperature range of sodium being available in a single phase, viability of decay heat removal by natural circulation, and ample grace period specific to SFRs. It is also recommended to provide AM measures for case when a siphon-breaking operation fails upon multiple sodium leakage events.

Requirement 9

9. Appropriate AM measures shall be implemented against ATWS and LOHRS events so as to practically eliminate the possibility of reactor vessel (RV) failure induced by the molten fuel. Those events resulting in the loss of containment function shall be precisely investigated.

The requirement is appropriate, since advantageous safety features specific to SFRs are to be taken into account.

For ATWS events represented by a ULOF accident, in which significant core damage cannot be prevented, the containment function may be challenged by mechanical and thermal consequences during accident progression. The containment vessel (CV) pressure buildup caused by combustion of sodium ejected from the reactor plug due to the mechanical energy release in the core is no longer a major concern, based on the detailed investigation of event sequences based on up-to-date knowledge. For the thermal consequence, inherent advantageous features of sodium, staying in liquid phase for a wide temperature range and excellent heat transfer capability, can be fully utilized in detailed investigation of event progression. As a result, an approach to ensure containment function by attaining IVR of molten core materials is adequate and justifiable in SFRs, and it is not necessary to assume the reactor vessel failure in the evaluation of the effectiveness of containment function. The need for accurate evaluation of the event progression during ATWS, the load to RV, and the feasibility of IVR with evaluating the impact of uncertainties, are evident.

For LOHRS-type events, the advantageous features of SFRs are identified such that the system pressure of SFRs is low and hence it is possible to keep the coolant level for core cooling against the failure of reactor coolant boundary with the design measure such as the guard vessel. Considering the large grace period for significant core damage and the redundant design provision for decay heat removal as well as additional AM measures, such as inherent and reliable heat removal by natural circulation and the use of maintenance cooling system, the possibility of LOHRS-type events leading to RV failure and thereby resulting in loss of containment function could be practically eliminated.

Requirement 10

10.	*Measures shall be deployed by design in nuclear facilities with sufficient safety margins against natural phenomena, such as earthquakes, tsunamis, and other possible natural phenomena, by assessing risks using the PRA method and so on. Adequate AM measures shall be established, considering features of SFRs, against natural phenomena beyond the design-basis scale, by recognizing their consequences and durability of facilities.*

The requirement to assess the risk of external events using PRA, provide design measures, and implement AM measures against events beyond design basis, is appropriate, reflecting the lessons learned from the 1F Accident. Additional attention should be made to assess complex scenarios that involve multiple consecutive hazards (e.g. earthquake and tsunami) and common cause failures following the external event. The assessment should consider the influence of external events, not only on the plant itself but also on the whole infrastructures surrounding the site which might be similarly affected. In addition to the natural hazards, it is recommended to include the consideration on external industrial hazards if necessary.

Requirement 11

11.	*Measures shall be taken against intentional large-aircraft crashes and other terrorism to prevent occurrence of SAs and to mitigate their consequences. Considerations shall be simultaneously given to features of facilities, such as arrangements or layouts of natural circulation cooling loops and the ACSs, and effectiveness of AM measures as well.*

The necessity to investigate the effect of aircraft crash and other terrorist attacks on the loss of safety function for core cooling and containment function has been adequately identified including implementation of preventive and mitigative measures considering the effectiveness of AM measures and site specific conditions.

The need for the harmonization of the assessment methodologies for the safety and security related design is among the explicit objectives in international consensus to improve the overall safety of future nuclear plants.

Requirement 12

12. *Appropriate AM measures shall be implemented against hydrogen explosion, such as measurement of hydrogen concentration, hydrogen discharge from the CV, controlled small-scale hydrogen combustion, and so forth. Existing facilities at present shall be fully utilized from the viewpoint of preparing reasonable preventive measures against BDBAs.*

The philosophy to take appropriate AM measures against hydrogen explosion, as one of the safety enhancement measures, reflecting the 1F Accident, is adequate considering the possibility of hydrogen generation from sodium chemical reactions beyond design basis. At the same time it is understood that such AM measures are not necessarily required in SFRs because massive hydrogen generation will not occur in the SFR reactor core.

Requirement 13

13. *Adequate AM measures shall be taken to prevent a fuel failure stored in EVST and spent fuel pools considering the features of facilities.*

The requirement is adequate for preventive measures against fuel failure in spent fuel storages considering the design characteristics of SFRs.

Requirement 14

14. *Habitability shall be secured for central control rooms and emergency operation centers. Adequate radiation source terms shall be estimated considering features during accidents in SFRs. Facilities shall be designed considering shielding, ventilation, etc. from the viewpoint of limiting radiation exposure of operators and other needed personnel.*

It is appropriate to estimate source term considering the characteristics of the event progressions in the accidents of SFRs, which are different from those in LWRs, in assessing the habitability in the central control room and the emergency operation center. It is necessary to include additional requirement of independence of the central control room and the emergency operation center, which permits to exclude simultaneous failure of the two due to common cause.

Requirement 15

15.	*Measures shall be implemented to monitor the conditions of the reactor and plant during an SA. Parameters to be monitored shall be adequately identified in view of features of SFRs, progression of events such as possible representative accident sequences, and environmental conditions. Devices for measurement shall be classified depending on the significance of parameters to be monitored, and their aseismic durability shall be secured.*

This requirement appropriately identifies the need to implement adequate devices to monitor the reactor and the plant conditions during SA. The achievement of IVR can be confirmed by various instrumentations that are already available.

Requirement 16

16.	*Needed measures shall be taken so that expedient AM measures can be implemented during SAs. System preparedness, operation procedures, documents and manuals, equipment and facilities, and others shall be established. And education and training shall be performed as well. These are required to enable emergency responses in an expedient and flexible manner against SAs. Effectiveness of AM measures shall be continuously improved by using the results of PRAs and so forth. Its effectiveness, at the same time, shall be also adequately and reversely reflected to PRAs by continuously assessing it using the results of training.*

This requirement is appropriate as general approach of safety assurance and continued improvement. Namely, sufficient considerations are given to implement, improve and enhance the AM measures specific to the SFR plant. AM measures should be elaborated to plant-specific severe accident management guidelines (SAMGs) with the objective to provide operating staff guidelines for SA management to prevent and mitigate core damage, to maintain containment integrity and to minimize offsite releases. It is appropriate to require that such guidelines should be periodically assessed against operating experience and updated knowledge. In line with the international safety standards for LWR, in addition to SAMG, emergency operating procedures (EOPs), should be elaborated for the operators.

4. Conclusion

The Report “Safety Requirements Expected to the Prototype Fast Breeder Reactor Monju” was reviewed by the leading international experts on SFR safety. The major results of the review are summarized as follows:

(1) It was agreed, among the international reviewers, that the basic concept for prevention of severe accidents and mitigation of their consequences was appropriate, since the selection of accident sequence groups and representative accident sequences are made systematically and comprehensively with adequately considering the insight from PRA.

(2) The mechanical and thermal consequences of ATWS events in Monju were evaluated adequately considering various uncertainties. For the mechanical consequences, it was confirmed that the integrity of reactor coolant boundary is maintained against the mechanical energy release resulted from recriticality events even in an upper bound scenario in which a recriticality event takes place. For the thermal consequence, it was evaluated that degraded core material is retained on the lower-core structures inside the reactor vessel. This scenario of IVR reflects the inherently safe and reliable decay heat removal capability of SFRs and in agreement with up-to-date SFR safety approach.

(3) A long grace period before core damage in LOHRS-type events in SFRs provides a large safety margin and multiple protective measures, including heat removal by natural circulation using the auxiliary cooling systems and by the use of the maintenance cooling system. A guard vessel is sufficient in SFRs to maintain the reactor coolant level necessary for core cooling, since the system pressure of SFRs is low. These justify the concept for ensuring containment function of Monju by preventing the core damage.

(4) The requirement for the external events is also appropriate, reflecting the lessons learned from the 1F accident, to use PRA approach, provide design measures, and implement AM measures against external events beyond design basis, considering the consequences of the events and design features and tolerances of SFR structures, systems and components.

Acknowledgment

The invaluable contribution and kind cooperation of the international experts who actively participated in the present international review are gratefully acknowledged. The international review would not have been possible without their dedicated and helpful effort.

Appendix 1

List of Reviewers

Name	Country	Affiliation 1	Affiliation 2	Title	Review meeting
Ammirabile Luca	EU	EC Joint Research Centre, Institute of Energy	Nuclear Reactor Safety Assessment Unit	Project Leader/ Scientific officer	
Tsige Tamirat Haileyesus	EU	EC Joint Research Centre, Institute of Energy	Nuclear Reactor Safety Assessment Unit	Principal Administrator, Directorate General, Scientific Officer	
Liu Yizhe	CN	China Institute of Atomic Energy (CIAE)	General Design Division, Department of fast Reactor Research and Design	Deputy Director	
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Appendix 2

Excerpt from the Report, “Safety Requirements Expected to the Prototype Fast Breeder Reactor ‘Monju’”

(Ed.) Advisory Committee on Monju Safety Requirements

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In July 2013, Nuclear Regulation Authority (NRA) has enforced new regulatory requirements in consideration of severe accidents for the commercial light water reactors (LWR) and also prototype power generation reactors such as the sodium-cooled fast reactors (SFR) of “Monju” based on TEPCO Fukushima Daiichi nuclear power plant accident (hereinafter referred to as “1F accident”) occurred in March 2011.

Although the regulatory requirements for SFR will be revised by NRA with consideration for public comments, Japan Atomic Energy Agency (JAEA) set up “Advisory Committee on Monju Safety Requirements” consisting of fast breeder reactor (FBR) and safety assessment experts in order to establish original safety requirements expected to the prototype FBR “Monju” considering severe accidents with knowledge from JAEA as well as scientific and technical insights from the experts.

This report summarizes the safety requirements expected to Monju discussed by the committee

Keywords: Fast Breeder Reactor, Monju, Safety Requirements, Severe Accident, Accident Management, In Vessel Retention

4 Concept for Prevention of Severe Accidents and Mitigation of Consequences

4.1 Basic Concept

Measures to prevent significant core damage shall be deployed as the first-step measures against beyond the design-basis accidents (BDBAs), which corresponds to the fourth level in the defense-in-depth concept (hereinafter abbreviated as “Fourth Level”). As a second step, measures shall be also provided to ensure the containment function and properly mitigate the off-site release of radioactive materials assuming the failures of the preventive measures of significant core damage.

At the same time, measures for accident management (AM) shall be implemented not only for hardware of equipment and facility but also for software of operation, management and system preparedness based on the results of risk assessments or alternative methods, considering salient features of sodium-cooled fast reactors (SFRs).

Uncertainties on the frequency of failure occurence and human error are considered to be relatively larger in SFRs than those in light water reactors (LWRs) since operational experience of SFRs is scarce. Therefore, the increase of uncertainties in the frequency of accident sequences and the event progression scenario shall be taken into account in considering the safety measures against beyond-design-basis accidents (BDBAs), which correspond to the Fourth Level.

A safety review was performed in the licensing procedures of Monju, the already licensed reactor, assuming accidents corresponding to the Fourth Level (hereinafter called “Paragraph (5) events”). Monju has already been designed based on a principle to mitigate the consequences of accidents corresponding to the Fourth Level. Design measures have been implemented not only to prevent the significant core damage but also to avoid reactor vessel failure and contain radioactive materials with the assumption of the failure of all preventive measures for core damage. These measures are deployed against Anticipated Transient without Scram (ATWS) events as one of the Paragraph (5) events.

As the grace period for core damage from its initiation is in the range from several tens of seconds to several minutes in an ATWS event, the grace time is considered too short for operators to recognize the event and initiate the responding operation. A backup reactor shutdown system is then equipped in Monju as a substitutional and quickly functioning system to prevent core damage in case that all preventive measures fail against DBAs (especially measures for mitigating accident consequences, more specifically a reactor trip by the primary shutdown system).

A shielding plug of reactor vessel combined with its connecting bolts and a reactor containment vessel are installed to contain radioactive materials in case all prevention measures against core damage fail. The former can contain the pressurized sodium in the upward direction which is induced by the occurrence of core damage due to ATWS. The latter can withstand the temperature and pressure increase caused by the combustion of sodium ejected from the gaps around the shielding plug. In-Vessel Retention (IVR) of the debris can be achieved by employing the thick structure of bottom plate of the Low Pressure Plenum (LPP), which can withstand the fall of the damaged core in the downward direction. In addition, cooling systems are equipped to steadily cool down the debris.

On the other hand, as for the measures against Loss of Heat Removal System (LOHRS) events, preventive measures against core damage can be expected by the actions of operators because the grace period is sufficiently long for core damage from its initiation (from several to several tens of hours). In addition, the following redundant or diverse measures are equipped in Monju for core cooling:

(1) Any single one out of the three reactor auxiliary cooling systems (ACSs) has the sufficient capacity to remove the decay heat,
(2) The cooling system for maintenance has the same capability to remove the decay heat by itself, which is independent of the ACSs,
(3) The water-steam system is available to cool down the system by supplying steam to the steam generators, etc. in an emergency situation.

Designs are employed in Monju based on a principle to prevent core damage by these redundant or diverse core cooling measures against LOHRS events.

It is important to adequately assess the functionality of these measures from the viewpoint of operation, management and system preparedness (software) in addition to the equipment and facility (hardware) for the above-mentioned distinguishing measures in Monju, following its design principle. These measures shall be implemented by referring to probabilistic risk assesments (PRAs) and others so that event sequences leading to loss of containment function can be practically eliminated. The risks to release significant radioactive materials shall be eliminated by employing these measures.

The following procedures are considered to be adequate for assuming and evaluating the conditions of BDBAs, which correspond to the Fourth Level.

(1) Selection of accident sequence groups considering the findings obtained from PRAs.
(2) Selection of a representative accident sequence for each accident sequence group.
(3) Identification of preventive measures for core damage against the selected representative accident sequences.
(4) Assessments on the effectiveness of preventive measures against representative accident sequences.

The selection of accident sequence groups to be investigated shall be based on the results of assessments on internal and external events of the plant (especially earthquakes and tsunamis) obtained by PRAs or other alternative methods.

The accident sequence groups for ensuring containment function to be investigated are basically the same as those for preventing core damage. An accident sequence becomes a threat to containment function only when preventive measures against significant core damage fail. Therfore, it is justified to select the accident sequences with the failure of preventice measures for severe core damage as the accident sequences for the assessment of measures eunsuring containment function.

Furthermore, the measures to avoid the siginificant radiation exposure of operators and emergency supporting personnel who stay in the central control room and emergency operation center, shall be clarified considering the purposes of each facility.

4.2 Selection of Accident Sequence Groups

4.2.1 Prevention of Significant Core Damage

Accident sequence groups shall be selected among not only the internal and external events but also postulated initiators apecific to SFRs. The following basic concepts are important in selecting the accident sequence groups.

(i) Regarding plant internal events, a comprehensive PRA shall be conducted and its results determine the accident sequence groups.

(ii) Regarding plant external events, an event tree analysis shall be performed for the earthquakes and tsunami which exceed the design-basis earthquake intensity, reflecting the experience of the accident at the TEPCO Fukushima Dai-ichi nuclear power plant (1F accident). The accident sequence groups shall be selected based on the results of the event tree analysis.

(iii) With respect to postulated initiators specific to SFRs, investigations shall be carried out on whether or not the issue needs to be taken into account as an accident sequence group.

The details of each item are described below.

(1) Accident sequence groups selected among plant internal events.

In the PRA, it is assumed that the design-basis measures (hereinafter called “measures against DBAs”) are only considered in the safety evaluation. In this PRA regarding the accidents caused by internal events hereinafter called “PRA considering only measures against DBAs”, each accident sequence is selected based on the results of “PRA considering only measures against DBAs”. Figure 4.2.1-1 shows an example for Monju.

In the conventional level-1 PRA, both the measures against DBAs and the available measures for the event sequences, in which the measures against DBAs fail, are included as the preventive measures. The latter measures are called hereinafter “measures against SAs”. The effectiveness of each measure will be judged whether or not the core is damaged. The core damage frequency (CDF) will be evaluated by considering all the available measures against each initiator of accidents, as shown in Figure 4.2.1-2 (This level-1 PRA is hereinafter called a “PRA considering all the available measures”.). Therefore, when PRA is evaluated considering all the available measures, the accident sequences derived from the analysis of their contributions to CDF will be limited to the ones assuming that all the measures against SAs fail.

On the other hand, the accident sequences to evaluate the effectiveness of measures against SAs need to be selected deterministically from a statistical population both with the success and the failure of measures against SAs. Therefore, the PRA considering only measures against DBAs was performed.

Events leading to core damage in SFRs can be categorized into the following 3 accident sequence groups based on the results of a level-1 PRA on the internal events, when focusing attention on safety functions of the plant to prevent core damage.

(a) ATWS

In the DBA event, a control rod insertion is usually required to shutdown the nuclear reactors (corresponding to primary control rod insertion in Monju). Failure of this function can be assumed as one of the accident sequences. Actuation of a backup reactor shutdown system is then required in such a case (corresponding to the backup reactor shutdown system in Monju). By deploying this backup system, the reactor will be safely shutdown and the core will be stably cooled without any core damage.

ATWS events can be classified into the following 3 groups, focusing on the differences in event progression from its initiation of core damage. The representative scenarios of each group are shown in Table 4.2.1-1 in brief.

- ✓ ATWS at a loss of flow: A reactor trip signal is generated by an initiator such as a loss of off-site power. All the (three) pumps in the primary heat transport system (PHTS) are automatically tripped. Although the reactor-trip breaker in the primary shutdown system succeeds to open, the reactor is not able to shutdown due to non-insertion of the primary control rods caused by any reason. The event progresses within the time range of several tens of seconds in this case.
- ✓ ATWS at an overpower: The reactor is not able to shutdown in case of overpower condition such as inadvertent and continuous withdrawal of control rods with the failure in opening the reactor-trip breaker in the primary shutdown system. The primary control rods are not inserted. The event progresses within the time range of several tens of seconds in this case.
- ✓ ATWS at a loss of reactor heat removal function: A failure occurs in the reactor shutdown without inserting control rods in the core when a loss of feed water. The event progresses within the time range of several minutes in this case.

The required capacity of the equipment (magnitude of negative reactivity to be inserted by the backup control rods for reactor shutdown) to prevent core damage is the same for all these accident sequences. The ATWS event at a loss of flow however provides the most severe accident consequence due to the shortest grace period for implementing the measures to prevent core damage.

(b) Loss of reactor level (LORL)

An accident sequence is possible to decrease the primary coolant level in RV less than the lower limit necessary for the core cooling by the PHTS (EsL) as illustrated in Figure 4.2.1-3. This is induced by a leakage in the primary coolant piping near the reactor vessel (RV) or below the level of the PHTS (SsL) during a decay heat removal operation after the reactor shutdown caused by a primary coolant leakage.

The primary coolant level will be maintained and the core will be stably cooled even in this case, if a siphon break is achieved by operators as shown in Figure 4.2.1-4.

(c) Protected Loss of Heat Sink (PLOHS)

Decay heat shall be removed even when the reactor is shutdown at an occurrence of a DBA. An accident sequence is postulated in Monju to lose the heat removal capability by means of forced circulation using the ACS. This event can occur at the failures of decay heat removal using all the three ACSs, when multiple active equipments fail simultaneously such as the pony motors of the main circulation pumps in the secondary heat transport system (SHTS).

The heat removal by natural circulation can be expected by the ACS in this case, and it will be secured that the core is stably cooled as shown in Figure 4.2.1-5.

(2) Accident sequence groups selected among events caused by earthquakes and tsunami hits.

An event tree will be analyzed which will lead to the core damage considering all the available measures, including those to mitigate consequences of DBAs in case of events caused by earthquakes and tsunami. The accident sequence groups are selected based on the analysis results.

An example of the event tree is shown in Figure 4.2.1-6 among the event trees assuming an earthquake as an initiator leading to core damage, which includes the heading with the minimum aseismic margin. The integrity of the electric power grid cannot be expected under such a severe earthquake motion beyond the design-basis (hereinafter called “Ss”). An off-site electric power is therefore assumed to be lost in this case, and the success and failure of mitigation measures for consequences of DBAs are evaluated.

The function of emergency diesel generators (DGs) cannot be guaranteed under an earthquake of which motion exceeds 1.25Ss. Decay heat removal by forced circulation fails when all the alternate electric power supply is lost (hereafter called Station Black Out “SBO”). And decay heat removal by natural circulation cannot be expected (is ignored) when considering only the mitigation measures for consequences of DBAs. Core damage occurs in this case.

The smaller the magnitude of earthquake motion is, the higher its occurrence frequency, and vice versa in general. The frequency to lose the safety functions of relevant equipment can be represented by the earthquake occurrence assuming that the functions will be lost when the magnitude of earthquake motion exceeds the aseismic design margin of the equipment.

The higher frequency to lose the functions is anticipated on the equipment with the least aseismic design margin based on this assumption. The aseismic design margin for the DGs is the least among the equipment presented in the event tree in Figure 4.2.1-6, which implies that the frequency to lose the functions is the highest for the DGs. An SBO is thus selected as the postulated accident sequence group based on the assumption mentioned above.

The actual aseismic design margin becomes larger than the evaluated values for the DG, since it was evaluated by the comparison of the demonstrated acceleration margin of DG by test and the responding accelerations in both horizontal and vertical directions at the exact location where the

DG is installed.

It is preferred to analyze and select the accidental scenario based on the event trees or PRAs using the acceleration of which the function actually fails. In the analysis of event tree above, the accidental scenario was selected by concentrating the area where the acceleration is close to the Ss so that the occurrence frequency becomes higher than others.

A wider spectrum of accident sequences will be considered if larger earthquake motions are assumed. However, the occurrence frequency of earthquake motions much larger than Ss will be lower and its uncertainty becomes larger. It is required to investigate this kind of lower occurrence frequency events including those close to the Ss by referring to PRAs and others, so as to enhance safety continuously by preparing and establishing AM measures against these events.

An event tree is presented in Figure 4.2.1-7, which includes the heading with the minimum margin against tsunami, among the event trees assuming a tsunami as the initiator which leads to core damage. The sea water pumps used in the component cooling system to feed cooling water to the DGs will be immersed under the sea water level, when the tsunami height exceeds 6.4m above sea level which corresponds to the height of floodwall surrounding the sea water pumps.

This disables the operation of these sea water pumps, and thus the DGs will lose its function leading to an SBO. An SBO shall be selected as an important accident sequence group as the same way in case of earthquake.

(3) Investigations concerning postulated initiators specific to SFRs.

Investigations were conducted concerning possible initiators, especially the following events, in light of features of SFRs. As a result, it was judged that these issues need not be selected as an accident sequence group.

(a) Instantaneous flow blockage in a whole fuel sub-assembly.

Instantaneous flow blockage in a whole fuel sub-assembly is the event when an instantaneous and complete flow blockage occurs at the entrance part of a fuel sub-assembly. This event assumes larger coolant flow blockage than that considered in DBAs (within a single sub-channel of a fuel sub-assembly) and larger local flow blockage considered in Paragraph (5) accidents (a blockage of 2/3 of a fuel pin bundle). The occurrence frequency to initiate this type of event is estimated to be extremely low. And the safety design is employed in Monju in order to exclude such a possibility of the flow path being instantaneously and completely blocked at the inlet of a fuel sub-assembly. This can be attained by deploying anti-blockage orifice configurations based on the lessons learned from the past overseas experiences.

(b) Passing of a large bubble through the core.

Passing of a large bubble through the core region is assumed to occur an insertion of positive reactivity which leads to core damage. This is an initiating event beyond the bubble-passing-through accident in DBAs. The DBA was hypothetically assumed in order to confirm the safety under such a condition that a pulsed positive reactivity is inserted to the core by initiators apecific to SFRs. The dipped plates are installed just below the sodium surface in the RV to suppress the sloshing. And the gas venting holes are located at the core support plate in Monju. It is judged that the reactivity insertion caused by passing of a large bubble through the core can be excluded by these preventive measures against bubbles.

The following 4 types of accidents are selected as the primary accident sequence groups to assess the effectiveness on preventive measures against core damage.

- ✓ Anticipated Transient without Scram (ATWS)
- ✓ Loss of Reactor Level (LORL)
- ✓ Protected Loss of Heat Sink (PLOHS)
- ✓ Station Blackout (SBO)

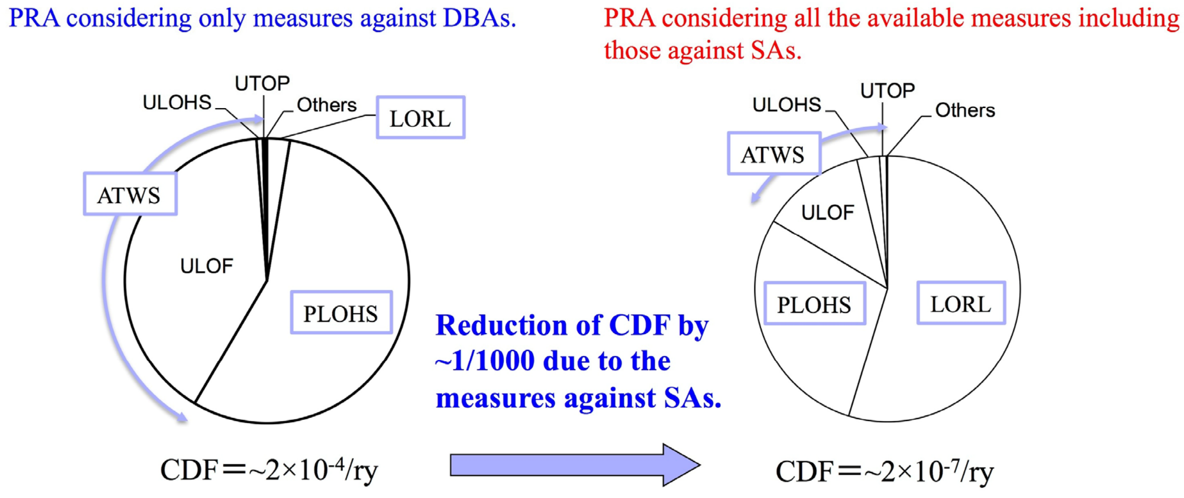
Representative accident sequences shall be selected for effectiveness assessment on preventive measures against core damage among the accidents included in each sequence group mentioned above. This procedure however is skipped here and will be discussed in Paragraph 4.2.4 together with the accident sequences leading to the loss of containment function without any core damage. The only difference between these two is that the latter assumes failure of first-step preventive measures against core damage.

Table 4.2.1-1 Typical scenarios for ATWS events.

ULOF	Typical scenario : • The control rods (of the primary shutdown system <a measure against DBAs>) are not inserted by a reactor trip signal due to a loss of off-site power etc., whereas the circulation pumps in the PHTS are tripped. Event sequence : • A mismatching in power to flow rate (rated power + loss of flow) ⇨ Insertion of the control rods (of the backup shutdown system <a measure against SAs>) ⇨ Safe and stable cooling of the core without damage
UTOP	Typical scenario : • The control rods (of the primary shutdown system) are not inserted due to failure to open the reactor-trip breaker, although a reactor-trip signal is generated by inadvertent and continuous withdrawal of control rods. Event sequence : • (Ditto)
ULOHS	Typical scenario : • The same scenario mentioned just above but caused by a loss of feed water. A loss of flow at the water-steam system occurs, whereas all the circulation pumps in the PHTS and SHTS continue to operate at the rated power. Event sequence : • A loss of heat sink (rated power operation of the core + continued operation of the PHTS + that of the SHTS + a loss of flow at the water-steem system) ⇨ Insertion of control rods (of the backup shutdown system) ⇨ Safe and stable cooling of the core without damage

Note) When the reactor is shutdown by measures against DBAs, the control rods of the primary shutdown system are inserted by completion of opening of the breakers for power feeding to them at around 0.9 seconds later after generation of a reactor trip signal due to a decrease in normal bus voltage. When the reactor is shutdown by measures against SAs (BDBAs), on the contrary, the control rods of the backup shutdown system are inserted by opening of the breakers for power feeding to them at around 1.9 seconds later (an example of results on DBA-based analysis) triggered by a secondary reactor-trip signal (a decrease in rotation speed of the PHTS circulation pumps) in failure of generation of a trip signal due to a decrease in normal bus voltage. The needed durations of time to insert the rods into 85% of the full stroke are the same for both the primary and backup shutdown systems to be 1.2 seconds (by a DBA-based analysis).

Note) The BDBAs are selected to envelope multiple accident sequences as a basic concept. Occurrence of multiple failure is therefore selected as the accident sequence for ULOF events, more specifically a combination of failures in generating a primary reactor -trip signal, in opening the trip breaker for the primary shutdown system, and in inserting the primary control rods.



The following accident sequence groups are identified to be most influential based on a PRA on the internal events:

- ATWS
- LORL
- PLOHS

Categorization of typical events leading to core damage in SFRs and identified accident sequence groups (shaded columns)

Category 1*1	Category 2*2
ATWS: Anticipated Transient without Scram	ULOF: Unprotected Loss of Flow
	UTOP: Unprotected Transient Over Power
	ULOHS: Unprotected Loss of Heat Sink
LOHRS: Loss of Heat Removal System	LORL*3: Loss of Reactor Level
	PLOHS*4: Protected Loss of Heat Sink

*1) Category 1 is classification by the basic safety functions in nuclear reactors.
*2) Category 2 is classification by the differences in accident progression from the onset of an initiating event to core damage. This is also a classification focused on the essential safety functions needed for the prevention of core damage. The followings are the indispensable safety functions to prevent core damage in Monju. Loss of functions among any one of these will lead to a core damage. (as listed below).
1) reactor shutdown, 2) securing of reactor sodium level, and 3) decay heat removal.
*3) Loss of heat removal function due to loss of maintaining reactor sodium level function.
*4) Loss of heat removal function whereas the reactor sodium level being maintained.

Fig.4.2.1-1 Accident sequence groups identified among the internal events (Results of level 1 PRA)

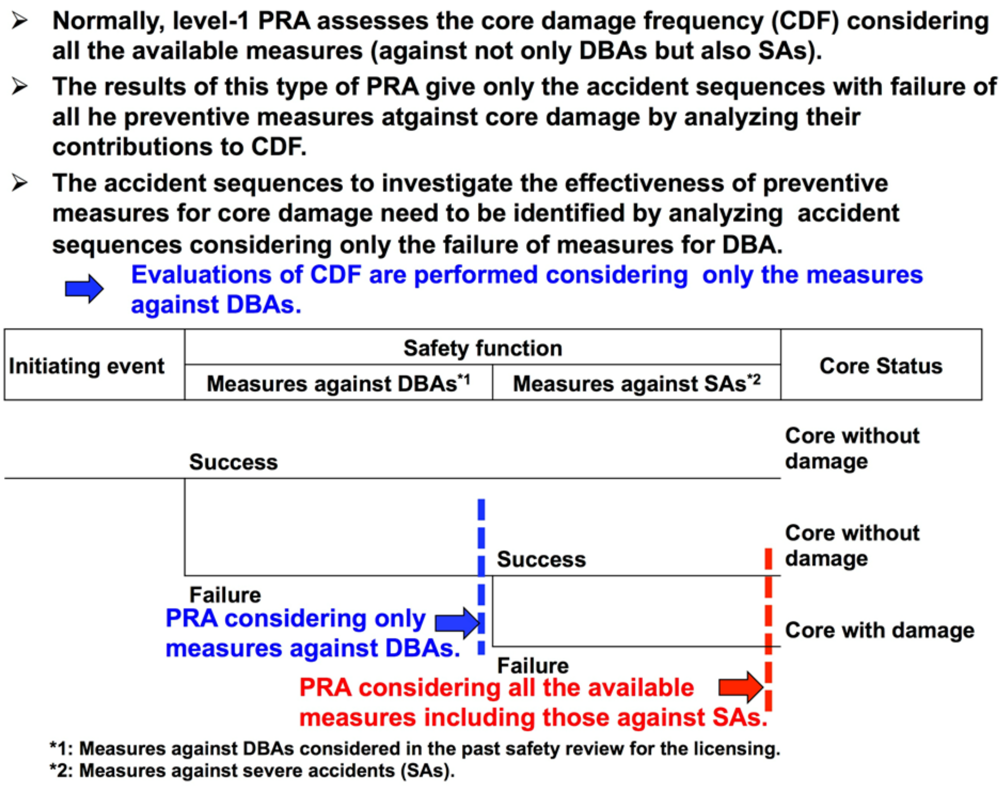


Fig.4.2.1-2 Concept for selecting candidate events to be investigated.

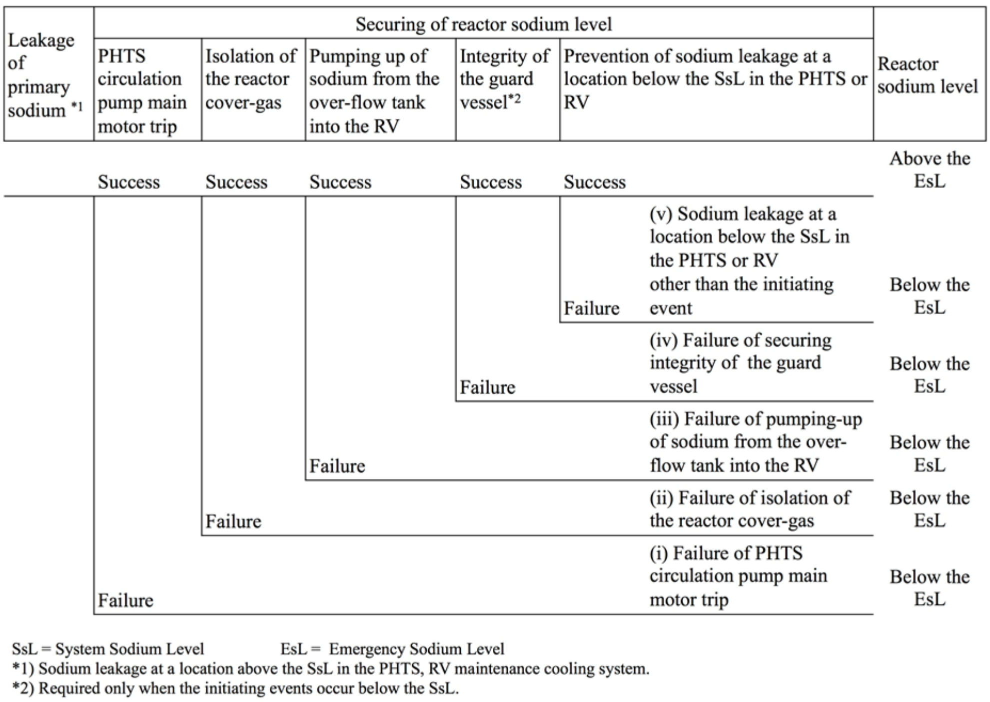


Fig.4.2.1-3 Event tree for securing reactor sodium level

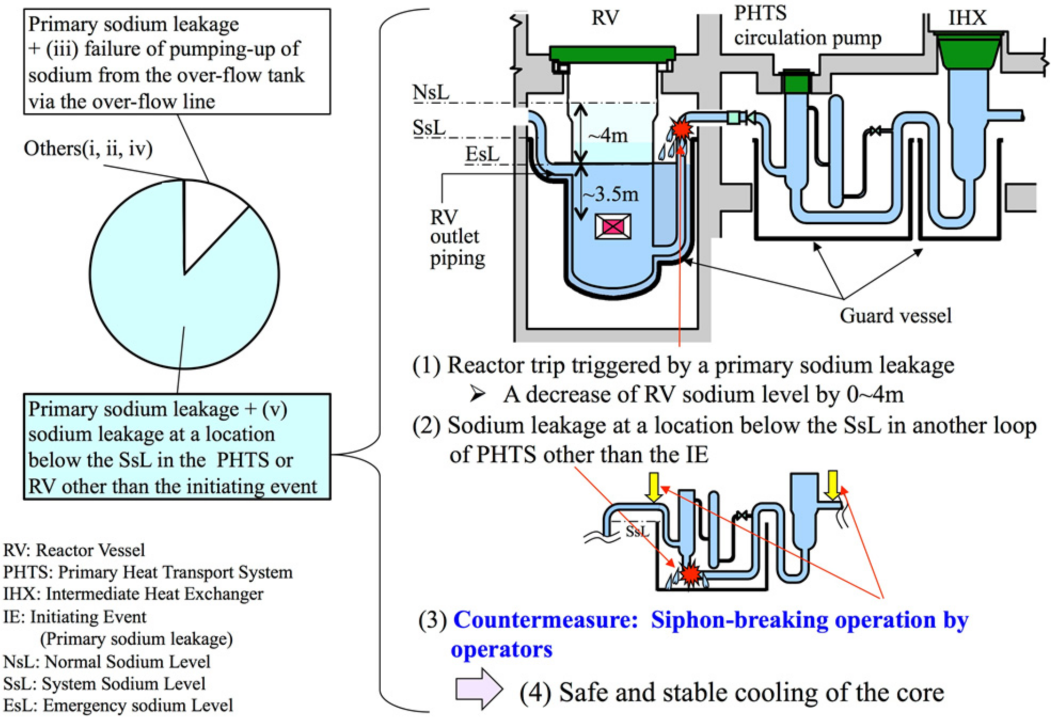


Fig.4.2.1-4 Identification of the representative accident sequence for LORL
((i) to (v) correspond to those shown in Fig.4.2.1-3)

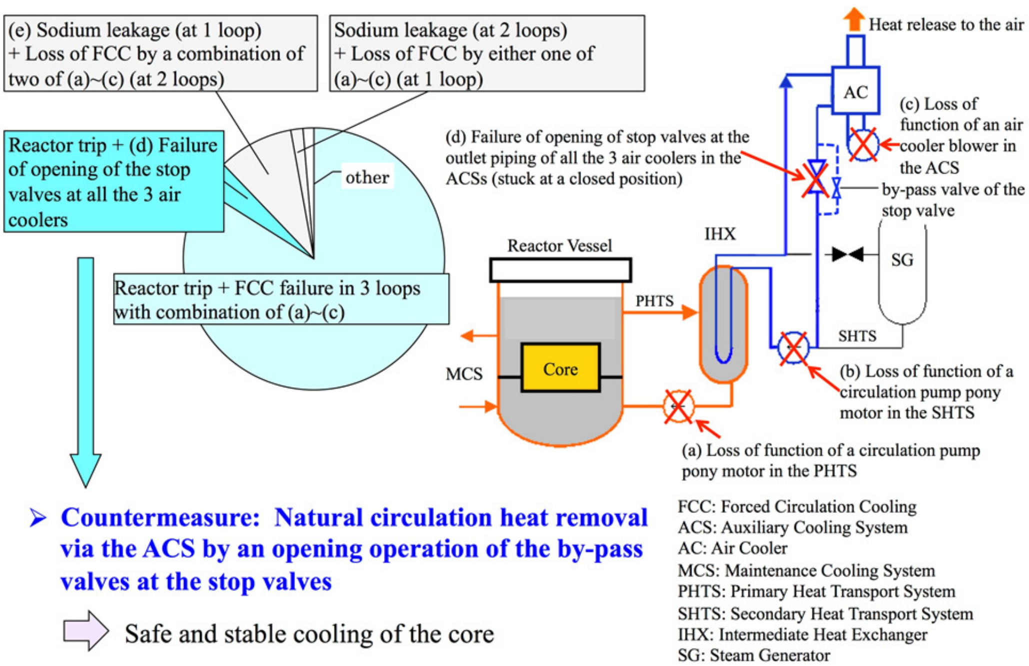


Fig.4.2.1-5 Identification of the representative accident sequence for PLOHS

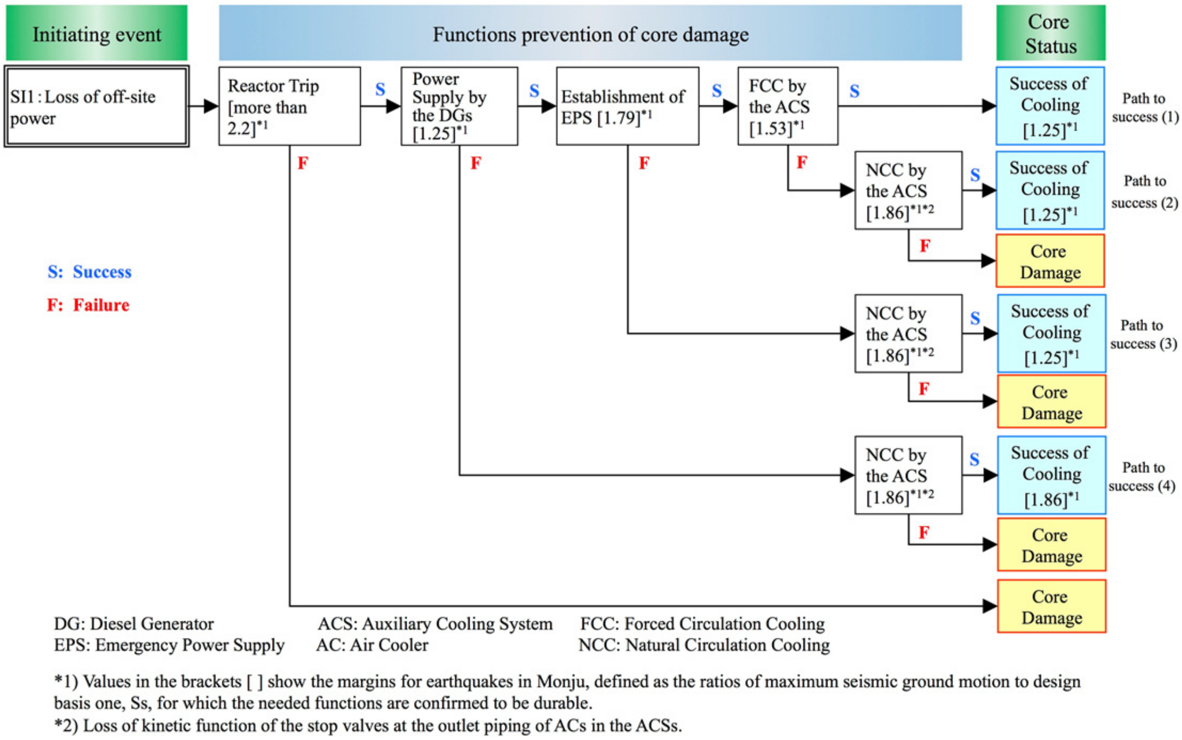


Fig.4.2.1-6　Event tree for loss of off-site power with safety margins for earthquake

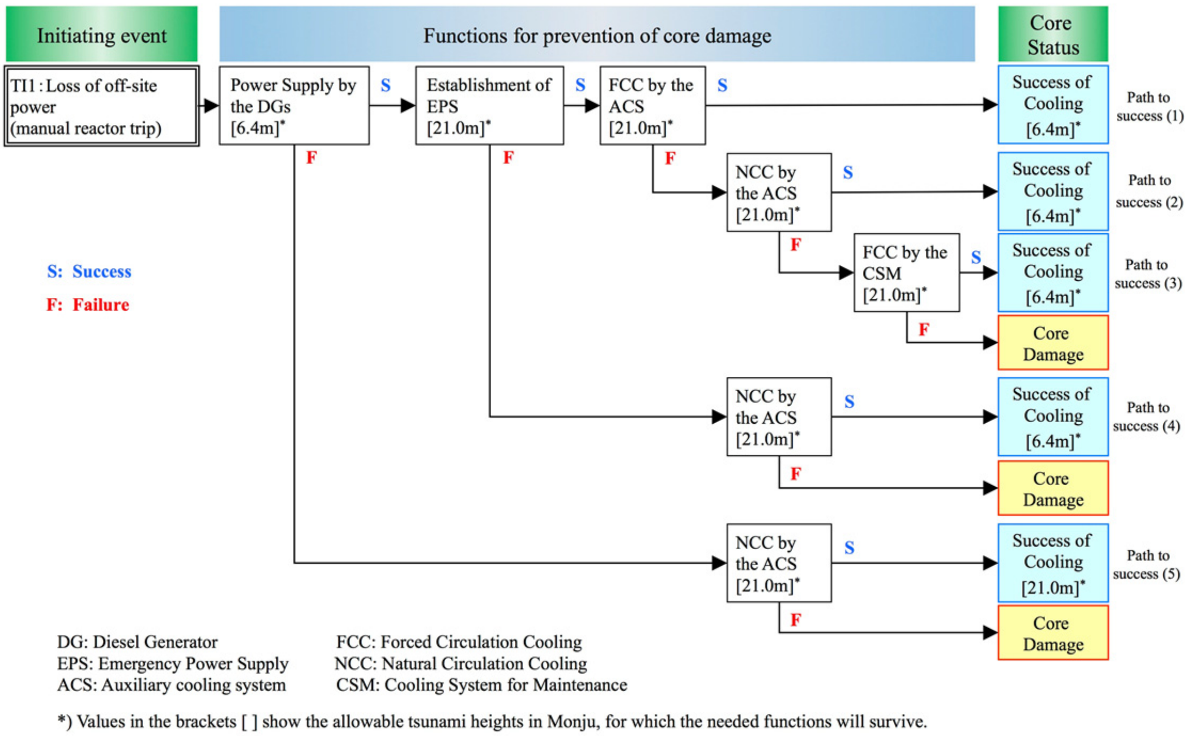


Fig.4.2.1-7 Event tree for loss of off-site power with allowable tsunami height

4.2.2 Prevention of Fuel Failure Stored in Spent Fuel Storage Tanks.

Spent fuels of the Monju are stored under the sodium in the Ex-Vessel fuel Storage Tank (EVST) until the decay heat decreases sufficiently after they discharged from the reactor. The cooled spent fuels are subsequently stored and cooled under the water in the spent fuel pool. These facilities are designed to be able to maintain sub-criticality during fuel storage both in the EVST and spent fuel pool.

Possible accident sequence groups are as shown below.

(1) EVST

The EVST is designed to transfer the decay heat from the spent fuel to the air coolers via forced circulation of sodium in the EVST as the 3 independent cooling systems. The cooling systems are designed to allow a natural circulation in case that the forced circulation fails. An outer vessel is provided so as to maintain sodium level for spent fuel cooling even if a sodium leaks from the inner vessel.

The possible accident sequence groups are the followings based on these features:

"Accidents to raise the sodium temperature" caused by loss of cooling function, and

"Accidents to lower the sodium coolant level" in the fuel storage tank caused by small-scale leakage from a penetration part of piping on the tank wall.

(a) An accident to raise the sodium temperature in the EVST can be induced by a loss of decay heat removal function by forced circulation due to failures of all the (three) pumps in the cooling system loops. Decay heat can be removed by the transition to a natural circulation mode even in such a case.

(b) An accident to lower the sodium level is caused by a sodium leakage from a penetration part of the piping on the vessel wall as shown in Figure 4.2.2-2. The sodium coolant level is maintained higher than that needed for spent fuel cooling in such an event.

(2) Spent fuel pool

Decay heat released from the spent fuels stored in the spent fuel pool is designed to be transferred and dispersed to the sea water as shown in Figure 4.2.2-3. The water in the spent fuel pool is cooled by forced circulation using the spent fuel pool cooling and purification system. The outlet piping to drain the pool water is located at a level above the top of spent fuels in order to ensure enough radiation shielding and decay heat cooling. The inlet piping to pour the pool water, on the contrary, is located at a level of the bottom of spent fuels, and a check valve is located at the inlet piping.

The possible accident sequences are "Accidents to raise the water temperature" caused by loss of cooling function, and "Accidents to lower the water level" caused by water leakage.

(a) An accident to raise the water temperature would occur at a loss of the spent fuel pool cooling and purification functions, and make-up water system as well. The decreased decay heat provides a sufficient grace period of around 70 days to prevent the decrease of the water level due to water evaporation. The needed water level can be maintained by pouring the water by the on-site firefighting team.

(b) An accident to lower the water level would occur when the water leakage occurs at a lower part of piping in the spent fuel pool cooling and purification system. This leakage can be induced, by assuming a sticking of the check valve being opened at the same time as illustrated in Figure 4.2.2-4. However, the needed water level can be maintained by stopping the water drainage by siphon break of inlet piping.

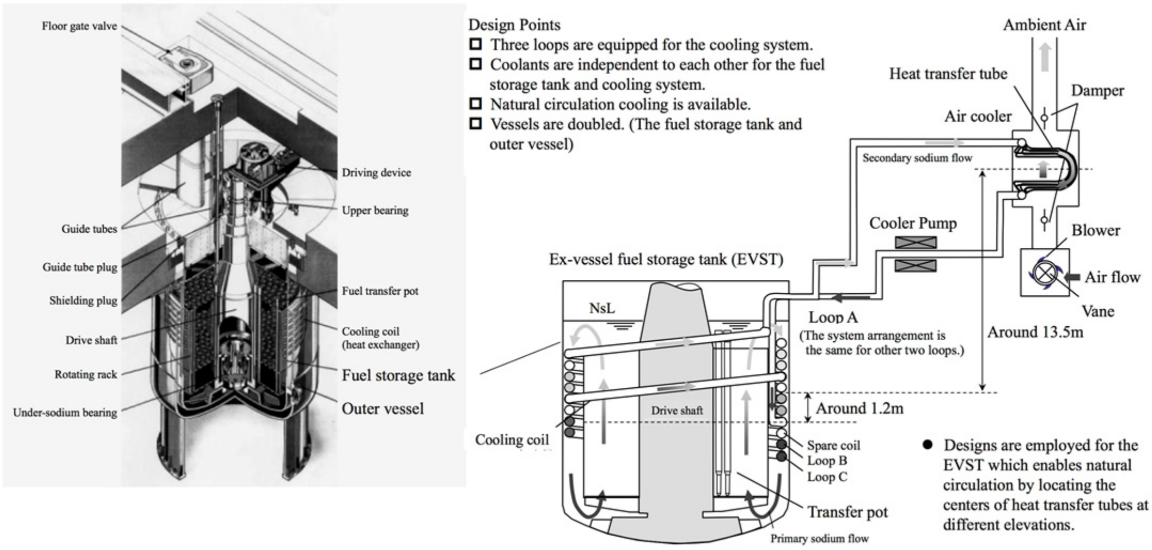
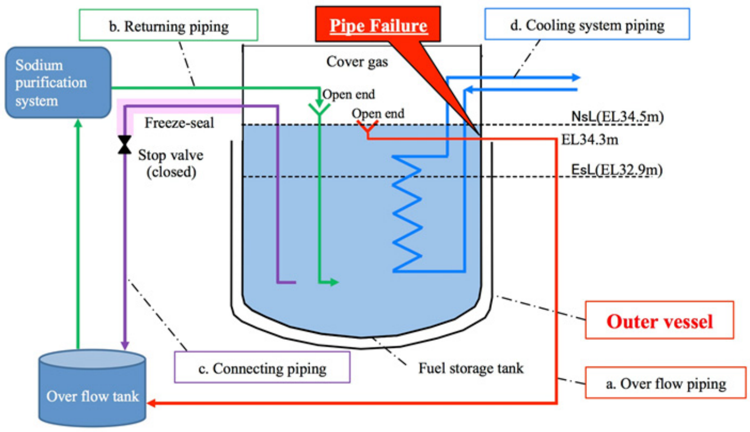


Fig.4.2.2-1 Overview of EVST



a. Over flow piping: This piping penetrates the wall of the fuel storage tank at a location below around 20cm from the NsL. The sodium in the tank will be spilled in case that a penetration part of this piping is damaged.

b. Returning piping: A part of this piping is opened to the cover-gas region. This piping therefore is designed not to allow for leakage of sodium by the siphon effect even assuming its rupture.

c. Connecting piping: This piping is freeze-sealed at a part above the stop valve to the fuel storage tank wall. And the stop valve is always closed. (Do not opened in the normal operation, etc.)

d. Cooling system piping: The sodium in the cooling system is independent of that in the fuel storage tank. And its piping is highly located above the NsL.

Fig.4.2.2-2 Location of piping penetrations and sodium level of the fuel storage tank.

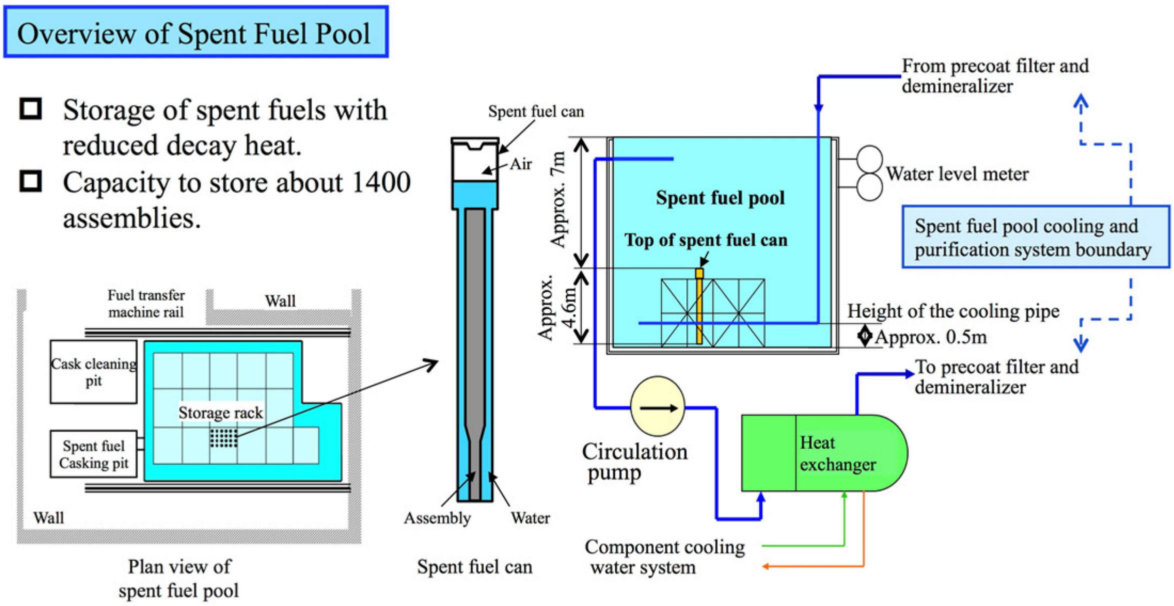


Fig.4.2.2-3 Overview of the spent fuel pool.

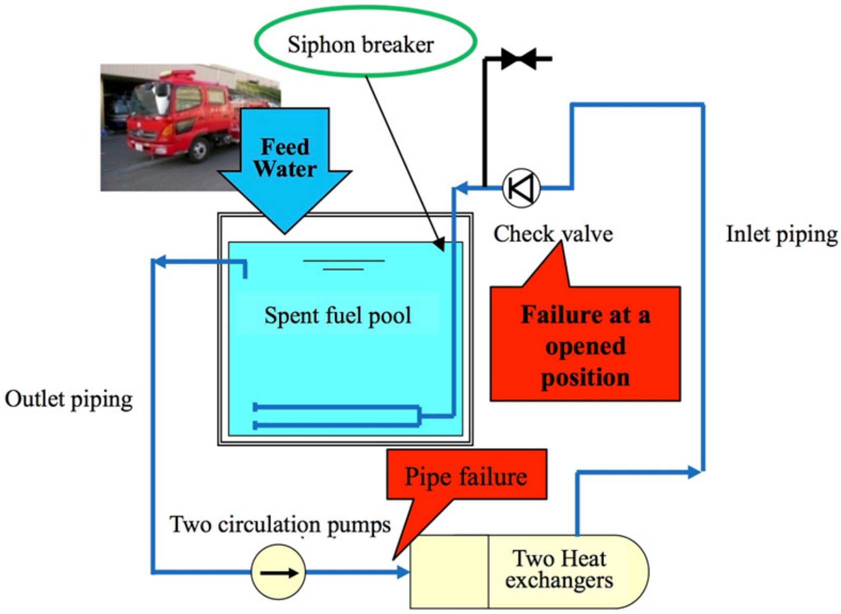


Fig.4.2.2-4 Assuming an accident to lower the water level in the spent fuel pool.

4.2.3 Prevention of Fuel Damage Loaded in the Core during Reactor Shutdown.

Sodium temperature of the cooling system and decay heat during the reactor shutdown are lower than those during rated-power operation. These features of SFRs during reactor shutdown extend the grace period to prevent the fuel damage longer than that during power-generation operation. On the other hand, the coolant sodium level in the RV is occasionally decreased from the normal sodium level at the rated-power operation as shown in Figure 4.2.3-1, and the number of available cooling loops will be reduced by draining the sodium in the HTSs during a maintenance mode.

The initiating events and their progresses leading to fuel damage are basically the same in both during reactor shutdown and during rated power operation. Accident sequence groups during reactor shutdown shall be selected by referring to the results of PRA during rated power operation considering the similarity of both conditions.

The possible accident sequence groups are shown below.

(1) Unintended reactivity insertion.

This type of events corresponds to ATWS during power-generation operation. The reactivity of the reactor is controlled by means of control rods only, and all the control rods are inserted into the core during reactor shutdown. In this situation, inadvertent and simultaneous withdrawal of only 2 control rods is assumed considering the designs to prevent inadvertent withdrawals and relevant operational procedures not to allow such an operation. The core would still be maintained at a sub-critical state with the residual 17 control rods being fully inserted during the reactor shutdown.

(2) Leakage of reactor coolant sodium.

This type of events corresponds to LORL during power-generation operation. The coolant sodium level in the RV during maintenance is maintained at a level lower than that during power generationg operation, and it will be further decreased by a leakage of primary coolant sodium. However, fuel damage can be prevented by decay heat removal using the maintenance cooling system, even assuming such a condition.

(3) Protected loss of heat sink

The event progression of PLOHS is basically the same as that occurred in case of the transition from a failure of forced circulation of coolant sodium to natural circulation. A failure of decay heat removal by forced circulation at loop B can be assumed during maintenance of loops A and C, where the number of available cooling loops becomes the minimum. However, fuel damage can be prevented by transitioning to natural circulation of the coolant sodium to remove the decay heat using the ACS even under such a condition, in the same way as is applied during power-generating operation.

(4) SBO

An SBO is caused by an external event which is the same as the case of events during power-generation operation, and the progress of both events is basically same. A loss of off-site power together with a simultanious loss of on-site emergency DG results in an SBO leading to a failure of decay heat removal by forced circulation during maintenance. However, fuel damage can be prevented by transitioning to the decay heat removal by natural circulation even in such a case.

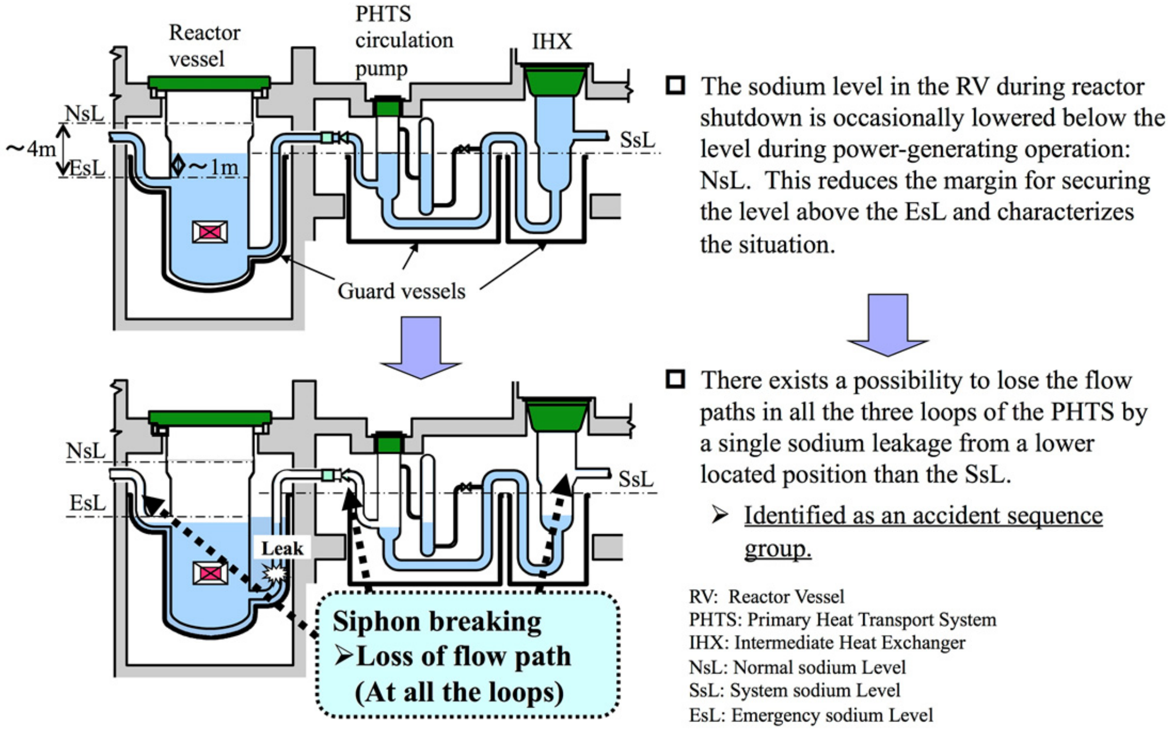


Fig.4.2.3-1 Sodium level in the reactor vessel during reactor shutdown (for maintenance)

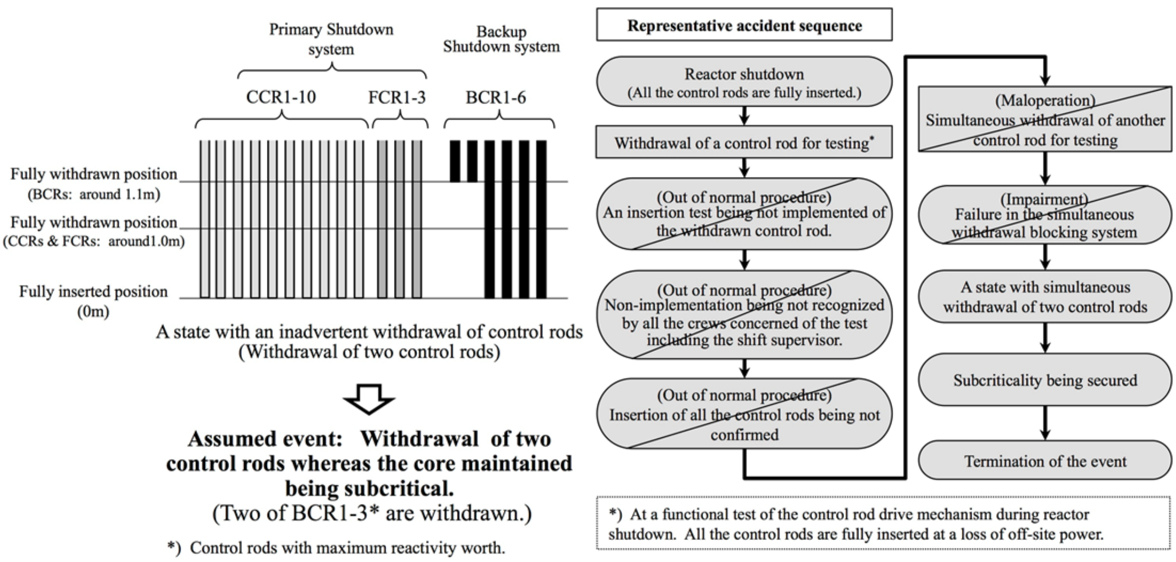


Fig.4.2.3-2 A state of the control rods inserted into the core during reactor shutdown

4.2.4 Ensuring contaiment function.

(1) Selection of Accident Sequence Groups

The accident sequence groups to ensure the containment function in SFRs are the same as those against core damage, as mentioned in Paragraph 4.2.1. An accident sequence becomes a genuine threat to containment function only when the measures to prevent the significant core damage fail. Therefore, the effectiveness of measures to ensure containment function shall be assessed on such accident sequences among those shown in Paragraph 4.2.1 with the failure of preventicve measures. The accident sequence groups to be investigated are as follows.

(a) Anticipated Transient Without Scram (ATWS)
(b) Loss of Reactor Level (LORL)
(c) Protected Loss of Heat Sink (PLOHS)

As the progress of SBO events occurred in the core is the same as that of PLOHS, an SBO which has already been included in Paragraph 4.2.1 is not considered here to avoid duplication. The accident sequences included in the selected groups shall be basically identified based on the results of PRAs.

(2) Selection of representative accident sequences.

(a) ATWS

This accident sequence group can be divided into 3 sub-groups when focusing on the differences in their progress of events from its initiation to core damage. They are Unprotected Loss of Flow (ULOF), Unprotected Transient Overpower (UTOP), and Unprotected Loss of Heat Sink (ULOHS). Core damage is presumed in ATWS events, because it is judged that the grace period to reach core damage from initiation of the accident is too short to implement the preventive measures by operators. Cooling capability of the damaged core and containing capability of the containment vessel (CV) are the issues to be investigated.

It is adequate to select the representative accident sequences for these 3 sub-groups based on the contribution ratios to CDF derived by PRAs considering all the available measures. Figure 4.2.4-1 presents the breakdown of accident sequences included in ULOF, UTOP and ULOHS, and selected representative accident sequences.

The accident sequences in ULOF can be categorized into the following 2 major items:

(i) Failure to open the electric power-supply breaker to delatch control rod clutching device, caused by a failure to open the reactor-trip breaker.

(ii)Failure to insert control rods, under the success to open the reactor-trip breaker.

The degree of consequences is the same for both cases, because the reactor shutdown fails by both the primary and backup reactor shutdown systems, and the events progress in a same manner as the time history of flow coast-down for both cases. Then case (ii) is selected as the representative accident sequence because of higher contribution ratio to CDF.

The primary contribution occupies over the half in UTOP events from inadvertent withdrawal of a control rod during rated-power operation. The second contributor is that during partial-power operation accounting for around 30%. The event progression shall be evaluated for UTOP in selecting representative accident sequences among either of them, which provides the more severe condition.

The event progression of ULOHS can be grouped into the following 3 types, as shown in Figure 4.2.4-1, based on the results of a PRA: LOHS at a single loop, LOHS at 3 loops, and LOF at a single loop. The event progression shall be also evaluated for ULOHS in selecting representative accident sequence among them, which gives the most severe condition.

The occurrence frequency of ULOHS can be significantly reduced by automating the AM measures against that relying on actions of operators (cognition of ULOHS occurrence and manual insertion of control rods) according to the analysis on the results of level-1 PRAs. Additional modifications of systems would be effective, that is adding an interlocking device for ULOHS prevention (a circuit for cognition of ULOHS occurrence and automatic insertion of backup control rods).

Assessments shall be performed on the actuation of this interlocking device, as a measure to ensure containment function, in assessing effectiveness of such measures under a ULOHS condition.

The subsequent event progressions after the actuation of the reactor-trip breaker are closely related to the actuation status of the breaker in Monju, due to the feature of its controlling circuit. The plant comes to a situation where the circulation pumps in the PHTS and others are automatically tripped by a reactor-trip signal, whereas the control rods are not inserted not by a failure in the opening of the reactor-trip breaker (succeeds in it) but by some other event (mechanical sticking, etc.). The subsequent event progression, in this case, is identical with that in a ULOF event, and therefore categorized into the same group. The pumps in the PHTS are tripped and the control rods are not inserted in this case.

On the contrary, all the equipment, including the pumps in the PHTS, is not tripped and continues its rated-power operation, when failure of opening at the reactor-trip breaker occurs, whereas the control rods are also not inserted due to continued power supply to their clutching devices.

This type of event progressions is categorized as those during UTOP when the event initiator is inadvesrtent control rod withdrawal, and during ULOHS when caused by some impairment to reduce heat removal capability from the PHTS.

The categorization of accident sequences regarding ATWS, as mentioned above, depends on the success and failure to open the reactor-trip breaker due to the feature of its controlling circuit. The success leads to ULOF and the failure results in UTOP or ULOHS depending on the initiator. With respect to the initiators of accidents, ULOF has no limiting condition. All the initiators issuing a reactor trip signal can cause a ULOF event. On the other hand, a UTOP or ULOHS event is caused by a limited initiator.

(b) LORL and PLOHS

The grace period is granted relatively long from several to several tens of hours to reach the core damage from its initiation regarding the events to lose the decay heat removal function. Sufficient grace time is therefore allowed to implement the preventive measures by operators against core damage. Containment function can be further kept by deploying multiple preventive measures against core damage being as independent as possible.

Containment function shall be ensured by preventing core damage in a continuous manner. Even though a first-step preventive measure fails, the following second-step preventive measure shall be effectively provided within several to several tens of hours.

The representative accident sequences need to be selected from a population of accident sequences referring to the contribution to CDF derived by PRAs considering only measures against DBAs. The same procedure can be employed in order to select these accident sequences as that for assessing core damage.

The representative accident sequences for LORL and PLOHS events therefore will be identified among those selected for core damage together with a failure of the first-step preventive measures (See Figures 4.2.1-4 and -5).

Table 4.2.4-2 summarizes the representative accident sequences to assess the effectiveness of the measures to ensure containment function for ATWS, LORL and PLOHS.

Table 4.2.4-1 Classification of accident sequences included in ULOHS-type events

Classified accident sequences	Description of events	Accident Sequences
LOHRS at a single loop	A circulation pump is tripped at a single loop of the SHTS (by the interlocking system in case of a heat-transfer-tube failure at an SG) at the inception of an initiating event. The circulation pumps will continue their operations at all the three loops of the PHTS and residual twos of the SHTS due to a failure in opening of the reactor-trip breaker. The residual circulation pumps of the SHTS in operation will be then tripped by the interlocking system along with the progress of the event. The PHTS is thus isolated and becomes to an adiabatic state. The reactor inlet temperature will be increased gradually whereas the circulation pumps will continue their rated operations at the three loops of the PHTS.	Decrease in flow of the SHTS (at 1 loop) + failure in opening of the reactor-trip breaker
		Failure of a heat transfer tube at an SG + failure in opening of the reactor-trip breaker
LOHRS at three loops	Circulation pumps are tripped at all the three loops of the SHTS by the interlocking system at the onset of an initiating event. Circulation pumps, on the contrary, will continue their operations at all the three loops of the PHTS due to a failure in opening of the reactor-trip breaker. The progress of the event will be the same as above after this situation.	Loss of main feed water flow + failure in opening of the reactor-trip breaker
		Turbine trip + failure in opening of the reactor-trip breaker
LOF at a single loop	A circulation pump is tripped at a single loop of the PHTS and the check valve is completely closed at the inception of an initiating event. The circulation pumps will continue their operations at all the three loops of the SHTS and residual twos of the PHTS due to a failure in opening of the reactor-trip breaker. The circulation pumps of the SHTS will be then tripped, at the corresponding loops to the residual ones of the PHTS in operation, by the interlocking system along with the progress of the event. The progress of the event to be followed will be the same as above. This event is Classified into LOF-type accidents when focusing on the initiator. It is however categorized into ULOHS-type accident sequences when considering progress of the event being a LOHS-type.	Decrease in flow of the PHTS (at 1 loop) + failure in opening of the reactor-trip breaker

Table 4.2.4-2 Representative accident sequences for effectiveness assessment on measures for ensuring containment function regarding ATWS, LORL and PLOHS

Accident Sequence Groups	Sub-Groups	Representative Accident Sequences
ATWS	ULOF	Off-normal transient + failure of control-rod insertion at both the primary and backup systems
	UTOP	Inadvertent control rod withdrawal during rated-power operation + failure in opening of the reactor-trip breaker, or The same sequence as above but during partial-power operation.
	ULOHS	LOHRS at a single loop, LOHRS at three loops, or LOF at a single loop.
LORL		Leakage of primary sodium + additional sodium leakage from a position located below the SsL in the PHTS during the decay heat removal operation + failure of siphon breaking operation
PLOHS		Off-normal transient + failure of heat removal by forced circulation due to failure in opening of the stop valves at the outlet piping of all the three air coolers in the ACSs (fail stuck at a closed position) + failure of heat removal by natural circulation due to failure in opening of all the three by-pass valves at the stop valves

Sub-groups	Representative Accident Sequences
ULOF	Off-normal transient + Control-rod-insertion failure (B)
UTOP	Inadvertent withdrawal of a single control rod during rated-power operation + failure of opening of the reactor-trip breaker (A), or The same as above but during partial-power operation (A).
ULOHS	LOHRS at a single loop (A), LOHRS at three loops (A), or LOF at a single loop (A).

(A): Failure to delatch for the control rod clutching device due to failure in opening of the reactor-trip breaker
(B): Control-rod-insertion failure with success in opening of the reactor-trip breaker*

*) Circulation pumps in the PHTS and other equipment are automatically tripped when a reactor trip signal is generated owing to a success in opening of the reactor-trip breaker. The event would then progress in a manner of ULOF, when a failure occurs in control-rod-insertion (by mechanical sticking, etc.) under this condition.

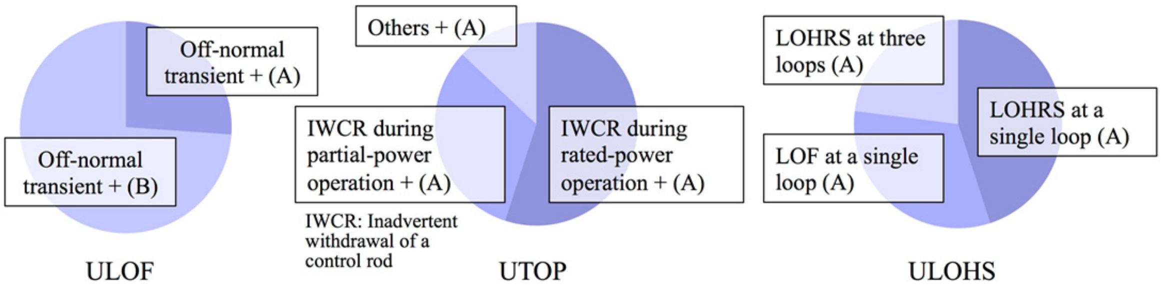


Fig.4.2.4-1 Group-wise contributions to CDF of each accident sequence and identified representative accident sequences for ATWS

4.3 Concept for Preventive Measures against Significant Core Damage

4.3.1 Prevention of Significant Core Damage

Table 4.3.1-1 summarizes the accident sequence groups, criteria for judgment, representative accident sequences, and corresponding preventive measures against core damage. The concept for the core damage preventive measures is as follows.

- ✓ In SFRs, heat removal by natural circulation can be expected as a core damage preventive measure insofar as the coolant path is secured even if the cooling by forced circulation fails. This is essentially different from LWRs, owing to the physical property of sodium. In the accident sequences with sodium leakage, the amount of sodium leakage will be limited in SFRs due to low system pressure and the coolant level neccesary for coolant circulation can be maintained by supplying the sodium to reactor vessel (RV). Requirements to the preventive measures of core damage in each accident sequence group, and examples of specific preventive measures are shown as follows.
  - ➢ In an ATWS, automatic preventive measures against core damage are required to avoid the damage, since any action by operators cannot be expected as the event progression is too fast. Automatic reactor shutdown by the backup reactor shutdown system is effective for this purpose in Monju.
  - ➢ Regarding a LORL event, an event is identified as a representative accident sequence. In this event sequence, the reactor has already been shut down by a precedent leakage in the primary loop and another sodium leakage is assumed to occur at a location below SsL (System sodium Level) in another primary heat transport system (PHTS) loop. The sodium level in the reactor vessel needed for circulation operation can be maintained by the following procedures in this accident sequence. The operator could perform siphon breaking operation at an elevated piping position after the sodium leakage detectection. This prevents pumping-out of the sodium from the RV by siphon effect, and promotes charging sodium flow in the elevated piping back into the RV.

    This operation will be performed by melting the frozen sodium (hereinafter called "Freeze Melt") at the vent-line and injecting the argon gas into the piping at an elevated piping position. It is important to ensure this siphon breaking operation by Freeze Melt and to maintain the sodium level in the reactor vessel.
  - ➢ Concerning LOHRS and SBO events, the temperature increase rate of sodium in the core and at the reactor coolant boundaries is moderate due to large heat capacity of sodium in the systems and of plant structural materials. A sufficient grace period therefore will be granted to significant core damage. Requirements are summarized as follows regarding core damage preventive measures by natural circulation decay heat removal.

- ✧ It is necessary to be able to prevent significant core damage by natural circulation heat removal of at least 1 out of 3 loops, concerning the core cooling capability by natural circulation.
- ✧ It is important to establish measures that ensure transition to a natural circulation heat removal mode, and hence batteries (direct-current power supply) are equipped to enable automatic transition. It is also of importance to prepare the operation procedures for manual transition in the central control room or at the field site, assuming unexpected failure of the automatic transition. For the manual operation of valves at the field site, it is important to enable quick removal of thermal insulator by the measures of packaging of the insulator. To preserve alternative access routes is also of importance even in case of the unavailability of the shortest route by a locking of the door.
- ✧ Preventive measures are required to exclude the sodium flow blockage considering that sodium is solidified at around 100 deg. C. Blockages can be caused by the excessive cooling by the air coolers during natural circulation heat removal, or the sodium freezing after reactor shutdown. Specifically, the operation procedures should be developed to open carefully the vanes and dampers while avoiding the excessive decrease of sodium temperature and securing the sodium flow path, for the manual operation of the air cooler. These procedures should be established and familiarized through training. The sodium temperature decrease can be retarded by totally shutting off the vanes and dumpers of the air coolers, even under the condition that the establishment of the alternate electric power is delayed. These procedures are required to be well prepared in advance.
- ✧ The operation of natural circulation heat removal needs to be continuously measured and monitored. This measurement and monitoring in the central control room will be disabled after full discharge of the batteries, in case of connection failure to the alternate electric power supply or its fuel exhaustion. The sodium temperature, however, can be measured by transportable instrument, even under such a condition, and the needed procedures should be prepared and familiarized through training.
- ✧ Major safety measures for an SBO in Monju are summarized in Table 4.3.1-2. A plant walkdown has been carried out by the advisory committee member for the representative safety measures such as,

  (i) The manual transition operation to the natural circulation mode by operators,

  (ii) The installation of transportable electric power-supply vehicles and substitute sea water pumps for restoration of diesel generator cooling, and

  (iii) The emergency access routes to Monju site, etc..

The results are shown in Table 4.3.1-3. The natural hazards specific to the area of Japan-Sea side should be considered, such as salt damages, wet snows, avalanches, tornados and so on, for the electoric power-supply vehicles and substitute sea-water pumps to be stored outdoors.

- ✓ The following procedures are the same in concept to that described in the safety review guidelines for effectiveness assessments on commercial reactors (LWRs).
  - (i) Representative accident sequences are to be selected by considering characteristics of SFRs,
  - (ii) Progression of events is to be analyzed under the condition that core damage preventive measures are effective, and
  - (iii) Effectiveness of these measures is to be assessed by comparing the results with the criteria for core damage established considering characteristics of SFRs.

  In other words, the characteristics of SFRs should be considered in selecting the representative accident sequences, identifying the core damage preventive measures, and establishing the criteria for judgment on the preventive measures. The concept in the safety review guidelines for LWRs is applicable with some modifications to remaining items of the effectiveness assessment. The criteria for judgment, however, should be specifically established for Monju due to the differences in coolant, cladding, and structure and material of the reactor coolant boundaries.

Table 4.3.1-1 Preventive measures against significant core damage

Accident sequence groups (Outline)	Criteria for judgment on core damage prevention	Representative accident sequences	Core damage preventive measures
(1) Anticipated Transient without Scram (ATWS) Failure to open of the trip circuit breaker in the primary shutdown system at an off-normal condition requiring a reactor trip. / Failure to insert sufficient number of primary control rods under success of trip circuit breaker opening.	(a) Maximum coolant temperature shall be under its boiling point. Moreover, fuel cladding shall not be damaged. (b) Function of the coolant boundaries shall be maintained. (c) Sufficient reactor sodium level shall be secured for decay heat removal. (Whereas, a temporal decrease of the sodium level is allowed, that does not affect the fuel integrity.)	Loss of coolant flow in core. + Failure of primary control rod insertion.	Reactor shutdown by the backup shutdown system.
(2) Loss of Reactor Level (LORL) A decrease of sodium level in the reactor vessel below the EsL (Emergency sodium Level) by multiple leakages at the primary heat transport system (PHTS), etc..		Sodium leakage at the PHTS. + Another leakage at a location below the SsL (System sodium Level) in the PHTS piping.	Interception of pumping-out of the sodium from the reactor vessel (RV) by siphon breaking operation at a elevated piping position.
(3) Loss of Heat Removal System (LOHRS) Loss of decay heat removal function by forced circulation in all the 3 loops due to loss of functions of the blowers, valves, vanes and dampers of the air coolers at the auxiliary cooling system (ACS), or the pony motors of main circulation pumps in the PHTS or SHTS.		Loss of opening function of the outlet stop valves at the air coolers in the ACS.	Natural circulation heat removal by the ACS by opening of the bypass valves at the air cooler outlet stop valves.
(4) Station Blackout (SBO) Loss of decay heat removal functions due to a station blackout.		Loss of off-site power source. + Loss of on-site emergency power.	Natural circulation heat removal by the ACS triggered by an SBO signal.

Table 4.3.1-2 Representative safety measures against a station blackout in Monju.

	Items	Representative safety measures
(1) Securement of electrical power supply.	Deployment of electric-power-supply vehicles and electric-power cables.	- Two of 300kVA electric-power-supply vehicles are deployed at a location where they would not be influenced by tsunamis.
	Installation of electric-power connecting board.	- Common connectors are equipped which enables to share the electric power-supply vehicles deployed at the Mihama nuclear power station of Kansai Electric Power Co. (KEPCO) and Tsuruga nuclear power station of the Japan Atomic Power Co. (JAPC). - Three connection points are available to be selected depending on the debris distribution condition.
	Introduction of an air-cooled electric-power-supply system in substitution for the emergency diesel generator (DG).	- An air-cooled electric-power-supply system (4000kVA, under preparation) is introduced which can substitute the power capacity of one DG, in addition to the electric-power-supply vehicles (300kVA×2).
	Preparation of substitute pumps in the sea water cooling system.	- Substitute pumps (×2) and a power supply system (×1) are prepared in case that the sea water pumps fail by a tsunami.
	Preparation of a substitute electric motor in the sea water cooling system.	- A substitute electric motor (×1) is prepared in case that electric motor fails in the reactor sea water cooling system by a tsunami.
	Strengthening of the floodwall.	- The floodwall is strengthened to prevent submergence of the sea-water pumps at the ACS by a tsunami. (The elevation of the floodwall is 6.4 m above sea level whereas the anticipated tsunami height is 5.2m.)
	Deployment of a wheel loader.	- A wheel loader (×1) is deployed for debris disposal.
(2) Securement of cooling capability.	Packaging of the sodium valve thermal insulator.	- Operability for the manual opening and closing of the sodium valves is improved to be able to quickly transition to or control natural circulation heat removal. (Approx. 15min/valve for opening and closing operation including removal of the thermal insulator.)

	Provision of access routes.	- Emergency ladders (×2) are installed in the air cooler room, which enables movement to upper and lower floors, to be able to access and operate the vanes and dampers in case that the door on the shortest route is blocked. (Multiple access routes are secured.)
	Measurement at the field site	- The status of natural circulation heat removal can be monitored by measuring the sodium temperature in the system at the field site by transportable instruments.
(3) Reinforcement of the emergency responce system.	Reinforcement of the responce organization.	- Responders for connecting the electric-power-supply vehicles, setting up the substitute sea-water pumps, and operating the wheel loader are standing by all day and night.
	Strengthening of communication measures.	- Power supply for the outdoor antenna of satellite phones is backed up. (by an emergency power source.) - The data-transfer lines of the monitoring posts are duplicated.
	Manuals and training.	- Needed operation manuals are prepared for SBO, securing cooling capability, and so on. - The operation procedures are familiarized through training, including exercises at night or under stormy weather.

Table 4.3.1-3(1/2) Results of plant walkdown by the "Advisory Committee on Basic Approach to Securing Safety of Monju" at the Monju site.

	Representative items examined by the field work.	Results of plant walkdown.
Responses by the operators at an SBO in the central control room. (Through the simulator training.)	Monitoring of the plant status in emergency.	- The plant statuses and external situations are monitored and recognized on needed timing, such as monitoring of the tidal level during a tsunami, the status of natural circulation heat removal (the throttling statuses of the valves, vanes and dampers at the air coolers, and the temperatures and flows of sodium coolant), etc.. - The status of natural circulation is adequately monitored on the control panel in the central control room or the fuel handling control room. The operation manuals for the process monitoring by portable circuit testers are well prepared assuming a condition that the electric-power-supply vehicles are unavailable and the batteries are fully discharged.
	Prediction of the event progress. / Judgment on the measures.	- The shift supervisor well predicted the progress of events and commanded the operators to promote the needed preparations. The operators smoothly implemented the manual start-up operation for natural circulation under the assumed situation, in surprise training, that the starting up of the natural circulation failed.
	Start-up operation for natural circulation.	- The transition operation is readily accomplished in the central control room to manually start up the natural circulation operation.
	Mutual coordination (communication) among the operators in emergency.	- The operators smoothly communicated each other despite the quick speaking by using abbreviations and acronyms of the facility.
	Preparatory status of the response manuals. / Training of operators.	- The response procedure manuals for SBO are well prepared, subsequently used with the preceding response manuals for loss of off-site power. - It was explained that the operators were trained to maintain and improve their depth of knowledge. This is

		accomplished by familiarizing themselves with the various emergency responses through simulator training, and acquiring knowledge on the progression of accidental events and corresponding preventive measures, considering features of SFRs through tabletop exercises.
Manual operations of the valves, vanes and dampers at the air coolers in the ACS at the field site.	Access routes from the central control room to the ACS room (A nitrogen-gas injected area under a sodium-leak condition).	- Two access routes are secured to the ACS room. Another access is also possible from outside the building (via the turbine building roof). - Unlocking is necessary to open the door to access the ACS room. The key is managed by the shift supervisor in a unified manner as one of the keys for the areas filled with nitrogen-gas. Thus, a mis-unlocking seems hardly to occur.
	Manual opening and closing operation of the sodium valves including removal of the thermal insulator.	- Removing operation of the thermal insulator is simplified due to its packaged structure. The amount of time required for operation is conservatively estimated to be approx. 10 minutes, including manual opening operation using the handle. - Additional scenarios should be developed step by step for simultaneous and multiple events, whereas, in the basic scenario, the piping room of the leaked loop will be blocked and filled with the nitrogen gas, excluding its contribution to natural circulation heat removal.
	Manual opening and closing operation of the dampers at the air cooler.	- The damper handles are located at a shoulder-level of the operators and can be operated readily.
	Upper part of the air cooler outlet duct (heat release point to the atmosphere.)	- Atmospheric temperature measurement can be used for monitoring the status of natural circulation at the upper part of the air cooler outlet duct as a backup measure. - The condensed water would be accumulated at around the air cooler outlet duct, although there are draining holes. The material integrity should be confirmed.

Table 4.3.1-3(2/2) Results of plant walkdown by the “Advisory Committee on Basic Approach to Securing Safety of Monju” at the Monju site.

	Representative items examined by the field work.	Results of plant walkdown.
Electric-power-supply vehicles and electric-power -connecting board.	Deployment statuses of the electric-power -supply vehicles and the electric-power -connecting board.	- The electric-power-supply vehicles are deployed at a location where they would not be influenced by tsunamis. The influences should be also investigated of tornados and wet-snow avalanches. Salt damages and corrosions are questioned from the viewpoint of long-term storage. - Emergency responders are always standing by on the Monju site even at night and on holidays, and they will be able to conduct the connecting and power-supplying operation using the electric-power-supply vehicles of 300kVA. - Three connecting points are available for the electric-power-supply vehicles of 300kVA allowing flexible choices depending on the situation. - Main specifications of the connecting board are common to that deployed in the Mihama nuclear power station of KEPCO and Tsuruga nuclear power station of the JAPC, which enables sharing of the power-supply vehicles with the neighboring power stations. - The electric-power-supply vehicle of 4000kVA is under preparation. No connecting operation will be needed because its power-supply cables will be permanently buried. - The measures against landslides are in progress. The possibility of avalanches caused by a combination of heavy snows and earthquakes should be also investigated (including their occurrence at a higher location which is out of sight from the site). -
Substitute sea-water pumps to resume one of the emergency	Status of deployment and location of the substitute sea water pumps.	- The substitute sea-water pumps, including related equipment, are deployed (stored) at a location where they would not be influenced by tsunamis. Salt damages and corrosions are questioned from the viewpoint of long-term storage. The influences should be also investigated of tornados and wet-snow avalanches.

diesel generators (DGs) after a tsunami.		- Emergency responders are always standing by on the Monju site to restore the pumps, including at night and on holidays. - Preparation of the extra pumps is confirmed against sea-water flooding in the underground-piping area.
Floodwall installation around the sea-water pumps.	Installation status of the floodwall.	- Installation of the floodwall with a height of 1.2m is confirmed. - It was confirmed that further raising of the floodwall height and measures against undertows were under investigation.
Wheel loader.	Deployment status of the wheel loader.	- A wheel loader for debris disposal is deployed at a location where it would not be influenced by tsunamis. Salt damages and corrosions are questioned from the viewpoint of long-term storage. The influences of tornados and wet-snow avalanches should be also investigated. - Emergency responders are always standing by on the Monju site to operate the wheel loader, including at night and on holidays.
Access routes to the Monju site.	Accessibility in case that the tunnel to the Monju site is blocked.	- Redundant access routes are confirmed along the sea coast and in the mountainside. The routes should be properly established in accordance with their purposes.
Others		- Smooth alternation of generation is required for a long-term project, such as Monju, based on the experience at the Fukushima Dai-ichi nuclear power plant of TEPCO. It is recognized to be of importance to assign experts of a wide variety of specialties, such as mechanics, electrotechnics, etc., with the knowledge of the plant from its construction phase, for the adequate response to accidents. - Confirmation is needed for the Monju dormitory, in which emergency responders are lodging and located near the Shiraki-Niu fault, regarding its earthquake resistance and furniture-fixing statuses.

4.3.2 Prevention of Fuel Damage in Spent Fuel Storage Tanks.

Table 4.3.2-1 summarizes the accident sequence groups, criteria for judgment, representative accident sequences, and corresponding preventive measures against fuel damage in spent fuel storage tanks.

The concept for preventive measures against damage of spent fuels is as follows.

- ✓ Spent fuels from SFRs are stored in the sodium in an Ex-Vessel fuel Storage Tank (EVST) until the decay heat decreases sufficiently after the removal from the reactor. The cooled spent fuels are subsequently stored and cooled under the water in a spent fuel pool. Preventive measures against damage of spent fuels in an EVST and spent fuel pool are as follows.
  - ➢ Natural circulation heat removal can be expected during an accident to raise the sodium temperature in an EVST, even in case that a loss of its decay heat removal by forced circulation occurs. The requirements for natural circulation heat removal are the same as that for the core in principle.
  - ➢ In regard to accidents to lower the sodium level in an EVST, penetration holes shall be located at a height sufficiently higher than the necessary sodium level for cooling when installing them for piping through the vessel wall, and thus the sodium level shall be maintained higher than the necessary level for cooling, even assuming a small-scale leakage caused by damage at a penetration part of piping.
  - ➢ The concepts can be also applied to SFRs for safety effectiveness assessment on spent fuel storage pools established in LWRs, because their system arrangements, representative accident sequences and fuel damage preventive measures are independent of reactor types.
- ✓ The safety review guidelines for LWRs are also applicable with some modifications to safety effectiveness assessment on preventive measures against significant fuel damage for spent fuel storage tanks in the same manner as that for the reactor cores. The criteria for judgment, however, should be specifically established for the Monju EVST due to the differences in coolant.

Table 4.3.2-1 Preventive measures against significant fuel damage in spent fuel storage tanks.

<table>
<tr><th colspan="2">Accident sequence groups
(Outline)</th><th>Criteria for judgment on fuel damage prevention</th><th>Representative accident sequences</th><th>Preventive measures against fuel damage</th></tr>
<tr><td rowspan="2">(1) EVST</td><td>- Accidents to raise sodium temperature.

Events to raise the sodium temperature in the EVST caused by a loss of its heat removal function.</td><td rowspan="2">(a) Function at the reactor coolant boundaries shall be maintained.

(b) Sodium level shall not be lowered below the needed level for heat removal (EL 32.9m).</td><td>Failure of heat removal by forced circulation caused by loss of functions in all the EVST circulation pumps.</td><td>Heat removal by natural circulation using the EVST heat removal system.</td></tr>
<tr><td>- Accidents to lower sodium level.

Events to lower the sodium level in the EVST caused by a small-scale leakage at a penetration part of piping through the vessel wall.</td><td>Small-scale sodium leakage at a penetration part of the overflow piping in the EVST.</td><td>(No additional measures are deployed because the sodium level is secured by itself higher than the necessary level for heat removal (EL 32.9m).)</td></tr>
<tr><td>(2) Spent fuel pool</td><td>- Accidents to raise water temperature.

Events to raise the water temperature caused by loss of heat removal function or water supply function in a spent fuel pool, resulting in lowering the water level by evaporation.</td><td>(a) Top of active height of fuels shall be immersed under the water.

(b) Water level necessary for radiation shielding shall be maintained.

(c) Sub-criticality shall be maintained.</td><td>Loss of functions at all the pumps in the spent fuel pool cooling and purification systems.
+
Failure of heat removal caused by loss of make-up water system function.</td><td>Maintaining the water level by substitutional water pouring. (By the on-site firefighting team.)</td></tr>
</table>

4.3.3 Prevention of Fuel Damage Stored in the Core during Reactor Shutdown.

Table 4.3.3-1 summarizes the accident sequence groups, criteria for judgment, representative accident sequences, and corresponding preventive measures against damage of fuels stored in the core during reactor shutdown.

The concept is as follows for preventive measures against the fuel damage.

- ✓ System sodium temperatures and decay heat level during reactor shutdown are lower than those during rated-power operation. The sodium level in the reactor vessel is decreased from the normal sodium level at the rated-power operation (hereinafter called “NsL”), and the number of available cooling loops is limited during a maintenance mode. The preventive measures against damage of fuels in the core under such conditions are as follows, considering the plant conditions mentioned above.
  - ➢ The core can be maintained at a sub-critical state even in case of an irregular control rod withdrawal owing to its well-prepared designs and operational procedure limitations. Thus, additional preventive measures against fuel damage are not required against unintended reactivity insertion accidents.
  - ➢ With respect to a leakage of reactor coolant (LORL), the sodium level in the reactor vessel can be maintained above a level at which heat removal by the maintenance cooling system is available. Thus, decay heat removal by the maintenance cooling system is effective as a preventive measure against fuel damage, even assuming a decrease of the sodium level below that necessary for heat removal by the PHTS (hereinafter called “EsL”).
  - ➢ Concerning LOHRS and SBO, natural circulation heat removal by the ACS is effective. The concept of its effectiveness assessment is the same as that during the rated operation.
- ✓ The concepts and approaches defined in the safety review guidelines for LWRs are also applicable with some modifications to safety effectiveness assessment on preventive measures against damage on fuels in the core during shutdown. The criteria for judgment, as is obvious, should be specifically established for Monju considering the differences in coolant, cladding, and structure and material of the reactor coolant boundaries.

Table 4.3.3-1 Preventive measures against significant damage on fuels stored in the core during reactor shutdown.

<table>
<tr><th>Accident sequence groups (Outline)</th><th>Criteria for judgment on fuel damage prevention</th><th>Representative accident sequences</th><th>Fuel damage preventive measures</th></tr>
<tr><td>(1) Unintended reactivity insertion

Reactivity insertion by an irregular control rod withdrawal during reactor shutdown.</td><td rowspan="4">(a) Core shall be maintained at a sub-critical state. (Whereas, a critical state during normal operation or that with a temporal and slight power increase without any influence on the fuel integrity is allowed.)

(b) Sodium level in the reactor vessel shall be maintained above a level necessary for decay heat removal. (Whereas, a temporal decrease of sodium level without any influence on the fuel integrity is allowed.)

(c) Function at the reactor coolant boundaries shall be maintained.</td><td>Irregular and simultaneous withdrawal of 2 control rods.</td><td>(The core is maintained at a sub-critical state with the residual 17 control rods being fully inserted. Thus, no additional measures are required.)</td></tr>
<tr><td>(2) Leakage of coolant (LORL)

Loss of reactor level caused by a piping damage at loop B during maintenance of loops A and C.</td><td>Sodium leakage at a location lower than the SsL in loop B during maintenance of loops A and C.</td><td>Decay heat removal by the maintenance cooling system.</td></tr>
<tr><td>(3) LOHRS

Loss of decay heat removal function by forced circulation caused by simultaneous failures of the air cooler blowers at all the 3 loops of the ACS, and pony motors in the PHTS or SHTS pumps, the same sequence of events as in that during power-generating operation.</td><td>Failure of forced circulation heat removal at loop B during maintenance of loops A and C.</td><td>Natural circulation heat removal by loop B using the ACS.</td></tr>
<tr><td>(4) SBO

Loss of decay heat removal function caused by an SBO, the same sequence of events as in a LOHRS.</td><td>Failure of forced circulation heat removal caused by an SBO (Loss of off-site power + Loss of on-site emergency power)</td><td>Natural circulation heat removal by loop B using the ACS.</td></tr>
</table>

4.4 Concept for Ensuring Containment Function

4.4.1 Anticipated Transient without Scram (ATWS)

(1) Basic concept.

In an ATWS event, fuel pins would be damaged and mechanical energy would be released by a power excursion induced by a prompt criticality. The containment function of sodium shall be maintained at the reactor coolant boundaries under such conditions. This function was once confirmed to be maintained in a safety assessment of the original licensing procedure for Monju. On the other hand, a possibility of ejected sodium combustion is conceivable by mechanical energy release and then its appropriate assessment is required.

An additional load, moreover, would be put on the containment vessel (CV) in case that the damaged fuel and other core materials are not retained within the reactor vessel (RV). Core damage caused by an LOHRS event, on the contrary, would be able to be prevented depending on the adequacy of the countermeasures, to be discussed in the next section.

Thus, it is important to evaluate in-vessel retention (IVR) behaviors of the damaged core and to clarify the occurrence possibility of an additional load to the CV during an ATWS event. The evaluations focusing on them are presented in this section.

The evaluations shall be performed in accordance with the following principles.

(a) Uncertainties in the actual phenomena shall be adequately considered.

(b) Uncertain phenomena occurring in a realistic event progression shall be evaluated by combining conservative approaches or models and others, considering the sensitivities of analytical models.

The most likely event progression in ULOF is prospected as follows based on the most up-to-date knowledge. A ULOF event is initiated by a failure of reactor trip at loss of off-site power resulting in a non-functioning of the reactor shutdown systems. Subsequently, the event progresses along with the following phases without a prompt criticality and mechanical energy release: (a) initiating phase, (b) transition phase, (c) post-accident-material-relocation (PAMR) phase, and (d) post-accident-heat-removal (PAHR) phase.

A large amount of sodium far below its boiling point is available in SFRs, insofar as the integrity of the reactor coolant boundaries are kept. Thus, it is possible at PAHR phase to remove the decay heat from the molten and dispersed core materials in the RV by the sub-cooled sodium. The decay heat then can be released to out of the reactor facility by the HTSs. Therefore, the event would be terminated without any damage on the reactor coolant boundaries, retaining the molten-core materials in the RV.

In the present evaluation, the uncertainties sthat may have influences on the event progression were considered within an physically reasonable extent. Possibility of mechanical energy release was evaluated together with its influences on stable cooling of the debris in the RV. Figure 4.4.1-1 shows an outline of the entire event progression.

(2) Most probable event progression.

(a) Initiating phase.

A up-to-date analytical tool, SAS4A, was employed for the analysis of the initiating phase. The sub-assemblies in the core are lumped depending on the similarity of the power and flow rate as shown in Figure 4.4.1-2. All the fuel pins in each group are represented by a pin with which the residual pins are assumed to behave identically. The behaviors are analyzed for these representative pins in each group concerning their coolant boiling, fuel-pin deformation failure, and fuel motion after the failure. The whole core damage behaviors are analyzed by integrating these pin-wise behaviors in the code during the initiating phase, more specifically, until the sub-assembly wrapper tubes are damaged.

The adequacy of analytical conditions in the evaluation has been confirmed as follows. A number of experimental data has been acquired on the fuel-pin-failure behavior by actually damaging fuel pins in the CABRI reactor in France. Quantitative understanding of the phenomena has been made available through this series of experimental researches and analyses and the degree of uncertainty has been rationalized in setting the analytical conditions for the evaluations of the initiating-phase event progression. The most probable case adopted above-mentioned adequately rationalized analytical conditions validated based on the CABRI experimental data.

Core components are grouped into multiple representative “channels”, as mentioned before, in the SAS4A analysis on the initiating-phase event progression. That is to say, core components such as fuel sub-assemblies are grouped into a certain number of representative channels depending on the similarity of their characteristics especially focusing on the “power”, “flow rate” and “burn-up“. In the present evaluation, a 33 channel division was employed so that each characteristic of fuel sub-assemblies were reflected in detail.

Specifically, 198 core fuel sub-assemblies in Monju were grouped into 32 representative channels depending on their power, coolant flow rate and burn-up as shown in Figure 4.4.1-3. Radial blanket fuel sub-assemblies were represented by a single channel. The evaluation was performed using a 1/3 sector model considering 120° rotational symmetry in the core layout.

Table 4.4.1-1 summarizes the rationalized analytical conditions based on the findings derived from the CABRI experiments. The axial location of pin failure in the sub-cooled (non-voided) sodium region significantly affects the possibility of a rapid power excursion induced by the super prompt criticality in the initiating phase. The axial location of pin failure in Monju is estimated to be higher than the core mid-plane height where the cladding temperature is relatively high and its strength is decreased. This is based on the latest knowledge derived from the CABRI experiments. Reactivity insertion by molten-fuel motion from the disrupted fuel pins should be negative under this condition. Such conditions were considered in the present evaluation.

Figure 4.4.1-4 presents analytical results by the SAS4A code on the transient behaviors of reactivity and power during the initiating phase. The maximum reactivity appears at 25s after the inception of the event, due to an increase in reactivity by the coolant boiling and molten-cladding dispersion. However, it does not exceed $1 and the maximum power remains below approx. 15 times of the rated one. Thus, the event moderately progresses without any mechanical energy release.

The reactivity then decreases according to molten-fuel dispersion after that. Both the reactivity and power are stabilized at approx. 27s after the onset of the event. These disrupted but stabilized core conditions at this timing (27s) by the SAS4A analysis was transferred to the subsequent analysis on the event progression during the transition phase.

(b) Transition phase.

A latest 3-dimensional analytical code, SIMMER-IV, was used in the transition-phase analysis. A conceptual diagram of the code is shown in Figure 4.4.1-5. The code intimately combines the following three major modules to provide a 3-dimensional neutronic-thermal-hydraulic-and-structural analytical capability in a systematic manner. A thermal-hydraulic module, the key component of the code, simulates the damaged core as a multi-phase, multi-component, and multi-velocity-field system to solve its fluid-dynamics. A neutronic module solves the space-dependent reactor kinetics depending on the material-distribution and temperature. A structural module analyzes the melting and damaging behavior of the structural materials. A systematic validation of the code has been performed in Japan under collaboration with the European research institutions.

The code has a 3-dimansional capability to simulate the molten-core material behavior realistically. The conventional 2-dimensional and in cylindrical coordinate code, SIMMER-III, has no such a capability to simulate spatially incoherent behavior of the materials. Axisymmetric and coherent fuel compaction toward the core center was inevitable in 2-dimensional cylindrical modelling. Moreover, molten-fuel discharge through the damaged control rod guide tubes (CRGTs) can be realistically simulated, which significantly affect the negative reactivity feedback effects during the transition phase.

Furthermore, findings obtained through the EAGLE experiments, performed in the National Nuclear Center (NNC) in Kazakhstan, were also reflected in the analytical conditions. Findings on the following phenomena, in particular, would dominantly determine the transition-phase event progression.

(i) Heat flux from the molten-core materials to the surrounding structural-material walls such as CRGTs.

(ii) Pressure generation behavior by fuel-coolant interaction (FCI: an interaction between the sodium inside a CRGT and molten-core materials surrounding it) at the moment of CRGT failure due to a thermal load from the surrounding molten-core materials.

Analytical results by SIMMER-IV on the event progression during the transition phase are shown in Figure 4.4.1-6. Increases in reactivity and reactor power are observed, in the early phase, caused by falling down of the molten fuel located at an upper part above the core during the initiating phase. The reactivity however transitions below $1 and the maximum power remains below approx. 20 times of the rated one. CRGTs are damaged and molten-fuel discharge through them is initiated simultaneously with these transients at approx. 2.5s after the onset of the transition phase. The reactivity is then decreased to a deep sub-critical state, approx. -$30, due to this molten-fuel discharge from the core. Thus, the event progression terminates without any mechanical energy release.

(c) PAMR and PAHR phases.

The molten-core materials, formed in the core, relocates to the low pressure plenum (LPP) through the CRGTs with neither the prompt criticality attainment nor the mechanical energy release, as mentioned above, in the most likely evaluation for the initiating and transition phases.

The molten-core materials discharged through the CRGTs are quenched and fragmented only at an earlier stage due to the limited sodium inventory in the LPP in the most likely case. And others discharged later are not fragmented, but accumulated after sodium dry-out in the LPP.

The amount of molten-core materials relocated to the LPP is approx. 20% of the core total inventory in the evaluation mentioned in the previous paragraph. Evaluations are to be carried out hereinafter under the above-mentioned condition for the PAMR and PAHR phases. On the other hand, it has been clarified that the molten-core materials can be stably cooled and retained in the LPP even though the amount is 50% of the original inventory. This is based on the results of an uncertainty evaluation, to be mentioned later, performed prior to the above mentioned evaluation.

The thermal load to the LPP in the most likely event progression is therefore below the half of that used in the uncertainty evaluation, because the amount of molten-core materials of 20%, relocated to the LPP, is below the half of that of 50% used in the uncertainty evaluation. The conclusions of the forthcoming uncertainty evaluation were that molten-core materials of 50% could be cooled and retained in LPP. Thus, it is concluded that the discharged molted-core materials from the core to the LPP would be stably cooled and steadily retained in it, and IVR of the molten core is attainable in consequences of a CDA.

(3) Evaluation on the influences of uncertainties dominant to the most probable event progression.

(a) Initiating phase.

(i) Uncertainties in nuclear parameters.

Evaluations were conducted focusing on the uncertainties in sodium void and Doppler reactivity, which would significantly affect the magnitude of reactor power excursion. Conclusions have been derived that uncertainty envelopes of ±20% for sodium void and

±14% for Doppler reactivity would be sufficient at the confidence level of 2σ, based on the most-recent knowledge of neutronics calculation.

The following factors were then assumed for each parameter referring to the nominal(best-estimate)-value-based most likely evaluation:

a) Positive part of void reactivity: 1.2 times of the nominal value.

b) Negative part of void reactivity: 0.8 times of the nominal value.

c) Doppler reactivity: 0.86 times of the nominal value.

As a result, it was indicated that the maximum reactivity transitions below approx. $0.95, and the maximum core-fuel average temperature remains approx. 2700K, resulting in no mechanical energy release.

(ii) Uncertainties in material motion.

Parametric surveys were performed to evaluate influence of uncertainties in the material motion modeling in SAS4A. The code has been validated by using the existing database of molten-fuel dispersion behavior acquired by the CABRI experiments conducted under a variety of fuel-pin-failure conditions. The following conditions are assumed.

a) Reducing the amount of accumulated FP gas in the disrupted fuel pin to 50% in order to reduce the driving force of the molten-fuel dispersion that provides negative reactivity feedback after thefailure of fuel pin.

b) Reducing the cladding strength to 50% in order to accelerate rapid increase of the void reactivity by the fuel pin failures in the sub-cooled sodium region.

c) Assuming falling down of fuel stub leading to a significant positive reactivity insertion.

The above-mentioned c) falling down of fuel stub means such a phenomenon that the solid fuel, remained un-melted at an upper part above the disrupted fuel region during a power transient in the initiating phase (hereinafter called “fuel stub”), falls down driven by the gas plenum pressure after losing its binding force by the cladding due to the temperature increase. This phenomenon has been observed in the CABRI in-pile experiments but only under the conditions that the fuel is melted at the central part of the core while the coolant is not or only partially boiled.

This is not the case with Monju where loss of cladding strength precedes fuel-pin failure due to early and prevailing coolant boiling in most part of the core region. Falling down of fuel stub was not observed under such a condition in the experiments. This condition therefore does not need to be considered as an uncertainty but was assumed just to be sure as a hypothetical condition considering its large positive reactivity feedbacks.

The results are as shown below and no mechanical energy release was confirmed.

	Maximum reactivity	Maximum core-fuel average temperature
a)	$0.85	2500K
b)	$0.85	2400K
c)	$1.0	2900K

(b) Transition phase.

The mechanical energy release in the transition phase is caused by the molten-fuel compaction in the core region. The uncertainty in this mechanically released energy was evaluated to a maximum extent by the following assumptions. The amount of molten fuel discharged from the core region was intentionally restricted to maximize that in the core, which contributes to the energy to be released. The amount of sodium entrainment at the CRGT failure was assumed to be 20g, a conservatively estimated value for the envelope evaluation, in contrary to the most likely one of 0.5g. An analysis was conducted under these conditions enhancing the molten-fuel compaction behavior in the core region due to its sloshing motion.

The transient changes in reactivity and reactor power of this analysis are depicted in Figure 4.4.1-7. The results show that the reactivity exceeds $1 and the reactor power exceeds approx. 100times of the rated one due to the molten-fuel compaction driven by sodium vapor pressure when the CRGT failure occurs at approx. 2.5s. A mechanical energy of approx. 35MJ is released consequently as a potential work assuming an isentropic expansion of the molten fuel to the atmospheric pressure.

(c) In-vessel response phase.

As a consequence of the re-criticality, presented in the transient-phase analysis mentioned in the previous paragraph, by cumulating possible uncertainties conservatively, the sodium in the upper plenum of the reactor vessel (henceforth called “sodium slug”) is accelerated upward due to increases in temperature and pressure of the molten-core materials. The accelerated sodium slug may put a mechanical load to the reactor coolant and cover-gas boundary by impacting the bottom of the shielding plug and the reactor vessel wall. An analysis on this series of core expansion behaviors was carried out using the SIMMER-III code. Figure 4.4.1-8 shows the conceptual diagram of the reactor vessel with its internal structures in a vertical sectional view together with its analytical modeling.

A mechanical energy of 150MJ was assumed as an isentropic expansion potential[1)] to the atmospheric pressure based on the results of a conservative transient-phase analysis by SIMMER-III. Only upward acceleration was considered of the sodium slug assuming the bottom and side walls of the core to be a rigid boundary. This assumption limits the expansion of the molten-core materials only to the upward direction. Formation of a large flow path was moreover assumed at the central region of each sub-assembly caused by a deformation of pin bundles being pressed toward the wrapper tubes at an above-the-core position due to the pressure generation in the core.

These hypothetical conditions are assumed to maximize the possible acceleration of the sodium slug considering that the uncertainty is large in geometry of the disrupted core. Figures 4.4.1-9 and -10 present the results of the analysis. The cover-gas volume is compressed to a minimum value of approx. 30m^3 at approx. 0.5s after the initiation of core expansion and no impact of the sodium slug is identified. The upward acceleration of the sodium slug occurs several times due to the generation of sodium vapor caused by an interaction between the sodium in the upper plenum and the high-temperature molten-core materials released from the core. However, the sodium slug does not impact with the shielding plug, and the molten-core materials are cooled by sodium approx. 2s later. Three peaks are observed in the cover-gas pressure with a maximum pressure of less than approx. 6 atm. And thus, deformations of the reactor vessel are within its elasticity.

Approx. 80% of the initial core fuel inventory is released to the upper plenum and accumulated at the bottom of the upper plenum in a debris form. The bottom of the upper plenum has an area of several times larger than that of the LPP. Thus, the accumulated core fuel is estimated to be stably cooled.

The mechanically released energy is evaluated to be approx. 35MJ as an isentropic expansion potential in the conservative transient-phase analysis, whereas the released energy is assumed to be 150MJ in the above-mentioned in-vessel core expansion behavior analysis. Thus, the integrity of the reactor coolant and cover-gas boundaries is sufficiently maintained with larger safety margins due to the much smaller mechanical load. An event progression evaluation of unprotected transient over power (UTOP) events is being required to confirm that the event progression of ULOF envelopes that of UTOP.

(d) PAMR and PAHR phases.

An evaluation was conducted assuming that 50% of the molten-core materials are discharged to the LPP to investigate an influence of uncertainties on event progresses during the PAMR and PAHR phases. In the evaluation, it is assumed that the ACSs and pony motors are normally actuated. Flow velocity field around the molten-core materials was evaluated under the total flow rate determined by the Super-COPD code as a boundary condition.

The evaluation was performed by a thermal-hydraulic analysis code, FLUENT, analytically modeling the geometry as shown in Figure 4.4.1-11. The sodium flow velocities of approx. 18cm/s and 0.4cm/s were obtained at the upper surface of the molten-core materials and undersurface of the core catcher, respectively.

A heat balance evaluation was carried out with a 1-dimensional model shown in Figure 4.4.1-12 with the obtained sodium flow velocities around the LPP as boundary conditions. The molten-core materials were divided into 10 regions in thickness to consider density-induced separation between fuel and steel. The molten-structural materials are assumed to be absorbed into the molten-core materials in case that the boundary temperature at the upper surface of the LPP bottom plate exceeds the melting point of the structural materials, where the molten-core materials are contiguous.

Heat removal capability was evaluated for the upper surface of the molten-core materials and the undersurface of the core catcher using the Nusselt-number-dependent correlation for heat transfer in a laminar-flow boundary-layer of sodium. Creep deformation was evaluated using the cumulative-damage-fraction method for the bottom plate of the LPP induced by its own and retained-molten-core material weights. Temperature distributions in the plate were considered, with which the molten-core materials are in contact. The judgment was made on stable cooling and retention according to this creep deformation. The plate was judged to be failed when the displacement exceeds approx. 100mm (corresponding to a cumulative damage of approx. 0.02 for Monju).

The mechanical strength was assumed to be lost at over 950 deg. C in the cumulative damage evaluation. Figure 4.4.1-13 shows the results. The heat from the molten-core materials can be stably removed and the materials can be steadily retained in the LPP under forced circulation operation by the pony motors. Thus, it can be concluded that IVR of the molten core is attainable as a consequence of a CDA. The heat removal capability should be also investigated of the molten-fuel remained in the core as an issue to be solved in the future.

(e) Summary

The present evaluation revealed that IVR of the disrupted core is sufficiently attainable not only in the most likely event progression but also in the most conservative one in which the utmost uncertainties are considered in the dominant reactor physics parameters, such as void reactivity, etc., and in the most-influential physical phenomena, such as FCI. It is also indicated that a grace period of several hours is available until the structure is disrupted that contributes to cooling down and retaining of the damaged core materials, if it is assumed with hypothetical degradation of cooling function during PAHR phase.

Reliability can be enhanced concerning the IVR of the molten core as a consequence of a CDA, by establishing the accident management measures such as operation procedures to manually actuate the pony motors using this grace period.

Table 4.4.1-1 Rationalized analytical conditions based on findings derived from the CABRI experiments

Items	Past licensing and reference analysis	Present analysis
Fuel pin failure in sub-cooled (non-voided) sodium region.	- Failure at melting of fuel with cross-sectional area of 50%.	- Cladding failure and fuel extrusion are judged by cladding strength and cavity pressure in pin.
Reactivity insertion by molten-fuel extrusion from disrupted fuel pins.	- Negative reactivity insertion is ignored and rapid axial expansion of ruptured area is assumed.	- Negative reactivity insertion is considered by molten-fuel extrusion.

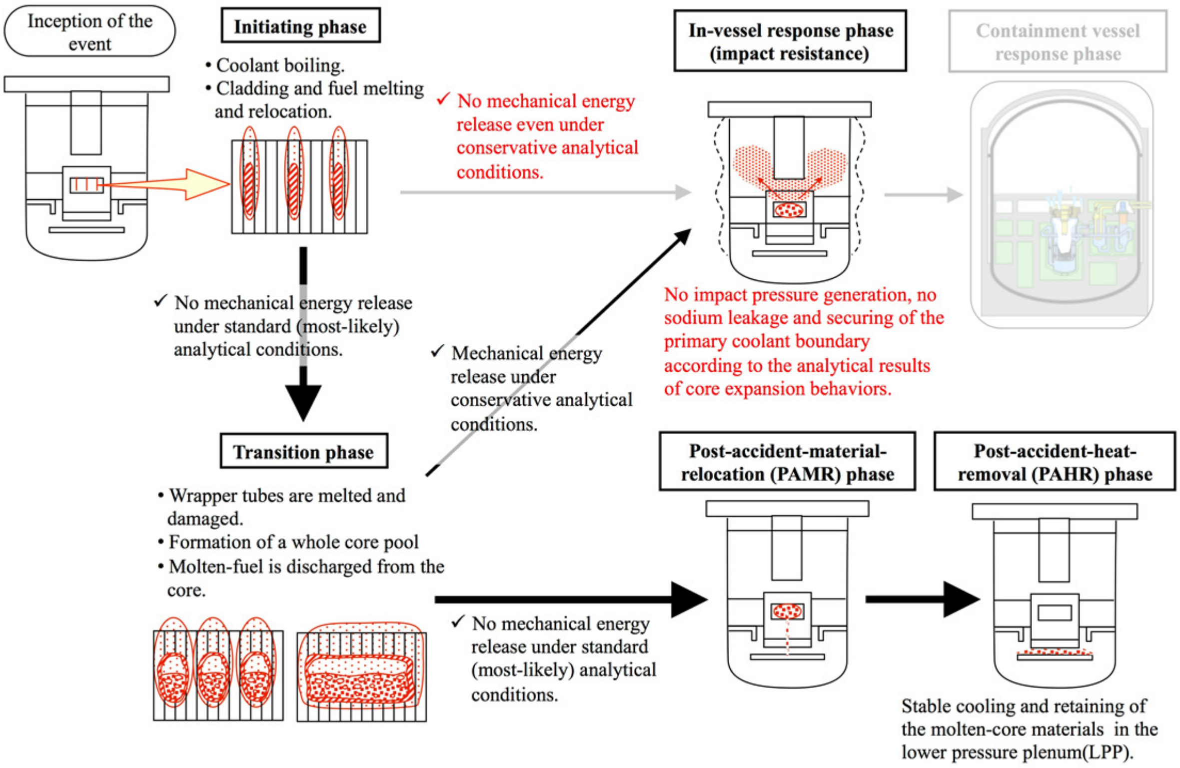


Fig.4.4.1-1 Event progression of ULOF

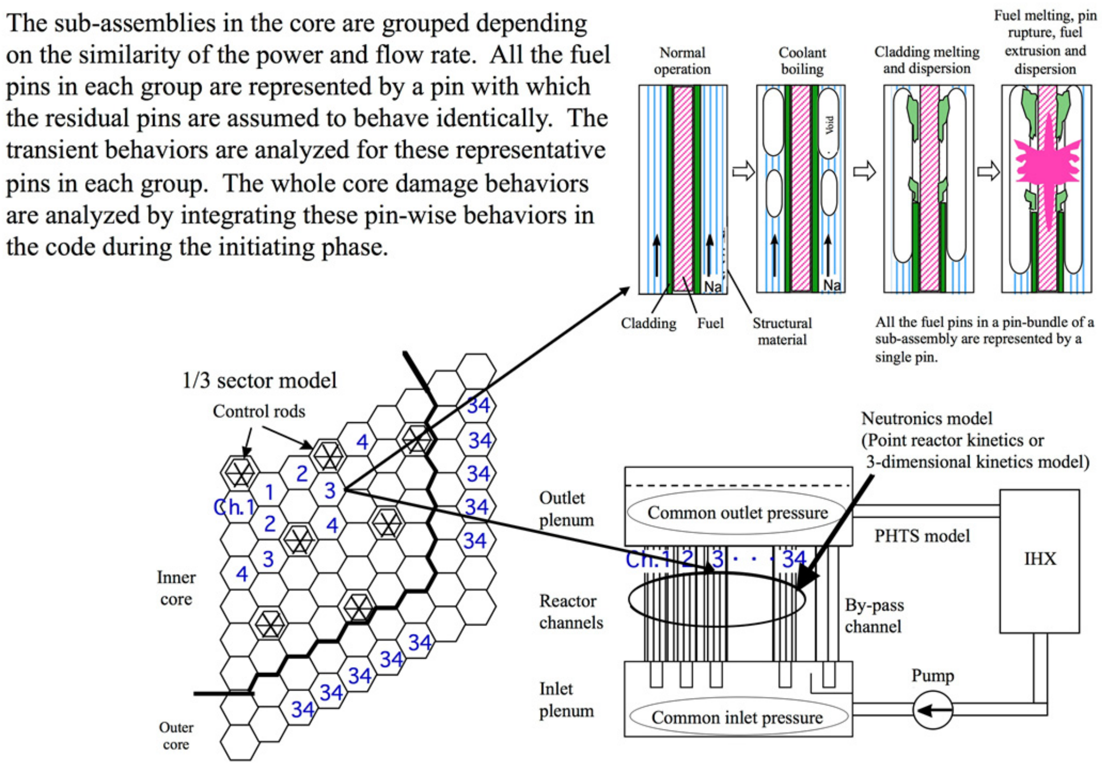


Fig.4.4.1-2 Outline of the SAS4A code

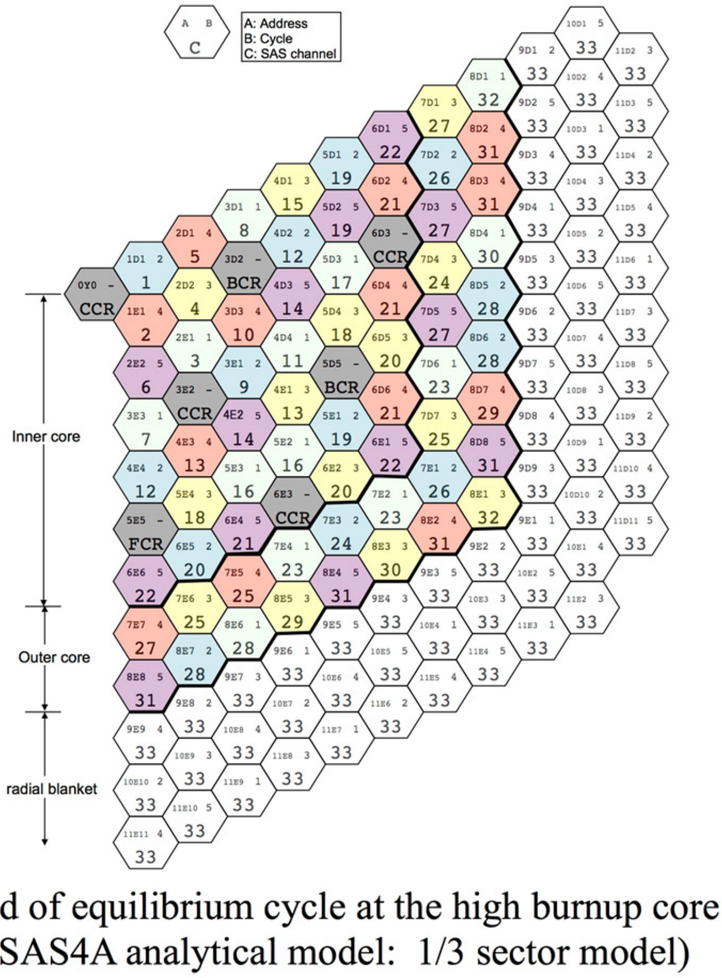


Fig.4.4.1-3 Grouping of fuel sub-assemblies into representative channels for event progression analysis during initiating-phase in Monju

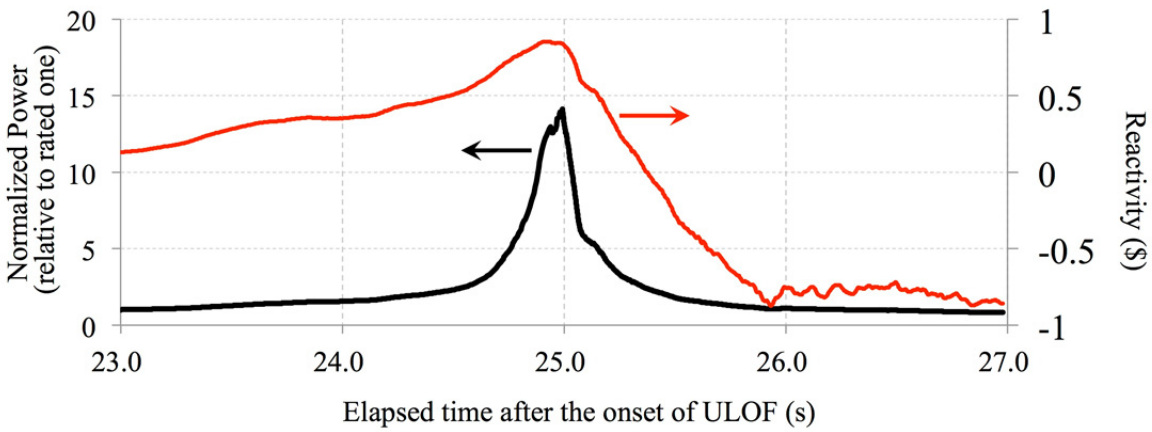


Fig.4.4.1-4 Power and reactivity transient in the most-likely analysis during the initiating phase in Monju by SAS4A.

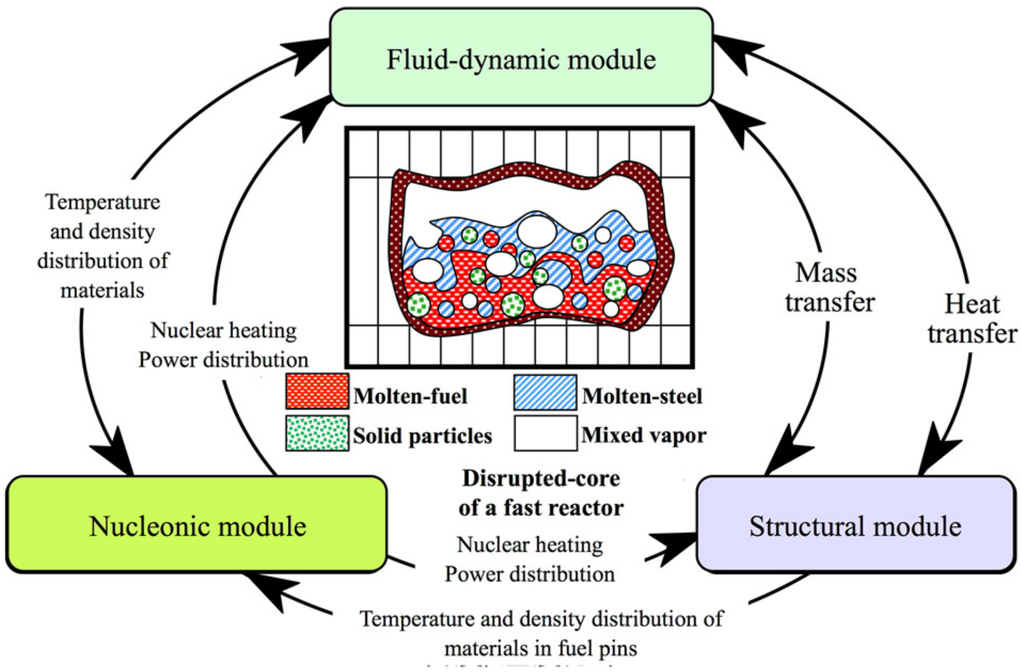


Fig.4.4.1-5 Conceptual diagram of the transition-phase analytical code, SIMMER-III and -IV

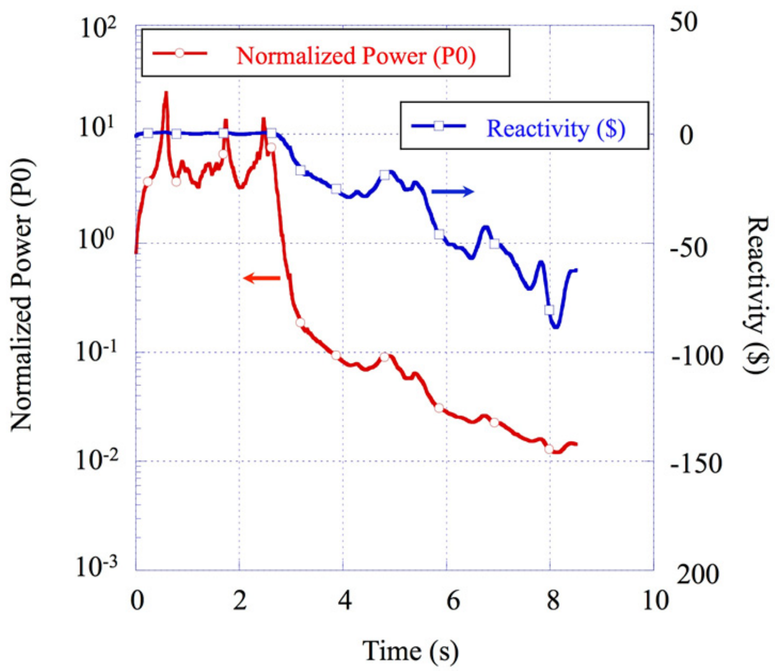


Fig.4.4.1-6 Transient of reactivity and power during transition phase derived by SIMMER-IV analysis on event progression in Monju.

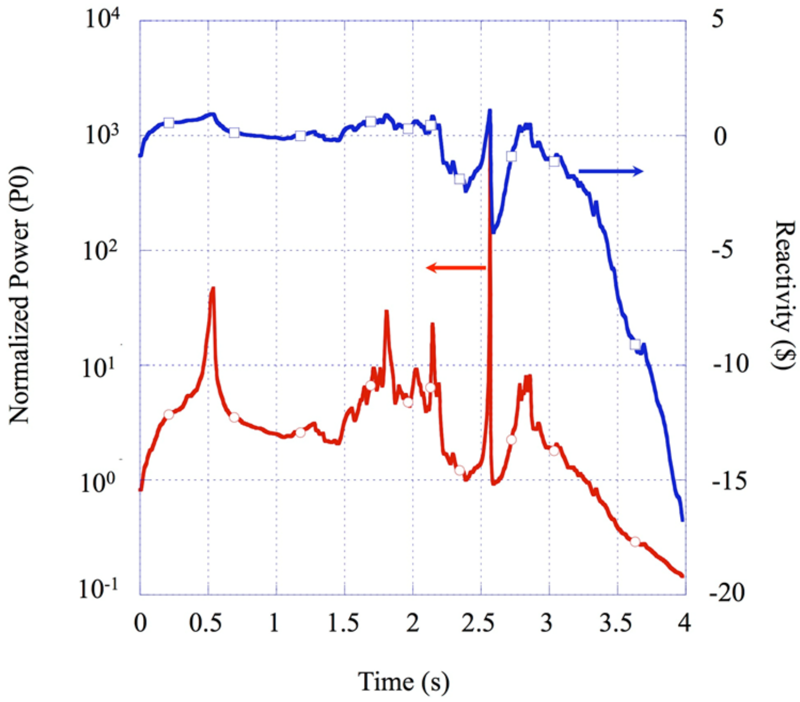
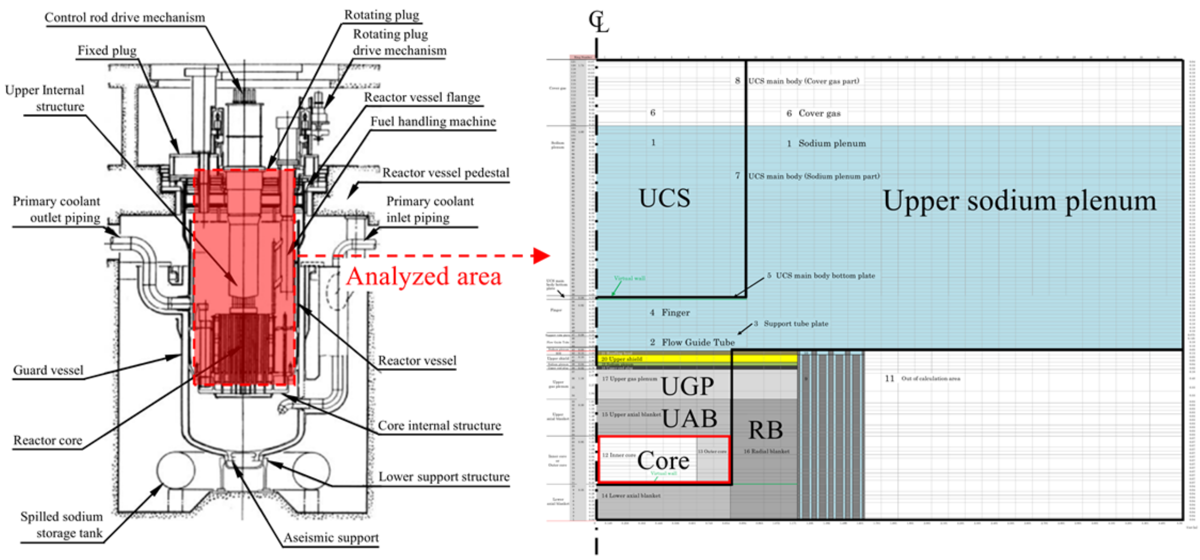


Fig.4.4.1-7　Transient of reactivity and power during transition phase derived by a conservative SIMMER-IV analysis on event progression in Monju.



Conceptual diagram of the reactor vessel　　　Analytical model (2 dimensional R-Z model)

Fig.4.4.1-8　Modelling of geometry for analysis of core expansion process during the reactor response phase in Monju

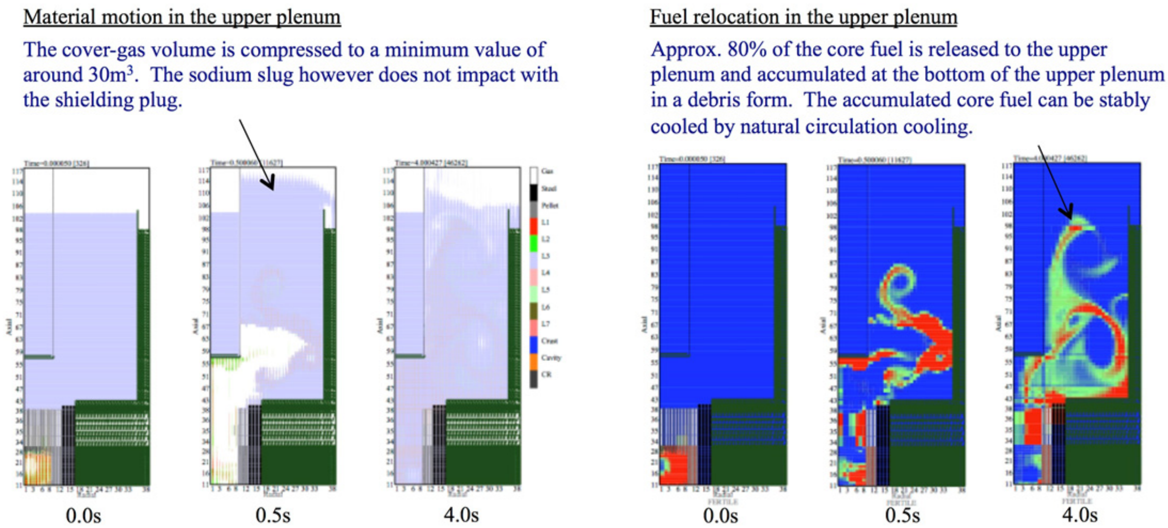


Fig.4.4.1-9 Material motion and relocation in the upper plenum during core expansion process in Monju

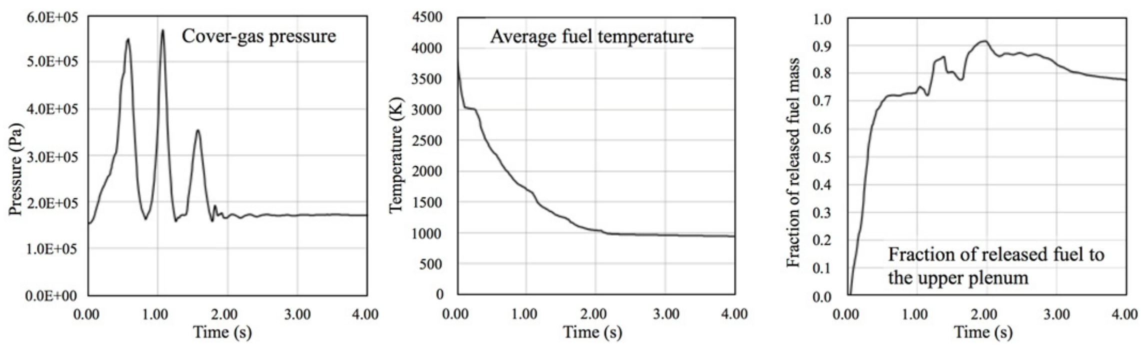


Fig.4.4.1-10 Analytical results of core expansion behaviors in Monju

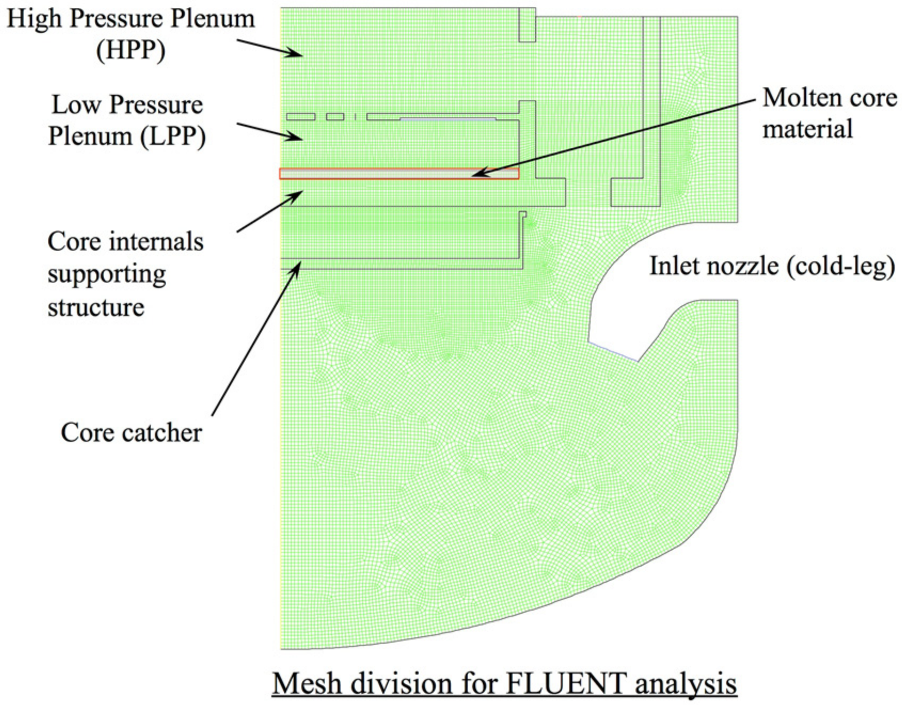


Fig.4.4.1-11 Modelling of geometry for hydraulic analysis of the lower plenum in Monju by the FLUENT code

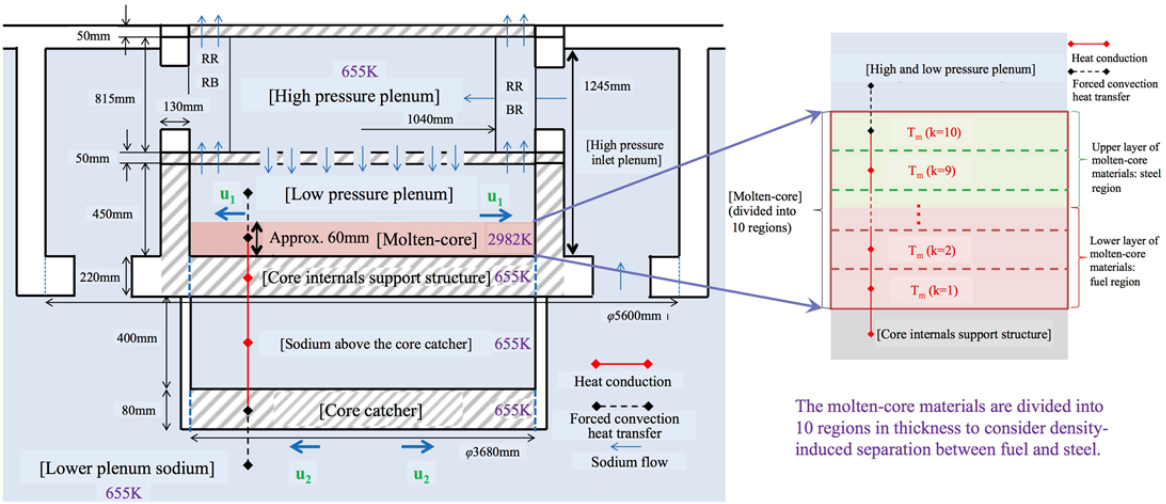


Fig.4.4.1-12 Modelling of geometry for analyzing cooling and retaining behavior of molten-core materials in the LPP

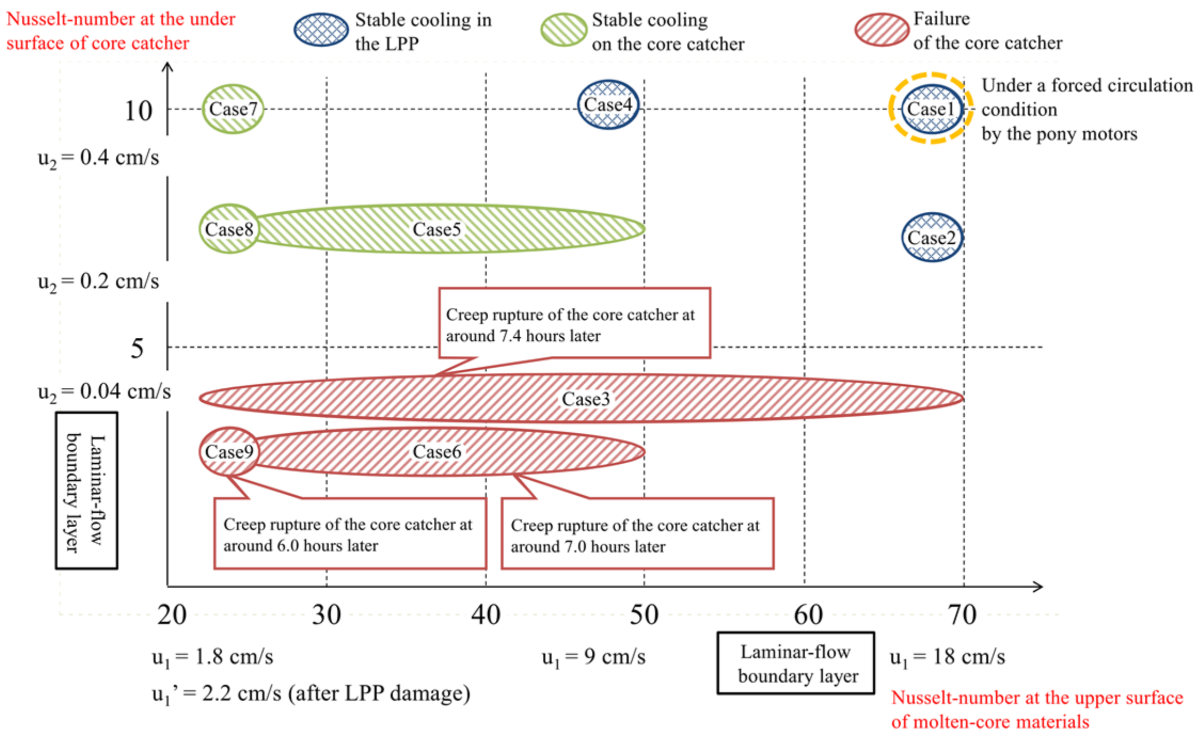


Fig.4.4.1-13 Analytical results of cooling and retaining behavior of molten-core materials in the LPP

4.4.2 Loss of Heat Removal System (LOHRS)

(1) Basic Concept

The measures for ensuring containment function during loss of heat removal system (LOHRS) events are aiming at securing the containment function by preventing core damage and thus maintaining the integrity of the reactor vessel. This would be accomplished by removing the decay heat using substitutional equipment even assuming that a failure occurs at preventive measures against core damage. The plant condition assumed here corresponds to a state where the preventive measures against core damage fail, irrelevant to occurrence of core damage itself. The measures for ensuring containment function therefore shall be as independent and diverse as possible so that they would be functional even assuming failures of the preceding mitigation measures for DBAs and preventive measures against core damage.

Providing redundant measures against SAs is of importance to enhance safety, preventing massive radioactive material release due to core damage induced by LOHRS-type events. Thus avoiding RV failure to keep its integrity is adequate as a concept for securing safety in SFRs to ensure containment function.

(2) Accident Sequences

The accident sequences can be represented by the following events regarding measures for ensuring containment function during a LOHRS-type event.

- Loss of reactor level (LORL) + Failure of preventive measures against core damage (Failure of measures for maintaining reactor level)
- LOHRS + Failure of preventive measures against core damage (Failure of heat removal by natural circulation, etc.)

Figure 4.4.2-1 depicts the event progression of LORL. The needed sodium level for core cooling is kept against reactor coolant leakage by designing the piping location at a higher position and the guard vessel with a limited volume, as mitigation measures against DBAs. In case of an additional reactor coolant leakage at a loop other than the loop with the original sodium leakage, siphon breaking operation will be performed at an elevated piping position by operators in the central control room. This event is beyond the conventional assumption and conducted as a preventive measure against core damage.

The RV is isolated from the damaged loop by this operation and thus the sodium level in the RV can be maintained above that needed for core cooling (EsL) and decay heat removal is enabled using remaining intact loops.

A possibility however still remains to result in the siphon breaks in all the loops by assuming the failure of preventive measures against core damage. This depends on the combination of locations to be damaged, resulting in the sodium level below the EsL. The containment function can be ensured even under such conditions by urgently activating the maintenance cooling system (MCS) by operators in the central control room. The flow path of the MCS can be maintained by design considering such conditions and the integrity of the RV can be kept by preventing core damage owing to the above-mentioned emergency operation by operators.

Figure 4.4.2-2 shows the event progression of LOHRS. Heat removal by forced circulation using the ACS is the first-step mitigation measure against DBAs as for the decay heat removal function. Heat removal by natural circulation is also possible using the ACS as a preventive measure against core damage, even in case that the above-mentioned DBA measure fails, by keeping the needed flow path. The containment function can be ensured by urgently activating the MCS implemented by operators in the central control room, even under a condition that the preventive measure fails at all the 3 loops of the ACSs. The integrity of the RV can be kept by preventing core damage owing to the above-mentioned emergency operation by operators.

Core damage and RV failure can be avoided by providing multiple measures against SAs as mentioned above.

(3) Overview of Concept for Ensuring Containment Function

The concept for ensuring containment function in SFRs is to be overviewed as follows based on the above-mentioned investigations, in contrast with the articles of the new NRA's regulatory requirements.

- Grace period for core damage has large safety margins in SFRs even under a failure of decay heat removal. This is owing to the wide temperature range of sodium being available in a liquid state, and the large heat capacity of sodium in the system and structural materials in the plant. Thus the available multiple preventive measures against core damage are enough so that core damage can be prevented with high reliability.

  Measures for ensuring containment function in SFRs therefore does not neccessarily require the assumption of RV failure by molten fuel, but the failures occur at preceding preventive measures against core damage. Deploying measures, to retain and cool down the core without uncovery of sodium in the RV, is effective in SFRs for ensuring containment function. (Corresponding to an article in the new regulatory requirements: Article 49)

  The CV is positioned as an essential system for reactor cooling in LWRs as a pressurized system. It plays a role of a pressure barrier against an accident of reactor pressure coolant boundary failure, etc. in such a pressurized system. An open-type guard vessel, etc., on the contrary, is sufficient in SFRs to maintain the reactor coolant level needed for core cooling against coolant leakage owing to the system being not pressurized.

  That is to say, SFRs do not require the CV, in contrary to LWRs, as a system to terminate the events within the RV for measures to ensure containment function. CV pressure relief systems or recirculator units, as in LWRs, are not needed. (Corresponding to an article in the new regulatory requirements: Article 50)

- It is effective to implement measures to practically eliminate the falling down of molten core even if the core damage occurs. This can be accomplished by retaining and cooling down the core without uncovery of sodium in the RV, attaining an IVR. (Corresponding to an article in the new regulatory requirements: Article 51)
- Hydrogen generation is unavoidable in LWRs due to the reaction between water and zirconium, included in the fuel cladding, at an early stage of core damage where no RV damage by molten fuel occurs. No such massive hydrogen generation occurs in SFRs, on the contrary, due to lack of such a reaction mechanism in core if RV failure is avoided by attaining IVR.

  It is thus effective to implement measures to practically eliminate massive generation of hydrogen by retaining and cooling down the core without uncovery of sodium in the RV, attaining an IVR. (Corresponding to articles in the new regulatory requirements: Articles 52 and 53)

(4) Concept for Instrumentation and Monitoring during Severe Accidents

Measurement parameters for monitoring and judgment of event progression shall be identified and established by each accident sequence, when an accident occurs leading to a possible loss of containment function. The parameters shall be adequately selected depending on the event progressions of representative accident sequences and environmental conditions, to be able to monitor and judge the attainment of IVR, i.e. RV does not fail, by the measures against SAs. The sodium level meters in the guard vessel, for example, can be used for the judgment in a PLOHS event, since the sodium level decreases in the guard vessel when the RV failure occurs in this type of event.

Core damage can be judged by the delayed neutron detectors or radiation detectors located in the CV. Attainment of IVR after core damage can be estimated by referring to readings and trends of a wide spectrum of detectors such as the sodium thermometers in the RV, neutron detectors, sodium level meters in the guard vessel, etc.. Detectors for the judgment of RV failure are available, such as the sodium level meters in the guard vessel, level meters and thermometers in the RV, underfloor atmospheric thermometers in the CV, contact-type sodium leak detectors, sampling-type ones, etc.. The judgment will be made by effectively combining these available detectors depending on the accident situation.

(5) Issues to be solved

- The sufficiency of the basis of defense-in-depth is to be consolidated by quantitative investigations in PRAs or effectiveness assessments of measurements, regarding the concept to ensure safety by preventing core damage using multiple measures against SAs.
- The accident sequences, which needs multiple measures against SAs, are to be specified

based on PRAs, etc. Additional measures are to be investigated to ensure the prevention of core damage, if necessary. Influences on existing systems and comprehensive risks should be also considered in these investigations because the operations of systems beyond conventional supposition may be performed.

(6) Summary

It is effective to retain and cool down the core without uncovery of sodium in the RV against LOHRS events in SFRs. A sufficient grace period for core damage enables this as a measure for ensuring containment function, effectively utilizing safety features of SFRs.

The enhancement of safety by providing redundant measures against SAs is of importance to prevent significant radioactive material release due to core damage induced by LOHRS-type events. Thus ensuring containment function by avoiding RV failure and keeping its integrity by this enhancement of safety, is adequate as a concept for ensuring safety in SFRs.

Moreover, measures shall be provided for monitoring. Procedures shall be prepared, system preparedness shall be established, and training shall be conducted. These are required in order to adequately and flexibly respond to an emergency concerning these measures against SAs.

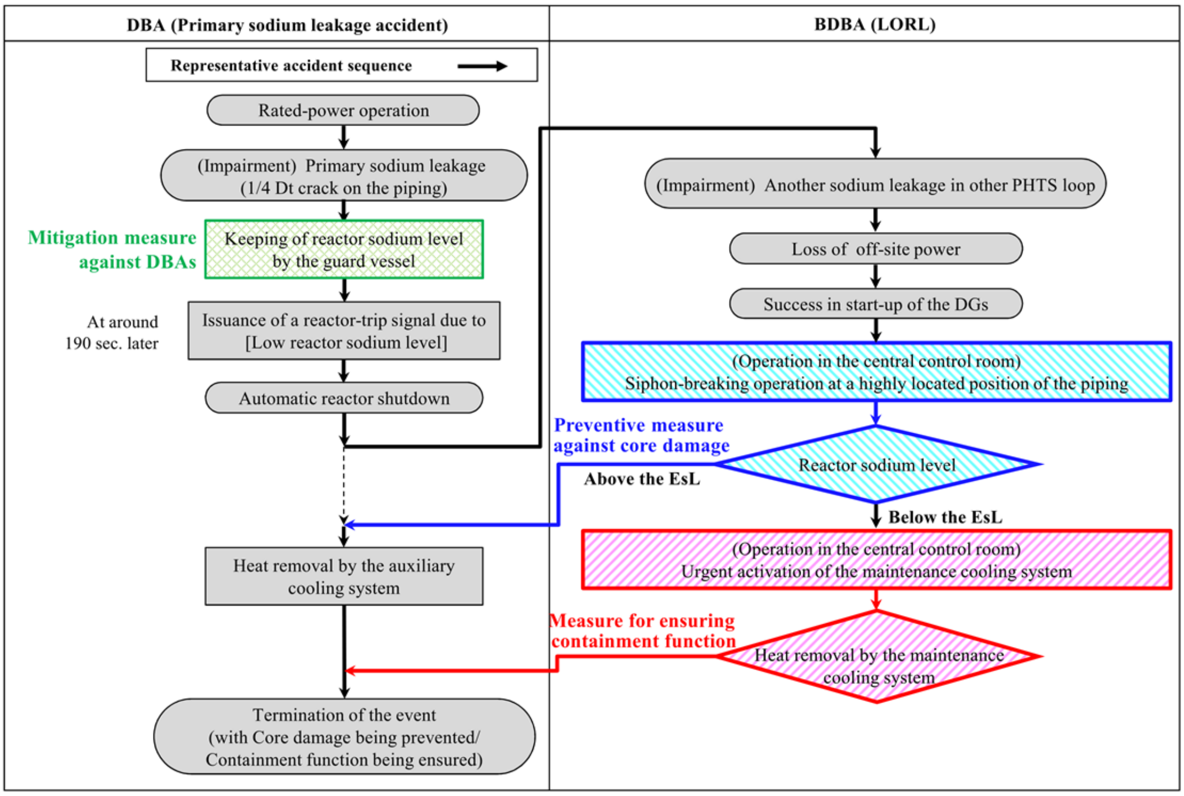


Fig.4.4.2-1 Event progression of LORL

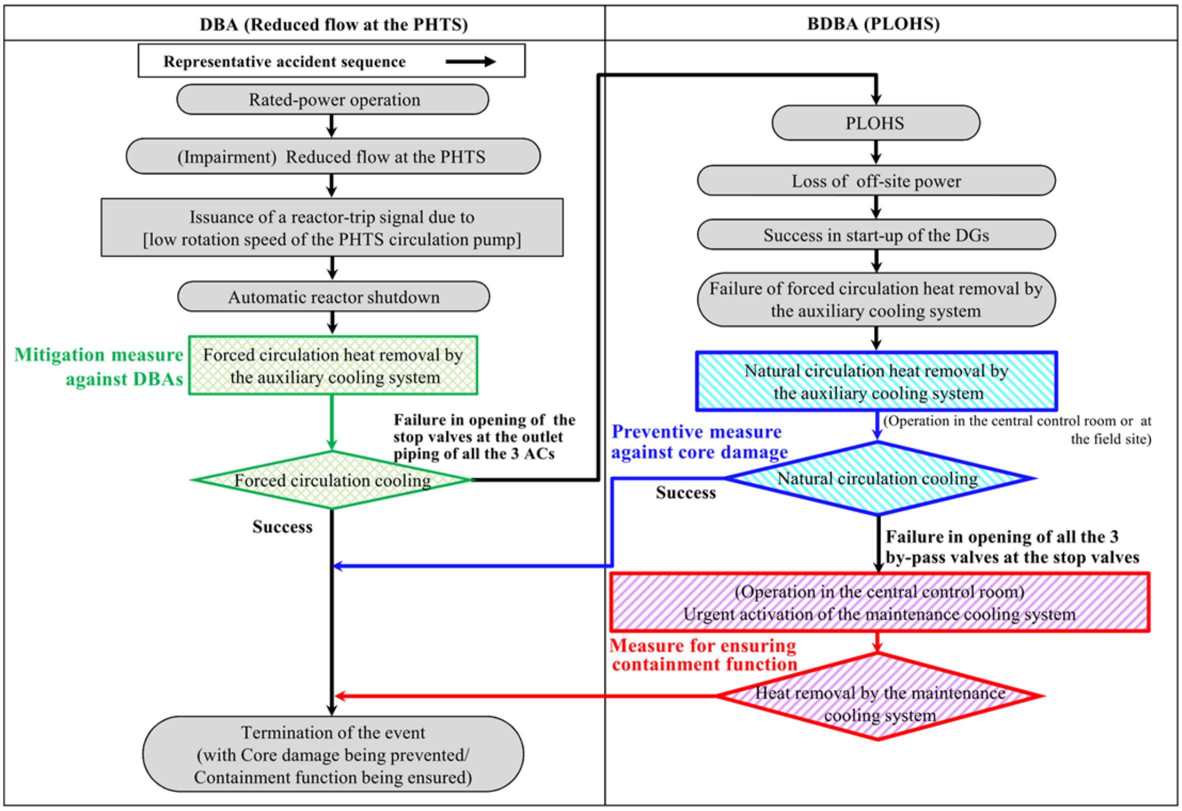


Fig.4.4.2-2 The event progression of PLOHS

7. Concept for Securing Safety of SFR

One of the safety features of Sodium-cooled Fast Reactors (SFRs) results from the characteristics of coolant sodium. The primary cooling system does not require pressurization for power generation owing to the high heat conductivity and boiling point of coolant sodium (883°C at atmospheric pressure). Reactor coolant level is able to be maintained above the reactor core by passive equipment such as guard vessels (GVs) without losing coolant from the reactor core, even if a leakage occurred from the primary heat transport system (PHTS) piping, unlike Light Water Reactors (LWRs). Therefore, depressurization of primary cooling system and operation of Emergency Core Cooling Systems (ECCSs) to cool down the core in case of accident which are necessary in LWRs are not required in SFRs. Moreover, cooling by natural circulation is possible in SFRs, which does not require the pumping, owing to the liquid sodium property in the wide temperature range. The heat capacity of coolant sodium and structures of the plant are relatively larger than the amount of decay heat. A sufficient grace period can be thus provided to significant core damage, even though a loss of heat removal occurred. The reliable measures will be implemented against design-basis accidents (DBAs). Moreover, decay heat can be continuously removed by manual operation either in the central control room or locally, even if the automatic operation leading to decay heat removal by natural circulation fails. Such diverse and robust accident management (AM) measures enable heat removal to be continued in such a case.

In SFR, sodium void reactivity might be positive at the central region of the core, therefore sodium boiling and cover gas entrainment into the primary cooling system need to be considered in the SFR safety design. Re-criticality induced by fuel compaction and consequent energy release are also to be considered when the core fuel is damaged and relocated, because the fuel inventory in the core significantly exceeds its minimum critical mass in SFRs. SFRs have different safety features from those of LWRs as mentioned above, and this needs to be sufficiently taken into consideration when deploying safety measures. Measures shall be taken by referring to the information obtained from PRAs and others in order to practically eliminate such accident sequences that may lead to the large-scale loss of containment function. And risks shall be reduced for the significant radioactive release.

In this report, the safety design of reactor shutdown systems has been examined for DBAs in Chapter 3, as well as for internal fire, internal flooding, water leak from the SGs, external hazards, and so forth. Possible accidents beyond design-basis (BDBAs) have been examined in Chapter 4 and the following chapters. As a result, sixteen important requirements are proposed in this chapter. These will clarify the essential concepts with respect to securing safety in Monju, considering lessons learned from the accident at the TEPCO Fukushima Dai-ichi nuclear power plant (1F accident).

(i) Basic Concept,

(ii) Concept for Measures against Design Basis Accidents (levels 1 to 3 of the defense-in-depth concept), and

(iii) Concept for Prevention of Severe Accidents (SAs) and Mitigation of Consequences (level 4 of the defense-in-depth concept).

It is especially emphasized that the concept for securing safety is examined from the viewpoint of what we learned from the 1F accident and what concepts are newly added.

Essential lessons learned from the 1F accident are as follows from the technical point of view, as shown in Chapter 2:

(1) Prevention of long-term loss of off-site power due to earthquakes,

(2) Prevention of loss of on-site power due to common cause failure, and reinforcement of emergency power supply,

(3) Enhancement of AM measures such as depressurization and water injection to cool down the core during a station blackout (SBO), and ensuring reliability of heat removal function at spent fuel pools,

(4) Prevention of failure of containment vessels (CVs) resulting in early or uncontrolled radioactive release, and

(5) Thorough strengthening of monitoring the plant statuses and management functions of the plant.

The 16 requirements are stated reflecting these lessons in the following viewpoints.

I. Basic Concept

Basic safety principles and concept to utilize the risk information (Requirement 1).

II. Concept for Measures against DBAs (levels 1 to 3 of the defense-in-depth concept)

Requirements are summarized to enhance measures against DBAs:

- Basic functions of reactor shutdown and decay heat removal at levels 2 and 3 (Requirements 2 and 3),
- Measures against internal fires and water flooding which may lead to common cause failures (Requirement 4), and
- Confirmation and enhancement of provisions against distinctive events which may occur with chemically active sodium coolant (Requirement 5).

These are basically not different from the existing regulatory requirements. Important points are stated in this report especially in relation to safety measures at level 4 of the defense-in-depth concept.

III. Concept for Prevention of SAs and Mitigation of Consequences (level 4 of the defense-in-depth concept)

Requirements on measures against SAs are summarized in the following items.

- Requirements on AM measures from the viewpoint of preventing events leading to the significant core damage, even if the events will be evolved to the state beyond the scope of DBAs (level 4 of the defense-in-depth concept) (Requirements 6 to 8),

- Requirements on further AM measures in case that Requirements 6 to 8 fail, for ensuring the containment function to avoid significant radioactive release and to mitigate consequences (Requirement 9),
- Requirements on diverse AM measures against external hazards such as natural phenomena based on the lessons from the 1F accident and intentional man- made events (Requirements 10 and 11),
- Requirements on measures for possible hydrogen explosion (Requirement 12),
- Requirements on strengthening AM measures for the events at sodium-cooled ex-vessel spent fuel storage tanks (EVSTs) and water-cooled spent fuel storage pools (Requirement 13),
- Requirements on ensuring safety operations in the central control rooms and emergency operation centers to support emergency responses after occurrence of an SA, as a part of diverse AM measures (Requirement 14),
- Requirements on monitoring of plant statuses when an SA occurred (Requirement 15), and
- Requirements on flexibility of AM measures in operation, management and system preparedness (software) to enhance its reliability (Requirement 16).

Important issues are also to be stated in relation to events within DBAs in these requirements.

In the past safety reviews of Monju licensing by regulator, the so-called Paragraph (5) events have been clarified of their positioning as stated in Chapter 2. As it is discussed in Chapter 4.7, the unprotected loss of flow (ULOF) and the unprotected transient overpower (UTOP) events shall be identified as representative “accident sequences” among the Paragraph (5) events from the viewpoint of their occurrence frequencies and consequences. Their occurrence frequencies can be estimated and the consequences lead to core damage if once they occur. In both events, measures to prevent the core damage and those to ensure containment function shall be deployed at level 4 of the defense-in-depth concept. And these measures shall be assessed their effectiveness and validity in the safety reviews. Therefore, these events shall be included in the events subject to the safety reviews.

Local fault (LF) and loss of piping integrity (LOPI) of the primary cooling system events[a)] are considered to be extremely low occurrence when considering their assumptions of initiating events, and these accident sequences will not lead to significant core damage even if they once occurred. Both events are essentially different from those identified based on the results of probabilistic risk assessments (PRAs) and shall be clearly distinguished in the safety reviews for SAs.

a) Postulated conditions for LFs and LOPI events were as follows in the past safety licensing of Monju:

- Hypothetical local over-power: Ten fuel pellets, of which the relative linear heating rates (LHRs) were twice of that in normal ones, were assumed to be inadvertently loaded into the core mid-plane position in a fuel pin.
- Hypothetical local flow blockage: A planar and impermeable blockage of 66% of the total flow area was assumed in a fuel subassembly.
- Large-scale pipe break at the primary heat transport system (PHTS): A double ended guillotine pipe break at the reactor vessel inlet was assumed.

As a reference, comments from an overseas expert are listed in Appendix 2 concerning the new regulatory requirements applied for Monju. Major comments are shown below:

(1) Current requirements includes ones that should not be applied to SFRs,

(2) Although measures against melt-through of reactor vessels (RVs) and guard vessels (GVs) are required in the current requirements, it is not likely to put SFRs in such a condition because a multiple failure scenario with much smaller likelihood has to be assumed, and

(3) In general, only major source of hydrogen generation in an SFR is byproduct of sodium-concrete chemical interactions, and its occurrence is preventable.

These were referred to when summarizing the concepts to deploy safety measures.

Sixteen requirements are listed below on the appropriate measures to be taken as "the concept for securing safety in Monju including SAs" based on the premises mentioned above.

I. Basic Concept

1. In order to secure the safety of people and to protect environment from accidents in nuclear facility, any risk shall be limited below a socially acceptable level. This shall be accomplished by preventing occurrence of accidents and mitigating their consequences based on the defense-in depth concept.

< Detailed descriptions >

In general, it is appropriate to secure safety of nuclear facilities which can be achieved by deploying measures at each level from 1 to 5, based on the defense-in-depth concept defined by IAEA. Each level in this defense-in-depth concept is defined as follows:

Level 1: "Prevention of abnormal operation and failures",

Level 2: "Control of abnormal operation and detection of failures",

Level 3: "Control of accidents within the design basis",

Level 4: "Control of severe plant conditions including prevention of accident progression and mitigation of severe accident consequences", and

Level 5: "Mitigation of radiological consequences of significant off-site releases of radioactive materials".

Safety of nuclear power plants is eventually secured by implementing safety measures at each level.

Prevention of malfunctioning is important by performing inspections and maintenance for enhancing measures against DBAs at levels 1 to 3. Confirmations shall be also performed of the conventional environmental conditions (especially natural phenomena including earthquake motions) set as a design standard. Durability against them shall be strengthened and preventive measures against common-cause failures shall be taken. The defense-in-depth is a concept for providing measures against uncertainties. Effective measures shall be adequately taken by recognizing uncertainties in representative events among SAs at level 4. These representative events shall be identified by the comprehensive PRAs or alternative analyses. The term of "measures" here means the entire AM measures from the viewpoint of not only equipment and facilities (hardware) but also operation, management, and system preparedness (software) to be stated in Requirement 16.

Continuous efforts shall be made with consistency both at each level and as a whole of the defense-in-depth concept in order to reduce the risks for people and the environment below a socially acceptable level. Utilization of risk information obtained by PRAs will be important, while recognizing their uncertainties, in order to make such comprehensive efforts to be effective and efficient. For the utilization of risk information, safety at the present state is required to be assessed probabilistically, but merely conducting such an assessment is not enough. Reliability of results shall be continuously improved by properly reflecting the results of maintenance activities and training for implementing AM measures. Substantial enhancement of safety shall be pursued as well through identifying inspective items of importance to safety and exploring more effective AM measures. Furthermore, continuous efforts should be taken in order to pursue a highest safety as a reasonably achievable level considering the following items:

- Technologies to improve reliability of safety functions,
- Technologies to precisely assess the plant behaviors and physical phenomena occurred during accidents,
- Technologies to quantify the influence of hazards to the plant with a wide spectrum of natural phenomena,
- Safety researches on technologies to evaluate the durability of each equipment and facility, and
- Establishment of effective maintenance technologies.

II. Concept for Measures against DBAs (levels 1 to 3 of the defense-in-depth concept)

2. Reactor Shutdown Systems: Reactor shutdown systems shall be equipped according to the concept of redundancy, diversity and independence. SFRs shall have multiple and independent reactor shutdown systems using control rods. At least one of those systems shall be able to shut the reactor down to a sub-critical condition and maintain sub-criticality during DBAs.

< Detailed descriptions >

Reactor shutdown systems in LWR consist of inserting a control rod and injecting a boric acid solution to the core to have redundancy or diversity and independence. It is difficult in SFRs to deploy such a system as injecting boric acid solution. Therefore, multiple and independent reactor shutdown systems by means of control rods (CRs) are installed in SFRs. This SFR reactor shutdown system has redundancy or diversity and independence by differentiating them in terms of the acceleration methods, rod-insertion schemes, trip circuits, and triggering signals at a time of reactor scram.

Two reactor shutdown systems are then installed in Monju, more specifically the primary and backup ones, respectively. Either of these systems can shut down the reactor independently with sufficient safety margins. Further improvements in reliability and continuous enhancement of safety are required to be pursued, although the systems have already been highly reliably arranged. These shall be conducted by analyzing causes of ATWS events induced by failure of control rod insertion and others

< Comparison with LWRs >

LWRs have reactor shutdown functions by injecting boric acid solution so called “Standby Liquid Control System (SLCS)” in addition to ones using control rods. The SLCS can put the reactor into a sub-critical state in case that emergency reactor shutdown by control rods fails. On the other hand, both the primary and backup reactor shutdown systems in Monju employ a control rod insertion method, where the backup shutdown system in Monju corresponds to the SLCS in LWRs.

< Corresponding articles in the new regulatory requirements >

Articles 10, 12-13, 15, 14, 24-25, 37, and 43-44.

3. Decay Heat Removal Systems: Decay heat removal systems shall be equipped in order to transport the decay heat generated in the core to the ultimate heat sinks. The heat transport systems and ultimate heat sinks shall not lose their functions and integrity.

< Detailed descriptions >

ECCSs are provided in LWRs against events such as primary system piping rupture in order to remove decay heat by urgently injecting cooling water into the core. Reliability is improved by making this system to be redundant or diverse and independent. Primary systems are not pressurized in SFRs, on the contrary, and there exists no such a possibility to lose the coolant directly from the system at a rupture of the boundaries. Decay heat removal will be therefore possible by using residual intact systems other than the damaged one, so far as multiple cooling systems are installed.

Sodium loops shall be secured to be intact and the circulations of coolant sodium shall be maintained for these decay heat removal systems. Sodium flow paths shall be confirmed to be kept from the viewpoint of anti-freezing and seismic durability. Manual operation procedures shall be prepared for vanes and dampers of air coolers, training of operators shall be performed, and system preparedness shall be established to surely implement these. These are required to secure the flow paths of coolant sodium so that the decay heat removal is available under any conditions even at an event of station blackout (SBO).

< *Comparison with LWRs* >

Coolant sodium in the RV can be retained by the GV and so on, even if a sodium leakage is occurred in the primary cooling system of SFRs, because the primary cooling system is not pressurized, contrary to LWRs. Therefore, a system such as an ECCS as deployed in LWRs is not required in SFRs. Decay heat removal by means of natural circulation is expected in primary and secondary coolant sodium. This is different from LWRs, in which the decay heat can be removed only by means of pump-driven forced water circulation.

However, making up sodium from outside the plant is impossible at a leakage of coolant sodium in SFRs, as is the case in LWRs. Sufficiency of sodium inventory stored in the cooling system shall be confirmed so that the needed sodium level in the RV, and coolant flow rate and flow paths are secured enough at the design-basis coolant leakage events. It also needs to be confirmed that the facilities shall be equipped in order to make up the coolant from inside the plant without difficulty during BDBA events.

< *Corresponding articles in the new regulatory requirements* >

Articles 10, 12-13, 14, 17, 21-22, 24, 37, 43, and 47-48.

4. Avoid the Common Cause Failure: Needed preventive measures shall be deployed against occurrence and expansion of internal fires and flooding that may lead to common cause failure. Statuses of these shall be reconfirmed. Additional measures shall be implemented if the existing ones are insufficient.

< *Detailed descriptions* >

Safety measures shall be enhanced against fires and flooding occurred inside the plant from the viewpoint of preventing the loss of safety functions caused by a common cause failure. Preventive measures against fires occurred inside the SFR plant are generally the same as those for LWRs. It should be noted that important safety functions in SFRs shall be assured with independent and redundant means by separating relevant systems. Measures shall be taken to prevent, detect and extinguish the fire, and mitigate its consequences. It is important to deploy appropriate facilities among fire-fighting installations using non aquerious materials such as inert gasses (carbon dioxide, nitrogen, etc.) applied in the area where sodium component is installed.

It is required to assess the consequences of internal flooding by identifying the facilities to be protected, considering the degree of importance on safety and also their safety functions specific to SFRs. The assessments shall be performed from the viewpoint of influences by immersion, water spray and steam in the same way as in LWRs. Especially, the consequences of flooding shall be confirmed not to affect the area, where sodium equipment specific to SFRs is installed, from the origin of internal flooding outside of the area.

< *Comparison with LWRs* >

Chemical properties of sodium shall be considered and designs on a basis of them shall be employed, when selecting fire-fighting installations against fires in SFRs. More specifically, appropriate equipments shall be deployed among fire-fighting installations using non aquerious inert gasses (carbon dioxide, nitrogen, etc.) for internal fires in SFRs where sodium component is installed. The influence of flooding shall be confirmed not to affect the area, where sodium component is installed, from the origin of internal flooding outside of the area.

< *Corresponding articles in the new regulatory requirements* >

Articles 8, 9, and 41.

5. Specific accidents concerning SFRs: Sufficient measures shall be taken against the sodium leakage and sodium-water reactions. Present facilities and measures which have already been implemented shall be reviewed to be valid for the sodium leakage in the secondary cooling system and the water leakage in SG, and these events shall be investigated whether or not to progress to SAs beyond the design-basis accidents. Additional measures shall be adequately taken, if necessary.

< *Detailed descriptions* >

Sodium used as coolant in SFR is chemically reactive, which induces the combustion under atmospheric condition and fierce reaction with water, therefore this chemical characteristics of sodium is required to be considered in the safety design. The following measures are therefore generally taken in SFRs as design-basis ones from their designing phases.

(1) Equipment containing liquid sodium with a cavity space is required that the cavity is filled with an inert-gas atmosphere so that the sodium does not directly contact with air.
(2) Systems and components important to safety to circulate sodium, are designed and operated to prevent the sodium from freezing, which loses their safety functions.
(3) Radioactive primary sodium is contained in a system configured to transfer the heat to the secondary system through the intermediate heat exchanger (IHX), so that the primary system is isolated.
(4) Sodium-water reactions may occur due to the water leakage from the heat transfer tubes of the steam generators (SGs). The system detects immediately the occurrence of a leakage, and is able to mitigate the consequences and terminate the reactions, when it once occurred. The secondary sodium system is isolated from the reactor core and other cooling loops via the IHXs. These configurations avoid any influences of sodium-water reaction on the other cooling loops. Systems secured to remove the decay heat are therefore separated.
(5) Design considerations are conducted for equipment important to safety not to lose the safety functions by consequences of chemical reactions of sodium, even if its leakage occurs.
(6) Equipments important to safety are separated in order to mitigate the consequences of sodium leakage.
(7) Steel linings and others are installed for the area where a sodium-containing equipment is installed, if necessary, to avoid direct contact of sodium with concrete.

Adequacy of these measures and possibilities of progression to SAs shall be reconfirmed in reference to the 1F accident, although measures have already been taken from the designing phase of the plant.

The room of primary cooling systems connected to the reactor vessel is filled in a nitrogen-gas in SFRs to prevent sodium combustion in case of sodium leakage. On the other hand, equipments of the secondary cooling systems such as sodium piping and so on are generally installed under air atmospheric conditions. Therefore it is required to confirm that appropriate measures shall be deployed against the sodium leakage events of secondary cooling systems. Leak rate and duration of sodium leakage are key issues to assess the consequences including sodium combustion. It is required to assume more severe conditions beyond DBAs bearing in mind a possibility to progress into SAs.

Leak rate and amount of water leakage from the breached tube are the dominant factors in safety assessments on sodium-water reactions in the SGs. It is required to investigate the possibility of the events beyond the scope of DBA assumptions. The progress of events is extremely rapid and AM measures might be unavailable in this type of events, although its frequency is estimated to be extremely low. Integrity of the boundary of the secondary sodium cooling system including that in the IHX, where the common boundaries of both the primary and the secondary sodium cooling systems are shared, is required to be examined concerning the possibility of progression into SAs.

Measures shall be further taken to secure the safety in the long term during a termination phase of the events until the recovery of the plant, after the events of sodium leakage and combustion, and sodium-water reactions occurred in the SGs.

< *Comparison with LWRs* >

Measures specific to SFRs are required which are different from LWRs. Sodium is chemically reactive, and heat and aerosols are produced by combustion when sodium is leaked from its boundary into the atmospheric environment at higher temperatures. Hydrogen can be generated by reactions of sodium with moisture released from concrete, under a condition that sodium contacts with concrete at a high temperature enough for concrete to release moisture. Measures are therefore required to quickly terminate sodium leakage and others for mitigating consequences of it. When a heat transfer tube of SG is failed, highly pressurized water and steam will be injected into the coolant sodium, and the temperature will increase rapidly by the heat of sodium-water reaction while generating hydrogen. The area of failed heat transfer tubes will be extended due to corrosive damage (wastage) caused by the gas jet impingement created by this sodium-water reaction, resulting in damaging the neighboring intact tubes. It is therefore required the early detection and early termination of the event. This event is not the case in LWRs which have the heat exchanger from water to water.

< *Corresponding articles in the new regulatory requirements* >

Articles 8, 12, 17, 21, 41 and 47.

IV. Concept to Prevent SAs and Mitigate the Consequences (level 4 of the defense-in-depth concept)

6. AM measures shall be adequately taken to be able to implement against anticipated transient without scram (ATWS) and loss of heat removal system (LOHRS) events which may lead to possible core damage. The measures shall be determined by considering both equipment and facilities (hardware), and operation, management, and system preparedness (software). Loss of safety functions and progress of events shall be adequately considered by referring to PRAs and others.

< *Detailed descriptions* >

Internal events leading to SAs are generally categorized into two types, ones caused by ATWS and others caused by LOHRS. Nuclear reactor facilities shall be equipped with needed measures to prevent the significant core damage when such events occurred leading to a possible SA. Moreover, needed measures shall be also implemented to prevent abnormal releases of radioactive materials to the environment when an SA occurred. The term of "needed measures" here denotes safety designs and AM measures including both the ones from the viewpoint of needed equipment and facilities (hardware), and operation, management, and system preparedness (software).

These events as specifically described in Chapter 4 are selected by referring to comprehensive PRAs assuming the loss of safety functions and the following accident scenario. It is especially required to deploy adequate measures against ATWS and LHORS events.

Some of ATWS events progress too rapid to take effective AM measures due to the core characteristics of SFRs. Adequate measures shall be established against these events to secure the safety in a long-term period, and the design margins need to be confirmed based on the updated knowledge. Grace periods for conducting the AM measures are relatively long in LOHRS events (several to several tens of hours), on the contrary, due to a large heat capacity of sodium in the cooling system and a large amount of structures in the plant, resulting in a gradual temperature increase of sodium. As the boiling point of sodium at atmospheric pressure is 883 °C, sodium exists in a single phase for a wide temperature range considered in SFR operation, and this physical property of sodium enables decay heat removal by natural convection due to the changes of sodium density depending on its temperature. Adequate measures shall be established against LOHRS events considering these features.

< Comparison with LWRs >

New regulatory requirements for LWRs address mandate accident-sequence groups to be assessed, whereas such groups are not specified for SFRs. Therefore, accident sequence groups for SFRs shall be identified by comprehensive PRAs or other alternative methods considering their occurrence frequencies and consequences. And the representative accident sequence shall be selected, as a result of combining loss of anticipated safety functions implemented against DBAs.

< Corresponding articles in the new regulatory requirements >

Articles 10, 12-13, 15, 17, 21-22, 24-25, 37, 43-44, and 47-48.

7. It shall be considered the ATWS events in SFRs proceed extremely fast due to its core characteristics, therefore the safety margins in design shall be carefully confirmed based on updated knowledge and experiences. Adequate measures shall be taken considering both aspects of needed equipment and facilities (hardware), and operation, management, and system preparedness (software).

< Detailed descriptions >

Backup reactor shutdown systems are installed in SFRs independent of primary ones as a first step at level 4 in the defense-in-depth concept to prevent the significant core damage. This backup systems are initiated when the reactor cannot be urgently shut down due to a failure in CR insertion of the primary reactor shutdown systems, which are the equipments for DBAs. It is also required to assume the failure of CR insertion using this backup systems, eventually all the reactor shutdown functions are lost in ATWS events. Measures to ensure the containment function shall be implemented, as a second step at level 4 in the defense-in-depth concept.

AM measures shall be taken against this type of events considering their progression speed being extremely fast. Considerations shall be also conducted on the fact that re-criticality and consequent mechanical energy release may occur due to positive sodium void reactivity at the central region of the core and the core being not in a maximum reactivity configuration.

ATWS-type events are represented by the following three types of events: “unprotected loss of flow (ULOF)”, “unprotected loss of heat sink (ULOHS)”, and “unprotected transient over power (UTOP)”. The reactor power will increase rapidly in cases of a ULOF- or ULOHS-type events due to positive reactivity insertion induced by sodium voiding. This is caused by heating the coolant sodium over its boiling point at the central part of the core subassemblies. This power increase may induce the melting of fuel pellet even in the subassemblies prior to sodium boiling. Molten fuel will be ejected into coolant flow paths due to the fuel cladding breach. The fuel-coolant interaction (FCI) will occur which induce an additional positive reactivity due to boiling the coolant sodium. This may lead to a power excursion resulting in possible significant core damage. On the other hand, such a power excursion will less happen in a UTOP event. Molten fuel ejection into coolant flow paths via breached fuel cladding, sodium boiling due to FCI will also occur in this type of events, which are caused by a gradual increase in reactor power under a constant coolant flow condition. These phenomena occur in series from the subassemblies with higher power to those with lower power, which avoid the rapid and simultaneous insertion of positive sodium void reactivity. As a result, the power excursion in a UTOP event would not likely occur.

Preventive measures are as follows:

- To reduce occurrence frequency,
- To deploy design measures to assure the integrity of coolant sodium boundary by accommodating the mechanical energy released as a consequent of the event, and
- To deploy sufficient AM measures to maintain the core cooling in a long-term and to retain the molten core so that it will not penetrate the RV.

These design considerations have already been confirmed to satisfy the safety criteria under a condition beyond the design basis events stated as the Paragraph (5) in Monju. The entire safety assessments on this issue shall be reconfirmed based on the updated knowledge and experiences.

< Comparison with LWRs >

LWRs deploy the systems that can put the reactor into a sub-critical condition in case of the failure to urgently shut down the reactor in order to prevent significant core damage. Shutdown operation of coolant recirculation pumps and injection of boric-acid solution into the reactor are applied for BWRs. Actuation of auxiliary feed-water pumps to suppress the reactor power and injection of boric-acid solution into the reactor are applied in PWRs.

< *Corresponding articles in the new regulatory requirements* >

Articles 10, 12-13, 15, 17, 21-22, 24-25, 37, 43-44, and 47-48.

8. AM measures against LOHRS events are extremely important considering the safety features of Monju, therefore the adequate measures shall be established from the viewpoints of equipment and facilities (hardware), and operation, management, and system preparedness (software).

< *Detailed descriptions* >

Protected loss of heat sink (PLOHS) and loss of reactor level (LORL) are categorized as LOHRS events. AM measures against these events shall be deployed to prevent core damage considering the safety features of SFRs.

Sodium exists in a single phase until the high temperature (boiling point: 883 °C at atmospheric pressure) as is previously mentioned. The safety of SFRs can be , therefore, achieved basically to secure the coolant level in the RV, flow paths and circulation of coolant sodium so as not to expose the core out of sodium in the RV, which enables heat removal by circulating the coolant sodium. Sufficient grace periods (several to several tens of hours) will be granted to bring the SFR core to a significant damage in this type of events, because temperature increase is gradual in general due to a large heat capacity of coolant sodium and a large amount of structures in the plant. Multiple AM measures will be available for securing the level and circulation of coolant sodium in RV by effectively using this grace period, even if the transition to decay heat removal by natural circulation fails. It is important to prevent the core damage by means of these measures. Hydrogen is generated if the molten core materials contact with the structure materials made of concrete. Safety measures against LOHRS-type events are therefore extremely important without RV melt-through and high reliability is required for these AM measures. The AM measures for these events are reasonably achievable in Monju due to the above-mentioned long grace period and efforts to enhance the safety should be continuously pursued. Especially, AM measures shall be improved on the following operational point of view in case of LORL-type events:

- Siphon breaking operation to terminate a sodium leakage from the piping,
- Redundant measures considering recovery phase, and
- Temperature control by heating the coolant piping to secure sodium liquidity, and so forth.

< *Comparison with LWRs* >

LWRs are equipped with the facilities to cool down the reactor core under a pressurized condition, even if the cooling function of equipement for DBA fails. These equipements against DBAs correspond to the Reactor Core Isolation Cooling (RCIC) system or the emergency condenser in BWRs, and auxiliary feed-water turbine driven pump in PWRs. Other AM measures are also available against failure of depressurization functions at equipements against DBAs by diversifying the driving mechanism of depressurization valves at the reactor pressure coolant boundaries. These are equipped to prevent significant core damage and failure of CVs.

When sodium leakage occurred from the sodium boundaries of primary cooling system, coolant boiling induced under depressurized condition which may expose the core out of coolant sodium will never occur in SFRs. Therefore, there is no need to implement an emergency coolant injection systems in SFRs. Sufficient grace periods will be granted to bring the SFR core to a significant damage in this type of events, because temperature increase is gradual in general due to a large heat capacity of coolant sodium and a large amount of structures in the plant as is previously mentioned. AM measures considering the safety features of SFR is very important and shall be implemented to maintain core cooling so that consequences of accidents will not influence outside the RV.

Sodium exists in a single phase until the high temperature, it is important to enhance diversity in safety measures by securing the core cooling using the natural circulation due to the change of sodium density depending on its temperature. Decay heat removal by natural circulation can be achieved without any electric power nor equipment such as component cooling systems. Even if the initiation operation for the decay heat removal by natural circulation fails, backup measures can be prepared by manual operation either at central control room or locally. Therefore, AM measures shall be pursued in terms of diversity, remote and local operation. These AM measures shall be continually improved to enhance its reliability by training the AM procedure, reflecting feedbacks obtained from the training results to update operation procedure and so on. Specific measures shall be further implemented to update the maintenance plan by introducing the PRA methods in order to enhance the reliability of equipment.

< *Corresponding articles in the new regulatory requirements* >

Articles 10, 12-13, 15, 17, 21-22, 24-25, 37, 43, and 47-48.

9. Appropriate AM measures shall be implemented against ATWS and LOHRS events so as to practically eliminate the possibility of RV failure induced by the molten fuel. These events resulting in the loss of containment function shall be precisely investigated.

< *Detailed descriptions* >

In the second step at level 4 of the defense in depth concept, it is assumed that all the measures to shutdown the core fail in case of the ATWS events. In this case, CV may fail due to the increase of pressure inside the CV owing to combustion of sodium which was ejected on the operating floor with air atmosphere in the CV by the mechanical energy generated in the core. Therefore, consequences of this events shall be thoroughly investigated.

Sufficient grace periods (several to several tens of hours) will be granted to bring the SFR core to a significant damage and the decay heat removal by natural circulation will be available in the LOHRS events, therefore multiple AM measures shall be implemented so that the possibility of RV failure by molten fuel is practically eliminated. This enables to prevent sodium-concrete reactions and molten-core-concrete interactions in the RV cavity caused by penetration of molten fuel to the RV cavity through the bottom of the RV.

AM measures shall be implemented to prevent a significant core damage and to consider the accident sequence groups which fails to prevent it. The accident-sequence groups to be assessed are the followings:

(1) Anticipated transient without scram (ATWS),

(2) Loss of reactor level (LORL), and

(3) Protected loss of heat sink (PLOHS).

Accident sequences of SBO-type events occurred in the reactor are included in those of PLOHS events.

Diversity shall be thoroughly considered in preventive measures against core damage in the assessment. Such measures depend on each accident sequence in which failure is assumed.

A sodium ejection and combustion may happen due to the consequent mechanical energy release induced by the prompt criticality during ATWS events, and these events shall be evaluated accurately. In-vessel retention (IVR) of the damaged core shall be assessed for ATWS events. Containment functions shall be evaluated by clarifying the load to the RV. The IVR shall be evaluated in accordance with the following principles, as mentioned in Sec. 4.4.

(a) Uncertainties in the phenomena shall be adequately considered based on realistic assumption.

(b) Uncertain phenomena occurring in a realistic event progression shall be evaluated by employing conservative approaches or models and others considering their sensitivities of the analytical models.

The followings are classified as LOHRS with the three PHTS loops unavailable. One is a failure in maintaing sodium level during LORL events, and the other is a failure in the decay heat removal by natural circulation using the auxiliary reactor cooling system (ACS) in three loops during PLOHS events. It is important to secure the decay heat removal in this type of events using a cooling measure independent of the PHTS so as not to uncover the core with the coolant sodium. Effectiveness of measures for ensuring containment function shall be assessed by investigating into the consequent event progression based on the heat removal capability of this cooling measure.

< *Comparison with LWRs* >

CV spray systems, air purification systems, and combustible gases control systems are equipped in LWRs against loss of containment function events. On the other hand, such phenomena to significantly raise temperatures and pressures inside the CV do not occur during any DBA event in SFRs, therefore CV spray systems shall not be applied to SFRs (stated in Article 49 in the new regulatory requirements: the same shall apply hereafter.).

Sufficient grace periods will be granted to bring the SFR core to a significant damage during LOHRS events due to a large heat capacity of coolant sodium and a large amount of structures in the plant as is previously mentioned. Therefore, it is effective to deploy diverse AM measures for retaining and cooling the core without uncovery of the coolant sodium in order to ensure containment function during this type of events in SFRs (Article 49).

The CV is positioned as an essential facility for reactor cooling in LWRs being highly pressurized and it works as the pressure barrier against an accident of reactor pressure coolant boundary rupture, etc. On the other hand, the SFRs are not pressurized and an open-type guard vessel is sufficient to maintain the reactor coolant level necessary for core cooling in case of coolant sodium leakage. This feature implies that SFRs have such a capability to terminate an event within the RV by means of the containment function against the coolant sodium leakage, without either CV pressure relief systems or coolant water recirculation units which are deployed in LWRs (Article 50), as mentioned in Sec. 4.4.

Retaining and cooling the core is attainable without unconvery of the coolant in the RV, leading to a success in IVR, even though the core is damaged, as also mentioned in Sec. 4.4. This can be sufficiently accomplished by strengthening the measures against this type of events, and their viability shall be assessed (Article 51).

Hydrogen generation from a reaction between coolant water and zirconium of cladding material is inevitable in LWRs, even at an early stage of core damage prior to the RV failure occurred by molten fuel. On the other hand, such a massive hydrogen will not be generated in the SFR core, therefore IVR enables to avoid RV failure, again as mentioned in Sec. 4.4. It is thus moreover effective to implement measures for retaining and cooling the core without uncovery of the coolant in the RV, that is attaining an IVR. These result in practically eliminating massive hydrogen generation as well, and again the viability of these shall be assessed (Articles 52 and 53).

< *Corresponding articles in the new regulatory requirements* >

Articles 10, 12, 14, 17, 21-22, 32-33, 37, 43, 47-53, 56, and 58.

10. Measures shall be deployed by design in nuclear facilities with sufficient safety margins against natural phenomena, such as earthquakes, tsunamis, and other possible natural phenomena, by assessing risks using the PRA method and so on. Adequate AM measures shall be established, considering features of SFRs, against natural phenomena beyond the design-basis scale, by recognizing their consequences and durability of facilities.

< *Detailed descriptions* >

Lessons learned from the 1F accident point out that common cause failures shall be taken into account induced by fierce natural phenomena. The term of “other possible natural phenomena” denotes flooding, strong winds (typhoons), tornados, freezing, heavy rain falls, heavy snow falls, lightning strikes, landslides, consequences of volcanoes, biological influences, forest fires, and others based on the natural environments of the plant site. Measures shall be implemented against these phenomena on a design-basis scale with sufficient safety margins by assessing risks using the PRA method and so forth. Investigations shall be also carried out into a possibility of natural phenomena beyond the assumed scale, when looking back the cause of 1F accident being occurrence of a huge tsunami beyond the conventional predictions. Weak points (cliff edges) shall be perceived, and features of SFRs shall be considered as well. Continuous efforts are required to pursue multiple AM measures by using facilities with high durability, and to continuously improve the facilities as well. Diversity of measures needs to be considered for the common-cause failures induced by fierce natural phenomena. Update knowledges and new findings shall be periodically reviewed to include the plant modification and revision of operational procedures, and efforts shall be made to deploy further AM measures and to improve reliability of facilities. These efforts will be pursued to reasonably attain the highest level of SFR safety.

< *Comparison with LWRs* >

The same natural phenomena as those considered in LWRs are also assumed for SFRs. Features specific to SFRs, however, shall be considered in identifying important safety-grade facilities and deploying AM measures against loss of them. These facilities shall be identified through investigations into effects of their failures on maintainability of safety functions.

< *Corresponding articles in the new regulatory requirements* >

Articles 3-6, 12, 14, 33, 38-40, 43, and 58.

11. Measures shall be taken against intentional large-aircraft crashes and other terrorism to prevent occurrence of SAs and to mitigate their consequences. Considerations shall be simultaneously given to features of facilities, such as arrangements or layouts of natural circulation cooling loops and the ACSs, and effectiveness of AM measures as well.

< *Detailed descriptions* >

The measures shall be implemented to prevent abnormal releases of radioactive materials to the environment in case that a core damage occurs, including its precedent phase, caused by terrorism such as intentional aircraft crashes. Basic concept for this measure is to terminate the accident by retaining and cooling the fuel materials within the RV.

It is required to investigate whether or not the safety functions are lost for the core cooling and containment and to implement the proper measures against the aircraft crash. Viability of AM measures for core cooling shall be examined in terms of separation on the geometrical layout of cooling flow paths considering the actual plant configuration. Assessments shall be moreover performed on consequences of failure in securing functions of reactor shutdown and cooling, even under a CV failure condition. Realistic assessments shall be conducted in examining the measures, considering the site-dependent conditions. AM measures shall be moreover established using transportable devices.

< *Comparison with LWRs* >

Measures against intentional large-aircraft crashes and other terrorism in LWRs are providing a specific SA management building separated from the reactor building with a needed separation distance (more than 100 meters, for example), in order to avoid simultaneous destruction of these, or containing the reactor in a monolithic building. Three loops of ACSs are dispersedly installed in Monju for decay heat removal even by natural circulation, the maintenance cooling system by forced circulation is also equipped. It is required to assess the effectiveness of these cooling systems considering that an ultimate heat sinks in Monju are air coolers (ACs).

< *Corresponding articles in the new regulatory requirements* >

Articles 7, 12, 18, 21-22, 42-43, and 47-48.

12. Appropriate AM measures shall be implemented against hydrogen explosion, such as measurement of hydrogen concentration, hydrogen discharge from the CV, controlled small-scale hydrogen combustion, and so forth. Existing facilities at present shall be fully utilized from the viewpoint of preparing reasonable preventive measures against BDBAs.

< *Detailed descriptions* >

There exists no concern in SFRs for massive hydrogen generation in the RV caused by chemical reactions between coolant water and zirconium used as the fuel cladding material, and the radiolytic decomposition of water, as is the case in LWRs.

Measures shall be implemented against sodium-concrete and sodium-water reactions in SFRs, which are major source of hydrogen-generation. Therefore, hydrogen generation shall be surely prevented in the PHTS as a basic concept.

In Monju, leaked sodium from the primary cooling system is retained in the GV, and a direct contact of sodium with concrete is prevented by steel linings which cover the floors, side walls and ceilings in the PHTS rooms. Combustion of sodium is also prevented by filling the PHTS rooms with an inert gas. Floors in the secondary heat transport system (SHTS) rooms are also covered with steel linings to prevent a direct contact of sodium with concrete. The leaked sodium from the SHTS will be immediately drained to the combustion suppression tanks located in the secondary dump tank rooms through floor draining pipings. It may be concerned the combustion of sodium due to air atmosphere in the SHTS rooms, however integrity of the floor linings is secured by detection of sodium leakage and draining operation of sodium in the SHTS in the early stage of events.

Hydrogen generated with sodium water reaction is discharged to the storage tank of sodium-water-reaction-products when a leakage occurs at a heat transfer tube of the SG. Trnsferred hydrogen is then combusted and released to the environment through the piping above the storage tank.

Preventive measures shall be reconfirmed to have sufficient safety margins to prevent hydrogen generation as the enhancement of safety measures following 1F accident. It is also required to investigate a discharge of generated hydrogen and suppression of hydrogen combustion by reducing the oxygen content in the SHTS rooms, assuming more hydrogen generation beyond the design basis. In order to enhance these effects, the methods shall be investigated to detect hydrogen generation by means of monitoring hydrogen content, etc. and AM measures shall be reinforced concerning discharging operation procedure at the possible locations of hydrogen generation.

< Comparison with LWRs >

It is important to deploy appropriate preventive measures against hydrogen generation caused by sodium-concrete and sodium-water reactions as the safety feature of SFRs using sodium as a coolant.

< Corresponding articles in the new regulatory requirements >

Articles 23, 32, 52-53, and 59.

13. Adequate AM measures shall be taken to prevent a fuel failure stored in EVST and spent fuel pools considering the features of facilities.

< Detailed descriptions >

Spent fuels of SFRs are discharged from the reactor and then stored at EVST filled with sodium until the decay heat is sufficiently decreased. They are subsequently transferred and stored at spent fuel pools filled with water after a certain decrease of the decay heat. Therefore, EVST and spent fuel pools are required to have the following capabilities.

(1) Any possibility to reach criticality by spent fuels shall be excluded.

(2) It is required to prevent melting of spent fuels due to decay heat. Systems shall be equipped to transport the decay heat to the ultimate heat sinks, and their purification systems shall be installed as well.

(3) Appropriate shielding capabilities shall be provided for radiation released from spent fuels.

(4) Liquid radioactive materials shall not be spilt out or leaked from spent fuel storage tanks. It should be surely detected in case such a leakage occurs.

(5) It is required to prevent freezing the liquid used of decay heat removal systems.

Concerning the above mentioned, measures shall be implemented for EVST against the increase of sodium temperature due to loss of cooling function, and against the decrease of sodium level in the RV due to sodium leakage. Diverse measures are available for the increase of sodium temperature in EVST using transportable devices because the temperature increase is moderate under a loss of cooling function, in the former case. It is also important to prevent the sodium freezing when implementing these measures. Measures to prevent overcooling and its monitoring, and to control the temperature shall be investigated. Adequate AM measures shall be implemented for the decrease of sodium level by utilizing the feature of equipments which have already been installed such as outer vessel as a measure against the sodium leakage, in the latter case.

The same requirements as for EVST are applied for the spent fuel storage pools, whereas the cooling material is water instead of sodium. More sufficient grace periods will be granted compared with those for the EVSTs, because the decay heat of spent fuels in the storage pool is further decreased. Considering this feature, the AM measures shall be implemented against loss of liquid level so as to prevent water drainage by siphon breaking operation, etc. and to feed water using transportable devices as well.

< Comparison with LWRs >

The EVST is a facility specific to SFRs. Measures shall be taken to monitor the off-normal and accident events occurred in EVST considering such features that the vessel of EVST is sealed and sodium is opaque.

The spent fuel storage pool of SFRs is similar to that of LWRs in terms of its specifications and layout. As the decay heat is sufficiently decreased which is a specific feature in SFRs, the AM measures for feeding water and so on will be effective considering these longer grace periods.

< Corresponding articles in the new regulatory requirements >

Articles 12, 16, 37, 43, 54, and 57.

14. Habitability shall be secured for central control rooms and emergency operation centers. Adequate radiation source terms shall be estimated considering features during accidents in SFRs. Facilities shall be designed considering shielding, ventilation, etc. from the viewpoint of limiting radiation exposure of operators and other needed personnel.

< *Detailed descriptions* >

It shall be secured to conduct emergency operations by plant operators staying at the central control room in the BDBA events. The adequate environment for the plant operators in the central control room would be assessed for the identified events in the past safety assessments. It is also appropriate to assume the most conservative release of radioactive materials by referring to so-called "hypothetical accident" for the site evaluations for nuclear installations in light of significance of the facility. It is required to design the functions of central control room and to evaluate the radiation dose of plant operators by examining the comprehensive accident sequences using PRAs. The environment in the central control room shall be maintained so as to perform the operation for ensuring containment function and to terminate the accident, which is necessary as a facility for the SA management.

Emergency responders and needed personnel shall be able to stay in the emergency operation center, even if radioactive materials are released at an abnormal level, in reference to 1F accident. Events shall be adequately assumed to assess the environment in this facility considering that progress of events in SFRs is different from those in LWRs. It is of importance to thoroughly investigate into purposes of installation and requirements on functions of the emergency operation centers in SFRs. Identification of accident sequences to be assessed is also important considering occurrence frequencies and consequences based on PRAs. The source terms for facility designs of such as shielding, ventilation, etc. shall be conservatively estimated against radiation exposure of needed personnel based on the analytical result on event progression of the identified accident sequences. Fission product (FP) elements such as cesium, tellurium, and others, and radioactive sodium and plutonium, specific to SFRs, will be released into the atmosphere, in addition to rare gases and iodine, assumed in the past safety assessments, when recalling the 1F accident. These elements shall be additionally considered to contribute to radiation exposure of needed personnel as a common issue to habitability assessments in the central control rooms.

< *Comparison with LWRs* >

Progresses of events in SFRs are different from those in LWRs in terms of processes from core damage to deterioration of containment function, behaviors of radioactive materials in the CV, relatively moderate event progressions depending on the events, and others.

Habitability in facilities for SA management shall be assessed by properly assuming events for design, depending on their purposes, based on PRAs. The purpose of central control rooms, for example, is providing a facility to ensure its habitability in case that measures for ensuring containment function are in success, leading to termination of the accident. And that of emergency operation centers is, on the contrary, offering a facility to secure its habitability even in failure of these. It is required to assume the release of nuclide specific to SFRs, such as radioactive sodium and plutonium, into the atmosphere as the source terms based on the event progression analyses considering the use of sodium as a coolant materials. The facilities used for SA management shall be designed to assure enough radiation shielding and ventilation to limit the radiation dose and to maintain the environment for the operators and personnel.

< *Corresponding articles in the new regulatory requirements* >

Articles 26, 29-30, 34, 60, and 62.

15. Measures shall be taken to be able to monitor the statuses of the reactor and plant needed during an SA. Parameters to be monitored shall be adequately identified in view of features of SFRs, progress of events such as possible representative accident sequences, and environmental conditions. Devices for measurement shall be classified depending on the significance of parameters to be monitored. And their aseismic durability shall be secured.

< *Detailed descriptions* >

Devices for measurement shall be installed for parameters to be monitored under conceivable conditions of use during SAs and others, such as temperatures, radiation doses, and other environmental conditions. Substitutional measures shall be provided assuming failures of these devices due to their impairments, such as measurement of alternative parameters effective for estimating the parameters in question, transportable devices for measurement, and so forth.

Measuring devices shall be classified depending on the importance of parameters to be monitored, and these shall be categorized depending on the requirements concerning aseismic durability, needed power supply, and others.

< *Comparison with LWRs* >

GVs are installed in SFRs to be able to maintain sodium level in case of RV failure. IVR can be therefore confirmed to be achieved by referring the signals and their trend obtained from various detectors such as the sodium level meters in the GV, sodium level meters and thermometers in the RV, and thermometers in the underfloor area of the CV, contact-type sodium leak detectors, sampling-type ones, and others. Multiple measuring devices, as is the case in LWRs, shall be installed to monitor the parameters concerning the judgement of achieving IVR. These shall be reflected in order to update operational procedures during accidents as part of AM measures.

SFRs employ sodium as cooling medium differently from LWRs, and thus there exist no need for considerations on hydrogen generation by chemical decomposition of the coolant. Sufficient safety margins have been confirmed and needed measures have been examined for hydrogen generation sources in SFRs, such that sodium-concrete and sodium-water reactions. Massive hydrogen generation, therefore, is excluded in a phase where RV failure by molten fuel is not induced. Needed devices for measurement, however, shall be further investigated assuming a worst-case scenario, as an AM measure.

Sufficient aseismic durability is required for the device to measure important parameters, such as levels of coolant sodium and its temperatures in order to confirm the status of the core during SAs. Diverse measuring methods shall be provided including devices with small aseismic margins in order to enable flexible responses by considering their priority.

< *Corresponding articles in the new regulatory requirements* >

Articles 12, 23, 26, 59, and 61.

16. Needed measures shall be taken so that expedient AM measures can be implemented during SAs. System preparedness, operation procedures, documents and manuals, equipment and facilities, and others shall be established. And education and training shall be performed as well. These are required to enable emergency responses in an expedient and flexible manner against SAs. Effectiveness of AM measures shall be continuously improved by using the results of PRAs and so forth. Its effectiveness, at the same time, shall be also adequately and reversely reflected to PRAs by continuously assessing it using the results of training.

< *Detailed descriptions* >

Operation manuals shall be prepared in two ways so that responses are possible in an expedient and flexible manner against SAs. One is manuals for needed operations along with progresses of identified representative accident sequences. And another is manuals in response to symptoms observed in the plant. Effectiveness of these manuals shall be assessed through training by using them. Well-organized systems shall be prepared by ensuring assignment of needed personnel.

Major manuals to be considered are the followings:

- Manuals to shutdown the reactor and to maintain in a sub-critical state in case that the emergency reactor shutdown system fails,
- Manuals to transport the heat from the core to the ultimate heat sinks, and
- Manuals to prevent CV failure.

These manuals shall be prepared on the following points, considering lessons learned from 1F accident:

(1) Criteria to judge the priority of operations and their implementations

Parameters shall be defined in nuclear facilities that are used as criteria for judgment on operations, together with their measurement methods (reactor powers, temperatures and flow rates of reactor coolant, etc.). Criteria for judgment shall be also defined on selection of operations and their implementations. Parameters to be defined shall be the ones that are directly measurable. Substitutional measures for estimation, however, shall be prepared, when being reminded the fact that some parameters could not be measured during the 1F accident due to damaging and others.

(2) Habitability in central control rooms and emergency operation centers, and access routes on the site

Central control rooms shall allow operators to remain there even during SAs. Emergency operation centers shall be able to accommodate needed personnel for issuing necessary instructions, communicating with inside and outside the plant, responding against SAs, and others, and allow for emergency responses during SAs. Accessibility to the plant site is required for these facilities as well.

Accesses to some plant sites could not be secured during 1F accident due to the dispersed debris by the tsunami hit and high radiation exposure. Operations at the plant site could be considered to be necessary in Monju for valves at the ACSs and CV isolation valves in order to ensure natural circulation flow paths. Access route to the plant site and pathways in the buildings need to be secured during SAs under an imposed condition of debris and radiation doses in order to obtain an accurate view of damage at the site, and to prepare the transportable devices, and other equipment.

(3) Substitutional measures in case that a part of managing facilities against SAs are not available, and others

Management procedures shall be clarified in the operational manuals in case that some managing equipements against SAs are not available. Emergency equipments were not available due to loss of electric power at 1F accident. Manual operations at the plant site are required in Monju, for example, when valves are not actuated to allow natural circulation flow paths. These procedures shall be clarified in the operational manuals.

(4) Electric power sources to secure needed instrumentations to perceive plant statuses and communications with inside and outside the plant

Instrumentations, computers, PHSs, and wired paging equipments were not available in 1F accident due to loss of electric power. It is therefore essential to secure electric power sources for plant instrumentations and computers to implement AM measures, and communications with needed sectors such as inside the plant, the Japanese government, local governments, and others.

(5) Assessments on consequences when conducting operations and on long-term stability

Points of attention shall be mentioned in the manuals on consequences of conducting instructed operations (both beneficial and detrimental), and maintenance of long-term stable states.

(6) Other issues such as criteria for transition of manuals, procedures for revisions, and so

forth

Criteria for permitting the change of operational manuals to be followed shall be clarified in case that multiple manuals are prepared to manage SAs and others. It shall be noted that the priority of operational procedures is reviewed based on the progress of actual events.

Procedures for revisions of manuals shall be clarified for revision control.

Organizations to implement and support against SAs and others shall be established so as to enable expedient and flexible responses. Responsible persons and individual duty shall be defined, and needed personnel shall be assigned in these organizations. The measures against SAs and others shall be effectively implemented by these organizations.

Effectiveness of AM measures is important to be continuously improved by using the results of PRAs and so on. Its effectiveness, at the same time, shall be also adequately and reversely reflected to PRAs by continuously assessing it using the results of the training. Continual pursuit of safety enhancement is important by introducing most-recent findings. Flexible AM measures shall be continually pursuited by using progresses in safety researches on designs, assessments, technologies for securing safety, PRA methods, and so forth, in view of responses to the external environments as well.

< Comparison with LWRs >

Sodium is used as the coolant material in Monju. Therefore, the pressure of cooling systems is nearly equal to atmosphere. Therefore, the sodium level in RV can be secured by passive equipments such as the GV against coolant leakage events. Depressurization of the cooling system and emergency coolant injection, as is the case in LWRs, are not required for core cooling during an accident in SFRs. As the core cooling is achieved by natural circulation in SFRs, the measures to cope with SAs and others are to secure the coolant flow paths. Operation procedures shall be prepared, on the other hand, taking note of heat and hydrogen generation. The former is generated by sodium combustion at a sodium leakage, and the latter is generated by sodium-water reactions, due to high chemical reactivity of sodium. Major facilities of Monju are located at an elevation of 21.0 meters above sea level on a developed site at a northern part of Tsuruga Peninsula.

The backyard of Monju plant is surrounded by mountains at altitudes between 300 and 600 meters above sea level. It is required to secure access routes, electric power sources, cooling waters in responding to accidents considering the field site of Monju plant.

< Corresponding articles in the new regulatory requirements >

Articles 11, 29-31, 35, 44, 47-48, 55, 57-59, and 60-63.

Appendix 3

Reviewers' Comments

4 Concept for Prevention of Severe Accidents and Mitigation of Consequences
4.1 Basic Concept

Basic concepts and procedure of assessments on the beyond design basis accidents corresponding to the level 4 of the defense-in-depth concept

The basic concepts and procedure are well defined.

Basic design concepts and evaluating procedure against Paragraph (5) events are adequate.

Within the basic concepts and procedure of the assessment of BDBA adequate consideration is given to the selection of accident sequences groups and representative accident sequences supported with insight from PRA, to the identification of preventive measures and to the assessments of the effectiveness of the preventive measures. The effects of uncertainties have been considered.

Concept of SA prevention and mitigation is explained within the context of defense-in-depth, the method to identify accident sequences leading to core damage and loss of containment function are well explained.

The analysis strongly relies on PRA studies. In general terms, the postulated initiating events should be identified on the basis of engineering judgement and a combination of deterministic assessment and probabilistic assessment. They shall include all foreseeable failures of structures, systems and components of the plant, as well as operating errors and possible failures arising from internal and external hazards (also but not only earthquakes and tsunamis), whether in full power, low power or shutdown states (including handling operations).

The design includes two independent fast acting shutdown systems, each one should be capable of terminating anticipated transients without action of the other system. The systems should be actuated by functionally diverse signals wherever possible. Equipment diversity should be included in the design to minimize the possibility of degradation of the shutdown system due to postulated common mode failures. Both shutdown systems are expected to use mechanical poison rods inserted from above the reactor. Differences in the absorber assembly design, in the control rod drive mechanism design and in the disconnect and insertion features should be envisaged.

Basic concept implemented for Monju safety analysis assumes implementation of the following procedures:
(1) Selection of accident sequence groups considering the findings obtained from PRAs.
(2) Selection of a representative accident sequence for each accident sequence group.
(3) Identification of preventive measures for core damage against the selected representative

accident sequences.
(4) Assessments on the effectiveness of preventive measures against representative accident sequences.
Thus, selection of representative accident sequences is based on PRA (for example, accident with total instantaneous blockage of FSA cross-section (TIB) is excluded from consideration referring to its low probability) and preventive measures against core damage are identified only for representative accident sequences in assumption that they overlap other accident sequences.
In my opinion safety analysis should be carried out not only for representative BDBAs, but for all technically feasible BDBAs and effectiveness of preventive measures should be evaluated for all BDBAs too.

4.2 Selection of Accident Sequence Groups
4.2.1 Prevention of Significant Core Damage

(1) Accident sequence groups selected among plant internal events

Accident sequence groups are properly selected.

Systematic approach considering the design characteristics of MONJU.

The accident sequence groups induced by plant internal events are determined based on comprehensive PRA. The accident groups leading to core damage in SFR are appropriately identified based on the results of a level-1 PRA and are in full agreement with current international practice.

Selection of accident sequences based on a comprehensive PRA considering internal events is sufficiently explained. Identified ATWS events, loss-of-reactor-level and protected loss-of-heat-sink events sufficiently cover the wide spectrum of internal events that need consideration.

Again the postulated initiating events should be identified on the basis of engineering judgement and a combination of deterministic assessment and probabilistic assessment.

(2) Accident sequence groups selected among earthquakes and tsunami hits

Accident sequence groups are properly selected.

Ditto (Systematic approach considering the design characteristics of MONJU).

The method implemented for the selection of the accident sequence group seems consistent.

The accident sequence groups induced by plant external event are selected based on the results of event tree analysis considering earthquake and tsunami. The approach followed for the selecting has identified the adequate accident sequence groups.

Selection of accident sequences due to external events is sufficiently outlined, and selection of the station blackout scenario as the representative case to address all major external initiators (based on the greater likelihood of diesel generators losing their function occurring both seismic and tsunami events) is properly justified.

It could be useful to include tsunamis in all flooding events and to add severe meteorological events (typhoons, snow, ice…).

(3) Investigations concerning postulated initiators specific to SFRs

Candidates are properly investigated and selected.

It is quite reasonable that instantaneous flow blockage in a whole FA and large bubble passing through core are excluded from the consideration of possible initiators.

The selected initiators are adequate for the assessment of the effectiveness of the protective measures against core damage. The judgment to consider the instantaneous flow blockage in a whole fuel assembly and the passage of large bubble through the core as practically eliminated is in agreement with current SFR safety approach.

The reviewer also fully agrees with exclusion of total instantaneous blockage and large gas bubble scenarios due to their very small likelihood of happening due to implemented MONJU design features.

Reviewer wants to suggest adding the information indicating that the large sub-assembly flow blockage is selected or not as an accident sequence group in other countries.

Large sodium fire is considered to be mentioned in view of ensuring containment function (sodium fire can be used for containment design basis).

Although probability of the TIB accident is extremely low, but it is feasible, therefore analysis of consequences of this accident for Monju would be reasonable for evaluation of its self-protection against this accident.

4.2.2 Prevention of Fuel Failure Stored in Spent Fuel Storage Tanks

(1) EVST

Accident sequence groups are properly selected.

Selection of accident sequence groups is proper considering the design characteristics of MONJU.

The postulated accident sequence groups are adequate for the assessment of the prevention of fuel failure in EVST of Monju.

Ex-vessel storage tank with redundancy (three independent coolant systems with forced circulation of sodium) and flexibility to rely on natural circulation provides robust capacity for decay heat removal. Outer vessel surrounding the tank should also provide sufficient measure to maintain sodium level.

Storage arrangements for both new and spent fuel shall have adequate margins to ensure sub-criticality under fault conditions including those of ingress of moderator and mis-loading by the operator.

A full criticality safety assessment shall be performed for all normal operating and fault conditions allowing for operator errors. This assessment shall be repeated if the fuel composition should be changed.

Firefighting equipment in areas of the fuel route where significant quantities of fuel are stored dry or under sodium should not depend on the use of water or other efficient neutron moderating materials.

(2) Spent fuel pool

Accident sequence groups are properly selected.

Ditto (Selection of accident sequence groups is proper considering the design characteristics of MONJU).

The postulated accident sequence groups are adequate for the assessment of the prevention of fuel failure in spent fuel pool of Monju.

Spent fuel pool provides sufficient grace period, and a concern for water leakage due to a stuck valve is properly addressed.

There is no indication of the failure to take into account leakage through the walls.

4.2.3 Prevention of Fuel Damage Loaded in the Core during Reactor Shutdown

Postulated accident sequence groups for the reactor at shutdown state

Accident sequence groups are properly selected.

Ditto (Selection of accident sequence groups is proper considering the design characteristics of MONJU).

The method implemented for the selection of the accident sequence group seems consistent.

The postulated accident sequence groups are adequate for the assessment of the prevention of fuel damage loaded in the core during reactor shutdown.

Reactivity insertion, coolant leakage, loss of decay heat removal, and station blackout scenarios adequately cover the spectrum of challenges MONJU can experience during a maintenance mode, and design provisions available are sufficient to avoid consequences of fuel failures during such scenarios.

4.2.4 Ensuring Containment Function

(1) Selection of accident sequence groups

Accident sequence groups are properly selected through adequate method.

Consistent approach.

The method implemented for the selection of the accident sequence group seems consistent.

The accident sequence groups are the same as those against core damage and are adequate for assessments of the effectiveness of measures for ensuring containment function.

Only the accidents with sequences that lead to significant core damage are identified as threats to containment function.

This reviewer feels that large sodium fires should also be included in the spectrum of challenges the containment structure is expected to withstand.

(2) Selection of representative accident sequences

(a) ATWS

Accident sequence groups are properly selected through adequate method.

Systematic approach considering the design characteristics of MONJU.

The method implemented for the selection of the accident sequence group seems consistent.

The selection of representative accident sequence groups for ATWS events is according to current SFR safety assessment practice. The analyses and results are in agreement with current SFR safety approach practice.

Selected events (ULOF, UTOP, ULOHS) adequately cover the ATWS spectrum.

(b) LOHRS

Accident sequence groups are properly selected through adequate method.

Ditto (Systematic approach considering the design characteristics of MONJU).

The method implemented for the selection of the accident sequence group seems consistent.

The selection of repetitive accident sequence groups for LOHRS events is according to current SFR safety assessment practice.

Representative accident sequences for initiating events that lead to loss of reactor level and heat sink are comprehensive.

4.3 Concept for Preventive Measures against Significant Core Damage

4.3.1 Prevention of Significant Core Damage

Concepts are well defined.

Preventive design measures and requirements are adequate considering the design characteristics of MONJU.

The concept for preventing significant core damage follows the safety review guidelines for LWR considering specific characteristics of SFRs. Representative accident sequences are selected, the progression of events is analyzed and the effectiveness of the measures is assessed.

Preventive measures largely based on assurance of natural circulation decay heat removal through at least one of the three loops, sodium coolant inventory maintenance, sodium heating system to avoid freezing, continuous monitoring should provide sufficient assurance against wide range of initiating events to avoid significant core damage. MONJU's safety features to ensure transition to natural circulation using batteries or manual transition and emergency procedures such as siphon breaking operation, installation of transportable electric power supply vehicles, and establishment of emergency access routes are additional measures that should provide an added layer of assurance for external events with magnitude beyond the design basis.

Diversity and redundancy in decay heat removal systems, including any necessary services or supplies, should be provided appropriate to reliability requirements of specific fault conditions and with each route contributing an appropriate fraction to the required reliability.

There should be physical segregation between decay heat removal systems and other potentially hazardous systems to prevent decay heat removal being jeopardized as a consequence of failure of these systems and by internal or external hazards.

The design of the decay heat removal system should promote natural circulation. This should be tested adequately at the commissioning stage.

4.3.2 Prevention of Fuel Damage in Spent Fuel Storage Tanks

Concepts are well defined.

Ditto (Preventive design measures and requirements are adequate considering the design characteristics of MONJU).

The preventive measures against spent fuel damage seem consistent with current practices.

The concepts for preventive measures against spent fuel damage for ex-vessel fuel storage tank are equivalent to the safety review guidelines for LWR.

4.3.3 Prevention of Fuel Damage Stored in the Core during Reactor Shutdown

Concepts are well defined.

Ditto (Preventive design measures and requirements are adequate considering the design characteristics of MONJU).

The preventive measures against damage of fuels stored in the core during reactor shutdown seem consistent with current practices.

The concepts for preventive measures against damage of fuel stored in the core during reactor shutdown are equivalent to the safety review guidelines for LWR.

Preventive measures based on maintaining the reactor at subcritical state (eliminating inadvertent reactivity insertion risk), maintaining the sodium coolant at a level to facilitate decay heat removal, and facilitating natural circulation should provide robust and sufficient capability to avoid core damage during reactor shutdown.

4.4 Concept for Ensuring Containment Function
4.4.1 Anticipated Transient without Scram (ATWS)

(1) Basic Concepts

Basic concepts are well defined.

Focus of evaluation and event progression are typical to Oxide fuel.

The basic concept follows established SFR safety assessment approach considering ULOF as a representative for ATWS events for the assessment of ensuring containment function. The evaluation of the accident progression indicates the feasibility of In-Vessel Retention (IVR) considering the most challenging conditions. The viability of IVR could be further enhanced by establishing appropriate accident management measures.

(2) Most likely progress of events identified for ATWS

The progress of events is properly identified.

The accident progression seems consistent with current understanding of the phenomena.

The accident progression without prompt recriticality and mechanical energy release consisting of the initiating phase, the transition phase, the post-accident material relocation and the post-accident heat removal is in agreement with current understanding of the phenomena.

This reviewer agrees with adequacy of the outlined progress of events with a reference to comment above.

(3) Evaluation on the influences of uncertainties dominant to the most probable event progression

Uncertainties are properly assessed with adequate method.

The evaluation of the influences of uncertainties seems consistent.

The evaluation of the influences of uncertainties in dominant factors on the most likely progress of events for ATWS enhances the conservativeness of the results.

State-of-the-art methods/codes are used to assess MONJU's response to ATWS events. Approaches to model MONJU's core and PHTS against accidents that lead to fuel melting appear to be more than sufficiently detailed and up to the standards required for a license review. Validity of analysis results have been confirmed by data from in-pile tests. Impact of uncertainties has been considered as a plus. Conclusions drawn based on the results of these analyses appear to be in-line with this reviewer's expectations for a robust SFR plant like MONJU.

4.4.2 Loss of Heat Removal System (LOHRS)

(1) Basic Concepts

Concepts are well defined.

Strategy for redundant measures coping with LOHRS is appropriate.

The described basic concepts for measures for ensuring containment functions during loss of heat removal system event are adequate. The focus is to prevent core damage maintaining the integrity of the reactor vessel by removing the decay heat using additional equipment assuming the failure of the preventive measure against core damage.

Based approach to assure containment function even with assumption of DBA mitigation measure failures and core damage prevention measure failures is more than adequate. Normally, frequency of such event sequences will fall well below the threshold for accidents in the residual risk category.

Putting MCS into operation should be possible to be made not only in the central control room, but in the emergency operation centers.

(2) Accident Sequences

Concepts are well defined.

Multiple measures against LOHRS is considered to be effective for preventing core damage and RV failure for MONJU design.

The approach followed for ensuring containment function during LOHRS-type events seems consistent with current practices.

The concepts for ensuring containment function during LOHRS-type events employ multiple measures including heat removal by natural circulation using the ACSs and the use of the maintenance cooling system. These multiple measures provide adequate protection with high reliability against core damage and reactor vessel failure even in the case of the failure of DBA mitigation measures.

Concepts presented adequately cover the two events considered (loss of reactor coolant level due to leakages, and less of decay heat removal capability) including various factors such as siphon breaks and various modes and venues to cool the core. This reviewer agrees with the characterization of the differences between the containment structures for LWRs and SFRs and agrees that the pressure relief systems and recirculator units are not needed for an SFR containment structure. This reviewer also agrees that, as long as the reactor core remains covered and natural circulation paths are open to achieve core debris coolability, failure of core support structure or primary coolant boundary can be practically eliminated. This reviewer believes that hydrogen explosions are not a threat to the MONJU containment.

(3) Summarized results of concepts for ensuring containment function in SFRS

Summary is well written.

Inherent & unique design features of SFR and key design measure of IVR for ensuring containment function are well addressed compared with high pressure LWR system.

There is no mechanical load for containment is LOHRS and ATWS events.

The approach followed for ensuring containment function seems consistent with current practices.

In SFRs, the grace time for core damage in LOHRS-type events has large safety margin. The multiple protective measures avoid core damage and reactor vessel failure. The safety margin is further enhanced by providing redundant measures. The described concepts are adequate for ensuring containment function.

This reviewer agrees with the adopted approach to achieve containment function by avoiding primary coolant boundary through redundant measures against severe accidents.

This is owing to the wide temperature range of sodium being available in a liquid state, the point is not the boiling temperature of sodium but the temperature for which we can ensure the resistance of the vessel.

(4) Concepts for instrumentation and monitoring during SAs

Concepts are well defined.

Intended overall measurement strategy is proper for monitoring the sequence of SA.

The concepts for instrumentation and monitoring during severe accident provide suitable means for engineering judgment of the accident progression.

In addition to the standard instrumentation, monitoring the sodium level in the reactor vessel, as well as potential leakages, and relying on delayed neutron detectors as an indication of fuel failures should provide sufficient assurance to achieve in-vessel retention.

Recommendation is that instrumentation and monitoring during SAs shall be provided not only in the central control room, but in the emergency operation centers.

7. Concept for Securing Safety of SFRs

Requirement 1

Appropriate general design requirement to address overall safety and reliability goals.

This requirement sets safety in design to achieve the highest level of safety that can reasonably be achieved consistent with the safety objectives by implementing the concept of defense in depth to prevent accidents and mitigate their consequences. The requirement is equivalent to the criteria 6 and 7 in SDC for Gen-IV SFRs.

Agreement would depend on what is socially acceptable. Regulatory requirements for dose limits should be used as a reference, not socially acceptable ones.

Some characteristics that would improve the safety level call for an as-exhaustive-as-possible identification of the risks that may impede the fulfillment of fundamental safety functions, the elimination of potential short sequences following the failure of a particular provision leading to major consequences without any possibility of restoring safe conditions, the rejection of potential "cliff edge effects" from deviation of physical parameters leading to major accidents, the availability of an adequate grace period during accidental situations and the absence of major sequences participating in an excessive manner to the global frequency of the damaged plant states.

Requirement 2

It is a typical design.

This requirement is adequately fulfilled by the two reactor shutdown system installed in Monju which have sufficient capability to reliably shutdown the reactor independently with sufficient safety margin under operation and accident conditions.

Requirement to assure reactor shutdown capabilities with redundancy and independence against DBAs is appropriate.

However, requiring diversity in a way that is accomplished in LWRs with a boron dilution system may be overprescriptive. This reviewer shares the view that MONJU's shutdown systems are sufficient to achieve safety goals.

Rather than limited to control rod, it would be better for scope of the requirements to be expanded to reactor protection system considering redundancy, diversity, independency, isolation, reliability, testability and so on.

Requirement 3

The three decay heat removal systems in Monju provide redundant provision for decay heat removal from the core. The heat is rejected into air from the secondary system both operating in forced circulation. Further, decay heat removal by natural circulation capability is feasible in both in primary and secondary systems.

Requirement to assure decay heat removal against DBAs is appropriate. Distinction of decay heat removal systems from high-pressure injection systems (ECCS) used in LWRs is important.

It would be necessary to address the redundancy, diversity and independency of decay heat removal system as a general design requirement.

The active part of decay heat removal systems should be equipped according to the concept of diversity and independence.

Appropriate measures shall be needed for leak detection and for ensuring the retention of sufficient sodium inventory in the secondary system as well as isolation capability.

The sentence in Requirement 3 "The heat transport systems and ultimate heat sinks shall not lose their functions and integrity" is rather general. In my opinion, more appropriate requirement is as follows: "Decay heat removal systems shall be equipped according to the concept of redundancy, diversity and independence excluding loss of their functions due to single failures".

Requirement 4

Appropriate design requirement against fire and flooding to preventing CCF.

Requirement to assure measures against fires and flooding to avoid common cause failures is appropriate.

The approaches against fires and internal flooding aim to enhance the safety measures in the plant to prevent a loss of safety function caused by common cause failure. The assessment of the consequences of internal flooding is necessary.

Requirement 5

Appropriate design requirement against sodium leakage and SWR including following-up measures.

The design-basis measures give adequate consideration to the specificity of sodium, including leak detection and mitigation of sodium water interaction.

Requirement to address reactive nature of sodium coolant as a unique SFR issue is appropriate.

Proper AM measures shall take into account toxic environment to extinguish fires.

Assessments to establish the adequacy of the measures in case of BDBA and if needed the implementation of additional measures are necessary.

Recommendation is to include into Requirement 5 consideration of validity of present measures against sodium leakage in the primary cooling system.

Requirement 6

Appropriate design requirement from the discussion of Ch. 4.

The accident sequence groups are identified by comprehensive PRA. The assessments of the consequence of ATWS and LOHRS, the identification of preventive measures and the effectiveness of the preventive measures have been performed adequately. Further, the need for the deployment of adequate accident management measures has been recognized.

Requirement 7

The safety margin is enough for MONJU.

Appropriate design requirement considering the design characteristics of MONJU. In PWR, diverse protection system (non-safety grade) is employed for reactor shutdown as a measure for ATWS in addition to auxiliary feed water and injection of boric-acid solution.

ATWS events may only occur as a result of multiple redundant and diverse safety provisions experiencing multiple failures. The design considerations for preventive measures for reduction of the occurrence frequency, the design measures to mitigate consequences and the AM procedures for IVR have been confirmed to satisfy the safety requirements.

How fast an ATWS events progress depends on the event (ULOHS could take significantly longer than ULOF).

It shall be considered the ATWS events in SFRs to proceed extremely fast due to its core characteristics… "extremely fast" is a bit too hard.

For ATWS, a key design impact is in providing a Plant Protection System (PPS) or shutdown system with an extremely low failure rate (<10-6 per year). This should be accomplished with two redundant, independent, fast acting systems, each system containing (1) diverse sensors, (2) diverse logic, and (3) diverse circuitry, and each actuating separate, diverse sets of neutron absorber rods. However, if sufficiently high reliability (or low failure rates) cannot be achieved and demonstrated then the other criteria or approaches become overriding in design. The acceptance criteria for primary systems integrity, fuel integrity, containment integrity, long term cooling and shutdown, and mitigating systems design may become dominant design considerations.

It should be noted that sodium is heated in ULOF accident over its boiling point not only at the central part of the core subassemblies, but at the top part of subassemblies too.

Requirement 8

Appropriate design requirement on AM measures against LOHRS considering the design characteristics of MONJU.

Adequate consideration has been given to the importance of highly reliable AM measures for LOHRS events. The need for improvement of some of the AM measure has been identified. Considering characteristics of SFRs such as the existence of sodium in a single phase over a large temperature range, the feasibility of decay heat removal by natural circulation and associated large grace period, the implementation of effective AM measures could significantly contribute to the improvement of the safety provisions.

See comment for item 37 (overall comment) below.

It is recommended to provide AM measures for case when siphon breaking operation fails.

Requirement 9

Appropriate design requirement to address the IVR of SFR to ensure the containment function.

Considering the large grace time for significant core damage and the possibility of decay heat removal by natural circulation for LOHRS, the implementation of multiple AM measures could practically eliminate the possibility of RV failure due to molten core material. For ATWS events, preventive measures against core damage are adequately considered. The need for accurate evaluation of the event progression, the load to RV and the feasibility of IVR has been identified considering the impact of uncertainties.

Requirement 10

Appropriate general design requirement on protection against natural phenomena reflecting the lessons learned from the 1F accident.

The investigation of the appropriateness of the provisions against external events on design basis with sufficient safety margins by assessing risks using PRA has been considered including further investigations to be carried out into the possibility of hazard beyond DBA in order to implement multiple and diverse AM measure considering update of knowledge and new findings. The approach is in agreement with current engineering practice.

A complete analysis of natural hazards is pivotal to a consistent safety assessment of a NPP. Additional attention should be devoted to assess complex scenarios that involve multiple consecutive hazards (e. g. earthquake + fire) as well as common cause failures following such events. The assessment should consider the impact of natural disaster, not only to the plant but to the whole infrastructures surrounding the site and likely to be similarly affected.

Requirement 11

The necessity to investigate the effect of aircraft crash and other terrorist attacks on the loss of safety function for core cooling and containment function has been adequately identified including implementation of adequate measures considering site specific conditions.

This requirement seems to include the security concern, not limited to aircraft crash.

The need for the harmonization of the assessment methodologies for the safety and security related design is among the explicit objectives formulated by WENRA to improve the safety of future nuclear plants (WENRA Statement on Safety Objectives for new Nuclear Power Plants, November 2010):
O5. Safety and security interfaces: ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.
The interpretation of this objective is detailed by WENRA within the "Safety of new NPP designs – Study by the WENRA Reactor Harmonization Working Group RHWG March 2013.

Requirement 11 related to potential external human-produced impact on SFR safety does not include consideration of external industrial impact. It is a recommendation to modify the first sentence of Requirement 11 as follows: "Measures shall be taken against external industrial impact, intentional large-aircraft crashes and other terrorism to prevent occurrence of SAs and to mitigate their consequences".

Requirement 12

Appropriate design requirement on preventive measures against hydrogen generated by sodium-water and sodium-concrete reactions.

There exists no concern for massive hydrogen generation in SFRs. Sodium-concrete and sodium-water reactions are major source of hydrogen. Sodium-concrete contact is prevented by the GV for a leak from primary system and by steel lining covering floors and walls. Combustion of sodium is prevented by the inertisation of equipment rooms. Hydrogen generated in SG is discharged to a storage tank. These measures are adequate to prevent massive hydrogen generation.

This reviewer believes that hydrogen explosions are not a threat to the MONJU containment, and agrees with JAEA approach to address this concern.

No need for SFR.

Requirement 13

Appropriate design requirement on preventive measures against fuel failure in spent fuel storages considering the design characteristics of MONJU.

The envisaged measures to prevent fuel failure in the EVST are adequate.

Requirement to fuel failure in ex-vessel storage tank and spent fuel pool is appropriate. This reviewer agrees with JAEA approach to address this requirement.

List of required capabilities for EVST and spent fuel pools is recommended to add with additional item 6 on page 81: "It is required to prevent decrease of liquid level in EVST and spent fuel pools below emergency level that leads to failure of decay heat removal from spent fuel".

Requirement 14

Appropriate design requirement on habitability of control rooms for accident management.

Adequate consideration has been given to assure the habitability of the control room and emergency operation centers.

Requirement to address control room and emergency operation center habitability is appropriate. This reviewer agrees with JAEA approach to address this requirement.

The progress of events in SFRs is different from those in LWRs.

AM measures should take into account the potential unavailability of instrumentation monitoring the status of the plant.

I think this criterion should include additional requirement of fully independence of central control room and emergency operation centers that permits to exclude simultaneous failure of central control room and emergency operation centers due to common reason.

Requirement 15

Appropriate design requirement on monitoring plant status during SA and measurement devices.

The need to install adequate devices to monitor the reactor and the plant during SA has been appropriately identified.

Requirement to address the need for monitoring core and primary coolant boundary during severe accidents is appropriate. This reviewer agrees with JAEA approach to address this requirement.

It is not easy to know the core situation in severe accident. As you mentioned, IVR can be confirmed to be achieved by sodium level meters, thermometers, contact-type sodium leak detectors, sampling-type ones. The core situation could be concluded by these measures.

Requirement 16

Appropriate design requirement on general AM measures.

Adequate consideration is given to the development and implementation of plant‐specific AM measures which could be developed to plant-specific Severe Accident Management Guidelines (SAMGs) with the objective to provide staff guidelines for SA management to mitigate core damage, to maintain containment integrity and to minimize offsite releases.

Requirement to address robust severe accident management system and guidelines is appropriate. This reviewer agrees with JAEA approach to address this requirement.

The AM is just a complementary measure for severe accident.

Any opinions or comments on safety requirements for SFRs as a whole (No choice is needed.)

The review is based on the information provided in the report. The views expressed are purely those of the reviewer.

The comments provided are based on the reviewer's knowledge and experience in SFR technology. Without having a detailed knowledge of Monju's Safety Analysis Report, some comments may result redundant.
The approach presented in the document is systematic and duly substantiated. The confinement of radioactive material should be ensured in all situations, normal and accidental, including situations with core degradation, and for all states of plant operation. Particular attention should be paid to controlling the risk of bypass, and the performance of containment under severe accident conditions, including the capability of the primary circuit to sustain the release of mechanical energy, the behaviour of the building, and the effectiveness of associated systems.

There is no firm regulatory framework on BDBAs yet in the US. NRC has the authority for establishing new requirements based on engineering judgment, operating experience, deterministic and probabilistic assessments to improve the safety by enhancing the plant's capabilities to withstand accidents that involve additional failures and are more severe than design basis accidents. However, NRC has done this only for limited conditions including station blackout, Anticipated Transients Without Scram, and certain external threats such as aircraft Impact, explosion or fire. The NRC is currently considering a recommendation from the staff (SECY-13-0132) to formally recognize these requirements as Design Extension (DE) requirements. Applicants seeking a COL under 10 CFR Part 52 are required to identify such conditions and address them in the Final Safety Analysis Report. The NRC staff's evaluation for compliance with these requirements is guided by the Standard Review Plan's "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors." The requirements 6-11 above appear to be consistent with NRC approach.

It would be useful to make prioritization in fulfillment of preventive measures and other AM measures for any BDBA under consideration.

Appendix 4

Score Sheet of Reviewer's Ratings

Ranking
5 : Completely supportive
4 : Rather supportive
3 : Neutral
2 : Rather negative
1 : Completely negative

No.	Chapter /Section	Check Item	Ashurko (RU)	Dufour (FR)	Kang (KR)	Lee (KR)	Liu (CN)	Luca (EU)	Ren (CN)	Sofu (US)	Tamirat (EU)	**Average**
1	4.1	Basic concepts and procedure of assessments on the beyond design basis accidents corresponding to the level 4 of the defense-in-depth concept	2	5	5	5	4	4	4	5	5	**4.3**
2	4.2.1	Methods and results of selecting the accident sequence groups extracted among ones induced by internal events of the plant	4	5	4	5	5	4	4	4	5	**4.4**

No.	Chapter /Section	Check Item	Ashurko (RU)	Dufour (FR)	Kang (KR)	Lee (KR)	Liu (CN)	Luca (EU)	Ren (CN)	Sofu (US)	Tamirat (EU)	**Average**
3	4.2.1	Methods and results of selecting the accident sequence groups from ones caused by earthquakes and tsunamis	5	4	5	5	5	4	4	5	5	**4.7**
4	4.2.1	Candidates for postulated initiators specific to SFRs to be examined	2	5	4	4	5	4	3	5	5	**4.1**
5	4.2.2	Postulated accident sequence groups in consideration of the features of EVST in Monju	5	5	5	5	5	4	4	5	5	**4.8**
6	4.2.2	Postulated accident sequence groups in consideration of the features of spent fuel pool in Monju	5	4	5	5	5	5	4	5	5	**4.8**
7	4.2.3	Postulated accident sequence groups for the reactor at shutdown state	5	5	5	5	5	4	3	5	5	**4.7**
8	4.2.4	Methods and results of selecting the accident sequence groups for assessments on effectiveness of measures for ensuring containment function	5	5	5	5	4	4	3	4	5	**4.4**

No.	Chapter /Section	Check Item	Ashurko (RU)	Dufour (FR)	Kang (KR)	Lee (KR)	Liu (CN)	Luca (EU)	Ren (CN)	Sofu (US)	Tamirat (EU)	**Average**
9	4.2.4	Methods and results of selecting the representative accident sequence groups for ATWS events	3	5	5	5	5	4	4	5	5	**4.6**
10	4.2.4	Methods and results of selecting the representative accident sequence groups for LOHRS events	5	5	5	5	5	4	4	5	5	**4.8**
11	4.3.1	Concepts for preventing significant core damage	5	4	5	5	4	4	4	5	5	**4.6**
12	4.3.2	Concepts for preventive measures against spent fuel damage	5	5	5	5	5	5	4	3	5	**4.7**
13	4.3.3	Concepts for preventive measures against damage of fuels stored in the core during reactor shutdown	5	5	5	5	4	5	4	5	5	**4.8**
14	4.4.1	Basic concepts (Philosophy)	5	5	5	5	5	5	4	5	5	**4.9**
15	4.4.1	Most likely progress of events identified for ATWS (initiating phase, transition phase, PAMR and PAHR phase)	5	5	5	-	5	4	4	5	5	**4.8**

No.	Chapter /Section	Check Item	Ashurko (RU)	Dufour (FR)	Kang (KR)	Lee (KR)	Liu (CN)	Luca (EU)	Ren (CN)	Sofu (US)	Tamirat (EU)	**Average**
16	4.4.1	Methods and results of assessing the influences of uncertainties in dominant factors on the most likely progress of events for ATWS (initiating phase, transition phase, RV response phase, PAMR and PAHR phase)	5	4	4	-	5	4	4	5	5	**4.5**
17	4.4.2	Basic concepts (Philosophy)	5	5	5	5	5	5	4	5	5	**4.9**
18	4.4.2	Concepts for ensuring containment function during LOHRS-type events	5	4	5	5	5	4	4	5	5	**4.7**
19	4.4.2	Summarized results of concepts for ensuring containment function in SFRs	5	4	5	5	5	4	4	5	5	**4.7**
20	4.4.2	Concepts for instrumentation and monitoring during SAs	5	5	5	5	5	5	5	5	5	**5.0**
21	7	Requirement 1	4	5		5	5	4	5	4	5	**4.6**
22	7	Requirement 2	5	5		3	5	4	5	4	5	**4.5**
23	7	Requirement 3	4	5		4	4	4	5	5	5	**4.5**

No.	Chapter /Section	Check Item	Ashurko (RU)	Dufour (FR)	Kang (KR)	Lee (KR)	Liu (CN)	Luca (EU)	Ren (CN)	Sofu (US)	Tamirat (EU)	**Average**
24	7	Requirement 4	4	5		5	5	5	5	5	5	**4.9**
25	7	Requirement 5	4	5		5	5	4	5	5	5	**4.8**
26	7	Requirement 6	5	5		5	4	4	5	4	5	**4.6**
27	7	Requirement 7	5	4		5	4	4	5	4	5	**4.5**
28	7	Requirement 8	4	5		5	5	5	5	4	5	**4.8**
29	7	Requirement 9	5	4		5	5	4	5	4	5	**4.6**
30	7	Requirement 10	5	5		5	5	5	5	4	5	**4.9**
31	7	Requirement 11	4	5		4	3	5	5	4	5	**4.4**
32	7	Requirement 12	5	5		5	3	5	3	5	5	**4.5**
33	7	Requirement 13	5	5		5	5	4	5	5	5	**4.9**
34	7	Requirement 14	4	5		5	5	5	5	5	5	**4.9**
35	7	Requirement 15	5	5		5	4	5	5	5	5	**4.9**
36	7	Requirement 16	5	5		5	4	4	5	5	5	**4.8**