GENES4/ANP2003, Sep. 15-19, 2003, Kyoto, JAPAN
Paper 1154

LMFBR Design and its Evolution: (3) Safety System Design of LMFBR

Hajime NIWA[1*], Shigenobu KUBO[2] and Kennichi KURISAKA[1]
[1] *O-arai Engineering Center, Japan Nuclear Cycle Development Institute, O-arai, Ibaraki, 311-1393, Japan*
[2] *The Japan Atomic Power Company, Tokyo, 101-0053 Japan*

In this paper, first, the evolution of the approach for safety design of liquid metal cooled reactors is briefly reviewed. A basic safety approach in designing LMFBRs is the defense-in-depth concept which is essentially the same as one taken in LWRs. The subject of severe accidents or CDAs(core disruptive accidents) has been and will be an important safety consideration for LMFBRs, which is addressed as beyond design basis events to provide or confirm an additional safety margin of the plant strictly designed for DBEs. Second, safety design requirement for future concept is disccussed. The safety targets are (1) sufficient smaller risk than existing one without off-site emergency responses (2) equal or higher level of safety compared with LWRs in the same generation. The elimination of recriticality concern as well as post-accident long-term debris cooling capability aiming at in-vessel retention for postulated CDAs is stressed. And third, the current safety design of the advanced loop type sodium cooled fast reactor is introduced that are considered in the framework of the Feasibility Study on Commercialized Fast Reactor Cycle Systems. The safety design of the plant is now in progress taking account of not only the system characteristics but also design studies and R&D experiences so far.

KEYWORDS: liquid metal cooled fast breeder reactor, sodium-cooled, core disruptive accident, in-vessel retention

I. Introduction

During the last several decades, liquid-metal-cooled fast breeder reactors (LMFBRs) have been developed widely in the world mainly because of the following two reasons. First fast neutron reactors with liquid-metal coolant, which has less neutron absorption and moderation characteristics than conventional light water reactors (LWRs), allow us to effectively breed nuclear fuel or burn plutonium and minor actinides while they generate electric power. Second good heat-transport characteristics of liquid metal can allow us to design a compact, high-performance and low-pressure reactor system. Historically more than twenty experimental and power-generating reactor plants have been actually constructed in the world and some of them are in operation, although commercialization of LMFBRs is yet to be pursued mainly because of necessity of further reduction in construction cost.

The Feasibility Study on commercialized Fast Reactor Cycle Systems in Japan(hereafter described as F/S) has been conducted since 1999 in order to clarify the perspectives for commercialized FR cycle by making the maximum use of its primary advantages so that the FR cycle can achieve economic competitiveness comparable to that of LWRs and other base power sources.[1)] Within the framework of F/S, the safety design principle is investigated and several kinds of design studies are now in progress. Among the designs for LMFBRs, the advanced loop type sodium cooled fast reactor is one of the promising candidates as a future commercialized nuclear power plant, which was contributed to discussion on the technology roadmap for Generation IV nuclear energy systems as the one of major option of the Sodium-Cooled Fast Reactor.[2)]

In this paper, first, the evolution of the approach for safety design of LMFBRs is briefly reviewed, second, safety design requirement for future concept is proposed, and third, the current safety design of the advanced loop concept is introduced that are considered in the framework of the F/S.

II. Evolution of the safety design approach[3)]

1. Key Safety Characteristics of LMFBRs

A basic safety approach in designing LMFBRs has been essentially the same as one taken in LWRs. Namely a reactor system shall be designed based on a defense-in-depth concept, with a primary emphasis on preventing and detecting abnormal occurrences. Then safety design measures shall provide appropriate means to shut the reactor down, cool the residual heat and contain radioactive materials within the reactor facility, by taking the safety characteristics of sodium coolant into account.

Even though the philosophy involved in the defense-in-depth concept has been universally accepted, what are technically included in it should not be regarded as solid static entities. An innovative reactor design with effective passive features to prevent and mitigate severe accidents may practically eliminate the need for off-site emergency procedure, which is defined as fifth defense line in the International Atomic Energy Agency document.[4, 5)]

The DBEs, commonly consisting of anticipated operational occurrences and (design basis) accidents, are postulated for safety analyses to confirm the validity of plant safety features and their functions in a conservative way. For LMFBRs, with a low pressure system and single-phase coolant system, the sequences of DBEs are rather benign. Even an incident of severe coolant leakage is unlike to lead to rapid core uncovery. Hence there is normally no need to

* Corresponding author, Tel. +81-29-267-4141,
Fax. +81-29-267-1676, E-mail: niwa@oec.jnc.go.jp

provide an emergency core cooling system that is commonly equipped in high-pressure LWR systems. The specific safety characteristics of coolant, however, must be carefully taken into account in LMFBRs. For example, chemical reactions of sodium are usually treated in safety analyses, since their consequences can be energetic when it comes into contact with air or water upon coolant boundary failure.

The subject of severe accidents or core-disruptive accidents (CDAs), historically called, has been and will be an important safety consideration for LMFBRs as well. A CDA in LMFBRs is characterized, in comparison with LWRs, by an energetics potential due to recriticality events in a fast-neutron reactor core, which is not designed in a most reactive configuration. That is, coolant voiding or fuel re-configuration can bring the reactor to a more reactive state with positive reactivity feedback mechanisms. CDAs were treated in the LMFBRs in the past in some ways. A common practice has been to regard CDAs as beyond-designed-basis events and to take an approach different from DBEs, with using a best-estimate oriented evaluation method.

2. Safety Approaches in 1970s and 1980s

The development of sodium-cooled fast reactors has a long history. There are several LMFBR plants, actually designed, built and or operated during the 1970s and 1980s, such as Superphenix in France, SNR-300 in Germany, The Clinch River Breeder Reactor Plant (CRBRP) in the United States and Monju in Japan. In a word, a rather coherent safety approach was taken in those LMFBRs developed in the 1970s and 1980s. Namely, these plants were designed based on the defense-in-depth principles with appropriate consideration on sodium reactions. Even though their early designs consider CDAs directly in the safety design, the treatment in safety evaluation is different from DBEs with best-estimate method and assumptions being commonly used. More recently they tend to be regarded clearly as an event category of beyond DBE. The purpose of CDA analysis is therefore to provide or confirm an additional safety margin of the plant strictly designed for DBEs.

(1) Superphenix [6,7]

Superphenix is a large pool-type commercial-size fast reactor of a 1240 MWe output. The design of the plant was made feasible from French experience in the former two fast reactors, Rapsodie and Phenix. The basic safety approach is to provide safety design measures based on a defense-in-depth concept. A special feature is that large hydraulic and thermal inertia of the coolant provides an inherently stable plant dynamics, as well as the negative reactivity feedback due to radial expansion of fuel and blanket subassemblies. For instance, a long grace time is available for more than 10 minutes before coolant boiling even in the event of unprotected (without scram) accidents.

CDAs are defined not as a part of DBEs but in a special accident category to define a containment design basis. The accident sequence is treated on a rational way in contrast with conservatively treated DBEs. It was evaluated that the reactor vessel and the roof (forming the intermediate containment) can accommodate the CDA energetics. An in-vessel core catcher system is provided to cool and retain post-accident core debris.

In spite of a later modification in the secondary cooling system to strengthen the plant capability to withstand large-scale sodium leakage, the government decision was made to stop the operation of the plant.

(2)SNR-300 [8,9,10]

The SNR-300 plant, a prototype fast reactor in Germany, is a loop-type reactor of a 327 MWe class. The basic safety approach is to provide safety design measures based on a defense-in-depth concept. Based on a multi-step licensing procedure in Germany, the requirements were specified by the licensing authority to provide design measures against the consequences of CDAs during an early conceptual design stage. Namely the reactor containment system is required to withstand a certain amount of mechanical energy released as a result of prompt-critical power excursion (recriticality). Further the reactor cavity is equipped with a special floor cooling device (ex-vessel core catcher). These imply that CDAs are treated very similarly to DBEs from the safety design points of view. However, the sequences of CDAs are generally analyzed in a more best-estimated-oriented way with reasonably considering phenomenological uncertainties.

The construction of SNR-300 was completed and fuel subassemblies were fabricated. The program was canceled, however, before fuel loading into the core, because the project judged that they cannot afford the maintenance cost any longer over a prolonged licensing procedure.

(3)CRBRP [11,12,13]

CRBRP is a loop-type fast reactor of a 380 MWe class. The basic safety approach is to provide safety design measures based on a defense-in-depth concept. A fast reactor core is usually designed with a two-zoned homogeneous core surrounded by the axial and radial blankets. One special innovation introduced in CRBRP was a heterogeneous core arrangement in which some blanket subassemblies are distributed in the active core. From the safety design point of view, this improved core design can effectively reduce the positive sodium void reactivity. The treatment of CDAs is a part of the licensing procedure but it is clearly defined that they are analyzed in a way different from DBEs. It is still required that structural and thermal margins beyond design basis should be provided to accommodate the consequences of CDAs.

The major components of CRBRP have been manufactured but the plant construction was canceled by the government decision, from the strong initiative of nuclear non-proliferation, that the U. S. withdrew from the breeder program with using plutonium.

(4) Monju [14,15]

Monju is a loop-type fast reactor in Japan of a 280 MWe class. The basic safety approach is again to provide safety design measures based on a defense-in-depth concept. The representative CDAs were analyzed as a part of the special accident category in a range of beyond-DBE, which was introduced because of limited operating experience in liquid-metal fast reactors in Japan. It was confirmed that both the mechanical and thermal consequences of CDAs could be accommodated within a reactor primary system boundary with limited influence on the containment.

The Monju plant was licensed and plant construction was completed. The operation of the plant was stopped since 1995 when the sodium leakage accident occurred in the secondary cooling system. An additional licensing procedure was finished in 2002 to improve the plant against sodium leakage events.

3. Safety Approaches in 1990s

In the next-generation LMFBR plants designed during the 1990s, a more advanced approach was taken for improved safety especially to cope with CDAs though it was regarded as a part of beyond-DBE category. Introduction of passive safety features is the common trend of this generation's design. On the other hand, consequences of CDAs are taken into account in some ways.

These advanced and innovative design concepts and related R&D achievements, especially those of DFBR are sound basis for LMFBR design study on-going F/S.

(1) European Fast Reactor [16,17]

An European Fast Reactor (EFR) program was a multi-national common project, for which France, Germany and United Kingdom were the members. The EFR was a large pool-type reactor of a 1500 MWe class with a MOX-fueled core. The basic EFR safety approach called "risk minimization" consists of: good balance between DBE and beyond-DBE behaviors, passive safety features for reactor shutdown and natural-convection decay heat removal, and damage limiting measures for residual risk. The primary objective of the risk minimization approach is to improve the accident prevention. Thus a set of passive shutdown measures are added as far as feasible to form a third shutdown level in addition to conventional two independent shutdown systems.

Even though the likelihood of CDAs can be well minimized, it was considered in this approach that they cannot be ruled out, because of uncertainties and limitation in human perception. The damage limiting measures are therefore considered with providing a containment system with a reasonable margin.

(2) ALMR [18,19,20]

The U. S. Advanced Liquid Metal Reactor (ALMR) is a small modular reactor with a metal-fueled core. One power block consisting of three reactor modules, has a net electric output of 465 MWe. The reactor concept is the power reactor innovative small module (PRISM). Metal fuel is operated at relatively low power and a large thermal expansion introduces negative reactivity feedback. Other passive safety features are provided by core radial expansion, axial expansions of control-rod drive lines and reactor vessel. The gas-expansion modules (GEMs) are added to mitigate unprotected loss-of-flow CDAs. In addition, the active shutdown system has a diverse system called the ultimate shutdown system, in which B_4C absorber spheres are dropped manually into the central channel of the core. These active and passive features make the reactor shutdown capability of PRISM very reliable.

Residual heat removal in the PRISM design is accomplished through several means including the reactor-vessel auxiliary cooling system, a reliable direct and natural-circulation cooling of the reactor vessel. Despite that the safety design approach has provided many active and passive preventive measures, representative CDAs are still evaluated to demonstrate the occurrence of core disruption is unlikely and to determine the margin of the containment, which is of a very compact and unique design.

(3) BN-800 [21]

The BN-800 fast reactor program in Russia is based on the previous fast reactors BN-350 and BN-600. The plant is a large pool-type commercial-size reactor of an 800 MWe class with a MOX-fueled core. Innovative passive safety features include: self-actuated shutdown system (SASS) for passive reactor shutdown and the special fuel subassembly design with an upper-sodium plenum, which upon voiding inherently introduces negative reactivity feedback due to enhanced neutron leakage.

In a licensing procedure, it is reported that CDAs were evaluated, given their occurrences, to some extent for various accident initiators as a part of beyond-DBE safety assessment.

(4) DFBR [22]

The demonstration fast breeder reactor (DFBR) program is the first commercial fast reactor program in Japan, in which Japan Atomic Power Co. has led. The DFBR plant is a large (600 MWe class) loop-type reactor with a MOX-fueled core. The reactor vessel, a primary pump tank and an intermediate heat exchanger tanks are connected by short inverse U-shape main coolant pipes with double walls. This eliminates the concerns of sodium leakage events. The passive reactor shutdown features that have been provided in design consist of an SASS and gas-expansion modules.

The treatment of CDAs was an extension of Monju: namely they are regarded as a part of beyond-DBE category and are evaluated on a best-estimate oriented basis to confirm a safety margin of the plant. The DFBR program has not evolved to a stage of actual plant construction, because the plant construction cost was considered to be expensive. It is for this reason that the development of fast reactors in Japan has been re-oriented toward the Feasibility Study for commercialized fast reactor and related fuel-cycle technology.

III. Safety design requirement for future LMFBRs[23]

1. Safety Targets and principle of F/S

Within the framework of F/S activity, safety targets and safety design requirement were investigated for future fast breeder reactor cycle systems. In utilizing the nuclear energy, it is necessary to recognize deeply the potential hazard of the nuclear energy, and to assure the sufficient level of safety in each stage of design, construction, operation and decommissioning of the nuclear facilities. The safety targets are determined as follows:

- To assure comparative or superior safety level to that of LWRs in the same generation as the advance reactors.
- To assure that the risk from the advanced reactors is smaller enough than the risk that already exists in the society without taking into account off-site emergency responses.

It can be said that these targets are basically consistent with the safety related user requirements both of the Generation IV project[2)] and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) / IAEA.[24)]

The defense-in-depth concept should be respected as the safety design principle of the advanced reactors while the risk-oriented approach would have certain roll in some considerations on the proportion or balance of different levels of the defense-in-depth. We believe the defense-in-depth is a proven way to achieve a high level of safety in advanced or innovative nuclear systems. Taking along with the concept, it is required to establish highly reliable system, which hardly causes of abnormal conditions. Then it is required the countermeasures both for accident prevention and mitigation. In the advanced design the priority would be put on prevention rather mitigation.

The target value of large off-site releases frequency is currently set as less than 1E-7/site year, which includes the nuclear fuel cycle facilities. This is set as a corresponding reference value to the above safety targets, which can be broken down into the requirements that the Core Damage Frequency (CDF) is less than 1E-6/ry and unreliability of containment capability is sufficiently small under representative CDAs. These target values is used within the framework of F/S.

Although practically it is not adequate to perform some detailed PSA at the early stage of conceptual design in order to check the target value satisfied, more simplified preliminary PSA, which is supported by some experiences of reactor applications, is quite useful for attaining well-balanced safety design or for elimination of any weak point or “cliff edge” effect that could occur during a severe accident as well as to see the feasibility for the target value in a global sense. The preliminary probabilistic assessment would be beneficial for systematically comprehending the safety (risk) characteristics of a plant with respect to a risk potential. Design improvement can be effectively made in such a way of appropriately controlling and minimizing the risk.

In designing an entire advanced reactor cycle system, a deep attention should be paid to physical and chemical characteristics (e.g., chemical activity, radiological toxicity, etc.) of material used in the system. Especially in the reactor systems, the accident should be naturally terminated inside the nuclear reactor plant with keeping the safety for the public by adopting the passive safety measures and by eliminating the re-criticality concern under CDAs. In this way, we are aiming at developing the FBR cycle systems considering the social acceptability.

In the conventional safety assessments as discussed in the previous sections, it has been shown by considerable amount of evaluations that the consequences of CDAs could be contained within the reactor plant even if the mechanical load by re-criticality event was assumed. However, it is not reasonable for the future plant to perform the similar evaluation from the viewpoint of public acceptance as well as avoiding excessive margin for structural design of reactor and containment structures. Therefore, provision of design measures for suppression of the mechanical energy release due to re-criticality under degraded core conditions is recognized as a crucial point.

By these design approach, we believe total risk can be minimized and thus the safety target can be achieved.

2. Safety Design Requirements of F/S

(1) General Requirement of Safety Design

Each concept of nuclear facility should be designed considering characteristic features of coolant, fuel and plant system, in addition to referring to the existing standards and guidelines used for current light water reactors, for safety assessment of the prototype FBR ‘Monju’, and so on.

(2) Requirements for Important Safety Functions

Reactor Shut Down:

- Enhancing the diversity of prevention and/or mitigation measure, utilization of passive safety features is encouraged.
- Operators action could be taken into account after a sufficient time length.

Heat Removal:

- Considering redundancy or diversity, and to achieve core cooling even if a failure of active measure is assumed.
- Failed systems are to be recovered easily by accident management.

Containment Capability:

- In order to reduce the risk reasonably, mitigation features against CDAs should be taken into account.
- The measure(s) should minimize and localize the accident consequences and achieve satisfactorily small unreliability of containment capability.
- The measure to satisfy post accident material relocation, heat removal and confinement of radioactive materials considering the event sequence of the selected CDAs.
- In addition, in consideration of the characteristics of a fast reactor core, the measure is required to prevent the significant mechanical energy release by re-criticality phenomena (eliminate the re-criticality issue).

IV. Safety design study in F/S

Among the designs for LMFBRs, the advanced loop type sodium cooled fast reactor is one of the promising candidates as a future commercialized fast reactor, which was contributed to discussion on the technology roadmap for Generation IV nuclear energy systems as the one of major option of the Sodium-Cooled Fast Reactor.[25,26,27,28,29](**Fig.1**) The power range is from 500 to 1500MWe and MOX fuel is selected as the reference core design while metal fuel core is also studied as an alternative one. The safety design of the plant is now in progress taking account of not only the system characteristics of the advanced loop concepts but also design studies and R&D experiences so far. The experiences and achievements of safety concerns as reviewed in the previous section are fully utilized in the design study.

Several new technologies have adopted to achieve the economical target, i.e., introduction of new system and materials such as three-dimensional (3D) seismic isolation systems, 12Cr-steel piping, and oxide dispersion strengthened ferrite steel (ODS) cladding, novel plant configuration such as two-loop system with compact reactor vessel, new sub-system such as the integrated IHX with pump.

An emphasis is placed on that a recriticality-free core concept with a special fuel assembly is pursued by performing both analytical and experimental efforts in order to realize the rational design for In-Vessel Retention (IVR). IVR is crucial point for LMFBRs, in which superior thermal characteristics of coolant can be utilized, to achieve the safety target with reasonable design of containment system.

(1) Prevention of abnormal occurrences

The system design with new material, subsystems and structures should have rational design margin difficult to be failed or not to result in the incidents. For this purpose, the new material, subsystems and structures will be carefully designed and fabricated based on the characteristics understanding and mock-up experiments.

As the basis of rational system design, the seismic isolation technology is one of the crucial issues, which would reduce plant materials and standardize the system design especially in Japan. The two-dimensional seismic isolation technology has been already developed and the related design guideline for fast reactor system has been published in Japan. However, the 3D seismic isolation technology should be developed in order to improve rationality, e.g., much smaller relative displacement among fuel assemblies, control rods, and the core support plate against vertical response acceleration. Also buckling of reactor vessel would be more easily avoided. The 3D seismic isolation system is now under investigation by producing several concepts.

The 12Cr steel, which has low expansion and has higher creep strength than conventional stainless steel, will be adopted to realize simplified plant configuration that makes it possible also to reduce the reactor building volume. The 12Cr steel has been utilized in fossil power plants, but material strength data such as creep fatigue strength has to be obtained considering severe thermal transient conditions, which are anticipated in the system with high thermal conductivity of sodium and high temperatures. Based on these experimental data and analytical methods, the structural design standard and the related criteria should be established.

The two loop system and compact reactor vessel design has raised several R&D needs to ensure the plant integrity related to thermal hydraulics, such as thermal stress, cover-gas entrainment, and flow induced vibration issues which would be facilitated by higher flow velocity of coolant in main piping. The integrated components with IHX and mechanical pump may enhance the fretting wear of IHX tubes. The evaluation methods for the fretting wear in 12Cr steel and related experimental data are necessary to be investigated to ensure its integrity.

The preventive maintenance efforts will be emphasized

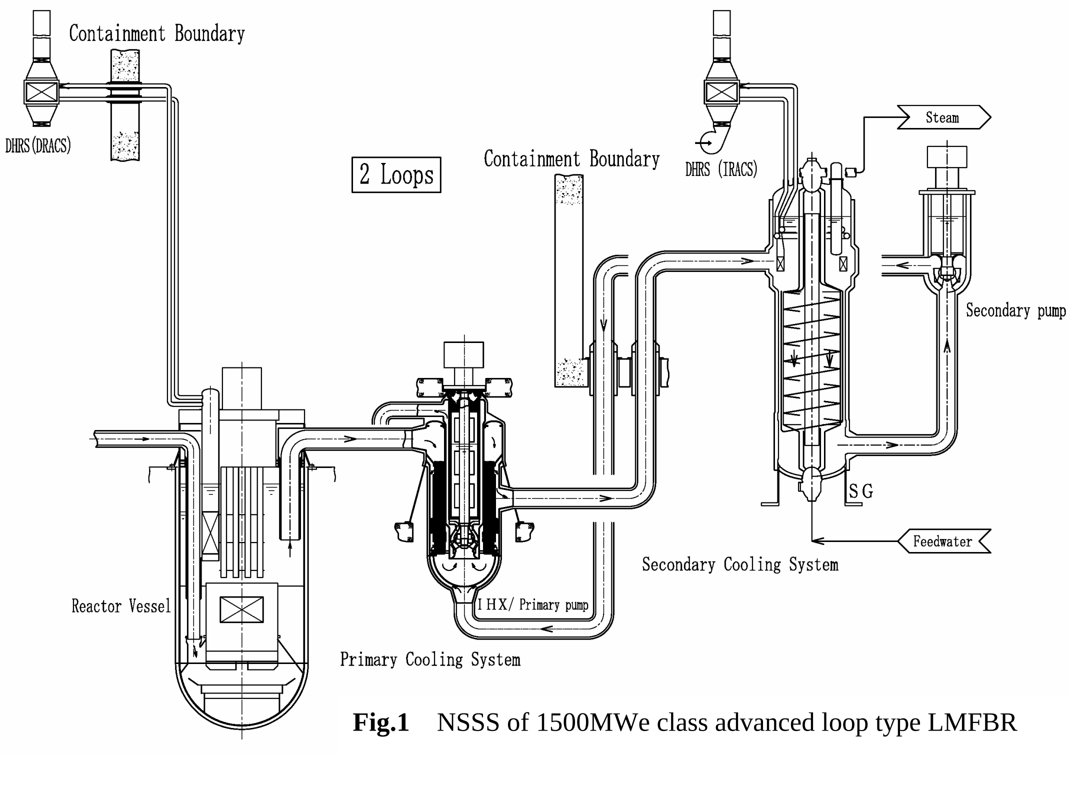

Fig.1 NSSS of 1500MWe class advanced loop type LMFBR

to ensure high plant availability. The improvement of the ISI&R technologies are crucial to confirm the integrity of in-sodium safety related structures and boundaries and to repair in-place quickly. For this improvement, the system and component design should be carried out taking account of the development targets for the following three elements, i.e., high quality sensors under 200deg.C sodium (sensor technology), accurate remote handling system like manipulators movable in narrow spaces (robotics), and high resolution and quick image processing system (image processing).

(2) Protection of incidents or transients into accidents

The early detection capability of safety protection system is significantly important to minimize unfavorable influences on plant systems, such as temperature increases in fuel, cladding and structures. From the safety point of view, the reliability of Reactor Shutdown System (RSS) and Decay Heat Removal System (DHRS) is essential. The safety criteria for fuel pins with ODS cladding tube and structures made of 12Cr steel are necessary to be developed.

The CDF less than 1E-6/ry can be achieved by providing redundant and independent safety function with diversity. The two independent RSSs have been designed within the DFBR Project in Japan, where the diversity of crucial parts of RSS are enhanced by adopting independent multi-signals for de-latching, different rod insertion mechanisms, different de-latch mechanisms, and rod structures such as solid and articulated. The preliminary PSA result showed the unavailability of RSS is sufficiently less than 1E-6/demand, even under the assumption of common cause failures.[30] As for the decay heat removal, it is great advantage of LMFBRs as non-pressurized system that the coolant level can be statistically maintained by guard vessels without coolant injection. The natural circulation capability is crucial to achieve higher reliability for a longer mission period, in addition to redundancy of the system and forced circulation function. These design features are adopted in the advanced loop concept so that the CDF less than 1E-6/ry would be satisfied.

In the advanced loop concept, two loop system without check valves is employed and thus the primary pump stick would become the severest accident in the DBEs comparing with the conventional three or four loops design. However some design adjustments, e.g., a delay time of the primary pump trip in the intact loop and a halving time of the primary flow rate within the reasonable range, make it possible to restrict the maximum cladding temperature less than the tentative safety criteria, which was set for the DFBR fuel design with the austenitic stainless steel cladding. Thereby the safety criteria for fuel pins with ODS cladding tube would be prepared through the irradiated material tests.

Detection system for local faults would be provided both fission gas and delayed neutron detection systems. The blockage formation process had been investigated in the experiments and the coolability had been analyzed in detail during DFBR design stage. The anticipated local blockage would become porous and fuel failure would not easily occur. The effectiveness of the detection system would be carefully examined with taking account of back ground level of radioactivity, transport delay time and stability of fault sight. Safety experiments to evaluate the gas blanketing effect and the behavior of fuel failure propagation for high burn up fuel pins would be necessary.

(3) Mitigation of accident consequences

The consequences of DBEs would lead to no radioactive release due to neither fuel pin failures nor penetrations from primary coolant to the containment. The pipe break of the cover gas piping would result in also no release of radioactive materials, since the double boundary system would be adopted in order to avoid sodium fire in the containment that is caused by accompanied sodium with gas leaking. There are small possibilities in the accidents during fuel handling system such as drop of fuel assembly, and accidents in the radioactive material storage tanks of clean up system. The confinement function would be adopted to mitigate the consequences in fuel handling system and similar design measure would be adopted in the clean up system. Nevertheless the containment would be provided as the final barrier to the environments. The closed reinforced concrete structure surrounding the reactor vessel is designed as the containment to ensure leak tightness with inner lining. Confinement area is provided outside of the penetrations of the containment, in which emergency gas treatment system is equipped.

(4) Prevention and mitigation of severe accidents

The probability of causing the severe core damage would be less than 1E-6/ry, but both the passive safety features and mitigation measures against CDAs would be provided to enhance safety. As the selection of these features, the great importance has been put on simple and reliable mechanisms together with demonstration capability.

(a) Passive Safety Features

A temperature sensing de-latch mechanism called Self Actuated Shutdown System (SASS) is adopted as the most promising passive shut down system developed from DFBR design study.[31] The Curie point type is used as the device, which is a pair of electromagnet and armature with temperature sensing alloy.(**Fig.2**) The magnetic flux is steeply decrease when the temperature of the sensing alloy reaches at the Curie point, then the absorber rod is passively de-latched and inserted by gravity. The Curie point of temperature sensing alloy can be changed by adjusting its composition so as to apply various reactor core concept not only MOX fuel but also Metallic fuel cores.

A restraint core concept with core barrel structure has been adopted in Japan. The cause of rod stuck has been studied by analyzing the contact modes between absorber rods and guide tubes through experiments. As a result, an articulated structure has been adopted in the upper most position of absorber rods to enhance insertion capability. The absorber rods are always inserted inside the guide tubes and locate just above the active core during operation requiring only 1m stroke for scram and it is designed to assure the clearance for insertion under conservative

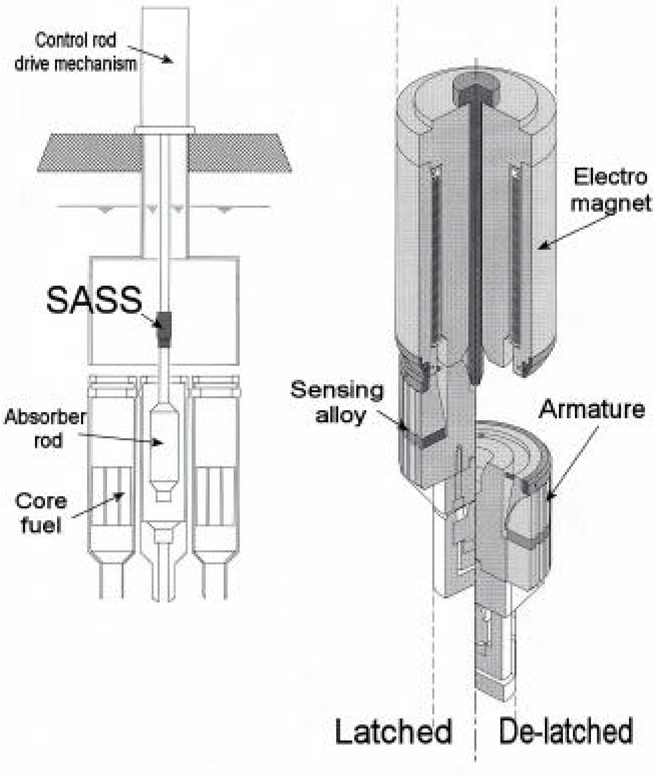

Fig.2 Self Actuated Shutdown System (SASS)

horizontal displacement between core head and the control rod drive lines taking a severe earthquake condition into account. By these design features, the absorber rod stuck is extremely unlikely and would not the cause of scram failure. Therefore possible cause for scram failure has been identified as multiple failures of safety protection systems and of de-latch mechanism. Sodium coolant, on the other hand, has superior characteristics as regard to thermal conductivity and heat transportation with sufficient sub-cool margin to boiling. Indeed there is ca 350 degree C. of sub-cool margin at the maximum cladding temperature point in the best estimate sense. And the absorber rod has a large negative reactivity worth and its motion is one-dimensional within a second by gravity. The increase of coolant temperature can be expected all types of ATWS, i.e. unprotected loss of flow (ULOF), unprotected transient over power (UTOP) and unprotected loss of heat sink (ULOHS). These are the reason why SASS is selected.

A comprehensive development work has been carried out both out-of-pile and in-pile experiments concerning the thermal aging, thermal fatigue and creep, thermal transients, and irradiated influences. The effectiveness of SASS has been confirmed by the transient tests using full mock-up SASS in an out-of-pile facility. The verification of integrity against thermal transients and of preventing spurious de-latch remain up to now. The thermal transient test is now under continuation and design effectiveness against spurious de-latch is scheduled to be verified through in-pile operations in the experimental fast reactor JOYO.

In the advanced loop concept, the natural circulation capability would be enhanced by reducing the pressure drop of core and adopting Intermediate Reactor Auxiliary Cooling System (IRACS) or Primary Reactor Auxiliary Cooling System (PRACS), that would increase the flow rate under natural circulation. And thus the passive cooling even under passive shutdown condition would be attained. The remaining point is natural circulation capability of Direct Reactor Auxiliary Cooling System (DRACS) that will be used under loop maintenance conditions. Design and analytical efforts accumulated in DFBR design study [32] will be reflected into future study and there seems less experimental needs.

Any other initiators entering CDAs such as Protected Loss of Heat Sink (PLOHS) would be also analyzed and provide redundant accident management procedures, since the double boundary system will not result in the loss of coolant and thus a large grace period would be ensured.

(b) Mitigation of CDAs

There provided passive safety features against fast sequences like ATWS and redundant accident managements against slow sequences, and then the probability leading to CDAs becomes negligibly small. Nevertheless the consequences of CDAs would be mitigated, since re-criticality potential in the course of CDAs has been regarded as one of the major safety issues in fast reactor core. Enormous efforts have been dedicated to the clarification of the accident scenario and the consequences of CDAs. Especially ULOF and UTOP scenarios have been historically investigated from the view of mechanical design margin against the super prompt excursion during the initiating phase (I/P) and against energetic re-criticality during the transition phase (T/P). The thermal design margin has been also investigated to ensure IVR[33].

The new safety design requirements has been set in F/S aiming at eliminating the energetic re-criticalities under CDA sequences by limiting the positive sodium void worth during I/P and enhancing the molten fuel discharge to prevent molten pool formation during T/P. The long-term coolability should be secured for the relocated fuel debris inside the reactor vessel by providing several design measures.

In the course of ULOF accident, power excursion due to coherent sodium boiling has been of major concern, because large sodium cooled core tends to have larger positive sodium void worth. In order to avoid severe energetics during I/P, the sodium void worth should be limited under certain value. In our study the reference value has been set less than six dollars (6$) for MOX core and 8$ for Metallic core based on both a theoretical consideration and experiences of analysis for various type of core design. The data-base for MOX fuel has been greatly accumulated through in-pile experiments in TREAT and CABRI and analyses for various types of core design have been carried out.[34] Those in-pile experiments give clear evidences for boiling propagation velocity and fuel dispersion velocity as the function of power level under ULOF conditions. By applying these data to reactor conditions, the void worth limitation has been obtained. SAS4A code is now being applied for some candidate core designs to clarify their feasibility in this matter.

As for the fuel discharge in T/P, special fuel assemblies have been proposed to enhance the molten fuel discharge as shown in **Fig.3**. The Fuel Assembly with Inner DUct Structure (FAIDUS) is a concept that a steel duct is installed as fuel escape path in every fuel assembly, while the Axial BLanket Elimination (ABLE) uses some portion of pin bundle structure of which axial blanket is removed. In both concepts every fuel assembly has its own fuel escape path so that sub-critical state due to sufficient fuel discharge from active core region can be achieved in the early stage of T/P. Since severe re-criticality events might occur by dynamic motion of whole core molten fuel pool, sufficient fuel discharge in the early stage is important to avoid severe re-criticality. In principle fuel discharge behavior can be confirmed by experiments because it is expected that the material motion is localized within a fuel assembly or vicinity of the fuel escape path.

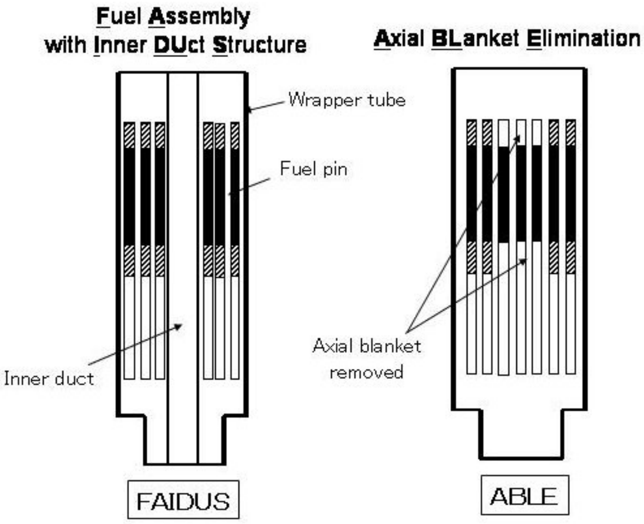

Fig.3 Fuel assembly design for enhancement of fuel discharge

Due to its larger diameter of fuel escape path, material motion in FAIDUS might have less uncertainty than ABLE. The experimental program named EAGLE is now in progress as the collaboration work between Japan and the Republic of Kazakhstan in order to clarify the fuel discharge capability of FAIDUS.[35)]

The FAIDUS assembly remains several R&Ds for fabrication, irradiation integrity and reprocessing. On the other hand, ABLE is expected to improve core performance than that of FAIDUS and less R&Ds are required for establishing related fuel technologies. However the fuel discharge process of ABLE contains more uncertainty including possibility of solid or low mobility fuel blockage. To ensure the elimination of re-criticality, analytical efforts are now undergoing. After the detailed understanding for possible fuel discharge process in ABLE, the experimental requirements would be clarified.

The design of the lower core support structure is crucial to cope with the post accident material relocation and heat removal for the fuel debris retention. Because the advanced loop concept has relatively small reactor vessel with large fuel inventory, multi-layer debris tray is required to hold full inventory of the fuel without dryout and re-criticality. Proper quenching and distribution as well as coolant convection are crucial points of this concept. Feasibility of this multi-layer concept should be confirmed by some experimental data.(**Fig.4**)

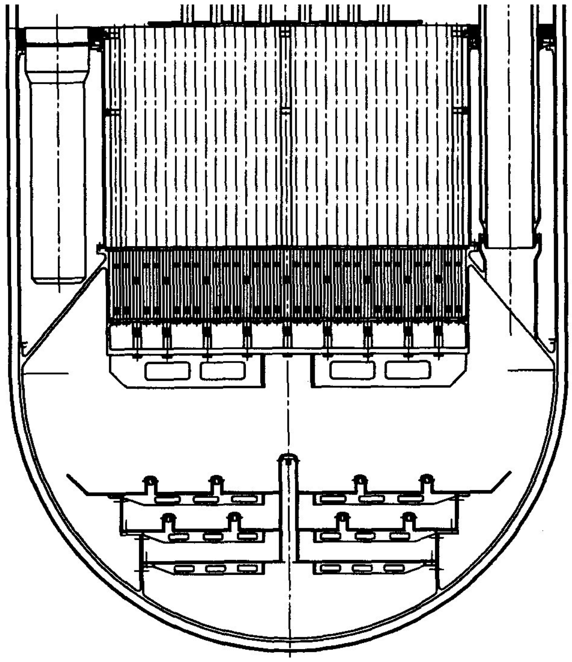

Fig.4 Core support structure and debris tray

With respect to Metallic fuel, further in-pile experiments, especially for ULOF conditions, would be required and analytical code would be modified for reactor analyses. The fuel discharge process would be also verified in the experiments.

As results of the above R&Ds, it is expected not only to realize the rational design against CDA consequences but also to resolve the major safety issue of FR core and thus to promote relief of the anxiety in the public.

(5) Specific issues for sodium coolant

The sodium related issues such as sodium leak and sodium-water reactions anticipated in the steam generators (SG) would be minimized, although even the conservative consequences will not connect to the reactor core safety.

(a) Sodium Leak

Lessons learned from a sodium leakage in Monju suggested us that the design should be more sophisticated against anticipated incidents, because the public fears its chemical activity appeared in the nuclear plants and the utilities has also anxiety that early restarting may be difficult after once sodium leak or sodium water reactions would occur. To cope with this requirement, the guard pipes are provided both primary and secondary cooling system and those annular regions would be filled with inert gas. All the penetrations for the guard pipes would be covered to ensure double boundary system. The shortening of piping adopted in the advanced loop design contributes to the reduction of plant material amount regardless of the double boundary concept.

The development of the Leak Before Break (LBB) concept for 12Cr steel is required in order to exclude the pipe break possibility and the resultant large scale sodium fire. The sodium leak detection would be easily accommodated in the annular region and thus the detectability for sodium leak would be enhanced to ensure the LBB concept. The rational design for guard pipe is ensured by the LBB concept, since the guard pipe would be conservatively designed and fabricated against the double-ended break without the LBB concept. Nevertheless the guard pipe would have a proper design margin against the consequences of double-ended break in order to eliminate the possibilities both for the abrupt coolant flow decrease in reactor core and the sodium fire potential. This design margin would be rationally taken into account in the best estimate manner, because such a break would be addressed as beyond DBEs.

(b) Sodium water reactions in SG

Currently a conventional single tube SG is addressed as reference design. The point is to enhance the reliability of early detection system for water leak. The earlier detection system especially against small leaks, which is sufficiently enveloped by DBEs, would be developed and adopted to prevent the propagation of tube ruptures and to make early restarting of plant operation possible. This is desirable to minimize loss of plant load factor. For safety concerns, the wastage data and high temperature creep strength for 12Cr steel is necessary to be obtained. A comprehensive evaluation method for tube rupture propagation behaviors would be developed with taking account of superimpose of wastage and overheating rupture modes, where the hydrodynamics of sodium and water-steam and the local coolability of water in tubes would be precisely evaluated. For this purpose the experimental efforts would be required.

Double-wall-tube SGs with secondary sodium system and a new type SGs with lead bismuth as the intermediate heat transfer medium are investigated as alternative designs. The most promising SG concept will be selected from the views of technical feasibility, construction cost and anxiety for the consequences of sodium water reaction. These efforts for more reliable and robust SG design can be significant to conquer the drawback of sodium cooled FR.

V. Conclusion

A basic safety approach in designing LMFBRs has been essentially the same as one taken in LWRs. Namely a reactor system shall be designed based on a defense-in-depth concept, with a primary emphasis on preventing and detecting abnormal occurrences. Then safety design measures shall provide appropriate means to shut the reactor down, cool the residual heat and contain radioactive materials within the reactor facility, by taking the safety characteristics of sodium coolant into account. .

The subject of severe accidents or CDAs has been and will be an important safety consideration for LMFBRs as well. CDAs were treated in the LMFBRs in the past in some ways since the fast reactor generally is not designed in a most reactive configuration. In 1970s and 1980s they are commonly regarded as an event category of beyond design basis event. The purpose of CDA analysis is therefore to provide or confirm an additional safety margin of the plant strictly designed for DBEs. In the next-generation LMFBR plants designed during the 1990s, a more advanced approach was taken for improved safety especially to cope with CDAs though it was regarded as a part of beyond-DBE category. Introduction of passive safety features is the common trend of this generation’s design. On the other hand, consequences of CDAs are taken into account in some ways.

Within the framework of F/S activity, safety target and safety design requirement were investigated for future fast breeder reactor cycle systems. The safety targets are (1) equal or higher level of safety compared with LWRs in the same generation (2) sufficient smaller risk than existing one without off-site emergency responses. In order to realize this target, safety design should be based on the defense-in-depth concept with the aid of PSA. Some design requirements are formulated to include: passive shutdown measures, reliable heat removal features such as with natural circulation, elimination of recriticality concern as well as post-accident long-term debris cooling capability aiming at in-vessel retention for postulated CDAs.

Among the designs for LMFBRs, the advanced loop type sodium cooled fast reactor is one of the promising candidates as a future commercialized fast reactor. The safety design of the plant is now in progress taking account of not only the system characteristics of the advanced loop concepts but also design studies and R&D experiences so far. Emphasis is placed on that a recriticality-free core concept with a special fuel assembly is pursued by performing both analytical and experimental efforts in order to realize the rational design for IVR.

Acknowledgment

This paper includes the outcome of collaborative study between JNC and JAPC(that is the representative of 9 electric utilities, Electric Power Development Company and JAPC) in the accordance with “the agreement about the development of a commercialized fast breeder reactor cycle system”.

References

1) H. Noda, “Current Status of FR Cycle System in Japan “, *Proc. of 8th Int. Conf. on Nucl. Eng.(ICONE 8)*, April 2-6, Baltimore, MD USA (2000).
2) U.S. DOE and the Generation IV International Forum, “A Technology Roadmap for Generation IV Nuclear Energy Systems”, December (2002)
3) Sa. Kondo, H. Niwa, “Safety issues and approach for liquid-metal reactors”, *OECD/NEA/CSNI Workshop on Advanced Nuclear Rector Safety Issues and Research Needs, OECD*, Paris, France, 18-20 February 2002

4) International Atomic Energy Agency, “The Safety of Nuclear Installations,” *IAEA Safety Series No. 110* (1993).
5) International Atomic Energy Agency, “Safety of Nuclear Power Plants: Design,” *IAEA Safety Standards Series No. NS-R-1* (2000).
6) P. Tanguy, “A French View on LMFBR’s Safety Aspect,” *Proc. Int. Mtg. On Fast Reactor Safety Technology*, Seattle, Washington, U. S. A., August, 1979.
7) J. Petit and J. Leduc, “Designers Safety Aspects for LMFBR’s in France,” *Proc. Int. Mtg. On Fast Reactor Safety Technology, Seattle*, Washington, U. S. A., August, 1979.
8) H. H. Hennies, R. Huper and G. Kessler, “The History of Fast Reactor Safety ^ Federal Republic of Germany,” *Proc. 1990 Fast Reactor Safety Meeting*, Snowbird, Utah, U. S. A., August 1990.
9) E. Kugler and S. Wiesmer, “Licensing Aspects in the Verification of the SNR 300 Design Concept against Hypothetical Accidents,” *Proc. Int. Mtg. On Fast Reactor Safety and Related Physics*, Chicago, Illinois, U. S. A., October 1996.
10) K. Traube, “Safety Design of SNR-300,” *Proc. Int. Mtg. On Fast Reactor Safety and Related Physics*, Chicago, Illinois, U. S. A., October 1996.
11) R. J. Slember, “Safety-Related Design Considerations for the Clinch River Breeder Reactor Plant,” *Proc. Int. Mtg. On Fast Reactor Safety and Related Physics*, Chicago, Illinois, U. S. A., October 1996.
12) L. E. Strawbridge, “CRBRP Structural and Thermal Margins beyond Design Basis,” *Proc. Int. Mtg. On Fast Reactor Safety Technology*, Seattle, Washington, U. S. A., August, 1979.
13) T. G. Theofanous and C. R. Bell, “Assessment of Clinch River Breeder Reactor Core Disruptive Accident Energetics,” *U. S. Nuclear Regulatory Commission NUREG/CR-3224* (1983).
14) Y. Fukuzawa, et al., “Safety Regulation and Licensing Experience of Liquid Metal-Cooled Fast Breeder Reactor in Japan,” *Proc. 1990 Fast Reactor Safety Meeting*, Snowbird, Utah, U. S. A., August 1990.
15) A. Hayashi, et al,, “Impact of Safety and Licensing Considerations on Monju,” *Proc. Int. Top. Mtg, on Fast Reactor Safety*, Knoxville, Tennessee, U. S. A., April 1985.
16) K. Ebbinghaus, et al., “The EFR Safety Approach,” *Proc. Int. Conf. On Design and Safety of Advanced Nuclear Power Plants (ANP’92)*, Tokyo, Japan, October 1992.
17) G. Heusener, G. Hubert and C. H. Mitchell, “EFR Safety Concept and Its Evaluation,” *Proc. Intl. Mtg. on Advanced Reactor Safety (ARS'94)*, Pittsburgh, Pennsylvania, U. S. A., May 1994.
18) J. E. Quinn, et al., “U. S. ALMR, A Multi-Mission Advanced Reactor Concept for the Next Century,” *Proc. Int. Conf. On Design and Safety of Advanced Nuclear Power Plants (ANP’92)*, Tokyo, Japan, October 1992.
19) W. Kwant, et al., “U.S. ALMR Sodium-Cooled Reactor Design and Performance,” *Proc. Int. Conf. On Design and Safety of Advanced Nuclear Power Plants (ANP’92)*, Tokyo, Japan, October 1992.
20) U. S. Nuclear Regulatory Commission, “Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,” *NUREG-1368* (February 1994).
21) Y.E. Bagdassarov, “Safety of Fast Neutron Reactor Power Units under Operation and Design in Russia,” *Nuclear Engineering and Design*, vol.173, p. 239 (1997).
22) M. Miura, et al., “Present Status of DFBR Design in Japan,” *Proc. Int. Conf. On Design and Safety of Advanced Nuclear Power Plants (ANP’92)*, Tokyo, Japan, October 1992.
23) Y. Kani, “Safety Approach for the Japanese Advanced Reactor Program,” *ANS President’s Special Session: Safety Considerations for Advanced Reactors*, Milwaukee, Wisconsin, U. S. A., June 2001.
24) B. Kuczera, P. Juhn and K. Fukuda, “Development Trends in Nuclear Technology and Related Safety Aspects”, *Proc. of 10th Int. Conf. on Nucl. Eng.(ICONE 10)*, April 14-18,Arlington, VA USA (2002).
25) Y. Shimakawa and S. Kasai, et al, "An Innovative Concept of Sodium-Cooled Reactor Pursuing High Economic Competitiveness", *Nuclear Technology,* **140**[1] , 1 (2002)
26) S. Kotake, et al., “The R&D issues necessary to achieve the safety design of Commercialized Liquid Metal cooled Fast Reactors”, *OECD/NEA/CSNI Workshop on Advanced Nuclear Rector Safety Issues and Research Needs, OECD*, Paris, France, 18-20 February 2002
27) T. Mizuno, et al., “LMFBR design and its evolution (1) Fuel design of LMFBR”, *Proc. Int. Conf. on Advanced Nuclear Power Plants and Global Environment, GENES4/ANP2003*, Kyoto, Japan, September, 2003.
28) N. Uto, et al., “LMFBR design and its evolution (2) Core design of LMFBR”, *Proc. Int. Conf. on Advanced Nuclear Power Plants and Global Environment, GENES4/ANP2003*, Kyoto, Japan, September, 2003.
29) M. Hishida et al., “LMFBR design and its evolution (4) An Innovative Concept of Sodium-Cooled Middle-Scale Modular Reactor Pursuing High Economic Competitiveness”, *Proc. Int. Conf. on Advanced Nuclear Power Plants and Global Environment, GENES4/ANP2003*, Kyoto, Japan, September, 2003.
30) K. Satoh, S. Kotake et al, "Application of the PSA method to a large scale FBR Design", *Proc. Int. Conf. of Design and Safety of Advanced Nuclear Power Plants (ANP’92)*, Vol.4 pp 39-1.1 - 39-1.7, (1992)
31) M. Morihata, et al., “Development of Self Actuated Shutdown System for FBR in Japan”, *Proc. 5th Int. Conf. on Nucl. Eng. (ICONE5)*, Nice, France, May 1996.
32) O. Watanabe and S. Kotake et al, "Study on Natural Circulation Evaluation Method for a Large FBR", *Proc.*

8th Int. Top. Meeting on Nuclear Reactor Thermal-Hydraulics, Vol.2,, pp829-838, (1997)
33) H. Endo and M. Ishida et al, "A. Study of the Initiating Phase Scenario of Unprotected Loss-of-Flow in a 600MWe MOX Homogeneous Core", *IAEA IWGFR Technical Committee Meeting on Material-Coolant Interactions and Material Movement and Relocation in LMFR*, PNC-OEC, Japan. (1994)
34) H. Niwa, "A Comprehensive Approach of Reactor Safety Research Aiming at Elimination of Recriticality in CDA for Commercialization of LMFBR", *Proc. Int. Symp.on the Global Environment and Nuclear Energy System*, Tsuruga/Japan, (1996)
35) T. Inagaki and K. Aizawa et al, "Role and Approach to the Recriticality Elimination with Utilizing the In-pile Test Reactor IGR", *2nd Int. Conf. on Non-Proliferation Problem*, Kurchatov, Republic of Kazakhstan, (1998)