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Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance (1)

- Interim Report of JNC/CEA Collaboration NWP-5(a) from 1999 to 2001: Common View and JNC's Contribution -

December, 2001

O-arai Engineering Center
Japan Nuclear Cycle Development Institute

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Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance (1)
- Interim Report of CEA/JNC Collaboration NWP-5(a) from 1999 to 2001: Common View and JNC's Contribution –

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ABSTRACT

This is an interim report describing the progress and the results of the collaborative research works between JNC and CEA on the safety logic in future fast reactors under the title of "Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance" from 1999 to 2001. This contains JNC's contribution and common view of both partners.

(1) Safety goals are proposed from JNC and CEA. Significant coherency is found such as to keep defense-in depth concept, mitigation measures against core melt are taken into account for containment design, "evacuation free" concept is pursued, quantitative safety target is also considered as well as deterministic approach, and improvement of social acceptance is considered from the development stage of the fuel cycle including nuclear power plants.

(2) Safety characteristics of each candidate coolant were compared and discussed. Gas-cooled fast reactor is a common interest area. Discussions are focused on: safety design requirements, safety evaluation events list, transient behavior analysis, core catcher designs, and so on.

(3) JNC's results include criticality map for predicting CDA behavior and consequences, and CDA analysis results of lead-cooled and gas-cooled fast reactors with SIMMER-III.

The collaboration on the action NWP-5a is recognized as being of great importance for the orientation of the innovative design studies.

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2) Innovative Systems Concepts Lab., Cadarache, CEA (presently Advanced Water Reactor Systems Lab., Cadarache)
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社会的受容性を備えたFBRを目指す合理的な安全確保論理の確立（１）
－CEA/JNC共研NWP-5(a)中間報告（1999-2001）：共通認識とJNCの成果－

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要　旨

本報はJNCとCEAとの先進技術協力に関する共同研究の一環として実施されている「社会的受容性を備えたFBRを目指す合理的な安全確保論理の確立」の中間報告書であり、1999年から2001年の間のJNCにおける成果と、両機関による討議により得た共通認識をまとめた。

(1) 両機関より安全性の目標が提案された。両者には、以下の点で整合性のあることが明らかとなった。「深層防護」の考え方を堅持すること、格納系の設計において炉心溶融緩和対策を考慮すること、「避難不要」概念を追求すること、決定論的アプローチに加えて定量的安全目標を考慮すること、原子炉プラントを含む燃料サイクルの開発段階から社会的受容性の向上を図ること。

(2) 各種冷却材の安全上の特徴を比較整理した。さらに共通の興味対象分野であるガス冷却高速炉について、安全設計要求、安全評価事象、事故時過渡挙動解析結果、コアキャッチャーの設計等の情報交換・討議を行い、共通認識の醸成を計った。

(3) JNCの成果として更に、炉心損傷時の事象推移を概略予測するための臨界形状マップ、鉛冷却炉の事故時過渡挙動解析結果等を報告した。

本分野における協力は革新的設計の研究を行う上で極めて重要であると考えられる。

1）システム技術開発部
2）フランス原子力庁（CEA）
3）要素技術開発部リスク評価研究グループ
4）システム技術開発部FBRサイクル安全設計グループ

Contents

1. Introduction

CEA and JNC agreed in 1998 to promote collaborative activities in the field of Advanced Technology R&D for Nuclear Energy. Among several New Work Packages (NWPs) defined for each field of collaboration, it was determined that "safety of future fast reactors" was treated in NWP-5. Because of the variety of issues to be treated in the area of "safety", it was agreed to separate some issues and in NWP-5(a) safety logic is considered under the title of "Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance."

The objectives and work scope of NWP-5(a) were defined as shown in Table 1, and the time schedule is shown in Fig. 1. Four meetings were held on NWP-5(a) from 1999 to 2001 (Table 2).

As shown in Table 1, the scope of this collaboration is broadly divided into two parts. The first part is discussion of the safety assurance logic including consideration on public acceptance. The second part is focused on much more technological discussion, the safety feature characterization of various reactor designs and comparison.

The contents of this report are designed so as to be consistent with the work scope as shown in Table 1. The first part of the work scope is discussed in Chapter 2, and the second part in Chapter 3. Because this is a report of collaborative work, each chapter in principle contains mutual discussions between CEA and JNC, and individual view of JNC. The parts of discussions are 2.2.2, 3.1.1, 3.2.2 and 3.3.

Table 1 Objectives and Work Scope of NWP-5(a): Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance

(1) Objectives of Collaboration

1. Establishment of a Rationalized Safety Assurance Logic Aiming at FBRs with Enhanced Social Acceptance

   Constitution of Coherent View on Fundamental Philosophy of Safety Assurance in Future Fast Reactors

2. Collaboration in Safety Research of Various Candidates for Future FR Systems

   Clarification and Comparison of Safety Characteristics of Innovative FR Systems

(2) Work Scope

1. Safety Approach aiming at Enhancement of Social Acceptance
   - 1.1 Information Exchange on present safety philosophy in Europe and in Japan
   - 1.2 Search for Joint safety approach about future innovative FRs, leading to social acceptance and in agreement with the universal safety principles.
   - 1.3 Joint recommendation, safety goals and criteria for future innovative FRs design

2. Respective Safety Aspects of various FRs Systems and Comparison
   - 2.1 Information Exchange on safety characteristics of FRs with various coolants, fuels and plant designs.
   - 2.2 Joint preliminary safety analysis of some FRs concepts and comparison as regards social acceptance.
   - 2.3 Identification of studies and R&D needs for in depth analysis and for validation of the most promising concepts.

Table 2 List of the Meetings on NWP-5(a)

Dec. 6, 1999	at Cadarache
May 16-18, 2000	Joint meeting with NWP-2 at O-arai
Dec. 11, 2000	at Cadarache
June 25-27, 2001	Joint meeting with NWP-2 at Cadarache

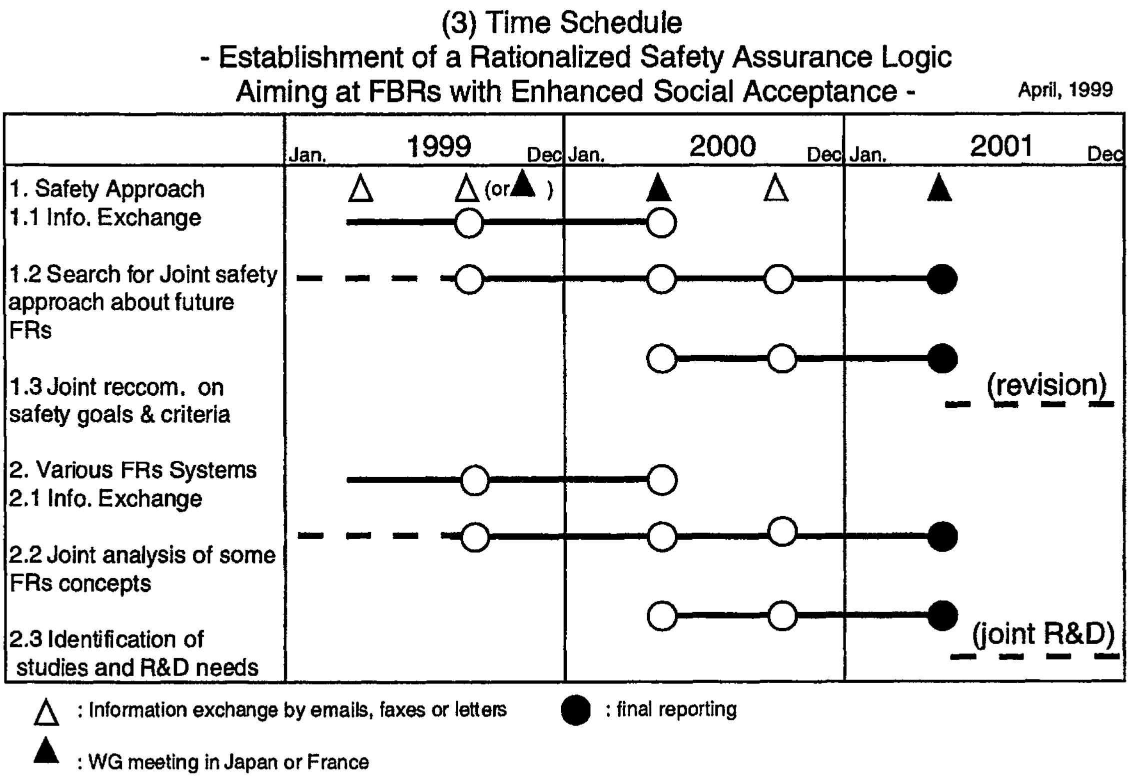

Figure 1 Time Schedule of NWP-5(a)

2. Safety Approach

2.1 Current safety approach in Japan

In this section, safety approach of Monju is summarised briefly, and some discussions of safety approach on DFBR (Demonstration FBR in Japan) are introduced.

2.1.1 Safety approach of Monju

The framework of safety assurance philosophy for Monju, a prototype FBR in Japan, is very similar to that for current LWRs except for the treatment of beyond design basis events (BDBEs). Safety design guideline was established based on the "Defense-in-Depth" philosophy, and "Safety Evaluation Policy for Monju" was approved and issued by Nuclear Safety Commission (NSC) in November 1980. The licensing experience of Monju is described in details in Ref. 1. The procedure of Monju Safety Regulations is summarized as follows:

0. To assess adequacy of LMFBR plant design
1. Selection of representative events
   - Abnormal transients during reactor operation
   - Accidents
2. Examples of candidates
3. Criteria for safety evaluation
4. Items specific to LMFBRs
5. BDBEs

   "Since operational experience of LMFBRs is limited, safety evaluation should also be made for postulated events with less likelihood and larger consequences than the accidents. Correlation should be made of the initiating events and preventive measures against accident progression to ascertain that release of radioactive materials to the environment is limited to a reasonably low level." (quoted from Safety Evaluation Policy for Monju)
6. Major accident & hypothetical accident to determine site suitability according to the guidelines for reactor siting.

   (1) Major accident: foreseeable from technical viewpoint (assume maximum release of radioactive materials among the above events)

(2) Hypothetical accident: inconceivable from technical viewpoint (postulate larger release)

Monju BDBE analysis and result are briefly summarized as follows:

1. Licensing activity in 1980 – 1982
2. following FFTF, CRBR (NUREG-0122), SNR-300 experiences,
3. ULOF: SAS3D, SAVE, VENUS-II/III, (SIMMER-II,) PISCES, PLUG, sodium fire code, PAHR code (NC network model)
4. 380 MJ (isentropic expansion of fuel to 1 atm) by SAS3D-VENUS calculation (conservative case)
5. 500 MJ for structural response analysis (380MJ + work energy caused by FCI)
6. sodium spillage: 400 kg to the containment volume (60,000 m3)
7. Xe, Kr = 1 %, I = 1%, Pu = 0.1% in the containment
8. Design leak rate of the containment (1%/day, tested), and
9. Long term coolability was confirmed.

Siting Evaluation Analysis (hypothetical accident):

(1) Xe, Kr = 100 %, I = 10%, Pu = 1% of inventory are assumed to be released in the containment, and

(2) The results are less than the exposure limit (0.25Sv for whole body and 3Sv for thyroid gland of an individual at the site boundary during 30 days).

PSA (probabilistic safety assessment) is not requested explicitly in the licensing of nuclear power plants (NPPs) in Japan because the licensing examination is performed at the stage of basic design (not detailed design). However, for LWRs, since 1992, the report to the licensing authority and NSC on the planning of accident management based on PSA is requested until startup of operation. This request is applied for newer reactors.

A full scope PSA was conducted for Monju. The results of the level-2 PSA were reported in Ref. 2. The level-1 PSA shows that the CDF (core damage frequency) of Monju is sufficiently below 10^{-5}/ry which was suggested by IAEA-INSAG group for newly installed reactors.

Monju ULOF case was re-evaluated in 1997 (Ref. 3) using latest knowledge with SAS4A and SIMMER-III. It was concluded that the evaluated work energy in an ULOF event is reduced to 110MJ (compared with 380MJ in the licensing) based on the latest analysis tools.

2.1.2 Discussions on Safety Evaluation Policy for J-DFBR

Discussions to formulate a framework of safety evaluation policy for DFBR, which was sponsored by STA (presently MEXT: Ministry of education, culture, sports, science and technology), gave us advanced ideas in treatment of CDAs (BDBEs) in the licensing. The draft of the DFBR safety evaluation policy indicates that CDA should be considered in the licensing procedure because:

(1) DFBR is yet a reactor under development, and also

(2) the consequence of recriticality could be severe.

Putting it the other way around, this means that CDA could be omitted from the licensing procedure if these issues are resolved in the future based on the results from safety R&D. Unfortunately the discussion has not been finalised because of Monju sodium-leakage accident in December 1995 and following reconsideration of DFBR project.

2.1.3 Discussion on Quantitative Safety Goal for Nuclear Safety Regulation in Japan

Nuclear Safety Commission of Japan established a special committee in February 2001 and started a discussion attempting to formulate a quantitative safety goal. The scope of discussion is widely ranged including NPPs, fuel cycle facilities, and waste disposal. It was originally planned to draw some conclusions within 2 years.

2.2 Proposal of safety goals & criteria

2.2.1 JNC's view

In this section, safety target in the development activity and safety design requirements in phase 2 of the feasibility study are described briefly. JNC's view is also shown in Appendix 1 and Ref.4.

Safety target in the development activity are summarized below.

- Recognizing deeply the potential hazard in the nuclear energy utilization, to assure sufficient safety level in each stage of design, construction, operation, and decommissioning.
- In the conceptual design stage, based on the defense-in-depth concept, high priority is put on accident prevention features.
- Mitigation features is considered against CDA (core disruptive accident).
- Comparative or superior safety level to that of LWRs in the same generation.
- The risk from the advanced reactors is smaller enough than the risk that already exists in the society without taking into account evacuation under severe accident.
- Consideration on physical and chemical characteristics of materials used in the system.
- Aiming at FR cycle system with social acceptability, taking the above into account.

And safety design requirements are summarized in the following.

(1) Safety Objectives Defined by Probabilistic Approach

Both the individual and social risk do not increase significantly

→Safety goal of JNC in the development activity: Frequency of large off-site releases is less than 1E-7/site year, which includes the nuclear fuel cycle facilities.

Based on above objectives, it is required quantitatively that the core damage frequency is less than 1E-6/ry and unreliability of containment capability is sufficiently small under representative CDAs.

NOTE: It should be emphasized that the above safety goal is proposed as a target in a development activity. This is not a goal to be considered in the regulatory activity. This goal could be re-considered if NSC would provide a safety goal in the regulatory context.

(2) Principles 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.

(3) 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).

2.2.2 Joint Safety Approach leading to social acceptance

JNC's view is presented briefly in Section 2.2.1, Appendix 1 and Ref.4.

CEA's view was presented in the meeting in June 2001, and reflected in the following descriptions.

Although CEA's view is focused on GCFR and JNC's view is rather generic, significant coherency is found in the following points.

(1) Defense-in-depth concept is also the fundamental principle for advanced reactors.
(2) Balance between prevention of accidents and mitigation of their consequences is considered.
(3) Although higher priority is put on prevention measures against severe core damage, mitigation measures against core melt are also taken into account for containment design for degraded-core with final recriticality-free configuration and for post-accident heat removal.
(4) Passive or inherent safety features shall be utilized provided that the uncertainty was demonstrated to be adequately small.
(5) Quantitative safety target is also considered as well as deterministic approach. Target CDF is lower than 10^{-6}/ry. Frequency of unacceptable release should be well below the CDF target (10^{-6}/ry).
(6) Detailed PSA are not required for systems without sufficient experience feedback. More appropriate method, and simpler one, is recommended for FR at project stage (the semi-probabilistic "Lines-of-Defense" method was extensively used for EFR and previous FRs). In order to check the global probabilistic objective (10^{-6}), application of the method will be able to check that the loss of each safety function (reactivity control, heat removal, confinement) is less than 10^{-7}/reactor year starting from each type of initiating event family.
(7) Measure shall be taken for elimination of any weak point or "cliff edge" effect that could

occur during a severe accident.

(8) "Evacuation free" concept is pursued. The radiological release shall be small corresponding to off-site emergency response and population evacuation. Any off-site impact and measure shall be as temporary (e.g. soil recovery).

(9) Improvement of social acceptance is considered from the development stage of the fuel cycle including NPPs. ("Acceptance as a total system" is taken into consideration.)

Both parties referred to IAEA/INSAG reports, and the objectives of Generation IV.

3. Safety Consideration for Various FRs Systems

3.1 Safety Comparison for Various FRs Systems

3.1.1 Safety Comparison

CEA and JNC have discussed on safety features of various FRs systems. The discussion has been focused on the safety characteristics caused by different coolant, rather than by different fuel. Because plant designs are not yet available for each coolant types, discussion was based on typical plant design concepts. Safety issues for each coolant type concept are listed in Table 3-1, and some comments are described below.

(1) Sodium cooled reactors

MOX-fueled sodium-cooled reactors have been developed for long time in various countries and its technology is fairly matured. Because sodium is not transparent, ISI and repair technologies should be continuously improved, though this is not directly a safety issue. Due to its chemical activated nature, sodium leakage/burning and sodium-water reaction in SG are still recognized as weak point that should be overcome.

(2) Gas cooled reactors

Gas-cooled fast reactor has different safety characteristics from those thermal neutron flux reactors because thermal capacity of the fuel in fast reactors is rather small. Therefore, coolability of the core in accident condition is weaker compared with liquid metal cooled reactors. Because the system pressure is high, containment of the coolant is also an issue to be considered. In the aspect of monitoring, leak localization measure is to be developed. As for coated particle fuels, the integrity of coating and its quality assurance could become one of the concerns because it will relate to the FP contamination of the heat transfer system in normal operation, which could influence to increase the workers exposure at maintenance of, for example, turbine. Usage of inert gas such as He might be advantageous in ISI and repair and social acceptance viewpoint.

(3) Lead or Lead-bismuth reactors

Corrosion issue is one of major concerns in heavy liquid metal (Pb or Pb/Bi) cooled reactors. Besides development of technologies and processes for corrosion prevention, potential of inspection capability needs to be improved for internal structures (e.g. core support). Freezing is a specific issue in Pb cooled reactors. LBB (leak before break) applicability is also not yet proven. LBB and DND (delayed neutron detector for fuel failure detection) technology are need to be developed. Shutdown devices should be developed to prove their high reliability. DHR should be validated but is not a major concern because of its high natural circulation potential. Because of the heavy coolant, anti-seismic design is important. This requirement could limit the size of the reactor. Higher boiling point is an advantage in safety point of view, though the cladding and structure integrity is much more concerned in accident conditions. Attention should be paid for toxicity of lead, and radiotoxicity of polonium.

(4) Water cooled reactors

Water-cooled fast reactors are designed based on the current advanced water-cooled reactors with thermal neutron spectrum. The technology is already matured, except for some parameters relating to the features for aiming at harder neutron spectrum. Coolability of narrower pin bundle, and CDA consequences taking into account the high Pu enrichment are major concerns. Usage of water is advantageous in ISI and repair and social acceptance viewpoint.

3.1.2 JNC's Investigation

In the feasibility study conducted at JNC, following candidate combinations of coolant and fuel are considered.

1) Na and MOX
2) Na and metal
3) Na and nitride
4) Pb/Bi and MOX
5) Pb/Bi and nitride
6) CO2 and MOX
7) He and nitride (pin or coated particle))
8) water and MOX

Na/metal has been studied in CRIEPI Japan, and water/MOX has been studied in JAERI, therefore JNC has started a collaboration with CRIEPI and JAERI.

Although we are not sure whether all of them could be a breeder, we are ready to start a preliminary investigation of safety characteristics of each system. First we will choose representative initiating events, and then will make a rough analysis of event sequences. Final target is to compare safety characteristics of the systems each other especially from the viewpoint of CDAs (neutronics potential and coolability), to extract fatal problems if any, and to list up the items for future R&Ds.

Review of the characteristics of various FRs

In the beginning of the study for safety design principle and fatal problems of safety, the characteristics of each plant design (mainly coolant aspect) have been reviewed and summarized as follows.

- neutronics performance: Na ~ Pb > gas > water
- coolability: Na > Pb > water > gas
- containability: Na ~ Pb > water > gas
- chemical stability: gas > water > Pb > Na
- negative void worth water > gas > Pb > Na

(based on the designs up to mid 2001)

1) Coolant: Na

Sodium is chemically active. Sodium fire and sodium-water reaction are to be considered.

2) Coolant: Pb

Because the melting point of steel is lower than the boiling point of lead, steel melting occurs prior to coolant boiling. This means that failure of support structure or reactor vessel could occur prior to coolant boiling. Therefore, the merit of high boiling point of lead cannot be fully utilised in accident conditions. Nevertheless, high boiling point introduce a gain (combined with high density) for passive DHR and prevent reactivity insertion through boiling.

3) Coolant: gas

Depressurisation accident is to be considered. Core catcher is needed if core damage is considered.

4) Coolant: water

LOCA is to be considered and ECCS is needed as well as LWRs. CDA and recriticality problem are also carefully considered if the Pu enrichment becomes higher.
As for supercritical-water type reactors, the initial safety margins should be reviewed (due to higher pressure and temperature) starting from proven technology, and normal operation stability should be confirmed (neutronic influenced by water parameters change).

5) Fuel: MOX

MOX fuel is currently most well known with a large experience feedback including transient behaviour for safety study. On the other hand, MOX is a 'hot' fuel (e;g. low conductivity) which influence the safety margins, and thermal behaviour is strongly changed by irradiation conditions (e.g. impact on local overpower transient). Use of MOX fuel is validated for sodium cooled FR, encompassing the safety aspects. For less efficient coolant needing high core volume fraction (lead or gas), fuel with high density and conductivity is generally preferred, but MOX fuel remains compatible.

6) Fuel: metal

Effect of eutectic formation in accident condition is to be considered. In order to keep integrity of the lower plate of the high pressure or low pressure plena in PAHR (post-accident heat removal) phase in CDA, a protective measure (such as Zr liner) might be needed.

7) Fuel: nitride

Because decomposition of the nitride fuel could be a fatal problem in accident conditions for sodium and heavy metal cooled reactors, its characteristics have been investigated. Decomposition phenomenon is endothermic reaction such as vaporization process, therefore instantaneous decomposition of all the fuel does not occur even if the fuel temperature exceeds its melting point. Only some fraction of the fuel, of which mass could be obtained from dividing the excess energy by the decomposition energy, will decompose.

Fuel coolant interaction between nitride fuel and sodium might be severer compared with that of MOX fuel and sodium because thermal conductivity of nitride is high. However, the effect of decomposed nitrogen gas on FCI phenomena is totally unknown.

Additional comments for reactor safety systems

(1) Passive and/or inherent safety features

Because passive features can be generally reliable but less efficient than active means, some passive means (e.g. DHR) are usually used as ultimate means after failure of active means, both implemented for the same safety function. Indeed, for limiting events sequence (assuming failure of “the first lines of defense” or in “accident” condition) relatively higher consequence level is acceptable compatible with capability of low efficient but reliable passive means.

Passive and/or inherent safety features will be further introduced in future reactors because its function is normally reliable and thus easy to be understood. However, considering those characteristics, the necessity of demonstration for these features is very important when the features are taken into account in safety assurance logic framework. An experiment using prototype reactor might be needed in some cases. Because the core characteristics are changing during burnup cycle, experiments at BOC and EOC are both needed if such features are concerned.

(2) CDA and recriticality issues

Although the prevention of accidents is highly focused in future reactors, mitigation is also taken into account in designing. Because of the characteristics of fast neutron reactors, CDA and recriticality issues should be adequately considered in development phase. Not only the mechanical energy but also thermal energy are produced if recriticality event occurs. Some comments are described below for each coolant types reactors.

Sodium cooled reactors: CDA characteristics are carefully studied, and measures for elimination of recriticality are proposed. Experimental programme is also progressing for studying fundamental phenomenology in fuel relocation.

Gas-cooled reactors: Direct attack of hot core material onto the reactor vessel could occur at recriticality event. Because cooling of the core material is critical, core catcher design is considered in the design stage.

Lead or lead/bismuth cooled reactors: Event sequence and consequence of CDA are not yet clarified. Because the fuel density is similar to the coolant, it is not clear whether fuel goes up or down at cladding melting. If it goes down, recriticality at core bottom becomes concern. If it goes up, fuel gathering at coolant surface becomes concern. Although the debris cooling issues might be easier by lead or lead/bismuth coolant, structure integrity is more critical at elevated temperature.

Water-cooled reactors: Severe accident sequences are widely studied for current LWRs. Therefore it is rather easy to find out the situation where recriticality issues should be considered. Because the reactor vessel is rather small due to lack of radial blanket subassemblies, recriticality at vessel bottom should be considered as well as at the original core region. If a recriticality event would occur, fuel temperature could be high and vapor explosion becomes another concern. Direct attack of hot core material onto the reactor vessel should also be considered in such a situation. Ex-vessel recriticality event should also be investigated. Phenomena specific to safety of water-cooled reactors (risk of hydrogen formation) remain to be considered in addition to new issues relating to fast neutron core feature.

3.2 Analysis of FRs Concepts

3.2.1 CDA Analysis of Gas-cooled and Lead-cooled Reactors

3.2.1.1 Recriticality potential map

In order to provide a straightforward insight about recriticality issues in CDA of fast reactors, static neutronic calculations have been performed to survey critical height of degraded core for various parameters as follows:

(1) fuel: MOX, metal, nitride,

(2) Pu enrichment is a parameter,

(3) reflector : bare, steel, lead, sodium, water,

(4) porosity: 0 ~ 50 %,

(5) mixture: vacant, sodium, lead, steel, water in porosity, and

(6) geometry: slab, cylinder, cone.

One of the typical results is shown in Fig. 3.1. One can easily compare this critical height and the height of degraded core, and then obtain a feeling of the likelihood about recriticality in CDA sequences.

3.2.1.2 Gas-cooled Rectors

An ULOF accident behavior in 3600MWth CO2-cooled MOX fueled reactor was analysed using SIMMER-III code. Brief descriptions are found in Appendix 2. The CDA characteristics of gas cooled reactor are summarized as follows:

- Small heat capacity and cooling capability of CO2
    - ➢ Coherent event progression throughout the core
- Sparse pin bundle
    - ➢ Hydraulic diameter ~ 12mm (typical LMFBR ~3mm)
    - ➢ Possibility of molten fuel removal through pin bundle
- Reactivity insertion mechanism
    - ➢ Loss of steel from the core after cladding melting (total loss of cladding from the core: =7.5$)
    - ➢ Fuel motion after pin disruption.

Based on the preliminary analysis of ULOF event, the following observations were derived:

- The cladding removal from the core drives recriticality in the initiating phase of CDA in CO2-cooled FBR.
- Fuel disruption occurs almost coherently throughout the core due to the small heat capacity and cooling capability of coolant.
- The molten fuel escapes to the lower plenum through the lower axial blanket due to the large hydraulic diameter of pin bundle and short blanket length.
- Thus, although the recriticality is inevitable, the mechanical consequences of the recriticality in CO2-cooled FBR are expected to be milder than sodium cooled reactor.
- Remaining issues:
  - ➢ Uncertainties on the initial ramp rate by cladding motion
  - ➢ Effect of blanket length and hydraulic diameter of pin bundle
  - ➢ Examination of gas heating upon fuel disruption

It should be mentioned that the thermal impact on the reactor vessel is not studied in this preliminary analysis. This thermal consequence should be treated much more carefully compared with sodium-cooled reactors.

3.2.1.3 Lead-cooled reactors

An ULOF accident behavior in 700MWth lead-cooled MOX fueled reactor was analysed using SIMMER-III code. Brief descriptions are found in Appendix 2. The CDA characteristics of lead-cooled reactor are summarized as follows:

- Boiling point of coolant > Melting point of steel
  - ➢ Cladding begins to melt in single phase coolant and goes upwards by buoyancy.
- Density of coolant is near to Density of MOX fuel at higher temperature
  - ➢ Disrupted fuel moves almost together with the coolant.
  - ➢ Sedimentation of fuel will be slow.
- Reactivity is inserted by;
  - ➢ Loss of steel from the core after cladding melting (total loss of cladding from the core: ~7$),
  - ➢ Void reactivity (inner core:~7$)
    - ✧ Fission gas blowout after cladding failure
    - ✧ Fission gas release upon fuel melting

- ➢ sedimentation of disrupted fuel.

Based on the preliminary analysis of ULOF event, the following observations were derived:

- The rate of reactivity insertion by cladding removal is less than several $/s which does not drive energetic recriticality.
- Fission gas release upon fuel melting has a potential to drive recriticality, but this must be re-examined using detailed meshing.
- Generally, the event progression in the CDA of lead-cooled FBR becomes mild due to the high density and high boiling point of lead.
- Before the achievement of final sub-criticality, the fuel must be brought out from the core by lead boiling or structure disruption. Recriticality possibility during these event need to be investigated.
- Remaining issues:
  - ➢ reactivity insertion by fuel sedimentation, and
  - ➢ consideration of natural circulation effect on fuel motion.

It should be mentioned that this analysis is not yet completed because of lack of knowledge about solid MOX fuel motion in liquid lead.

3.2.2 Discussions specific to Gas-cooled reactors

Some discussions on the safety of gas-cooled reactors were described in this section.

(1) List of safety events to be evaluated

JNC has prepared a list of events to be considered in safety assessment for gas-cooled reactors in the feasibility study. CEA pointed out some events to be added and both partners discussed on it. Table 3.2 shows the results of discussion. One should be careful that events in Table 3.2 will be changed depend of the system design, however, this could be utilized as a comprehensive check list.

(2) Core Catcher

Both parties recognize the necessity of core catcher for gas-cooled fast reactors.

Nevertheless, design investigation for minimizing core damage accident frequency remains an open task to be pursued.

A few designs of in-vessel core catcher were studied at JNC and proposed to CEA, and comments were provided based on the experience of LWR's accident management. Although no concrete conclusions have been derived from the discussion yet, both have agreed to keep this issue in future collaboration.

3.3 Identification of R&D needs

The common interest area for both parties is the safety aspects of gas-cooled fast reactors. Both are interested in helium as coolant gas.

Fuel type is not yet fixed in both parties. Because the safety characteristics of the fuel is a critical issue in reactor safety, discussion and recommendation to the design will be continued keeping in mind the safety target and the list of safety events to be evaluated.

Safety analyses of design basis and beyond design basis events are also the common interest area in order to increase the reliability of the analysis tools through comparative analysis and mutual discussion.

Both parties understand that SIMMER code is useful for the CDA analysis of gas-cooled reactors, continuation of SIMMER-III collaboration between JNC and CEA is agreed utilising EJCC framework. Model improvement, model validation, and reactor application will be planned and conducted under tight cooperation between NWP5(a) and EJCC side.

Items of model development will be clarified when the design of the reactor, especially for the fuel and subassemblies, becomes clearer. Validation experiment(s) will be considered coherently.

Core catcher issue has been identified as an important one to assure the containment capability and to meet the safety target of the future reactors. More deep investigation and associated experimental programme should be considered in the next step.

Table 3.1 Comparison of Safety Issues among Typical Fast Reactor Plant Design Concepts with Various Coolant Types

Issues / coolant	Sodium	Gas	Lead or Lead-bismuth	Water
Fuel type	MOX	MOX or nitride pin/coated particle	MOX or nitride	MOX
Prevention (should be adequately prevented for normal operation)	Na/air, Na/H2O	Gas leak	corrosion, freezing, issues due to high density	leakage (but proven technology)
Monitoring issues	redundancy (proven technology)	Leak localisation measure is to be developed	LBB, DND (need to be developed)	proven technology
Shutdown	Diversity, reliable, SASS	To be developed	To be developed Relilability not proven	proven technology
DHR	To prevent common mode in large size plant	Search for ultimate backup mean	To be validated (NC head is large)	proven technology
Reference fault	LOF Local melting	Mass flow loss Depressurization accid.	Blockage in core	LOCA
Aggravating	Na boiling, void effect, MFCI	Advantage inert, Water ingress?	Central void, freezing, earthquakes	more tight pin-bundle, re-entry of coolant at LOCA
1st barrier (e.g. cladding)	good knowledge	Advantage of coated particles for high temperature Unknown technology for fast neutron Issues of integrity inspection	Nitride/bond/pin Validation	good knowledge
Primary containment (RV & piping)	need for 2 barriers	Important for DHR	LBB applicability	Conventional function + consideration of recriticality
Secondary containment	mainly for sodium risk	Need for residual pressure Cont. spray?	Conventional function	Conventional function + consideration of recriticality
Core melting Out-site release?	Conventional approach	Post-accident DHR (in/ex-vessel core catcher?)	recriticality in later phase, and vessel heatup	recriticality due to high enrichment
Social acceptance	Na reaction, void, energetic accidents	Inert gas (advantageous), CDA consequence unknown	radiotoxicity (quantity?), CDA consequence unknown	water (advantageous), CDA consequence unknown
Mature technology	yes	no	no	yes

Table 3.2 List of Events to be Evaluated for Gas-Cooled Fast Reactors (1/2)

	coolant	CO_2[1]	He[2]	He[2]
	fuel type (pin/coated particle)	pin	c. p.[3]	pin
	fuel (MOX/MN)	MOX	MN	MN
	turbine (steam/direct gas)	steam	gas	gas
Abnormal Operational Transient				
(1)	abnormal change of reactivity/power distribution			
a.	C/R withdrawal from sub-crit condition	X	X	X
b.	C/R withdrawal at normal operation	X	X	X
c.	drop of C/R and mismatch	X	X	X
(2)	abnormal heat generation/removal			
a.	abnormal in primary coolant flow	X	X	X
b.	abnormal in water/steam flow (steam turbine concept)	X		
c.	abnormal water flow in pre- or inter-cooler (GT concepts)		X	X
d.	loss of off-site power	X	X	X
e.	loss of load	X	X	X
f.	failure of GT (e.g. inadvertent opening of by-pass valve)		X	X
g.	failure in the normal shutdown cooling system	X	X	X
(3)	others			
a.	local failure of the first barrier (cladding or coating)	X	X	X
b.	small leakage of HX pipe in boiler (water ingress)	X		
c.	small leakage of HX pipe in cooler (water ingress)		X	X
d.	fuel handling faults (cooling issue and reactivity insertion)	X	X	X
e.	internal and external hazards (fire, flooding, severe weather, etc.)	X	X	X
Accident				
(1)	reactivity increase			
a.	C/R rapid withdrawal	X	X	X
b.	C/R ejection		X	X
c.	fuel slumping accident	X		X
d.	steam into the core	X	X	X
(2)	reduction of cooling capability			
a.	seizure of components for primary flow (compressor, circulator)	X	X	X
b.	closure of any valve in the primary circuit	X	X	X
c.	main feed water pump stick	X	X	X
d.	failure at PCRV penetration / rapid depressurization (LOCA)	X	X	X
e.	blockage of coolant channel in the fuel subassembly	X	X	X
f.	failure of internal structure leading to core by-pass flow	X	X	X
g.	loss of redundant system (if not diversified) used for structure cooling	X	X	X
h.	loss of station service power (station blackout) for long duration	X	X	X
(3)	radiological material release to environment			
a.	primary coolant leakage	X	X	X
b.	failure of off-gas system	X	X	X
c.	malfunction of primary circuit valves (inadvertent opening)	X	X	X
(4)	others			
a.	failure of HX pipe in boiler	X		
b.	failure of HX pipe in pre- or inter-cooler		X	X
c.	ingress of foreign substances in the primary circuit (e.g. air ingress)	X	X	X
d.	loss of load at shutdown state (maintenance operation)		X	X
e.	blade failure in turbine/compressor (turbine missile)		X	X
f.	core loading error (severer than abnormal category)	X	X	X
g.	internal and external hazards (fire, earthquakes, explosion, aircraft crash, etc.)	X	X	X

Table 3.2 List of Events to be Evaluated for Gas-Cooled Fast Reactors (2/2)

		coolant	$CO2^{1)}$	$He^{2)}$	$He^{2)}$
		fuel type (pin/coated particle)	pin	c. p.[3)]	pin
		fuel (MOX/MN)	MOX	MN	MN
		turbine (steam/direct gas)	steam	gas	gas
Beyond Design Basis Event					
	(1)	reactor shutdown capability or reactivity insertion event			
	a.	positive reactivity insertion + failure of scram (UTOP)	X	X	X
	b.	loss of coolant flow + failure of scram (ULOF)	X	X	X
	c.	loss of heat sink (time margin) (ULOHS)	X	X	X
	d.	small loca + failure of scram	X	X	X
	e.	large water ingress depending on design concept	X	X	X
	(2)	cooling capability after reactor shutdown			
	a.	loss of all AC power (PLOHS)	X	X	X
	b.	double-ended guillotine rupture of largest pipe	X	X	X
	c.	delayed double leakage for design concept with double envelope	X	X	X
	d.	local fuel meltdown	X	X	X
	(3)	others			
	a.	severe accident considered as CDA reference or for elimination of any weak point (cliff edge effect)	X	X	X
Residual Risk situations[4)]					
	a.	large reactivity insertion (e.g. core support failure, core compation, serious ejection of control rod)	X	X	X
	b.	primary circuit failure leading to instantaneous total depressurisation (serious turbine missile)	X	X	X
	c.	Other severe cases	X	X	X

1) steam turbine concept

2) gas turbine concept

3) cated particle fuel type

4) In CEA, their consequences are not studied, but their frequency must be checked as very low < 10^{-7}/ry, by LOD (line of defence) method, because they represent sequences leading to loss of one of the 3 safety functions.
In JNC, these events are conseidered comprehensively in PSA study.

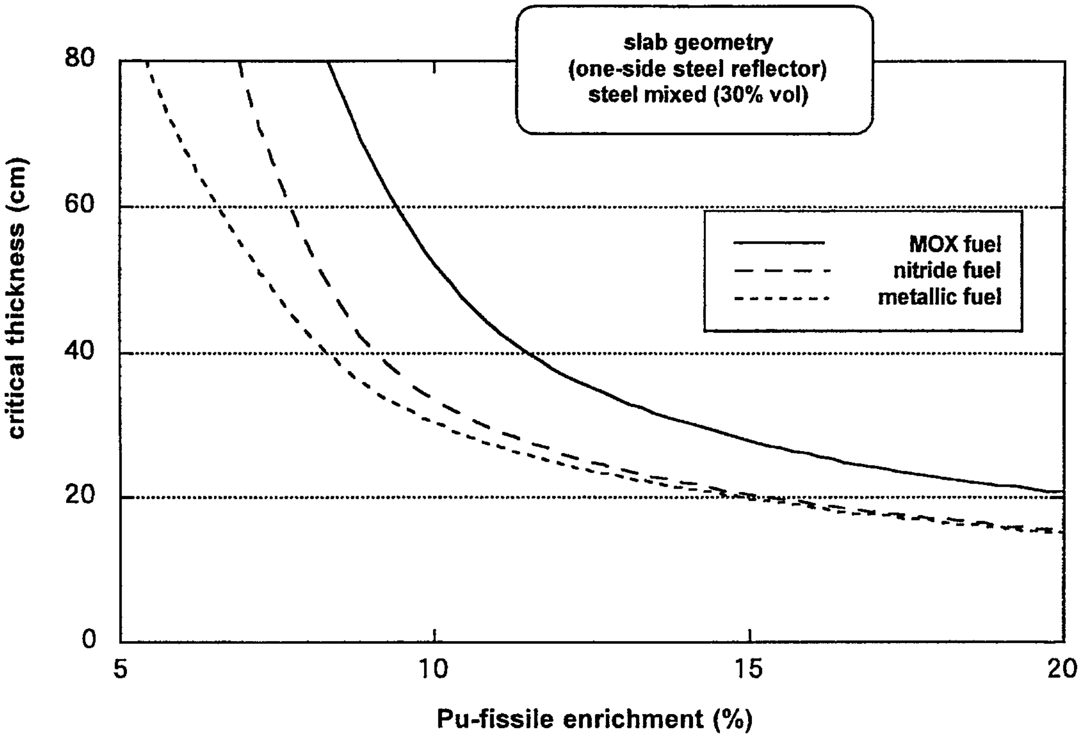

Fig. 3.1 Critical thichness of various fuel types (slabgeometry with 30%vol. steel mixed)

4. Concluding remarks

(1) Safety goals are proposed from JNC and CEA. Significant coherency is found such as to keep defense-in-depth concept, mitigation measures against core melt are taken into account for containment design, "Evacuation free" concept is pursued, quantitative safety target is also considered as well as deterministic approach, and improvement of social acceptance is considered from the development stage of the fuel cycle including NPPs.

(2) Gas-cooled fast reactor is a common interest area. Discussions are focused on: safety design requirements, safety evaluation events list, transient behavior analysis, core catcher designs, and so on. SIMMER-III code is recognized as a convenient and useful tool for future reactor analysis.

(3) Orientations for future work were identified as follows:

- Joint safety analyses (design-basis and beyond design-basis events) for different Fast Neutron Systems,
- Safety recommendations for the general design of future gas cooled Fast Reactors/Cycle systems,
- Identification of R&D needs for safety
- Launching of a joint R&D programme on safety issues, including calculation tools and related validation experiments

The collaboration on the action NWP-5a is recognised as being of great importance for the orientation of the innovative design studies.

5. References

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2. S. Kondo, et al., "Integrated Analysis of In-Vessel and Ex-Vessel Severe-Accident Sequences," *1990 Intl. Fast Reactor Safety Mtg.*, Snowbird, Utah, U.S.A., August 1990.
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4. Y. Kani and H. Niwa, "Safety Approach for the Japanese Advanced Reactor Program," ANS National Meeting, President's Special Session, June 19, 2001, Milwaukee, WI, USA

APPENDIX 1

Safety Target and Safety Design Requirement in Phase 2 of FS

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1. Safety Target in the Phase 2 of F/S

Although the "Safety target" in the R&D of the F/S at JNC is currently being discussed, they will be summarized as follows.

"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. Based on this understanding, in the conceptual design stage of the advanced reactor, the safety objectives is determined as follows:

- **To assure comparative or superior safety level to that of LWRs in the same generation as the advance reactors, and**
- **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 evacuation under the nuclear accident.**

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. On the basis of the defense-in-depth concept, it is required to put a high priority on accident prevention. Especially in the reactor systems, the accident should be naturally terminated inside the nuclear reactor plant with keeping the safety for the public. This should be achieved by means of adopting the passive safety measures for accident prevention, and adequate measures to mitigate the accident consequence postulating core disruptive accident (CDA).

Elimination of the re-criticality issue should be considered.

In this way, we will aim at developing the FBR cycle systems considering the social acceptability."

The points that should be emphasized here are as follows.

(1) Existence of the potential hazard in utilization of the nuclear energy should be always remembered, which is clearly described at the beginning in the White Paper on Nuclear Safety 2000 that was issued by the NSC. In addition, although the designer might have an illusion such that the safety is sufficiently enhanced by design, in order to attain the safety objectives, it is important to make an effort to break down the level of safety in each stage of construction, operation, and decommissioning, as well as aiming at the high level of safety in designing.

(2) What we can do in the conceptual design stage is only to design the systems so that the accident does not occur and that its consequence under the accident could be mitigated. The fundamental concept is defense-in-depth. However, in the level of small risk, we consider the cost versus benefit and the concept of ALARA by introducing the risk concept.

(3) Since it is thought that in the beginning period of introducing advanced reactors they will be operated together with the modified light water reactors, the concept of "comparative or superior safety level to that of LWRs" is essential.

(4) As one of indices to confirm adequacy of the safety design, the concept of risk is adopted. There are several proposals about the quantitative safety objectives. In our case, the objective is defined as to assure that increment of the risk generated by introducing the FBR cycle systems to the public near the facility related to the FBR cycle systems is smaller enough than the risk that already exists in the society.

(Quantitative description is given in the design requirements.) Furthermore, the objective is to attain the level of safety without taking into account any evacuation. Of course, if such evacuation is considered, the risk becomes smaller.

(5) A high priority will be placed on the countermeasure for accident prevention. To enhance the accident prevention, the passive safety measure is adopted as well as high quality assurance in design and fabrication, assuring the adequate safety margin, etc. Particularly in the development stage of advanced reactors, it is important to confirm and assure the feasibility of the reactor concepts or the structural integrity, and to extract development issues to be further resolved. For these purposes, it is essential to accumulate experiences through irradiation test of fuel and materials, mockup test of components, as well as computer simulation.

(6) In addition, the accident consequence is adequately mitigated so that the cliff-edge does not appear even if CDA is postulated. According to the FBR core characteristics, in addition to the issue of heat removal from degraded core, there is possibility to generate mechanical energy induced from re-criticality under CDA because change of fuel arrangement may result in positive reactivity effect. So, the system should be designed so as to be able to eliminate the re-criticality issue under CDA so that the accident is naturally terminated inside the nuclear reactor plant with keeping the safety for the public.

(6) The social acceptability should be judged by the society, and it is not suitable to the objective of development side. So, the objective was described as an effort target that we “aim” at being accepted by the society as a result. However, as consideration to social acceptability, research and investigation are implemented, and their results are reflected in design examination as much as possible.

(7) In addition, it is thought that the seismic risk is important in Japan and that external factors such as an earthquake are, of course, included in the risk to be considered. In this context, development and adoption of a three-dimensional (3D) seismic isolation system is considered in the feasibility study, where two basic structural concepts are pursued. One is 3D base isolation of a whole nuclear island, and the other is a combination of horizontal base isolation and vertical isolation at component level. Development projects of isolation devices for these concepts are underway. Technical feasibility and potential benefits on enhanced safety and economy of the 3D isolation are also assessed.

2. Safety Design Requirements

Requirements on design related to safety for future reactors consist of following 3 points:

- Safety objectives defined by probabilistic approach.
- Principles of safety design.
- Requirements for three important functions.

(1) Safety Objectives Defined by Probabilistic Approach

It is required that both the individual and social risk do not increase significantly when FBR cycle is introduced as a power generation system.

→ Safety goal of JNC: Frequency of large off-site releases is less than 1E-7/site year, which includes the nuclear fuel cycle facilities.

Based on above objectives, it is required quantitatively that the core damage frequency is less than 1E-6/ry and unreliability of containment function is sufficiently small under representative CDAs.

We obtained the value of 1E-7/site year using the conventional Japanese risk data. Since the same method as NRC (Ref.1, NUREG-0880) was used here, there is no need to go into details. However, it is desirable to define the risk using a unit of ' / site year' instead of ' /reactor year' in Japan, because one site usually has several power plants and also the co-location concept of reactors and nuclear cycle facilities is considered for future FBR cycle system.

(2) Principles of Safety Design

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

(3) Requirements for Function of 'Reactor Shut Down', 'Heat Removal' and 'Containment'

'Reactor Shut Down':

There are two options for design of additional reactor shut down system. One is to add a passive shut down system or feature. The other option is to add a passive mechanism which can moderate accident progression and can allow the operator managements to terminate the accident. If a system is designed to prevent the core damage in a short time and it is shown that the operator managements prevent the core damage, there is no need to add a passive shut down system. (But CDF must be less than 1E-6/ry.)

'Heat Removal':

Decay heat removal systems after the reactor shut down are designed considering redundancy or diversity, and to achieve core cooling even if a failure of dynamic mechanism is assumed. In addition, the failed systems are designed so as to expect the recovery of the function by accident management.

'Containment':

In order to reduce a risk reasonably under representative CDAs, the following measures are required which consider features of the plant system and event sequence. This measure can minimize and localize the accident consequence and achieve satisfactorily small unreliability of containability.

- 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).

The features to be described here are summarized below.

Future power plants we are studying are required to eliminate re-criticality issue, which is the characteristic of the fast reactor, and to keep the containment function by the post accident heat removal even if CDAs are assumed. Of course, there is no need to add special measures if a design satisfies above requirements against CDAs. (It is acceptable to add and reduce measures according to the plant characteristics.)

We have discussed about the concept of 'CDA-Free', which does not require the measure for CDAs (core catcher, for example), as a goal with high safety objective. However, by the following reasons, we decided to require the design to equip the containment function which takes into account the CDAs consequence:

- It seems rather difficult to prove a CDF value much smaller than 1E-6/ry when external factors are taken into account. It is the case especially in advanced reactor concepts on which we do not have experiences. We think that the range of explanation accountability which a developer could cover is limited within achieving core damage frequency less than 1E-6/ry (or not larger than 1E-6/ry), keeping the containment function under representative CDAs, and thus achieving negligible risk increase.

Here, elimination of re-criticality issue has to be touched. In conventional safety assessments, evaluations have shown that the effect of CDAs could be contained within the reactor plant even if the mechanical load by re-criticality phenomena 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 structural design of reactor vessel or in-core structures. Therefore, a design concept is being studied for sodium-cooled FBR in which core melt extension is limited within each sub-assembly and fuel escape from the core is enhanced. In order to confirm the effectiveness of such design, the experimental study called EAGLE project (Ref.2) has started. This project uses the out-of-pile experimental facility and safety test reactor called IGR in National Nuclear Center in the Republic of Kazakhstan. The results are going to be included in the design study of the F/S.

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APPENDIX 2

CEA/JNC meeting NWP-1a, NWP-2a and NWP-5a May 16-18, 2000

Preliminary Safety Comparison of Sodium/Gas/Lead concepts

by

Yoshiharu TOBITA (compiler)
SRS/ATD/OEC/PNC

[1]

Objectives of this study

- Ellucidate the characteristics of CDA in each candidate.
  - Sodium cooled reactor (reference plant)
  - Lead cooled reactor:
    - high density and boiling point of Lead
  - Gas (CO2) cooled reactor:
    - negligible void reactivity and sparse pin bundle
- Examine the existence of crucial problems
- Evaluate the necessity of dedicated measures to exclude the energetic recriticality

[2]

Characteristics of CDA in Lead-cooled FBR

- Boiling point of coolant > Melting point of steel
  - The cladding begins to melt in the single phase coolant and goes upwards by buoyancy.
- Density of coolant < Density of fuel (~5%)
  - The disrupted fuel moves almost together with the coolant.
  - The sedimentation of fuel will be slow.
- Reactivity is inserted by;
  - loss of steel from the core after cladding melting (total loss of cladding from the core: ~7$),
  - void reactivity (inner core:~7$)
    - FP gas blowout after cladding failure
    - FP gas release upon fuel melting
  - sedimentation of disrupted fuel.

[3]

Calculation method in the CDA analysis

- Initiating phase analysis by SIMMER-III
  - Limitation of SAS4A
    - difficulty in replacing the coolant properties
    - the melting of cladding in prior to the coolant voiding
  - The event progression will be mild so that the detailed pin model will not be necessary.
- FP gas blowout model is transplanted from SAS4A to SIMMER-III
  - Blowout from both the upper and lower plenum is treated.
  - Test calculation confirmed;
    - the same blowout velocity with SAS4A, and
    - mass conservation.

[4]

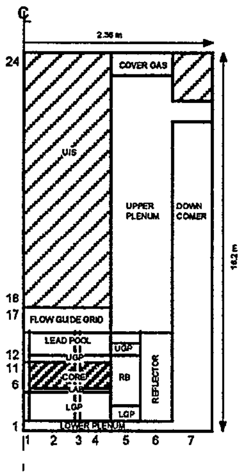
Specification of Lead-cooled FBR analyzed

■ Fuel :	MOX
■ Coolant:	Lead
■ Power:	700 MWth / 300 MWe
■ Inlet/Outlet temperature:	420/540 ℃
■ Core height/diameter:	120/216 cm
■ Height of LAB/UAB:	10.0/2.2 cm
■ Lower/upper gas plenum length:	29.8/127.5 cm
■ Fuel volume fraction:	31.21 %
■ Av. Burnup:	143 MWD/t
■ Fuel smear density:	0.82TD
■ Breading ratio:	1.1

[5]

Calculation condition and procedure

- ULOF is selected as the initiating event.
  - Flow halving time of 10.0s (BREST-300)
- Simplified geometry
  - One radial mesh for inner and outer core : limitation of computer resource
  - High coherency of event progression
  - Necessity for detailed meshing before final conclusion
- Procedure
  - Steady state calculation for 30.0 s with nominal power and flow rate
  - Inlet pressure reduction to cause the flow coast down.

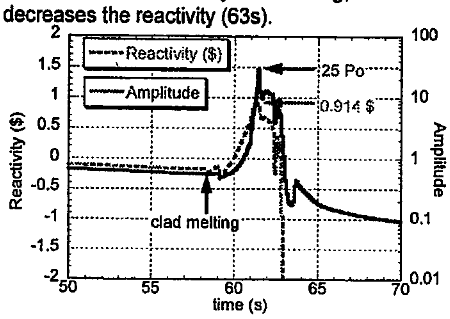


[6]

The result of CASE1 (1)

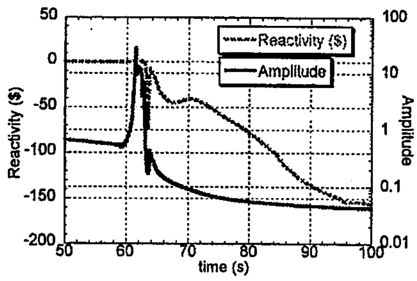
■ CASE1: without FP gas blowout

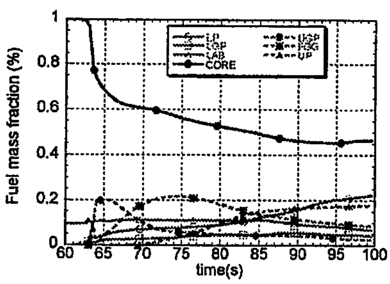
- The reactivity increases only by the escape of molten cladding from the core.
- Cladding starts to melt at 2/3 of the core height.
- Maximum ramp rate ~ 7 \$/s, Energy release ~20FPS.
  - Local fuel temperature < 3400K, fuel quenching by Lead, no mechanical load to reactor vessel
- The voiding of inner core occurs by lead boiling, but the simultaneous fuel dispersion decreases the reactivity (63s).



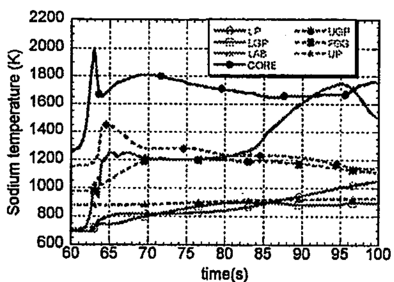
[7]

The result of CASE1 (2)





- C/W failure around 62s and fuel escapes downward through CRGT.
- The fuel inventory in the core is about 50% at 100 s (23% in lower plenum and 18% in upper plenum).
- High temperature lead (~2000K) stays close to the core.

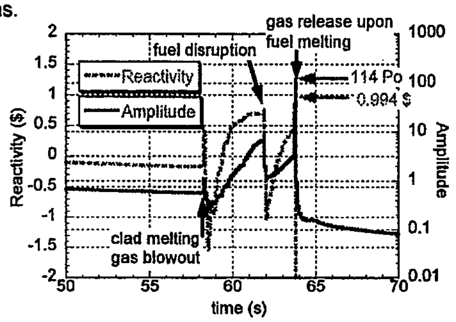


[8]

The result of CASE2 (1)

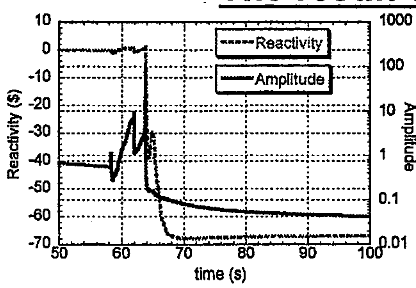
■ CASE2: with FP gas blowout and release upon fuel melting

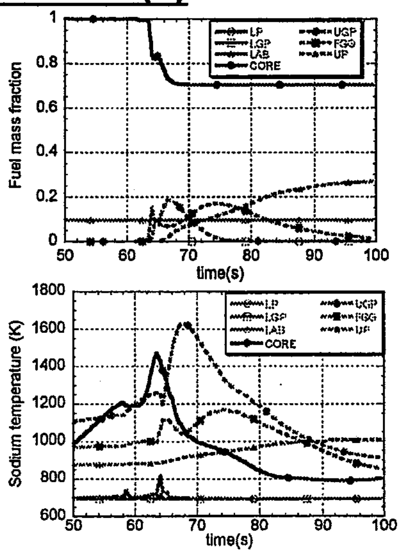
- FP gas blowout does not drive the recriticality, because the void extends only upward and the effect of neutron leakage becomes dominant.
- The FP gas release after fuel melting causes reactivity increase (~14 $/s), however it does not drive the recriticality in this case (~17FPS).
- The reactivity decreases very rapidly by Doppler reactivity and fuel dispersal by FP gas.



[9]

The result of CASE2 (2)





- CRGT wall does not fail.
- The fuel inventory in the core is about 70% at 100 s (30% in upper plenum).
- The temperature of the lead will be kept lower than 1000K after 100 s.

[10]

Conclusion of lead-cooled FBR analysis

- The rate of reactivity insertion by cladding removal is less than several $/s which does not drive energetic recriticality.
- The FP gas release upon fuel melting has a potential to drive recriticality, but this must be re-examined using detailed meshing.
- Generally, the event progression in the CDA of lead-cooled FBR becomes mild due to the high density and boiling point of lead.
- Before the achievement of final sub-criticality, the fuel must be brought out from the core by the lead boiling or the structure disruption. Recriticality possibility during these event need to be investigated.
- Remaining issues:
  - analysis with detailed meshing,
  - reactivity insertion by fuel sedimentation and
  - consideration of natural circulation

[11]

Characteristics of CDA in CO2-cooled FBR

- Small heat capacity and cooling ability of CO2
  - Coherent event progression throughout the core
- Sparse pin bundle
  - Hydraulic diameter ~ 12mm (Typical LMFBR ~3mm)
  - Possibility of molten fuel removal through pin bundle
- Reactivity insertion mechanism
  - Loss of steel from the core after cladding melting (total loss of cladding from the core: ~7.5$)
  - Fuel motion after pin disruption

[12]

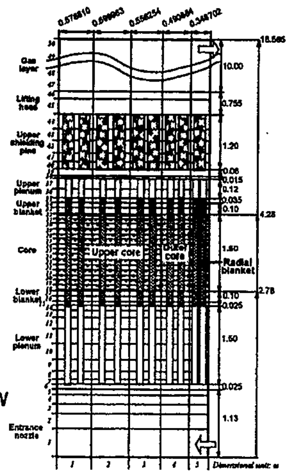
Specification of CO2-cooled FBR analyzed

■ Fuel :	MOX
■ Coolant:	CO2
■ Power:	3600 MWth / 1440 MWe
■ Inlet/Outlet temperature:	240/525 ℃
■ Core pressure	42 bar
■ Core height/diameter:	150/480 cm
■ Height of LAB/UAB:	10.0/10.0 cm
■ Gas plenum length:	162 cm
■ Fuel volume fraction:	31.21 %
■ Av. Burnup:	120 MWD/t
■ Fuel smear density:	0.88TD
■ Breading ratio:	1.05

[13]

Calculation condition and procedure

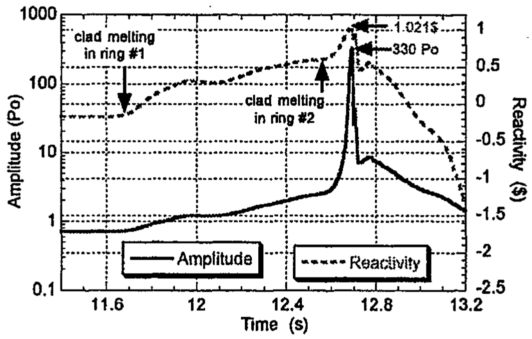
- Initiating phase analysis by SIMMER-III
- ULOF is selected as the initiating event.
  - Flow halving time of 4.4 s
- Geometry
  - 3 radial meshes for inner core and 1 radial mesh for outer core
  - Control rod channels are neglected.
- Procedure
  - Steady state calculation for 30.0 s with nominal power and flow rate
  - Inlet pressure reduction to cause the flow coast down.



[14]

Results of CO2 reactor analysis (1)

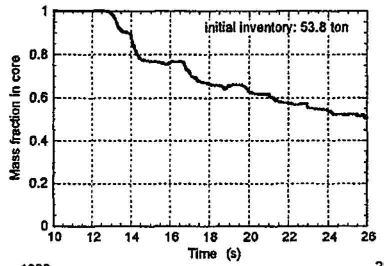
- Cladding melting at mid-plane and upward motion by coolant flow (~20% of nominal flow)
- Recriticality by steel loss reactivity (max. ramp rate ~8 $/s, energy production~10FPS, local fuel temperature peak ~ 3400K)
- Inner core fuel disruption in 30 ms after recriticality.

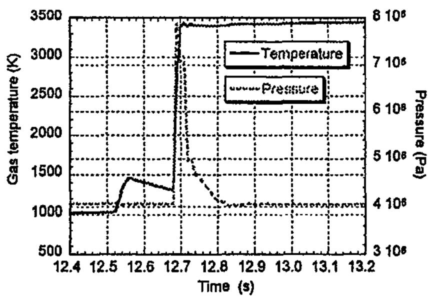


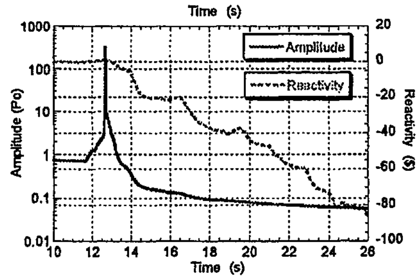
[15]

Results of CO2 reactor analysis (2)

- Reactivity decrease
  - Fuel dispersion in core by heated gas pressure (~few ten ms after recriticality)
  - Fuel relocation to lower plenum (~25s) through the fuel pin bundle







[16]

Conclusion of CO2-cooled FBR analysis

- The cladding removal from the core drives recriticality in the initiating phase of CDA in CO2-cooled FBR.
- The fuel disrupts almost coherently throughout the core due to the small heat capacity and cooling capability of coolant.
- The molten fuel escapes to the lower plenum through the lower axial blanket due to the large hydraulic diameter of pin bundle and short blanket length.
- Thus, although the recriticality is inevitable, the mechanical consequences of the recriticality in CO2-cooled FBR are expected to be milder than sodium cooled reactor.
- Remaining issues
  - Uncertainties on the initial ramp rate by cladding motion
  - Effect of blanket length and hydraulic diameter of pin bundle
  - Examination of gas heating upon fuel disruption

[17]

Concluding Remarks

- The ULOF accidents are analyzed both in Pb and CO2 cooled FBR.
- The consequences of CDA in these reactors become milder than sodium cooled reactor because of
  - high density and boiling point of Lead in Pb-cooled reactor and
  - sparse pin bundle in CO2-cooled reactor.
- The dedicated measures for the exclusion of recriticality may not be required for these reactors.
- Several remaining issues need to be clarified to draw the final conclusion.

[18]