ANL-GenIV-099

Progress Reports for Gen IV Sodium Fast Reactor Activities FY 2007

Nuclear Engineering Division

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Progress Reports for Gen IV Sodium Fast Reactor Activities FY 2007

by
A. Tentner and J. Cahalan
Nuclear Engineering Division, Argonne National Laboratory

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Contents

Page

1. Introduction

An important goal of the US DOE Sodium Fast Reactor (SFR) program is to develop the technology necessary to increase safety margins in future fast reactor systems. Although no decision has been made yet about who will build the next demonstration fast reactor, it seems likely that the construction team will include a combination of international companies, and the safety design philosophy for the reactor will reflect a consensus of the participating countries. A significant amount of experience in the design and safety analysis of Sodium Fast Reactors (SFR) using oxide fuel has been developed in both Japan and France during last few decades.

In the US, the traditional approach to reactor safety is based on the principle of defense-in-depth, which is usually expressed in physical terms as multiple barriers to release of radioactive material (e.g. cladding, reactor vessel, containment building), but it is understood that the 'barriers' may consist of active systems or even procedures. As implemented in a reactor design, defense-in-depth is classed in levels of safety. Level 1 includes measures to specify and build a reliable design with significant safety margins that will perform according to the intentions of the designers. Level 2 consists of additional design measures, usually active systems, to protect against unlikely accidental events that may occur during the life of the plant. Level 3 design measures are intended to protect the public in the event of an extremely unlikely accident not foreseen to occur during the plant's life. All of the design measures that make up the first three levels of safety are within the design basis of the plant. Beyond Level 3, and beyond the normal design basis, there are accidents that are not expected to occur in a whole generation of plants, and it is in this class that severe accidents, i.e. accidents involving core melting, are included. Beyond design basis measures to address severe accidents are usually identified as being for <u>prevention</u> of progression into severe accident conditions (<u>prevention</u> of core melting) or for <u>mitigation</u> of severe accident consequences (<u>mitigation</u> of the impact of core melting to protect public health and safety). Because design measures for severe accident prevention and mitigation are beyond the normal design basis, established regulatory guidelines and codes do not provide explicit identification of the design performance requirements for severe accident accommodation.

The treatment of severe accidents is one of the key issues of R&D plans for the Gen IV systems in general, and for the Sodium Fast Reactor (SFR) in particular. Despite the lack of an unambiguous definition of safety approach applicable for severe accidents, there is an emerging consensus on the need for their consideration for the design.

The US SFR program and Argonne National Laboratory (ANL) in particular have actively studied the potential scenarios and consequences of Hypothetical Core Disruptive Accidents (HCDA) for SFRs with oxide fuel during the Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor Plant (CRBRP) programs in the 70s and 80s. Later, the focus of the US SFR safety R&D activities shifted to the prevention of all HCDAs through passive safety features of the SFRs with metal fuel in the Integral Fast Reactor (IFR) program, and the study of severe accident consequences was de-emphasized.

The goal of this paper is to provide an overview of the current SFR safety approach and the role of severe accidents in Japan and France, in preparation for an expected and more active collaboration in this area between the US, Japan, and France.

2. Characterization of Japanese (JSFR/JAEA) SFR Safety Approach

This section provides a summary of the safety approach to severe accidents currently employed in Japan in the design of the Japan Sodium Fast Reactor (JSFR). Information summarized in this section was collected from presentations at technical meetings by staff of the Japan Atomic Energy Agency (JAEA). Information on proprietary design details in the original presentations has been excluded from this summary.

2.1 Safety Targets and Design Principles

The Japanese SFR safety targets were set aiming at world wide acceptance. The SFR design must ensure:

(1) A comparable or superior safety level to that of same-generation LWRs.
(2) A risk much lower than the risks encountered in daily activities, without taking into account the need for offsite emergency responses

These targets are consistent with the safety-related goals or user requirements in the Generation IV project and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO/IAEA).

The safety design principle is Defense-in-depth, with the following defense levels:

1. Prevention of abnormal occurrences
2. Control abnormal operations
3. Control accidents
4. Manage severe accidents
5. Offsite emergency response

While levels 2 and 3 above refer to Design Base Events (DBE), the severe accidents referred to in level 4 are not DBEs but are considered Design Extension Conditions (DEC). The defense-in-depth approach is illustrated in Fig. 1. The defense levels 2 and 3 (DBE) rely on two independent Reactivity Shutdown Systems (RSS), the primary RSS and the secondary RSS. The Decay Heat Removal System (DHRS) for these events includes redundant systems with passive operation, such as the Direct Reactor Cooling System (DRACS) and the Primary Reactor Cooling System (PRACS).

The Defense-in-depth strategy is complemented by a Risk-informed approach, with the goal of an estimated large offsite release frequency lower than 10^{-6} /site-year. Other risk related goals include: a) Core damage frequency (CDF) less than 10^{-6} /reactor-year (ry), and b) Unreliability of containment capability sufficiently small under representative Core Disruptive Accident (CDA) conditions. A preliminary Probabilistic Safety Assessment (PSA) based on existing reliability data base (including data from JOYO, MONJU, EBR-II, FFTF, and Japanese LWRs) resulted in CDF << 10^{-6} /ry : a) Anticipated Transient Without Scram (ATWS): 10^{-8} /ry to $3x10^{-8}$ /ry, b) Loss of Reactor Level (LORL): $4x10^{-9}$ /ry, and c) Protected Loss of Heat Sink (PLOHS: $2x10^{-8}$ /ry

Prevention against and mitigation of Severe Accidents relies on several elements:

1. An additional passive self-actuated shutdown system (SASS)
2. Conditions for the elimination of severe re-criticality:
    - The sodium void worth is less than 6 dollars.
    - The core height is less than or around 1m.
    - Enhanced molten fuel discharge from the core region.
3. Long term cooling of fuel debris (In Vessel Retention)

2.2 The Role of Severe Accidents

Protection against re-criticality is a central issue in the consideration of severe accidents. The fast reactor core is not in its maximum reactivity geometry. The minimum critical mass is less than several hundred kg, or fuel in several subassemblies (SA) (MOX, Pu-fissile 15%, sphere, no reflector). During a severe accident that leads to fuel melting in several fuel subassemblies, providing an early molten fuel escape from the core can avoid re-criticality and a severe power burst. This target scenario and provides the motivation for modified subassembly designs that enhance the early molten fuel escape from the core.

Two independent RSSs + SASS (2 strong + 1 weak) are sufficient to reduce the occurrence frequency of ATWS to very small value (around 10^{-8} /ry). However:

- Rapid accident progression of ATWS could result in early large release (Cliff-edge effect).
- PSA for advanced reactors with less operating experiences: 11,000 reactor years of experience for LWRs, but only 300 reactor years for SFRs.
- Many Gen-IV plants will be constructed and be used for a long term period in various societies.
- The re-criticality issue is an important issue in fast reactor safety and public concern is rather high.

Therefore, JAEA considers that the re-criticality issue should be resolved, and thus it should be considered in design to avoid a large mechanical energy release, but the cost should be minimized. Improvement of counter measures is essential in order to minimize the impact on core performance and fabrication cost.

An important counter measure against re-criticality in the JSFR is a fuel assembly designed to facilitate the early molten fuel escape from the core. A sequence of fuel assembly designs have evolved to minimize the impact on core performance and fabrication cost. The current reference design is Modified-FAIDUS (Fuel Assembly with Inner Duct System), with a corner channel for fuel escape. The current R&D target relies on a slim (thin wall) Control Rod Guide Tube (CRGT) to provide the paths for molten fuel escape. This design allows normal fuel subassembly fabrication and requires no fuel displacement in normal operation.

The Post Accident Heat Removal (PAHR) for JSFR is designed to prevent the debris bed from reaching the limit conditions for a coolable debris bed, thus achieving in-vessel debris retention:

- Critical thickness : >30cm
- Cooling limit (bed dry out condition with porosity 0.5)
  - 10 cm for bed formation just after shutdown
  - 15 cm for bed formation 1000 second after shutdown

The basic idea that underlies the PAHR design is to broaden the fuel debris bed as much as possible inside the reactor vessel:

- Upward relocation and in-place cooling inside the core will help to reduce the amount of the molten fuel which might reach the bottom of the reactor vessel (RV). For the fuel relocating upward, the intermediate isolation plate can retain a debris bed with up to 40% of the core fuel within the limit conditions.
- For the downward relocating fuel, the core support structure and the multi-layered debris tray can retain 100% of the fuel within the limit conditions. The core support structure is designed to protect against direct molten fuel jet attack, while multi-layered debris tray at the bottom of the reactor vessel is designed for debris retention the debris bed height limits of cooling and sub-critical state.

2.3 Core Disruptive Accident Analysis

CDAs are not DBA but BDBE. They are assessed in the licensing and the assessment must be conducted with "realistic" (best estimate) analysis codes. Selected BDBEs, including ULOF, UTOP, LOPI, and Local Fault are evaluated in order to confirm the safety margin, or no cliff-edge effects. ULOF was recognized as an envelope of core damage effects and the whole ULOF scenario and consequences were evaluated for DFBR, Monju, and JSFR. The phenomenological BDBE event tree in ULOF, potentially leading to CDAs is illustrated in Fig. 2 [Ref. 3]. A ULOF analysis was completed for JSFR with modified FAIDUS assemblies. The initiating phase was calculated with the SAS4A code and showed limited fuel melting (12.6%) just after the initial power transient termination. Continuation of the SAS4A calculations for another 3 s showed a gradual increase of the fuel melting (34.6%) with a relatively small sub-criticality. The core conditions at this time were then used as initial conditions for the SIMMER III calculations. After the failure of the inner duct, the fuel escape was driven by the pressure difference between the core (about 10.0 atm) and upper sodium plenum (about 2 to 2.5 atm). The modified-FAIDUS expelled 90 % of molten materials in core. The fuel escape behavior in several typical sub-assemblies with various power levels was integrated to estimate the whole core behavior. The fuel relocation after the failure of the inner tube was calculated using the SIMMER code. It appears - but it is not explicitly stated in Refs. 1 and 4 - that SIMMER and SAS4A continued to work simultaneously in different fuel assemblies after the SIMMER initiation. The power transient was taken from SAS4A analysis, accounting the decrease of reactivity and power by fuel escape. The results, illustrated in Fig. 3, support the feasibility of the avoidance of severe re-criticality by fuel escape from the core in the early phase of CDA.

The fuel relocation results from SAS4A and SIMMER calculations were used to evaluate the longer term post-accident heat removal (PAHR). In these calculations, which considered the decay heat just after shutdown, 40% of the fuel was located on the intermediate isolation plate and 60% remained inside the core. The PAHR simulation used 3D fluid dynamics inside the RV and 1D net-work flow for the primary circuit with DRACS x 1 and PRACS x2. The outermost row of the core, radial blanket and radial shield were available for cooling paths, while the rest of the core region was assumed to be totally blocked. The results indicated that the decay heat balanced the removed heat around 30 minutes after the start of the transient event and the maximum temperature of the debris support structure remained below 700°C. Thus, it the structure integrity could be preserved.

2.4 Summary

CDAs are evaluated in the category of Beyond Design Base Events (BDBE) or Design Extension Conditions (DEC). ATWS/ULOF is one of the major concerns for the severe accident consideration. Both CDA prevention and mitigation features are included in the JSFR design:

- Prevention: SASS
- Mitigation: Elimination of severe re-criticality and in-vessel debris retention

R & D work is under way for the development, enhancement, and evaluation of these CDA prevention and mitigation capabilities. Severe core damage can be ruled out from the design and accident management considerations by achieving sufficiently low occurrence probability.

Improvement of the M-FAIDUS fuel assembly is considered essential to reducing the design impact. The current R&D target is the Slim CRGT design, which will be analytically and experimentally studied in detail. A SIMMER-III/IV collaboration is considered desirable by JAEA for this purpose.

3. Characterization of French (AREVA/CEA) SFR Safety Approach

This section provides a summary of the safety approach to severe accidents currently employed in France in the design of sodium fast reactors. Information summarized in this section was collected from presentations at technical meetings by staff of Areva and the Commissariat à l’énergie atomique – Cadarache (CEA). Information on proprietary design details in the original presentations has been excluded from this summary.

3.1 EFR Safety Approach

The European Sodium Fast Reactor (EFR) is based on a pool-type design. The EFR safety approach is based on risk minimization. It includes:

- Comprehensive and balanced safety concept by harnessing the favorable features of the sodium fast reactor
- Avoiding weak points and cliff edge effects in the beyond design basis area
- Prevention of accidents and minimization of their consequences if they occur
- ALARA principle

CDA might occur in case of:

- Failure of the shutdown function
- Failure of the decay heat removal (DHR) function
- "Exotic" initiators: e.g., fast structural failure or insertion of large gas amount into the core

Improvement of CDA prevention requires improvement of the shutdown and heat removal functions.

Improvement of the shutdown function is obtained by adding a third shutdown system, which operates both passively and actively in case of postulated failure of the two basic shutdown systems. The two basic shutdown systems are designed for efficient operation in case of serious imbalances between produced and removed power:

- Failure of primary pumps
- Loss of main heat sink
- Withdrawal of absorbers

The objective of the third shutdown system is to relegate unprotected transients such as ULOF, ULOHS, and UTOP outside the realm of technical imagination.

Improvement of the DHR function is obtained through complementary DHR measures, including the implementation of a debris tray for long term retention and coolability of molten core materials.

3.2 The Role of Severe Accidents

The post-EFR generation of European commercial sodium fast reactors is designed with consideration of CDAs. After improvement of safety functions, no CDA scenario is credible. The frequency of a combination of CDA initiators with postulated failure of the first and second shutdown systems and the third shutdown level is significantly below 10^{-7} per year. CDAs are Beyond Design Conditions.

CDA analyses, traditionally based on ULOF, are performed to assure that there are no cliff-edge effects. They also provide appreciation of the relative importance of core characteristics in order to obtain a good balance between design and beyond design requirements.

Reasonable containment measures are provided for the mitigation of radiological releases in beyond design conditions. They include:

- Improvement of the primary containment for increased resistance against mechanical energy release from CDA
- Implementation of a debris tray for long term retention and coolability of molten core
- Definition of beyond design basis plant states for demonstrating the effectiveness of secondary containment

3.3 Core Disruptive Accident Analysis

CDA analyses are performed to identify the core characteristics which minimize the consequences of CDAs. The sodium fast reactor characteristics lead to the consideration of the ULOF scenario for assessing the core behavior.

3.4 Summary

CDAs are evaluated in the category of Beyond Design Base Events (BDBE) or Design Extension Conditions (DEC). ATWS/ULOF is one of the major concerns for the severe accident consideration. Both CDA prevention and mitigation features are included in the EFR design:

- Prevention: A third shutdown system, both passively and actively activated
- Mitigation: Enhanced in-vessel coolability and retention

R & D work is under way for the development, enhancement, and evaluation of these CDA prevention and mitigation capabilities. Severe core damage can be ruled out from the design and accident management considerations by achieving sufficiently low occurrence probability.

4. ANL Experience Related to Core Disruptive Accidents

ANL has extensive experience in the area sodium fast reactor (SFR) severe accidents analysis. During the US FFTF and CRBRP projects, in the 70s and 80s, ANL played a lead role in the study of severe accidents for SFRs with oxide fuel pins. The research included in-pile and out-of-pile experiments for the study of severe accidents phenomena as well as an extensive code development effort for the quantitative evaluation of the postulated severe accident consequences. Specialized modules of the SAS4A code were developed to describe the material relocation during UTOP events (PLUTO) and ULOF transients (LEVITATE). These modules allowed SAS4A to perform detailed whole core analyses of the initiating phase of postulated severe accident such as ULOF, UTOP, and LOF driven TOP. Through international agreements, SAS4A was shared with research organizations in Japan, France, Germany, and UK and became the de-facto worldwide standard for the study of the initiating phase of postulated severe accidents in oxide fuel SFRs.

With the re-focusing of the US SFR program in the US on the metal fuel SFR, the ANL experimental and analytical activities related to severe accidents have shifted to metal fuel phenomenology. Existing modules of SAS4A such as DEFORM were modified to allow the modeling of metal fuel pins, and a new module was developed to describe the pre-clad-failure in-pin fuel relocation (PINACLE). A series of metal fuel experiments was performed to study the severe accident phenomena typical for metal fuel SFRs, and these experiments were analyzed with the SAS4A (metal fuel) code.

In the meantime, the development and validation of the SAS4A oxide fuel code was continued by an active collaboration involving Japan, Germany, France, UK, and the EU. ANL continued to participate in some of the meetings of this group and to provide technical advice as needed. The SAS4A code continues to be used actively in Japan, France, and Germany for the study of the initiating phase of postulated SFR severe accidents. The SAS4A code, including the LOF material relocation module LEVITATE, was used for the ULOF analyses performed for severe accident analyses during the re-licensing of the Monju reactor in Japan.

Current research activities in Japan and France are focused on the development of the SIMMER code for the analysis of the transition phase. SIMMER was initiated in analyses performed in Japan after SAS4A, at the time of the failure of the inner tube, when radial molten fuel relocation can become significant. Thus, the interfacing between SIMMER and SAS4A becomes an important area of the severe accident analysis where ANL can provide valuable expertise. ANL has initiated a study and model development of the multi-dimensional material relocation in fuel assemblies during the US New Production Reactor (NPR) project, when the development of a 2-D version of the LEVITATE code (DIANA) was undertaken.

ANL has also performed extensive SAS4A analyses of the inherent safety characteristics of metal and oxide fuel SFRs in order to quantify the ability of various passive reactivity feedbacks, such as radial core expansion and control rod drive line expansion, to help prevent the occurrence of severe accidents.

5. Conclusions

Despite substantial differences in the reactor and plant design there are many similarities between the safety approaches to severe accidents in Japan and France. In both cases the CDAs are considered in the plant design. The goal is to improve both the prevention and mitigation functions.

The central element in the prevention approach is the addition of a third, passively activated, shutdown system. While the other two redundant reactivity shutdown systems are considered for Design Base Events, the third shutdown system is considered for severe accidents. When the third shutdown system is considered, the combined probability occurrence of CDA initiators and failure of all three shutdown systems is significantly below 10^{-7} per year. Thus the CDAs are excluded from the Design Base Events, and are considered in the safety analysis as Design Extension Conditions (DEC) or Beyond Design Base Events (BDBE).

The mitigation approach if focused on preventing re-criticality and assuring in vessel retention of core materials if a CDA does occur. Both JSFR and EFR include a debris tray, designed to ensure the retention of the relocated core materials, to protect the reactor vessel from direct contact with the molten fuel, and to maintain a debris bed that is both sub-critical and coolable. A feature unique to the JSFR design is the modified fuel assembly, M-FAIDUS, designed to allow an early escape from the core of the molten fuel and thus prevent a severe power burst during the transition phase.

CDA analyses, traditionally based on ULOF, are performed for JSFR and EFR to assure that there are no cliff-edge effects and to provide an understanding of the relative importance of core characteristics for the design and beyond design requirements. In Japan these analyses have used the SAS4A code for the initiating phase and on the SIMMER code for the transition phase. A modified version SIMMER named PAMR is being developed for the analysis of long-term post accident events. In France the codes used for the initiating phase analysis include SAS4A, FRAX, and PHYSURAC, while SIMMER is used for the transition phase analysis. The exact current capabilities and limitations of these codes are not clear from the available published information and, if ANL will assume a more active role in the evaluation of the SFR severe accidents, should be the subject of future technical discussions with experts in Japan and France.

As part of the US GNEP program it is expected that a prototype SFR will be built, probably by a consortium including US, Japanese, and/or French companies. The safety approach in general and the treatment of severe accident issues will thus be strongly influenced by the Japanese and/or French approaches to CDAs outlined in this report. ANL has played a leading role in the evaluation of postulated severe accidents for both oxide fuel and metal fuel SFRs and retains considerable expertise and international recognition in this area. Assuming an active role in the international SFR severe accident safety interactions will allow ANL to better understand the status of current related activities in Japan and France, their limitations and the areas where improvement is needed. This increased participation in the international SFR severe accident safety collaborations will allow ANL to provide expert advice to the DOE GNEP program in the area of SFR severe accident safety and to effectively represent the US DOE position in this area in the international arena.

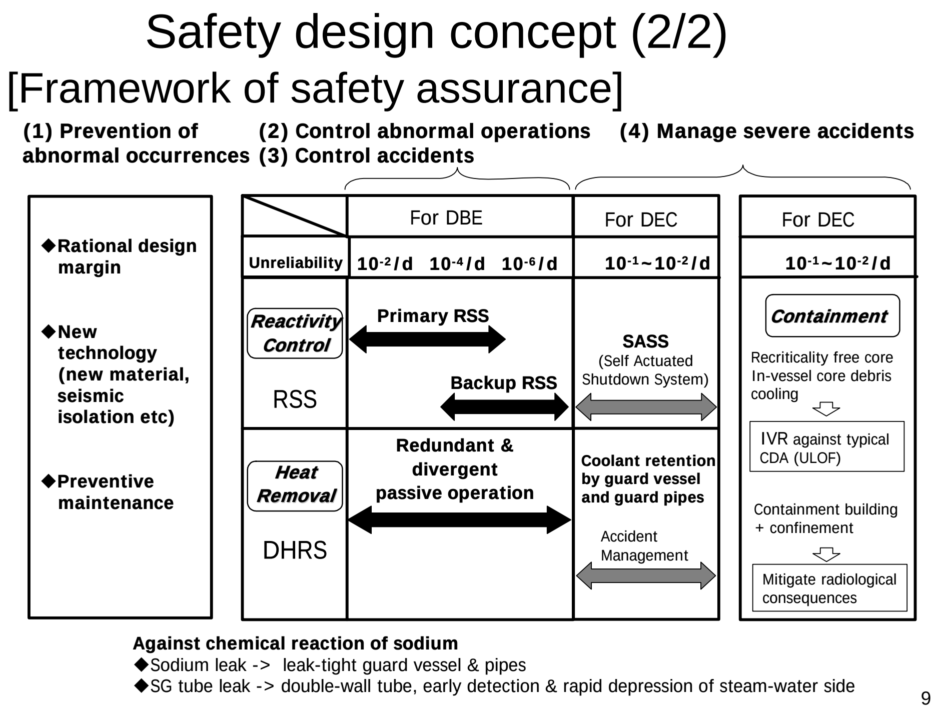

Fig. 1 Safety assurance strategy for JSFR (1)

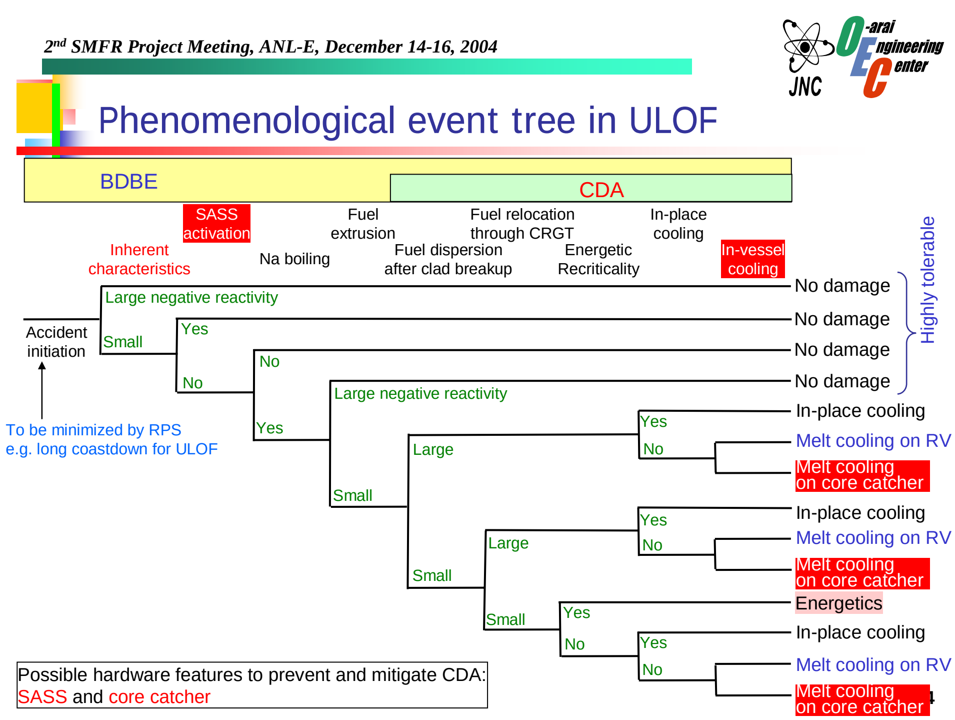

Figure 2. Phenomenological BDBE event tree in ULOF (3)

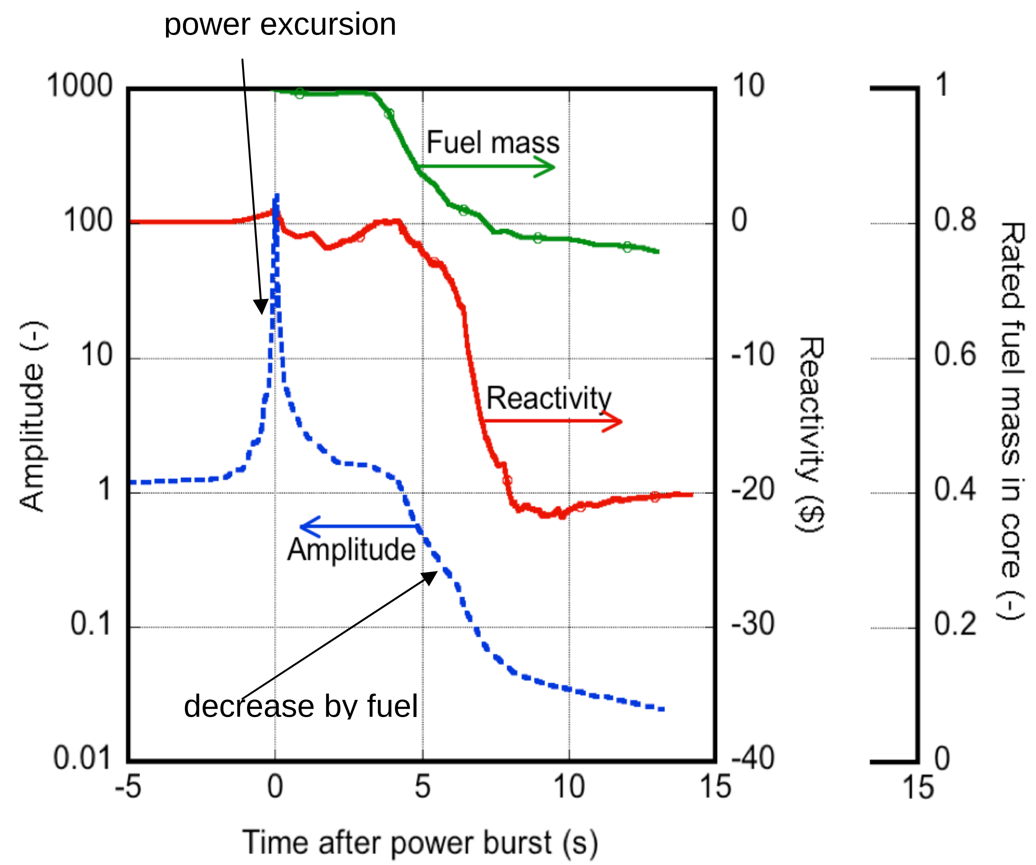

Figure 3. Power, reactivity, and fuel mass history during ULOF
in JSFR with modified FAIDUS (1)

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4. I. Sato, T. Fukano, K. Kawada, Y. Tobita, H. Yamano, S. Fujita - " Status of SAS4A and SIMMER-III/IV at JAEA, Severe Accidents Workshop, Cadarache, France, Dec 8-9, 2005

5. B. Carluec - "LMFBR Severe Accidents - EFR Safety Approach", Severe Accidents Workshop, Cadarache, France, Dec 8-9, 2005

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Appendix A

Abbreviations

ATWS - Anticipated Transient without Scram
BDBE - beyond Design Base Events
CDA - Core Disruptive Accidents
CDF - Core Damage Frequency
CRDL - Control Rod Drive-Line
CRGT - Control Rod Guide Tube
CRSS - Control Rod Stop System
DBE - Design Base Events
DEC - Design Extension Conditions
DFBR - Demonstration Fast Breeder Reactor
DHRS - Decay Heat Removal System
DRACS - Direct Reactor Auxiliary Cooling System
FA - Fuel Assembly
FS - Feasibility Study
GEM - Gas Expansion Modules
GNEP - Global Nuclear Energy Partnership
HCDA - Hypothetical Core Disruptive Accidents
IVR - In-Vessel Retention
JSFR - Japanese Sodium Fast Reactor
LOF - Loss of Flow
LOHS - Loss of Heat Sink
LOPI - Loss of Pipe Integrity
LORL - Loss of Reactor Level
NSSS - Nuclear Steam Supply System
PAHR - Post-Accident Heat Removal
PAMR - Post-Accident Material Relocation
PLOHS - Protected Loss of Heat Sink
PRACS - Primary Reactor Auxiliary Cooling System
RSS - Reactivity Shutdown System
RV - Reactor Vessel
RVACS - Reactor Vessel Auxiliary Cooling System
SASS - Self-Actuated Shutdown System
SFR – Sodium Fast Reactor
SMFR - Small Modular Fast Reactor
SG - Steam Generator
TOP - Transient Over-Power
ULOF - Unprotected Loss of Flow
ULOHS - Unprotected Loss of Heat Sink
USS - Ultimate Shutdown System
UTOP - Unprotected Transient Over-Power

Consultants' Meeting on the IAEA Coordinated Research Project (CRP)
"Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel"

J. Cahalan

Contents

1. Introduction

The author attended this preparatory meeting on 11 May in Tsuruga, Japan, at the MONJU reactor site. The meeting was organized by Alexander Stanculescu (IAEA) and hosted by the Japan Atomic Energy Agency (JAEA). The objective of the meeting was to perform preliminary planning for a coordinated research project to perform benchmark analyses of a flow test performed in MONJU in 1995. The meeting was attended by JAEA staff from MONJU and O-arai Engineering Center (OEC), staff from Central Research Institute of Electric Power Industry (CRIEPI - Japan), staff from Osaka University, staff from Commissariat à l'énergie atomique – Cadarache (CEA - France), staff from Indira Gandhi Centre for Atomic Research (IGCAR – India), staff from Korea Atomic Energy Research Institute (KAERI – Korea), staff from Institute of Physics and Power Engineering (IPPE – Russia), and the author.

2. Meeting Agenda

<u>9:00-10:00</u>

Part 1 Introduction of the Natural Convection CRP by IAEA

(i) Outline of the CRP (brief introduction of IAEA's rationale and implementation framework given by the TWG-FR)

(ii) Background of the CRP proposal (brief history of the proposal, starting with the Japanese manifestation of interest through official approval)

(iii) Implementation plan of the CRP (IAEA Program and Budget 2008 – 2009, procedures and rules for potential participants, etc)

<u>10:00-12:00</u>

Part 2 Confirmation of technical issues and discussions on action plans
Introduction of the Monju tests (JAEA)

(i) Presentation of the Monju plant trip test in December 1995
  - Plant configuration
  - In-vessel structure
  - Instrumentation
  - Test sequence

(ii) Test result

(iii) Boundary conditions

(iv) JAEA's post analysis compiled in 1997

<u>12:00-13:00</u> Lunch

<u>13:00</u>

Report and needs/requests for the technical content and outputs of the CRP from the potential participants:

(i) JAEA: validation needs for the methodology (modeling and new strategy of detailed thermal hydraulic analysis) to be applied for commercialized FBR design

(ii) Other Japanese participants (Universities, other institutions)
(iii) Other TWG-FR members

Part 3 Conclusions and definition of the preliminary action plan for the CRP:

(i) Technical content
(ii) Indication of interest to participate
(iii) Agreement on draft of official IAEA CRP proposal
(iv) Date and venue of the kick-off Research Coordination Meeting

Closing of the meeting

3. Meeting Summary

The meeting was opened by A. Stanculescu who described the meeting objectives. JAEA staff presented descriptions of MONJU and the 1995 natural convection test, the measured test results, and results of JAEA analyses performed in 1997. CRIEPI staff then presented results of test analysis performed with the CERES computer code, followed by presentations by delegates from KAERI, IPPE, and ANL on analysis capabilities in Korea, Russia, and the US. It was agreed by the attendees that the proposed CRP benchmark would be appropriate, and that planning should proceed. A. Stanculescu indicated that he would include provision for the CRP in the IAEA budget proposal for the coming year, and would organize the CRP if funding was obtained. The US traveler affirmed participation by ANL, subject to budget availability.

4. Background Situation Analysis

MONJU is a 714 MWth (280 MWe) loop-type prototype fast breeder reactor. It is fuelled with mixed uranium-plutonium oxide fuel and cooled by sodium (three loops). Initial criticality was achieved in April 1994, and connection to the grid in August 1995. In December 1995, while performing operation tests with the reactor at 40 % power, a sodium leakage incident occurred in the secondary sodium circuit. The reactor has remained shut down since then. JAEA (and before its creation, JNC and JAERI) have performed an extensive investigation of the causes of the leakage incident and a comprehensive safety review. Based on a thorough licensing procedure, the permit for plant modifications (including, among other, various countermeasures against potential sodium leakage) was issued in December 2002 by the Ministry of Economy, Trade and Industry (METI). JAEA has started preparatory work for modification, as soon as the local Fukui Province governor has granted approval in February 2005. The main modification work is in progress since September 2005. First criticality is expected to be achieved in the first quarter of 2008, followed by the start-up program. The main objective is to resume MONJU operation as soon as possible and achieve the initial goals of demonstrating the operational reliability of a fast reactor power plant, in particular with regard to sodium handling technology.

Thermal hydraulics of hot pool forms an important part of sodium cooled fast reactor vessel design. With reference to MONJU, due to the limited (as compared to the present) computational capabilities available at the time when the reactor was designed (roughly three decades ago), the reactor vessel design work had to rely heavily on a large number of mock-up experiments and numerical analyses based on pessimistic assumptions. Large margins had to be incorporated into the definition of the relevant parameters to account for rather important uncertainties. While this approach was appropriate at that time, since MONJU's main mission (as a prototype) was to confirm the technical feasibility of the sodium cooled fast reactor concept, current emphasis in fast reactor design work is shifting towards the demonstration of the economical competitiveness of power plants equipped with sodium cooled fast reactors. Therefore, current reactor component (in this case vessel) design must make use of state-of-the-art simulation codes that were developed on the basis of sophisticated physical models, and are made possible due to the vast improvement in computational technology. Moreover, current simulation methods have vastly increased the application fields of these codes to, e.g. the precise description of the transport of the delayed neutron precursors from failed fuel subassemblies, the confirmation of thermocouple locations, the identification of potential zones of thermal striping, gas entrainment possibilities, the analysis of the oscillatory nature of thermal stratification effects, etc.

However, the new numerical analysis methods require extensive validation efforts, preferable through collaborative international efforts. Recognizing that an IAEA initiated CRP is particularly well suited to provide the necessary international framework for such a validation effort, the specialists from the Member States of IAEA's Technical Working Group on Fast Reactors (TWG-FR) felt strongly the need for this CRP and recommended IAEA to implement it. The major activities to be carried out under this CRP, i.e. analytical/numerical benchmark exercises and comparisons with experimental data obtained at MONJU, will provide a wide basis for the validation/qualification of codes/methodologies being employed by the various Member States. Ultimately, the outcome of the CRP will be a contribution to more accurate simulation methods allowing reducing the margins and delivering cost effective design solutions, with enhanced safety features.

5. Overall Objective

The overall objective of the CRP is to improve the Member States' numerical simulation capabilities of complex thermal hydraulics of sodium cooled fast reactors. It is felt that a necessary condition towards achieving this objective is a wide international validation effort of the data and codes currently employed for the simulation of the various physical effects involved in this field. Towards realising this, it is expected that the experts from the interested Member States will contribute by participating and applying to the common benchmark exercises the different methodologies being used by them. In the first stage, the benchmark exercises will focus on the numerical simulation of the natural convection phenomenon in the upper plenum of the MONJU reactor vessel, for which coolant temperature and flow data were measured during the original MONJU start-up experiments, and will be made available by JAEA to the CRP participants.

6. Specific Research Objectives

For the first stage of the CRP, JAEA will submit to the participants the experimental data relevant to the envisaged benchmark exercise, specifically the sodium thermal stratification measurements performed in the MONJU reactor vessel upper plenum during a plant trip test conducted on 1 December 1995 with the reactor at 45% thermal power level (corresponding to

40% electrical power level). The trip test was meant to deliberately introduce abnormality in the condenser as triggering event. The subsequent scenario played out according to the following sequence: turbine trip (closure of the steam stop valve) → reactor trip (insertion of all the control rods) → sodium pump trip (only pony motors driving the pump and ensuring heat removal).

Process parameters data (coolant flow rates and temperatures at various locations) were measured during the above mentioned transient. JAEA will provide these data to the CRP participants along with the detailed description of the relevant geometry and other initial conditions, as needed by the participants to perform the respective simulations. The participants will apply their own methodologies (computer codes, modelling approaches, assumptions/simplifications, boundary conditions, etc) in their numerical simulations. Inter-comparisons between the various calculation results and between the calculations and the experimental data, including the results of the original analyses performed by JNC with the help of the multi-dimensional computer code AQUA (published in 1997), will be performed by the CRP participants. Based on the results of these inter-comparisons, subsequent investigations by the participants in the CRP will identify remaining open issues and further R&D needs to resolve them.

Based on the outcome of the first stage of the CRP, the possibility to extend the activities of the CRP to benchmark analyses of tests planned during the upcoming MONJU start-up experiments will be investigated. By the same token, given expressed views of the CRP participants in the consultants' meeting, the possibility to extend the scope of the CRP to benchmark exercises based on experimental data necessary for validating the coupled thermal hydraulics of the hot pool and the remaining systems will be discussed in upcoming meetings among the CRP participants, the TWG-FR members and the TWG-FR's Scientific Secretary.

7. Relationship to IAEA's Sub-Program Objectives

The CRP will be implemented as a programmatic activity of the IAEA Project 1.1.5.2 "Technology Advances in Fast Reactors and Accelerator Driven Systems" starting with the IAEA Program and Budget Cycle 2008 – 2009. The Project 1.1.5.2 has the objective, among others, to enable Member States to make informed decisions on the development of new or advanced fast reactor designs, and to increase cooperation between Member States in achieving advances in fast reactor technology development through international collaborative R&D. Given its objectives, as stated in Section 3 of this Meeting Report, the CRP clearly responds to the objectives of the IAEA Project 1.1.5.2.

8. Action Plan (Activities)

Member States with past and/or ongoing fast reactor programs are invited to participate in the CRP. The following institutes in Member States and international organizations have informally indicated their interest in participating:

China	China Institute of Atomic Energy
France	Commissariat à l'Energie Atomique CEA
India	Indira Gandhi Centre for Atomic Research

Japan	Japan Atomic Energy Agency
Rep. of Kazakhstan	Kazatomprom
Rep. of Korea	KAERI
Russian Federation	State Scientific Centre Institute of Physics and Power Engineering (IPPE) Obninsk
Switzerland	PSI
USA	ANL
OECD/NEA, Paris	

Following the establishment of an international team by putting in place research agreements and contracts, JAEA discloses the detailed data (see Sections 5 and 6 of this Meeting Report) to all the participants in the CRP. During this first preparatory stage of the CRP, IAEA, JAEA and the participants exchange information mainly by electronic communication means. To achieve its objectives, the CRP will comprise, during its actual implementation phase, the following coordinating activities:

I. First (kick-off) research consultant's meeting (RCM) to identify lead organizations among the CRP participants for the various topics/work packages, produce an agreed upon list of detailed tasks as well as work plans and deadlines, identify responsibilities for competing tasks, and to establish an outline and responsibilities for completion of the final CRP report (NE Series publications report).

II. Second and third RCMs to review progress of technical work and NE Series publications report status, and identify needed improvements and/or modifications to the tasks and/or work plans. In particular, at the 2nd and 3rd RCMs, the participants will discuss the possibility to extend the activities of the CRP to benchmark analyses of tests performed during the on-going MONJU start-up experiments, as well as to benchmark exercises aiming at validating simulation methods of the coupled thermal hydraulics of the hot pool and the remaining systems (see Section 4 of this Meeting Report).

III. Fourth RCM (if necessary) to review the status of the technical wok and perform an overall review of the CRP results, provide the final input to the NE Series publications report and finalize the draft of the NE Series publications report, identify open issues and actions to resolve them, and outline the road ahead as well as the Agency's role.

The estimated duration of the CRP is 3 – 4years. The schedule of near-term activities and actions arrived at in this consultants' meeting is as follows:

Submission of a short write-up by the participants in this consultants' meeting giving the details of their perspective and requirements with respect to the CRP, which will be included in the formal Meeting Report	**End May 2007**
The TWG-FR Scientific Secretary to release the Meeting Report as IAEA working material, incorporating the comments from the participants	**Mid June 2007**
JAEA to provide the consolidated experimental and other relevant	**1 July 2008**

data for the benchmark exercises in the form of a technical report, taking into account the participants' requirements	
Kick-off RCM (tentative)	**September 2008**

The sequence of activities is detailed in the chart below:

Activities	Year (0)	Year (1)	Year (2)	Year (3)	Year (4)	Year (5)
	2007	2008	2009	2010	2011	2012
1. Setting up the CRP team	X	X				
2. Convene 1st (kick-off) RCM		X				
3. Convene 2nd RCM			X			
4. Convene 3rd RCM				X		
5. Convene 4th RCM (if necessary)					X	
6. Issue of the final CRP report (NE publications series report)						X

Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439-4842

www.anl.gov